

Transient Testing of Nuclear Fuels Performed in the Original Operation of TREAT

Third Edition

Chemical and Fuel Cycle Technologies Division
Nuclear Science and Engineering Division

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Third Edition

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Executive Summary

This report provides summary information on experiments performed in the Transient Reactor Test Facility (TREAT) during its original operation from 1959-1994, and more specifically, before the TREAT core was upgraded at the end of the 1980s. The tests performed since about 1970 are summarized in considerable detail. Earlier tests are noted by their designated series and by some parameters that help to identify and distinguish each series from the others. The detailed summary descriptions are intended to help inform future researchers regarding the large, historical, empirical data base generated from that 35-year span during which hundreds of TREAT experiments were performed to investigate and demonstrate the transient behavior of a wide variety of nuclear fuels during severe off-normal and accident conditions -- using direct nuclear heating of the test samples. That empirical basis is foundational and valuable for guiding future qualification and safety evaluation of improved fuel designs and for future transient test planning and performance in TREAT. The second edition of this report added descriptions of the following test series: D, E, EOS, F, H, J, R, RX, and S. This third edition includes descriptions of an additional four test series: the HUT, HOP (& HUC), ORNL-TR, and PNL.

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1 Introduction

The Transient Reactor Test Facility (TREAT) is an experimental reactor designed for the transient testing of nuclear reactor fuels and other materials [1]. TREAT was designed and built by Argonne National Laboratory (ANL) at a site that is now the Idaho National Laboratory (INL) [2]. ANL (as a collaboration between ANL-East and ANL-West) operated TREAT for use in hundreds of experiments from 1959 until 1994, when the reactor was placed in non-operational standby. In 2013, the United States Department of Energy (U.S. DOE) identified the need for a resumption of transient testing in the United States to support the development and qualification of new, advanced nuclear reactor fuels [3] and decided to restart TREAT for that purpose. TREAT was successfully restarted by INL in November 2017 [4]. Moving forward, TREAT will be operated by INL to support new transient experiment programs.

Future operation of TREAT will benefit from the information produced during the extensive prior utilization of the facility, which included a wide variety of experiments. Knowledge of those prior experiments, including how they helped to develop the ability to understand and predict the transient behaviors of novel, advanced fuels, provides a foundation for qualification of numerous fuel forms already tested, as well as planning and designing new future tests [5].

The several editions of this overall summary report have been prepared sequentially with the intent of giving higher priority to tests that were conducted later in the historical time period (1959-1994) since those tests generally were more prototypic and addressed the fuel designs, performance, and safety issues relevant to the more recent reactor designs. The various test series added to each successive edition were selected also with the intent of grouping similar series together, such as general fuel type and transient conditions. It is planned to continue this approach in preparing future editions. In particular, the next two editions are planned to focus on approximately 15 to 20 test series (comprising about 90 tests) most of which are related to the transient overpower response of non-oxide ceramic fuels. The subsequent edition(s) will address the remaining 15 to 20 series. The latter comprise roughly 350 simple phenomenological and screening-type tests that were performed during the first decade of TREAT experimentation as the transient response characteristics of fast and thermal reactor fuels were first being explored. Because of the relative simplicity, close (“cookie-cutter”) similarities, and less available information regarding these tests, it is anticipated that one-page summaries will be prepared for groups of tests within series -- rather than for individual tests as has been the case so far -- and, in fact, preliminary drafts of many such group summaries already exist for those tests.

A brief summary of the entire set of historic TREAT experiments is provided in Appendix A. Further information on the broader set of experiments performed in TREAT can be found in the TREXR (TREAT Experimental Relational) Database, a compendium of reference documents and data which has been developed over the past several years at ANL [6].

In addition to the selected experiment set summaries, this report also provides brief background information on the TREAT reactor and the general nature of TREAT experimentation.

2 TREAT Description

TREAT is a graphite-moderated, graphite-reflected, air-cooled reactor fueled with 93% enriched UO_2 dispersed in graphite. The fuel is arranged in approximately four-inch-square fuel elements

consisting of a ~four-foot-long zircaloy-clad central fuel region and ~two-foot-long aluminum-clad graphite reflector regions above and below the fuel region. The reactor core can accommodate a maximum of 361 TREAT fuel elements in a 19x19 array. For irradiation experiments, a small number (typically two, occasionally a few more) of central TREAT fuel elements are replaced with an experiment vehicle containing the test sample(s). Frequently, an additional row (or half row) of fuel assemblies is also removed to provide an unimpeded path for fast neutron travel between the test sample and an ex-core system of collimator and detectors, collectively called the TREAT Fast Neutron Hodoscope. The hodoscope was used to provide real-time monitoring of fast neutrons emanating from test fuel samples located at the core center.

The core is controlled by B₄C-bearing control rods, arranged in three banks:

1. Control/shutdown rods, used to set the reactor to a critical state,
2. Computer-controlled transient rods, capable of high-speed travel, used to introduce reactivity changes which drive TREAT transients, and
3. Compensation/shutdown rods, used to compensate for the reactivity addition which occurs when a TREAT experiment vehicle is removed.

The core is cooled by forced air flow provided by two blowers. Because of the short duration of most power transients, the cooling effect is insignificant during the transients. Cooldown after transients typically takes several hours. The reactor core is enclosed in a radial graphite reflector surrounded by a concrete bioshield. Both the reflector and bioshield have holes to accommodate instrumentation.

TREAT transients are performed by introducing reactivity changes via movement of the transient rods from a critical pre-transient configuration (in which the transient and control/shutdown rod banks are both partially inserted within the core). There are three general categories of TREAT transients:

1. Temperature-limited (“burst”) transients, in which a quick (“step”) insertion of reactivity is caused by withdrawing the transient rods at their maximum speed, resulting in a bell-shaped burst in power which is constrained by the large, prompt negative temperature reactivity feedback provided by the heating of the TREAT fuel graphite;
2. Shaped transients, in which complex motions of the transient rods, under full computer control, are used to produce a specific core power-time history different from a simple power burst; and
3. Extended transients, lasting for several minutes at relatively low power, during which the reactor power is controlled by a combination of manual adjustment of control rod positions and computer control of transient rod positions. (Extended transients were historically used only in the STEP series of experiments which were performed in the mid-1980s.)

Major upgrading of the facility in the late 1980s included changing the locations of the control and transient rod banks in the core. The current, symmetrical arrangement, with 18 control rods and each of the three independent control rod sets operated simultaneously as a “bank,” is referred to as the “upgraded core.” The previous configuration, which was not symmetric, had fewer rods, and used two pairs of transient rods operated sequentially, is called the “pre-upgrade core.” The changed configuration resulted, of course, in significant differences in the characteristics of the spatial neutron flux distribution across the core and in the core-to-test-fuel neutronic coupling. With the exception of the M8 Calibration Experiment (M8CAL), which was performed in the upgraded core, all of the experiments discussed in this report were performed in the pre-upgrade core. A set of Inconel-clad fuel elements fabricated for use in a planned “TREAT Upgrade” core, which would have enabled TREAT to generate the harder neutron spectrum and higher fluence required for

experiments using 37-pin test-fuel bundles within an exceptionally-large loop, was fabricated but never used in the reactor.

A key performance metric in TREAT experiments is the total energy deposition (TED) achieved in the test sample(s). The TED is related to TREAT core operation through a parameter called the power coupling factor (PCF). The PCF is the ratio of test sample power (or energy) per unit mass to the reactor total core power (or energy). It is typically expressed in units of W/g/MW or J/g/MJ. The PCF may change over the course of a transient, due to changes in core conditions including temperature, rod position, and the neutron flux reaching the test sample; this is captured in a parameter called the transient correction factor (TCF). Typically, the PCF and TCF were characterized for a particular experiment via a series of calibration runs performed prior to the experiment itself, in order to plan how the TREAT reactor power needed to be controlled during the experiment in order to achieve the desired energy deposition and temperature conditions in the tested sample.

3 TREAT Experiment Program History

During the 35-year period from 1959 to 1994, roughly 800 experiments were performed in TREAT to investigate the response of various nuclear fuels and fuel element designs to off-normal and accident-related transients, with and without the presence of water or sodium coolants. These experiments supported the evolving needs of U.S. civilian nuclear reactor development programs (LWR, LMFBR, and IFR). As those needs evolved, experiment requirements (which were driven by a need to ensure prototypic conditions for the test samples) spurred enhancements of TREAT's capabilities for performing increasingly complex experiments.

Most of the early experiments that have been conducted in TREAT were performed in rapid succession during the first several years of TREAT operation. They were designed to investigate fuel-water and fuel-sodium interactions for a wide range of fuel/cladding materials and fuel/coolant temperatures. Those experiments used small samples of unirradiated fuel in simple, mass-produced capsule-type containments heated using basic TREAT power transients (initially only bursts and later constant power "flattops") with application to both light-water-cooled reactors (LWRs) and sodium-cooled fast reactors (SFRs).

Starting in the mid-1960s, the experiment programs transitioned to predominantly address oxide fuels for liquid-metal-cooled fast breeder reactor (LMFBR) development, both for severe accident evaluations and for qualification of fuels for use in demonstration reactors. Further advances in TREAT's automatic control system allowed the production of power transients that could be tailored in shape, intensity, and duration. At the same time, the need arose and persisted to test larger fuel samples (longer pins and multi-pin bundles) under more-prototypic conditions, primarily to guide the development and validation of computer models and codes that describe off-normal and accident transient behaviors. This resulted in experiment designs (particularly ones that provided flowing coolant) that were more complex and had greater cost and schedule requirements. These more complex experiments could be performed at a rate of only about ten to fifteen experiments per year. Correspondingly, there was diminishing practical opportunity to investigate large parametric matrices of test conditions or to gain evidence of repeatability of observations. Each experiment thus became essentially unique, even though experiments

continued to be planned and performed in “families” (sets, or series) within which key parametric changes were investigated.

The oxide-fuel LMFBR experimentation was pursued until the mid-1980s when attention was focused briefly on investigating the radiological source term during specific LWR accident scenarios, and also on investigating the behavior of modern metallic fuel for SFRs under severe transient overpower conditions. Experimental approaches, techniques, and equipment varied as needed during this time period in order to best address the variety of experiment objectives.

In general, TREAT experiments sought to create conditions that would reveal phenomena and demonstrate integral behavior of fuels and core materials that occur during reactor accidents, and to observe and measure those phenomena and behaviors. Topics of interest included:

- transient-induced fuel and cladding damage (composition and microstructural changes; and effects related to fission gas, chemical interactions, mechanical interactions, melting, and in-pin material vaporization),
- pre-failure fuel motions and coolant voiding,
- cladding failure thresholds,
- fuel-coolant interaction energetics,
- post-failure motions of fuel, cladding, and coolant,
- coolant channel blockages caused by disrupted fuel and cladding,
- and the complex inter-play among these behaviors.

The results of TREAT experiments were used for a variety of purposes, including:

- to help develop basic understanding,
- to acquire data relevant to developing empirical correlations which describe important behavior,
- to acquire data for validating mathematical models of certain phenomena, and
- to provide evidence useful in validating codes that describe complex integral behaviors such as might occur during the early stages of severe reactor accidents.

4 Historical Experiment Series Included in This Report

In the following sections of this report, an overview is given for each test series selected for inclusion in this edition of the report, using a common format for each overview. The format includes information in each of the following categories: the testing purpose, the approach taken, limitations involved in the approach, the in-pile experiment vehicle, the test fuel, imposed test conditions, results, and applications. Information is also provided on the power coupling determination, and references are indicated where the reader can find additional information.

More-detailed information is provided regarding each individual test by means of one-page summary descriptions grouped by experiment series in separate appendices at the end of the report. These one-page summaries provide an at-a-glance overview of each test (or sub-group of tests, as appropriate). They include descriptions of the test sample(s), reactor operating conditions, measurements performed during and post-irradiation, and a brief statement of the key results.

These one-page summaries and the primary source materials used to compile them can also be found in TREXR [6]. (Note: one-page summaries in TREXR that are not included in this report should be considered preliminary drafts.)

The 23 test series for which detailed summaries have been developed and included in this edition of the report comprise a total of 128 tests. These tests and key parameter information regarding them are indicated in Table 4-1. (Note that Edition 2 added the following series to the First Edition report: D, E, EOS, F, H, J, R, RX, and S. The additional series included in Edition 3 are: HUT, HOP (& HUC), ORNL-TR, and PNL.) For the HOP and HUT tests, the transient numbers are also shown to unambiguously identify tests that had been historically renamed.

Table 4-1. Key Characteristics of the Tests Addressed in This Report

Test	Fuel material	Bond	Clad	Geometry	Coolant	Imposed coolant flow	Overpower	Fuel failure / disruption
CDT1	Irrad MOX	He	HT9	1-pin	Sodium	Steady	Slow	Minor
CDT2	Irrad MOX	He	HT9	1-pin x 3	Sodium	Steady	Mod. Fast	Minor
CDT3	Irrad MOX	He	HT9	1-pin x 3	Sodium	Steady	Mod. Fast	Moderate
C01	Fresh MOX	He	SSTL	1-pin	NaK	(none)	Fast	Major
C02	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Major
C03	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Major
C04	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Slow	Moderate
C05	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Slow	Minor
C06R	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Very fast	Major
D1	Fresh UO2	He	SSTL	7-pin	Steady	Steady	Steady	None
D2	Fresh UO2	He	SSTL	7-pin	Steady	Steady	Steady	None
E1	Fresh UO2	He	SSTL	1-pin	Sodium	Steady	Very fast	None
E2	Fresh UO2	He	SSTL	1-pin	Sodium	Steady	Very fast	Major
E3	Irrad UO2	He	SSTL	3-pin	Sodium	Steady	Very fast	Major
E4	Fresh MOX	He	SSTL	1-pin	Sodium	Steady	Very fast	Major
E6	Irrad MOX	He	SSTL	1+6-pin	Sodium	Steady	Very fast	Major
E7	Irrad MOX	He	SSTL	7-pin	Sodium	Steady	Very fast	Major
E8	Irrad MOX	He	SSTL	7-pin	Sodium	Steady	Very fast	Major
EBT1	Irrad MOX	He	D9	1-pin	NaK	(none)	Mod. fast	Major
EBT2	Irrad MOX	He	D9	1-pin	NaK	(none)	Mod. fast	Major
EBT3	Irrad MOX	He	SSTL, D9, HT9	1-pin x 3	Sodium	Steady	Mod. slow	None
EBTB	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Mod. fast	Major
EOS1	Fresh MOX	He	SSTL	1-pin	(none)	(NA)	Very fast	Major
EOS2	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Very fast	Major
EOS3	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Very fast	Major
F1	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Steady	Major
F2	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Very fast	Major
F3	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Steady	Major
F4	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Steady	Major

F6	Irrad MOX	He	SSTL	1-pin	(none)	(NA)	Steady	Major
H1	Fresh MOX	He	SSTL	1-pin	Sodium	Steady	Very fast	None
H2	Fresh MOX	He	SSTL	1-pin	Sodium	Steady	Fast	Major
H3	Irrad MOX	He	SSTL	1+6-pin	Sodium	Steady	Mod. fast	None
H4	Irrad MOX	He	SSTL	1+6-pin	Sodium	Steady	Mod. fast	Major
H5	Irrad MOX	He	SSTL	1+6-pin	Sodium	Steady	Mod. fast	Minor
H6	Irrad MOX	He	SSTL	7-pin	Sodium	Steady	Mod. fast	Major
HOP 1-6A	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Variable	Moderately slow	Minor (in only one pin)
HOP 3-1A	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
HOP 3-2A (tr. 1452)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
HOP 3-2B (tr. 1511)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
HOP 3-2D (tr.1518)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Moderate
HOP 3-1B (tr. 1542)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
HOP 3-2C (tr. 1543)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
HOP-PTO 1-2A	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Moderately slow	None
HOP-PTO 3-2E	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Fast	None
HUC-PTO 2-2A	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Continuous flow reduction	(steady power)	None
HUT 1-6A	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Slow	None
HUT 1-8A	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Continuous flow reduction at constant power	Slow overpower at steady flow rate	(posttest condition unknown)
HUT-L2	Fresh MOX	He	SSTL	1-pin x 3	Sodium	Steady	Slow	None
HUT 3-2A (tr. 1642)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	Moderate
HUT 3-2B (tr. 1518)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	Minor
HUT 3-5A (tr. 1614)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	None
HUT 3-5B (tr. 1898)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	None

HUT 3-6A	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Minor
HUT 3-6B	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Moderate
HUT 3-7A (tr. 1634)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	None
HUT 3-7B (tr. 1645)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Very fast	Major
HUT 5-1A (tr. 2032)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Slow	Failure extent unknown
HUT 5-2A (tr. 1600)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	None
HUT 5-2B (tr. 1648)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	Major
HUT 5-5A (tr. 1577)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	Moderate
HUT 5-5B (tr. 1903)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	None
HUT 5-7A (tr. 1592)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	None
HUT 5-7B (tr. 1637)	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Moderately fast	Failure extent unknown
HUT 5-9A	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Slow	None
L02	Irrad MOX	He	SSTL	7-pin	Sodium	Steady	Fast	Major
L03	Irrad MOX	He	SSTL	7-pin	Sodium	Steady	Slow	Major
L04	Irrad MOX	He	SSTL	7-pin	Sodium	Coastdown	Fast	Major
L05	Irrad MOX	He	SSTL	7-pin	Sodium	Low	Fast	Major
L06	Irrad MOX	He	SSTL	7-pin	Sodium	Low	Fast	Major
L07	Irrad MOX	He	SSTL	7-pin	Sodium	Low	Very fast	Major
M1	Irrad U-5Fs	Na	SSTL	1-pin	(none)	(NA)	Steady	Minor
M2	Irrad U-5Fs	Na	SSTL	1-pin x 3	Sodium	Steady	Mod. Slow	Minor
M3	Irrad U-5Fs	Na	SSTL	1-pin x 3	Sodium	Steady	Mod. Slow	None
M4	Irr + fresh U-5Fs	Na	SSTL	1-pin x 3	Sodium	Steady	Mod. Slow	Minor
M5	Irrad UPuZr	Na	D9	1-pin x 2	Sodium	Steady	Mod. Slow	None
M6	Irrad UPuZr	Na	D9	1-pin x 2	Sodium	Steady	Mod. Slow	Minor
M7	Irrad UPuZr, U-Zr	Na	D9, HT9	1-pin x 2	Sodium	Steady	Mod. Slow	Minor
ORNL-TR-1	Fresh MOX (pellet and Sphere-Pac)	He	SSTL	1-pin x 3	Sodium	(none)	Fast	None
ORNL-TR-2	Fresh MOX (pellet and Sphere-Pac)	He	SSTL	1-pin x 3	Sodium	(none)	Fast	None

PINEX2	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	None
PINEX3	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Mod. fast	Minor
PNL 1-1	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Major
PNL 1-2	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast -?	None
PNL 1-7	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast -?	None
PNL 1-8	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast	Minor
PNL 2-1	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast -?	Moderate
PNL 2-10	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast -?	None
PNL 2-11	Irrad MOX	He	SSTL	1-pin	NaK	(none)	Fast -?	Major
R3	Fresh UO2	He	SSTL	1-pin	Sodium	Coastdown	Steady	Major
R4	Fresh UO2	He	SSTL	7-pin	Sodium	Coastdown	Steady	Major
R5	Fresh UO2	He	SSTL	7-pin	Sodium	Coastdown	Steady	Major
R6	Fresh UO2	He	SSTL	7-pin	Sodium	Coastdown	Steady	Major
R7	Fresh UO2	He	SSTL	7-pin	Sodium	Coastdown	Steady	Major
R8	Fresh UO2	He	SSTL	7-pin	Sodium	Coastdown	Steady	Major
R9	Fresh UO2	He	SSTL	7-pin	Sodium	Steady	Mod. fast	Major
R12	Fresh UO2	He	SSTL	7-pin	Sodium	Steady	Mod. fast	Major
RFTL1	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Mod. fast	Minor
RFTL2	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Slow	None
RFTL3	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Slow	None
RFTL4	Irrad MOX	He	SSTL	1-pin x 3	Sodium	Steady	Fast	None
RX1	Fresh UO2+SSTL	(NA)	none	cylinder	(none)	(NA)	Steady	Major
RX2	Fresh UO2+SSTL	(NA)	none	cylinder	(none)	(NA)	Steady	Major
S2	Fresh UO2	He	SSTL	5 (+4)-pin	Sodium	None	Very fast	Minor
S3	Fresh UO2	He	SSTL	5 (+4)-pin	Sodium	None	Very fast	Major
S4	Fresh UO2	He	SSTL	5 (+4)-pin	Sodium	None	Very fast	Major
S5	Fresh UO2	He	SSTL	5 (+4)-pin	Sodium	None	Very fast	Major
S6	Fresh UO2	He	SSTL	7-pin	Sodium	None	Very fast	Minor
S7	Fresh UO2	He	SSTL	7-pin	Sodium	None	Very fast	Major
S8	Fresh UO2	He	SSTL	7-pin	Sodium	None	Very fast	Major
S11	Fresh UO2	He	SSTL	1-pin	Sodium	None	Very fast	Major
S12	Fresh UO2	He	SSTL	1-pin	Sodium	None	Very fast	Major
STEP1	Irrad UO2	He	Zircaloy-4	4-pin	Steam	Steady	Very slow	Major
STEP2	Irrad UO2	He	Zircaloy-4	4-pin	Steam	Steady	Very slow	Major
STEP3	Irrad UO2	He	Zircaloy-4	4-pin	Steam	Steady	Very slow	Major
STEP4	Irrad UO2	He	Zircaloy-4	4-pin	Steam	Steady	Very slow	Major
TS1	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Slow	Minor
TS2	Irrad MOX	He	SSTL	1-pin	Sodium	Steady	Slow	Major

NOTES:

Geometry

"1-pin," "3-pin," "4-pin," and "7-pin" refer to that number of identical pins in a single cluster

"1+6-pin" refers to 1 pre-irradiated pin surrounded by 6 fresh pins

"5 (+4)-pin" means 5 fuel pins and 4 hollow dummy pins interspersed in a cluster

"1-pin x 3" refers to 3 pins in separate flowtubes with common inlet and outlet flow headers

Overpower	Rate refers to the rise time to peak power. Very fast = 0 to 0.3 s; Fast = 0.3 to 1.5 s; Moderately-fast = 1.5 to 3 s; Moderately-slow = 3 -10 s and includes 10 ¢/s simulations; Slow = 10 - 60 s and includes 5 ¢/s simulations; Very slow > 60 s.
Fuel failure/disruption	Refers to severity of the damage to the fuel sample: "None" means no cladding breach and at most minor fuel changes "Minor" means significant fuel changes, and usually includes small breaching of cladding "Major" means extensive cladding damage and gross redistribution of fuel sample materials

5 CDT-series Experiments

The CDT-series tested full-length, top-plenum, annular design (fuel, insulator, and reflector), irradiated MOX fuel pins in flowing sodium under fast and slow TOP (transient overpower) scenarios to show pre-failure fuel extrusion potential. A one-page summary of each CDT test is included in Appendix B.

Number/Time

The CDT-series consisted of three experiments performed in 1987 on fast-reactor FFTF-irradiated mixed-oxide fuel of the latest designs considered for deployment in the FFTF core. Five pins were exposed to severe overpower transients, comparing effects of solid-pellet versus annular-pellet designs at two power ramp rates. The results spanned the cladding failure thresholds and provided data useful for validating transient fuel-behavior models [7, 8, 9, 10].

Purpose

The three CDT tests were performed to provide failure threshold and pre-failure fuel-motion data for two designs of advanced FFTF-type mixed-oxide fuel pins, at two (medium and high) burnup levels, when exposed to different overpower transients (5 ¢/s and 1 \$/s reactivity insertion rates) relevant to FFTF severe accident safety assessment. The data were intended to be used for comparing the transient response of the two pin designs and evaluating the predictive capability of various fuel-pin transient performance codes and pin-failure models. The two pin designs differed by one having solid fuel and axial blanket pellets and the other having annular fuel and axial blanket pellets.

Approach

The tests were performed in a manner that would allow close comparison of the transient behaviors of the different fuel pins under conditions as prototypic of the simulated FFTF accident transients as feasible within the limitations of the TREAT reactor (e.g., utilizing the maximum-allowed TREAT energy). Conditions were intended to be created that would cause the fuel pins to marginally reach cladding failure. The set of test pins and test conditions was chosen to form a tight matrix of a few parameters in order to maximize the simplicity of making comparisons between the resulting behaviors. Thus, the same 5 ¢/s reactivity insertion transient was used in two tests (CDT-1 and CDT-3), and solid-pellet designs and annular-pellet designs were tested at both overpower ramp rates. Furthermore, solid-pellet designs of both medium and high burnups were subjected to the same (1 \$/s) transient, simultaneously with a pin of annular-pellet design. Sodium flow rates were chosen to achieve nominal FFTF conditions prior to the initiation of the power ramp.

Limitations

Although the transient behavior of an annular-pellet pin design was intended to be compared to the transient behavior of a solid-pellet pin design, the annular-pellet pins were of medium burnup

whereas the solid-pellet pins were of high burnup. The fast fluence exposure of the cladding of the annular-pellet pins was correspondingly lower than that of the solid-pellet pins.

The accuracy of the empirical determination of the power coupling between the TREAT reactor core and the fuel in each of the pins tested was diminished by unexpected deficiencies in the test vehicles that were used, as noted below.

Because of the high temperature of molten oxide fuel, fuel expulsion from a failed pin into the coolant channel would tend to locally void the channel of sodium and quickly melt through the thin-walled flowtube. A breach in the flowtube where fuel is moving would affect the subsequent fuel motion dynamics.

Experiment Vehicle

Tests CDT-1 and CDT-3 were performed using two nominally-identical Single-Pin Test Loops (SPTLs). Test CDT-3 used a modified Mark-IIIC loop containing three fuel pins. Each pin was located within its own flowtube, with several thermocouples attached to the flowtube's outer surface. The flow rates through the three flowtubes in test CDT-2 were nominally identical. In all three tests, each flowtube was initially surrounded by an argon-filled space to limit heat transfer radially outward from the flowtube.

It was discovered during posttest examinations of tests CDT-1 and CDT-3 that the expansion bellows near the top of each flowtube had failed. (It was unknown whether the failure had occurred before or during the test.). These failures allowed sodium to inadvertently enter and fill the spaces around the flowtubes, forcing out the gas that had been there, and contacting the flowtubes' outer surface and the attached thermocouples. The presence of the sodium subsequently affected heat transfer through the flowtube wall and measurement of the flowtube temperatures. This condition, if it had existed before the test, would have affected the interpretation and accuracy of the in-situ heat-balance measurements which were performed to determine the test-fuel-to-TREAT power coupling. Data intended to be obtained from the tests were limited also by premature failure of many of the flowtube thermocouples.

Test Fuel

All five of the pins tested were of helium-bonded mixed-oxide fuel clad in HT-9 ferritic alloy, with 0.914 m fuel column height and a 6.5 cm axial blanket pellet stack at both ends of the fuel. Two pins (for CDT-1 and CDT-2) were of solid-pellet design of high-burnup (116-118 MWd/kg) and $17\text{--}18 \times 10^{22}$ n/cm² fast fluence. Two other pins (for CDT-2 and CDT-3) were of annular-pellet design (1.40 mm as-fabricated internal diameter) of medium burnup (63-65 MWd/kg) and $9.4\text{--}9.9 \times 10^{22}$ n/cm² fast fluence. The fifth pin (also for CDT-2) was similar to the annular-pellet pins except that it had solid pellets.

Conditions

CDT-1 and CDT-3 each subjected a single fuel pin to a power transient that increased on a 23-s e-folding period, corresponding to a 5 ¢/s reactivity insertion rate in FFTF. In contrast, CDT-2 subjected three fuel pins, each in its own flowtube, to a power transient that increased on a 0.67-s e-folding period, corresponding to a 1 $\text{\$/s}$ reactivity insertion rate in FFTF. In each case, the power increase was continued until the cladding failure threshold would be reached or nearly reached. Leading into the overpower transient in each test was a 10-s hold at a level generating approximately nominal fuel power (P_0) for fuel of that burnup in FFTF (34-37 kW/m). In CDT-2, the power in the medium-burnup solid-pellet pin was about 14% higher than in the medium-burnup annular-pellet pin.

In CDT-1 and CDT-3, the unfortunate failure of the bellows near the top of the flowtubes diminished the ability to know the thermal conditions in the test fuel and in the sodium adjacent to the fuel pins.

In CDT-2 the bellows remained intact, but the combination of temperatures indicated by the thermocouples, the sodium flowrate indicated by the flowmeter, and the assumed fuel pin power were mutually inconsistent, leading the analysts to conclude that the actual flow rate was 20% lower than indicated by the flowmeter.

Results

During the three tests, two of the five pins failed. Both of the medium-burnup pins (one solid-pellet and the other annular-pellet design) in fast-ramp test CDT-2 withstood a peak power of about 17 P_0 but breached just after peak power, as the power was rapidly dropping due to the reactor scram. The three other pins remained intact but apparently had been taken to near their failure thresholds. The high-burnup solid-pellet pin survived 4.0 P_0 in slow-ramp test CDT-1 and 17 P_0 in test CDT-2. The medium-burnup annular-pellet pin in slow-ramp CDT-3 survived 4.5 P_0 despite experiencing 20% more fuel enthalpy and 90°C higher peak coolant outlet temperature than did the high-burnup solid-pellet pin in CDT-1.

Substantial axial, pre-failure, in-pin fuel motion occurred in all five pins, to the extent that the spring in the pin plenum was found to be fully compressed in each case – an upward movement of about 6 cm.

Axial fuel motion in the solid-pellet pin in CDT-1 began at a power of about 3.6 P_0 , when about 75% of the transient energy had been generated. About 6 cm length of molten fuel extruded into the region above the initial fuel column, lifting the upper axial blanket and plenum tube until the spring became fully compressed. In CDT-3 axial fuel motion in the annular-pellet pin began at about 3.8 P_0 . The fuel penetrated the annulus of both the upper and lower blankets along their entire lengths, and ~10 g of fuel (5% of the fuel inventory) entered the plenum above the axial blanket. Posttest neutron radiography of CDT-3 showed a large axial variation of fuel density along the original fuel region.

In CDT-2, in both the high-burnup and medium-burnup solid-pellet pins, axial fuel motion within the upper blanket region occurred at peak power, and by the end of the test, the upper blanket had been pushed up about 5 cm in the high-burnup pin and about 7.5 cm in the medium-burnup pin. The high-burnup pin remained intact, but the medium-burnup pin failed just above the axial midplane, by meltthrough according to posttest computations. Posttest examination showed the failure extending over a region 7 to 25 cm above the midplane (i.e., from 0.58 to 0.77 relative fuel height). By the end of the CDT-2 test, molten fuel in the medium-burnup annular-pellet pin had reached and coated the inside of the spacer tube as high as 33 cm above the upper blanket column. The upper blanket was raised about 5 cm by axial fuel motion. As with the medium-burnup solid-pellet pin, the pin failed just above the axial midplane, by meltthrough according to posttest computations. The observed rapid loss of fuel just after peak power, and the fuel accumulation in a region centered in a region about 10 cm above the fuel midplane, was interpreted as being a consequence of midplane failure. Within 20 ms of cladding failure, molten fuel from that breached annular-pellet pin penetrated the flowtube in which that pin was located and spread into the gas-filled region beyond.

Posttest metallographical examinations indicated that in the solid-pellet pin in CDT-1 the fuel areal melt fraction was 47% at $\frac{1}{4}$ height of the fuel column, 50% at midheight, and 56% at $\frac{3}{4}$ height. When compared with code computations, these values suggest that the actual fuel pin power was about 20% less than expected from pretest determinations. The melt fractions in that pin were notably lower than measured in the annular pin tested in CDT-3, which were 54% at $\frac{1}{4}$ fuel height, 75% at the midplane, and 68% at $\frac{3}{4}$ fuel height. The high-burnup solid-pellet pin that survived in the fast-ramp CDT-2 test had measured melt fractions of 61%, 68%, and 75% at $\frac{1}{4}$, half, and $\frac{3}{4}$ height levels, respectively.

The posttest measured peak cladding strain level in CDT-1 solid-pellet pin was 3.7% (at the top of the initial fuel column and above, of which 2.7 % occurred during the transient). In the cladding of the annular-pellet pin in CDT-3, however, there was only 1.4% strain (all occurring during the transient). The surviving, high-burnup solid-pellet pin in CDT-2 showed total strain of $\sim 7\%$ near the midplane, $\sim 4\%$ near the fuel top, and $\sim 10\%$ about 5 cm above the fuel top.

Applications

Results from the three CDT transient overpower tests showed that the latest FFTF-design mixed-oxide pins could survive to power levels greater than 4 Po during 5 ¢/s reactivity-ramp-rate accidents and up to 17 Po during 1 $\text{\$/s}$ accidents. The pins exhibited substantial pre-failure axial fuel movement, including through the center of annular blanket pellets, which could significantly introduce a negative reactivity effect on the accident. Significantly-lower transient-induced cladding strains in the annular-pellet design pins than in the solid-pellet design pins despite higher energy deposition in the annular-pellet pin, is attributable to the design difference. The test results were useful in evaluating transient fuel behavior models and the acceptability of the fuel designs for use in FFTF.

Calibrations in TREAT

Power coupling between the TREAT core and the test fuel was experimentally determined prior to the test transient by performing heat balance transients using the actual test fuel and test vehicle. The heat-balance transients were run at a steady, relatively-low power level representative of nominal pin power.

6 CO- and LO-series Experiments

The LO-series tested 7 seven-pin bundles of full-length, bottom-plenum, annular, fresh and irradiated MOX fuel pins in flowing sodium under fast and slow TOP (transient overpower) and a range of TUCOP (transient undercooling-driven overpower) conditions. The CO-series tested single, full-length, bottom-plenum, annular, fresh and irradiated MOX pins, 3 under fast TOP in stagnant NaK, and 2 under slow TOP and 1 under TUCOP in flowing sodium [10, 11, 12, 13, 14, 15, 16, 17]. A one-page summary of each CO-series test is included in Appendix C; one-page summaries of the LO-series tests are in Appendix D.

Number/Time

Six CO-series tests (CO1 through CO5, plus CO6R) and seven LO-series tests (LO1 through LO7) were performed during 1980-1983 within the UKAEA/USDOE PFR/TREAT Program.

Purpose

The LO- and CO-series tests were performed to investigate the timing and location of cladding failure and the pre-failure and post-failure fuel motions during overpower and loss-of-flow severe

accident transients in large mixed-oxide-fueled fast reactors in which the fuel pins contain annular fuel and are grid-spaced and of bottom-plenum design and in which coolant voiding causes a positive reactivity addition resulting in a significant power burst. In conjunction, the tests were run to validate existing computational models of fuel pin failure and to strengthen the understanding of the relative roles of cladding-failure mechanisms.

Approach

The testing program consisted of tests on single pins (CO-series) and on seven-pin bundles (LO-series). The former provided azimuthal uniformity in fuel-pin thermal conditions, whereas the latter provided for effects of larger fuel mass and multi-pin geometry with multiple coolant channels. Tests CO1 through CO3 were performed in stagnant NaK-filled capsules. The other tests were performed in flowing-sodium loops. Key parameters among the collection of tests included fuel burnup (medium and high), overpower ramp rate (slow and fast), and (for the loss-of-flow accident simulations) degree of coolant channel voiding when fuel was expelled into the channel (incipient voiding, partially voided, fully voided).

Limitations

The UK-design fuel differed from typical US-designed fuel in that the fission-gas plenum was below the fuel, and the pins were not wire-wrapped but were instead designed for use with grid spacers (which were used in the LO-series tests). Radial power depression in the fuel pins and within fuel-pin bundles due to self-shielding of the test fuel within TREAT's thermal neutron spectrum was a significant factor in designing and planning the tests to provide maximum feasible prototypicality in radial temperature profiles through the pin. (For example, outer coolant channels in the pin-bundle tests LO3 through LO7 were enlarged to compensate for the higher fuel power near the bundle periphery.) Hydraulic characteristics of the test vehicle internals above and below the test fuel region were not fully prototypic of the above-core and below-core hydraulic situations in commercial plants and thus affected the prototypicality of the post-failure material motions. Melting of some of the flowtube after contact with fuel added molten steel mass to the dispersing molten fuel and cladding. TREAT energy capability allowed only about 20 full-power seconds of test-fuel energy generation in the TUCOP tests LO4-LO7. Conditions in static capsule tests were not considered relevant regarding application of post-failure fuel motions.

Experiment Vehicle

Three different test vehicle designs were used: (a) NaK-filled single-pin capsules for CO1, CO2, and CO3 (used to obtain early information) in which the fuel pin was thermally bonded by NaK to a massive nickel heat sink, (b) flowing-sodium single-pin loops for CO4, CO5, and CO6R (which provided greater thermal-hydraulic prototypicality than did the capsules), and (c) seven-pin-bundle flowing-sodium loops for LO1 through LO7 (which provided greater prototypicality regarding multi-pin and multi-channel effects during post-failure interactions of fuel, cladding, and coolant). In the pin-bundle tests, the pins were separated by grid spacers. In LO1-LO3 a thick upper captor grid, representative of a fuel-pin hold-down device in a full scale reactor, was located above the top of the pin bundles. In LO4-LO7, immediately above the tops of the fuel pins was an 18.3-cm-long bundle of seven cladding tubes, each tube containing a steel rod equal in diameter to the fuel; this simulated the above-core structure of a full-scale reactor where molten material might freeze after pin failure and disruption.

Test Fuel

The fuel pins in all of the tests were of helium-bonded mixed U-Pu oxide fuel in stainless steel cladding. The MOX fuel pellet stacks in the pins were 91.4 cm high, with depleted urania breeder columns above and below the fuel. The fission gas plenum in the pin was located below the fuel

column, separated from the fuel by a molybdenum knitmesh plug. Fresh (unirradiated) pins were used in C01, L01, and L06; pins of 9 at% burnup were used in C03 and C05; the pins used in the other tests had ~4 at% burnup. The 9% burnup fuel had been irradiated at a higher power level (27.7 kW/m burnup-averaged) than the 4% burnup fuel (17.1 kW/m burnup averaged), causing the fission gas retention to be about the same for both fuels.

Conditions

Fast (5 \$/s reactivity ramp simulation) transient overpower (TOP) conditions were applied in the three single-pin capsule tests (C01, C02, and C03 on fuel of 0%, 4% and 9% burnup, respectively) and in the seven-pin loop tests (L01 and L02 on fuel of 0% and 4% burnup, respectively). Slow (10 ¢/s reactivity ramp simulation) overpower transients were applied in two of the single-pin loop tests (C04 and C05 on fuel of 4% and 9% burnup, respectively) and one seven-pin loop test (L03 on fuel of 4% burnup). Simulations of transient undercooling-driven over-power (TUCOP) conditions - caused by undercooling conditions and coolant voiding in a large fast reactor with positive sodium-void reactivity coefficient -- were generated in one single-pin loop test C06R and four seven-pin loop tests L04 through L07 (using 0% burnup fuel in L06 and 4% burnup fuel in C06R, L04, L05, and L07). The key variable in the TUCOP tests was the degree of voiding in the coolant channel at the time and location where fuel was initially expelled from the pin(s) - no voiding, some voiding, or complete voiding.

Results

In general, there was little or no prefailure fuel motion, only some axial expansion in some tests. Fresh fuel under fast TOP conditions failed near the fuel midplane. Over the burnup range tested (0 to 9%), the power-to-fail under fast TOP (5\$/s) conditions was roughly 24 times nominal power and under slow TOP (10 ¢/s) conditions was three times nominal power. Upon cladding failure, fuel motion was extensive and predominantly upward, in most tests dispersing to an extent corresponding to a negative reactivity addition of approximately 4 to 18% if it had occurred in a large fast reactor. (The worth analysis assumed a cosine-squared axial worth distribution with fuel-width at half maximum being 80% of the fuel height.) The exceptions were C04 in which the worth decrease was about 30%, and L05 in which there was a 7% worth decrease at pin failure but a final net worth increase of 4%. Failed fuel tended to accumulate at the sites of the grid spacers, and its upward motion was strongly limited by the presence of the upper captor grid in the pin-bundle tests. In the TOP tests, vaporization of coolant created pressure that expelled fuel out both ends of the flow channel. In the TUCOP tests with preirradiated fuel (L04, L05, and L07) upon pin failure an accumulation of fuel formed gradually at the failure site, grew axially, and then dispersed into regions above and below the original fuel zone. The fresh fuel in L06 was less mobile. Upper and lower blockages (one of which was complete and the other either complete or nearly so) were found posttest in each of the seven-pin tests.

Applications

Combining models with the observed results provided good understanding of cladding failure mechanisms for fuel pins of the design tested, including internal pressurization, cladding melting, and swelling of hot solid fuel. Explanation of the initial fuel release from the pin was obtained from computations of the amount of molten fuel and existence or absence of a solid fuel crust at the time and location of failure. The observations were generally consistent with results of several computational models used to predict and analyze the tests performed with sodium flow.

Calibrations in TREAT

Power coupling between the TREAT core and the test fuel was experimentally determined for tests C01, C02, and C03 by means of irradiating and radioactivity analysis of a fresh fuel pin (designated

as the US/UK CAL activity) as well as by means of pre-test reduced-power heat-balance transients of the test pin within the test capsule. Power coupling determinations for CO4 and CO5 were made by TREAT irradiation of a fresh fuel pin and monitor wires in the activities designated as CO4-CAL, and similarly for CO6R by activities designated as CO6-CAL. Similar activities were conducted for LO1 and LO2 (LO1-CAL), for LO3 (LO3-CAL), and for LO4-LO7 (LO4-CAL). In addition, heat-balance transients were typically run using the actual test fuel and vehicle prior to the final transient for each test.

7 M-series Experiments

The M-series tests focused on in-pin fuel motion in pre-irradiated metallic fuel pins (of circa 1980 metallic fuel design) during severe transient overpower, with cladding failure by rapid fuel-cladding metallurgical interaction [10, 18, 19]. A one-page summary of each M-series test is included in Appendix E.

Number/Time

Seven M-series tests (M1 through M7) were performed during 1985-1987 in the pre-upgraded core. In anticipation of future tests in the series, neutronic calibration irradiations (M8CAL) were performed in the subsequently upgraded core.

Purpose

The M-series tests were performed to investigate in-pin motion of the latest designs of sodium-bonded fast-reactor metallic fuel during severe transient overpower accidents. Of particular interest was the potential for substantial pre-failure fuel motion ("extrusion") as a potential means of early negative reactivity insertion during an overpower accident. Another objective was to determine the nature of transient-induced cladding failure by the combined action of fuel-cladding metallurgical interaction and fission gas pressure. Evaluation of effects of fuel pin plenum pressure, retained fission gas, and sodium-logging of the fuel were of primary interest. The study of post-failure, in-channel fuel motion of the metallic fuel was not a major objective.

Approach

Test M1 heated two open-ended segments of an irradiated fuel pin at constant power in a dry-capsule environment (no coolant) and relied on high-speed photography to observe fuel motion out of the cladding. The following information refers to tests M2 through M7. Two or three intact fuel pins were heated with an exponentially-increasing power transient in a flowing-sodium loop. A total of 15 fuel pins were tested. Each pin was tested in a separate coolant environment so that more than one pin could be tested simultaneously, with a different flow rate past each pin, to permit different power/flow ratios between pins. The fuel pins were separated far enough to essentially eliminate neutronic effects of one pin upon the other. Fuel burnups ranged from zero to nearly 10 at%. A main objective was to preserve, for posttest examination, the fuel conditions just before, and just after, cladding failure. It was also an objective to limit post-failure fuel motion so that the sodium loop would be able to be reused in subsequent tests.

Limitations

The availability of irradiated metallic fuel for TREAT testing was limited at the time. U-Fs fuel was used in M1 through M4 as a substitute for U-Pu-Zr fuel. Fuel in D9 cladding was used as a substitute for fuel in reference HT9 cladding. Inherent limitations of the TREAT reactor transient energy led to use of an 8-second exponential period in the power transient to ensure the heating rate became high enough to cause rapid fuel-cladding metallurgical interaction (at about 1350 K interface temperature) to be a major (or dominant) contributor to cladding failure (assisted by plenum

pressure). This heating rate was higher than prototypic for unprotected transient overpower accidents in sodium-cooled fast reactor designs of interest at that time.

Experiment Vehicle

Test M1 used a dry capsule of a type that had been used in previous tests. Laser illumination of the test samples was employed, with high-speed cameras recording the fuel response to the power transient. Tests M2 through M7 used the Mark-III sodium loop, with fuel holders designed for three pins in separate sodium flow tubes (M2, M3, and M4) or two pins in separate flow tubes (M5, M6, and M7). Orifices in each flow tube were sized such that the flow rate past each pin would achieve the desired heating objectives for each pin.

Test Fuel

Tests M1 through M4 used irradiated 316 SS-clad, sodium-bonded U-5Fs fuel of burnups 3.5 at% in M1 and of burnups ranging from 0.3 to 7.9 at% in M2 through M4. Tests M5 through M7 tested five D9-clad U-19Pu-10Zr fuel pins of burnup ranging from zero to 9.8 at% and one HT9-clad U-10Zr fuel pin of 2.9 at% burnup. All irradiations had been done in EBR-II. Active fuel columns were nominally 35 cm high.

Conditions

Test M1 applied steady power to the sample; no flowing coolant was present. Tests M2 through M7 brought power up to approximately nominal fast-reactor fuel power and then increased the power on an 8-second exponential period. The power was abruptly dropped at incipient cladding failure or immediately after cladding failure. Flowing sodium coolant was used during these six tests.

Results

Fuel extrusion was observed in all of the fuel pins tested, but varied substantially among the several fuel types. The low and medium burnup U-5Fs fuel extruded greatly prior to cladding failure, apparently because of the relatively-large amount of retained fission gas in that fuel. The high burnup U-Fs fuel and the U-19Pu-10Zr fuel extruded much less. Cladding failure occurred promptly when the fuel-cladding interface temperature reached approximately 1350 K in the U-5Fs and in the U-Pu-Zr fuel. Little fuel-cladding metallurgical interaction occurred in the U-Zr fuel pin despite reaching that range of interface temperature, apparently related to its higher fuel melting point. Initial post-failure fuel motion was consistently upward toward the failure site at the fuel-plenum interface, with ejected fuel being swept upward out of the original fuel zone. Material leaving the failure site was likely a molten alloy of fuel and cladding constituents due to the significant amount of fuel-cladding metallurgical interaction that preceded cladding breach.

Applications

Fuel extrusion modeling was aided by the wide range of fuel burnups and compositions tested, which provided insights into the relative roles of retained fission gas and in-fuel logged sodium. The 1350 K criterion for rapid penetration of cladding by fuel-cladding metallurgical interaction was confirmed by the results, although the reason for the lagging interaction in the U-Zr fuel was unclear. The few percent pre-failure extrusion of fuel (beyond thermal expansion) would cause a substantial negative reactivity effect in a large reactor undergoing an unprotected overpower transient.

Calibrations in TREAT

Experimental determination of the power coupling between the test fuel and the TREAT core was determined by irradiations in TREAT of fresh fuel pins and fissile monitor wires, which were then radiochemically analyzed. Such calibration activities were performed for test Mi (designated MI-

CAL), for tests M2, M3, and M4 (designated M2-CAL and M4-CAL), and for M5, M6, and M7 (designated M7-CAL). Later, an unusually-extensive set of neutronic calibration irradiations (designated M8-CAL) was performed to compare the neutronic coupling differences between the then-newly upgraded TREAT core and the pre-upgrade core.

8 TS-series Experiments

The TS-series tests investigated the prefailure axial fuel extrusion and the time and location of cladding failure in solid-pellet FFTF-type preirradiated MOX fuel pins during a slow (5 ¢/s) TOP [9, 10, 20, 21]. A one-page summary of each TS-series test is included in Appendix F.

Number/Time

Two TS-series tests (TS-1 and TS-2) were performed during 1983-84.

Purpose

The tests were performed to determine the time and location of failure of FFTF-type mixed-oxide (MOX) fuel pins during a slow (5 ¢/s) overpower transient.

Approach

In both tests, a single pin of FFTF design and pre-irradiated in FFTF was tested in flowing sodium. Heating of the test fuel was programmed to match the heating that would occur in that type of fuel if such an accident were to happen in FFTF. The time of cladding failure was determined by sudden changes in flow rate due to vaporization of sodium by hot fuel expelled from the cladding. The location of failure was determined by the fast neutron hodoscope and thermocouples located axially along the wall of the flow tube surrounding the fuel pin.

Limitations

The only significant non-prototypicality apparently was the effect of the thermal neutron spectrum in TREAT resulting in a test-fuel radial power profile that was strongly peaked at the periphery of the fuel, unlike in a fast spectrum core such as FFTF. (This effect is typical of experiments in TREAT.) Post-failure fuel motion characteristics, which were not among the main objectives to determine, were affected by the prompt post-failure penetration of the flowtube, with molten fuel and cladding thereafter accessing additional space beyond that barrier.

Experiment Vehicle

Both tests were performed in Single-Pin Test Loops (SPTL-Type B) designed specifically for full-length, top-plenum fuel pins, such as FFTF driver pins. In flowing sodium. The loops were instrumented with flow, pressure, and temperature sensors, as well as acoustic detectors. The loop occupied the space of two TREAT fuel assemblies. A thin-walled flowtube, instrumented with many thermocouples attached to its outer surface, surrounded the test fuel pin and provided the boundary for sodium flow adjacent to the pin. Surrounding the flowtube was an alumina sleeve that served as a barrier to further penetration by molten fuel-pin materials after cladding failure. An inert gas space was between the flowtube and alumina sleeve. Dysprosium neutron filters located near the top and bottom of the test fuel were included to shape the axial power distribution in the fuel so as to reasonably match the axial power profile in FFTF.

Test Fuel

In both tests, the pin was a helium-bonded mixed-oxide (MOX) pin of FFTF-prototypic design (91.4 cm fuel length; 20% CW type 316 stainless steel cladding of 5.84 mm OD, 0.38 mm wall thickness,

wire wrapped, solid dished pellets of 4.94 mm diameter and ~85% TD smear density, two uranium-dioxide solid pellets above and below the fuel, and a nickel reflector rod above the upper insulator pellets, with a spring holddown in the upper gas plenum). The fuel pin in TS-1 had been pre-irradiated in FFTF to a burnup of approximately 2 MWd/kg at 41 kW/m peak linear power. The pin in TS-2 had been FFTF-irradiated to ~6 MWd/kg with a linear power that was 36 kW/m at the peak burnup. Based on examination of a sibling pin from the FFTF irradiation, the test pin fuel had a central void of diameter 0.5 mm at the fuel midheight and ~0.2 mm near the ends.

Conditions

To generate test-fuel temperature versus time at a rate as representative as during a 5 ¢/s (~21 s period) FFTF overpower transient as possible, the TREAT power history was programmed to account for the changing TREAT-to-test fuel power coupling during the transient, and the sodium flow rate was adjusted upward by 10%. The reactor was programmed to scram immediately upon indication that the inlet sodium flow rate in the test vehicle dropped by 50%. The initial part of the TREAT power history (before the 5 ¢/s period began) provided about 10 s of near-nominal power to preheat the test fuel.

Results

In TS-1, cladding failure (and immediate reactor scram) occurred ~22 s into the power transient, at a test-fuel power 3.1 times the nominal 41 kW/m power. About 1.7 cm of axial extrusion of molten fuel (approximately the free travel length allowed by the plenum spring) occurred approximately 5 s before failure. In TS-2, cladding failure occurred ~24 s into the overpower ramp, at a test-fuel power 3.4 times its end-of-life level of 36 kW/m, preceded approximately 4 s earlier by ~2.5 cm of axial fuel extrusion. In both tests, the cladding failed in the upper half of the fuel column ($X/L = 0.84$ in TS-1) and rapidly penetrated the flowtube, allowing for more space into which the molten fuel-pin material and sodium could move.

Applications

The tests demonstrated that at low and medium burnups, FFTF-design MOX fuel pins can withstand a slow transient overpower event up to more than three times nominal power before cladding failure. They also showed the existence of significant pre-failure axial extrusion of molten fuel, which event was important (according to posttest analyses) in prolonging the time of failure during the tests.

Calibrations in TREAT

Neutronic/radiochemical and thermal-hydraulic calibrations, referred to as TS-CAL, were performed to determine the power coupling between reactor and test fuel as a function of time during the specific transient planned for the experiments. This information was collected from irradiations of monitor wires, a fresh fuel pin, and preliminary lower-power irradiations of the test fuel pin in the SPTL with and without sodium flow.

9 EBT-series Experiments

The EBT-series investigated and compared the response of short, fast-reactor, irradiated MOX fuel and blanket pins, with various claddings, to unprotected overpower transients up to cladding failure [9, 22, 23]. A one-page summary of each TS-series test is included in Appendix G.

Number/Time

Four tests were performed in the EBT series (assuming EBTB was considered part of the EBT series), all during the 1983-84 time period.

Purpose

Three of the tests (EBT-1, EBT-2, and EBT-3) were performed to test irradiated mixed-oxide (MOX) fast-reactor pins under unprotected transient overpower (TOP) conditions of 10 ¢/s or 50 ¢/s reactivity ramp rate (relative to the FFTF reactor) to show how the transient response of the fuel pin differs in relation to the fuel's cladding material (316 SS, D9, or HT9). The fourth test (EBTB) investigated the response of a 316SS-clad MOX blanket fuel pin when subjected to a similar unprotected TOP.

Approach

EBT-1 and EBT-2 were single-pin tests in static NaK designed to directly compare the transient response of D9-clad irradiated MOX fuel with the results of a prior TOP test (in the HUT series) on 316SS-clad MOX. All three of those tests were performed with the same test vehicle and generated virtually identical thermal conditions in each pin up to cladding failure. EBT-1 used a high-burnup pin, and EBT-2 used a medium-burnup pin. (The HUT-series test was initially designated HUT5-1B and later renamed to HUT5-2B; in the TREXR database it is referred to as HUT5-1648, reflecting its TREAT transient number.) Test EBT-3 simultaneously tested three pins, each having a different cladding material (316 SS, D9, or HT9) in flowing sodium. The EBTB test with the blanket pin was also a single-pin test in flowing sodium.

Limitations

EBT-1 and EBT-2 were run in static NaK in order to be direct counterparts to the prior companion test (which tested a 316 SS-clad pin in static NaK) although, because the coolant was static and not flowing, some loss of prototypicality in the pin's temperature distribution during the test would result. The TREAT thermal neutron spectrum caused strong power peaking near the periphery of the fuel, which is typical in TREAT tests and not representative of a fast spectrum.

Experiment Vehicle

Tests EBT-1 and EBT-2 were performed in a vehicle of the same design as used in the previous HUT-series test, i.e., a static capsule with thermal neutron shielding in which the test fuel pin was NaK bonded to a nickel heat sink. EBT-3 and EBTB were performed in Mark-IIIC sodium loops with axial neutron flux shaping. In EBT-3, each of the three pins were located inside of its own flow tube, with each flow tube thermally isolated from the other.

Test Fuel

Solid-pellet helium-bonded MOX fuel pins were used in EBT-1, EBT-2, and EBT-3, and a MOX blanket pin was used in EBTB. All of the pins had been pre-irradiated in the fast spectrum of the EBR-II reactor. The D9-clad pins in EBT-1 and EBT-2 had been pre-irradiated to 9.4% and 4.2% burnup, respectively. The three pins in EBT-3 had been pre-irradiated to burnups approximately in the 8 to 9% range, with peak linear powers of approximately 37 kW/m and peak fluence of $\sim 6 \times 10^{22} \text{ n/cm}^2$. The 316 SS-clad blanket fuel pin in EBTB had been pre-irradiated to about 3% burnup and a cladding fluence of $1 \times 10^{23} \text{ n/cm}^2$.

Conditions

In all four tests, the power transient began with a constant-power plateau lasting for several seconds (in order to preheat the test fuel at roughly the nominal pre-irradiation power level) prior to the overpower portion of the transient. The overpower portion began immediately thereafter and

rose at a rate corresponding to either a 50 ¢/s (EBT-1, EBT-2, and EBTB) or 10 ¢/s (EBT-3) FFTF-type reactor reactivity ramp insertion rate continuing until cladding failure, or incipient failure, occurred. The power was abruptly terminated by TREAT scram at a pre-determined time or upon indication of test-pin cladding failure. Thermal conditions in the test pins were affected by heat transfer to the static NaK in EBT-1 and EBT-2 and by the flowing sodium in EBT-3 and EBTB.

Results

The cladding of the pins in EBT-1, EBT-2, and EBTB failed at power levels several times their nominal power (five times nominal power in EBT-1 and EBT-2 and seven times nominal power in EBTB), followed by extensive motion of molten fuel, gross disruption of the fuel (or blanket) pin with extensive relocation of molten fuel into the coolant channel. On the other hand, none of the three pins in EBT-3 failed; their peak powers had reached the range 2.4 to 2.9 times their steady-state irradiation power. In EBT-1, the cladding of the high-burnup pin first breached near the top of the fuel column (at a power about 7% lower than in the corresponding HUT test), followed by significant fuel expulsion to above the test fuel region and significant voiding of fuel along the length of the fuel column. In EBT-2, the cladding of the medium-burnup pin first breached near the fuel midplane (at a power about 18% lower than in the HUT test), followed first by fuel loss from the top and bottom of the fuel column and then fuel movement from the midplane to the bottom of the fuel column. The blanket pin in EBTB apparently failed (at 7 times nominal power) first near the fuel midplane, with ejected molten fuel then being swept up to the top of the fuel column where it accumulated and blocked the coolant flow channel. The three unfailed pins in EBT-3 experienced significant (about 36% areal fraction) fuel melting but little or no transient-induced cladding strain.

Applications

The tests demonstrated the ability of the fuel pins to withstand the imposed power transients up to several times their nominal steady-state irradiation power. Data from the tests were useful in evaluating the predictive capability of fuel-pin behavior codes and to help support the safety basis for irradiations of advanced fuel designs in FFTF.

Calibrations in TREAT

For EBT-1, EBT-2, and EBT-3, the power coupling between the test fuel and the TREAT core was measured before each test by heating the test fuel, inside the test vehicle, using a relatively-low-level, steady power transient, measuring the resulting temperature rise of the coolant and surrounding structure, and computing the fuel power needed to produce the measured thermal effects. No separate TREAT irradiations of additional pins were involved in these calibration activities. For EBTB, a separate calibration irradiation in TREAT (EBTB-CAL), using a neutronic mockup of the EBTB test vehicle, was performed on a fresh blanket fuel pin of design similar to the EBTB test pin and also on flux monitor wires, followed by radiochemical analysis of those items to determine fission density and axial profile, thus yielding the desired power coupling information for the EBTB test.

10L-series Experiments

The L-series tests demonstrated the fuel-motion responses of pre-irradiated fast-reactor-type MOX fuel pins to several different loss-of-flow accident conditions including strong overpower ramps [24, 25, 26, 27, 28, 29, 30, 31, 32, 33] [34, 35, 36, 37]. A one-page summary of each L-series test is included in Appendix H.

Number/Time

Eight L-series experiments were performed during the time period 1970 - 1978.

Purpose

The tests were performed to measure the timing and magnitude of spatial redistributions of fuel during simulations of various aspects of a fast-reactor hypothetical loss-of-flow (LOF) accident. Behaviors of fuel of various types, under various imposed conditions, pertinent to a range of subassembly types (power level, burnup, power level) would be demonstrated in order to indicate how each type might respond to, and subsequently affect, the course of the accident. The results were to be useful in validating analyses of the loss-of-flow accident for a MOX-fueled CRBR fast reactor.

Approach

Because this test series was the first to simulate loss-of-flow conditions in TREAT experiments in sodium, and because of limited availability of pre-irradiated fuel of the types preferred (particularly prototypic-length fast-flux-irradiated pins of FFTF/CRBR-type fuel pins), and because of then-existing limitations of the TREAT control system, it was necessary for the prototypicality of the simulations to progress from test to test as experimental capabilities and fuel availability improved with time. The initial tests would need to utilize constant-power “flattop” transients, whereas shaped transients (e.g., flattops concluded with a programmed reactivity ramp to simulate a LOF-driven overpower situation) would be used when the TREAT control program was enhanced to produce them. Constant coolant flow during tests would be used until the sodium loop system became capable of providing programmed coolant flow coastdowns. Short-fuel-length irradiated pins (representing the top of the fuel of full-length FFTF-type pins) would be used until longer fuel pins became irradiated, and the longer pins would be used despite being irradiated in a thermal flux because no fast-flux-irradiated long-fuel pins would become irradiated until much later. Single pins and small bundles of pins, with pins of various burnups and pre-irradiation fuel restructuring, would be used depending upon which type of reactor subassembly would be represented in the particular test. Reactivity ramps would be included in the test if the test was intended to simulate fuel in lower-power subassemblies responding to a power transient caused by reactivity effects in the lead assemblies of a reactor.

Limitations

Aspects of the approach taken in conducting this series of tests were chosen because of the existing limitations in availability of fuel for testing and capabilities of testing facilities. These were described above. Additional limitations needed to be taken into account: some in test planning and some in posttest analyses. One was the size of the fuel sample (one, three, or seven pins), limited by safety reasons or fuel availability, which would affect how well the test sample would represent a much-larger array of pins in a reactor subassembly. Another was the ability to design a test environment of the pins such that the pins could be adequately heated to high temperatures in initially-flowing sodium with a representative fuel-to-sodium mass ratio, and with adequate instrumentation to monitor the thermal conditions of the fuel, and simultaneously with adequate fuel containment during the meltdown. The loop test section designs that were used provided for good conditions to the point of fuel dispersal. After being contacted by hot fuel, however, the steel flow tube surrounding the fuel pins readily melted and breached, opening up new lateral spaces for molten fuel, steel, and sodium to flow into, and contributing to the amount of molten steel in the meltdown mass. Experience with the sodium loop was still being developed, and problems with the loop flowmeters were initially encountered that, in some tests, significantly reduced the amount of information obtained.

Experiment Vehicle

Mark-II integral sodium loops were used in all of these tests. The loops provided the flowing sodium and the thermal and pressure containment of the test fuel. Thermal-neutron-absorbing layers (flux “shaping collars”) were located around the loop periphery to provide a suitable axial power profile in the test fuel. Inside the loop test section was the test train that held the test fuel sample and provided the sodium flow area and flow perimeter to best represent conditions in a large assembly of pins. Test L1 used a single-pin test train. The flow perimeter in the seven-pin-bundle tests (L2, L3, and L4) and the three-pin-bundle tests (L5 through L8) was a fluted stainless steel tube surrounded by an inert-gas space. The flutes in the tube were geometric representations of adjacent fuel pins in a larger array.

Test Fuel

All of the tests used FFTF-type helium-bonded mixed-oxide fuel pins but with fuel regions shorter than standard FFTF fuel pins. The pins in L1 and L2 were not pre-irradiated. Tests L3 and L4 used fast-spectrum-irradiated, medium-burnup pins; in L3 the fuel had been low-power irradiated, whereas in L4 the fuel had been high-power-irradiated. Fuel pins in tests L1 through L4 had fuel columns 34.3 cm high (the height of the EBR-II core in which the pins in L3 and L4 had been pre-irradiated). Pins for tests L5 through L8 were all pre-irradiated in the thermal flux of the GETR reactor and had fuel column heights of 86.4 cm, only 5% less than the FFTF core height. For L5 through L7, the pre-irradiation of all of the fuel pins resulted in moderate-power fuel structure; their burnup was 8% for L5 and 3% burnup for L6 and L7. Test L8 used pins like those in L6 and L7 except for having been irradiated at low power.

Conditions

Without a capability for shaped power transients or coolant flow coastdown at the beginning of this series, and being an initial (therefore exploratory) LOF test, test L1 subjected only a single, unirradiated pin to a rough simulation of a LOF condition by exposing the pin to a succession of steady-power runs, each with a lower coolant flow rate than the previous run, with the intent of closely approaching, but not exceeding, the cladding failure threshold. Test L2 included the improvements of testing a seven-pin bundle with a coolant coastdown, still using fresh fuel and a constant-power run with flow coastdown, with conditions exceeding cladding failure. Flow coastdowns were used in all of the subsequent tests. Tests L3 and L4 added the feature of using fast-spectrum pre-irradiated fuel pins (with 34.4 cm fuel height as in L1 and L2), again in seven-pin bundles and with a constant-power run in each test. Tests L5 through L8 also used pre-irradiated fuel, but now with 84.4 cm fuel height (although with thermal-spectrum pre-irradiation), and significantly ramped up the power after a few seconds at steady power. In L5 the peak power reached about six times nominal, in L6 about ten times nominal, in L7 about 23 times nominal, and in L8 about 75 times nominal. The strong overpower parts of the transient in the last four tests all generated about the same amount of TREAT energy and thus (if the power coupling did not change as the test fuel dispersed) about the same amount of test fuel energy. In each test, an objective of avoiding the generation of significant fuel vapor pressure was achieved.

Results

In L1, unexpected sodium boiling apparently occurred, possibly under the spacer wire wrap of the pin and the adjacent in-channel thermocouple, causing severe flow anomalies and probably strong azimuthal and axial temperature gradients in the pin. The pin remained intact but was highly distorted. In contrast, the seven fuel pins in test L2 (the first test performed in TREAT in which a fuel pin was destroyed by exposure to loss-of-flow conditions) were highly disrupted, with gross movement of molten fuel and steel and formation of upper and lower flow blockages that posttest

appeared to be complete. Gross disruption of the original fuel region and accumulation of molten steel and fuel near the top and bottom of the original fuel region likewise characterized all of the subsequent tests. Generally much of the flowtube adjacent to the fuel regions was also melted, thus contributing to the overall molten fuel mass and allowing molten material to flow into the surrounding gas-filled space. Details of fuel motion were provided by the fast neutron hodoscope and varied considerably from test to test (and thus probably also did the steel motion). Final fuel and steel configurations also varied in their details, but generally little mixing of the two materials was found posttest.

Applications

The fuel motions observed in the later tests in the series were evaluated in terms of the reactivity changes they would produce in a large fast reactor such as CRBR. For this purpose, selected CRBR fuel worth axial distributions were used. This provided a means by which the results of accident analysis models and codes could be compared to the test results. The SAS accident analysis code was developed through its various editions at the time (SAS1A, SAS2A, SAS2B, SAS3A, and SAS3D) in parallel with the L-series tests as the two programs collaborated with and assisted each other. A combination of the predictive analyses and empirical results from the test provided a basis for the development of scenarios describing the sequence of key LOF events in each test.

Calibrations in TREAT

Typically the final transient in each test was preceded by a heat-balance transient at a reduced, steady power and nominal flow rate in order to experimentally verify the power coupling between the test fuel sample and the TREAT core. The later tests in the series involved neutronic calibrations performed by TREAT-irradiation and radiochemical analysis of a fresh fuel pin representing the pins to be transient tested, thereby determining the absolute power coupling between the fuel and TREAT core. Pellets from the calibration pins were also core-drilled to obtain samples for radiochemical analysis to determine radial fission profiles.

11 PINEX-series Experiments

The PINEX tests investigated the potential for in-pin axial fuel motion in special, annular-design low-burnup MOX fuel pins undergoing 3 \$/s and 50 ¢/s unprotected reactivity-ramp power excursions [38, 39]. A one-page summary of each PINEX-series test is included in Appendix I.

Number/Time

Two PINEX-series tests, PINEX-2 and PINEX-3, were performed in 1978.

Purpose

The tests were performed to investigate the potential for in-pin pre-failure axial fuel motion to significantly mitigate severe hypothetical transient overpower accidents in mixed-oxide fueled fast reactors. A second purpose of the tests was to evaluate the capability of the pin-hole imaging system to monitor the motion of nuclear fuels in TREAT experiments.

Approach

Fuel pins were designed with an axial hole through the fuel, through the two insulator pellets above the fuel, and through the reflector above the insulator pellets, in order to provide a pathway for molten fuel to escape from the fuel region prior to cladding failure. The timing and extent of axial fuel motion was to be determined in each of two tests that differed only in the rate of power rise during the overpower burst, i.e., one representing a 3 \$/s unprotected reactivity ramp in FFTF and the other representing a 50 ¢/s ramp rate. The relationship between TREAT power and sample

power was established such that the TREAT power level would rise high enough during the transient (above 1000 MW) to permit adequate signal strength to the pinhole (PINEX) fuel detection system that was to be evaluated.

Limitations

The thermal spectrum used in the pre-irradiation of the test fuel samples created a different radial distribution of fuel restructuring and fission-gas retention than a fast spectrum would have generated. In addition, the thermal spectrum in TREAT caused the effect (typical in TREAT tests) of depositing heat preferentially near the periphery of the test fuel sample. These effects affected the timing and progress of fuel melting and fission gas release in the test fuel pin that differed from what would have occurred in a fast spectrum. The effects would be expected to differ between the two tests due to the different power rise rates and the resultant different progression of the radial temperature profile during the overpower part of the transient. In addition, due to the static coolant, the axial temperature gradient in the fuel was not peaked toward the top of the fuel as it would have been in a flowing coolant environment, and this would have affected the axial progression of fuel melting above the midplane. The extent to which these non-prototypicalities affected the outcome of each of the tests either qualitatively or quantitatively was addressed by analyses.

Experiment Vehicle

Both tests used the same type of test vehicle designed to contain a single fuel pin in a static NaK coolant environment surrounded by a thick nickel heat sink. Thermal-neutron attenuation was provided by boron-containing layers around the outside of the vehicle. Heaters located axially adjacent to the test fuel provided an initial axial temperature profile. The vehicle was designed such that it would house the fuel pin during the pin's pre-irradiation in the General Electric Test Reactor (GETR) and also serve as the test vehicle during the TREAT transient test.

Test Fuel

Identical specially-designed helium-bonded mixed-oxide fuel pins (25 wt% Pu, 75 wt.% U) were used in both tests. The fuel pellets were annular, with 0.81 mm diameter fabricated central hole. The pellet stack was 86.4 cm high. Above the stack were two annular UO₂ pellets having 1.78 mm ID. Above the insulator was a 49-mm long nickel rod with 1.70 mm diameter central hole. The fuel pins were nominally of FFTF design, with Type 316 SS cladding (20% cold-worked) of 5.84 mm outer diameter and 0.38 mm thickness. The fuel pins were pre-irradiated in the GETR thermal-spectrum reactor to approximately 2 at% burnup.

Conditions

Each test was designed, within the limits of the TREAT reactor and the test vehicle that was used, to produce test fuel thermal conditions associated with unprotected reactivity ramp excursions in FFTF as closely as practical, specifically a 3 \$/s ramp in PINEX-2 and a 50 ¢/s ramp in PINEX-3. At the beginning of each test, the fuel was preheated at nominal power for a few seconds, followed immediately by a power rise corresponding to the desired ramp rate. In each test, the TREAT reactor was programmed to scram at the predicted time of initial fuel melting, anticipating that the axial fuel motion that would begin at that time would, if it were to happen in the core of a fast power reactor, cause a negative reactivity insertion and power reversal. The stagnant NaK environment, compared to a flowing-coolant environment, of the pin was deemed to be an acceptable substitute in these tests. TREAT power during each transient rose above the 1000 MW level needed for acceptable signal to the PINEX fuel detector system.

Results

The fuel pin transient behavior differed substantially between the two tests, presumably due to the difference in how the different heating rate (and heat loss rate to the coolant) caused the fuel melting progression into fission-gas-containing regions of the fuel. Fuel in PINEX-2 became free to move axially within the fuel central hole axially past the top of the fuel and into the insulator, reflector, and plenum. This apparently decreased the temperature and pressure loading on the cladding, and the cladding remained intact. In contrast, with the slower transient in PINEX-3 it appeared that the central void near the top of the fuel did not open to allow significant axial fuel movement through that area, and gross disruption of cladding and fuel resulted. The PINEX detector system provided fuel location information useful in the analysis of the test, supplementing the fuel-motion information provided by the TREAT hodoscope.

Applications

The ability of MOX fuel to flow axially, prior to pin failure, within annular-design fuel pins as used in these tests was demonstrated to be feasible, but dependent upon the particular heating conditions and retained fission gas distribution in the fuel. Results from the real-time fuel monitoring data and from detailed posttest analysis of the intact fuel pin from PINEX-2 yielded particularly valuable information for comparison with code predictions.

Calibrations in TREAT

Power coupling between the test fuel and TREAT core was experimentally determined by measurements taken during low-energy transient irradiations of the actual test fuel in the test vehicle. No separate irradiations of calibration fuel pins were made. (The PINEX-3A test was subsequently performed to explore the sensitivity of the pinhole-intensified imaging system to low fuel specific power levels.)

12RFT-series Experiments

The RFT tests showed that FFTF driver fuel pins can withstand unprotected overpower transients well beyond the secondary PPS (Plant Protective System) limit and provided data for calibrating transient fuel behavior codes [10, 40, 41, 42, 43, 44]. A one-page summary for each RFT-series test is included in Appendix J.

Number/Time

There were four tests in the RFT series, performed between 1982 and 1984. In addition, a preliminary test RFT-CAL-L was performed to provide data on power calibration, loop thermal-hydraulics, and fuel pellet design impact on cladding strain.

Purpose

The objectives of the RFT series were: (1) to demonstrate the capability of FFTF reference driver fuel pins, irradiated in FFTF, to accommodate an FFTF secondary plant protective system (PPS) - terminated overpower transient without damage, (2) to verify that a significant margin to failure exists by extending the transient overpower beyond the secondary PPS limit of 1.25 times normal rated power, and (3) to establish transient-induced measurable changes in the test pins as a basis for fuel pin performance code correlation, improvement, and validation. Responses of FFTF driver fuel pins of various burnups and irradiation power levels were evaluated for reactivity ramp rates of 5 ¢/s, 50 ¢/s, and 1 \$/s.

The objectives of RFT-CAL-L were to provide data needed for pretest analysis and planning of the RFT tests, i.e., regarding power coupling between FFTF driver fuel pins and the TREAT core and regarding sodium loop thermal-hydraulics for calibrating the loop thermal-hydraulic model, and also to provide data on fuel pin transient performance for use in calibrating fuel pin transient performance codes.

Approach

In the RFT series, fuel pin responses to unprotected overpower transients would be tested and evaluated for several fuel burnup and irradiation-power levels and for several reactivity ramp rates. Test conditions were set to closely match code-predicted fuel and cladding temperatures during the transient, and to reach conditions involving substantial fuel melting, measurable cladding strain, but no cladding failure. To best match the thermal transient in the test fuel and cladding with the thermal transient analytically computed for the FFTF unprotected reactivity excursion, both the test-fuel power and the coolant flow rate were specified to be higher (generally 10-15%) than the conditions which the pins experienced in FFTF; this adjustment helped to compensate for the depressed radial power profile in the test fuel due to the TREAT thermal neutron spectrum. The transients were to continue long enough to cause significant fuel melting and cladding strain but not to reach cladding failure conditions. Posttest examination of the pins would provide key data for comparison with analytical code predictions of fuel pin thermal-mechanical response.

Limitations

The thermal neutron spectrum in TREAT produces a large radial flux depression toward the center of test fuel samples, in contrast to the profile that is present in a fast spectrum. This effect on the fuel pin temperature history during the transient was reduced by adjusting the starting conditions of fuel temperature and power from their nominal FFTF conditions.

Experiment Vehicle

All of the tests in this series used Mark-III integral sodium loops containing a three-pin test train in which each pin is located in a separate flowtube. Instrumentation included flowmeters on the loops and numerous thermocouples on the test trains. Flux shaping collars on the loops were included to provide a suitable axial power profile in the test fuel.

Test Fuel

In the four RFT series tests, all of the fuel pins were FFTF helium-bonded MOX driver fuel pins that had been irradiated in FFTF. In RFT-L1 and RFT-L2, the pins had been irradiated to very low (about 0.2%) burnup, three pins at 26-29 kW/m and three at 41 kW/m. The pins in RFT-L3 and RFT-L4 had been irradiated in FFTF at 35-38 kW/m to two burnups (2.6% and ~5.3%). Both types of pins were exposed to a 5 ¢/s overpower transient (in RFT-L3) or to a 1 \$/s transient (RFT-L4).

In the RFT-CAL-L test, all three pins were unirradiated. Two were FFTF driver pins. The third was a specially-designed MOX fuel pin with fully enriched uranium and several fuel sections of various pellet inner and outer diameters and fuel density, and one section was made with a radial slot.

Conditions

In the RFT series, the 0.2% burnup pins of each irradiation power were subjected to a 50 ¢/s overpower in RFT-L1 reaching a peak power of about 5 times nominal; in RFT-L2 pins of that burnup were subjected to a 5 ¢/s overpower reaching 3.4-3.7 times nominal in the lower-power-irradiated pins and 2 times nominal in the higher-power-irradiated pin. In RFT-L3 and RFT-L4, pins of both burnups were subjected to either a 5 ¢/s overpower reaching 1.8 times nominal or to a 1 \$/s overpower reaching about 6.8 times nominal.

In the RFT-CAL-L test, the fuel was initially subjected to a 1 \$/s overpower spike (to produce fuel melting and fuel-cladding interaction in the specially-designed fuel pin), and then was subjected to two lower-power transients to build up fissions needed for radiochemical analysis.

Results

All of the RFT tests produced substantial fuel melting, and measurable cladding strain occurred in many of the tests. All of the pins survived to power levels well beyond the secondary PPS limit. The failure threshold was exceeded in one of the pins (in RFT-L1). Detailed posttest characterization of the test pins was performed for use in comparison with analytical predictions.

In RFT-CAL-L, the special fuel pin underwent fuel melting and cladding strain, the latter being a strong function of the fuel design features (annular, fuel-cladding gap size, fuel density, presence of radial slot). Neutronic and thermal-hydraulic calibration data were obtained for use in preparing the RFT tests.

Applications

The test data were used for detailed comparison with code-calculated predictions of fuel melting and cladding strain.

Calibrations in TREAT

In addition to the RFT-CAL-L test, heat-balance transients at constant test-fuel power in TREAT were performed before each test to determine the power coupling between the test fuel and the TREAT core. During these transients, the test-fuel temperatures were kept below their prior irradiation temperatures in FFTF. In some cases, flux monitor wires were attached to the outside of one or more flowtubes for use in measuring the axial flux profile posttest.

13STEP-series Experiments

The STEP-series tested 4 four-pin bundles of short, pre-irradiated LWR pins to failure in flowing steam and collected and characterized volatile fission products transported downstream [45]. A one-page summary of each STEP-series test is included in Appendix K.

Number/Time

Four Source-Term Experiments Project (STEP) tests STEP-1 through STEP-4 were conducted during 1984-1985 by an international consortium headed by EPRI and including Ontario Hydro of Canada, US Department of Energy, US Nuclear Regulatory Commission, and Belgonucleaire.

Purpose

The tests were performed (a) to characterize fission products and structural materials that may be released from light-water reactor (LWR) fuel during postulated risk-dominating pressurized-water reactor (PWR) and boiling-water reactor (BWR) severe accidents and transported downstream, and (b) to generate and collect data on associated fuel-rod heat-up, cladding oxidation, and failure. The intended characterization focused on the physicochemical properties of the biologically-important volatile fission products released early in such transients.

Approach

In each test, pre-irradiated fuel pins were subjected to a flowing steam environment at pressures and temperatures which, at the time of release of volatile fission products and noble gases from the pins, would be consistent with a particular accident scenario. Various types and orientations of

surfaces were provided downstream upon which fission products could collect during the tests. Deposited volatile fission products would be examined and characterized posttest.

Limitations

With the relatively-short test fuel, conditions were simulated for the time during the reactor accident scenarios when the water level has fallen below the axial mid-height of the fuel. The means of aerosol sample collection was limited by the available space within the vehicle. A small pin-bundle was used to represent full-size pin arrays, requiring a flow tube with high surface area per pin. Fuel heating rate in STEP-2 had to be higher than desired because of facility limitations.

Experiment Vehicle

A square-arrayed four-pin bundle in each test was surrounded by a high-density zirconia tube within which steam flowed past the fuel pins. The steam was generated by an ex-pile system. After the steam passed through the pin bundle and fission-product collectors it was received and condensed ex-pile. The sample collection tree included many coupons of a wide variety of materials. Some coupons were parallel, and some perpendicular, to the flow. Two aerosol canisters were mounted alongside the pin plenum region above the fuel columns. Each canister contained three chambers, each of which was opened for a predetermined time interval during the test. Each canister contained 14 sample collection stages, each of which contained fine wire impactors, a settling plate, and a sample coupon or additional settling plate. The flowtube was instrumented with platinum-platinum rhodium thermocouples. Hydrogen partial pressure was also monitored.

Test Fuel

In all of the tests, the fuel pins were composed of a 1 m-high stack of UO_2 pellets, clad in Zircaloy-4, and pre-irradiated in the Belgian BR-3. Peak burnups were about 35 GWd/T in the pins used in STEP-1, -3, and -4 and about 31 GWd/T in the pins used in STEP-2. During the preirradiation, the fuel power level had been relatively low for the fuel in STEP-1, intermediate for the fuel in STEP-2, and high in the fuel for STEP-3 and STEP-4. The fuel had cooled for several years prior to being tested in TREAT. These test fuel elements simulated the central portion of much-longer PWR or BWR elements.

Conditions

STEP-1 simulated the conditions of a large-break loss-of-cooling-accident (LOCA) with assumed failure of the emergency core cooling system (ECCS), the “AD” sequence; it was thus performed at low system pressure. STEP-2 simulated BWR conditions due to failure of both the high-pressure ECCS and the long-term decay heat removal system, the “TQUW” sequence; it was also performed at low system pressure. STEP-3 and STEP-4 both simulated conditions resulting from transients in PWR due to failure of feedwater systems combined with failure to recover electric power, the station blackout “TMLB” sequence; both were performed at high system pressure (about 8 MPa). A simulated PWR silver-indium-cadmium control rod was included in STEP-4. The incoming steam in all tests was at 644 K (700°F) and at a flow rate predicted for the accident being simulated. TREAT power was provided over approximately 20 minutes at a level that would cause test-fuel fission power needed to simulate decay heating and to offset heat losses from the fuel-pin bundle. A novel TREAT reactor control scheme was developed for these tests, involving both automatic control and manual control of the transient and control rods.

Results

Calculated maximum fuel temperatures were about 2900 K in STEP-1, 2700 K in STEP-2, 2200 K in STEP-3, and (with the power reduction caused by the simulated control-rod material) considerably

lower than 2200 K in STEP-4. In STEP-3 and -4, the temperatures were much lower than expected and indicated that heat had been removed upward much more than in STEP-1 and -2. Whereas pin damage in STEP-1 and -2 was severe throughout the central portion and top portion of the fuel column, in STEP-3 and -4 it was mostly limited to the upper third of the fuel column. Heat from the hydrogen generation corresponded to cladding oxidation of (approximately) 90% in STEP-1, 70% in STEP-2, 40% in STEP-3, and 30% in STEP-4. Fission products that were collected on the sample tree were cesium, iodine, tellurium, molybdenum, and rubidium. Fuel element materials tin, zirconium, and uranium were also collected. Greater amounts of fission products were released in the low-pressure tests (STEP-1 and -2) than in the high-pressure tests (STEP-3 and -4) because strong natural-convective cooling under the higher pressure resulted in a lower fuel temperature. In STEP-1 and -2, released material increased the flow-path resistance in the main exit steam line, which caused increases in system pressure. Cesium, rubidium, and a trace of iodine were the only fission products found in the STEP-3 deposits, and none were found in STEP-4 deposits. The tests provided detailed information regarding aerosol concentration and size distribution for conditions accurately simulating the early phase of severe LWR accidents. The chemistry of the deposits was complex. Information was obtained on the collocation, chemical form, and morphology of cesium and iodine and the other detected fission products. It was observed that a marked tendency exists for component elements of ceramic and stainless steel structures to be volatilized and transported in high-pressure steam but much less in low-pressure steam.

Applications

The test results enhanced the database for testing and improving source-term models in accident analysis codes: they identified volatile fission products and extended the body of nuclear aerosol characterizations, they provided data regarding the magnitudes and release rates of fission products from degraded fuel elements as well as the physical and chemical characteristics of the released fission products, and they provided information on aerosol formation and transport that can be applied to transport mechanisms of the volatiles and condensed aerosols. The results were compared with computations of release masses, compositions, and fuel temperatures at time of release.

Calibrations in TREAT

Numerous low-power and transient irradiations of flux monitor wires, and low-power irradiations of four calibration fuel pins were performed in TREAT prior to the STEP tests in order to determine the power coupling between the TREAT core and the test fuel during the planned test transients.

14R-series Experiments

The R-series tests demonstrated the coolant, cladding, and fuel dynamics during unprotected loss-of-flow and transient overpower conditions with unirradiated, FFTF-type oxide fuel pins in sodium flow [46, 47, 48, 49, 50, 51, 52, 53, 54, 55] [56, 57, 58, 59, 60, 61, 62, 63, 64, 65] [66, 67, 68, 69]. A one-page summary of each R-series test is included in Appendix L.

Number/Time

Eight R-series experiments (R3 through R8, R9, and R12) were performed during the time period 1973 to 1977.

Purpose

The tests were performed to provide data to validate SAS code predictions of the timing and sequence of coolant boiling and voiding, cladding melting and runoff, and fuel melting and relocation during simulated FFTF (FTR) fast-reactor unprotected loss-of flow (ULOF) and 50 ¢/s

unprotected transient overpower (UTOP) conditions using beginning-of-life (actually zero burnup) full-length fuel pins in a flowing-sodium environment.

Approach

The test series utilized a new TREAT experiment system (a “once-through” sodium loop) in which the sodium flow rate could be programmed to follow a prescribed coastdown timed with the TREAT power transient. Because the Mark-I and -II package-type sodium loops were not long enough to accommodate the full-length test fuel pins, and because the use of fresh (unirradiated) fuel would not require double containment for safety purposes, a fundamentally-different test-vehicle design was used, one that involved an ex-core sodium supply and receiver system connected to a simple in-pile piping loop. Because the vehicle design was new, preliminary calibration and check-out testing was performed (referred to as “tests” R1 and R2). Instrumentation was provided in the test vehicle to detect the onset of sodium boiling, the coolant voiding dynamics, fuel pin failure, fuel-coolant interaction pressure, and fuel motions. The test series was planned to sequentially build upon the results of previous tests in the series so as to provide increasing prototypicality from test to test. Thus, test R3 was a single-pin proof test of the experiment system; the subsequent tests would use seven-pin bundles. Test R4 would show the complete ULOF meltdown sequence, whereas test R5 would show only up to the cladding relocation phase. Test R6 repeated test R4 to gain fuel motion details that were not obtained in R4, in order to support the planning of R7. Test R7 included a power burst timed to occur when fuel motion began, to represent the neutronic effect of the fuel motion. In test R8, three of the pins were pressurized to represent fission-gas-release effects after cladding failure. Tests R9 and R12 (there were no R10 and R11 tests) investigated the effects of a 50 ¢/s UTOP, with R9 driving the fuel well beyond cladding failure and R12 terminating the test promptly upon cladding failure.

Limitations

The use of fresh fuel, although not representative of fuel during its in-reactor life, provided the advantage of not involving contributions to the complexity of severe-accident progression due to fuel burnup. The reduced simplicity would be advantageous to validating the accident analysis modeling that was in its early stages of development. Utilization of a small bundle of fuel pins resulted in the usual loss of prototypicality involved in TREAT experiments, compared to behaviors in full-size subassemblies of pins. Also, as typical of TREAT testing, relocation of test fuel during a transient changes the power coupling between the test fuel and the TREAT core, making it difficult to estimate the power per unit mass in the test fuel after fuel motion begins, as the fuel density affects the self shielding. In the R-series, the flowtube in each test was a thin-walled stainless steel tube, which was readily penetratable by hot fuel. When such penetration occurred, fuel-pin material became free to enter the additional volume outside the flowtube. In addition, melting of the flowtube added steel to the mass of molten cladding in the fuel-steel system being tested. Furthermore, when the flowtube breached, the non-condensable gas outside the flowtube wall generally entered under pressure and affected the on-going molten-material dynamics. Also, in all but one of the ULOF tests, no power effect from neutronic feedback from early coolant and cladding motions was included.

Experiment Vehicle

The test vehicle was the R-series loop designed specifically for these tests. The in-core part of the system was a long, pipe loop, with one leg sized to accommodate a seven pin bundle of FFTF-type fuel pins. Enough space was provided radially around the fuel pins and flow channel to accommodate thermocouples and a molybdenum heat barrier to protect the stainless steel primary containment pipe wall. Flow meters and pressure transducers (and, in the later tests, acoustic sensors) were also provided along the in-core test section. The ex-core part of the system consisted

of sodium supply and receiver tanks and the associated equipment for heating, flow control, etc. The flowtube in each test was stainless steel with 0.5 mm wall thickness, of circular cross section for the single-pin test R3 and hexagonal for the other tests. Radially surrounding the flowtube was a thick molybdenum tube. In test R3, the Mo tube was 9.4 mm radially from the flowtube; in the other tests it was only about two to three mm from the flowtube. In every test, the intermediate annular volume between the flowtube and Mo tube was filled with non-condensable gas.

Test Fuel

All of the tests used unirradiated FFTF-type fuel pins, having 91.4 cm of unirradiated UO₂ fuel, helium-bonded to type 316 stainless steel cladding. The UO₂ in the single fuel pin in R3 and in the six peripheral fuel pins in the seven-pin bundles of the other seven tests was 14% enriched; the UO₂ in the central pins in all of the seven-pin bundles was 20% enriched to compensate for the lower power couplings in those central pins. Two unenriched UO₂ pellets (together 1.3 cm high) were located at each end of the fuel column, separating the fuel from 14.4 cm-long Inconel reflector rods at each end. Above the upper reflector rod was a spring-loaded plenum tube within the cladding, and then the 2.5-cm top end cap. Below the lower reflector rod was the 3.5-cm lower end cap. The overall length of the fuel pin (element) was 237 cm. In test R8 (only), three of the seven pins were pressurized to represent fission gas pressure.

Conditions

The initial thermal-hydraulic conditions in each test were selected such that, within the first three or four seconds after reaching nominal power, temperatures in the fuel and coolant would approximately represent normal operating conditions in a high-power FFTF subassembly. For the ULOF tests, this was following by a programmed reduction in sodium flow rate, but (except for test R7) with no change in TREAT power until scram. Test R7 included a power excursion to represent the reactivity effect of fuel slumping. For the UTOP tests, on the other hand, the initial preheat was followed by a power transient simulating an FFTF 50 ¢/s reactivity insertion, and no change in applied inlet sodium pressure was made. The timing of test termination by reactor scram depended upon the objectives of each individual test.

Results

The results of the tests are described in terms of the timing of the events characteristic of the sequences (boiling onset, flow reversal, cladding failure, cladding and fuel motions, inlet and outlet flow blockage formation, pressure pulses) and final condition and distribution of materials. In the ULOF tests R3 through R7, cladding failures occurred between 0.9 and 1.2 s following inlet flow reversal, and flowtube failure occurred 2.0 to 2.5 seconds after cladding failure. In ULOF test R8 (with three of the pins pressurized), cladding failure occurred 0.7 s after flow reversal, and the flowtube failed 0.08 s thereafter. In all of these ULOF tests, 30% to 100% of the flowtube in the test melted and joined the rest of the molten material. Complete inlet flow blockages occurred in all of the ULOF tests, and outlet flow blockages occurred also in tests R4, R5, and R7. The inlet flow blockages were largely once-molten steel, varying in tests R4 through R8 from 8 to 20 cm height (thickness). In R3, R4, and R5 there remained some segments of pellet stacks. In R7 (with the power spike) most of the fuel, and all of the adjacent cladding and flowtube, had melted. The overpower transient in UTOP test R9 continued approximately one second after inlet flow reversal occurred, causing more than 90% of the fuel to melt and 53% of the fuel to move upward beyond the original fuel region (as high as 47 cm beyond), and resulted in a 27-cm-high accumulation of material blocking the outlet and also a blockage at the inlet. In contrast, test R12 was programmed to automatically terminate during inlet flow reversal, resulting in minor failures of all seven pins, localized near the top of the active fuel region, and 90% of the fuel remaining within the cladding.

Applications

The tests were planned and their results were evaluated using SAS code computations. This is a frequent approach when the complex interacting phenomena that will occur in the TREAT experiments are not well known, understood, or predictable. The applicability of the test results in validating the computational models was limited to the extent that the models were able to describe certain aspects of the experiment that the models were never intended to address. Such aspects are particularly the effect of flowtube breach and melting, introduction of additional void space and pressurized non-condensable gas into the test fuel region when the flowtube failed, and unknown changes in fuel power density as the fuel moved. Nevertheless, the tests provided experimental evidence of important features involved in the two major accident categories simulated, and behaviors occurring early in the disruptive phase of the tests were significantly relevant to the models.

Calibrations in TREAT

Test R1 provided radiochemical data on sample power and power distributions.

Sample power was obtained from radiochemical measurement of the barium-lanthanum activity in sample pellets. Axial power shape was obtained from foils placed along the axis of the test train. Foils on the surface of the pin were used to obtain the radial maximum-to-average power ratio. Test R2 was a heat-balance run made to confirm average calibration data and to check out the flow-decay characteristics of the system. Additional calibrations (M8CAL) obtained further information important to both the ULOF tests and the UTOP tests M9 and M12 regarding axial and azimuthal power shapes at the test fuel, and included neutronic effects of a recent change in the TREAT core slotted elements.

15E-series Experiments

The E-series tests investigated fuel dispersal and fuel thermal energy to coolant work conversion of fresh and preirradiated oxide fuel pins failing into flowing sodium in \$3/s FTR overpower transients [70, 71, 72, 73, 74, 75, 76, 77, 78, 79] [80, 81, 82, 83, 84, 85, 86]. A one-page summary of each E-series test is included in Appendix M.

Number/Time

Seven tests (E1 through E4, E6 through E8) were performed during 1969-1974.

Purpose

The first tests in the series were intended to investigate the potential for energetic oxide fuel-coolant interaction (causing significant fuel-energy to work conversion ratio) during an unprotected LMFBR transient overpower excursion in a flowing-sodium environment beginning with intact fuel pin(s). After those tests demonstrated that energy conversion was small, more-prototypic testing was performed to simulate LMFBR \$3/s transient overpower hypothetical accident conditions relevant to the FTR/FFTF.

Approach

The first two tests (E1 and E2) were intended to indicate whether fuel-energy to work conversion would be significant when oxide fuel pins failed in a flowing-sodium environment. Both tests used natural-burst TREAT power transients to heat single, small, unirradiated, UO₂ fuel elements. In both tests, the fuel pin was surrounded by six hollow dummy pins. E1 was intended not to drive the fuel pin much (if any) beyond failure. Because the cladding did not fail but instead allowed extensive, in-pin axial fuel motion, a modification to the pin design and additional transient energy were used for test E2, resulting in nearly-complete fuel release from the cladding and highly fragmented fuel

debris, but very low energy conversion. Test E3 subjected a bundle of three, high-gas-content preirradiated UO₂ fuel pins to a natural burst transient more energetic than in E1 and E2, to include the potential for fission gas to fragment fuel before the fuel was expelled from the cladding. The test resulted in considerably-more fuel-pin damage and higher energy conversion than in E2, but the conversion efficiency was still low (~0.3%). Test E4 was intended to be similar to test H2 by subjecting a single, fresh, mixed-oxide pin to an overpower transient, but E2 used a much-faster power rise than the rise in H2. Test E6, involving a seven-pin bundle of mixed-oxide pins (one preirradiated pin surrounded by six fresh pins), was essentially identical to test H4 in that both tests used a “shaped” transient that included a short pre-heat segment followed by an overpower burst, but the burst in E6 simulated a hypothetical \$3/s reactivity accident in FFTF/FTR whereas the burst in H4 simulated a 50 ¢/s accident. Test E7 basically repeated E6 but with all seven of the pins in the bundle having been preirradiated, characterized by a high-power microstructure. Test E8 was nearly identical to E7 but tested a seven-pin bundle of low-power-microstructure preirradiated pins.

Limitations

The sodium loops used in the tests were not specifically designed to facilitate posttest computations of the work done by postfailure coolant dynamics caused by fuel-to-coolant heat transfer. This, combined with some questionable flowmeter data, led to significant uncertainties in the hydraulic analysis, but since the energy conversion was found to be consistently low, the level of uncertainty did not affect the general conclusion that energetic fuel-coolant interactions would not be a significant safety issue for the range of conditions studied in the tests. The test series was generally exploratory. Prototypicality (representation of subassembly conditions) regarding post-failure fuel motion was limited but was improved in the later tests E6-E8 by including additional coolant flow channels (initially using dummy “pins,” progressing to a preirradiated pin surrounded by six fresh fueled pins, then to seven fueled pins). Nevertheless, because only short fuel pins were available, and because the loop design could accommodate only short pins, the hydraulic prototypicality in the tests (radially and axially) remained limited. For example, in test E6 the flow area per pin was 30% larger than in an FTR assembly, and the axial pressure gradient was about 10% of that in an FTR assembly, deficiencies which significantly affected the prototypicality of post-failure coolant motions. In addition, the ability of hot fuel to readily melt the steel flowtube caused non-prototypic addition of molten steel (adding to melted cladding and spacer wires) to the system. Melt-through of the flowtube opened pathways for through which fuel and molten steel could move laterally into surrounding void spaces, and allowed contact between the molten materials and the surrounding, relatively-cool, steel outer wall. Such effects influenced postfailure material motions, freezing, and blockage formation. Use of a combination of both fresh and preirradiated fuel pins within the same bundle caused diminished the prototypicality of the postfailure material motions (although final distribution of the pre-irradiated fuel was determined by radiological measurement).

Experiment Vehicle

All seven tests were performed in Mark-II integral sodium loops designed for TREAT meltdown experiments on up to seven-pin bundles of EBR-II-length fuel pins. Test trains (containing the experiment fuel pin(s), the flow tube surrounding the pins, and associated structure) were inserted into the main vertical pipe of the loop. Each test train was designed specific to the number and type of fuel pins to be used in the test: one in E4, three in E3, and seven in the other tests. The loops (above and below the region occupied by the test fuel) a flowmeter, pressure transducer, and thermocouples. Additional thermocouples were included in the test trains. The loops were positioned at the center the TREAT core such that the midheight of the test fuel was located at the midheight of the core.

Test Fuel

Tests E1, E2, and E3 used unirradiated UO₂ fuel. The fuel column height was 305 mm in E1 and E2 and 145 mm in E3. Tests E4, E6, E7, and E8 tested U-Pu oxide fuel (MOX) fuel. Only one, unirradiated, pin was tested in E4. In E6 a single preirradiated pin was surrounded by six fresh pins. Seven-pin bundles or preirradiated pins were tested in E7 (high-power fuel microstructures) and E8 (low-power fuel microstructure). The fuel column heights were 305 mm in E1 and E2, 145 mm in E3, and 343 mm in E4, E6, E7, and E8. In all of the tests the fuel was helium-bonded to stainless steel cladding (OD = 7.37 mm in E1 and E2, 4.42 mm in E3, and 5.84 mm in E4, E6, E7, and E8).

Conditions

All of the tests started at $\sim 400 \pm 20^\circ\text{C}$. The initial sodium flow velocity from test to test ranged from 3.3 to 5.4 m/s, with corresponding flow rate per pin ranging from 80-160 cm³/s. The power transients in all of the tests included a sharp power burst with full-width-at-half-maximum of 0.15 s (initial period of 35 ms) in E3 and ranging from 0.23 s to 0.32 s in the other tests (initial periods ranging from 66 ms to 189 ms). The transients in tests E1 and E2 were simple, reactivity-step transients (the only type possible at that time, since the computer-controlled reactivity control system had not yet been installed in TREAT). The transients in the later tests used a short, initial constant-power phase to preheat the test fuel prior to the power burst, with burst full-width-at-half-maximum in the range 230-310 ms

Results

In test E1, the fuel pin did not fail even though up to 80 areal% of the fuel had melted and the cladding strain reached 1.1% along the fuel column region. In-pin axial fuel motion was extensive, with some fuel reaching the top of the pin plenum. Results from the other tests in the series consistently showed (based upon coolant voiding dynamics) that the thermal-to-work energy conversion due to fuel-coolant heat transfer upon oxide fuel failing into flowing sodium was acceptably low, for fresh fuel and for irradiated fuel with low-power-structure and with high-power structure (including relatively-low and high fission-gas retention in the fuel). Conditions causing cladding failure were demonstrated, and in some tests, axial location of failure was also demonstrated. Axially unrestrained fuel could be greatly dispersed axially prior to (and preventing) cladding failure. Fuel pins that were more of the FFTF design, on the other hand, failed prior to much in-pin fuel motion and resulted in released fuel and molten cladding forming substantial blockages of the coolant channel at the top and/or bottom of the fuel column. Intimate mixing of fuel and steel was generally not observed. Penetration of the flowtube caused unprototypical additional complexity in the subsequent fuel and cladding motions, including eventual formation of blockages.

Applications

The pre-failure and early-post-failure behavior of fresh and irradiated LMFBR oxide fuel under fast transient heating aided the evaluation of computational models and accident analysis code(s), for which little or not previous in-pile test results in a flowing sodium system were available.

Calibrations in TREAT

Determinations of the neutronic coupling between the test fuel and the TREAT core during the experiments were made by irradiations of fresh pins at relatively low power and energy in TREAT and by computations.) The fuel irradiations were followed by measurements of fissions density in the fuel samples. In some cases, small foils were placed around the fresh fuel pin(s) and analyzed post-irradiation to determine the azimuthal variation of neutron flux around the pin. In the later test(s), monitor wires were irradiated both at low-power steady-state (as were the fresh pins) and separately during the power transients planned for the tests, to determine the how much the

neutronic coupling changes during the transient. The power coupling for test E4 was determined using measurements from the calibration measurements made for TREAT test H-1. Tests E6 and E7 used results from the H-3 power calibration measurements. The calibration for E8 involved the full range of monitor-wire and fresh fuel irradiations and measurements; it also involved using a newly-developed ultrasonic technique for cutting annular rings from several pellets from a fresh pins irradiated in TREAT calibration runs, followed by radiological analysis of the relative fissions per ring, yielding a measure of the radial fission profile in the test fuel.

16H-series Experiments

The H-series tests investigated the response of MOX fuel of various pre-irradiation conditions when subjected to FFTF 50 ¢/s to 1 \$/s unprotected overpower transient simulations in flowing sodium [83, 56, 80, 87, 88, 70, 89, 90, 91, 92] [93, 94, 95, 96, 97]. A one-page summary of each H-series test is included in Appendix N.

Number/Time

Six H-series experiments were performed during the time period 1970 to 1977.

Purpose

The test series addressed fuel response (cladding failure, initial fuel and cladding relocations, and coolant energetics) to conditions representative of hypothetical 50 ¢/s to 1 \$/s unprotected FTR overpower transients. The tests were intended to extend the earlier, preliminary/scoping overpower tests of fast-reactor mixed-oxide fuel to include fuel types and transient conditions specific to the FFTF Fast Test Reactor. This required greater prototypicality regarding the fuel design and fuel pre-irradiation conditions, and also regarding the test fuel power-time history and spatial power distribution and coolant environment during the transient test.

Approach

Fuel pins based on the FFTF fuel design, but short enough to be preirradiated in EBR-II and accommodated in the available Mark-II TREAT sodium loop, would be tested using fuel pins spanning a range of pre-irradiation conditions including fresh, low-to-intermediate power irradiation, and high-power irradiation. Variation of the transient behavior due to the associated range of fuel microstructural and fission-gas retention characteristics would be investigated, such as conditions causing cladding failure and ensuing initial fuel and cladding relocations. Improved prototypicality relative to earlier overpower transient tests would be provided by using a combination of features such as fast-flux irradiated fuel, multi-pin bundles, thermal-neutron-filtered test vehicles to provide better radial and axial fission-density profiles in the test fuel, shaped TREAT power transients, and ability to scram the TREAT reactor upon indications of cladding failure during the transient. Some of these features had only recently become available.

Limitations

The Mark-II integral TREAT sodium loop design had been optimized for testing small bundles of available pre-irradiated fuel pins under fairly-prototypic conditions causing cladding failure within the capability of the TREAT reactor. The loop-reactor system was limited practically to test seven-pin bundles of highly-enriched fuel pins if the pins were to be heated to failure. The limited inventory of suitable pre-irradiated fuel pins resulted in the need, in some of the tests, to test the pre-irradiated pin with six surrounding fresh pins to help simulate a prototypic radial thermal-hydraulic environment. Test conditions were computationally designed to be as prototypic as possible using available thermal-hydraulics and transient fuel behavior models which, as with the existing empirical basis, were still in their early stages of development, resulting in considerable

uncertainty in predictive capability. In addition, experimental techniques for performing tests of this type were based on a limited foundation of prior testing experience upon which to design such experiments. Test fuel and cladding temperature distributions were less prototypic in the two tests (H1 and H2) that did not include a preheat phase before the power burst. On the other hand, the use of thermal neutron filtering and neutron-attenuating axial shaping collars on the loops helped to make the radial and axial power profiles in the test pins more prototypic. The use of fueled pins (although unirradiated) surrounding the pre-irradiated fuel pin in some respects tended to provide a more-prototypic environment for that pin, if the pre-irradiated pin should fail prior to the fresh pins; however, the capability to clearly detect which pin type (pre-irradiated or fresh) in the bundle failed first was not generally present. Extended post-failure material motions in such mixed-bundle tests, although providing useful evidence of multi-pin bundle behavior, would not fully display the behavior representative of a bundle of all identically-pre-irradiated pins. Thin stainless steel structures (such as the flowtube wall around the pins) contacted by hot fuel during a test could readily be melted through, thereby adding to the inventory of molten steel in the system and allowing molten materials to penetrate into regions outside the flowtube, creating non-prototypic material relocation scenarios.

Experiment Vehicle

Mark-II integral flowing-sodium loops were used in all of the H-series tests. Thermal-neutron-absorbing layers (flux “shaping collars”) were located around the loop periphery to provide a representative axial power profile in the test fuel. Test train (fuel holder) designs varied from test to test, depending on whether one pin (H1 and H2) or seven pins (H3 through H6) were tested and also depending upon design assumptions made regarding potential for post-failure heat transfer and material motions. The flowtube (approximately 0.8 mm thick) and surrounding outer tube in each test train was made of stainless steel. The test train designs for the three tests in which post-failure in-channel fuel motions were significant (H2, H4, and H6) had the following features (moving radially outward from the flowtube): in H2 the flowtube was surrounded by a 0.55-mm wide evacuated annulus, then a 0.8-mm-thick stainless steel tube, then a 10-mm thick sodium annulus; in H4 the flowtube was surrounded by 2.4-mm-wide helium gap, then a 0.9-mm-thick stainless steel tube; in H6 the flowtube was surrounded by an inert-gas annulus and then a stainless steel tube. In the seven-pin tests, the flowtube periphery was fluted in six azimuthal places to partially represent the geometry of an additional ring of pins and thus provide a more-prototypical local flow area around the periphery of the bundle. The vehicles were outfitted with flowmeters, pressure transducers, and thermocouples.

Test Fuel

Except for test H1, all of the tests used helium-bonded mixed-oxide fuel pins that were like standard FFTF-fuel pins but shorter (38 cm fuel column height in H1, 34 cm in the other tests). In H1, the fuel pin was of PFR design and was tested in TREAT oriented with the plenum down, toward the coolant inlet. All pins had type 316 stainless steel cladding. In H1 and H2, the single pins tested were fresh. In each of the tests H3, H4, and H5, a single pre-irradiated fuel pin was tested as the central pin in a seven pin bundle with the other six (peripheral) pins being fresh. In H6, a bundle of seven pre-irradiated pins was tested. The pre-irradiated pins had the following irradiation characteristics: in H3, H4, and H5 the fuel had been irradiated to 3.4 to 4.4 at.% burnup at about 35 kW/m linear power; whereas in H6 the fuel had been irradiated to 6 at.% burnup at 30 kW/m. The fuel in the H3 and H5 pre-irradiated pins had intermediate-power microstructure, whereas the pre-irradiated fuel in H4 had high-power microstructure, and in H6 the fuel had low-to-moderate-power microstructure.

Conditions

All of the tests were performed using flowing sodium under constant sodium-pumping conditions and subjected the test fuel to overpower transients. In H1 and H2, the transients were natural burst transients generated by an initial reactivity step in TREAT. The other four tests, using improved, computer control of the reactor, included an initial, short lower-power segment to preheat the test fuel prior to the burst. The burst shapes were similar among all the tests but can nevertheless be roughly compared according to their full-width-at-half-maximum durations: 0.36 s in H1, 1.3 s in H2, ~0.6 s in H3, 1.06 s in H4, 1.0 s in H5, and ~1.5 s in H6. The transients differed in part because the test-fuel power levels developed during TREAT transients typically depend not only upon the TREAT power but also upon the power coupling between the test fuel and the TREAT core specific to the particular test fuel and loop hardware design used in the test, which differed from test to test. For each test, the test transient was designed to produce fuel conditions closely approaching, or else definitely exceeding, the fuel pin failure threshold, depending upon the test objective. Significant fuel energy generation after initial cladding failure was intended to be avoided in order to preserve evidence of the early post-failure state and distribution of the fuel and cladding.

Results

The tests displayed a wide range of outcomes, including pins that did not fail (H1 and H3), pins that experienced only mild failure (H5), and pins that were grossly disrupted (H1, H4, and H6). Because of the shaped transient used in tests H3 through H6, the pre-irradiated test-fuel linear power to failure among those tests can be reasonably compared: 140 kW/m without failure of the H3 intermediate-power-structure pin, 170 kW/m at failure of the H4 high-power-structure pin, ~180 kW/m at failure of the H5 intermediate-power-structure pin, and ~200 kW/m at failure of the low-to-moderate-power-structure pin. The fuel pin in H1 (as in H3) remained intact.

In H3, the pre-irradiated pin was heated just short of reaching the fuel solidus, whereas the fresh pins experienced up to 40% areal melting. Posttest the preirradiated pin provided detailed information regarding the post-transient characteristics of the fuel microstructure and retained fission-gas distributions. The intact pins from H3 also provided radiochemical data from which a direct determination of the power coupling between the test fuel and TREAT core during the test was made.

In H5, the pre-irradiated pin fortuitously experienced only a minor, localized cladding breach (at fuel height $x/L=0.84$) with minimal fuel release; the fresh pins were only slightly disrupted, and although little fuel motion occurred overall, the posttest flow rate was only 40% of the initial flow rate.

Gross fuel pin disruption occurred in H2, H4, and H6, resulting in complete (or nearly complete) flow blockages. Upward, in-pin fuel movement and mild sodium boiling occurred before cladding failure in H2 but not in any of the other five tests. In the tests involving large fuel motion, those large motions did not start until 0.2-0.5 s after cladding failure, although smaller motions may have occurred earlier in H2 in order to cause the detected flow blockage 50 ms after cladding failure. Large-mass movement of fuel within coolant channels to regions above or below the original fuel zone occurred only in H2. The flowtubes in H2, H4, and H6 breached, and much fuel moved into the surrounding space in H4 and H6 (but not in H2). Most of the fuel column was disrupted in H2 and H4, but mainly only the top half was disrupted in H6. In none of the tests involving significant post-failure fuel motion was significant coolant energy generated, and the peak pressure measured in the test series was on the order of 12 MPa (in H6).

Applications

The tests provided some of the first in-pile results of the behavior of pre-irradiated FFTF-like fuel pins during simulations of hypothetical FFTF 50 ¢/s to 1 $\text{\$/s}$ unprotected overpower transients in flowing sodium, including cladding failure threshold, transient-induced microstructural and fission-gas retention characteristics, and initial post-failure fuel and coolant dynamics. The tests also demonstrated subsequent fuel and coolant dynamics as coolant attempted to re-enter the hot fuel region, fuel melting of nearby steel structures, and propensity of the molten fuel and steel to form blockages in the coolant channel. The results were useful for comparing with analytical models, for helping to provide preliminary, empirical evidence of the associated transient behaviors, and to guide the design and planning of subsequent tests that would provide conditions of greater prototypicality.

Calibrations in TREAT

Measurements of the test-fuel to TREAT-core power coupling were made using various methods including irradiation of fuel pins and uranium monitor wires during low-power irradiations in TREAT, irradiation of monitor wires during high-power transients in TREAT, posttest radiochemical analysis of the H3 unfailed fuel pins, and (for H6) low-power irradiations of monitor foils placed around the periphery of fresh pins in a seven pin bundle, plus radiological measurements of radial segments from those pins.

17EOS-series Experiments

The EOS-series tests probed the potential for fast reactor MOX fuel to disperse by fission gas pressure prior to generating high fuel vapor pressure during extremely fast high-power transients [98, 99, 100, 101, 102, 103]. A one-page summary of each EOS-series test is included in Appendix O.

Number/Time

Three EOS-series tests were performed during 1977.

Purpose

The EOS tests were performed to provide scoping information regarding whether, during rapid energetic power bursts in fast reactors, fission gas in irradiated mixed-oxide fuel can significantly disperse fuel prior to the generation of high fuel vapor pressure. The data were needed to support modeling of fuel motion in the target range of fuel heating rates, within which little was known at that time. In particular, for such power bursts, answers to the following question were sought in the tests, for both moderate-power irradiated fuel and high-power irradiated fuel: (a) does fission gas provide a dispersive mechanism for the fuel or does the gas separate from the fuel without causing significant fuel dispersal?, (b) if the fission gas does cause significant dispersal, what is the timing of that dispersal relative to the fuel reaching its solidus (or what fuel specific energy is needed to initiate dispersal or slumping in irradiated fuel versus in fresh fuel), and what mode of dispersal results?, and (c) if fission gas does not cause significant dispersal, does the fuel slump before significant fuel vapor pressure develops? The mode of fuel dispersal would depend largely on whether the fission gas had first caused grain boundary separation as the fuel solidus was being approached, resulting in formation and dispersal of finely-divided solid particles (dust) or whether the fission gas release occurred as the fuel liquidus was reached, causing fuel expansion and dispersal of larger masses of solid or melting fuel. If, prior to dispersal, the fuel was in the form of fine particles, its dispersal was anticipated appear as a generally uniform redistribution of fuel radially within the test capsule. Otherwise, the dispersal (of partially-melted chunks) was anticipated to be relatively localized.

Approach

The fuel dispersal characteristics of fuels having significantly-different fission gas content and distributions would be tested for comparison: fresh fuel in EOS-1, high-power-structure irradiated fuel in EOS-2, and medium-power-structure irradiated fuel in EOS-3. The fresh fuel pins would not undergo fuel melting or cladding failure, thereby allowing posttest characterization of transient-induced fuel grain growth, cracking, densification, and plasticity for fuel mechanics code calibration. The tests would be performed without coolant, simulating conditions in a voided coolant channel. The TREAT power transient would be fast enough to produce essentially adiabatic heating of the sample.

Limitations

Because the power transients used in the tests were so narrow and reached such high power, the planned test conditions were practically at the limit of TREAT experiment capability regarding accuracy and precision of measuring fuel motion (by the hodoscope), determining semi-empirically the test-fuel power, enthalpy, and temperature during the transient (by neutronics measurements, thermal-hydraulic computations, and physical properties information), and detecting the time of cladding failure (by the thermocouple response). The signal strength reaching the hodoscope was too low to allow the precise determination of the pin-failure time in EOS-1; in EOS-2 the useful data provided meager definitive information; in EOS-3 the signal was sufficient to permit good discernment of fuel axial displacement but not radial displacement.

Experiment Vehicle

The test vehicle used in all three tests consisted mainly of a pair of dry, instrumented capsules, each accommodating a short fuel pin within a boron nitride enclosure, bounded by sealed stainless steel containment walls. The capsules were located in tandem, located one above the other within the TREAT core. The upper capsule was intended to house the test fuel pin; the lower capsule was to house an unirradiated "control" pin.

Test Fuel

Both of the pre-irradiated helium-bonded mixed U-Pu oxide (MOX) pins used in EOS-2 and EOS-3 had been irradiated in EBR-II to ~ 0.8 at% burnup. The pin designated for EOS-3 was a larger-diameter pin than the one designated for EOS-2. The EOS-2 pin linear power was ~ 37 kW/m resulting in the development of high-power-structure fuel (central void and concentric zones of columnar, equiaxed, and unrestructured grains), whereas the EOS-3 pin linear power was ~ 27 kW/m resulting in medium-power-structure fuel (only zones of equiaxed and unrestructured grains). After the pins had been pre-irradiated, they were cut open, the cladding tubes were shortened, and the remaining cladding was refilled with a 10.2-cm stack of the pre-irradiated MOX pellets, with boron nitride pellets at each end of the stack. After the cladding was re-sealed (at 2 atm pressure), the cladding wall was reduced full length to only 0.13 mm thickness. (The fission-gas content of the fuel in the EOS-3 pin was calculated to be about 50% greater than the fission-gas content of the fuel in the EOS-2 pin. However, considering the fission-gas density variation by fuel microstructure, it was estimated that the fission-gas-induced effects between the two pins in the EOS tests would be approximately as if the EOS-3 pin contained only 25% more gas than the EOS-2 pin.) The fresh pin used as the test pin in EOS-1 was initially of the same design as the EOS-2 pin and was similarly modified. The fresh control pin in each test was similar to its companion test pin in the test, except that its cladding thickness had not been reduced.

Conditions

The power transient used in each test was essentially the same: a "natural" reactivity-step-input resulting in a burst rising on a 23-25 ms period and having a full-width-at-half-maximum of 110-

120 ms. The power generated in the test fuel pins depended, as usual, on the magnitude and variation of the coupling between the test (or fresh/control) fuel and the TREAT core during the transient. In the EOS tests, peak test-fuel power levels reached approximately 100 times nominal power for those fuels. There was no coolant in the test. Near the power peak (when the test fuel pins failed), the time duration between reaching fuel solidus and fuel liquidus temperatures was about 20 ms.

Results

Disruption of the test pins began near peak power, at about the time the fuel solidus was reached in the preirradiated test fuel or the liquidus was reached in the fresh test fuel. (It was assumed that some fuel vapor pressure was needed to disperse the fresh fuel.) Test instrumentation was unable to accurately discern the onset of fuel disruption on the unusually-short time scale of events in the test. It was estimated that disruption of the medium-power-structure fuel in EOS-3 began at a fuel enthalpy roughly 100 J/g less than the enthalpy in the high-power-structure fuel when the latter began to disrupt. It was also estimated that the pre-irradiated fuel disruption began at roughly 200-300 J/g lower than when the fresh fuel disruption began. In EOS-3, significant axial motion of the test fuel did not occur until about 60 ms after the onset of axial fuel relocation (at peak power) was detected, and even larger axial fuel motion occurred during the next 10-100 ms as the power transient was decreasing below its half-power level. In all of the tests, axial test-fuel motion was generally more upward than downward. In EOS-2, upward fuel motion appeared to be more energetic or extensive than in EOS-1, the posttest appearance of the fuel material was like that in EOS-1, and posttest evidence indicated that after fuel movement ended and the fuel solidified, molten steel tended to drain downward. The remains of the fuel in EOS-3 were more porous (density about 8.4 g/cm³) than the remains in EOS-1 and EOS-2 (9.3 to 10.7 g/cm³). Pellets from the fresh fuel pin remained in good condition in the test, with no evidence of exterior cracks and no increase in diameter.

Applications

The test data (coupled with calculated test-fuel temperatures) appeared to provide evidence for the onset of fuel motion near the fuel solidus for preirradiated fuel and the fuel liquidus for fresh fuel during these extreme transients. The rate at which fuel mass dispersed axially relative to the rate at which the fuel temperature increased to and beyond the fuel liquidus was apparently not able to be well determined, for a variety of reasons. The role of fuel vapor relative to the roles of fission gas and steel vapor pressure in the larger irradiated-fuel dispersal events following initial cladding failure was therefore unclear.

Calibrations in TREAT

The relationship between the TREAT core power and the fission power generated in the EOS-series test and control pins was determined by irradiating fresh fuel pins (which were neutronicallly similar to the EOS pins to be transient tested) and enriched-uranium monitor wires in neutronic mockup hardware representing the actual test hardware. Radiological measurements of the irradiated samples, together with the measured reactor energy values, provided the information needed to determine the associated power coupling factors as well as the axial power profile in the test fuel. Corrections to those factors due to fuel burnup for the pre-irradiated fuel were computed. In addition, analysis of annular samples cored from the enriched-uranium-bearing calibration fuel pins provided information on the radial power profile in the fuel.

18F-series Experiments

The F-series tests investigated the motion of irradiated LMFBR mixed-oxide fuel following sodium voiding and dryout during a hypothetical loss-of-flow accident [104, 105, 106, 107, 108, 109]. A one-page summary of each F-series test is included in Appendix P.

Number/Time

Five F-series tests were performed: F1 and F2 in 1974, F3 and F4 in 1980, and F6 in 1984.

Purpose

The goal of the tests was to determine the timing and mode of fuel-column disruption of mixed-oxide fast reactor fuel during simulations of lead-subassembly conditions in hypothetical loss-of-flow accidents in LMFBRs with moderately-positive sodium void reactivity. The conditions involve fuel breakup in channels voided of sodium coolant after cladding has drained or dispersed. Potential disruption modes of the fuel column include column buckling, fuel swelling (before or after the solidus is reached), fragmentation, and liquid fuel jet formation, with strong dependence on the role of the fission gas initially retained in the fuel.

Approach

All tests were performed on single, modified, EBR-II-irradiated fuel pins located in capsules without coolant. Generally, the fuel was heated at a steady power until the fuel disrupted, except that in F2 a power burst was included at the end a steady-power preheat phase. Two different approaches were taken in the test designs in accordance with changes in test objectives. In F1 and F2, the fuel pins, after having their plena depressurized, were tested inside a surrounding, nuclear-heated wall intended to provide a more-prototypic environment for the fuel pin as it disrupted during the test. In tests F3, F4, and F6, details of the nature of initial fuel-column breakup dependence on fuel heating rate were sought by including an optical fuel-motion diagnostics system and using smaller fuel samples.

Limitations

The nuclear heated wall in F1 and F2, although having the advantage of providing a prototypically-hot environment up to the initial fuel dispersal in F1 and F2, strongly diminished the hodoscope signal signal to background/noise ratio. Melting of the UO₂ in the nuclear heated wall in F2 added UO₂ to the fuel UO₂ mass and tended to confuse the interpretation of the fuel motion that followed. Lack of prior knowledge regarding the rapidity of initial fuel breakup under these test condition resulted in the optical fuel-diagnostics system used in F3 and F4 being found to be inadequate and the improved optical system used in F6 to providing limited useful information. The significant additional test-fuel energy that was generated resulted, in most cases, in gross melting and relocation of the fuel after initial fuel relocation.

Experiment Vehicle

Two different designs of test capsules were used, one for F1 and F2, the other for F3, F4, and (as modified) for F6. The F1/F2 capsule contained a single EBR-II-length fuel pin (34.3 cm fuel column) inside a 0.76-mm-thick wall composed of tungsten-UO₂ cermet with the U enriched. The wall, which would be nuclear-heated during the test, was intended to prevent non-prototypical freezeout of fuel and cladding and also provide a well-defined geometry for fuel motion. Radially beyond the heated wall were stainless steel containment barriers. The annular space between the fuel pin and heated wall was sized to represent the cross sectional area of a coolant channel in a reactor. Thermocouples were positioned slightly above and slightly below the 34.5 cm active test fuel region to indicate the temperature of fuel or cladding that might arrive at those locations during the test. The F3/F4

capsule, in contrast, was designed for optical viewing (using ex-reactor high-speed cameras) the 5 cm test-fuel column of the fuel pins used in those tests. The capsule and optical system F3 and F4 relied on the self-illumination of the sample (which proved inadequate) and was subsequently modified for F6 to allow the use of laser illumination of the sample.

Test Fuel

The fuel in all five tests was mixed oxide pellets helium-bonded to stainless steel cladding, pre-irradiated in the fast flux of EBR-II. In F1 and F2 the fuel had been pre-irradiated at 38-40 kW/m, to a burnup of 2.35 at% for F1 and to 0.35 at% for F2, generating a high-power structure in both fuels, but with a larger central void and smaller unrestructured zone in the lower-burnup fuel. Short fuel pins, containing 5-cm of fuel that had been preirradiated at 26-29 kW/m to 9 at% burnup and having no central void, were tested in F3, F4, and F6.

Conditions

The TREAT power transient in tests F1, F3, F4, and F6 was a constant-power run, at generating test fuel linear power of ~38 kW/m in F1, 82 kW/m in F3, 216 kW/m in F4, and ~90 kW/m in F6 (prior to fuel disruption). In F2, however, the transient began with a constant-power segment generating 41 kW/m in the test fuel for several seconds but ended with a ~450 ms full-width-at-half-maximum power burst peaking at about eleven times the initial constant power level. There was no coolant in any of the tests.

Results

In F1 and F2, initially the cladding melted and ran down to the bottom of the fuel region, followed by extensive swelling of fuel at the ends of the fuel region and melting of the rest of the fuel. In F1, the molten fuel remained within the original fuel region, but it ran down from the top half of the fuel region into the bottom half. Scram occurred about a second after fuel motion began. The hodoscope did not indicate any large fuel-motion event. The heated wall remained intact and radially confined the molten materials. In F2 just before the power burst began, cladding had melted and drained to the bottom, leaving the fuel pellet stack standing. When between 30-50% of the burst energy had been generated, two significant upward fuel motion events occurred involving fuel in the top half of the original fuel region, some moving more than half the length of the fuel column above the original top of the fuel. As the power dropped, much of the fuel that had moved up dropped down to the bottom half of the fuel pin. At an unknown time during the transient, the UO₂ phase of the heated wall melted, added its mass to the fuel UO₂ inventory, and allowed a considerable amount of fuel to leak out into the surrounding space. In F3 and F4, the fuel pin disintegrated before the fuel had melted, and its disintegration was too fast (within ~1.5 ms) for the details to be discerned by the optical fuel-motion diagnostics system. In F4, the disintegration may have occurred even before the cladding had melted. In F6, the fuel pin experienced a local break or tear in the still-solid cladding, followed by a sudden release of extremely hot material that filled the 2.5 cm viewing area within 1 ms.

Applications

The relative timing of cladding melting, fuel melting, and initial fuel motion of irradiated fuel in dry coolant channels as illustrated by these F-series tests allowed for conclusions to be drawn regarding the role of fission gas in fuel disruption and initial fuel dispersal under the imposed test conditions. In F1 and F2, fuel motion began after the cladding had drained off. In F1, it appeared that fission gas did not prevent an axial fuel collapse nor cause a gross axial dispersion of fuel prior to the fuel collapse that occurred, but a froth of fuel and fission products likely retarded fuel collapse until the fission products separated from the partially-molten fuel. The highly-swollen fuel radially filling the space inside the heated wall at the top and bottom of the original fuel region may have prevented

molten fuel from moving past those locations. In F2, much fuel was initially driven upward far above the top of the original fuel region. Tests F3, F4, and F6 showed that initial fuel disruption may occur prior to melting of the fuel or cladding and that release of a substantial amount of hot material from the cladding can occur on the scale of one or two milliseconds. Data directly showing the mode of fuel disruption was not able to be obtained.

Calibrations in TREAT

The relationship between the TREAT core power and the fission density in the test fuel sample located within the two different F-series in-core experiment hardware systems was radiochemically determined from samples of fresh fuel, and enriched-uranium monitor wires and foils, that had been irradiated in TREAT within a neutronic mockup of each F-series test vehicle. Concentric samples that were core-drilled from the TREAT-irradiated fresh fuel were also radiochemically analyzed for fissions. The coupling between the heated wall in F1 and F2 and the TREAT core was similarly determined.

19D-series Experiments

The D-series tests demonstrated pre-failure response of fresh FFTF-like UO_2 fuel pins subjected to near-steady, localized overpower causing substantial fuel melting but with cladding remaining cool [110]. A one-page summary of each D-series test is included in Appendix Q.

Number/Time

Two D-series tests (D1 and D2) were performed in 1971.

Purpose

The tests intended to empirically-determine the conditions required to achieve the release of a small amount of molten fuel from failure of a fresh UO_2 fuel pin of FFTF-like design into sodium coolant under conditions approximating normal coolant flow and temperature, and to provide information useful in determining the mechanism and cause of the fuel release, the extent of any molten-fuel/coolant interaction, the nature of any flow impediment caused by the fuel release, and whether any damage to adjacent pins tends to propagate additional pin failure.

Approach

A seven-pin bundle of fresh, FFTF-like fuel pins fueled with UO_2 would be tested in flowing sodium in a constant-power transient with the power level and sodium coolant flow sufficient to likely cause cladding breach by the end of the transient. A key feature of the test would be to generate over-heating of a small section of the central pin in the bundle by over-enrichment of a few pellets of fuel near the fuel-column axial midplane. When none of the pins failed in D1, the test was repeated (test D2) using an identical but untested central pin and a 70-75% higher power in a second attempt to cause pin failure.

Limitations

These were the first seven-pin tests performed in flowing sodium in a TREAT loop and were performed without the benefit of a significant empirical basis upon which to predict conditions for fuel pin failure in flowing sodium. The phenomenological details associated with failure of fresh fuel in the range of moderate overpower thermal conditions utilized in the tests were not well known at the time.

Experiment Vehicle

Both tests used Mark-IIA integral sodium loops, with seven-pin test trains, instrumented with flow meters, pressure transducers, and thermocouples.

Test Fuel

All pins in both tests were of FFTF-like design, but with short (34.3 cm) fuel columns, containing fresh UO_2 pellet stacks helium bonded to type 316 stainless steel cladding. So that the fuel in the central pin would have about the same fission density during the test as the fuel in the six edge pins, the UO_2 in the central pin had a higher enrichment (26 at%) than the fuel in the edge pins (20 at%). To cause localized cladding failure in the central pin, several UO_2 pellets, occupying a 3.8-cm length at the fuel midplane of the central pin, were fully-enriched to 93 at%.

Conditions

Both tests subjected the test fuel to near-constant power for several seconds, with the fuel pin bundle cooled by flowing sodium. The conditions in D1 were 40 kW/m (82 kW/m in the fully-enriched section) for ~ 13 s with 850 cm^3 sodium flow at an inlet temperature increasing from 415°C to 570°C. The conditions in D2 were ~ 70 kW/m (~ 140 kW/m in the fully-enriched section) with 870 cm^3 sodium flow at an inlet temperature that increased from 425°C to 555°C.

Results

The test fuel pin remained intact in each test despite considerable fuel melting. In the region of 93% enriched fuel, the fuel melting was up to 55 areal% in D1 and up to 75 areal% in D2. The maximum melting in the 26%-enriched region in the central pin in D2 was 40 areal%. In some locations, molten fuel had moved through spaces in the pellets and reached the cladding where it froze, causing practically no effect on the cladding. The outlet sodium temperature at the end of the power transient in D2 was 690°C.

Applications

The two tests showed that, regarding fresh MOX fuel pins of prototypical FFTF design, short-term pin failure due to local over-enrichment of fuel is not likely to occur during near-steady fuel linear powers up to 140 kW/m if the cladding is adequately cooled. Correspondingly, the tests provided evidence that fresh FFTF-design MOX-fueled pins can operate at nearly-normal coolant conditions for short periods of time with large amounts of molten fuel and not fail. In addition, significant useful information gained from these first TREAT sodium loop tests on seven-pin bundles guided the performance of later loop tests.

Calibrations in TREAT

The ratio of test-fuel fission density to TREAT core integrated power was determined by the irradiation of a seven-pin bundle of fresh fuel pins under low-power and also the irradiation of monitor wires under low-power and power transient conditions, and then radiologically analyzing the irradiated samples to determine fissions per gram. Posttest thermal-hydraulic calculations tended to verify the experimentally-determined test-fuel-to-core power coupling ratio.

20S-series Experiments

The S-series tests investigated the energetics due to thermal interaction between oxide fuel and sodium following cladding failure, over severe fuel heating scenarios and coolant conditions at cladding failure [111, 112]. A one-page summary of each S-series test is included in Appendix R.

Number/Time

Nine S-series (S-2 through S-8, S-11, and S-12) were performed during 1968-1972.

Purpose

The tests were performed to investigate the pressure pulses and their impulses that would be generated upon release of hot uranium oxide from fuel-rod cladding upon cladding failure. They were mainly scoping tests to determine whether the power-excursion-driven meltdown of multi-pin arrays of clad oxide fuel in sodium would result in the very high pressures (perhaps up to several hundred MPa) and the high efficiency of thermal-to-mechanical energy conversion (perhaps as high as 30%) that were calculated to be thermodynamically possible in such a system. Posttest estimations would be made of the efficiency of the conversion of test-fuel thermal energy to mechanical work that occurs during such fuel-sodium interactions. Heating conditions necessary to cause cladding failure were to be discovered early in the investigation. Determining the dependence of the energetics upon the fuel conditions (particulate or molten) and coolant conditions (temperature and void fraction) at the time of cladding failure, plus coolant channel geometry was a key part of the investigation. The size distribution of fuel particulate in posttest debris would help in developing an understanding of the fuel-coolant heat transfer characteristics that occurred. The test results would be compared with results of ex-reactor tests involving injection of molten UO_2 into sodium.

Approach

All of the tests used short, unirradiated, stainless-steel clad uranium-oxide-fueled fuel rods immersed in stagnant sodium within strong autoclave-type test capsules. The power transients used in all of the tests were short-duration bursts (in the range 104 to 220 ms full-width-at-half-maximum) generated by a step-input of TREAT reactivity and controlled only by the natural negative temperature feedback reactivity of the TREAT core. Coolant temperatures, pressures, and pressure impulse energies were measured. Four tests (S-2 through S-5) used nine-rod bundles comprised of five fueled rods and four dummy rods (hollow cladding tubes) in a square array. Three tests (S-6 through S-8) used bundles of seven fueled rods in a hexagonal array. Two tests (S-11 and S-12) used single fueled rods.

Limitations

Achieving fuel-rod cladding failure in stagnant sodium prior to sodium boiling required a high fuel-heating rate, generally starting from a low coolant temperature. Because the capability of predicting cladding failure conditions for helium-bonded uranium oxide fuel under severe power transients was still in an early stage of development, the in-reactor testing program initially involved some degree of trial and error in gaining the required knowledge. TREAT power bursts were limited by the available reactivity, transient rod speed, and negative neutronic feedback characteristics of the reactor. Power-pulse width, varying inversely with total pulse energy, limited the range of options for achieving the desired fuel radial temperature/melting profile and coolant temperature at the time the cladding reached its local failure temperature-pressure condition.

Experiment Vehicle

The basic capsule used in all of the tests was a robust, cylindrical, stainless steel pipe that served as the primary containment wall of the stagnant-sodium test vehicle. The inside of the pipe provided a cavity of 3.8 cm diameter and 34 cm long to accommodate the fuel rod or rod bundle and Zircaloy-2 liner that formed the radial boundary of the region that contained the fuel rods and sodium. A piston was located at the end of the cavity and was connected to a linear motion transducer. Outside the primary containment were heaters, insulation, and (for tests S-4 through S-12) a 0.63 cm-thick

graphite annulus, all surrounded by a thin-walled stainless steel secondary containment. A pressure transducer and several thermocouples were also positioned to measure sodium conditions.

Test Fuel

All fuel rods were about 34 cm long and contained ~15-cm-high stacks of enriched UO₂ pellets helium bonded to stainless steel cladding. All contained one atmosphere pressure at room temperature, except for the fuel rods in test S-5. In test S-5, the fuel rods were evacuated to prevent the possibility of release of the fill gas at cladding breach from blanketing the interaction between released fuel and coolant.

Conditions

All tests were performed with burst-type transients. The transients in tests S-2 through S-7 had initial periods in the range 34 to 62 ms and full-widths-at-half-maximum in the range 175-220 ms. The transient in test S-8 had initial period of 27 ms and FWHM of 140 ms. Tests S-11 and S-12 used transients of 23 ms initial period and 104 ms FWHM. Temperatures of the sodium and fuel rods at the start of each test (165-190°C in tests S-2 through S-5, ~500°C in tests S-6 through S-8, and 140°C in tests S-11 and S-12), and incorporation of heat sinks radially surrounding the sodium, were chosen to achieve the desired combination of fuel, cladding, and sodium temperatures at the predicted time of cladding breach, taking into consideration the associated rise-time of the power burst that needed to be used for the test. The initial pressure in the capsule was 0.1 MPa in all tests. The transients in S-2 through S-8 generated between approximately 1.9 and 3.3 kJ/g of fuel, whereas the transients in S-11 and S-12 generated approximately 7 kJ/g of fuel. In several of the tests, a considerable fraction of the total energy was generated after the cladding failed: about 30% post-failure energy in S-4 and S-7, nearly 50% in S-8, and about 70% in S-11 and S-12. Analyses showed that only in test S-3 was sodium boiling likely to have occurred prior to cladding failure, that the possibility of pre-failure boiling was marginal in tests S-7 and S-8, and unlikely in tests S-4, S-5, and S-6. In tests S-11 and S-12, the shortest burst possible in TREAT was used so that a high fuel energy would be reached well before the cladding exterior temperature reached the sodium boiling point.

Results

In the tests, the measured pressure-time impulse correlated well with the observed change in momentum of the piston. Peak recorded pressures (less than 20 MPa), fuel particulate size (generally greater than 100 microns), and calculated thermal-to-mechanical energy conversions (less than 0.2%) indicated molten-fuel-coolant interactions that were non-explosive and of low efficiencies, under the fuel-coolant contacting conditions that actually occurred. The actual conditions, particularly the sodium temperature and amount of sodium vapor formation immediately preceding cladding failure, were not well known, however. This was due to uncertainties in temperature measurements, limited utility of data from the linear motion transducers (which locked in place before the full energy of the sodium slug could be absorbed), and two-dimensional heat-generation and heat-transfer effects that were not sufficiently analyzed. The most-finely-divided fuel residues were resulted from test S-5 in which the fuel pins had been evacuated, thereby nearly completely eliminating the role of fill gas in the fuel-coolant interaction. In test S-8, a mobile slurry of UO₂ in molten steel with very high thermal conductivity apparently was formed.

Applications

The peak fuel-sample power levels generated in the experiments were in the range of interest for FFTF hypothetical core disassembly accident studies, but they were significantly higher than for FFTF transient overpower excursions. The fraction of solid to molten fuel at the time of cladding

breach, combined with the unprototypically-high coolant temperatures at the time of cladding breach, plus the relatively-large heat-sink area, affected the direct applicability of the results. The low energy conversion ratios indicated in all of the tests were in all cases much less than theoretical maximum ratios. Tests S-11 and S-12 demonstrated that fuel fragmentation by fuel vapor pressure, followed by intimate fuel-coolant mixing and rapid heat transfer needed for an energetic fuel-coolant interaction, did not occur.

Calibrations in TREAT

A preliminary test designated S-1 was performed using a low-power transient in TREAT to determine the power coupling between the test fuel and the TREAT core. That coupling was measured by relating the fission-energy input to a test fuel sample and the fission energy generated by the core. In addition, wet-chemical analysis of the fuel from one of the pins that remained intact during transient test S-2 provided a measure of the power coupling factor. Similar calibration irradiations were performed (one prior to test S-6, another prior to test S-7, and a third prior to tests S-11 and S-12) to determine the power coupling values for those different test-vehicle designs and fuel sample configurations.

21RX-series Experiments

The RX-series tests were performed to investigate the behavior of a mixture of molten UO_2 and boiling steel with internal fission heating simulating the “transition phase” of an LMFBR accident [113, 114]. A one-page summary of each RX-series test is included in Appendix S.

Number/Time

Two RX-series tests (RX1 and RX2) were performed during 1982-1983.

Purpose

The tests were to provide scoping information regarding the boil-up of a molten UO_2 -steel pool under both “open pool” and “closed pool” conditions, with internal pool heating simulating the decay heat level during the “transition phase” of a hypothetical LMFBR accident. (The open versus closed condition refers to the amount of void volume above the pool.)

Approach

In the tests, a column of fuel-steel mixture containing enriched UO_2 would be fission heated at a constant rate within a fission-heated UO_2 boundary resulting in the fuel melting and generation of steel vapor. Axial movement of the boiling fuel-steel pool would be monitored by the TREAT fast neutron hodoscope and by a variety of thermocouples located at several axial positions within and above the initial fuel region.

Limitations

These were the first tests of their kind and involved considerable uncertainty regarding the type of pool behavior that would be observed. Temperature instrumentation would need to be located where they would be in contact with the pool, requiring the need for them to withstand the extremely-high temperatures. Performance capability of the selected ultrasonic thermometer and its boron nitride protective sheath under such uncertain conditions was not well known. Use of a nuclear heated wall was also an experimental feature for which little prior experience existed.

Experiment Vehicle

Both tests used test vehicles of nearly identical design. In each, a cylindrical stack of enriched- UO_2 -steel slugs (the “fuel”) was surrounded by an enriched- UO_2 cylinder of 24.6 mm inner diameter and

4.8 mm thick (the nuclear-heated wall). Radially outward from the nuclear wall were thin heat shields composed, successively, of tungsten, molybdenum, graphite, and stainless steel, all within stainless steel primary and secondary containments. High temperature thermocouples were located at various elevations above the fuel and in vertical holes drilled within the fuel. In RX1, an ultrasonic thermocouple, capable of measuring temperatures at several axial locations, extended into the top of the fuel within a BN sheath. Thermal-neutron filters were placed around the region where fuel would possibly be present as it disrupted during the test, to help control the fission power density in the fuel mass as it moved. The free volume above the fuel was ~5 liters in RX1 and ~0.5 liter in RX2, to provide conditions representative of “open pool” and “closed pool,” respectively.

Test Fuel

The fuel consisted of a stack of ten slugs of a hot-pressed mixture of fresh 87.6 wt% UO_2 – 12.4 wt% stainless steel. Each slug was 2.33 cm diameter and 1.27 cm high. In most of the slugs, the UO_2 was enriched to ~93%. The UO_2 in the top and bottom slugs in RX1, and in the bottom slug in RX2, were enriched to only 50%, however. Vertical holes were drilled in the slugs as needed to accommodate the thermocouples (and BN sheath in RX1).

Conditions

The fuel in each test was heated a nearly-constant heating rate of ~20 W/g up to the time of significant fuel relocation. (As test fuel reconfigures during TREAT tests, its power coupling with the TREAT core correspondingly changes.) The power transient lasted for about 150 s. Fission heating of the nuclear heated wall during the transient helped to provide a hot environment for the fuel.

Results

The main fuel motion began in both tests at roughly 110-120 s into the transient. In RX1, at the time of initial motion, steel vaporization had been predicted to begin. Also at that time, however, the boron nitride sheath of the ultrasonic thermocouple reached a temperature where it began decomposing, thereby removing heat needed for its decomposition and adding considerable nitrogen gas to the system. (It could not be determined whether the decomposition occurred within pool or only above the pool.) The top of the fuel moved upward several times, ultimately reaching a level ~22 cm above its initial location. Test RX2 did not include the ultrasonic thermometer or the BN sheath. In that test, the top of the fuel pool moved gradually upward by 1.6 cm during the first 90 s of the test. The main fuel-motion of the test, however, was of the upper quarter of the fuel column moving radially beyond the heated wall to the tungsten heat shield and then downward. By the end of the test, a small amount of fuel had also moved to a point 12 cm above the initial location of the fuel top.

Applications

In each test, an artifact significantly complicated the test and added considerable complexity, uncertainty, and loss of prototypicality. The BN decomposition in RX1 interfered with drawing conclusions regarding the role of steel vapor, since the nitrogen gas may have been partly generated within the pool and, in any case, would have increased the system pressure enough to affect the steel vaporization rate. In RX2, the penetration of the heated wall, which allowed some of the pool to leak radially to into a cooler annular region, affected the analysis of the fuel pool axial movement. Nevertheless, the tests provided new and unique evidence of pool behavior against which analyses could be compared and assumptions evaluated.

Calibrations in TREAT

Power coupling between the TREAT core and the RX test fuel and nuclear wall was measured by irradiating identical samples of the test fuel and nuclear wall in TREAT under low-power conditions.

The resulting coupling factors were then adjusted by the ratio of fissions generated in monitor wires irradiated under low power and power transient conditions intended for the RX tests. Radial and axial variation of the power coupling were computed.

22 J-series Experiments

The J-series investigated the response of pre-irradiated FFTF-like mixed-oxide fuel pins to the slowest transient overpower simulation possible in TREAT that results in cladding failure of such pins [115, 116, 117]. A one-page summary of each J-series test is included in Appendix T.

Number/Time

Only one test (J1) was performed, in 1979.

Purpose

The first test in the series was intended to show that a slow-period LMFBR transient overpower simulation causing cladding failure in an FFTF-like MOX fuel pin could be performed within the limitations of TREAT. The results would provide evidence of the failure threshold and post-failure fuel motions of preirradiated MOX fuel during such a transient. The test results would also be used in planning subsequent tests on full-length MOX fuel pin bundles in a longer, Mark-III sodium loop.

Approach

The J1 test subjected a seven-pin bundle of identical pre-irradiated mixed-oxide FFTF-like MOX fuel pins to conditions representing the slowest hypothetical unprotected FFTF overpower transient that could be simulated in TREAT and would (per pretest calculations) definitely cause failure of the test fuel pins. The TREAT power transient would be automatically terminated upon initial cladding failure, assumed to be indicated by a decrease in inlet flow rate of at least 50%.

Limitations

Due to pin-to-pin self shielding, the power generated in the central pin was only 81% of the power generated in the edge pins, and there was a significant azimuthal power variation within each edge pin. (There was little power difference among the edge pins, however: +/- 4%.)

Experiment Vehicle

The test vehicle was a Mark-II integral flowing-sodium loop instrumented with pressure transducers and flow meters at the inlet and outlet. Dysprosium layers around the test vehicle periphery hardened the neutron spectrum and helped to provide a representative axial power profile in the test fuel. The test train within the loop included ~0.8-mm-thick stainless steel flowtube that accommodated a seven-pin bundle. The flowtube was fluted inward in six azimuthal places to partially represent the geometry of an additional ring of pins and thus provide a more-prototypical local flow area around the periphery of the bundle. Surrounding the flowtube was a stainless steel outer tube. Between the flowtube and outer tube was an inert-gas annulus in which thermocouples and strain gauges were located, attached to the outer surface of the flowtube.

Test Fuel

The seven fuel pins contained stacks of 75%UO₂-25%PuO₂ (79 at% enriched U) dished pellets, with total fuel height of 34.3 cm. The pellet stacks were helium bonded to 5.84 mm diameter, 0.38 mm wall, 20% cold-worked 316 stainless steel tubing. All pins had been pre-irradiated in a fast (EBR-II) spectrum to ~7 at% burnup.

Conditions

The J1 power transient consisted of a 1.3 s preheat at about 70-75 kW/m (peak axial) in the edge pins followed by a linear power rise for 3.5 s reaching ~100-110 kW/m at pin failure, whereupon the power was rapidly dropped to a low value. The sodium coolant flow rate was 900-940 cm³/s prior to flow disruption at pin failure.

Results

Three of the seven pins failed, all being edge pins of the bundle. Prior to failure, some of the fuel columns of the edge pins had elongated by up to ~ 3 cm. The failures were no more than ~ 3 cm long and extended axially slightly into the region occupied by UO₂ insulator pellet above the fuel. The failures involved about half the circumference of the cladding. Essentially all of the fuel that was molten in those pins when they failed escaped from the cladding (a total of ~53 grams from the three pins) and remained near the failure sites. After some apparently anomalous post-transient flow behavior, the final flow rate was about 70% of the initial flow rate.

Applications

The J1 test results were unique in that it was the first time in a TREAT test that oxide fuel failed and released a significant quantity of fuel and yet resulted in a coolable geometry, that is, no complete coolant flow blockage formed. The test also confirmed that failure of oxide fuel pins during a simulation of a slow LMFBR overpower accident could be achieved in TREAT.

Calibrations in TREAT

Because the test vehicle and fuel pin bundle in test J1 were so similar to the ones used in TREAT test H6, the results from the H6 calibration were used for J1 after being adjusted for the minor difference in the as-irradiated neutronic characteristics of the different fuel pins used in the two tests. The H6 calibration included low-power irradiations of fresh pins in a seven pin bundle (including monitor foils placed around the periphery of the pins), plus low- and high-power irradiations of monitor wires. The irradiated samples were then measured for fission density, including axial and radial relative density distributions in the fuel pins. The resulting calibration factor was supported by measurements from a J1 pre-test heat balance transient using the actual test hardware.

23HUT-series Experiments

The HUT series tested the response of FFTF-like mixed-oxide fuel pins to conditions simulating 3 ¢/s, 50 ¢/s, and 3 \$/s FFTF/CRBR reactivity insertions unterminated by the plant protection system. A one-page summary of each HUT-series test is included in Appendix U [118, 119, 120, 121, 122, 123, 124, 125, 126, 127].

Number/Time

Nineteen HUT tests were performed during 1973-1979. Three of the tests were performed in sodium loops and the other 16 in static NaK capsules. Midway during the testing program (about spring 1976), many of the capsule tests were renamed, and some were given the same name as a previous test in the series. Early test reports published before the names were changed referred of course to the tests by their original names, whereas later reports used the final names but generally also included a key relating the final names to the original names. To help reduce confusion, the TREXR database uses the final names, and, for additional clarity, also uses secondary names that include the unique four-digit number of the TREAT transient used in performing the test. The following table indicates the relationships among the names used for the capsule tests in this series.

TREAT Transient Number	Final Name Officially Assigned to the Test	TREXR Name Also Used for the Test	Original Name Officially Assigned to the Test
1642	HUT 3-2A	HUT3x1642	HUT 3-1A
1518	HUT 3-2B	-----	HOP 3-3C
1614	HUT 3-5A	HUT3x1614	HUT 3-3A
1898	HUT 3-5B	HUT3x1898	-----
1924	HUT 3-6A	-----	-----
1928	HUT 3-6B	-----	-----
1634	HUT 3-7A	HUT3x1634	HUT 3-5A
1645	HUT 3-7B	HUT3x1645	HUT 3-5B
2032	HUT 5-1A	HUT5x2032	-----
1600	HUT 5-2A	HUT5x1600	HUT 5-1A
1648	HUT 5-2B	HUT5x1648	HUT 5-1B
1577	HUT 5-5A	HUT5x1577	HUT 5-3A
1903	HUT 5-5B	HUT5x1903	-----
1592	HUT 5-7A	HUT5x1592	HUT 5-5A
1637	HUT 5-7B	HUT5x1637	HUT 5-5B
2133	HUT 5-9A	-----	-----

Purpose

Qualification of the mixed-oxide fuel pin design for use in core loadings in FFTF and later in CRBR required experimental verification of the margins to cladding failure, and the ability of transient fuel behavior codes to adequately predict those margins, over a range of reactivity insertion events not terminated by the reactor's plant protection system. Data regarding the fuel pin response up to and including cladding failure were needed for that purpose, primarily regarding pre-irradiated fuel.

Approach

Fast-flux-irradiated fuel was needed, and because EBR-II was the only available fast-flux irradiation facility, the fuel pins to be tested were fabricated with fuel columns having height equal to that of the EBR-II core and fuel pin overall lengths able to be accommodated in EBR-II. Otherwise, the fuel pins basically matched the reference FFTF pin design. A test matrix was designed to span the relevant parametric ranges of fuel irradiation power, burnup, and fluence, the FFTF reactivity insertion rate, and the cladding failure threshold during the reactivity-induced power transient. For simplicity, most of the tests were performed on single pre-irradiated fuel pins located within static NaK coolant, but three tests (each with three pins) were performed in sodium. Data were taken as needed to support the validation of existing fuel-pin transient thermal-mechanics codes. Fuel motion monitoring using the TREAT Fast Neutron Hodoscope was included in all tests except three of the static NaK capsule tests.

Limitations

Several features resulted in test conditions that were less than fully prototypical of those calculated for FFTF and CRBR under the assumed reactivity insertion scenarios considered. Yet, the achieved conditions lay within the range needed for code validation and subsequent use of the codes to predict fuel transient response under fully-prototypical hypothetical conditions. Thus it was considered acceptable that: (a) in the capsule tests (in which the coolant was static), the anticipated transient response of the fuel pin would be controlled mainly by radial heat-transfer effects; (b) the axial variation of fuel micro-structure in the as-irradiated fuel pin reflected the EBR-II axial power profile rather than the FFTF profile; (c) the fuel cladding fluence-to-burnup ratio was determined

by the EBR-II neutron spectrum rather than the FFTF spectrum; and (d) the radial profile of the fission power generated in the test fuel during the transients in TREAT's thermal-neutron environment was considerably peaked toward the fuel periphery, even with the neutron filtering provided by a thermal-neutron absorber on the test vehicle. Most of these non-prototypicalities are common in TREAT experimentation.

Experiment Vehicle

The static NaK capsule vehicle was designed to contain a single fuel pin having a diameter equal to the reference FFTF fuel pin design diameter and length suitable to be accommodated in EBR-II. Surrounding the fuel pin was a 1.3-mm thick NaK annulus bounded by a nickel heat sink generally (in most tests) of 7.8 mm thickness. Thermocouples were located at several axial locations within the heat sink, with their tips either barely in contact with the NaK annulus or embedded within the nickel. Radially outward from the heat sink were thicknesses of NaK and structural walls to meet containment safety requirements. Around the periphery of the vehicle primary containment wall was a B6Si thermal neutron filter to harden the TREAT neutron spectrum impinging on the test fuel sample. The thickness of the neutron filter varied axially to eliminate the natural power peaking at the ends of the fuel column where the fuel was less self-shielding. Heaters located axially adjacent to the test fuel provided an initial axial temperature profile. Mark-II sodium loops were used in three of the tests. The fuel pin holders (test trains) in the loops were designed to accommodate three test fuel pins in parallel, each in its own flow tube. Instrumentation included thermocouples, flow meters, and pressure transducers.

Test Fuel

The fuel pins in all of the tests contained stacks of mixed oxide fuel (75 wt% UO₂ -- 25 wt% PuO₂, with O/M = 1.96 – 1.99) helium bonded to 20% cold-worked type 316 stainless steel cladding -- with one possible exception: the gas in the EBR-II-irradiated pins used in HUT 5-9A may have been removed prior to the test. The cladding radial dimensions (5.84 mm OD, 0.38 mm thickness) and fuel smeared density (86-89%) were essentially the same as the reference designs for the FFTF and CRBR reactors. In all of the tests performed in static NaK capsules, the pins had fuel column lengths of 34.3 cm and overall pin lengths of 101 to 155 cm. Pins used in the two sodium loop tests had prototypic fuel-column length (91.5 cm) and overall-pin length. All of the fuel pins for the capsule tests had been preirradiated in a matrix of burnup (6 to 112 MWd/kg), fluence (1 to 11 x 10²² n/cm²), and peak cladding temperatures (455 to 510 °C). Two of the loop tests (HUT 1-6A and HUT 1-8A) used EBR-II irradiated fuel; the third loop test (HUT-L2) used fresh FFTF-prototypic-length fuel pins.

Conditions

Basically, each test exposed the fuel samples to one of three types of power transients, representing either a 3 ¢/s, 50 ¢/s, or 3 \$/s reactivity insertion ramp untermiated by the FFTF or CRBR plant protection system. The power rise used to represent a hypothetical 3 \$/s reactivity accident was roughly exponential with a period of approximately 150 to 260 ms. The 50 ¢/s transient had a power rise corresponding to an exponential rise with a period between two and three seconds. The TREAT reactor was scrammed at the peak power and total energy generation desired to meet the test objectives. In the capsule tests with static coolant, the initial temperature and axial temperature distribution within the region radially surrounding the fuel pin were established (using electric heaters) before the transient to help generate test pin transient temperatures during the TREAT transient that would closely match those predicted for an FFTF pin during the simulated reactivity accident. In the loop tests, the initial sodium temperature and flow rate were chosen with the same goal.

Twelve of the tests subjected medium burnup (30-54 MWd/kg) fuel pins to either 50 ¢/s or 3 \$/s transients, performed in pairs, as shown in the following table. Test names that are followed by “(F)” indicate which tests caused cladding failure. Two pairs of tests (all performed with 3 \$/s transients) did not result in spanning the cladding failure threshold.

PAIRS OF CAPSULE TESTS	Pre-irradiation Power of Medium-burnup Fuel		
	High Power (39-41 kW/m)	Intermediate Power (26-33 kW/m)	Low Power (14-19 kW/m)
Simulated Reactivity Insertion			
50 ¢/s	HUT 5-2A HUT 5-2B (F)	HUT 5-5A (F) HUT 5-5B	HUT 5-7A HUT 5-7B (F)
3 \$/s	HUT 3-2A (F) HUT 3-2B (F)	HUT 3-5A HUT 3-5B	HUT 3-7A HUT 3-7B (F)

In addition to the tests shown in the “Pairs of Capsule Tests” table immediately above, seven other tests were performed: HUT 3-6A, HUT 3-6B, HUT 1-6A, HUT 1-8A, HUT-L2, HUT 5-1A, and HUT 5-9A. Four of those additional tests were performed on fuel irradiated to high burnup (112 kW/m). Two of those high-burnup fuel tests (HUT 3-6A and HUT 3-6B) were performed in capsules on intermediate-power-irradiated pins and caused fuel pin failure. In each of the other two high-burnup fuel tests, three pins were tested in sodium loops (HUT 1-6A with intermediate power-irradiated pins and HUT 1-8A with low-power-irradiated pins). In loop test HUT 1-6A, during a preliminary check-out transient at constant power, sodium flow was planned but did not occur; post-run analysis indicated that the still-intact fuel pins had sustained enough damage to provide useful technical information but the damage was too great for the pins to be suitable for use in the 3 ¢/s overpower transient that was intended to be performed next on that fuel. The other loop test (HUT 1-8A) on high burnup fuel (pre-irradiated at low power) included five preliminary relatively-low constant-power transients, the last of which was accompanied by a continuous reduction of the coolant flow rate. The final transient included a mild power rise (about 50% overpower in 20 s) without a flow reduction. Results from that test are unavailable. A third three-pin loop test (HUT-L2) subjected unirradiated full-length FFTF-style fuel pins to a 50 ¢/s transient with flowing sodium coolant and did not cause cladding failure. The remaining two of the seven additional tests (HUT 5-1A and HUT 5-9A) were 50 ¢/s simulations performed in capsules. HUT 5-1A tested a high-power, low burnup pin to failure; HUT 5-9A tested a low-power, low-burnup pin.

Results

In most cases where all, or a large fraction, of the cladding remained intact during the test, posttest examinations provided important evidence of the fuel and cladding damage, including locations of fuel melt fronts, fuel swelling, gas release from fuel, cladding swelling, and cladding breach location. The power-time history of the test fuel during the test was determined by pre-test measurements of the axial flux profile in the test fuel, calorimetric measurements and computations that related TREAT power to test-fuel power, and measurement of the TREAT reactor power during the test. Little or no information provided by the hodoscope for any of the HUT tests was found. [It is noted that the fact that cladding failure occurred in HUT 5-5A but not in HUT 5-5B is consistent with the value of TREAT integrated power in HUT 5-5B calculated from the TREAT P(t) curve (~580 MJ) but not consistent with the value quoted in the test reports (880 MJ), the latter possibly being an initial typographical error.]

Applications

The results from the tests were used to validate, over a wide range of important parameters, computer codes that were designed to predict the cladding failure of fuel pins of the type tested. The parameters included as-irradiated fuel conditions and damage to the pins caused by the test, i.e., from near cladding failure to marginally beyond cladding failure. The validation process took account of the observation that, in many of the capsule tests, there was considerable difference between (a) temperatures indicated by the various thermocouples in the heat sink adjacent to the fuel pins and (b) temperatures computed by heat-transfer codes using (i) the fuel sample power-time history calculated using results of pre-test calorimetry of test samples heated in the capsule and (ii) measured TREAT reactor power-time history during the test. Because of uncertainties in measured power calibration, the analyses generally scaled the power-time histories to best match the temperatures indicated by the thermocouple data. (The temperatures indicated by the thermocouples were assumed to be accurate in the reports, although it appears that no analysis was made of how imperfect thermal coupling between the thermocouples and the heat sink in which the thermocouples were embedded may have affected the ability of the thermocouples to accurately indicate the actual temperatures of the heat sink or the NaK at the location of the sensing point of each thermocouple.)

Calibrations in TREAT

Power calibration measurements were generally made by transient heating of the fueled test vehicles using relatively low-energy transients and then determining the amount of heat generated in the sample via calorimetry.

24HOP/HUC-series Experiments

The HOP/HUC series tested pre-irradiated FFTF-like mixed-oxide fuel pins under simulations of 3 ¢/s and 3 \$/s reactivity insertion events, generally terminated by the FFTF/CRBR plant protection system. A one-page summary of each HOP/HUC-series test is included in Appendix V [128, 119, 129, 127, 130, 131, 132, 133].

Number/Time

Ten tests with test-name prefixes HOP, HOP-PTO, or HUC-PTO were performed during 1972-1977. Seven tests had names with prefixes HOP ("HEDL OverPower"), two had names with prefixes HOP-PTO ("HEDL OverPower with Post-Transient Operation"), and one had prefix HUC-PTO ("HEDL UnderCooling with Post-Transient Operation").

Part way through the testing program, five HOP tests were renamed. Two of those renamed tests were given the same name as had been used in a previous test in the series. Early test reports published before the names were changed referred, of course, to the tests by their original names; later reports used the final names but also included information relating the final names to the original names. To help reduce confusion, the TREXR database uses the final names, and, for additional clarity, also uses secondary names that include the unique four-digit number of the TREAT transient used in performing the test. The following table indicates the relationships among the names used for these re-named HOP tests.

TREAT Transient Number	Final Name Officially Assigned to the Test	TREXR Name Also Used for the Test	Original Name Officially Assigned to the Test
1452	HOP 3-2A	HOP3x1452	HOP 3-3A
1511	HOP 3-2B	HOP3x1511	HOP3-3B
1518	HOP 3-2D	HOP3x1518	HOP 3-3C
1542	HOP 3-1B	HOP3x1542	HOP 3-2A
1543	HOP 3-2C	HOP3x1543	HOP 3-2B

Two other tests were referred to in the literature using inconsistent names. The test that used TREAT transients 1806 through 1809 was referred to as either HOP-PTO 2-2A or HUC-PTO 2-2A; in the following text, this test will be referred to as HUC-PTO 2-2A. The test that used TREAT transients 1798 through 1800 was referred to as either HOP-PTO 3-2C or HOP-PTO 3-2E. It will be referred to below as HOP-PTO 3-2E.

Three other tests (HOP 1-6A, HOP 3-1A, and HOP-PTO 1-2A) were not renamed.

Purpose

The HOP tests were performed to (a) evaluate prototypic FFTF mixed-oxide pin performance during postulated 3¢/s and 3 \$/s FFTF accident transients terminated by the FFTF plant protective system (PPS - Secondary System), (b) to demonstrate that the performance requirements for the FFTF fuel pin can be met and thus confirm the conservatism of the FFTF fuel pin design basis, and (c) to extend the mixed-oxide fuel pin transient behavior data base beyond FFTF goal conditions. The HOP tests were not intended to relate directly to conditions causing cladding failure nor to post-failure behavior. Information (from posttest examinations) regarding transient-induced fuel changes (changes in cladding dimensions, fission gas release, and microstructural changes) was desired. The possibility of the posttest examination also yielding information regarding conditions relevant to cladding failure mechanisms was anticipated.

The HOP-PTO tests were intended to subject pre-irradiated FFTF-like fuel pins to mild overpower transients with some pins destructively-examined post-transient and other pins continued in further steady-state irradiation in EBR-II to much-higher burnups in order to determine the effect of the transient on the fuel burnup limit.

The HUC-PTO test was performed for reasons similar to the HOP-PTO tests except that the transient was an undercooling transient rather than an overpower transient.

Approach

The ten tests were defined to be part of a set of proof tests to verify prototypic FFTF pin performance in response to 3 cents/s and 3 dollars/s transients, with nine of the tests simulating transients terminated by the FFTF plant protective system (PPS) and one test simulating an unterminated transient. The fuel pins in all tests would be of FFTF-like design, pre-irradiated in EBR-II. It was intended that all of the transients in these tests would be mild enough to generally not cause cladding breach or significant cladding damage. Six of the tests would each subject single pins in static NaK capsules to fuel conditions calculated for 3 \$/s events. Five of those six 3 \$/s simulations would involve transients terminated by the FFTF plant protective system (PPS), and one would be a simulation not terminated by the PPS. The other four of the ten tests in this series would each test three pins in flowing sodium to two or more transients. In each case, the first transient would be a 3 cents/s protected simulation. The subsequent transient(s) would be the

same as the initial transient with the same or different sodium flow rate, or a 3 \$/s reactivity insertion simulation, or a transient including a continuous reduction in coolant flow rate.

All of the tests would use fuel in the 24 to 48 MWd/kg burnup range except that a low-burnup pin would be tested in one of the PPS-terminated 3 \$/s tests in static-NaK, and very-high-burnup pins would be tested in one of the 3 ¢/s PPS-terminated tests in flowing sodium.

One or more pins from each of three of the tests in flowing sodium (the tests with “PTO” in the test name) would be returned to EBR-II for post-transient operation to high burnup.

Test conditions (initial coolant temperature, initial coolant axial temperature distribution, TREAT preheat/flattop power level and duration, and TREAT power-time history for the overpower segment of the transient) were selected apparently with the goal of generating the same time-dependent fuel and cladding temperatures in the test fuel as were calculated would occur in fuel of that burnup if the pins were in the FFTF core during the postulated reactivity insertion event.

Limitations

The radial power profile in the test fuel was (as is typical in most TREAT tests) highly peaked near the fuel periphery due to the thermal neutron spectrum in the TREAT core. This affected the degree to which the radial temperature profile in the fuel and cladding during the test represented the profile that would be generated in an actual reactivity insertion accident in a fast reactor such as FFTF. An additional influence on the fuel-to-cladding temperature difference is attributable to the use of static coolant, although that influence may have been relatively small because of the fast, 3 \$/s transients used in the capsule tests. In addition, determination of the power coupling between the test fuel and the TREAT core was challenging in the capsule tests due to thermal effects around the thermocouples that were used and to the need for a calorimetric approach.

Experiment Vehicle

Two types of test vehicles were used. One was a capsule designed to test single pins in a static NaK environment. In that capsule, the fuel pin was radially surrounded by a 1.6-mm-thick NaK annulus, the outer boundary of which was the inner surface of a cylindrical, 15-mm-thick nickel heat sink. Thermal-neutron filtering was provided by boron-containing layers around the outside of the vehicle, except that no such neutron filtering was present on the capsule used during the HOP 3-2A test. Heaters located axially adjacent to the test fuel provided an initial axial temperature profile. The vehicle was designed such that it would house the fuel pin during the pin’s pre-irradiation in the General Electric Test Reactor (GETR) and also serve as the test vehicle during the TREAT transient test.

The tests in sodium were each performed on three fuel pins, located in individual stainless steel flowtubes in parallel array surrounded by inert gas and housed within boron-filtered Mark-IIIC integral flowing-sodium loops. The flowtubes in the PTO-designated tests were of 9.3 mm ID and 0.9 mm thick. In test HOP 1-6A, the flowtube dimensions were 8.1 mm ID and 0.7 mm thick.

Test Fuel

All of the fuel pins used in these tests were comprised of solid-pellet hypostoichiometric mixed uranium-plutonium oxide helium bonded to stainless steel cladding. Because the fuel had been irradiated in EBR-II, the fuel column lengths were essentially equal to the EBR-II core height (~34 cm). The fuel burnups were in four groups: ~ 7 MWd/kg for HOP 3-1A, ~45 to 48 MWd/kg for HOP 3-2C, ~112 MWd/kg for HOP 1-6A, and ~25 to 35 MWd/kg for the other seven tests.

Conditions

Test conditions for the six single-pin tests in static NaK are listed in the table below in decreasing order of apparent severity. All six tests simulated conditions associated with a 3 \$/s event. In each test, the power transient began with a quick rise to a constant-power (“flattop”) segment of the transient. The flattop duration was 3.2 to 5.9 s and in the range 66 to 75 MW, except for test HOP 3-2A (which had no neutron filtering and thus a higher power coupling factor) for which the flattop power was only 27 MW. The flattop durations were in the range 2.9 to 5.9 s. Immediately following the flattop segment was a sharp power rise to a peak power, and then a sudden drop to low power (typically, a reactor scram). All of the tests were intended to simulate PPS-terminated events, except for test HOP 3-2D which was intended to simulate an event not terminated by the PPS. For reasons that are unclear, the severity of some of the tests was considerably different than had been planned. Two of the PPS-terminated tests (HOP 3-1B and HOP 3-2C) were inadvertently overpowered by ~25%, but yielded results relevant to cladding integrity. One of the tests (HOP 3-2A) was inadvertently underpowered by ~7%, resulting in an insignificant peak fuel temperature.

The power coupling factors for tests HOP 3-1A, HOP 3-1B, HOP 3-2B, and HOP 3-2C were all reported to be in the range 3.8 to 4.6 W/g per TREAT MW; the PCFs for HOP 3-2A and HOP 3-2D were not found, but the value for HOP 3-2D was likely to also have been in that range.

Single-Pin Tests in Static NaK

	Fuel Pre-irradiation Conditions		Overpower Segment of the Transient (P_{FT} indicates the flattop power level)			
Test Name (HOP-prefix)	Burnup (MWd/kg)	Peak Power (kW/m)	Peak Power (P_{peak}/P_{FT})	Relative Power Rise Rate from Flattop $[(P_{peak} - P_{FT}) / P_{FT}]$ per sec	Energy From End of Flattop to Peak Power (in terms of P_{FT} – seconds)	Test Results
3-2D	30	31 - 34	16.2	9.70	3.30	Cladding severed at midplane; fuel motion
3-1B	24.5	42	15.4	4.93	0.89	Incipient fuel melting; ~0.35% cladding strain
3-2C	45 - 48	42	12.5	4.26	0.76	No fuel at liquidus; 1 mm dia. central void
3-1A	7.5	31 - 34	12.5	3.75	0.74	No fuel at liquidus; 1 mm dia. central void
3-2B	33	31 - 34	10.1	3.04	0.71	Little or no change in fuel or cladding
3-2A	30	31 - 34	4.7	2.56	0.89	Maximum fuel temperature less than during pre-irradiation and no additional cladding strain

All four of the tests in flowing sodium (HOP 1-6A, HOP-PTO 1-2A, HOP-PTO 3-2E, and HUC-PTO 2-2A) began by first exposing the three fuel pins to transient conditions representing those that the fuel would experience during a 3 \$/s PPS-terminated event. The TREAT power transient for that initial simulation was the same for each test: a fast rise to the power flattop, a hold at power for about five seconds, then a gradual rise to peak power during the next eight to nine seconds, then a

reactor scram. The power coupling factors in all of the tests were approximately 3.8 W/g-MW for the hottest of the three pins and about 3.4 W/g-MW for the two cooler pins. All four tests began with the flowing sodium at 375°C.

In test HOP-PTO 1-2A, three ~31 MWd/kg burnup pins in flowing sodium were exposed to four nearly-identical transients, all intended to simulate fuel conditions during a PPS-terminated 3 cents/s event. The 60 MW flattop of ~5 s duration was followed by a 9-s power rise which peaked at ~77 MW, and then a reactor scram. The total TREAT energy generated in each transient was ~950 MJ. The sodium flow rate was nominally the same during all four of the transients.

In test HOP 1-6A on ~110 MWd/kg burnup fuel, two identical transients simulating fuel conditions in PPS-terminated 3 cents/s events were performed. In each of the transients, a flattop at 45 MW was followed by an 8-s rise to a peak power of 57 MW, yielding a total TREAT energy of 704 MJ at the end of the transient. The flow rate was 106 g/s during the first transient and 118 g/s during the second transient, both about 10 to 20% lower than had been planned. Also, the power level in the hottest of the three pins (the one with the highest power coupling factor) during the power flattop was about 12% higher than that pin had experienced during its pre-irradiation in EBR-II.

In test HOP-PTO 3-2E on three 31 MWd/kg burnup fuel pins, the initial transient was a ~60 MW flattop followed by a ~8 s rise to a peak power of ~77 MW, yielding a total of 950 MJ (identical to the initial transient for HOP-PTO 1-2A). It was followed by a second transient identical to the first. Then a third transient was performed in which intended fuel conditions were to represent a PPS-terminated 3 dollar/s event. In the latter transient, a power flattop of 50 MW was followed by a 0.4 s rise to peak power of ~430 MW, then scram, for a total of 460 MJ TREAT energy.

In test HUC-PTO 2-2A, three ~31 MWd/kg fuel pins were subjected to four transients. The first three transients were the same as the initial transients used in tests HOP-PTO 1-2A and HOP-PTO 3-2E. The fourth transient consisted of a power flattop at 63 MW lasting ~18 s, with a continuous coolant flow reduction from 100% flow rate down to 58% flow rate starting at ~ 5 s after the beginning of the flattop, followed by reactor scram.

Results

The main results from the single-pin tests in static NaK were as indicated in the preceding table. In test HOP 3-2D, a high concentration of fuel was found near the midplane where the cladding had been severed, and some fuel had traveled to locations beyond each end of the initial fuel region. From the midplane failure elevation to an elevation about 4 cm below were localized breaches where the cladding had melted through. Everywhere else the cladding remained intact. In test HOP 1-6A, the hottest pin breached, whereas the two cooler pins did not. The breach was of pin-hole size at the fuel midplane. Whether the breach occurred during the first or second transient is unknown. The breach was the only observed change; no measureable cladding swelling, fuel melting, or fuel relocation occurred in that pin or in either of the two unfailed pins from that test. In all of the pins from the "PTO"-type tests there was no detected change in the fuel or cladding condition resulting from the transient(s), and pin(s) from each PTO test were returned to EBR-II, as planned, and further irradiated to ~120 MWd/kg without cladding breach.

Applications

The tests helped to demonstrate the ability of irradiated FFTF-type fuel to withstand conditions representative of PPS-protected postulated 3 ¢/s and 3 \$/s reactivity insertion events relevant to FFTF and CRBR, as pertinent to fuel performance requirement for those plants. Test HOP 3-2D (which involved significant cladding failure and fuel motion) became the lead experiment for the

HUT-series tests in TREAT. Specific applicability of the tests depended on results of detailed computational analyses that estimated fuel conditions during the tests (taking account of unprototypicalities and power uncertainties) and how well those conditions matched conditions predicted for fuel behavior in FFTF.

Calibrations in TREAT

The power coupling factors for the tests were determined by calorimetry, using fresh (unirradiated) fuel pins, calibration hardware similar to the test hardware, and TREAT flattop irradiations. Consideration was also given to adjusting the power coupling values to more closely align with measurements taken by thermocouples embedded in the nickel heat sink wall during the test transients.

25PNL-series Experiments

The PNL-series were the first transient overpower tests of EBR-II-irradiated FFTF-like MOX fuel pins. They investigated fuel pin pre-failure damage and failure threshold using unshaped burst transients. A one-page summary of each PNL-series test is included in Appendix W [128, 123, 134, 135].

Number/Time

Seven tests were performed during the period 1970 through 1974. The names assigned to the tests are the same as the identification numbers of the fuel pins used in the tests.

Purpose

The tests were performed to help confirm the FFTF fuel pin design and to verify the design's performance using damage criteria.

Approach

Fuel of FFTF-like design that had been pre-irradiated in the EBR-II fast neutron flux would be subjected to natural-burst transients in TREAT, with the peak power and total energy varying from test to test in order to cause cladding failure in some tests but only pre-failure damage in others. All would be single-pin tests with the fuel pin located in static NaK coolant. Parameters to be varied among the tests included burnup (low, moderate), fuel-pellet design (solid, annular), and fuel smeared density (79%, 84%, and 88% of theoretical).

Limitations

Because these tests were performed before TREAT was able to produce shaped transients, only natural burst transients were used. In addition, thermal-neutron filtering of TREAT experiment vehicles had not yet begun, resulting in the fuel surface-to-center power ratio during the TREAT transient being about four to one in these tests. Consequently, both the fuel-pin radial temperature profiles at the beginning of the power burst and during the power burst were not as prototypic as in later TREAT testing programs. The use of static coolant rather than flowing coolant added to the non-prototypicality but was probably less important due to the short duration of the power bursts.

Experiment Vehicle

All of the tests were performed in capsules that positioned a single fuel pin in a static NaK annulus surrounded by a nickel heat sink. Temperatures in or near the NaK annulus were measured by thermocouples. The capsules had no thermal-neutron filtering.

Test Fuel

The fuel pins in each test were of FFTF-like stainless-steel-clad helium-bonded mixed-oxide fuel pin design, except with shorter fuel columns (34.3 cm) equal to the EBR-II core height. The fuel pins in all of the tests were preirradiated at peak power levels in the range 27 to 31 kW/m. The pins for tests PNL 1-1, PNL 1-2, PNL 1-7, and PNL 1-8 reached burnups in the range 8 to 9 MWd/kg. The pins for tests PNL 2-1, PNL 2-10, and PNL 2-11 reached burnups of 41 to 43 MWd/kg. Three of the pins (PNL 1-1, PNL 1-2, and PNL 2-1) were of solid-pellet design and had as-fabricated fuel smeared densities of 87-88% of theoretical. The other four pins had annular pellets (1.37 mm diameter central hole) and smeared densities of 84% for PNL 1-7 and PNL 1-8 and 79% for PNL 2-10 and PNL 2-11.

Conditions

The initial NaK temperature and the description of the TREAT power transient were available for only two of the seven tests: PNL 1-1 and PNL 1-8. Both of those were described as starting at 150 oC NaK temperature and subjecting the fuel sample to a natural burst (Gaussian-like shape) power transient. The peak TREAT power and burst full width at half maximum were 124 MW and 0.97 s for PNL 1-1 and 139 MW and ~0.95 s for PNL 1-8. The corresponding reported fuel sample peak power and total energy generation were ~250 kW/m and 1.23 kJ/g for PNL 1-1 and ~280 kW/m and 1.3 kJ/g for PNL 1-8. It is likely that the same starting temperature and similar power transients were used for the other tests.

Results

The cladding failed in tests PNL 1-1 (gross failure; 80% of the fuel melted), PNL 1-8 (localized failure; no fuel melting), PNL 2-1 (separated into two pieces; failure extended from $x/L = 0.37$ to 0.52), and PNL 2-11 (gross failure; failure extended from $x/L = 0.82$ to 0.93). The cladding of the pins in the other three tests remained intact, no fuel in those pins melted, and post-test fuel fission-gas release measurements were made. Of the total amount of fission gas generated in the fuel during its preirradiation, the fraction released during the preirradiation (from sibling pin analysis) and the fraction released during the transient were approximately 8% and 24% in PNL 1-2, 8% and 18% in PNL 1-7, 8% and 45% in PNL 1-8, and 50% and 78% in PNL 2-10. (The 8% release during preirradiation for PNL 1-7 and PNL 1-8 assumes the same as reported for PNL 1-2). It was concluded from the tests that (a) fission gas can be released from low-power-irradiated fuel pins prior to fuel melting, (b) intra-granular fission gas bubble formation and growth can appreciably increase the fuel volume, and (c) during transient heating of the type used in these tests, cladding can fail before a significant amount of fuel melts.

Applications

The results of the tests helped initiate a database upon which the adequacy of the FFTF fuel pin design was later confirmed. The series was soon followed by two similar series of TREAT tests (HOP-series and HUT-series) on EBR-II-irradiated FFTF-design fuel pins in which more-prototypic transient heating conditions were used.

Calibrations in TREAT

The TREAT and test fuel power levels indicated for tests PNL 1-1 and PNL 1-7 corresponded to a test-fuel to TREAT power coupling factor of about 2 kW/m per MW. The basis for this value was not available.

26 ORNL-TR-series Experiments

The ORNL-TR tests investigated the response of sol-gel manufactured Sphere-Pac mixed-oxide fuel pins to severe transient overpower conditions, in comparison to the response of pelletized fuel pins. A one-page summary of each ORNL-TR-series test is included in Appendix X [136, 137, 138, 139, 140].

Number/Time

Two tests were performed during 1969. (The tests are reported in the literature with names simply TR-1 and TR-2. To avoid confusion with other tests having prefix TR in their test names, these two tests are designated here with the prefix ORNL-TR, with ORNL indicating the performing organization.)

Purpose

The tests were intended to compare the transient response of Sphere-Pac fuel with pelletized fuel under conditions that simulate severe power excursions.

Approach

The transients were designed to melt about 25% of the 81% dense fuel in the peak power region of the test capsule in test ORNL-TR-1 and 50% of the 81% dense fuel in ORNL-TR-2 using TREAT power bursts, pins with essentially zero burnup, and a static sodium environment for the pins. The parameters involved in the tests were the fuel column geometry (Sphere-Pac or pelletized), fuel smear density (~89% or ~81%), and the relative power and total energy generation among the pins.

Limitations

Differences in the power coupling among the three pins in each test limited the extent to which pin-to-pin fuel response could be compared. Relative to the highest-power pin in the test, the powers in the other two pins were 88% and 67%. In ORNL-TR-1, the Sphere-Pac and pelletized pins differed in power and energy by as little as 12%, but in ORNL-TR-2 they differed by 33%. Differences in the fuel-to-cladding heat transfer between the two fuel types added difficulty to fuel temperature predictions.

Experiment Vehicle

Both tests used the same test-vehicle design. The fuel pins were held in tandem (vertically stacked one above the other), within a static sodium annulus, with pin A at the top and pin C at the bottom. A means of measuring the fuel length change during the transient was incorporated into the vehicle used in ORNL-TR-2 (and possibly also in the ORNL-TR-1 vehicle). One or more thermocouples were present to measure temperatures near the sodium. No other details of the test vehicle were found.

Test Fuel

The fuel in all six of the test pins was (U-20%Pu)-O₂ made using the sol-gel process. Three of the pins (pin B in ORNL-TR-1 and pins A and B in ORNL-TR-2) had fuel columns of Sphere-Pac particles. The other three pins had solid-pellet fuel columns. None of the fuel had been irradiated prior to being used in the TREAT tests. The cladding of all of the pins was type 304 stainless steel tubing of 6.45 mm OD and 0.41 mm wall thickness, helium bonded to the fuel. The fuel smeared density of all pins was in the range 79-81 % of theoretical, except for pin A in ORNL-TR-1 which had a smeared density of 89%.

Conditions

In both tests, the fuel pins were located in a static sodium annulus. The initial sodium temperature in both tests was in the range 410-414 °C for pins A and C and 436-449 °C for pin B. Information regarding the exact shape of the TREAT power transient was not available, but likely the transients were basically natural (unshaped) power bursts. The reported peak TREAT power was 202 MW in both tests, and the reported TREAT total energy generation (integrated power) was 140 MJ in ORNL-TR-1 and 172 MJ in ORNL-TR-2. The power coupling between the TREAT core and the test fuel pins differed among the pins, presumably due to such factors as fuel smeared density, pin location within the TREAT core, effects of the thermal neutron flux in TREAT, and uncertainties associated with estimation of power coupling factors and measurements of TREAT core power and energy. The reported pin peak-axial linear power and energy generation values are as shown in the table below.

Pin	Fuel Type	Smear density (% TD)	Initial sodium temp. (°C)	Peak linear power (kW/m)	Total energy release (kJ/g)	Radius of equiaxed zone posttest (r/R)
TR-1A	Pellet	88.9	410	577	1.63	0.85
TR-1B	Sphere-Pac	80.9	449	505	1.57	1.00
TR-1C	Pellet	79.7	414	386	1.22	0.00
TR-2A	Sphere-Pac	82.0	400	576	2.17	1.00
TR-2B	Sphere-Pac	80.9	436	533	2.04	1.00
TR-2C	Pellet	80.2	410	343	1.32	0.70

Results

None of the six fuel pins failed during the tests, and the amount of cladding diameter change was minimal or none. The only pins in which fuel melted were the two Sphere-Pac pins in ORNL-TR-2; 50% areal melting at the axial midplane was measured in both pins. Fuel central voids formed only in the three Sphere-Pac pins. Equiaxed grain growth extended radially to the outer edge of the fuel in all of the Sphere-Pac pins but only to 85% of the fuel outer radius in pelletized pin A in ORNL-TR-1 and to 70% in the pelletized pin in ORNL-TR-2. No equiaxed grain region formed in the cooler pelletized pin in ORNL-TR-1. No region of low fuel density developed in any of the pins. The fuel columns of all three of the Sphere-Pac pins shortened as a result of the transient.

Applications

The results provided some of the first experimental information regarding the behavior of sol-gel manufactured Sphere-Pac design MOX fuel pins during a severe overpower transient.

Calibrations in TREAT

Power calibration data were obtained by means of TREAT-transient irradiations of flux monitor samples in the test hardware and via measurement of coolant temperature rise during a relatively-low-power transient on the actual ORNL-TR-1 fuel pins.

27 Conclusions

TREAT was utilized for hundreds of transients tests performed on nuclear reactor fuels during its original operation from 1959 until 1994, when the reactor was placed in non-operational standby. Future operation of TREAT will benefit from the information produced during this prior utilization

of the facility, which included a diverse range of experiments. Knowledge of those prior experiments provides a foundation for planning and designing new future tests.

This report provided summary information on selected sets (series) of historic experiments to illustrate the range of experiments (and TREAT's capabilities) that characterized historic TREAT. The experiment sets addressed in this report were chosen from among the several dozen experiment types or experiment series that were performed. Further information on the broader set of experiments performed in TREAT can be found in the TREXR (TREAT Experimental Relational) Database, a compendium of reference documents and data which has been developed over the past several years at ANL [6].

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Appendix A: List of All Experiment Series

Table A.1 includes brief information about each of the many test series that were performed in TREAT prior to 1994. Each series is indicated by a test prefix. The number of tests associated with a test prefix may be several, dozens, or even hundreds, but rarely only one. Where there were many tests, the range of test parameters might have been large, requiring the brief information items in the table to be stated either more generally or more extensively. For some test prefixes, not enough information about the tests was available to provide much (if any) information into the table, but that does not imply that no such information exists anywhere.

Table A.1. Summary Information Regarding All TREAT Historic Experiment Series

Test Name Prefix	Test Sample	Imposed Environment / Conditions	General Outcome of Test(s)
A	Unclad pellet(s)	Inert gas; overpower and/or undercooling	(TBD*)
AIUCPH	Fresh Na-bonded UC pins in Na	Overpower	Limited fuel damage; cladding failure threshold
AN	Thermal reactor, aluminum clad, ring-shape	Flowing steam	(only calibration was performed)
ANLPUC	Fresh gas-bonded UC pins	Inert gas; overpower	(TBD)
ALUXHARRISON	(TBD)	(TBD)	(TBD)
ALUXIVINS1	(TBD)	(TBD)	(TBD)
APED	Fresh Na-bonded UO ₂ ; single pin	Stagnant Na; undercooling and overpower	Limited damage; in-pin fuel motion
APT	(TBD)	(TBD)	(TBD)
AX	Fresh, Na-bonded UC	Flowing Na	Severe damage
BCL	Mixed nitride	(TBD)	(TBD)
BI	(TBD)	(TBD)	(TBD)
C	Fast reactor UO ₂ ; fresh and irradiated; with and without cladding	Stagnant Na; overpower	Ranged from limited fuel damage to breached cladding
CDT	Fast reactor MOX; irradiated	Flowing Na; overpower	In-pin fuel motion; cladding breach
CEN	Fresh (few irradiated) oxides and metals in pellet, rod, and plate geometries, with and without cladding	Stagnant water, steam, or inert gas; undercooling and/or overpower	Limited damage; cladding failure threshold, fuel and cladding interaction with coolant
CENT	Fresh, unclad carbide fuel with internal cooling	Stagnant inert gas; undercooling with overpower	(TBD)
CO	Fast reactor MOX; fresh, low, and medium burnups	Static NaK or flowing sodium; overpower; undercooling plus overpower	Pre-failure fuel motion; cladding failure threshold; mild cladding failure; early post-failure disruption
CS	(TBD)	(TBD)	(TBD)
D	Fast reactor UO ₂ ; fresh; pin bundle	Overpower	Limited fuel damage through mild cladding failure
E	Fast reactor UO ₂ ; fresh, low, and medium burnup; single pin and pin bundle	Flowing sodium; overpower	Pre-failure fuel motion; cladding failure threshold; fuel and cladding disruption and dispersal
EBRMF2-2pin	(TBD)	(TBD)	(TBD)
EBT	Fast reactor MOX; low and medium burnup; single pin and pin bundle	Static NaK or flowing sodium; overpower	Pre-failure fuel motion; mild cladding failure
EM1	Irradiated fast reactor metallic	Static inert gas; overpower	(TBD)
EOS	Fast reactor MOX; fresh and low burnup; single pin	Stagnant inert gas; overpower	Cladding failure and fuel motions driven by fission-gas

			or fuel vapor pressure
F	Fast reactor MOX; low and medium burnup; single pin	Stagnant inert gas; overpower	Fuel disruption and dispersal driven by fuel vapor or fission gas
FA	Irradiated fast reactor metallic	Static inert gas; overpower	
FC	(TBD)	(TBD)	(TBD)
FCT	(TBD)	(TBD)	(TBD)
FCX	(TBD)	(TBD)	(TBD)
FRF	Fresh UO ₂	Flowing steam	(TBD)
GASREL	Fast reactor MOX, medium burnup	Flowing sodium; overpower	Mild cladding failure; fission gas release
GRIST	Gas-cooled fast reactor oxide; pin bundle	Flowing helium	(test not performed)
H	Fast reactor MOX	Flowing sodium; overpower	Pre-failure fuel motion through post-failure fuel and cladding disruption and dispersal
HC	Helium-bonded carbide	Flowing sodium; overpower	
HN	Helium-bonded nitride	Flowing sodium	(TBD)
HOP	Fast reactor MOX; low, medium, and high burnups	Static NaK or flowing sodium	Varied from no changes to localized failures
HUCPTO	Fast reactor MOX, medium burnup	Flowing sodium; undercooling	(TBD)
HUT	Fast reactor MOX, low, medium and high burnup	Static NaK; static or flowing sodium	Pre-failure fuel damage; cladding failure threshold
IDRP	Fast reactor U-fissium alloy; fresh	Flowing sodium	(TBD)
IRT	Fast reactor U-fissium alloy; fresh	Flowing sodium	Instrumentation response and calibration
J	Fast reactor MOX pins in bundle; medium burnup	Flowing sodium; overpower	Cladding failure and post-failure material interactions and motions
L	Fast reactor MOX; fresh, low, and medium burnup; pin bundle	Flowing sodium; undercooling and undercooling+ overpower	Cladding failure, post-failure material interactions and motions
LO	Fast reactor MOX; fresh and low burnup; pin bundle	Flowing sodium; overpower, undercooling, and undercooling+ overpower	Pre-failure fuel motions; cladding failure threshold; post-failure material interactions and motions
M	Fast reactor U-fissium U-Zr, and U-Pu-Zr; pin bundle	Flowing sodium; overpower	Pre-failure fuel motions; cladding failure threshold; post-failure material interactions and motions
MF	(TBD)	(TBD)	(TBD)
MFT	(TBD)	(TBD)	(TBD)
MWT	Fresh U-Al or U ₃ O ₈ plates	Static water	(TBD)
NRPA	(TBD)	(TBD)	(TBD)
ORNL-FPR	Fresh and irradiated HFIR	Static gas or steam-air;	(TBD)

	UO ₂	overpower	
ORNL-TR	Fresh sol-gel MOX, Sphere-Pac or pellet	Static sodium	Fuel restructuring or melting; no failures
PBF	Fresh UO ₂ -ZrO ₂	Static gas; overpower	(TBD)
PHOTO62	U-Fs or Fermi-A metallic	Static gas; overpower	(TBD)
PINEX	Fast reactor MOX; low burnup;	Static NaK	Pre-failure fuel motion
PNL	(TBD)	(TBD)	(TBD)
R	Fast reactor UO ₂ ; fresh; pin bundle	Flowing sodium; overpower, undercooling, undercooling+overpower	Pre-failure fuel motions; cladding failure threshold; post-failure material interactions and motions
REMET	(TBD)	(TBD)	(TBD)
RFT	Fast reactor MOX; low and medium burnup; single pin	Flowing sodium; overpower	Limited fuel damage; pre-failure fuel motion
RP49	(TBD)	Static gas; overpower	(TBD)
RX	Fast reactor UO ₂ +steel; fresh fuel; monolithic mass	No coolant; heating to simulate melt-down accident transition phase	Melting and gross fuel relocations
S	Fast reactor UO ₂ ; fresh; single pin and pin bundle	Stagnant sodium; high overpower burst	Molten fuel-coolant interaction, including with fuel vapor effects
S(roman)	UO ₂ , UC, U-fissium and other metal alloys; fresh and low burnup; steel and zirconium claddings; bare fuel or single pin	No coolant or stagnant sodium or NaK coolant; overpower	Fuel disruption; cladding failure threshold; post-failure fuel dispersal
SC	UC	Flowing sodium; overpower	Cladding failure threshold; post-failure material motions
STEP	LWR UO ₂ ; medium burnup; pin bundle	Flowing steam; sustained power generation after cladding breach	Fission product release, chemical reactions, transport, and deposition.
Th20U**	Fresh Th-U	Static gas; overpower	(TBD)
TR-1 and TR-2	(see instead ORNL-TR)	(see instead ORNL-TR)	(see instead ORNL-TR)
TR (but not TR-1 or TR-2)	U-fissium; fresh and low burnup; single pin or pin bundle	Flowing sodium or flowing inert gas; overpower and undercooling+overpower	Cladding failure threshold; post-failure material interactions and motions
TS	Fast reactor MOX; medium burnup; single pin	Flowing sodium; overpower	Pre-failure fuel motion; cladding failure threshold; post-failure material interactions and motions
UL	UC; fresh and low burnup; single pin	Overpower	Pre-failure fuel motion; cladding failure threshold; post-failure material interactions and motions
UN/PuN	(TBD)	(TBD)	(TBD)
23nT ¹	LWR UO ₂	Water coolant;	Cladding failure threshold;

		overpower	post-failure material interactions
* TBD indicates that information is yet to be determined. **Prefix Th20U was assigned in the absence of a known title.			

Appendix B: One-page Summaries of CDT-series Tests

One-page summaries of each of the CDT-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test CDT1

Conducted 9 June 1987

Transient No. 2742

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	10-s steady-state flat-top to 33.6 kW/m, 28-s 5¢/s TOP to peak power of 160.9 kW/m, scram
Composition	UO ₂ (Natural) – 24.6% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Solid pellet	Test vehicle & experiment system	Modified Mark IIIC Loop
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Smear density	84.3% of theoretical	Posttest coolant flow condition	No blockage
CLADDING		Posttest condition of test sample(s)	No cladding failure, severe fuel motion/voiding, max. cladding strain of 2.5%
Material	Thermo-mechanical treated HT9 ferritic alloy	Posttest measurements & information	Neutron radiography, ceramography, macro and metallographic examination, gamma scanning, full plenum spring compression
OD	0.686 cm	Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)
Thickness	0.053 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	116 MWd/kg		
Peak linear power	33.6 kW/m		
Peak internal clad temperature	575 °C		
Fluence (spectrum)	1.70 X 10 ²³ n/cm ² ; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
CDE fuel pin w/ HT9 cladding from the FFTF assembly ACO-1, Plenum: (23.6 cm ³ of He + 1 cm ³ of Xe + 1 cm ³ of Kr) at 1 atm, annular insulator pellets			

KEY SUMMARY INFORMATION ABOUT THE TEST
Experiment CDT-1 was conducted to investigate the effects of a LMFBR transient overpower on Core Demonstration Experiment (CDE) type fuel pins. This test provided transient performance data which supported the irradiation of CDE in the Fast Flux Test Facility to its goal burnup of 900 equivalent full power days. In order to provide a test environment which better simulated the characteristics of a LMFBR core, the test train was designed to minimize heat transfer between the fuel/sodium region and the peripheral structure with an insulating argon-filled region, known as the adiabatic region. Unfortunately, a crack in the flowtube allowed sodium to flood into this area, causing an increase in thermal conductivity. This affected the axial and radial temperature profiles, although it was not considered an issue for test validity. In spite of the higher power level and higher measured cladding temperature of CDT-3, the solid pin design of CDT-1 displayed more transient-induced strain than the annular fuel pin of CDT-3. This is a strong indicator that the annular fuel design allows for fuel-region pressure mitigation. Approximately 5.8 cm of fuel motion from the fuel region toward the upper plenum region was observed, providing evidence of an inherent safety feature which could potentially lessen the consequences of an overpower transient. After posttest analysis, it was concluded that the large margin to failure and the high cladding ductility of these pins allowed them to accommodate greater strain than similar pins clad in 20% cold-worked stainless steel. It was also noted that further CDE burnup was supported by the expected reduction in the rate of transient damage associated with the increase of burnup (due to reduced power levels and therefore lower temperatures).

Summary of TREAT Test CDT2

Conducted 11 August 1987

Transient No. 2757

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	10-s steady-state flattop to approx. 35 kW/m, 2-s 1 \$/s TOP to peak power of approx. 662 kW/m, scram
Composition	ACO-1: UO ₂ (natural)– 24.6% PuO ₂ FO-2: UO ₂ (depleted) – 26% PuO ₂	Fuel sample configuration in the test	Three pins in separate flow tubes
Geometry	Pellet (ACO-1 solid; FO-2 solid or annular)	Test vehicle & experiment system	Modified Mark IIIC Loop
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Smear density	ACO-1: 84.3% of theoretical FO-2 Solid: 84.3% of theoretical FO-2 Annular: 80% of theoretical	Posttest coolant flow condition	No blockage
CLADDING		Posttest condition of test sample(s)	Cladding failure and severe fuel movement/voiding (FO-2 solid and annular only), maximum cladding strain of 9.5% (ACO-1 Solid only), full plenum spring compression
Material	Thermo-mechanically treated HT9 ferritic alloy	Posttest measurements & information	Neutron radiography, ceramography, macro & metallographic examination, gamma scanning
OD	0.686 cm	Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)
Thickness	0.053 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	ACO-1 Solid: 116 MWd/kg FO-2 Solid: 64 MWd/kg FO-2 Annular: 65 MWd/kg		
Peak linear power	ACO-1 Solid: 32.9 kW/m FO-2 Solid: 37.3 kW/m FO-2 Annular: 36.0 kW/m		
Peak internal clad temperature	ACO-1: 597 °C FO-2: 549 °C		
Fluence (spectrum)	ACO-1: 1.79 X 10 ²³ n/cm ² FO-2: 0.98 X 10 ²³ n/cm ² ; Fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
CDE fuel pin w/ HT9 cladding from the FFTF assemblies ACO-1 and FO-2, annular insulator pellets			

KEY SUMMARY INFORMATION ABOUT THE TEST

Experiment CDT-2 was conducted to investigate the effects of a LMFBR transient overpower on Core Demonstration Experiment (CDE) type fuel pins. This test provided transient performance data which supported the irradiation of CDE in the Fast Flux Test Facility to its goal burnup of 900 equivalent full power days. Extensive pre-failure axial fuel movement from the high-worth fuel region to the low-worth plenum region was observed in the FO-2 annular pin, providing evidence of an inherent safety feature which could potentially lessen the consequences of an overpower transient. After posttest analysis, it was concluded that the large margin to failure and the high cladding ductility of these pins allowed them to accommodate greater strain than similar pins clad in 20% cold-worked stainless steel. It was also noted that further CDE burnup was supported by the expected reduction in the rate of transient damage associated with the increase of burnup (due to reduced power levels and therefore lower temperatures).

Summary of TREAT Test CDT3

Conducted 31 July 1987

Transient No. 2752

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	10-s steady-state flattop to approx. 35 kW/m, 2-s 1 \$/s TOP to peak power of approx. 662 kW/m, scram
Composition	UO ₂ (natural) – 26.0% PuO ₂	Fuel sample configuration in the test	Three pins in separate flow tubes
Geometry	Annular pellet	Test vehicle & experiment system	Modified Mark IIIC Loop
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Smear density	80.0% of theoretical	Posttest coolant flow condition	No blockage
CLADDING		Posttest condition of test sample(s)	Cladding failure and severe fuel movement/ voiding (FO-2 solid and annular only), maximum cladding strain of 9.5% (ACO-1 solid only), full plenum spring compression
Material	Thermo-mechanically treated HT9 ferritic alloy	Posttest measurements & information	Neutron radiography, ceramography, macro and metallographic examination, gamma scanning
OD	0.686 cm	Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)
Thickness	0.053 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	63 MWd/kg		
Peak linear power	34.6 kW/m		
Peak internal clad temperature	556 °C		
Fluence (spectrum)	0.94 X 10 ²³ n/cm ² ; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
CDE fuel pins w/ HT9 cladding from the FFTF assembly FO-2, annular insulator pellets			

KEY SUMMARY INFORMATION ABOUT THE TEST
Experiment CDT-3 was conducted to investigate the effects of a LMFBR transient overpower on Core Demonstration Experiment (CDE) type fuel pins. This test provided transient performance data which supported the irradiation of CDE in the Fast Flux Test Facility to its goal burnup of 900 equivalent full power days. In order to provide a test environment which better simulated the characteristics of a LMFBR core, the test train was designed to minimize heat transfer between the fuel/sodium region and the peripheral structure with an insulating argon-filled region, known as the adiabatic region. Unfortunately, a crack in the flowtube allowed sodium to flood into this area, causing an increase in thermal conductivity. This affected the axial and radial temperature profiles, although it was not considered an issue for test validity. In spite of the higher power level and higher cladding temperature of CDT-3, the annular fuel pin of CDT-3 displayed less transient-induced strain than the solid pin design of CDT-1. This is a strong indicator that the annular fuel design allows for fuel-region pressure mitigation. The annular fuel and insulator pellets also provided a channel for approx. 10 g of molten fuel to escape into the upper plenum region. This pre-failure axial fuel movement from high-worth to low-worth regions associated with annular fuel pins could potentially be implemented as a passive safety measure. After posttest analysis, it was concluded that the large margin to failure and the high cladding ductility of these pins allowed them to accommodate greater strain than similar pins clad in 20% cold-worked stainless steel. It was also noted that further CDE burnup was supported by the expected reduction in the rate of transient damage associated with the increase of burnup (due to reduced power levels and therefore lower temperatures).

Appendix C: One-page Summaries of CO-series Tests

One-page summaries of each of the CDT-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test CO1

Conducted 5 November 1981

Transient No. 2238

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Helium-bonded MOX	Transient conditions	\$5/s ramp insertion
Composition	UO ₂ (natural U) – 31.5 wt% PuO ₂	Fuel sample configuration in the test	Single full-size fuel pin
Geometry	Annular pellet, flat ends	Test vehicle & experiment system	Single-pin capsule with a massive nickel heat sink at the periphery of the NaK coolant.
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movement (by hodoscope and PINEX camera)
Smear density	80.0% of theoretical	Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
CLADDING		Posttest condition of test sample(s)	Gross disruption
Material	316 stainless steel	Posttest measurements & information	Location of fuel failure, fuel and cladding distribution, neutron radiography, optical microscopy
OD	0.584 cm	Posttest analyses	Thermal-mechanical analyses
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	No pre-irradiation		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
Fuel: 161.7 g; pellets natural UO ₂ (10 cm upper/solid, 46 cm breeder lower/annular); bottom plenum 15.4 cm ³			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test CO1 was performed to determine the cladding breach timing and location during a \$5/s reactivity insertion rate overpower transient. It was a companion test to single-pin tests CO2 and CO3, which were performed in stagnant NaK on medium-burnup and high-burnup fuel, respectively. There was no pre-failure length change of the fuel, within the hodoscope's sensitivity limits (about +/- 6 mm). Post-test radiographs of the test pin revealed that severe pin disruption occurred at the bottom and axial mid-plane of the fuel column. Cladding failure occurred at roughly 24 times the nominal power of 44 kW/m for the type of pin tested, which analysis indicated was mainly by a melt-through. There was apparently a heat transfer problem (perhaps a bubble in the NaK bond) in the capsule near the bottom of the fuel pin that enhanced a melt-through failure at that location. A related test using a seven-pin bundle of fuel pins of this type in a flowing sodium loop was test LO1.

Summary of TREAT Test CO2

Conducted 23 November 1981

Transient No. 2299

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Helium-bonded MOX	Transient conditions	\$5/s reactivity ramp insertion
Composition	UO ₂ (natural U) - 31.5 wt% PuO ₂ Oxygen-to-metal ratio = 1.985	Fuel sample configuration in the test	Single full size fuel pin
Geometry	Annular pellet, flat ends	Test vehicle & experiment system	PFR/TREAT Test Vehicle Assembly (stagnant NaK capsule)
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movements (by hodoscope and PINEX camera)
Smear density	80.0% of theoretical	Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
CLADDING		Posttest condition of test sample(s)	Gross disruption
Material	316 stainless steel	Posttest measurements & information	Neutron radiography, optical microscopy, location of fuel failure, fuel and cladding redistribution,
OD	0.584 cm	Posttest analyses	Thermal-mechanical analyses
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~4 at% of HM		
Peak linear power	30 kW/m equivalent		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel 159.1 g; breeder pellets natural UO ₂ (10 cm upper/solid, 46 cm lower/annular); bottom plenum 15.4 cm ³ at 4.19 atm (cold). As-irradiated fuel: low-power structure, 0.9 mm central hole, total fission gas release 1.61×10^{-3} moles = 29%.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test CO2 was performed to determine the cladding breach timing and location during a \$5/s reactivity insertion rate overpower transient on low burnup (~4 at%, peak) fuel. It was a companion test to single-pin tests CO1 and CO3, which were performed in stagnant NaK on zero-burnup and high-burnup fuel, respectively. There was no pre-failure length change of the fuel, within the hodoscope's sensitivity limits (about +/- 6 mm). Cladding failure which occurred at roughly 24 times nominal power (44 kW/m for pins of this type), apparently was caused by internal pressurization and the elevated cladding temperature at the top of the pin. Some fuel cladding mechanical interaction occurred in the lower part of the pin. A related test using a seven-pin bundle of fuel pins of this type in a flowing sodium loop was test LO2.			

Summary of TREAT Test CO3

Conducted 2 December 1981

Transient No. 2300

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Helium-bonded MOX	Transient conditions	\$5/s reactivity ramp insertion
Composition	73 wt% UO ₂ (23.5% enriched), 27 wt% PuO ₂	Fuel sample configuration in the test	Single full-size fuel pin
Geometry	Annular pellet, flat ends	Test vehicle & experiment system	PFR/TREAT Test Vehicle Assembly (static NaK capsule)
Fissile height	91.44 cm	Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movements (by hodoscope and PINEX camera)
Smear density	81.5% of theoretical	Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
CLADDING		Posttest condition of test sample(s)	Gross disruption
Material	316 stainless steel	Posttest measurements & information	Location of fuel failure, fuel movement and deposition, extent of fuel-cladding mechanical interaction, impact of internal pressurization, neutron radiography, optical microscopy
OD	0.584 cm	Posttest analyses	Thermal-mechanical analyses
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9 at% of HM		
Peak linear power	54.0 kW/m (27.7 kW/m burnup-averaged)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel 166.6 g; breeder pellets natural UO ₂ (10 cm upper/solid, 46 cm lower/annular); bottom plenum 15.4 cm ³ at 4.19 atm (cold). As-irradiated fuel: medium-to-high-power structure			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test CO3 was performed to determine the cladding breach timing and location during a \$5/s ramp insertion rate overpower transient on high burnup (9 at%, peak) fuel. It was a companion test to single-pin tests CO1 and CO2, which were performed in stagnant NaK on zero burnup and low-burnup fuel, respectively. There was no pre-failure length change of the fuel, within the hodoscope's sensitivity limits (about +/- 6 mm). Cladding failure occurred at 25-45 ms before the TREAT power peak, at a fuel elevation about one-fourth of the fuel height (which was attributed to a problem in the experiment. Cladding failure apparently was caused by fuel cladding mechanical interaction (FCMI), but strain measurements were not made. Failure by internal pressurization was delayed relative to CO2 by the lower cladding temperatures. The melt-through possibility was very low for CO3. The low failure location (about 1/4 above the bottom of the fuel column) is also consistent with an FCMI, since no unusual temperature changes were detected in that region, as they were in CO1. The cladding melted near the bottom of the fuel and completely penetrated the knitmesh plug at the fuel bottom.

Summary of TREAT Test CO4

Conducted 15 July 1982

Transient No. 2370

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 120 g/s; 0.47 MPa system pressure, 670 K
Form	Helium-bonded MOX	Transient conditions	Fast rise to ~44 kW/m immediately followed by a 15-s period power increase; scram immediately upon pin failure
Composition	UO ₂ (natural U) - 31.5 wt% PuO ₂ Oxygen-to-metal ratio = 1.985	Fuel sample configuration in the test	Single fuel pin
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark IIIA Mod 1 loop with Single-Pin Test Train
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant temperature, flow rate, fuel movement (by hodoscope)
Smear density	79% of theoretical	Posttest coolant flow condition	Nearly the same as the pre-test flow rate.
CLADDING		Posttest condition of test sample(s)	Cladding intact up to fuel height $x/L \sim 0.98$. Local breach at fuel top. Fuel dispersal from cladding. Unmelted fuel on interior wall of cladding along most of fuel region.
Material	316SS, 20% CW	Posttest measurements & information	Optical microscopy, neutron radiography, photography, x- radiography, autoradiography
OD	0.584 cm	Posttest analyses	Thermal-mechanical analysis of fuel pin
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.2 at% of HM		
Peak linear power	30 kW/m (17.1 kW/m burnup-averaged)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel 159.1 g/pin; breeder pellets natural UO ₂ (10 cm upper/solid, 46 cm lower/annular); bottom plenum 15.4 cm ³ at 4.19 atm (cold). As-irradiated fuel: low-power structure, 0.9 mm central hole, total fission gas release 1.61×10^{-3} moles=29%.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test CO4 was performed to characterize the failure behavior of irradiated, low-power-structure, bottom-plenum fuel pins when subjected to unprotected slow transient overpower (10 cents/s reactivity insertion) conditions beginning at a high power of 44 kW/m. The fuel pin failed at the top of the fuel region at about 123 kW/m axial peak power. About 44% of the fuel was expelled from the pin. Fuel was expelled from regions along most of the original fuel zone, according to the melt radius of the fuel within the cladding. The flowtube was breached during the failure event, which substantially increased the flow area. Post-failure fuel motion was dispersive, resulting in a reactivity worth decrease equivalent to ~30% in a large fast reactor. Cladding strain was measured over most of the fuel region and was about 0.050% at a fuel height of $x/L = 0.87$. The seven-pin bundle counterpart to this test is test LO3.			

Summary of TREAT Test CO5

Conducted 2 December 1982

Transient No. 2405

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 120 g/s; 0.33 MPa system pressure
Form	Helium-bonded MOX	Transient conditions	Fast rise to ~44 kW/m immediately followed by a 15-s period power increase; scram immediately upon pin failure
Composition	73 wt% UO ₂ (23.5% enriched), 27 wt% PuO ₂	Fuel sample configuration in the test	Single fuel pin
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark IIIA Mod 1 Loop with Single-Pin Test Train
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant temperature, flow rate, fuel movement (by hodoscope)
Smear density	81% of theoretical	Posttest coolant flow condition	Fully-blocked flow channel
CLADDING		Posttest condition of test sample(s)	Cladding intact up to fuel height $x/L \sim 0.94$. Localized cladding breach. Fuel dispersal from cladding. Unmelted fuel on interior wall of cladding along most of fuel region.
Material	316SS, 20% CW	Posttest measurements & information	Optical microscopy, neutron radiography, photography, x-radiography, autoradiography
OD	0.584 cm	Posttest analyses	Thermal-mechanical analysis of fuel pin.
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9 at% of HM		
Peak linear power	54 kW/m (27.7 kW/m burnup-averaged)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel 166.6 g; breeder pellets natural UO ₂ (10 cm upper/solid, 46 cm lower/annular); bottom plenum 15.4 cm ³ at 4.19 atm (cold). As-irradiated fuel: medium-to-high-power structure, similar overall gas retention as 4% burnup fuel in test CO4.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test CO5 was performed to characterize the failure behavior of irradiated, medium-to-high-power microstructure, bottom-plenum fuel pins when subjected to unprotected slow TOP (10 cents/s reactivity insertion) conditions beginning at a high power level of 44 kW/m. The fuel expanded axially ~ 11 mm prior to cladding failure. The fuel pin failed at the top of the fuel region at about 131 kW/m axial peak power. About 33% of the fuel was expelled from the pin. Fuel was expelled from regions along most of the original fuel zone, according to the melt radius of the fuel within the cladding. Post-failure fuel motion was dispersive, resulting in a reactivity worth decrease equivalent to ~15% in a large fast reactor. Cladding strain was measured over most of the fuel region and was about 0.37% at a fuel height of $x/L = 0.87$.			

Summary of TREAT Test CO6R

Conducted 20 July 1983

Transient No. 2467

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium, inlet temperature 670 K, 0.184 MPa system pressure
Form	Helium-bonded MOX	Transient conditions	Sub-nominal initial flow rate, several seconds at ~22 kW/m, then two-stage power burst, first increasing power by 3x then burst of ~0.40 s FWHM increasing power by another factor of 18. Peak power ~1300 kW/m.
Composition	UO ₂ (natural U) - 31.5wt% PuO ₂ Oxygen-to-metal ratio = 1.985	Fuel sample configuration in the test	Single pin
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Single-Pin Test Loop
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant temperature, coolant flow rate, coolant pressure, fuel distribution
Smear density	81% of theoretical	Posttest coolant flow condition	(not relevant to the accident scenario due to the large amount of excess energy generated)
CLADDING		Posttest condition of test sample(s)	Gross disruption
Material	M316 stainless steel	Posttest measurements & information	Neutron radiography
OD	0.585 cm	Posttest analyses	Fuel-pin thermal-mechanics, coolant thermal-hydraulics, fuel motion
Thickness	0.077 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.2 at% of HM		
Peak linear power	~30 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel 159.1 g/pin; breeder pellets natural UO ₂ (10 cm upper/solid, 46 cm lower/annular); plenum 15.4 cm ³ at 4.19 atm (cold). As-irradiated fuel: low-power structure, 0.9 mm central hole, total fission gas release 1.61×10^{-3} moles=29%.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
The objective of test CO6R was to determine the time and location of fuel-pin failure during a type of hypothetical transient-under-cooling-driven overpower (TUCOP) accident. Fuel axial expansion (~ 5 mm) and some coolant voiding occurred prior to fuel pin failure. Cladding failed at approximately 960 kW/m at a relative fuel column height of 0.79, as the power was still rising. More than half of the burst energy was generated after the cladding failed, resulting in gross disruption of cladding and fuel. Post-failure dispersive fuel motion was equivalent to a reactivity worth reduction of ~18% in a large fast reactor. The seven-pin counterpart to this experiment was test LO7.			

Appendix D: One-page Summaries of LO-series Tests

One-page summaries of each of the PFR/TREAT CO- and LO-series tests are included here. These summaries can also be found in TREXR [6]

Summary of TREAT Test LO1

Conducted 24 November 1980

Transient No. 2246

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	Steady power for ~4 s (~31 kW/m at midheight of hottest pin), then power burst peaking at ~1000 kW/m in 0.79 s, then scram
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.289 Oxygen-to-metal ratio 1.965	Fuel sample configuration in the test	7-pin, grid-spaced bundle
Geometry	Annular pellet, flat ends	Test vehicle & experiment system	Mark IIIA loop, dysprosium axial shaping collar, B ₄ Si & dysprosium neutron filters
Fissile height	91.4 cm	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion (hodoscope)
Smear density	78.5% of theoretical	Posttest coolant flow condition	Complete upper blockage of once-molten fuel; metal blockage at the bottom of the initial fuel region
CLADDING		Posttest condition of test sample(s)	Severe cladding failure and dispersive fuel motion, complete pin disruption in upper two-thirds of fuel zone
Material	SS316 20% CW	Posttest measurements & information	Macro- and micro examination, neutron radiography, metallographic examination, sibling pin analysis,
OD	0.584 cm	Posttest analyses	Thermal hydraulic, cladding failure, fuel and cladding motion, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	No pre-irradiation		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
U.S.-mfg. PFR-like pins; 163 g/pin fuel mass; Central hole in fuel: 0.165 cm diameter; Annular breeder pellets of natural UO ₂ (9.4 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold)			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test LO1 subjected fresh MOX fuel to a simulation of a \$5/s transient overpower accident. No prefailure axial fuel expansion or fuel motion occurred, within the sensitivity limit (about +/- 6 mm) of the hodoscope. There was severe cladding failure with complete pin disruption in the upper two-thirds of the fuel region. Fuel released into the coolant channel was relocated overall upward, with its dispersal corresponding to a net loss of reactivity worth of about 10% in a large fast reactor. The captor grid in the flow channel above the fuel pins presented a significant impediment to fuel motion as well as a fuel collection site. The posttest material configuration included a complete upper blockage by once-molten fuel and a metal blockage at the lower end of the fuel region. It was concluded that cladding failed by melt-through. In comparison with a companion test (LO2) on pre-irradiated fuel, the failure in LO1 occurred much later, and less fuel disruption resulted. The reactivity loss from fuel disruption was only slightly more than in LO2.			

Summary of TREAT Test LO2

Conducted 5 March 1982

Transient No. 2327

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	Steady power for 4.1 s (~28 kW/m at the midheight of the hottest pin), then power burst peaking at ~1000 kW/m in 0.73 s, then scram
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.985	Fuel sample configuration in the test	7-pin bundle; grid-spaced pins
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark IIIA Loop, dysprosium axial shaping collar, B ₆ Si and dysprosium neutron filters
Fissile height	91.4 cm	Measurements /diagnostics during test	Flow rate, coolant pressure, temperatures, fuel distribution
Smear density	79% of theoretical	Posttest coolant flow condition	Complete upper blockage of once-molten fuel/cladding material
CLADDING		Posttest condition of test sample(s)	Severe cladding failure, net upward dispersive fuel motion, complete pin disruption in upper 3/4 of fuel zone
Material	SS316 20% CW	Posttest measurements & information	Macro and micro examination, neutron radiography, metallographic examination
OD	0.584 cm	Posttest analyses	Thermo-hydraulic analysis, fuel and cladding motion analysis, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.2 at% of HM		
Peak linear power	~ 30 kW/m equivalent		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Prototypic PFR fuel pins; 159.1 g/pin fuel mass ; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold) ; As-irradiated low-power structure, 0.09 cm central hole, 1.61 x 10 ⁻³ moles total fission gas release (29% of gas generated).			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test LO2 subjected fresh MOX fuel to a simulation of a \$5/s transient overpower accident. It was a counterpart to fresh-fuel test LO1, but using moderate-burnup fuel. No prefailure axial fuel expansion or fuel motion occurred, within sensitivity limit (about +/- 6 mm) of the hodoscope. The cladding failed, apparently by a rapid pressurization due to fission gas release from the fuel (much earlier than the fuel pins failed in LO1). There was complete pin disruption in the upper three-fourths of the fuel region, and fuel loss was mainly from the bottom half of the fuel region. As in LO1, the fuel motion was net upward. The degree of cladding failure and extent of fuel disruption in LO2 was greater than in LO1, but the reactivity loss due to fuel dispersal occurred more slowly and was slightly less than the 10% that occurred in LO1. The upper captor grid presented a significant restraint to post-failure motion of molten fuel. By the end of the test, the fuel region has become blocked at the top and bottom by solidified metal.			

Summary of TREAT Test LO3

Conducted 5 May 1982
Transient No. 2342

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	Rapid power rise first to 71 kW/m at midplane of hottest pin; then further increase on 14-s period. Reactor failed to scram at initial pin failure (at 122 kW/m), continuing for 2.5 s longer on 14-s period generating 46% more reactor energy than planned.
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.980	Fuel sample configuration in the test	7-pin hexagonal bundle; grid-spaced pins
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Modified Mark IIIA loop with dual ALIPs, dysprosium shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion (hodoscope)
Smear density	80% of theoretical	Posttest coolant flow condition	Complete blockage promptly upon initial pin failure which then melted as the extra energy was generated. Final blockage at top of sodium pump.
CLADDING		Posttest condition of test sample(s)	Extremely severe cladding and fuel relocation (gross disruption), molten fuel and steel relocation throughout loop components
Material	SS316 20% CW	Posttest measurements & information	Macro and micro examination, neutron radiography, metallographic examination, fuel and cladding distribution
OD	0.584 cm	Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.2 at% of HM		
Peak linear power	~ 30 kW/m equivalent		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Prototypic PFR fuel pins; 159.1 g/pin fuel mass ; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold) ; As-irradiated low-power structure, 0.09 cm central hole, 1.61 x 10 ⁻³ moles total fission gas release (29% of gas generated).			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test LO3 was conducted to investigate the effects of a slow transient overpower accident on UK Prototype Fast Reactor (PFR) pin failure and fuel motion. Cladding failure was caused by the internal cavity pressurization resulting from fuel expansion and fission gas release and occurred at a fuel height x/L = ~0.8. Fuel dispersion soon after cladding failure corresponded to a reactivity loss of about 6% in a large fast reactor. A software error prevented the planned scram from occurring upon initial pin failure and led to continued fuel heating and resulting fuel and cladding movement. Although it was predicted that the oxide fuel would limit dispersal by solidifying to form coolant channel blockages, the severe overheat allowed for the re-melting and movement of fuel. This molten fuel tended to collect within the ALIP sodium pump inlet, causing damage to the experimental hardware.			

Summary of TREAT Test LO4

Conducted 17 February 1983

Transient No. 2422

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	He-bonded MOX
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.985
Geometry	Annular pellets with flat ends
Fissile height	91.4 cm
Smear density	79% of theoretical density

CLADDING	
Material	SS316 20% CW
OD	0.584 cm
Thickness	0.038 cm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	4.12 at% of HM
Peak linear power	~ 30 kW/m equivalent
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; fast spectrum

OTHER FUEL SAMPLE INFORMATION	
Prototypic PFR fuel pins; 159.1 g/pin fuel mass ; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold) ; As-irradiated low-power structure, 0.09 cm central hole, 1.61 x 10 ⁻³ moles total fission gas release (29% of gas generated).	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium, initially at 30% of nominal
Transient conditions	Several seconds at 44 kW/m (average pin at the midplane) with flow reduction by 50% in 5 s inducing boiling and triggering power rise with 0.44 s period for one e-fold, then rise on ~0.18 s period reaching ~920 kW/m in ~0.35 s, then scram. Total energy ~1500 kW/kg in average pin at midplane.
Fuel sample configuration in the test	7-pin hexagonal bundle; grid-spaced pins
Test vehicle & experiment system	Mark-IIIC loop; dysprosium shaping collars
Measurements /diagnostics during test	Pressure transducers, flowmeters, thermocouples, fuel motion (hodoscope)
Posttest coolant flow condition	Total upper blockage of once-molten fuel and steel around grid spacers, lower blockage of once-molten steel
Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and relocation)
Posttest measurements & information	Neutron radiography, macro examination, fuel and cladding distribution
Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test LO4 simulated high-power, lead-subassembly transient-undercooling-driven overpower (TUCOP) conditions by causing sodium voiding prior to the power burst. (The first part of the power burst was to simulate reactivity increase due to coolant voiding in lead channels in a large fast reactor; the second part of the burst was to simulate reactivity from voiding progression into assemblies of intermediate power.) Prior to cladding failure there was significant fuel expansion but no fuel motion. Pin failure occurred at fuel height x/L= 0.66 with fuel escaping into mostly-voided channels when the cladding was weak or molten. This was followed by severe cladding meltthrough, gross fuel and cladding disruption, flowtube meltthrough, and dispersive fuel movement. A total upper blockage of once-molten fuel and steel formed around a grid spacer above the fuel region, and a lower blockage of once-molten steel filled most of the flow channel just below the bottom of the fuel region. The fuel motion corresponded to a net loss of approximately 14% reactivity worth in a large fast reactor.	

Summary of TREAT Test LO5

Conducted 1 April 1983
Transient No. 2432

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; flow rate held at 36% of nominal
Form	He-bonded MOX	Transient conditions	Several seconds at 44 kW/m (average pin at midplane), then power rise with 0.44 s period for one e-fold, then rise on ~0.18 s period reaching ~920 kW/m in ~0.35 s, then scram. Total energy ~1500 kW/kg in average pin at midplane.
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.985	Fuel sample configuration in the test	7-pin hexagonal bundle; grid-spaced pins
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark-IIIC loop, dysprosium shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Smear density	79% of theoretical density	Posttest coolant flow condition	Total blockage of once-molten fuel and steel around upper breeder region, lower blockage of once-molten steel
CLADDING		Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), , flowtube melt-through, pellet stack collapse
Material	SS316 20% CW	Posttest measurements & information	Neutron radiography and tomography, macro examination, fuel and cladding distribution
OD	0.584 cm	Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.12 at% of HM		
Peak linear power	~ 30 kW/m equivalent		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Prototypic PFR fuel pins; 159.1 g/pin fuel mass ; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold) ; As-irradiated low-power structure, 0.09 cm central hole, 1.61 x 10 ⁻³ moles total fission gas release (29% of gas generated).			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test LO5 simulated intermediate-power-subassembly conditions by causing the planned power burst to occur while the sodium coolant was just beginning to boil. (The first part of the power burst was to simulate reactivity increase due to coolant voiding in lead channels in a large fast reactor; the second part of the burst was to simulate reactivity from voiding progression into assemblies of intermediate power.) Prior to cladding failure there was significant fuel expansion but not fuel motion. Weakened cladding failed into two-phase sodium slightly below the fuel midplane, followed by severe cladding failure and dispersive fuel movement. This resulted first in a 7% reduction in reactivity worth but ending in a fuel distribution corresponding to an overall reactivity gain of 4% in a large fast reactor. By the end of the test, a total blockage of once-molten fuel and steel at a grid spacer near the upper breeder region and a partial lower blockage of once-molten steel at bottom of the fuel region had developed.			

Summary of TREAT Test LO6

Conducted 9 September 1983

Transient No. 2486

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium, initially at 30% of nominal
Form	He-bonded MOX	Transient conditions	Several seconds at 44 kW/m (average pin at the midplane) with flow reduction by 50% in 5 s inducing boiling and triggering power rise with 0.44 s period for one e-fold, then rise on ~0.18 s period reaching ~920 kW/m in ~0.35 s, then scram.
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.985	Fuel sample configuration in the test	7-pin hexagonal bundle; grid spaced pins
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark-IIIC loop, dysprosium shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Smear density	79% of theoretical density	Posttest coolant flow condition	Upper blockage of once-molten fuel and steel, complete lower blockage of once-molten steel
CLADDING		Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), flowtube melt-through
Material	SS316 20% CW	Posttest measurements & information	Neutron radiography and tomography, macro examination, fuel and cladding distribution
OD	0.584 cm	Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	No pre-irradiation		
Peak linear power	(not applicable)		
Peak internal clad temp.	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
Prototypic PFR fuel pins; 159.1 g/pin fuel mass; ~1.8 mm central hole in fuel; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold).			

KEY SUMMARY INFORMATION ABOUT THE TEST
LO6 was a fresh-fuel counterpart to LO4, providing a comparison of effects due to fission gas (in irradiated fuel) vs. fuel vapor pressure (in fresh fuel) on fuel dispersal. Lead subassembly conditions were created by causing sodium voiding prior to the planned power burst. (The first part of the power burst was to simulate reactivity increase due to coolant voiding in lead channels in a large fast reactor; the second part of the burst was to simulate reactivity from voiding progression into assemblies of intermediate power.) Prior to cladding failure there was significant fuel expansion but not fuel motion. Cladding failed at fuel height location $x/L=0.72$, followed by severe cladding disruption and dispersive fuel movement. The fresh fuel was less mobile than was the preirradiated fuel in LO4 (and in LO5 and LO7). The final fuel configuration corresponded to a reactivity worth decrease of ~ 13% in a large fast reactor. Blockage of once-molten fuel and steel filling two-thirds of flow channel at top of fuel region; 18-cm-long complete lower blockage of once-molten steel at bottom of original fuel region.

Summary of TREAT Test LO7

Conducted 16 December 1983
Transient No. 2520

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; flow rate held at 43% of nominal
Form	He-bonded MOX	Transient conditions	Several seconds at 44 kW/m (average pin at midplane), then power rise with 0.44 s period for one e-fold, then on ~50 ms period to ~1610 kW/m in ~0.17 s, then scram. Total energy ~1600 kW/kg in average pin at midplane.
Composition	UO ₂ (natural U)-PuO ₂ ; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.985	Fuel sample configuration in the test	7-pin hexagonal bundle; grid-spaced pins
Geometry	Annular pellets with flat ends	Test vehicle & experiment system	Mark-IIIC loop; dysprosium shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Smear density	79% of theoretical	Posttest coolant flow condition	Total blockage of once-molten fuel and steel around spacer grids, lower blockage of once-molten steel
CLADDING		Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), flowtube melt-through, pellet stack collapse
Material	SS316 20% CW	Posttest measurements & information	Neutron radiography and tomography, macro examination, fuel and cladding distribution
OD	0.584 cm	Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.17 at% of HM		
Peak linear power	~30 kW/m equivalent		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Prototypic PFR fuel pins; 159.1 g/pin fuel mass ; Annular breeder pellets of natural UO ₂ (10 cm upper, 46 cm lower); 15.4 cm ³ bottom plenum at 4.19 atm (cold) ; As-irradiated low-power structure, 0.09 cm central hole, 1.61 x 10 ⁻³ moles total fission gas release (29% of gas generated).			
KEY SUMMARY INFORMATION ABOUT THE TEST			
LO7 simulated intermediate subassembly conditions by causing the planned power burst to occur such that pins would fail into single-phase sodium. Prior to cladding failure there was significant fuel expansion but not fuel motion. The faster burst compared to the one in tests LO4-LO6 was to allow investigation of the special case of possible fuel accumulation near the failure site in fast-burst transient-undercooling-driven overpower accidents. Pin failure occurred initially near the midplane when the cladding was still strong, with fuel escaping into a channel that was essentially unvoided at that time and location. The resulting coolant dynamics were much stronger than in LO4 and LO5 and were accompanied by sharp pressure spikes suggestive of molten fuel-coolant interaction. Cladding damage was severe, and dispersive fuel movement corresponded to an overall reactivity worth net loss of approximately 4%. There was a total blockage of once-molten fuel and steel around an upper spacer grid and a nearly-complete blockage of once-molten steel at the bottom of the original fuel region.			

Appendix E: One-page Summaries of M-series Tests

One-page summaries of each of the M-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test M1

Conducted 10 May 1985

Transient No. 2608

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant; inert gas environment
Form	Lateral segments of clad fuel cut from fuel pin	Transient conditions	Rapid power rise to a level at which the test fuel is heated at 70 °C/s, holding at that level for 30 s, then scram.
Composition	U-5Fs	Fuel sample configuration in the test	One pin segment open to 0.255 MPa containment; the other pin segment encased in quartz tube at 2.67 MPa pressure
Geometry	Two fuel segments, separated	Test vehicle & experiment system	F-series type of dry capsule
Fissile height	1.27 cm, each segment	Measurements /diagnostics during test	High-speed photography with laser illumination; thermocouples spot-welded to each sample; pressure transducers; fuel motion by hodoscope
Smear density	~75% of theoretical (17.6 g/cc as-cast)	Posttest coolant flow condition	Not applicable
CLADDING		Posttest condition of test sample(s)	The segment in high pressure became slightly bent. The segment in low pressure lost a piece of fuel and also bent.
Material	316 stainless steel, SA	Posttest measurements & information	Neutron radiography
OD	0.442 cm	Posttest analyses	Not available
Thickness	0.0305 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	3.5 at. %		
Peak linear power	(not available)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Sodium was vacuum-distilled out of the samples before TREAT testing to prevent or limit sodium vapor from obscuring the visual observation of the fuel.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
The purpose of test M1 was to collect preliminary information regarding the transient-induced extrusion behavior of metallic fuels within cladding. One pin segment was held at low pressure to simulate pin failure conditions, while the other was encased in quartz and held at high plenum pressure simulating the plenum pressure in intact fuel pins. During the test, a chunk fell off the low-pressure segment, followed by a plume of smoke that obscured the remainder of the experiment from the camera's view. The smoke was apparently sodium vapor and other volatiles that had been trapped within the fuel. The fuel in both pins remained intact, and little fuel extrusion or swelling occurred.			

Summary of TREAT Test M2

Conducted 21 March 1985

Transient No. 2596

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; same flow rate for each pin
Form	Sodium-bonded metallic fuel	Transient conditions	Power rose to nominal power (39 kW/m), held for 1.5 s, then increased exponentially on an 8-second period, reaching peak power of about 4 times nominal.
Composition	U-5Fs	Fuel sample configuration in the test	3 pins, each within its own separate thin-walled stainless steel flow tube, with a He-Ar gas mixture around the flow tubes
Geometry	Cast pin	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm, as fabricated	Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
Smear density	~75% of theoretical (17.6 g/cc as-cast)	Posttest coolant flow condition	Coolant channel pressurization upon cladding breach caused sharp flow anomaly, including sharp decrease in inlet flow rate. Final posttransient flow rate was about 75% of the initial flow rate.
CLADDING		Posttest condition of test sample(s)	0.3% burnup pin remained intact; other two pins failed with substantial fuel loss from cladding. Cladding failure located at top of fuel region of each failed pin.
Material	316 stainless steel, SA	Posttest measurements & information	Neutron radiograph; detailed posttest metallurgical examinations
OD	0.442 cm	Posttest analyses	Cladding failure analysis; pre- and post-failure fuel motion analysis; thermal-hydraulic analysis
Thickness	0.0305 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Various; see below		
Peak linear power	39.4 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Three pins: peak axial burnups of 0.3 at. %, 4.4 at. %, and 7.9 at. %. Each pin was wire-wrapped, with 0.124 cm diameter wire on a 15.2-cm pitch.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test M2 investigated the effects of fuel motion before and after cladding failure during an overpower transient with power rise rate fast enough to cause cladding failure primarily by rapid cladding penetration by formation of low-melting-point fuel-cladding phases and assisted by plenum pressure. Axial power distribution was 1.10 peak/average. Transient pre-failure axial fuel growth of the low, medium, and high burnup fuel columns was measured to be 16%, 15%, and 3%, respectively. Power was immediately shut down after the first cladding failure, at approximately four times nominal power. The failures occurred at the top of the fuel column in the two higher-burnup pins, after which the fuel (or fuel-cladding melt) was rapidly ejected through the breach and moved upward in the flow channel, leaving empty space in the bottom quarter of each pin's original fuel region. This rapid dispersal of fuel can be explained as having been due to sodium vaporization rather than fission gas pressure. Approximately 50% of the fuel's volume melted before cladding failure occurred. Coolant pressure pulses peaked at about 1.7 MPa.

Summary of TREAT Test M3

Conducted 11 April 1985

Transient No. 2599

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; same flow rate for each pin
Form	Sodium-bonded metallic fuel	Transient conditions	Power rose to nominal power (39 kW/m), held for 1.5 s, then increased exponentially on an 8-second period, reaching peak power of about 3.8 times nominal.
Composition	U-5Fs	Fuel sample configuration in the test	3 pins, each within its own separate thin-walled stainless steel flow tube, with a He-Ar gas mixture around the flow tubes
Geometry	Cast pin	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm, as fabricated	Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
Smear density	~75% of theoretical (17.6 g/cc as-cast)	Posttest coolant flow condition	No significant changes
CLADDING		Posttest condition of test sample(s)	All pins intact. Significant axial fuel elongation occurred in the lowest burnup pin.
Material	316 stainless steel, SA	Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
OD	0.442 cm	Posttest analyses	Cladding failure predictions, pre-failure fuel motion analysis, thermal-hydraulic analysis
Thickness	0.0305 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Various; see below		
Peak linear power	39.4 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Three pins: peak axial burnups of 0.3 at. %, 4.4 at. %, and 7.9 at. %. Each pin was wire-wrapped, with 0.124 cm diameter wire on a 15.2-cm pitch.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test M3 (like Test M2) investigated the effects of fuel motion during an overpower transient with power rise rate fast enough to cause cladding failure primarily by rapid cladding penetration by formation of low-melting-point fuel-cladding phases and assisted by plenum pressure. However, adjustments were made to the coolant flowrates in each channel to cause the lowest-burnup pin to reach failure (or incipient failure) at the time that the higher burnup pins reached incipient failure. Axial power distribution was 1.10 peak/average. The transient was terminated after the power reached 3.8 times nominal. None of the pins failed during the test. Fuel expanded axially by 18% in the low burnup pin and by 4% in the medium and high burnup pins.

Summary of TREAT Test M4

Conducted 13 January 1986
Transient No. 2683

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	Sodium-bonded metallic fuel	Transient conditions	Power rose to nominal power (39 held for 1.5 s, then increased exponentially on an 8-second period, reaching peak power of about 4 times nominal.
Composition	U-5Fs	Fuel sample configuration in the test	3 pins, each within its own separate thin-walled stainless steel flow tube, with a He-Ar gas mixture around the flow tubes
Geometry	Cast pin	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm, as fabricated	Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
Smear density	~75% of theoretical (17.6 g/cc as-cast)	Posttest coolant flow condition	Coolant channel pressurization upon cladding breach caused sharp flow anomaly, including sharp decrease in inlet flow rate. Final posttransient flow rate was approximately 90% of the initial flow rate.
CLADDING		Posttest condition of test sample(s)	The 2.4 at% burnup pin failed, with a small breach near the top of the fuel column. The bottom quarter of the fuel column remained in place. About 1/3 of the rest of the fuel was expelled from the cladding. The other two pins remained intact.
Material	316 stainless steel, SA	Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations.
OD	0.442 cm	Posttest analyses	Cladding failure analysis, pre- and post-failure fuel motion analysis, thermal-hydraulic analysis
Thickness	0.0305 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Various; see below		
Peak linear power	39.4 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Two pre-irradiated pins (burnups 4.4 at% and 2.4 at%) and one fresh (unirradiated) pin were tested.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test M4 focused on three issues pertaining to pre-failure fuel elongation: (1) the role of trapped sodium versus fission gas in very-low-burnup fuel, (2) the amount that would occur in fuel of burnup between the very-low and medium burnup fuels tested in M2 and M3, and (3) additional information to resolve the discrepancy measured in the 4.4 at% burnup fuel tested in M2 and M3. Compared to M2 and M3, a higher flow rate and lower sodium inlet temperature caused the fuel and cladding temperatures at the time of failure to be more prototypic of fast reactor conditions. The fuel melt radius was therefore larger at failure. The transient was quickly terminated when the 2.4 at% burnup pin failed, at about 4.1 times nominal power for that pin. Fuel (or fuel-cladding melt) was ejected from that pin through a small breach near the top of the fuel column and swept upward. The 4.4 at% burnup fuel elongated about the same amount as in M3. The fresh fuel elongated about 3 %, due to local swelling at the top, then slumped to about 90% of the initial height after the transient was terminated. Data from M2, M3, and M4 confirm that significant fuel extrusion does not occur until melting, and fission gas serves as the primary cause of extrusion, although sodium boiling also has an effect at low plenum pressures.			

Summary of TREAT Test M5

Conducted 15 August 1986

Transient No. 2714

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; 0.4 MPa system pressure
Form	Sodium-bonded metallic fuel	Transient conditions	Power increased quickly to nominal power, then increased exponentially on an 8-s period.
Composition	U-19Pu-10Zr	Fuel sample configuration in the test	Two pins, each located in its own 0.038 cm-thick stainless steel flowtube with inert gas between the flowtubes. Dimples in flowtubes centered the pins in the tubes.
Geometry	Cast fuel slug	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm (as-fabricated)	Measurements /diagnostics during test	Coolant inlet and outlet pressures, temperatures, and total flow rates, plus flow rate in each flowtube; temperatures on the outer surfaces of each flow tube; fuel motion by hodoscope
Smear density	72.5 % of theoretical 11.2 g/cc	Posttest coolant flow condition	Coolant boiled at end of test
CLADDING		Posttest condition of test sample(s)	Both fuel pins remained intact
Material	D9 Steel	Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
OD	0.584 cm	Posttest analyses	Cladding failure, pre- and post-failure fuel motion, thermal-hydraulics
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	0.8 at% in one pin; 1.9 at% in the other		
Peak linear power	36 to 40 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
5.0 cc as-fabricated gas plenum volume.			
2 to 3% as-irradiated axial swelling.			
Wire wraps were removed prior to the test.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test M5 investigated the pre-failure axial expansion of IFR zirconium-based fuel, which was not available in time to be used in tests M2, M3, or M4. The test was intended to heat the fuel pins not quite to failure (so as to be able to reuse the loop for a subsequent test). After an initial transient that reached 3.3 times nominal power, the hodoscope data showed no fuel movement occurred. The same power transient was then rerun, but the sodium flow rate was reduced by 22%. Both pins remained intact. The higher flowrate in the second run caused the fuel temperatures and melt radius to be less than prototypic. Peak power-to-flow ratio was 4.4 times nominal in the 0.8% burnup pin and 4.2 times nominal in the 1.9% burnup pin. Transient-induced fuel elongation was 2 to 3%. There was evidence of eutectic fuel-cladding interaction in which a multi-phase alloy was formed, a behavior not previously observed in the U-Fs fuel of tests M2 - M4.

Summary of TREAT Test M6

Conducted 6 February 1987

Transient No. 2734

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; 0.4 MPa system pressure.
Form	Sodium-bonded metallic fuel	Transient conditions	Power was raised first to approximately nominal power, then increased exponentially on an 8-s period until cladding failure was reached, then sharply reduced.
Composition	U-19Pu-10Zr	Fuel sample configuration in the test	Two pins, each located in its own 0.038 cm-thick stainless steel flowtube with inert gas between the flowtubes.
Geometry	Cast fuel slug	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm (as-fabricated)	Measurements /diagnostics during test	Coolant inlet and outlet pressures, temperatures, and total flow rates, plus flow rate in each flowtube; temperatures on the outer surfaces of each flow tube; fuel motion by hodoscope
Smear density	72.5 % of theoretical 11.2 g/cc	Posttest coolant flow condition	Temporary reversal of flow in the flowtube of the 5.3 at% burnup pin.
CLADDING		Posttest condition of test sample(s)	5.3 at.% pin failed, releasing fuel into coolant flowtube.
Material	D9 Steel	Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
OD	0.584 cm	Posttest analyses	Cladding failure analysis, pre- and post-failure fuel motion analyses, thermal-hydraulic analysis
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	1.9 at% in one pin; 5.3 at% in the other		
Peak linear power	36 to 40 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
5.0 cc as-fabricated gas plenum volume. 2 to 3% as-irradiated axial swelling. Wire wraps were removed prior to the test.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test M6 was identical to M5 but used fuel samples with higher levels of pre-irradiation. The 5.3 at.% pin failed at 4.1 times nominal power, at the top of the fuel column; the 1.9 at% burnup pin reached 4.2 times nominal power and remained intact. The failure of the 5.3 at% pin was accompanied by the upward expulsion into the coolant channel of two thirds of the fuel (or fuel-cladding melt) through the localized failure, yet more than half of the original flow rate was maintained in that flowtube. Pre-failure elongation was 4% in both pins, supporting the observation in M5 (relative to M2, M3, and M4) that IFR fuels exhibit less elongation than U-Fs fuel, probably because the higher operating temperature of ternary fuel leads to lower concentrations of retained fission gas.			

Summary of TREAT Test M7

Conducted 15 October 1987

Transient No. 2775

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium; 0.4 MPa system pressure
Form	Sodium-bonded metallic fuel	Transient conditions	Power increased exponentially for 8 s until cladding failure was reached, then sharply reduced.
Composition	U-19Pu-10Zr in one pin; U-10Zr in the other	Fuel sample configuration in the test	Two pins, each located in its own 0.038 cm-thick stainless steel flowtube with inert gas between the flowtubes
Geometry	Cast fuel slug	Test vehicle & experiment system	Mark-IIIC sodium loop
Fissile height	34.3 cm (as-fabricated)	Measurements /diagnostics during test	Coolant inlet and outlet pressures, temperatures, and total flow rates, plus flow rate in each flowtube; temperatures on the outer surfaces of each flow tube; fuel motion by hodoscope
Smear density	72.5 % of theoretical 11.2 g/cc	Posttest coolant flow condition	Temporary reversal of flow, followed by restoration of 0.7 times prefailure flow rate in flowtube of failed pin.
CLADDING		Posttest condition of test sample(s)	U-19Pu-10Zr pin failed, releasing fuel into the coolant flowtube through a small breach near the top of the fuel column. The U-10Zr pin did not fail.
Material	D9 steel for the U-Pu-Zr fuel; HT9 steel for the U-Zr fuel	Posttest measurements & information	Neutron radiography, isotopic gamma scanning, laser profilometry, plenum gas analysis, macro- and microphotography
OD	0.584 cm	Posttest analyses	Cladding failure, pre- and post-failure fuel motion, thermal-hydraulics
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9.8 at% in U-Pu-Zr pin; 2.9 at% in U-Zr pin		
Peak linear power	38 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
5.0 cc as-fabricated gas plenum volume. 2 to 3% as-irrad. axial fuel swelling (UPuZr pin), 8 to 9% as-irrad. axial fuel swelling (UZr pin), 1.09 max/avg axial burnup distribution, Wire wraps were removed prior to the test.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
In test M7, the binary (UZr) fuel sample survived a power of 4.8 times nominal, with up to one-third of the cladding thickness being lost locally near the top of the fuel column due to transient-induced fuel-cladding interaction. The ternary (UPuZr) fuel pin failed at 4.0 times nominal, rapidly ejecting half of its fuel (or molten mixture of fuel and cladding) through a breach in the cladding at the top of the fuel column into the coolant flowtube, where the melt continued moving upwards. As in tests M5 and M6, at peak power about half of the fuel inventory was molten.			

Appendix F: One-page Summaries of TS-series Tests

One-page summaries of each of the TS-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test TS-1

Conducted 8 August 1983

Transient No. 2474

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium, 103-114 g/s
Form	He-bonded MOX	Transient conditions	6-s hold at 240 W/g; then rise on 5 ¢/s TOP (20.85-s exponential period) to cladding failure; then immediate reactor scram
Composition	UO ₂ (Natural) – 22.4 % PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Solid, dished pellet	Test vehicle & experiment system	Single Pin Test Loop (SPTL) – Type B, with dysprosium collars to axially shape the neutron flux reaching the test fuel
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flow sensors, fuel motion (hodoscope)
Smear density	85.6% of theoretical	Posttest coolant flow condition	Incomplete blockage; 90% of initial flow rate, but the flow tube was breached.
CLADDING		Posttest condition of test sample(s)	Cladding breach was at 79 cm above fuel bottom and less than 1 cm long.
Material	20% cold-worked type 316 SS	Posttest measurements & information	Neutron radiography, ceramography, cladding strain measurement
OD	0.584 cm	Posttest analyses	TEMECH thermal-mechanical analysis of fuel, cladding, and flow tube, including cladding failure time and location.
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~2 MWd/kg (0.2% of HM)		
Peak linear power	41 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF driver fuel pins, wire wrapped; 4.8 cm Inconel reflector above the two UO ₂ solid upper insulator pellets; 18.5 cm ³ gas plenum volume filled at 1 atm at room temperature			
KEY SUMMARY INFORMATION ABOUT THE TEST			
This was the first FFTF-type fuel pin to be transient tested in TREAT. The overpower transient closely matched the fuel heating that would occur in FFTF during a 5-cents/s reactivity insertion. The test goal was to determine the time and location of cladding failure in a low-burnup pin and to provide information for comparing with analytical predictions. Some axial fuel movement occurred 5 s before the cladding failed. Failure occurred at a fuel height of 84%, at about 3.1 times the nominal power of 41 kW/m, and resulted in an unusually small cladding breach. Molten fuel moved upward within the fuel pin, pushing the upper insulator pellets and reflector upward enough to fully compress the spring in the pin plenum. Molten fuel that left the cladding penetrated the thin flow tube wall and entered a gas space radially beyond. The relatively-intact condition of the fuel pin after the test allowed detailed examination of the condition of the pin at the time of failure. The cladding was nearly penetrated by cracks near the axial elevation of the cladding breach. The fuel that remained within the pin had a large central hole that had apparently been occupied by molten fuel. The fuel melt radius fraction (r/R) was 0.81 at $X/L = 0.25$, 0.86 at $X/L = 0.47$ and 0.75, and 0.75 at $X/L = 0.88$. Test TS-1 was the low-burnup counterpart of medium-burnup-fuel test TS-2.			

Summary of TREAT Test TS-2

Conducted 11 January 1984
Transient No. 2524

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium, 113 to 103 g/s
Form	He-bonded MOX	Transient conditions	7-s hold at 34 kW/m; then rise on 20.8-s exponential period (5 ϕ /s TOP simulation) to cladding failure; then immediate reactor scram
Composition	UO ₂ (Natural) – 22.4 % PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Solid, dished pellet	Test vehicle & experiment system	Single Pin Test Loop (SPTL) – Type B, with dysprosium collars to axially shape the neutron flux reaching the test fuel
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flow sensors, fuel motion (hodoscope)
Smear density	85.6% of theoretical	Posttest coolant flow condition	10% of the initial flow rate, but the flowtube wall had been extensively breached
CLADDING		Posttest condition of test sample(s)	The fuel pin was destroyed in the fuel region above X/L = 0.6.
Material	20% cold-worked type 316 SS	Posttest measurements & information	Neutron radiography, posttest examination of remains
OD	0.584 cm	Posttest analyses	TEMECH thermal-mechanical analysis of fuel, cladding, and flow tube, including cladding failure time and location.
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~6 MWd/kg		
Peak linear power	41 kW/m BOL, 36 kW/m EOL		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
Wire wrapped; 4.8 cm Inconel reflector above the two UO ₂ upper insulator pellets; 18.5 cm ³ gas plenum volume filled at 1 atm at room temperature			
KEY SUMMARY INFORMATION ABOUT THE TEST			
This was the second FFTF-type fuel pin to be transient tested in TREAT. The overpower transient closely matched the fuel heating that would occur in FFTF during a 5-cents/s reactivity insertion. The test goal was to determine the time and location of cladding failure in a – medium-burnup pin and to provide information for comparing with analytical predictions. About 2.5 cm of axial fuel relocation above the original fuel column (apparently fully compressing the spring in the pin plenum) began about 4 s before the cladding failed. Failure occurred at about 3.4 times the nominal end-of-life power of 36 kW/m and resulted in extensive cladding destruction and multiple breaches of the flow tube in the axial region X/L = 0.6 – 0.9. More than half of the fuel inventory was ejected from the cladding. The radial melt fraction (r/r) of the fuel was 80% at X/L = 0.25 and 87% at X/L = 0.59.			
Test TS-2 was the moderate-burnup-fuel counterpart of low-burnup-fuel test TS-1.			

Appendix G: One-page Summaries of EBT-series Tests

One-page summaries of each of the EBT-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test EBT-1

Performed 25 April 1983

Transient No. 2441

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	He-bonded MOX	Transient conditions	5.6 s preheat at ~40 kW/m, then 50 ¢/s reactivity ramp overpower to 270 kW/m peak
Composition	Pu/(U+Pu) = 0.29; 81% enr. U	Fuel sample configuration in the test	Single pin
Geometry	Solid pellet; 4.95 mm diameter	Test vehicle & experiment system	HUT-series static capsule; with boron thermal neutron filtering; pin NaK-bonded to 7.8 mm-thick nickel circumferential heat sink
Fissile height	34.3 cm	Measurements /diagnostics during test	Thermocouples, fuel motion (hodoscope)
Smear density	90.2% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Massively and extensively disrupted
Material	D9 alloy, 20% cold worked	Posttest measurements & information	Neutron radiography, gas sampling
OD	5.84 mm	Posttest analyses	(unknown)
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9.4% of HM (90 MWd/kg)		
Peak linear power	34.8 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	7.0x10 ²² n/cm ² fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
65 g of fuel in the pin.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test EBT-1 duplicated the same test-fuel thermal conditions as in TREAT Test HUT5-2B (designated as HUT5-1648 in TREXR), which was performed on a 316 SS-clad pin of design similar to the pin in EBT-1 but had only half the burnup. It also duplicated the thermal conditions in test EBT-2, performed on a similar D9-clad pin but of 4.1% burnup. In EBT-1 the cladding breached at 220 kW/m (~ five times normal power) near the top of the fuel column, about 0.5 s before test termination. The major fuel motion occurred within 20 ms duration and involved fuel expansion above the fuel column and also to the sides of the pin. Fuel was also detected escaping from the cladding in the axial region 0 to 7 cm above the midplane. Significant fuel expulsion to above the test fuel region occurred, with significant voiding of fuel along the length of the pin. The power at initial cladding breach was ~7% lower than in HUT5-2B and 10% higher than in EBT-2. The location of initial breach was the same as in HUT5-2B but much higher than the midplane initial breach in EBT-2.

Summary of TREAT Test EBT-2

Performed 19 April 1983

Transient No. 2438

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	He-bonded MOX;	Transient conditions	5.6 s preheat at ~40 kW/m, then 50 ¢/s reactivity ramp overpower to 270 kW/m peak
Composition	Pu/(U+Pu) = 0.29; 81% enr. U	Fuel sample configuration in the test	Single pin
Geometry	Solid pellet; 4.95 mm diameter	Test vehicle & experiment system	HUT-series static capsule; with boron thermal neutron filtering; pin NaK-bonded to 7.8 mm-thick nickel circumferential heat sink
Fissile height	34.3 cm	Measurements /diagnostics during test	Thermocouples, fuel motion (hodoscope)
Smear density	90.2% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Greatly disrupted over most of the fuel column length
Material	D9 alloy, 20% cold worked	Posttest measurements & information	Neutron radiography, gas sampling
OD	5.84 mm	Posttest analyses	(unknown)
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~4.2% of HM		
Peak linear power	35 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	3.1x10 ²² n/cm ² fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
65 g of fuel in the pin.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test EBT-2 duplicated the same test-fuel thermal conditions as in TREAT Test HUT5-2B (designated as HUT5-1648 in TREXR), which was performed on a 316SS-clad pin of design and burnup similar to the pin in EBT-2. It also duplicated the thermal conditions in test EBT-1, performed on a similar D9-clad pin but of 9.4% burnup. In EBT-2 the cladding breached at 200 kW/m (~ five times normal power) near the fuel midplane, about 0.5 s before test termination. The major fuel motion occurred within 10 ms duration near the fuel midplane, accompanied by fuel loss from the top 20% and bottom 20% of the fuel column. About 0.25 s later, fuel suddenly moved from the midplane to lower regions of the fuel pin. The power at initial cladding breach was ~18% lower than in HUT5-2B and 10% lower than in EBT-1. The midplane location of first cladding breach differed significantly from the near-top of fuel locations of initial breach in HUT5-2B and EBT-1.

Summary of TREAT Test EBT-3

Performed 23 February 1984

Transient No. 2543

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	6-s steady state of between 35-38 kW/m, then 10 ϕ /s overpower reaching peak of 91-100 kW/m at 10 s from the start of the overpower, then scram
Composition	1: $\text{UO}_2 - 25.0 \text{ PuO}_2$ 2: $\text{UO}_2 - 29.3 \text{ PuO}_2$ 3: $\text{UO}_2 - 29.3 \text{ PuO}_2$ All $\text{Pu}/(\text{Pu}+\text{U}) = 0.25$	Fuel sample configuration in the test	Three pins, each in a separate flowtube
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIC sodium loop with axial thermal neutron filters
Fissile height	34.3 cm	Measurements /diagnostics during test	Flowmeters, thermocouples, pressure transducers, fuel motion monitor (hodoscope)
Smear density	~86% of theoretical	Posttest coolant flow condition	No blockages
CLADDING		Posttest condition of test sample(s)	No cladding failures, fuel melting in all pins, maximum strain 0.7% (in the HT9-clad pin)
Material	1: 316 SS, 20% cold worked 2: D9 austenitic alloy, 20% CW 3: HT9 ferritic alloy, air quenched, tempered	Posttest measurements & information	Neutron radiography, macro examination, profilometry, ceramography
OD	5.84 mm	Posttest analyses	TMECH and SIEX code analyses
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	82 to 92 MWd/kg		
Peak linear power	1: 38 kW/m 2: 37 kW/m 3: 36 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	~6.3 x 10^{22} n/cm ² (fast spectrum)		
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test EBT-3 was conducted to compare the effects of a hypothetical unprotected LMFBR transient overpower accident on three fuel pins with different types of cladding irradiated to near-nominal exposure goals. Test conditions were established to allow all three pins to fail, but no failures occurred. It is believed that axial extrusion of molten fuel into the solid-insulator-pellet area served as a pressure relief mechanism for the central cavity. The peak power levels reached in the pins (in terms of multiples of their prior steady-state irradiation power) were 2.9 for the HT9-clad pin, 2.6 for the D9-clad pin, and 2.4 for the 316SS-clad pin. The 0.7% local peaking in cladding strain in the HT9-clad pin was just above the top of the fuel and apparently due to movement of molten fuel slightly above the fuel column. The peak strain elsewhere in that pin was 0.3% and in the other two pins was 0.1%. The central void in each pin was filled discontinuously by once-molten fuel, and the maximum areal melt fraction was about 36%.			

Summary of TREAT Test EBTB

Performed 12 March 1984

Transient No. 2550

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	(96% UO ₂ , 4% PuO ₂)
Geometry	Solid dished pellets
Fissile height	34.4 cm
Smear density	95% of theoretical

CLADDING	
Material	316 SS, 20% cold-worked
OD	12.85 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	3%
Peak linear power	(see "Other Fuel Sample Information" below)
Peak internal clad temperature	(not available)
Fluence (spectrum)	1.0x10 ²³ n/cm ² (fast spectrum)

OTHER FUEL SAMPLE INFORMATION	
The test pin had been previously irradiated at steady state, which generated 3% peak cladding strain. It was then subjected to a load-following and slow overpower test in EBR-II.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
Transient conditions	7-s preheat at low power, then 50 ¢/s overpower transient, ended by scram on flow signal ~8 s later
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Mark-IIIC sodium loop, with dysprosium collars to prevent power peaking at the ends of the blanket fuel column
Measurements /diagnostics during test	Thermocouples, flow meters, fuel motion monitor (hodoscope)
Posttest coolant flow condition	Complete blockage
Posttest condition of test sample(s)	Extensive damage, especially over the top half of the blanket fuel region
Posttest measurements & information	Neutron radiography, gamma scanning, destructive examination, macrophotography
Posttest analyses	LIFE-4

KEY SUMMARY INFORMATION ABOUT THE TEST

Test EBTB was performed to determine the ability of MOX blanket fuel rod to withstand an unprotected 50 ¢/s fast reactor overpower transient. The cladding breached when the power level in the pin reached 300 kW/m (600% overpower compared to the pin's steady state irradiation power). Molten fuel expelled from the pin caused a sharp sodium flow anomaly, which triggered a reactor scram as planned. A postfailure scenario most consistent with the data is that the cladding first breached by near the fuel midplane. Molten fuel expelled through the breach was swept upward in the coolant channel to the top of the fuel region where it collected, formed a blockage, and melted through the flowtube wall. The resulting reduced cooling of the above-midplane part of the pin caused gross overheating, melting, and destruction of that part of the fuel column. There was no breach in the flowtube at the midplane. Where pellets remained relatively intact, there were no changes in blanket fuel structure nor in the fuel cracking pattern due to the transient. The initial cladding failure was deduced to be due to fuel-cladding mechanical interaction. The areal melt fraction of the fuel was 26% at x/L=0.13, ~60% at x/L=0.5, and 85% at x/L=0.86. Channel blockage was present along almost the entire upper half of the fuel column.

Appendix H: One-page Summaries of L-series Tests

One-page summaries of each of the L-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test L1

Performed 15 September 1970
Transient Nos. 1136, 1137, and 1138

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂ , with the UO ₂ being 65% enriched
Geometry	Solid pellet
Fissile height	34.3 cm
Smear density	Unknown, but probably 87% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(none)

OTHER FUEL SAMPLE INFORMATION

The fuel pin was spiral-wrapped with a 1 mm diameter spacer wire.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium, set at 2.35 m/s pretest in the first transient, ~3.8 m/s in the second transient; unknown in the third transient
Transient conditions	Three 12-15 s duration power transients, each intended to produce pin power ~15% higher than nominal.
Fuel sample configuration in the test	Single pin geometry, surrounded by a flowtube with 1.2-mm clearance between pin cladding and flowtube wall
Test vehicle & experiment system	Mark-IIA integral sodium loop
Measurements /diagnostics during test	Thermocouples, flowmeters, pressure transducers, and the fast-neutron hodoscope fuel motion detector.
Posttest coolant flow condition	No evidence of permanent flow blockage
Posttest condition of test sample(s)	Fuel pin was highly warped but cladding remained unbreached. Evidence of localized incipient cladding melting and bulging just below the top of the fuel column.
Posttest measurements & information	Neutron radiography, eddy current examination of the cladding, cladding ovality measurement
Posttest analyses	Analyses using SAS1A

KEY SUMMARY INFORMATION ABOUT THE TEST

Test L1 was kept relatively simple because it was the first test to explore the ability of the sodium loop to create loss-of-flow simulations in assemblies of sodium-cooled FFTF-type fuel pins. Thus, the test fuel was not pre-irradiated, and it was planned to approach, but not exceed, the cladding failure threshold of the fuel pin when operating at 15% over design power. Programmed flow coastdown was not available, so successively more-severe power-to-flow mismatch conditions were created by subjecting the fuel pin to three power transients, each producing about 15% overpower in the test pin, but using successively-lower constant sodium flow rate from one transient to the next.

During the first transient, as the TREAT power was rising to the flat-top level, the inlet sodium flow quickly decreased by ~40% (where it remained for the rest of the power transient), the outlet flow rate decreased by ~20%, and the hodoscope detected movement of test-fuel pellets within the cladding. Thereafter, the outlet flow signal rose steadily, rising to ~2.5 times the initial flow signal by the end of the power transient. This unexpected behavior was possibly caused by local sodium boiling where the wire wrap and adjacent thermocouple contacted the cladding, which would have caused much of the flow channel to be voided during the transient. By the end of the transient, the fuel pin had become permanently warped, with the distortion reaching the limit allowed by the restraining wall of the surrounding test section, and all of the instrument signals had returned to their pretransient levels.

The second transient initially overshoot the intended power level by ~60% and then gradually reached the intended level by the end of the transient. When the power peaked, the inlet and outlet flow rates both suddenly dropped by more than half. The inlet flow rate remained at that value until the end of the power transient when it abruptly returned to its initial level. The outlet flow signal stayed at its 50% level for about 2 s, then steadily rose to twice its initial value by the end of the power transient. [No flow signals were recorded during the third transient, and the impact of the transient is unknown.]

Summary of TREAT Test L2

Performed 28 July 1972

Transient No. 1464

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂ , with the UO ₂ being 65% enriched
Geometry	Solid pellet, dished ends
Fissile height	34.3 cm
Smear density	87% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(none)

OTHER FUEL SAMPLE INFORMATION	
The fuel pin was spiral-wrapped with a 1 mm diameter spacer wire. The plenum pressure was 1 atm at room temperature. Above the fuel column were 1.27 cm of UO ₂ insulator pellets with a 15 cm-long Inconel reflector rod above it.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium; programmed reduction in flow rate down to 10% in 9.5 s during power transient
Transient conditions	Essential constant power of ~36 kW/m (pin-bundle average) for ~23 s
Fuel sample configuration in the test	7-pin bundle of wire-wrapped pins inside a fluted flow tube
Test vehicle & experiment system	Mark-II integral sodium loop with axial flux-shaping collars (axial max/avg power = 1.08) test train with argon gas space radially surrounding the flow tube
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, sodium void detection, fuel motion monitoring by the fast neutron hodoscope
Posttest coolant flow condition	Complete flow blockages near bottom and top of test fuel region
Posttest condition of test sample(s)	Highly disrupted; with much fuel and cladding melting, but also with some fuel pellets unmelted or partially melted. Some melted fuel contained steel particles.
Posttest measurements & information	Neutron and X-ray radiography, destructive examination of the remains
Posttest analyses	Analyses using the SAS2B code.

KEY SUMMARY INFORMATION ABOUT THE TEST

In Test L2, a preliminary checkout/calibration heat-balance transient was run with nominal sodium flow rate. A second preliminary transient was run with lower flow rate to check the flow-coastdown capability of the loop and to approach but not exceed the cladding failure threshold of the fuel pin.. A structural anomaly found prior to the final transient was identified posttest distorted the flow distribution among the several fuel pins but apparently did not have had much effect at lower flow rates during the experiment. Flow oscillations at the outlet began toward the end of the flow coastdown, indicating sodium boiling, voiding, and then flow chugging. Temperatures suggesting dryout were reached about 2 s later. The first upward flow eruption was believed to have caused the posttest-observed upward relocation of the plenum section of the fuel elements, upward movement of the reflector rods and top insulator pellets, and compression of the plenum spring. Within 2 s thereafter, the flow channel was completely plugged. At about this time, the upper third of the fuel columns slumped non-uniformly and non-coherently, with fuel initially spilling out of the cladding at $\sim x/L = 0.75$. This was followed by a quiescent period of about 4 s, ending in two large movements of fuel in both directions away from the midplane area. Little or no fuel passed above or below the original fuel region. The lower flow blockage was complete only over a distance of about 1 cm. It was composed mostly of solidified steel surrounding partially intact fuel pellets. The flow blockage at the top of the fuel column was about 4 cm long, porous, and capped a thin layer of steel. Some of the steel apparently had vaporized.

Summary of TREAT Test L3

Performed 2 January 1973

Transient No. 1483

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂ , with the UO ₂ being 62% enriched
Geometry	Solid pellet, dished ends
Fissile height	34.3 cm
Smear density	87.5% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	3.5 at. %
Peak linear power	30 kW/m initially; then 38 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; fast spectrum

OTHER FUEL SAMPLE INFORMATION

Each fuel pin was spiral-wrapped with a 1 mm diameter spacer wire. Plenum pressure was 1 atm at room temperature. Above the fuel column were 1.27 cm of UO₂ insulator pellets with a 15 cm spacer tube above it. The irradiated fuel had no central void and ~51% fission gas retention.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium; programmed reduction in flow rate beginning about 2 s after the start of the constant-power part of the transient, decreasing to ~10% of the initial value within 5 s
Transient conditions	Power rise for short preheat, then power constant at ~30 kW/m for 27 s, then scram
Fuel sample configuration in the test	7-pin bundle of wire-wrapped pins inside a fluted flow tube
Test vehicle & experiment system	Mark-IIA integral sodium loop with axial flux-shaping collars (axial max/avg power = 1.08) test train with argon gas space radially surrounding the flow tube
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the fast neutron hodoscope
Posttest coolant flow condition	Complete blockage of flow channels.
Posttest condition of test sample(s)	Highly disrupted fuel pins, with extensive removal of fuel from the mid-region of the fuel column and upward displacement of fuel into region above the original fuel zone
Posttest measurements & information	Neutron and X-ray radiography, destructive examination of the remains
Posttest analyses	Analyses by SAS2A, THTE

KEY SUMMARY INFORMATION ABOUT THE TEST

Test L3 was the first test of preirradiated FFTF-type fuel in a loss-of-flow simulation. Early in the transient, lateral squirming of the fuel region began and continued for the first half of the constant-power part of the transient. Failure of the fluted flowtube appeared (from thermocouple failures) to occur about half-way through the transient. Fuel motion continued simultaneous with additional melting of the flowtube by molten fuel. There was a marked loss of fuel from the axial center part of the fuel region with upward fuel movement. About 4 s before scram, the central part of the fuel region suddenly voided within 150 ms, with about 10% of the fuel moving upward slightly beyond the original fuel zone. Lesser fuel motions continued until scram. There was no indication of outlet flow chugging as had been observed in Test L2 on fresh fuel. By the end of the test, a blockage consisting of melted steel around the fuel elements was present at the extreme bottom of the original fuel region. Just above that blockage the fuel pellets appeared to have remained unmelted. Additional melted steel was present below the blockage. A partial blockage of molten steel and porous melted fuel had been formed at the top of the original fuel region. Along the original fuel region, most of the fluted flowtube had melted away, and fuel that was present was no longer in pellet form – except at the extreme ends of that region where large masses of melted fuel had accumulated around partially-intact fuel pellets. There was little penetration of molten steel into the fuel. Bulk melted fuel had areal porosity in the 30-40% range. No evidence was found to indicate that the tops of the fuel pins might have moved up or down as a unit.

Summary of TREAT Test L4

Performed 23 March 1973

Transient No. 1503

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂ , with the UO ₂ being 77% enriched
Geometry	Solid pellet
Fissile height	34.3 cm
Smear density	85.0% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	4.3 at. %
Peak linear power	30 kW/m initially; then 38 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; fast spectrum

OTHER FUEL SAMPLE INFORMATION

Each fuel pin was spiral-wrapped with a 1 mm diameter spacer wire. Plenum pressure was 1 atm at room temperature. Above the fuel column were (in series) 17 cm of UO₂ insulator pellets, 13 cm of Inconel reflector, and 29 cm of spacer tube. The irradiated fuel microstructure was characteristic of high-power-irradiated fuel, with central void and ~44% fission gas retention.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium, with 6-s flow coastdown to zero flow beginning at end of preheat
Transient conditions	2-s preheat at 52 kW/m, then constant power at ~33 kW/m (average over the pin cluster) for 23 s, then scram,
Fuel sample configuration in the test	7-pin bundle of wire-wrapped pins inside a fluted flow tube surrounded by an inert gas space
Test vehicle & experiment system	Mark-IIA integral sodium loop with axial flux-shaping collars (axial max/avg power = 1.08) test train with argon gas space radially surrounding the flow tube
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the fast neutron hodoscope
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Gross destruction of most of the fuel region. Accumulations of fuel and steel formed partial blockages near the top and bottom of the original fuel region (but the flowtube had been breached)
Posttest measurements & information	Neutron and X-ray radiography, destructive examination of the remains
Posttest analyses	Analysis with SAS2B.

KEY SUMMARY INFORMATION ABOUT THE TEST

In Test L4, coolant boiling apparently began (per indications of channel voiding) ~5.4 s after the start of the 6-s flow coastdown. Pressure increases about 1.5 s later suggest cladding failure and release of gas from pin(s). A rapid increase in flowtube temperatures near the top of the fuel region ~3.5 s later was interpreted as indicating sodium film dryout in the coolant channel. The hodoscope showed the pins to be very flexible and squirming by this time. About 2 s later, flowtube thermocouples were contacted by materials above the steel melting point, and the hodoscope showed the presence of fuel columns leaning against the loop wall outside the flowtube. During the next 6 s, some fuel moved from above midplane to below midplane, likely in the region outside the flowtube. Two significant pressure-generating events (eructations) occurred in the coolant channel about 2 s and 6 s later, during which fuel moved both upward and downward, and then some moved toward the axial midplane. The top third of the fuel did not slump. Little fuel motion happened during the next 3.5 s until scram occurred. Posttest radiography showed that all of the fuel elements had failed, and masses of fuel (with voids through them) were present at the top and bottoms of the fuel columns, with little fuel in between them. Some fuel had moved to above the original top of the fuel region, extending higher than in test L2 but not as high or as widely distributed as in test L3. At the bottom of the fuel columns, fuel melting was significantly less than had occurred in tests L2 and L3.

Summary of TREAT Test L5

Performed 7 November 1974

Transient No. 1606

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium with an initial rate 40% of FFTF nominal, then coasting down for 4.7 s to 15% of the initial flow rate starting 4 s into the initial preheat phase of the transient.
Form	He-bonded MOX	Transient conditions	~9-s preheat at a constant 36 kW/m. Then a reactivity step was initiated (when cladding melting was predicted to have begun), raising the power on a 0.835 s period and reaching peak power (nearly six times the preheat power) 1.6 s later, then scram.
Composition	UO ₂ -75%PuO ₂ , with the UO ₂ being 30% enriched	Fuel sample configuration in the test	3-pin bundle of wire-wrapped pins inside a 1.47-mm thick fluted stainless steel flow tube surrounded by an inert-gas-filled region
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIC integral sodium loop
Fissile height	86.4 cm	Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the 0.5-m fast neutron hodoscope (over only 60% of the test fuel region)
Smear density	85.0% of theoretical	Posttest coolant flow condition	No sodium flow could be achieved posttest.
CLADDING		Posttest condition of test sample(s)	Gross failure of all pins over nearly the entire length of the fuel region.
Material	SS 316 (20% CW)	Posttest measurements & information	Neutron radiography, gamma-ray scans, macro- and micro-examinations of the fuel remains
OD	5.84 mm	Posttest analyses	Analyses with SAS3A
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	8 at.%		
Peak linear power	40 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence unavailable; thermal spectrum		
OTHER FUEL SAMPLE INFORMATION			
Fuel microstructure approximated moderate-power-structure FFTF fuel. The plenum in each pin was back-filled after irradiation with Xe and He at 1 atm. Above and below the fuel column were 13 mm lengths of UO ₂ insulator pellets. Above the upper insulator was a 7.6 cm long Inconel reflector, then a 23 cm long stainless steel tube.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test L5 was planned to simulate the conditions in moderate-power FFTF subassemblies during a loss-of-flow accident in which fuel slumping in high-power subassemblies causes a power excursion. The actual power burst rose about half as fast and reached only half the peak power than planned, but generated approximately the planned test-fuel energy. There were indications of sodium boiling and flow reversal (voiding) ~0.3 s before the power burst started. Thermocouple failures indicating steel melting occurred after reactor scram. Most of the fuel motion occurred after scram (when the reactor power had dropped so far that the signal to the hodoscope was too low to be useful). Fuel apparently moved from near the fuel midplane and accumulated at fuel axial locations $x/L = 0.11$ to 0.15 and $x/L = 0.59$ to 0.66 . Only close to the top and bottom of the original fuel region was the original geometry maintained. Nearly complete blockages of solidified steel filling the flow channel around the fuel elements formed between 7 and 33 mm above the top of the original fuel column (at the bottom of the reflector rod). A blockage also formed between 8 and 25 mm above the bottom of the fuel region and consisted of steel that had filled almost the entire cross section of the flow tube including most of the area originally occupied by fuel. The cladding tubes were intact below the lower blockage but completely destroyed above the blockage. Just below the upper blockage was an accumulation of many fuel chunks of size 0.5 to 3 mm, mixed with steel. Chunks of fuel were present everywhere above and below the region $x/L = 0.2$ to 0.8 . It appeared that most of the molten cladding moved either upward to form the upper blockage or laterally to mix with melted parts of the flowtube before draining. All disrupted fuel remained between the upper and lower cladding blockages and little was found outside the flowtube.			

Summary of TREAT Test L6

Performed 18 October 1978

Transient No. 2068

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂
Geometry	Solid pellet with dished ends
Fissile height	86.4 cm
Smear density	85.5% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	3 at. %
Peak linear power	41 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence unavailable; thermal spectrum

OTHER FUEL SAMPLE INFORMATION

Fuel microstructure approximated moderate-power-structure FFTF fuel. The plenum in each pin was back-filled after irradiation with Xe and He at 1 atm. Above and below the fuel column were 13 mm lengths of UO₂ insulator pellets. Above the upper insulator was a 7.6 cm long Inconel reflector, then a 23 cm long stainless steel tube.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium with an initial rate 40% of FFTF nominal, then coasting down starting 4 s into the initial preheat phase of the transient.
Transient conditions	Constant power at ~36 kW/m for 9 s; then rapid power increase to 10 times nominal. Scram rod motion began 200 ms after peak power (allowing hodoscope viewing during the expected time of fuel motion).
Fuel sample configuration in the test	3-pin bundle of wire-wrapped pins inside a 1.47-mm thick fluted stainless steel flow tube surrounded by an inert-gas-filled region
Test vehicle & experiment system	Mark-IIIC integral sodium loop
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the 1.2-m fast neutron hodoscope
Posttest coolant flow condition	Posttest coolant flow capability information was unavailable.
Posttest condition of test sample(s)	Gross disruption of the original fuel region.
Posttest measurements & information	Neutron radiography
Posttest analyses	Analyses of fuel redistribution in terms of equivalent worth changes.

KEY SUMMARY INFORMATION ABOUT THE TEST

Test L6 duplicated the sequence of test L5 (flow coastdown, coolant voiding, cladding motion, and then fuel motion) until the initiation of the power increase, and generated the same amount of test-fuel energy up to peak power. When tests L5, L6, and L7 were being planned, it was considered possible to terminate an LOF accident by early fuel dispersal driven mainly by release of entrained fission gases, so a test objective was to avoid generating significant fuel vapor pressures. It was calculated that the maximum fuel temperatures during the test did not exceed 4000 °C, and thus no significant fuel vapor pressure was generated. The change in axial distribution of fuel during the test, as determined by the hodoscope, can be described in terms of the corresponding changes in fuel worth if those fuel motions were to occur in a fast power reactor (using the central subassemblies of CRBR as an example). In L6, the fuel worth declined by ~3% during the 200 ms between peak power and scram. No pressure pulses were observed during the test.

Summary of TREAT Test L7

Performed 10 September 1977
Transient No. 1932

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂
Geometry	Solid pellet with dished ends
Fissile height	86.4 cm
Smear density	85.5% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	3 at %
Peak linear power	41 kW/m ("nominal")
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence unavailable; thermal spectrum

OTHER FUEL SAMPLE INFORMATION	
Fuel microstructure was like moderate-power-structure FFTF fuel. Plenum back-filled after irradiation with Xe and He at 1 atm. Above and below the fuel column were 13 mm lengths of UO ₂ insulator pellets. Above the upper insulator was a 7.6 cm long Inconel reflector, then a 23 cm long stainless steel tube.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium with an initial rate 40% of FFTF nominal, then coasting down starting 4 s into the initial preheat phase of the transient.
Transient conditions	~36 kW/m for 9 s; then rapid power increase to ~23 times nominal. Scram rod motion began after peak power to allow adequate hodoscope signal level during the expected time of fuel motion.
Fuel sample configuration in the test	3-pin bundle of wire-wrapped pins, inside a 1.47-mm-thick 16-mm-OD fluted stainless steel flow tube, surrounded by an inert-gas-filled region, bounded by a 0.9-mm-thick 25-mm-OD SS "inner tube."
Test vehicle & experiment system	Mark-IIIC integral sodium loop
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the 1.2-m fast neutron hodoscope
Posttest coolant flow condition	Information not available
Posttest condition of test sample(s)	Great disruption of the original fuel region.
Posttest measurements & information	Neutron radiography
Posttest analyses	Analyses of fuel redistribution in terms of equivalent worth changes. Analyses using SAS3D (with PRIMAR-3)

KEY SUMMARY INFORMATION ABOUT THE TEST

Test L7 (like Test L6) duplicated the sequence of tests L5 and L6 (flow coastdown, coolant voiding, cladding motion, and then fuel motion) until the initiation of the power increase, and generate the same amount of test-fuel energy up to peak power as in L5. When tests L5, L6, and L7 were being planned, it was considered possible to terminate an LOF accident by early fuel dispersal driven mainly by release of entrained fission gases, so a test objective was to avoid generating significant fuel vapor pressures. It was calculated posttest that the maximum fuel temperature exceeded 4000 °C (above which significant fuel vapor pressure would be generated) beginning about half way between peak power and scram. Instrumentation indicated that sodium boiling started about 0.3 s before the power burst began, and that voiding continued slowly for about 0.6 s. At that point (when the power was 17 times nominal, 0.13 s before peak power), sodium was rapidly ejected from the fuel region, with no pressure pulses being detected. The important fuel movements occurred during the 200-ms period between peak power and scram. At the end of the test, most of the fuel from the bottom 60% of the fuel region had relocated to the top 40% of the fuel region and slightly higher, and much fuel had moved outside the flowtube wall and collected in a large-diameter mass near the top of the fuel region. Only a little fuel moved axially more than ~2 cm above the original top of the fuel region, and little had moved beyond the original bottom of the fuel region. The change in axial distribution of fuel during the test, as determined by the hodoscope, can be described in terms of the corresponding changes in fuel worth if those fuel motions were to occur in a fast power reactor (using the central subassemblies of CRBR as an example). In L7, the fuel worth declined by ~10%, at a nearly-constant rate of ~60¢/s per dollar from the onset of fuel motion to scram. However, at the time the peak fuel temperature was calculated to have reached 4000 °C (calculated), the equivalent fuel worth had decreased by only ~4¢/s per dollar relative to intact geometry.

Summary of TREAT Test L8

Performed 9 May 1978
Transient No. 2007

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	UO ₂ -75%PuO ₂ (U was 30% enr.)
Geometry	Solid pellet with dished ends
Fissile height	86.4 cm
Smear density	85.5% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	3 at. %
Peak linear power	20.3 kW/m ("nominal")
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence unavailable; thermal spectrum

OTHER FUEL SAMPLE INFORMATION

Irradiated fuel was unrestructured, to represent low-power FFTF fuel. Plenum back-filled after irradiation with Xe and He at 1 atm. Above and below the fuel column were 13 mm lengths of UO₂ insulator pellets. Above the upper insulator was a 7.6 cm long Inconel reflector, then a 23 cm long stainless steel tube. About 10% fission gas released from pre-tested fuel.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium with an initial rate of ~ 180 cm ³ /s, then coasting down at ~20 cm ³ /s per second starting 0.5 s before the beginning of the power hold
Transient conditions	2.4-s hold at 3.9 times nominal power, then power burst of 255 ms full-width-at-half-maximum that peaked at 75 times nominal
Fuel sample configuration in the test	3-pin bundle of wire-wrapped pins, inside a 1.47-mm-thick 16-mm-OD fluted stainless steel flow tube, surrounded by an inert-gas-filled region, bounded by a 0.9-mm-thick 25-mm-OD SS "intermediate tube," surrounded by a 0.9-mm-thick 31-mm-OD "outer tube."
Test vehicle & experiment system	Mark-IIICB integral sodium loop
Measurements /diagnostics during test	Temperatures, coolant flow rates, coolant pressures, fuel motion monitoring by the 1.2-m fast neutron hodoscope
Posttest coolant flow condition	Posttest coolant flow capability is unavailable.
Posttest condition of test sample(s)	Gross disruption of the original fuel region.
Posttest measurements & information	Posttest destructive examination of the fuel remains.
Posttest analyses	Analyses of fuel redistribution data in terms of equivalent worth change. Analyses with SAS3D, LEVITATE, and PLUTO2

KEY SUMMARY INFORMATION ABOUT THE TEST

Test L8 achieved its objectives of causing initial failure of a low-power-structure pin to occur at a power level at least 30 times nominal, with failure into almost-boiling flowing sodium. The total test-fuel energy generated was similar to that in tests L5, L6, and L7. Sharp pressure pulses and flow anomalies occurred as the power was rising past 75 times nominal power, indicating cladding failure. Fuel motion that occurred within 0.1 s after cladding failure amounted to an equivalent reactivity worth loss of 15-20% (if described in terms of the corresponding changes in fuel worth if those fuel motions were to occur in a fast power reactor [using the central subassemblies of CRBR as an example]). The worth value did not change during the next 0.15 s as the power level began dropping from its peak. Much of the flowtube melted, except at the top and bottom end of the original fuel region. Melting of the "intermediate tube" occurred mostly above the fuel midplane. The fuel posttest axial distribution was as follows: ~10% was below the original fuel region (some fuel as far as 21 cm below); ~20% within the bottom half of the original region (but practically no fuel at $x/L=0.37-0.45$); ~60% in the upper half of the original fuel region; and ~10% above the original fuel region (some extending as far as 40 cm above). Over the range $x/L=0.57$ to 0.73 , the fuel was limited radially by the "outer tube." From $x/L=0.85$ to 1.00 it was limited radially by the "intermediate tube." Generally, the fuel was porous and brittle. Some fuel that moved downward had to flow within the annular space between the flowtube and the intermediate tube because spongy was fuel blocking the interior of the flowtube. There was no indication that the melted cladding moved upward, and there was no accumulation of melted steel in the upper part of the original fuel region or above. Much of the fuel contained finely-dispersed steel.

Appendix I: One-page Summaries of PINEX-series Tests

One-page summaries of each of the PINEX-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test PINEX-2

Performed 16 January 1978

Transient No. 1986

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	He-bonded MOX	Transient conditions	Initial power flattop, then a power ramp to simulate a 5 \$/sec overpower transient. Power burst full-width at half maximum was ~0.2 s.
Composition	25 wt % PuO ₂ , 75 wt % UO ₂ (natural U)	Fuel sample configuration in the test	Single pin
Geometry	Annular pellets (0.81 mm ID)	Test vehicle & experiment system	GETR/TREAT Mark II type capsule, with axial power shaping such that the top of the fuel received about 40% power compared to the fuel midplane
Fissile height	86.4 cm	Measurements /diagnostics during test	PINEX pinhole imaging system; fuel motion (hodoscope), temperatures near fuel pin
Smear density	86.1% of theoretical density	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	No cladding failure. 10 g of molten fuel ejected upward into reflector region and fission gas plenum. Maximum cladding strain of 1.0%. Fuel melting occurred over relative fuel height 0.15 to 0.70.
Material	316 SS, 20% CW	Posttest measurements & information	Neutron radiography, x-radiography, ceramographic microstructural examination
OD	0.584 cm	Posttest analyses	Code comparison among the ORIGEN, TEMP, LAFM, HOTPIM, FUMO-T, FRAX-2, and PINEX-AR codes.
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	19.5 GWd/MTM (~2 at% of HM)		
Peak linear power	41 kW/m		
Peak internal clad temperature	(unknown)		
Fluence (spectrum)	Fluence is unavailable; thermal spectrum (GETR)		
OTHER FUEL SAMPLE INFORMATION			
Fuel mass = 165 g. Upper UO ₂ insulator column 4.83mm OD, 1.78mm ID, 12.7 mm high. Reflector ID=1.70mm, length 4.93 mm. Initial plenum pressure of 1.6 MPa. The pre-irradiation caused closure of top 15 cm of the central void in the fuel.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
PINEX-2 was conducted to demonstrate the mitigation effects of internal fuel motion in a hypothetical transient overpower accident and to verify the PINEX pinhole diagnostic system concept. An annular design was employed to promote fuel relocation to the plenum. The power transient was designed to provide sufficient fuel melting and fission gas release to make fuel motion likely and to be high enough for fuel detection by the PINEX camera system. Scram was programmed to occur at the time of initial fuel melting, to reflect the possible negative reactivity effect that axial fuel extrusion would have in a power reactor. The pin survived to ~27 times nominal power. The PINEX system capability was demonstrated as intended. The hodoscope detected six fuel-motion events, including three after scram. Posttest examination of the pin showed once-molten fuel filled the passage through the reflector, more than half the available volume in the pin plenum, and the fuel column central void above and below the region of gross melting.			

Summary of TREAT Test PINEX-3

Performed 27 February 1980

Transient No. 2196

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	He-bonded MOX	Transient conditions	Initial low-level steady-state at 40 kW/m for 5.5-s duration followed by a 50¢/s TOP to maximum power of 300 kW/m
Composition	25 wt % PuO ₂ , 75 wt % UO ₂ (natural U)	Fuel sample configuration in the test	Single pin
Geometry	Annular pellets (0.81 mm ID)	Test vehicle & experiment system	Nickel heat sink static capsule in containment vessel, B ₄ C axial flux shaping collars
Fissile height	86.36 cm	Measurements /diagnostics during test	Fuel motion (hodoscope), photography (pinhole imaging system), thermocouples
Smear density	86.1% of theoretical density	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Massive cladding failure, no significant redistribution of fuel to outside of pin, only the top 18% of the fuel and cladding in the initial fuel region remained intact
Material	316 SS, 20% CW	Posttest measurements & information	X-radiography, neutron radiography, destructive examination, microstructure analysis
OD	0.584 cm	Posttest analyses	Analyses with TEMP, LAFM, FUMO-E codes.
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	19.5 GWd/MTM (~2 at% of HM)		
Peak linear power	41 kW/m		
Peak internal clad temperature	(unknown)		
Fluence (spectrum)	Fluence is unavailable; thermal spectrum (GETR)		
OTHER FUEL SAMPLE INFORMATION			
Fuel mass = 165 g. Upper UO ₂ insulator column 4.83mm OD, 1.78mm ID, 12.7 mm high. Reflector ID=1.70mm, length 4.93 mm. Initial plenum pressure of 1.6 MPa. The pre-irradiation caused closure of top 15 cm of the central void in the fuel.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
PINEX-3 was conducted to demonstrate the mitigation effects of internal fuel motion in a hypothetical transient overpower accident and to verify the PINEX pinhole diagnostic system concept. An annular design was employed for the fuel, reflectors, and insulators to promote fuel relocation to the fission gas plenum. The power transient was designed to provide sufficient fuel melting and fission gas release to make fuel motion likely and to be high enough for fuel detection by the PINEX camera system. Scram was programmed to occur at the time of initial fuel melting, to reflect the possible negative reactivity effect that axial fuel extrusion would have in a power reactor. Cladding failure occurred without ejection of large quantities of molten fuel into the fission gas plenum. Fuel relocated from below the midplane to above the midplane, followed by gradual net downward movement of fuel. Posttest examination showed some fuel in the upper reflector region, a small plug of fuel in the lower insulator pellet, and a likely blockage in the axial hole in the upper fuel region. Results from the PINEX system were compared with those from the hodoscope.			

Appendix J: One-page Summaries of RFT-series Tests

One-page summaries of each of the RFT-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test RFT-L1

Performed August 1982

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded MOX
Composition	Pin 1: UO_2 - 22% PuO_2 Pin 2: UO_2 - 22% PuO_2 Pin 3: UO_2 - 27% PuO_2 (Natural UO_2 , O/M = 1.96)
Geometry	Solid pellet
Fissile height	Pin 1: 91.2 cm Pin 2: 91.4 cm Pin 3: 91.8 cm
Smear density	~85% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	Pin 1: 0.21% HM Pin 2: 0.23% HM Pin 3: 0.15% HM
Peak linear power	Pin 1: 41 kW/m Pin 2: 41 kW/m Pin 3: 29 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; fast spectrum

OTHER FUEL SAMPLE INFORMATION

FFTF driver fuel pins; 145 mm long Inconel reflector above upper insulator pellets
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TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
Transient conditions	Short (~6 s) preheat at slightly more than nominal power, then FFTF 50 ϕ /s overpower ramp reaching about 5 times nominal pin power in ~2.4 s, then scram. Pin 1 power was 5% higher than Pin 2 power and 4% lower than Pin 3 power.
Fuel sample configuration in the test	Three pins in separate flow tubes
Test vehicle & experiment system	Mark-IIIA sodium loop, with axial flux-shaping collars
Measurements /diagnostics during test	Thermocouples, flowmeters
Posttest coolant flow condition	(not available)
Posttest condition of test sample(s)	Cladding of only Pin 3 failed (at $x/L = 0.8$).
Posttest measurements & information	Macro-examination, neutron radiography, gamma scanning, fission gas sampling, ceramographic examination
Posttest analyses	Analyses using TEMECH

KEY SUMMARY INFORMATION ABOUT THE TEST

Test RFT-L1 was conducted to investigate the effects of a 50 ϕ /s unprotected overpower transient on very low burnup FFTF driver fuel pins of different irradiation power levels. The test conditions were selected to cause fuel melting and cladding strain but no cladding failure. The failure of Pin 3 may have been due to the fact that its power level was lower than that of the other two pins, and it had been irradiated at a lower temperature than the other pins. A hole was found in the flowtube of Pin 3 near the elevation where Pin 3 failed. All pins had fuel melting and central void formation but no internal fuel motion. The posttest condition of the fuel pins provided data for use in evaluating, improving, and validating computer models for predicting transient fuel pin behavior.

Summary of TREAT Test RFT-L2

Performed 22 June 1983
Transient No. 2457

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	Short (~7 s) preheat at pin powers between 1x and 2x nominal, then 5 ϕ /s overpower transient reaching pin powers of 3.7 times nominal (Pin 1), 2.0 times nominal (Pin 2), and 3.4 times nominal (Pin 3)
Composition	Pin 1: UO_2 - 27% PuO_2 Pin 2: UO_2 - 22% PuO_2 Pin 3: UO_2 - 27% PuO_2 (Natural UO_2 , O/M = 1.96)	Fuel sample configuration in the test	Three pins, each in a separate flowtube
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIa sodium loop, with axial flux-shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters
Smear density	85% of theoretical	Posttest coolant flow condition	Essentially the same as pretest.
CLADDING		Posttest condition of test sample(s)	All pins remained intact.
Material	SS 316 (20% CW)	Posttest measurements & information	Macro-examination, neutron radiography, gamma scanning, fission gas sampling, ceramographic examination
OD	5.84 mm	Posttest analyses	Analyses using TEMECH, SAS4A, FPIN2, and LIFE codes
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Pin 1: 1.5 MWd/kg Pin 2: 2.3 MWd/kg Pin 3: 1.5 MWd/kg		
Peak linear power	Pin 1: ~27 kW/m Pin 2: ~41 kW/m Pin 3: ~26 kW/m		
Peak internal clad temperature	Pin 1: 522 °C Pin 2: 603 °C Pin 3: 516 °C		
Fluence (spectrum)	Fluence not available; fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF driver fuel pins; 145 mm long Inconel reflector above upper insulator pellets			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test RFT-L3 was conducted to investigate the effects of a 50 ϕ /s unprotected overpower transient on very low burnup FFTF driver fuel pins that had been irradiated at different power levels. The test conditions were selected to cause fuel melting and cladding strain but not cladding failure. Peak power reached in the fuel was 3.4-3.7 times nominal in Pins 1 and 3 (the two lower-power-irradiated pins) and 2.0 times nominal in Pin 2 (the higher-power-irradiated pin). The maximum measured strains were 0.56% in Pin 1, none in Pin 2, and 0.14% in Pin 3. All pins had fuel melting and central void formation. The maximum melt radius was ~0.75 r/R in Pins 1 and 3 and ~0.60 r/R in Pin 2, all peaking at a height x/L ~0.6. The posttest condition of the fuel pins provided data for use in evaluating, improving, and validating computer models for predicting transient fuel pin behavior.

Summary of TREAT Test RFT-L3

Performed 30 January 1984

Transient No. 2531

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	~7-s preheat at steady power, then 5 ¢/s overpower ramp for ~13 s reaching peak pin powers of 72-80 kW/m (~1.8 times nominal), then scram
Composition	UO ₂ - 23% PuO ₂ (Natural UO ₂ , O/M = 1.96)	Fuel sample configuration in the test	Three pins, each in a separate flowtube
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIa sodium loop, with axial flux-shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters
Smear density	85% of theoretical	Posttest coolant flow condition	Essentially the same as pretest.
CLADDING		Posttest condition of test sample(s)	No cladding failures.
Material	SS 316 (20% CW)	Posttest measurements & information	Neutron radiography, gamma scanning, profilometry, destruction examination
OD	5.82 mm	Posttest analyses	Analyses with SAS4A and TEMECH
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Pin 1: 2.6% HM Pin 2: 5.2% HM Pin 3: 5.2% HM		
Peak linear power	Pin 1: ~35 kW/m Pin 2: ~36 kW/m Pin 3: ~38 kW/m		
Peak internal clad temperature	Pin 1: 463 °C Pin 2: 485 °C Pin 3: 481 °C		
Fluence (spectrum)	Pin 1: 0.418 x 10 ²³ n/cm ² Pin 2: 0.869 x 10 ²³ n/cm ² Pin 3: 0.870 x 10 ²³ n/cm ² Fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF driver fuel pins; 145 mm long Inconel reflector above upper insulator pellets			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test RFT-L3 was conducted to investigate the effects of a 5 ¢/s unprotected overpower transient on low-to moderate burnup irradiated FFTF driver fuel pins. The test conditions were selected to cause fuel melting and cladding strain but no cladding failure. The fuel reached about 1.8 times nominal power, resulting in fuel melting and central void formation. Measurable cladding strain occurred only in the lower-burnup pin. The posttest condition of the fuel pins provided data for use in evaluating, improving, and validating computer models for predicting transient fuel pin behavior.			

Summary of TREAT Test RFT-L4

Performed 9 February 1984

Transient No. 2535

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	He-bonded MOX	Transient conditions	~6.2-s preheat at steady power, then 1 \$/s overpower ramp for ~125 s reaching peak pin powers of 265-298 kW/m (~6.8 times nominal), then scram
Composition	UO ₂ - 22% PuO ₂	Fuel sample configuration in the test	Three pins in separate flowtubes
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIa sodium loop, with axial flux-shaping collars
Fissile height	91.4 cm	Measurements /diagnostics during test	Thermocouples, flowmeters
Smear density	85% of theoretical	Posttest coolant flow condition	Essentially the same as pretest.
CLADDING		Posttest condition of test sample(s)	No cladding failures.
Material	SS 316 (20% CW)	Posttest measurements & information	Gamma scanning, neutron radiography, profilometry, destructive examinations
OD	5.82 mm	Posttest analyses	Analysis with SAS4A and TEMECH codes.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Pin 1: 2.6% HM Pin 2: 5.4% HM Pin 3: 5.3% HM		
Peak linear power	Pin 1: 35 kW/m Pin 2: 36 kW/m Pin 3: 38 kW/m		
Peak internal clad temperature	Pin 1: 517 °C Pin 2: 520 °C Pin 3: 530 °C		
Fluence (spectrum)	Pin 1: 0.423 x 10 ²³ n/cm ² Pin 2: 0.877 x 10 ²³ n/cm ² Pin 3: 0.872 x 10 ²³ n/cm ² Fast spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF driver fuel pins; 145 mm long Inconel reflector above upper insulator pellets			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test RFT-L4 was conducted to investigate the effects of a 1 \$/s unprotected overpower transient on low to intermediate burnup FFTF driver fuel pins. The test conditions were selected to cause fuel melting and cladding strain but no cladding failure. The fuel pins reached about 6.8 times nominal power and experienced fuel melting, central void formation, and (in the lower burnup pin only) measurable cladding strain. The posttest condition of the fuel pins provided data for use in evaluating, improving, and validating computer models for predicting transient fuel pin behavior.			

Appendix K: One-page Summaries of STEP-series Tests

One-page summaries of each of the STEP-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test STEP1

Performed 19 June 1984

Transient No. 2568

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing steam
Form	He-bonded cm uranium dioxide	Transient conditions	20.6 min. at low pressure with an initial heatup rate of 1.8 K/s
Composition	UO ₂ 8.26 wt% ²³⁵ U in U (as fabricated)	Fuel sample configuration in the test	4-pin bundle
Geometry	Solid pellet stack inside cladding	Test vehicle & experiment system	Single-pass steam system, fission-product sampling system, fission gas and hydrogen collection systems
Fissile height	100 cm	Measurements /diagnostics during test	Pressures, temperature profiles, steam flow rate, hydrogen concentration, fuel motion (hodoscope)
Smear density	9.77 g/cm ³	Posttest coolant flow condition	Fission product build-up caused partial, temporary blockage
CLADDING		Posttest condition of test sample(s)	Severe fuel damage over 90% of the fuel column (in central & upper part)
Material	Zircaloy-4	Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry, chemical composition analysis, particle size classification
OD	0.950 cm	Posttest analyses	SOAP computer code validation, fission product analysis, thermodynamic analysis, thermal-hydraulics, hydrogen generation and release
Thickness	0.063		
PRE-IRRADIATION CHARACTERISTICS			
Burnup	33.1–36.4 GWd/T (axial average)		
Peak linear power	21.9 – 26.7 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; thermal spectrum		
OTHER FUEL SAMPLE INFORMATION			
BR-3 elements from Belgonucleaire			
Fill gas: He at 20 atm			
KEY SUMMARY INFORMATION ABOUT THE TEST			
STEP-1 investigated the release of fission products and the interaction between steam and fission products during the simulation of a PWR large-break loss-of-coolant accident with failure of emergency-core cooling injection. Calculations indicate a peak fuel temperature of 2900 K. Hydrogen generation corresponded to approximately 90% cladding oxidation. Damage was severe throughout the central and top portions of the fuel column. An early, large release of volatile fission products was observed at relatively low temperatures, believed to have been associated with the cladding ballooning that occurred. Fuel motion was overall upward, with some fuel movement toward the midplane during the time of vigorous oxidation. Partial blockage of released material occurred, which was spontaneously relieved. Fission product deposits collected during the transient were composed mainly of cesium, tin, molybdenum, rubidium, and possible silicon. It was found that the chemistry of fission-product deposits was complex and that fuel and cladding particles were released by mechanical processes.			

Summary of TREAT Test STEP2

Performed 20 November 1984

Transient No. 2586

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded uranium dioxide
Composition	UO ₂ 8.26 wt% ²³⁵ U in U (as fabricated)
Geometry	Solid pellet stack inside cladding
Fissile height	100 cm
Smear density	9.77 g/cm ³

CLADDING	
Material	Zircaloy-4
OD	0.950 cm
Thickness	0.063 cm

PRE-IRRADIATION CHARACTERISTICS	
Burnup	30.6 GWd/T (axial average)
Peak linear power	30.1-30.2 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; thermal spectrum

OTHER FUEL SAMPLE INFORMATION	
BR-3 elements from Belgonucleaire	
Fill gas: He at 20 atm	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing steam
Transient conditions	20.5 minutes at low pressure with a fuel heatup rate of 2.1 K/s (slowest achievable in test but intended to simulate an accident having an initial heatup rate of 0.3 K/s)
Fuel sample configuration in the test	4-pin bundle
Test vehicle & experiment system	Single-pass steam system; fission-product sampling system, fission gas and hydrogen collection systems
Measurements /diagnostics during test	Pressures, temperature profiles, steam flow rate, hydrogen concentration, fuel motion (hodoscope)
Posttest coolant flow condition	Fission product build-up caused partial blockage
Posttest condition of test sample(s)	Extensive fuel damage in the upper three-fourths of the fuel column
Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry, chemical composition analysis, and particle size classification
Posttest analyses	SOAP computer code validation, fission product analysis, thermodynamic analysis, thermal-hydraulics, hydrogen generation and release

KEY SUMMARY INFORMATION ABOUT THE TEST

STEP-2 investigated the release of fission products and the interaction between steam and fission products during the simulation of a BWR accident caused by failure of the high-pressure emergency core cooling and long-term decay heat removal systems, with automatic depressurization and temporary low pressure coolant injection assumed to occur. Calculations indicate a peak fuel temperature of 2700 K. Hydrogen generation corresponded to approximately 70% cladding oxidation. Damage was severe throughout the central and top portions of the fuel column. An early and partial blockage of released material occurred which (unlike in STEP-1) was not spontaneously relieved, so the test was terminated before the pressure increase could have caused a facilities problem. A large release of fission products was observed at relatively low temperatures, believed to have been caused by cladding ballooning. Fission product deposits during the transient were composed mainly of cesium, tin, molybdenum, rubidium, and silicon, accompanied by tellurium. Iodine was co-located with high levels of nickel and no cesium. It was found that the chemistry of fission-product deposits was complex and that fuel and cladding particles were released by mechanical processes.

Summary of TREAT Test STEP3

Performed 30 January 1985

Transient No. 2588

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing steam
Form	He-bonded uranium dioxide	Transient conditions	21.4 min. at high pressure with heatup rate of 1.1 K/s
Composition	UO ₂ 8.26 wt% ²³⁵ U in U (as fabricated)	Fuel sample configuration in the test	4-pin bundle
Geometry	Solid pellet stack inside cladding	Test vehicle & experiment system	Single-pass steam system, fission-product sampling system, fission gas and hydrogen collection systems
Fissile height	100 cm	Measurements /diagnostics during test	Pressures, temperature profiles, steam flow rate, hydrogen concentration, fuel motion (hodoscope)
Smear density	9.77 g/cm ³	Posttest coolant flow condition	No blockage
CLADDING		Posttest condition of test sample(s)	Major fission product release; severe fuel damage limited to upper third of fuel column
Material	Zircaloy-4	Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry, chemical composition analysis, and particle size classification
OD	0.950 cm	Posttest analyses	SOAP computer code validation, fission product analysis, thermodynamic analysis, thermal-hydraulics, hydrogen generation and release
Thickness	0.063 cm		
PRE-IRRADIATION CHARACTERISTICS			
Burnup	35.2-36.0 GWd/T (axial average)		
Peak linear power	33.6 – 34.2 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; thermal spectrum		
OTHER FUEL SAMPLE INFORMATION			
BR-3 elements from Belgonucleaire			
Fill gas: He at 20 atm			

KEY SUMMARY INFORMATION ABOUT THE TEST
STEP-3 investigated the release of fission products and the interaction between steam and fission products during the simulation of a PWR station blackout by creating transient conditions typical of the failure of the feedwater system combined with failure to recover station electrical power. Calculations indicate a peak fuel temperature of 2200 K, which was lower than expected. Hydrogen generation corresponded to approximately 40% cladding oxidation. Damage was severe over the upper three-fourths of the fuel column. Fuel movement was overall upward. A smaller release of fission products occurred in this high pressure test than in the low-pressure tests STEP-1 and -2 because convective flow was induced due to the high pressure conditions, resulting in lower fuel temperatures. Cesium, rubidium, and a trace of iodine were the only fission products found in the deposits. However, the test also demonstrated the tendency for component elements of ceramic and stainless steel structures to be volatilized and transported in high pressure steam. Aerosol deposits were found to contain mainly silicon and iron.

Summary of TREAT Test STEP4

Conducted 2 March 1985

Transient No. 2590

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing steam
Form	He-bonded uranium dioxide	Transient conditions	23.8 min. at high pressure (8 MPa) with a fast heatup rate (initially 1.0 K/s)
Composition	UO ₂ 8.26 wt% ²³⁵ U in U (as fabricated)	Fuel sample configuration in the test	4-pin bundle; control rod material present (to simulate PWR control rod)
Geometry	Solid pellet stack inside cladding	Test vehicle & experiment system	Single-pass steam system, fission-product sampling system, fission gas and hydrogen collection systems
Fissile height	100 cm	Measurements /diagnostics during test	Pressures, temperature profiles, steam flow rate, hydrogen concentration, fuel motion (hodoscope)
Smear density	9.77 g/cm ³	Posttest coolant flow condition	No blockage
CLADDING		Posttest condition of test sample(s)	Upper third of the control rod material melted and flowed downward; severe fuel damage was limited to the upper quarter of the fuel column.
Material	Zircaloy-4	Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry.
OD	0.950 cm	Posttest analyses	SOAP computer code validation, fission product analysis, thermodynamic analysis, thermal-hydraulics, hydrogen generation and release
Thickness	0.063 cm		
PRE-IRRADIATION CHARACTERISTICS			
Burnup	34.9-36.0 GWd/T (axial average)		
Peak linear power	33.6 – 34.5 kW/m		
Peak internal clad temp.	(not available)		
Fluence (spectrum)	Fluence not available; thermal spectrum		
OTHER FUEL SAMPLE INFORMATION			
BR-3 elements from Belgonucleaire			
Fill gas: He at 20 atm			

KEY SUMMARY INFORMATION ABOUT THE TEST
STEP-4 investigated the release of fission products and the interaction between steam and fission products during the simulation of a PWR station blackout by creating transient conditions typical of the failure of the feedwater system and failure to recover electrical power. The test was planned to be a duplicate of STEP-3 except that it also contained a simulated silver-indium-cadmium PWR control rod. Calculations indicate a peak fuel temperature below 2200 K. Hydrogen generation corresponded to approximately 30% cladding oxidation. Fuel movement was overall upward. No fission products were found in the deposits. However, the test demonstrated the tendency for component elements of ceramic and stainless steel structures to be volatilized and transported in high pressure steam. Aerosol deposits were found to contain mainly iron with silicon.

Appendix L: One-page Summaries of R-series Tests

One-page summaries of each of the R-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test R-3

Performed 19 October 1973

Transient No. 1526

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	316 °C sodium; flow at 7.3 m/s for 6 s after reaching steady power, then dropping rapidly to a low level, followed by a more-gradual decay until boiling began.
Form	He-bonded oxide	Transient conditions	Steady reactor power initially generating ~31 kW/m average in the test fuel
Composition	UO ₂ , 14% ²³⁵ U in U	Fuel sample configuration in the test	Single fuel pin
Geometry	Solid pellets with dished ends	Test vehicle & experiment system	Gas-driven, once-through loop. Circular stainless steel flowtube (7.61 mm ID, 0.5-mm thick) surrounded by a molybdenum reflector tube (27.4 mm ID, 1.5 mm thick), with a 9.4-mm wide gas-filled annulus in between
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion detector (hodoscope)
Smear density	9.4 g/cm ³	Posttest coolant flow condition	Complete inlet blockage due to accumulation of steel.
CLADDING		Posttest condition of test sample(s)	A spiral-distorted column of bare fuel pellets, which subsequently collapsed.
Material	316 SS 20% CW	Posttest measurements & information	Neutron radiography, macro-examination
OD	0.584 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors. 0.8-mm wire wrap.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test R3 was the first of the R-series tests and served as a proof test for subsequent seven-pin tests. (R1 and R2 were calibration and checkout tests). When local boiling began, the flow velocity had decreased to 1.8 m/s (22% of the initial flow rate) and was decreasing at 12%/s. Inlet flow reversal occurred 1.7 s later. Relative to the time of flow reversal (reference time = 0), the following sequence of events occurred: initial cladding failure at 1.2 s, flowtube failure at 3.6 s, and reactor power termination at 10.5 s. (In locations where the flowtube had disappeared, radiant heat loss from hot fuel pins to the much-cooler Mo tube became large. Significant oscillations of the inlet flow occurred after boiling began, including before inlet flow reversal. No significant pressure events were recorded. The hodoscope indicated that the fuel column moved into a helical shape in two phases, the first one limited radially by the cladding, and the second one limited radially by the Mo reflector. The fuel column was still upright when the reactor power was brought down, but when neutron radiographs were taken later, the fuel column had collapsed. Posttest examination revealed that over the length of the fuel column, almost all of the cladding, wire wrap, and flowtube had melted. A large mass of solidified steel had formed around the bottom insulator pellets of the fuel pin (just below the active fuel column) completely blocking the flow channel. Below the insulator pellets, the flowtube and cladding were essentially undamaged. A small amount of once-melted steel was located around the insulator pellets above the active fuel.			

Summary of TREAT Test R-4

Performed 9 November 1973

Transient No. 1527

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	320 °C sodium; at 7.5 m/s for 3 s after reaching steady power, then gradual decrease to 2.1 m/s during 2.1 s, then gradual decrease to 1.5 m/s during 4 s, then decrease to zero in 0.5 s.
Form	He-bonded oxide	Transient conditions	Steady reactor power initially generating ~31 kW/m average in test-fuel bundle
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Fuel sample configuration in the test	7-pin hexagonal bundle
Geometry	Solid pellets with dished ends	Test vehicle & experiment system	Gas-driven, once-through loop. The hexagonal 0.5-mm thick stainless steel flowtube was surrounded by a molybdenum tube, with an ~2.5 mm intermediate gas gap.
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion detector (hodoscope)
Smear density	9.4 g/cm ³	Posttest coolant flow condition	Complete flow blockages at the inlet and outlet.
CLADDING		Posttest condition of test sample(s)	Gross meltdown of cladding and flowtube.
Material	316 SS 20% CW	Posttest measurements & information	Neutron radiography, macro-examination
OD	0.584 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test R4 focused on coolant and cladding behavior under conditions representative of the central region of a high power-to-flow subassembly in the FTR (FFTF) beginning of life core, providing data to validate the SAS code voiding model. The objective was to allow the test to proceed as far into the destructive stage of the sequence as the TREAT limitation on integrated power would permit, well beyond the start of molten-fuel motion. Sustained channel voiding began when the coolant velocity had fallen to 1.83 m/s and the flow decay rate was 12%/s. Relative to the time of boiling-induced inlet flow reversal (reference time = 0), the following sequence of events occurred: initial cladding failure at 1.13 s, upper coolant-channel blockage complete at 2.5 s, flowtube failure at 3.14 s causing a major flow perturbation due to nitrogen gas entry from outside the flowtube, lower coolant-channel blockage complete at 5.20 s, much fuel moved downward during the next 5 s, and reactor power termination at 10.15 s. No significant pressure events were recorded. Posttest examination revealed that the steel cladding and flowtube was completely gone from the upper two-thirds of the initial test-fuel region. A complete, 0.3-cm-thick steel planar blockage had formed at the upper insulator pellets. At the bottom, a 15-cm-thick steel blockage had formed extending radially to the molybdenum liner and extending downward 5 cm into the region of the lower reflector rods. A large mass of once-molten fuel was located on top of and partially buried within this blockage. Near the fuel midplane, fuel melting was complete, and most of the fuel had drained downward, leaving a thick, irregular crust on the wall of the Mo liner. In the upper part of the original fuel region, fuel melting was not complete, with only center melting of pellets. Although the molten fuel from the pellet centers had drained out, the sintered pellets remained in their stacked configuration in that region. The hodoscope data could not be processed.

Summary of TREAT Test R-5

Performed 6 December 1973

Transient No. 1528

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	320 °C sodium; at 7.5 m/s for 3 s after reaching steady power, then gradual decrease to 2.1 m/s during 2.1 s, then gradual decrease to 1.5 m/s during 4 s, then decrease to zero in 0.5 s.
Form	He-bonded oxide	Transient conditions	Steady reactor power initially generating ~31 kW/m average in test-fuel bundle
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Fuel sample configuration in the test	7-pin hexagonal bundle
Geometry	Solid pellets with dished ends	Test vehicle & experiment system	Gas-driven, once-through loop. The hexagonal 0.5-mm thick stainless steel flowtube was surrounded by a molybdenum tube, with an ~2.5 mm intermediate gas gap.
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion detector (hodoscope)
Smear density	9.4 g/cm ³	Posttest coolant flow condition	Small fraction of initial sodium flow rate possible, corresponding to ~99% plugging (at the inlet).
CLADDING		Posttest condition of test sample(s)	Extensive melting and relocation of cladding. Little or no fuel melting.
Material	316 SS 20% CW	Posttest measurements & information	Neutron radiography, macro-examination
OD	0.584 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test R5 focused on coolant and cladding behavior under conditions representative of the central region of a high power-to-flow subassembly in the FTR (FFTF) beginning of life core, to examine initial cladding motion caused by vapor streaming. The objective was to reproduce the conditions of test R4 but to terminate the test before fuel melting to preserve evidence of early motion of molten cladding. Sustained channel voiding began when the coolant velocity had fallen to 1.83 m/s and the flow decay rate was 12%/s. Relative to the time of boiling-induced inlet flow reversal (reference time = 0), the following sequence of events occurred: initial cladding failure at 1.14 s, upper flow-channel blockage at 2.6 s, flowtube failure at 3.29 s causing a major flow perturbation due to nitrogen gas entry from outside the flowtube, reactor power termination at 4.45 s, and lower flow-channel blockage 6.20 s. No significant pressure events were recorded. Posttest examination revealed that the cladding and flowtube were completely gone from the upper two-thirds of the initial test-fuel region. A thin, incomplete plug had formed in the coolant channel around the upper insulator pellets. An 8-cm-thick steel plug had formed at the bottom of the initial fuel region, which caused a nearly-complete flow blockage. The lower plug contained intact fuel pellet stacks. Above the lower plug the pellet stacks had become considerably jumbled, as also occurred at the top of the initial fuel region, but in between those locations the fused pellet stacks remained intact. There was very little centerline melting among the pellets.			

Summary of TREAT Test R-6

Performed 9 August 1974

Transient No. 1596

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	320 °C sodium; at 7.4 m/s for 3 s after reaching steady power, then linear decrease to 3.3 m/s in 1.5 s, then linear decrease to 1.5 m/s in 6.5 s, then decrease to zero in 0.5 s. (Boiling began at 2 m/s.)
Form	He-bonded oxide		
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Transient conditions	Steady reactor power initially generating ~28 kW/m until scram.
Geometry	Solid pellets with dished ends	Fuel sample configuration in the test	7-pin hexagonal bundle
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Test vehicle & experiment system	Gas-driven, once-through loop. The hexagonal 0.5-mm thick stainless steel flowtube was surrounded by a molybdenum tube, with an ~2.5 mm intermediate gas gap.
Smear density	9.4 g/cm ³	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion detector (hodoscope)
CLADDING		Posttest coolant flow condition	Complete flow blockage at the inlet.
Material	316 SS 20% CW	Posttest condition of test sample(s)	Gross meltdown of cladding and flowtube. Pellet stacks of three pins still standing.
OD	0.584 cm	Posttest measurements & information	Neutron radiography, macro-examination
Thickness	0.038 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test R6 focused on coolant and cladding behavior under conditions representative of the central region of a high power-to-flow subassembly in the FTR (FFTF) beginning of life core, particularly to obtain fuel slumping information for input to planning the subsequent R7 test (which would include a power pulse consistent with reactivity addition due to fuel slumping). Test R6 reproduced test R4 as closely as possible and obtained fuel-motion data using the hodoscope. Relative to the time of boiling-induced inlet flow reversal (reference time = 0), the following sequence of events occurred: initial cladding failure at 0.90 s, flowtube failure at 3.00 s, structural melting beginning at 3.65 s, an inlet blockage forming at 4.35 s, downward fuel motion beginning at 5 s and ending at 7 s, and the reactor scrambling at 8.2 s. No major pressure events were recorded. Posttest examination revealed that three pins retained their column shape, although many of the pellets had become hollow. The hexagonal flowtube had melted totally along the length of the active fuel section. At the bottom of the fuel elements, cladding and spacer wires had completely melted. The steel from the flowtube and cladding, combined with some molybdenum from the wall surrounding the flowtube, had filled spaces around pellet stacks to form a massive blockage about 18 cm high near the original bottom of the fuel region. In the upper blanket region was a steel blockage extending across, but not totally filling, the area inside the flowtube. The top 15% of the fuel region ($x/L = 0.85$ to 1.0) was practically empty. Elsewhere in the original fuel region the fuel was mainly a collection of short stacks of pellets or individual pellets (or pieces of pellets) in a jumble; the pellets were unmelted or partially-melted. The greatest evidence of fuel melting was among the pellets in the region $x/L = 0.4$ to 0.8. No fuel was found above or below the original ends of the fuel columns. Within the once-melted fuel, finely-dispersed metallic particles were generally found.

Summary of TREAT Test R-7

Performed 27 September 1974

Transient No. 1602

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	315 °C sodium; at constant 7.3 m/s during the first 3 s after reaching steady power, then flow coastdown decreasing to 40-50% of full flow within about 2 s.
Form	He-bonded oxide	Transient conditions	Reactor power generating 27 kW/m for 17 s followed by a power pulse having 0.18 s full-width-at-half-maximum, initiated 6.2 s after inlet flow reversal, with peak 15 times the steady-state power
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Fuel sample configuration in the test	7-pin hexagonal bundle
Geometry	Solid pellets with dished ends	Test vehicle & experiment system	Gas-driven, once-through loop. The hexagonal 0.5-mm thick stainless steel flowtube was surrounded by a molybdenum tube, with an ~2.5 mm intermediate gas gap.
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, acoustic sensor, fuel motion detector (hodoscope)
Smear density	9.4 g/cm ³	Posttest coolant flow condition	Complete flow blockage at the inlet.
CLADDING		Posttest condition of test sample(s)	Gross meltdown
Material	316 SS 20% CW	Posttest measurements & information	Neutron radiography, macro-examination
OD	0.584 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test R7 simulated an unprotected loss-of-flow situation in the central channels of a high power-to-flow subassembly in an FTR (FFTF)-like reactor including a power excursion due to reactivity feedback associated with fuel slumping. Conditions for motion of molten core materials driven by fuel and steel vapor pressure within a region plugged at top and bottom by steel blockages were predicted. The power pulse was timed to coincide with the onset of fuel slumping using analytical predictions and results from test R6. The R7 apparatus was nearly identical to that of test R6, but (unlike in the previous R-series tests) the annular gas space outside the flowtube was filled at a reduced pressure so that when the flowtube began melting, the gas pressure would approximately equal the sodium channel pressure. The hodoscope viewed only the axial-center half of the original fuel region ($x/L=0.25$ to 0.75). Relative to the time of boiling-induced inlet flow reversal (reference time = 0), the following sequence of events occurred: initial cladding failure at 1.24 s, upper flow channel blockage at 2.8 s, flowtube failure at 3.72 s, melted structural material beginning to move at 4.8 s, a significant amount of fuel having moved radially beyond the flowtube by 5.5 s, the power spike beginning at 6.2 s, ~40% of the fuel initially in the region $x/L=0.25$ to 0.75 (250 g) moving axially out of that region, an inlet blockage forming at 6.6 s, ~170 g of fuel returning to the region $x/L=0.25$ to 0.75, the reactor scrambling at 7.7 s, and then ~120 g of fuel again leaving the region $x/L=0.25$ to 0.75. No major pressure events were recorded. Posttest examination revealed that claddings, spacer wires, and the flowtube were completely melted from $x/L=0$ to 1.0. Fuel pellets filled the inside of the Mo tube in a disarray from $x/L=0$ to 0.06 and in irregular stacks from $x/L=0.06$ to 0.12. Melted steel (but no melted fuel) completely filled the spaces between these pellets radially outward to the Mo wall. A dense mass of fuel, most of which had previously been melted, filled the region $x/L=0.13$ to 0.25. Above it, from $x/L=0.25$ to 0.5, was a less-dense, sponge-like mass. From $x/L=0.5$ to 0.6 was a dense fuel mass that occupied much of the area within the Mo tube. On top of that was a more-dense fuel accumulation. There was another accumulation of fuel at the top of the original fuel region ($x/L=0.97$-1.05). An incomplete flow blockage was present at the top of the original fuel column region.			

Summary of TREAT Test R-8

Performed 7 May 1976

Transient No. 1780

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	315 °C sodium; at constant 7.15 m/s during the first 3 s after reaching steady power, then flow coastdown.
Form	He-bonded oxide	Transient conditions	Steady reactor power initially producing 31.8 kW/m in the test fuel.
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Fuel sample configuration in the test	7-pin hexagonal bundle
Geometry	Solid pellet w/dished ends	Test vehicle & experiment system	Gas-driven, once-through loop. The hexagonal 0.5-mm thick stainless steel flowtube was surrounded by a molybdenum tube, with an ~2.5 mm intermediate gas gap.
Fissile height	91.44 cm (prototypic of FTR power fuel design)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, acoustic sensor, fuel motion detector (hodoscope)
Smear density	9.4 g/cm ³	Posttest coolant flow condition	Complete flow blockage at the inlet.
CLADDING		Posttest condition of test sample(s)	Gross meltdown (except for a length of partially-intact cladding expelled from the fuel region)
Material	316 SS 20% CW	Posttest measurements & information	Neutron radiography, macro-examination
OD	0.584 cm	Posttest analyses	Thermal-hydraulics, cladding motion and fuel motion (SAS4A)
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
Three pins were pressurized with Xe to nominal end-of-life pressure (1.4 MPa). Four pins had beginning-of-life pressure. 173 g fuel per pin. 14.4 cm upper and lower Inconel reflectors.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test R8 investigated the effect on the loss-of-flow sequence of the release of pressurized non-condensable plenum gas into the coolant channel. Sodium voiding began 9.5 s into the coastdown when the coolant velocity had decreased by 71%. The inlet flow reversed 0.7 s later. Relative to the time of flow reversal (reference time = 0), the following sequence occurred: cladding failures began at 0.71 s (above the midplane, with the three pressurized pins failing within 20 ms), the flowtube breached at 0.78 s (and 2% of the fuel moved above the original fuel region), sodium was expelled from the fuel region from 1.5 to 2.2 s, there was single attempt by the inlet sodium to re-enter the fuel region, an inlet blockage formed at 2.5 s, and downward fuel motion began at 3.8 s and continued until scram at 7.8 s. The downward fuel motion was into the lower third of the original fuel region and radially beyond the flowtube to the surrounding Mo tube. Posttest examination showed there was no steel blockage in the upper insulator and reflector region, although much of the original steel there had melted. There was essentially no fuel above the original fuel region. The top parts of two of the pressurized pins had been driven upward 6 cm and 10 cm, limited by a mechanical restraint. The other pressurized pin had been mechanically severed 9 cm below the top of the original fuel column, and the long cladding segment above the severed location had been driven 74 cm upward, without restraint. The bottom 5 cm of the latter cladding segment showed an axial rip, apparently indicating the location and mode of initial cladding breach. The long cladding segment appeared to have slid off of the top pellets and reflector rod during its ejection. The entire upper quarter of the original fuel region ($x/L = 0.75$ to 1.0) had slumped down, and fuel had significantly penetrated into the region outside of the flowtube. The top 30% of the flowtube along the fuel column ($x/L = 0.7$ to 1.0) had also melted away, some of it apparently moving downward along the inner surface of the molybdenum tube. Practically no fuel remained in the fuel region $x/L = 0.6$ to 1.0. Some fuel remained in the region $x/L = 0.1$ to 0.6, but from $x/L = -0.1$ to 0.1 there was a dense accumulation of fuel on top of steel extending radially to the Mo tube.			

Summary of TREAT Test R-9

Performed 9 April 1976

Transient No. 1776

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 7.6 m/s, 371 °C inlet temperature
Form	He-bonded oxide	Transient conditions	5.2-s flattop preheat at 33 kW/m (bundle axial average), then 50¢/s TOP burst (~1.5-s e-fold period). Reactor scrammed 3.5 s into burst.
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U	Fuel sample configuration in the test	7-pin hexagonal bundle
Geometry	Solid pellet with dished ends	Test vehicle & experiment system	Gas-driven once-through sodium loop. A 0.5-mm-thick hexagonal stainless steel flowtube surrounded the fuel pin bundle.
Fissile height	91.44 cm (prototypic of FFTF/FTR)	Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, acoustic microphone, flux monitor wires, fuel motion detector (hodoscope)
Smear density	Outer pins: 9.40 g/cm ³ Inner pin: 9.39 g/cm ³	Posttest coolant flow condition	No flow was possible.
CLADDING		Posttest condition of test sample(s)	Gross melting and extensive redistribution of fuel and cladding, with about half of the fuel relocated downstream of the initial fuel region. Complete upper flow blockage and possibly complete lower flow blockage.
Material	316 SS 20% CW	Posttest measurements & information	Gamma, X-ray, and neutron radiography; macro-examinations, X-ray spectroscopy
OD	0.584 cm	Posttest analyses	Fuel and cladding motion (SAS4A), thermal-hydraulics
Thickness	0.038 cm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not preirradiated)		
Peak linear power	Not applicable		
Peak internal clad temperature	Not applicable		
Fluence (spectrum)	Not applicable		
OTHER FUEL SAMPLE INFORMATION			
173 g fuel per pin; 14.4 cm long upper and lower Inconel reflectors; 8.67 cm ³ plenum volume, sealed at 1 atm at 24 °C.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test R-9 investigated the coolant, cladding, and fuel dynamics resulting from an FFTF 50 ¢/s transient overpower accident on beginning-of-life full-length fuel and provided data for comparison with SAS computations. Coolant boiling was detected 2.58 s into the overpower ramp, and initial pin failure occurred 0.14 s later, when the bundle-averaged fuel power was 150 kW/m. The initial pin failure did not cause an inlet flow reversal or high-pressure pulses, but there was significant voiding of the test-fuel region, and the coolant flowtube failed (creating a path for materials to move into the annular space outside the flowtube). The inlet flow briefly reversed at 0.05 s, 0.14 s, and 0.67 s after initial pin failure. Fuel motion began 0.07 s after the first reversal, with upward fuel motion into the fuel pin plenum region (which continued during the next 0.45 s). At the second inlet sodium flow reversal, most of the fuel region was voided of sodium, although some sodium may have flowed into the annulus outside the breached flowtube. Beginning 0.09 s before the third flow reversal, much more fuel began moving to and above the top of the original fuel zone and continued to do so for the next 0.40 s. Downward fuel motion began 0.06 s after the third inlet sodium flow reversal, with fuel moving below the original fuel region. The reactor scrammed 1.02 s after initial fuel pin failure, followed 0.5 s later by a significant (but not very energetic) sodium flow event. Inlet coolant flow oscillations of decreasing magnitude occurred as the sodium repeatedly attempted to reenter the hot voided fuel zone. Misalignment of the hodoscope collimator prevented quantitative mass analysis of the fuel-motion data. Posttest examination showed that over 90% of the test fuel had melted, the flowtube had melted over the upper 75% of the original fuel region (and above) and had been redistributed, and about 53% of the fuel had been swept downstream above the active fuel zone. The upper parts of three of the pins had been pushed 23 to 42 cm above their initial locations, and the upper axial reflectors had completely melted. The coolant channel had become totally blocked by a 27-cm-high accumulation of fuel and steel centered at an elevation 47 cm above the original fuel region. Some material had relocated as high as one meter above the original fuel region. The lower reflector region had become considerably blocked by a 4.5-cm high accumulation of fuel and possibly once-molten metal.

Summary of TREAT Test R-12

Performed 23 June 1977

Transient No. 1915

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	He-bonded oxide
Composition	UO ₂ Outer pins: 14% ²³⁵ U in U Inner pin: 20% ²³⁵ U in U
Geometry	Solid pellet with dished ends
Fissile height	91.44 cm (prototypic of FFTF/FTR design)
Smear density	Outer pins: 9.40 g/cm ³ Inner pin: 9.39 g/cm ³

CLADDING	
Material	316 SS 20% CW
OD	0.584 cm
Thickness	0.038 cm

PRE-IRRADIATION CHARACTERISTICS	
BU	(Not pre-irradiated)
Peak linear power	Not applicable
Peak internal clad temperature	Not applicable
Fluence	Not applicable

OTHER FUEL SAMPLE INFORMATION	
173 g fuel per pin; 14.4 cm long upper and lower Inconel reflectors; 8.67 cm ³ plenum, sealed at 1 atm at 24 °C.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium at 7.6 m/s, 371 °C inlet temperature
Transient conditions	5-s preheat at 34 kW/m (bundle axial average), then 50¢/s TOP burst (~1.6-s e-fold period). Reactor scrammed 2.8 s into burst (at 170 kW/m average) due to inlet flowrate drop.
Fuel sample configuration in the test	7-pin hexagonal bundle
Test vehicle & experiment system	Gas-driven once-through sodium loop. A 0.5-mm-thick hexagonal stainless steel flowtube surrounded the fuel pin bundle.
Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, acoustic microphone, flux monitor wires, fuel motion detector (hodoscope)
Posttest coolant flow condition	Same as initial flow rate.
Posttest condition of test sample(s)	Failure of all seven pins. 10% of fuel escaped from cladding, of which half relocated to above the initial fuel region. The flowtube also failed.
Posttest measurements & information	Gamma, X-ray, and neutron radiography; macro-examinations, X-ray spectroscopy
Posttest analyses	Fuel and cladding motion (SAS4A), thermal-hydraulics

KEY SUMMARY INFORMATION ABOUT THE TEST

An objective of test R12 was to investigate post-failure fuel coolability in an FFTF 50 ¢/s transient overpower accident simulation with reactor scram occurring immediately upon initial disruptive pin failure. Test R12 was to be compared to test R9, in which the power transient was identical to the one in R-12 but continued long after pin failure. As in test R9, a steel heat sink adjacent to the bottom of the fuel caused 25% lower power along the bottom third of the test fuel than along the top third. The power burst in R12 was automatically terminated when the inlet flow rate dropped by 60%, which occurred 2.82 s into the burst when the major coolant dynamics event of the test occurred (at a reactor energy within 5% of the energy at initial pin failure in test R9). However, failure of one or more fuel pins had occurred when only two-thirds of the reactor burst energy had been generated, accompanied by minor coolant perturbations, flowtube breach, and 2% of the fuel relocating above the fuel columns. During the event causing scram, expelled fuel moved back into the fuel-column region. After scram, 6% of the original fuel mass moved from the bottom half of the original fuel zone to above the midplane. Full coolant flow was reestablished posttest.

Posttest examination showed that all seven pins failed in a highly-localized region at the top of the active fuel zone, the hexagonal flowtube had been breached at that location, about 5% of the fuel inventory had been swept downstream out of the active fuel zone, and 90% of the fuel remained within the cladding tubes. The breaches of five of the peripheral pins faced the central pin; the breach of the other peripheral pin faced the flowtube breach. The breaches were due mainly to cladding melting; there was no cladding diameter increase anywhere along the fuel columns. Some molten fuel evidently flowed downward within the cladding. By the end of the test, the central pin had lost about 13% of its fuel, and the peripheral pins had each lost 5-20%. The fuel axial distribution was roughly uniform except sharply lower at the fuel top, and the magnitude of the loss varied linearly with the measured fuel peak-axial areal melt fraction, which varied from 55-88%. The plenum springs had compressed 7-22 mm among the test pins due to the transient.

Appendix M: One-page Summaries of E-series Tests

One-page summaries of each of the E-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test E1

Performed 2 July 1969

Transient No. 1281

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium flow at 3.8 m/s (~ 1100 cm ³ /s) and 410 °C
Form	Uranium oxide	Transient conditions	A 0.27 s full-width-at-half-maximum power burst driven by a reactivity step, generating ~1.7 kJ/g in the test fuel, and ~1900 kW/m at peak power
Composition	UO ₂ , 13% enriched U	Fuel sample configuration in the test	Single clad fuel pin surrounded by six helium-filled hollow cladding tubes ("dummy pins"); all cladding wire-wrapped
Geometry	Clad pellet stack	Test vehicle & experiment system	Mark-II integral sodium loop
Fissile height	~305 mm	Measurements /diagnostics during test	Temperature, flow rates, pressures, fuel motion (hodoscope)
Smear density	9.9 g/cm ³ (90% of theoretical)	Posttest coolant flow condition	Same as initial flow rate
CLADDING		Posttest condition of test sample(s)	No cladding failure but extensive axial in-pin fuel relocation.
Material	Type 304L stainless steel, annealed	Posttest measurements & information	X-radiography, neutron radiography, macro-photography, micro-photography, cladding diameter measurements
OD	7.37 cm	Posttest analyses	Thermal-hydraulics, cladding mechanics
Thickness	0.51 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
There was only a stainless-steel coil spring on top of the UO ₂ fuel-pellet stack.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test E1 was the first in a series of meltdown tests on stainless-steel-clad oxide fuel in a flowing-sodium environment allowing coolant expulsion and reentry. The total test-fuel energy generation was chosen to be similar to that in stagnant-sodium TREAT test S-3. Gross in-pin axial fuel motion began late in the transient when the power had dropped to about half its peak and ~1.6 kJ/g had been generated in the test fuel. The cladding remained intact despite experiencing a strain of ~1.1% along the original fuel-column region, the latter being much larger than could have been due to hoop stress caused by gas pressure. Failure was probably prevented by the axial fuel motion. The fuel had moved upward within the cladding, some reaching the top of the plenum, resulting in about a dozen large voids, roughly spaced axially about every two to three cm along nearly the entire length of the pin. The voids, apparently formed by large gas bubbles, were rounded at the top of the void and tapered at the bottom. The maximum areal melt fraction of the fuel was ~80%, with an outer rim remaining unmelted, although some melted fuel penetrated cracks in the rim and contacted the cladding.			

Summary of TREAT Test E2

Performed 4 September 1969

Transient No. 1283

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium flow at 4.0 m/s (~1100 cm ³ /s) and 400 °C
Form	Uranium oxide	Transient conditions	A 0.24 s full-width-at-half-maximum power burst driven by a reactivity step, reaching ~3400 kW/m peak linear power in the test fuel and generating ~2.7 kJ/g.
Composition	UO ₂ , 13% enriched U	Fuel sample configuration in the test	Single clad fuel pin surrounded by six helium-filled hollow cladding tubes ("dummy pins"); all cladding wire-wrapped
Geometry	Clad pellet stack	Test vehicle & experiment system	Mark-II integral sodium loop
Fissile height	~305 mm	Measurements /diagnostics during test	Temperature, flow rates, pressures, fuel motion (hodoscope)
Smear density	9.9 g/cm ³ (90% of theoretical)	Posttest coolant flow condition	Nearly complete flow blockage
CLADDING		Posttest condition of test sample(s)	Cladding failure; fuel extremely fragmented and extensively dispersed
Material	Type 304L stainless steel, annealed	Posttest measurements & information	X-radiography, macro-photography, micro-photography, cladding diameter measurements, electron microprobe analysis
OD	7.37 mm	Posttest analyses	Thermal-hydraulics, cladding mechanics
Thickness	0.51 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
On top of the UO ₂ fuel pellet stack was a depleted 1.3-cm-long UO ₂ fuel pellet; then a 5-cm long, 12-g, stainless steel rod.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test E2 was the first test in TREAT on the meltdown of stainless-steel-clad oxide fuel in a flowing-sodium environment allowing coolant expulsion and reentry. Cladding failure occurred late in the transient when the power had dropped to about half its peak and about 1.8 kJ/g had been generated in the test fuel. Extensive failure of the central pin cladding occurred along the top 4 cm of the fuel stack. To a lesser extent, the cladding also melted through along a spiral path following the downstream side of the spiral spacer wire. The maximum calculated areal fuel melting before failure was 90%. The fuel was extremely fragmented and dispersed, with packed fines nearly completely blocking the flow channels. The insulator pellet and stainless steel rod had moved upward about 8 mm. There was no fuel left in the cladding except for ~6% in the form of a 0.3 mm thick apparently unmelted circumferential shell of fuel against the bottom third of the cladding inner surface. Nearly all of the expelled fuel had become finely divided by the end of the test. The fines had become packed among the cladding tubes. The central pin cladding strain was 1.5% along the original length of the fuel column. There was no evidence of chemical reaction between fuel and cladding. The six dummy tubes were melted through on the sides facing adjacent regions of massive failures of the central pin. Conversion of fuel energy to work on the coolant was ~0.06%.

Summary of TREAT Test E3

Performed 19 June 1970

Transient No. 1322

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 400 °C and 4.4 m/s (366 cm ³ /s; flow cross sectional area = 0.82 cm ²)
Form	Uranium oxide	Transient conditions	Reactivity-step burst transient having 35-ms initial period, 150 ms full-width-at-half-maximum, producing a peak fuel power of 11.4 kW/g and a pin-cluster-averaged fuel energy of 2.2 kJ/g.
Composition	UO ₂ , 13.3% enriched U.	Fuel sample configuration in the test	Three-pin bundle; fluted flow boundary to simulate neighboring pins
Geometry	Pellet stack in cladding	Test vehicle & experiment system	Mark-IIA sodium loop; no thermal neutron filter; sodium annulus outside of the flowtube
Fissile height	145 mm	Measurements /diagnostics during test	Temperatures at various locations; flow rates and pressures at inlet and outlet; fuel motion (hodoscope)
Smear density	~9.2 g/cm ³	Posttest coolant flow condition	Complete flow blockage
CLADDING		Posttest condition of test sample(s)	Highly disrupted
Material	Type 304L stainless steel (annealed)	Posttest measurements & information	Neutron radiography, macro-examinations, microphotography
OD	4.42 mm	Posttest analyses	Thermal-to-work energy conversion, severe-accident fuel-pin computations
Thickness	0.23 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.8, 4.8, and 6.3 at%		
Peak linear power	(not available)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available (thermal spectrum)		
OTHER FUEL SAMPLE INFORMATION			
Fuel O/M = 2.019. Overall pin length = 22.5 cm. Spacer wire of 1.2 mm diameter. Gas release fraction ~23%. Central void radius ~0.25 mm.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test E3 aimed to obtain data on the severity of energy conversion to work resulting from failure of high-gas-content irradiated-oxide LMFBR fuel into coolant-filled channels in a hypothetical transient-overpower (TOP) accident. The data were to help determine whether trapped fission gas would tend to fragment the fuel expelled from the fuel pin and thereby promote an energetic fuel-coolant thermal interaction, or whether the fission gas would blanket the fuel and inhibit fuel-coolant heat transfer. It was the first test to study the consequences of irradiated oxide fuel under TOP conditions. Cladding failure, indicated by sudden coolant channel voiding prior to coolant boiling, occurred 15 ms before peak power, when the highest local energy input in the hottest pin was ~860 J/g. The void growth rate was ~9.5 cm³/ms, leading to complete voiding of the fueled region within about 12 ms from the time of failure. Gross fuel motion, generally a steady upward fuel dispersal, began within ~30 ms after the cladding failure and was essentially complete in about 100 ms. Peak recorded pressures were at most 6 MPa. The fuel was likely at incipient melting, or had just begun to melt, when the cladding failed. The ratio of coolant work done to the total fission energy generated by the time the work was done was about 0.3%. Posttest examination showed that all of the cladding had been destroyed except for at the extreme upper end of the fuel pin. About two-thirds of the total fuel and cladding inventory moved upward, out of the original fuel region, where it formed a flow blockage. Much of the fuel in the blockage was highly fragmented. Some cladding moved downward, but not in sufficient quantity to cause a flow blockage at the inlet. There was little or no interaction between the fuel and the surrounding steel structures in the original fuel-column region. There was some mixing of fuel and steel, but not on a large scale.			

Summary of TREAT Test E4

Performed 17 November 1970

Transient No. 1346

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Fast reactor mixed oxide
Composition	72%UO ₂ -28% PuO ₂ ; 65% enr. U
Geometry	Clad pellet stack
Fissile height	343 mm
Smear density	9.76 g/cm ³ (88.04% of theoretical)
CLADDING	
Material	Type 316 stainless steel
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	(Not pre-irradiated)
Peak linear power	(Not applicable)
Peak internal clad temperature	(Not applicable)
Fluence (spectrum)	(Not applicable)
OTHER FUEL SAMPLE INFORMATION	
1-mm-diameter wire wrap.	
TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium at 5.4 m/s (142 cm ³ /s) and 425 °C
Transient conditions	Power transient produced by a 2.25% $\Delta k/k$ reactivity step, with 66 ms initial period, generating 740 MW-s with full-width-at-half-maximum of 230 ms.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Mark-IIA integral sodium loop, with B ₂ Si filtering. Outward from the 0.8-mm-thick stainless steel flowtube was a 0.55-mm-wide evacuated annulus, then a 0.8 mm-thick stainless steel tube, then a 10-mm-thick sodium annulus.
Measurements /diagnostics during test	Temperatures, inlet and outlet pressures, inlet and outlet flow rates. (Hodoscope data were not available.)
Posttest coolant flow condition	Complete flow blockage at both inlet and outlet
Posttest condition of test sample(s)	Highly disrupted
Posttest measurements & information	Neutron radiography, axial gamma scan, macrophotography, microphotography, particle size analysis
Posttest analyses	Thermal-hydraulic, fuel-pin transient behavior

KEY SUMMARY INFORMATION ABOUT THE TEST
Test E4 subjected a fresh LMFBR fuel pin to a fast transient overpower excursion sufficient to cause pin failure. It was nearly identical to Test H2 which used a slower overpower transient. Upward in-pin fuel motion may have begun as early as 100 ms before cladding failure. Mild sodium boiling existed for 25 to 50 ms before failure. The cladding failed 85 ms after peak power (when ~70% of the transient energy had been generated and 1.6 kJ/g had been generated in the test fuel). A little fuel was promptly released upon failure. Within 10 ms after failure, the flow channel along the entire pin completely voided itself of sodium, and the outlet started to become blocked, apparently by freezing of upward-driven molten fuel. Three more sodium expulsion events occurred at the outlet, and the blockage was complete by ~50 ms after cladding failure. The flowtube likely failed during this time, which would have opened an entrance to the surrounding initially-evacuated annulus and may have caused the observed sudden flow reversal of the initially-expelled sodium slug. A minor secondary event occurred 65 ms after cladding failure, involving redistribution of much fuel and steel. No energetic fuel-coolant interactions occurred. Time-dependent fuel-motion data recorded by the hodoscope were found posttest to have been corrupted by an extraneous signal and rendered unusable. From the posttest examination, it appeared that the major point of failure was within 5 cm of the top of the fuel column. A large fraction of the fuel moved above the original fuel region. The entire cladding had melted along the original fuel region, and some of that steel formed a blockage near the inlet. There was no evidence of massive fuel relocation toward the inlet. The flowtube wall had been penetrated in many locations along the upper half of the original fuel region, and extensively melted in the region extending more than ~10 cm higher. Fuel had been pushed upward within the remaining intact cladding of the fuel pin plenum.

Summary of TREAT Test E6

Performed 18 January 1973

Transient No. 1487

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 400 °C and 615 cm ³ /s
Form	He-bonded MOX	Transient conditions	Preheat at 167 MW TREAT power (54 kW/m center pin, 65 kW/m edge pins) for 3 s, then a power ramp with 183 ms period reaching 2250 MW peak 0.52 s later, then scram. Power burst full-width-at-half-maximum was ~0.28 s.
Composition	75 wt% UO ₂ (77% enriched center pin; 65% enriched edge pins) 25 wt% PuO ₂	Fuel sample configuration in the test	Seven-pin bundle, with one irradiated pin surrounded by six fresh edge pins
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIA integral sodium loop with B ₆ Si filter. Stainless steel flowtube 0.89 mm thick surrounded by a 2.4 mm gas-filled annulus bounded by a 0.89-mm thick stainless steel outer wall.
Fissile height	343 mm	Measurements /diagnostics during test	Temperatures, coolant flow rate and pressure, fuel motion (hodoscope)
Smear density	9.23 g/cm ³ center pin 9.65 g/cm ³ edge pins	Posttest coolant flow condition	Complete flow blockage.
CLADDING		Posttest condition of test sample(s)	Highly disrupted, except near the bottom, with little axial dispersal of material.
Material	316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, axial gamma scanning, macro-examinations, microphotography
OD	5.84 mm	Posttest analyses	Thermal-hydraulics, severe-accident analysis
Thickness	0.39 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.6 at% HM (center pin) No burnup in edge pins.		
Peak linear power	39 kW/m (center pin)		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	4 x 10 ²² n/cm ² (center pin); fast spectrum (EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
FFTF/FTR-like wire-wrapped pins. Lower UO ₂ insulator: 13 mm long. Upper UO ₂ insulator: 13 mm in edge pins, 170 mm in center pin. Upper Inconel reflector above insulator was 152 mm long in edge pins, 127 mm in center pin. Total fuel mass in test = 466 g.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test E6 subjected a bundle of six fresh pins surrounding one high-power-irradiated pin to conditions simulating an LMFBR hypothetical \$3/s transient overpower accident. (The test was the fast-transient counterpart to the 50 c/s TOP test H4.) As the fuel heated during the power burst, the fuel pins bowed, and some fuel (5 to 10 g) may have moved slightly upward within the cladding of the hottest edge pins(s), from fuel axial center. Fuel-pin failure occurred 550 ms after the start of the overpower ramp (~15 ms after peak power), causing sudden reversal of inlet flow and sharp acceleration of outlet flow. Two more similar flow events followed: one at ~45 ms and one at ~90 ms after peak power (at TREAT power ~1900 MW and ~1100 MW, respectively), suggesting additional pin failures. Fuel dispersal, and rupture of the flowtube at a fuel relative height x/L= 0.65 to 0.8, were associated with the last of these events. Posttest examinations showed cladding melting as low as x/L=0.15 in the edge pins and as low as x/L=0.4 in the central pin. There was little fuel/steel debris in the flow channels in the lower third of the original fuel region. Considerable fuel/steel material had moved radially through the flowtube breach outward to the surrounding outer wall over the region x/L=0.7 to 1.0. Most of the upper half of the (preirradiated) fuel from the center pin had redistributed fairly uniformly along the length of the original fuel region, with extra accumulation at x/L=1.3 to 1.5, and had a sponge-like appearance. Most of the fuel/steel material that had dispersed axially (less than 2% of the total test fuel mass) was distributed throughout a 250-mm-long region above the fuel column. Most of the axially-dispersed fuel was the pre-irradiated fuel. The debris accumulation in the region x/L=0.8 to 1.05 caused the flow blockage.			

Summary of TREAT Test E7

Performed 9 March 1973

Transient No. 1499

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium at ~380 °C and initial flowrate of 3.9 m/s (570 cm ³ /s)
Form	He-bonded MOX	Transient conditions	~2.4 s at ~140 MW TREAT power (41 kW/m in central pin, 53 kW/m in 5 hottest edge pins), then a power burst rising on 180 ms period, reaching 2560 MW at 0.54 s after start of burst. Power dropped to <200 MW within next 0.4 s. Burst full-width-at-half-maximum = 310 ms.
Composition	75 wt% UO ₂ (77% enriched) 25 wt% PuO ₂	Fuel sample configuration in the test	7-pin bundle of nominally-identical pins
Geometry	Solid pellet	Test vehicle & experiment system	Mark-II integral sodium loop with B ₄ Si filtering. Fluted, 0.9-mm thick, stainless steel flowtube surrounded the pin bundle. Outside the flowtube was a 0.76-mm-thick stainless tube; a 3.2 mm annulus existed between the tubes.
Fissile height	343 mm	Measurements /diagnostics during test	Coolant and flowtube temperatures, coolant flow rate and pressure, fuel motion (hodoscope)
Smear density	9.2 g/cm ³	Posttest coolant flow condition	Complete flow blockage
CLADDING		Posttest condition of test sample(s)	Highly disrupted. All of cladding and nearly all of fuel had melted. Considerable fuel and steel above original top of fuel region.
Material	316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, macroscopic examination, optical microscopy
OD	5.84 mm	Posttest analyses	Thermal-hydraulic; hodoscope fuel-motion data analysis
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.2 at% HM		
Peak linear power	32-38 kW/m		
Peak internal clad temperature	~500 °C		
Fluence (spectrum)	(Fluence not available) Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTR/FTR-type wire-wrapped HEDL N-F fuel pins. 170 mm UO ₂ insulator above the fuel; 13 mm UO ₂ insulator below the fuel. Inconel reflector 127 mm long above the upper insulator pellets. High-power-structure fuel after irradiation.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test E7 subjected high-power-microstructure irradiated MOX fuel to a hypothetical \$3/s transient overpower LMFBR accident. Flow disturbances began about 400 ms into the power burst (at TREAT power of ~1500 MW). During the next 130 ms (as the TREAT power reached its peak), three successive jumps in the outlet flow rate occurred, and the inlet flow gradually slowed and then reversed direction, corresponding to coolant voiding in the fuel region. Thereafter, the outlet flow suddenly decreased substantially and then gradually decreased further, reaching zero at ~300 ms after peak TREAT power. The peak pressure recorded during the test was 0.5 MPa, at the inlet. Post-failure fuel motion was mostly lateral, occurring at various times, with small amounts of fuel moving to the top and bottom of the original fuel region. Posttest radiography and macro-examination of the remains revealed that a few partially-intact pellets remained at the original bottom of the fuel region and that a little fuel and steel had flowed below that region. The flowtube had melted almost totally within the original fuel region and above (fuel relative height $x/L = 0$ to 1.25). The cladding and spacer wires had melted completely within $x/L = 0$ to 1.0, and the fuel columns had melted from $x/L = 0.05$ to 1.0. Most of the steel movement had preceded the fuel movement, with steel extending to the wall radially beyond the flowtube. Large globules of steel were found within the fuel in some regions. Accumulations of fuel and steel were of low density between $x/L = 0$ and 0.35, were not present between $x/L = 0.35$ and 0.50, were of medium density between $x/L = 0.5$ and 0.8, were of high density between $x/L = 0.8$ and 1.0, and were of medium density above that point.

Summary of TREAT Test E8

Performed 9 July 1974

Transient No. 1588

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium at 400 °C and an initial flowrate of 575 cm ³ /s
Form	He-bonded MOX	Transient conditions	~2 s at 115 MW TREAT power (46 kW/m in test fuel), then a power burst reaching about 2180 MW TREAT power within ~500 ms (1.4 MJ/kg energy in the test fuel). Peak power (2535 MW) occurred about 80 ms later. Then scram.
Composition	75 wt% UO ₂ (65% enriched) 25 wt% PuO ₂	Fuel sample configuration in the test	7-pin bundle of preirradiated pins
Geometry	Solid pellet	Test vehicle & experiment system	Mark-IIIC integral sodium loop with Dy filtering. Fluted, 0.9-mm thick, stainless steel flowtube surrounded the pin bundle.
Fissile height	343 mm	Measurements /diagnostics during test	Coolant pressure, flow rate, temperatures, fuel motion
Smear density	85.5% of theoretical	Posttest coolant flow condition	Complete flow blockage
CLADDING		Posttest condition of test sample(s)	Grossly disrupted; pellet stack structure only in the bottom fourth of the fuel region.
Material	316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, gamma scanning, photography, optical microscopy
OD	5.84 mm	Posttest analyses	Thermal-hydraulics; hodoscope data analysis
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~5 at% HM		
Peak linear power	30 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	(Fluence not available) Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF/FTR-type wire-wrapped PNL-10 fuel pins; low power microstructure but with partial small central void. 356 mm UO ₂ insulator above the fuel; 169 mm UO ₂ insulator below the fuel. Inconel or stainless steel reflector 127 mm long above the upper insulator pellets.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test E8 subjected low-power-microstructure irradiated MOX fuel to a hypothetical \$3/s transient overpower LMFBR accident. The initial cladding failure occurred at 2180 MW TREAT power (test fuel power about 29 times the 30 kW/m nominal preirradiation power), when about 40% of the TREAT overpower burst energy had been generated. Failure was accompanied by 2% of the fuel mass being promptly ejected above the original top of the fuel column, rapid sodium voiding and inlet flow reversal, and low (1.5 MPa) pressure pulses at the inlet. Upward fuel dispersal occurred only during the first 80 ms after fuel motion began, after which a fuel-debris blockage apparently occurred, involving about 60 g of fuel. Subsequent fuel motions (all during the next 120 ms) occurred mostly within the original fuel region. The posttest distribution of fuel and steel was basically an accumulation of mostly steel plus some fuel located as far as 33 cm above the original fuel region, a dense accumulation of fuel and steel in the top half of the original fuel region (x/L=0.5 to 1.0, including complete melting of the flowtube), the region x/L=0.25 to 0.5 essentially voided of both fuel and steel, and some fuel and steel in the region x/L=0 to 0.25. All of the fuel in the upper two-thirds of the fuel columns apparently melted. Most of the steel that had melted moved above the top of the initial fuel region.			

Appendix N: One-page Summaries of H-series Tests

One-page summaries of each of the H-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test H1

Performed 2 February 1970

Transient No. 1298

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 0.4 m/s, with initial loop plenum pressure of 3 bar.
Form	Fast reactor U-Pu oxide	Transient conditions	Natural burst transient of 0.36 s full-width-at-half-maximum, generating 960 J/g in the test fuel.
Composition	U-15%Pu, U enriched to ~60% ²³⁵ U	Fuel sample configuration in the test	Single pin oriented with plenum below the fuel, toward the coolant inlet.
Geometry	Annular pellet, 0.76 mm ID	Test vehicle & experiment system	Mark-II sodium loop with B ₂ Si filtering. Single-pin test train with insulated flow-tube wall 8.28 mm ID, 0.79 mm thickness
Fissile height	37.6 cm	Measurements /diagnostics during test	Temperature, flow rate
Smear density	9.15 g/cm ³	Posttest coolant flow condition	Same as at the start of the test.
CLADDING		Posttest condition of test sample(s)	The fuel pin remained intact, with little effect due to the test.
Material	316 stainless steel	Posttest measurements & information	Visual inspection, radiation monitoring, neutron radiography, eddy current inspection of cladding, X-radiography
OD	5.84 mm	Posttest analyses	Thermal-hydraulics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	None		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(none)		
OTHER FUEL SAMPLE INFORMATION			
PFR-design fuel pin, but with a wire wrap of 0.8 mm diameter. Fuel mass = 69.7 g. An extensometer was located within the pin plenum.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test H1 was intended to be a preliminary, failure-threshold test to begin the H-series. Posttest visual inspection and radiation monitoring found no damage to the fuel pin caused by the experiment. Eddy current examination indicated possible abnormal regions in the cladding amounting to 10-20% of the wall thickness located in the fuel region 4 to 9 cm from the plenum end of the fuel. The insulator pellet at the end of the fuel column farthest from the plenum was badly fractured due to the transient. It was determined posttest that the extensometer was probably in contact with the end of the fuel column and bound within the cladding prior to the test (and thus possibly mechanically restraining axial fuel expansion). The fuel pellets became extensively cracked, and irregularities formed in the fuel central void, as a result of the transient.			

Summary of TREAT Test H2

Performed 11 June 1970

Transient No. 1317

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 6.3 m/s (165 cm ³ /s) and 400 °C
Form	Fast reactor U-Pu oxide	Transient conditions	Power transient produced by a 1.99% $\Delta k/k$ reactivity step, with 80 ms initial period, generating 490 MW-s with full-width-at-half-maximum of ~1.3 s.
Composition	75%UO ₂ -25% PuO ₂ ; 65% enr. U	Fuel sample configuration in the test	Single pin
Geometry	Clad pellet stack	Test vehicle & experiment system	Mark-IIA integral sodium loop, with B ₂ Si filtering. Outward from the 0.8-mm thick stainless steel flowtube was a 0.55-mm wide evacuated annulus, then a 0.8-mm thick stainless steel tube, then a 10-mm-thick sodium annulus.
Fissile height	34.4 cm	Measurements /diagnostics during test	Temperatures, inlet pressure, inlet and outlet flow rates; fuel motion monitoring by the hodoscope
Smear density	9.86 g/cm ³ (89% of theoretical)	Posttest coolant flow condition	Complete blockage.
CLADDING		Posttest condition of test sample(s)	Extensively damaged
Material	Type 316 stainless steel	Posttest measurements & information	Neutron radiography, macro-photography, microphotography, particle-size analysis
OD	5.84 mm	Posttest analyses	Thermal hydraulics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	(Not pre-irradiated)		
Peak linear power	(Not applicable)		
Peak internal clad temperature	(Not applicable)		
Fluence (spectrum)	(Not applicable)		
OTHER FUEL SAMPLE INFORMATION			
1-mm-diameter wire wrap			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test H2 subjected a fresh LMFBF fuel pin to a moderately-fast transient overpower excursion sufficient to cause pin failure. (It was nearly identical to test E4 which used a faster overpower transient.) Upward in-pin fuel motion may have begun as early as 100 ms before cladding failure. Mild sodium boiling existed for 25 to 50 ms before failure. The cladding failed 0.21 s after peak power (when the power level was only ~30% of peak power, ~90% of the transient energy had been generated, and 1.5 kJ/g had been generated in the test fuel). At failure, a little fuel was promptly released, the inlet flow reversed, and an outlet flow spike occurred. The flow channel at the outlet started to become blocked by relocated fuel and cladding within ~10 ms after cladding failure and was essentially completely blocked by ~50 ms. A major event occurred 200 ms after cladding failure, involving redistribution of much fuel and steel. A large fraction of the fuel moved above the original fuel region. There was no evidence of massive fuel movement toward the inlet, and no energetic fuel-coolant interactions occurred. Because the radial power profile in the fuel pin was strongly peaked toward the outer periphery, fuel melting began within 0.5 mm from the fuel outer surface and then moved inward. Thermal equilibrium conditions in the test-fuel pin were being approached when the pin failed; hence the test may be considered more characteristic of a scram-protected accident than an unprotected accident. Posttest examination revealed that the flowtube wall was penetrated near the top of the fuel column, allowing a large (2.5 times) increase in available volume for sodium to move into laterally. Little molten fuel or steel had entered that space, though. A large fraction of the cladding had been destroyed, and fuel had extensively dispersed out to the flowtube wall but mostly remaining axially within the original fuel region. The inside surface of the flowtube was extensively coated with fuel particles. A complete flow blockage had formed just above the original fuel top. The top of the fuel pin had moved upward ~4 cm.			

Summary of TREAT Test H3

Performed 29 June 1972

Transient No. 1456

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 4 m/s (740 cm ³ /s) and 375 °C
Form	Fast reactor U-Pu oxide	Transient conditions	Preheat at 55 kW/m for 3.0 s, followed by a 0.25 s power increase to ~140 kW/m, holding at that power for 0.4 s, then scram.
Composition	75 wt%UO ₂ – 25 wt% PuO ₂ 65% enriched U (central pin); O/M=1.97	Fuel sample configuration in the test	7-pin bundle, pre-irradiated central pin surrounded by six unirradiated pins
Geometry	Solid, dished pellets	Test vehicle & experiment system	Mark-IIA integral sodium loop with B ₄ C filtering and axial flux-shaping collars. Fluted stainless steel flowtube 0.9 mm thick in good thermal contact with 2.2 mm thick stainless steel outer wall.
Fissile height	34.2 cm	Measurements /diagnostics during test	Sodium temperature, flow rate, and pressure. Flowtube temperatures. Fuel motion using the hodoscope.
Smear density	88.9% of theoretical	Posttest coolant flow condition	Same as at the beginning of the test.
CLADDING		Posttest condition of test sample(s)	No cladding failures. No fuel movement.
Material	Type 316 stainless steel	Posttest measurements & information	Neutron radiography, gamma scanning fuel radiochemical measurements. Visual inspection; Macro- and microphotography. Areal fuel melt fraction. Cladding bow, strain, and eddy-current measurement. Laser gas sampling of the central-pin fuel.
OD	5.84 mm	Posttest analyses	Thermal-hydraulic, cladding mechanics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~3.5 at%		
Peak linear power	35 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	~3.2 x 10 ²² n/cm ² Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
FFTF-type fuel pins, one preirradiated at intermediate-power, the others unirradiated. The fuel column was bounded above and below by 1.3 cm of UO ₂ insulator pellets. A 15-cm long Inconel reflector was above the upper insulator pellets. Fuel mass = 68 g/pin.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test H3 was the first transient overpower failure-threshold experiment in TREAT that used irradiated fuel and also employed a TREAT transient that was shaped to produce test-fuel temperature distributions typical of steady state prior to the overpower excursion. The overpower ramp simulated a 0.5-1 \$/s reactivity ramp in FFTF/FTR. There were no pressure or flow events during the test, and no fuel motion was detected. Posttest examinations revealed no transient-induced cladding strain occurred. No fuel melting or radial swelling occurred in the central pin, but calculations indicated the peak fuel temperature was just short of the solidus. Measurements indicate up to ~40% of the fuel area in the hottest fresh pin exceeded the solidus. Relative to an untested preirradiated sibling pin, no grain growth occurred in the fuel of the preirradiated tested pin due to the transient.

Summary of TREAT Test H4

Performed 19 December 1973

Transient No. 1530

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at 3.5 m/s (730 cm ³ /s) and 370 °C
Form	Fast reactor MOX	Transient conditions	Preheat at 70 kW/m (235 MW TREAT power) for 2.5 s, followed by a power rise on a 0.96-s period reaching a peak of 654 MW TREAT power, with 1.06 s full-width-at-half-maximum.
Composition	75 wt%UO ₂ – 25 wt% PuO ₂ 77% enriched U (central pin) O/M= 1.97	Fuel sample configuration in the test	7-pin bundle, pre-irradiated central pin surrounded by six unirradiated pins
Geometry	Solid pellets but with full-length central void after pre-irradiation	Test vehicle & experiment system	Mark-IIA integral sodium loop with B ₆ Si filtering and axial flux-shaping collars. Fluted stainless steel flowtube 0.9 mm thick surrounded by a circular 0.9 mm stainless steel outer tube with a 2.4 mm annular helium gap in between
Fissile height	34.4 cm	Measurements /diagnostics during test	Sodium flow rate, pressure, and temperature. Flowtube temperature. Fuel motion (by hodoscope)
Smear density	84.6% of theoretical (as-fabricated preirradiated fuel); 87.5% fresh pins	Posttest coolant flow condition	10% of the initial flow rate.
CLADDING		Posttest condition of test sample(s)	Gross disruption of all seven fuel pins over the initial fuel region.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography
OD	5.84 mm	Posttest analyses	Thermal-hydraulic
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	4.4 at %		
Peak linear power	35 kW/m		
Peak internal clad temperature	510 °C		
Fluence (spectrum)	3.78 x 10 ²² cm ⁻² Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
Above the preirradiated fuel was a 17-cm high stack of UO ₂ pellets, then a ~13 cm Inconel rod. Above the fresh fuel was a ~1 cm stack of UO ₂ pellets, then a ~15 cm Inconel rod. All pins were wire-wrapped. Fuel mass 64.3 g/pin.			

KEY SUMMARY INFORMATION ABOUT THE TEST
In test H4, a preirradiated fuel pin containing a nearly axially-uniform distribution of high-power-structure fuel was tested to failure under conditions representative of a hypothetical 50¢ unprotected overpower transient in FFTF/FTR. Six fresh pins surrounding the pre-irradiated pin helped to simulate the local thermal-hydraulic environment in a subassembly of pins. Pin failure occurred at 580 MW (~170 kW/m in the test fuel) as the power was rising and when the sodium was still well below boiling. The thermocouple data suggested failure was near the fuel midplane. The flowtube was breached within 50 ms after cladding failure, and the sodium flow into the fuel region began decreasing. At ~0.30 s after cladding failure, the inlet flow reversed, and 20-25 g of fuel (about one-third of the inventory in a pin) moved upward from the middle third of the fuel region, with about 10 g moving to the top of the original fuel region. About 50 ms later, 30-35 g of fuel was driven downward from the upper half of the fuel region to the lower half, and 70 ms later fuel was dispersed upward from the bottom half of the fuel region. Several inlet sodium reentry-expulsion events followed. Due to earlier-than-predicted cladding failure and with no automatic reactor shutdown upon cladding failure, post-failure fission energy generation (25% of the burst energy) was greater than desired and resulted in the melting and relocation of nearly all of the fuel in the seven pins. Most of the fuel and cladding moved upward, with molten steel moving ahead of the fuel. A mass of molten steel froze just above the top of the original fuel column, partially blocking the coolant channels. Gross upward motion of fuel and steel occurred late in the test and was apparently stopped by the partial steel blockage, resulting in material containing most of the fuel from the seven pins collecting in a 10-cm-high region extending to just above the original top of the fuel columns and nearly filling the cross section bounded by the tube surrounding the flowtube. There were scattered deposits of fuel (mostly pre-irradiated fuel) on the outside of the cladding tubes adjacent to the pin plenum regions. A dense region of fuel debris formed over the bottom 5 cm of the original fuel column. Sodium flow posttest may have by-passed the lower blockage through the breached flowtube. The flow tube was significantly melted in the axial region near the fuel midplane.

Summary of TREAT Test H5

Performed 5 February 1973

Transient No. 1490

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium at ~3.3 m/s (~600 cm ² /s) and 350 °C
Form	Fast reactor MOX	Transient conditions	Preheat at 34 kW/m in the central pin (114 MW TREAT power) for 3.0 s, followed by a power rise on 1.1-s period with 1.0 s full-width at half maximum and peak of 600 MW TREAT power
Composition	75 wt%UO ₂ - 25 wt% PuO ₂ 65% enriched U (central pin) O/M=1.97	Fuel sample configuration in the test	7-pin bundle, pre-irradiated central pin surrounded by six unirradiated pins
Geometry	Solid pellets	Test vehicle & experiment system	Mark-IIA integral sodium loop with B ₆ Si filtering and axial flux-shaping collars. Fluted stainless steel flowtube 0.9 mm thick surrounded by a circular 0.9 mm stainless steel outer tube with a 2.4 mm annular helium gap.
Fissile height	34.1 cm	Measurements /diagnostics during test	Sodium temperatures, pressures, and outlet flow rate. Flowtube temperatures. Fuel motion using the hodoscope.
Smear density	88.6% of theoretical	Posttest coolant flow condition	40% of the initial flow rate
CLADDING		Posttest condition of test sample(s)	Slight, localized failure of the pre-irradiated pin. Mild failures of the fresh pins with considerable melting of cladding but little fuel release.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, macro-examination, cladding diameters, axial gamma scans, fission product distribution using electron microprobe
OD	5.84 mm	Posttest analyses	Thermal-hydraulic, cladding mechanics, coolant dynamics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	3.4 at%		
Peak linear power	34 kW/m		
Peak internal clad temperature	~550 °C		
Fluence (spectrum)	~3.2 x 10 ²² n/cm ² Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
Average fission gas retention in the pre-irradiated fuel was ~52%. Each pin contained ~67.5 g of fuel.			

KEY SUMMARY INFORMATION ABOUT THE TEST
In test H5, a preirradiated FFTF-type fuel pin containing intermediate-power-structure fuel (equiaxed and columnar grains but essentially no central void at the midplane) was tested to failure under conditions representing a hypothetical 50¢ unprotected overpower transient in FFTF/FTR. Six fresh pins surrounding the pre-irradiated pin helped to simulate the local thermal-hydraulic environment in a subassembly of pins. (Five of those six pins had been used in the periphery of the seven-pin cluster in TREAT test H3, where they had experienced no measurable cladding plastic strain and at most 40% of the fuel area above solidus.) Pin failure occurred at ~180 kW/m in the pre-irradiated pin, prior to coolant boiling. About 60-70 ms thereafter, one or more of the hottest fresh pins failed (at ~260 kW/m), and two small breaches were formed in the flowtube at elevations $x/L = 0.7$ and 0.9. The TREAT power began dropping fast starting 10 ms later. Because only ~15% of the overpower transient energy was generated after failure, the condition of the pin cluster immediately following pin failure was preserved. Little fuel motion occurred during the test. Peak pressure recorded was 1.5 MPa. During disassembly of the hardware, the pins were individually removed from the pin bundle. It was concluded from the posttest examination that the central, preirradiated pin was the first pin to fail. The failure site of that pin was at a fuel elevation $X/L = 0.84$ and extended axially only 2 cm, with a short spiral breach (following the spacer wire) extending immediately above. About 1 g of fuel was lost from the exposed fuel pellets at the failure site. Only 10% of the fuel cross section at that elevation had exceeded the solidus temperature during the test. At the fuel axial midplane ($x/L=0.5$), 25% of the fuel area had exceeded the solidus, was very porous, and was surrounded by, but not in close contact with, a shell of unmelted fuel. About 0.2 to 0.4% cladding strain was measured posttest in that pin. Cladding of the fresh pins melted above the axial midplane, apparently after coolant voiding began. No inelastic strain occurred in any of the fresh pin cladding where measurements could be made. The maximum fuel areal fraction above solidus in the hottest fresh pins was 50-60%.

Summary of TREAT Test H6

Performed 10 November 1977

Transient No. 1950

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium initially at ~6 m/s and 470 °C
Form	Fast reactor MOX	Transient conditions	Preheat for 2.25 s at 36 kW/m (~100 MW TREAT power) followed by a power rise on a 1.65 s period peaking at 685 MW. Overpower burst full-width-at-half-maximum was ~1.5 s.
Composition	UO ₂ -25%PuO ₂ ; 65% enriched U	Fuel sample configuration in the test	7-pin bundle of pre-irradiated pins
Geometry	Pellet stack	Test vehicle & experiment system	Mark-IIIC integral sodium loop with neutron filtering and axial-flux-shaping collars; fluted stainless steel flowtube surrounded by inert-gas annulus.
Fissile height	34.3 cm	Measurements /diagnostics during test	Sodium and flowtube temperatures, inlet and outlet pressure, inlet and outlet flow rate, fuel motion (using the hodoscope)
Smear density	85% of theoretical	Posttest coolant flow condition	Complete flow blockage
CLADDING		Posttest condition of test sample(s)	Top half of fuel region was highly disrupted. Bottom half was largely intact. Small amounts of fuel moved axially far above the original fuel region.
Material	316 stainless steel (20% cold-worked)	Posttest measurements & information	Neutron radiography
OD	5.84 mm	Posttest analyses	Thermal-hydraulic
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~6 at. %		
Peak linear power	30 kW/m		
Peak internal clad temperature	Not available.		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) spectrum		
Fuel mass ~64 g/pin. Above the fuel column was 17 cm of UO ₂ then 13 cm of stainless steel reflector. Pins were wire-wrapped.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test H6 subjected the test pins (a low-to-moderate-power structure pre-irradiated pin and six surrounding fresh pins) to a simulated hypothetical 50 ¢/s FFTF/FTR unprotected overpower transient. Pin failure was first indicated when the TREAT power was 530 MW, the pin linear power was ~200 kW/m, and two-thirds of the overpower burst energy had been generated. The inlet flow rate decreased only about 20% (too little to trigger TREAT scram), although the outlet flow rate spiked, and about 5 g of fuel was expelled upward in the flow channel and lodged on the cladding adjacent to the pin reflectors. A significant outlet flow acceleration occurred 0.15 s later, followed by outlet flow oscillations and a gradual decrease in the inlet flow rate. Some fuel accumulated in a region centered on the fuel midplane, followed by movement of about 3 g of fuel to above the active fuel region. About 0.48 s after initial failure (when 93% of the overpower burst energy had been generated), a larger event occurred, involving a large, rapid outlet flow increase, abrupt stopping of the inlet flow (triggering reactor scram), and a pressure spike of ~12 MPa at the inlet. The flowtube had failed by this time, and sodium was entering the gas space surrounding the flow tube. The event caused considerable fuel motion, with about 3% of the fuel inventory in the test moving to regions near the upper reflector, about the same amount moving to well below the active fuel zone (probably dropping within the gas annulus), and fuel also moving within the original fuel region. Posttest examination showed that the lower half of the active fuel column region remained nearly intact. Fuel from the top half of the fuel column region had moved to locations above or below the original fuel region. Some fuel had moved also into the gas annulus surrounding the flowtube. No axial movement of fuel pins (solid cladding containing fuel) was apparent.

Appendix O: One-page Summaries of EOS-series Tests

One-page summaries of each of the EOS-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test EOS-1

Performed 17 February 1977

Transient No. 1885

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant
Form	Fast reactor He-bonded MOX	Transient conditions	High-intensity, short duration TREAT power burst, initial period = 23 ms, full-width-at-half-maximum = 110 ms, generating peak power of 4300 kW/m in the test pin (approximately 100 times nominal power) and 935 kW/m in the control pin
Composition	75% UO ₂ , 25% PuO ₂ 65% enr. U in test pin; natural U in control pin; O/M ratio = 1.996	Fuel sample configuration in the test	Each pin in a separate sealed capsule. Capsules located in tandem vertically
Geometry	Solid pellet	Test vehicle & experiment system	Mark-II-C Capsule Containment Assembly (modification of part of a Mark-II-C loop), with boron thermal-neutron filtering. A 2.3-mm-thick boron nitride wall (I.D. = 8.74 mm) surrounded the test fuel pin. A thicker BN wall surrounded the control pin.
Fissile height	10.2 cm	Measurements /diagnostics during test	Pressure, temperature, fuel motion, including a thermocouple located close to the fuel column
Smear density	85% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Test pin failed, complete fuel dispersal radially and axially Control pin did not fail.
Material	316 stainless steel	Posttest measurements & information	Hodoscope scans, gamma radiography, micro-examination
OD	5.34 mm test pin; 6.60 mm control pin	Posttest analyses	Transient fuel behavior
Thickness	0.13 mm test pin; 0.76 mm control pin		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	None		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
Boron nitride pellets were located at each end of the fuel column.			

KEY SUMMARY INFORMATION ABOUT THE TEST

Test EOS-1 was conducted to provide experimental evidence of the effect of fission gas on fuel motion during a high-power burst in the absence of sodium coolant. It provided baseline data to compare with the effects of a similar transient on pre-irradiate fuels in tests EOS-2 and EOS-3. The test pin in EOS-1 was intended to fail. The control pin was designed to survive intact to supply calibration data and transient-induced microstructural changes for fuel behavior code validation. Based on data from the thermocouple located close to the fuel column, it was concluded that the fresh test fuel pin failed at or soon after peak transient power, just when calculations indicated that the test fuel liquidus was reached, when about half of the transient energy had been generated. It was presumed that some fuel vapor pressure was the cause of fuel being driven outward to contact the thermocouple (since no fission gas was present), and it was believed that the observed upward dispersal of the test fuel was caused by stainless steel vapor pressure generated from hot fuel-steel contact.

Summary of TREAT Test EOS-2

Performed 22 February 1977

Transient No. 1886

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant
Form	Fast reactor He-bonded MOX	Transient conditions	High-intensity, short duration power burst, initial period = 24 ms, full-width-at-half-maximum = 120 ms, generating peak power of 4055 kW/m in the test pin (approximately 100 times nominal power) and 920 kW/m in the control pin
Composition	75% UO ₂ , 25% PuO ₂ 65% enr. U in test pin; natural U in control pin Initial O/M ratio = 1.996	Fuel sample configuration in the test	Each pin in a separate sealed capsule. Capsules located in tandem vertically
Geometry	Solid pellet	Test vehicle & experiment system	Mark-II-C Capsule Containment Assembly (modification of part of Mark-II-C loop), with boron thermal-neutron filtering. A 2.3-mm-thick boron nitride wall (I.D. = 8.74 mm) surrounded the test fuel pin. (A thicker BN wall surrounded the control pin.)
Fissile height	10.2 cm	Measurements /diagnostics during test	Pressure, temperature, fuel motion, including a thermocouple located close to the fuel column
Smear density	85% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Test pin failed, complete fuel dispersal radially and axially Control pin: no cladding failure
Material	316 stainless steel, 20% CW	Posttest measurements & information	Hodoscope scans, gamma radiography, micro-examination
OD	5.34 mm test pin; 6.60 mm control pin	Posttest analyses	Transient fuel behavior
Thickness	0.13 mm test pin; 0.76 mm control pin		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Test pin: 0.85 at% of HM Control pin: no burnup		
Peak linear power	~37 kW/m		
Peak internal clad temperature	Not available		
Fluence (spectrum)	Fluence not available; Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
High-power-structure in pre-irradiated fuel (central void radius r/R= 0.18, columnar zone outer radius r/R=0.62, equiaxed zone outer radius r/R=0.75, unstructured zone outer radius r/R=1.0. The test pin was modified after pre-irradiation: the fuel column was shortened, BN pellets were placed at each end of the fuel column, the cladding was resealed at 2 atm. pressure, and the cladding thickness was reduced to 0.13 mm.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test EOS-2 was conducted to provide experimental evidence of the effect of fission gas on fuel motion of high-power-structure fuel during a fast high-power burst in the absence of sodium coolant, for comparison with results of EOS-3 (on medium-power-structure irradiated fuel) and EOS-1 (on fresh fuel). The test pin was intended to fail, while the control pin was designed to survive intact to supply calibration data and transient-induced microstructural changes for fuel behavior code validation. Loss of intact geometry of the test pin occurred at peak power (when roughly half of the transient energy had been generated), at a fuel specific energy roughly 200 J/g lower than at fuel failure in EOS-1 and ~100 J/g higher than at failure in test EOS-3. The time lag between the first-detected axial fuel motion and significant axial fuel motion was about 60 ms. Most of the fuel initially moved upward. The upward fuel motion appeared to be more energetic or extensive than in EOS-1. The posttest appearance of the fuel material was like that in EOS-1. Posttest evidence indicated that after fuel movement ended and the fuel solidified, molten steel tended to drain downward. Pellets from the fresh fuel pin remained in good condition in the test, with no evidence of exterior cracks and no increase in diameter.			

Summary of TREAT Test EOS-3

Performed 20 April 1977

Transient No. 1904

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Fast reactor He-bonded MOX
Composition	75% UO ₂ , 25% PuO ₂ 93% enr. U in test pin; 9% enr. U in control pin Initial O/M ratio = 1.975
Geometry	Solid pellet
Fissile height	10.2 cm
Smear density	85% of theoretical

CLADDING	
Material	304 stainless steel, annealed
OD	5.79 mm test pin; 7.06 mm control pin
Thickness	0.13 mm test pin; 0.76 mm control pin

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	Test pin: 0.81 at% of HM Control pin: no burnup
Peak linear power	~27 kW/m
Peak internal clad temperature	Not available
Fluence (spectrum)	Fluence not available; Fast (EBR-II) spectrum

OTHER FUEL SAMPLE INFORMATION	
Medium-power-structure in pre-irradiated fuel (no central void or columnar zone, equiaxed zone outer radius $r/R=0.48-0.62$, unstructured zone outer radius $r/R=1.0$. The test pin was modified after pre-irradiation: the fuel column was shortened, BN pellets were placed at each end of the fuel column, the cladding was resealed at 2 atm pressure, and the cladding thickness was reduced to 0.13 mm.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	No coolant
Transient conditions	High-intensity, short duration power burst, initial period of 27 ms, full-width-at-half-maximum = 120 ms, generating peak power of 4055 kW/m in the test pin (approximately 100 times nominal power) and 920 kW/m in the control pin
Fuel sample configuration in the test	Each pin in a separate sealed capsule. Capsules located in tandem vertically
Test vehicle & experiment system	Mark-II-Capsule Containment Assembly (modification of part of Mark-II-C loop) with boron thermal-neutron filtering. A 2.3-mm-thick boron nitride wall (I.D. = 9.14 mm) surrounded the test fuel pin. A thicker BN wall surrounded the control pin.)
Measurements /diagnostics during test	Pressure, temperature, fuel motion, including a thermocouple located close to the fuel column
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Test pin failed, complete fuel dispersal radially and axially, fuel was completely melted and had slumped downward. The control pin did not fail.
Posttest measurements & information	Hodoscope scans, gamma radiography, gamma scanning, micro-examination
Posttest analyses	Transient fuel behavior

KEY SUMMARY INFORMATION ABOUT THE TEST

Test EOS-3 was conducted to provide experimental evidence of the effect of fission gas on fuel motion of medium-power-structure fuel during a high-power burst in the absence of sodium coolant, for comparison with results of EOS-2 (on high-power-structure irradiated fuel) and EOS-1 (on fresh fuel). The test pin was intended to fail, while the control pin was designed to survive intact to supply calibration data and transient-induced microstructural changes for fuel behavior code validation. Loss of intact geometry of the test pin near peak power (when roughly half of the transient energy had been generated), earlier than in fresh-fuel test EOS-1 by roughly 300 J/g and earlier than in test EOS-2 by ~100 J/g. It is believed that axial fuel dispersal of the test fuel (generally upward) in EOS-3 was due to fission gas pressure increase. The time lag between first indications of fuel motion and significant fuel motion was about 60 ms. Posttest examination showed that, by the end of the test, all of the fuel had been melted, and most of the cladding had been melted. The molten steel was present as discrete lumps associated with fuel and as a finely-dispersed metallic phase within the fuel. Fuel material was more porous (density about 8.4 g/cm³) than the fuel remains in EOS-1 and EOS-2 (ranging from 9.3 to 10.7 g/cm³).

Pellets from the fresh fuel pin remained in good condition in the test, with no evidence of exterior cracks and no increase in diameter.

Appendix P: One-page Summaries of F-series Tests

One-page summaries of each of the F-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test F1

Performed 21 November 1974

Transient No. 1609

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant. Capsule was filled with Ar-3%He at ~1 atm pressure pretest.
Form	Fast reactor gas-bonded MOX	Transient conditions	Steady power (at ~38 kW/m peak-axial in the test fuel (and 27 kW/m in the heated wall) during the 9.7 s of the transient (until fuel motion altered the power coupling with the TREAT core). Reactor power remained constant and until scram.
Composition	75% UO ₂ – 25% PuO ₂ ; 75% enr. U	Fuel sample configuration in the test	Single pin
Geometry	Pellet stack in cladding	Test vehicle & experiment system	Dry, sealed capsule, filtered with B ₄ Si to harden the neutron spectrum. Dysprosium collars provided an EBR-II-like axial fuel power profile. Surrounding the test pin was tungsten-UO ₂ tube (enriched U), 7.87 mm ID, 0.76 mm thickness, wrapped in 0.05-mm-thick tungsten foil to provide a heated wall around the test fuel. Radially outward was a 0.5-mm thick polished Mo reflector and then a stainless steel heat sink.
Fissile height	34.3 cm	Measurements /diagnostics during test	Thermocouple above and below the test fuel region; capsule pressure transducer, hodoscope (fuel motion detection)
Smear density	(not available)	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Highly disrupted
Material	316 stainless steel	Posttest measurements & information	Neutron radiography, gamma scanning, macro- and microphotography
OD	5.84 mm	Posttest analyses	Thermal, fuel motion
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	2.35 at%		
Peak linear power	39.4 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) flux.		
OTHER FUEL SAMPLE INFORMATION			
Pre-irradiated fuel contained a central void (r/R=0.16, columnar region to r/R=0.65, equiaxed region to r/R=0.69, and unstructured region to r/R=1.00. The wire wrap on the pin was removed pretest. The plenum was vented, re-filled with Xe to 0.8 atm absolute pressure, and re-sealed before the test. About 37% of the fission gas generated during the pre-irradiation was retained in the fuel.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test F1 investigated the effect of fission gas on fuel motion at design power levels following cladding melting and drainage during a hypothetical LMFBF loss-of-flow accident simulation. Extensive fuel swelling apparently occurred prior to gross axial fuel motion. The hodoscope indicated that fuel-column bending began about 2 s into the 9.7-s-long transient and lasted for 5 s. Small masses of fuel moved downward during that time period. From 4.4 s to 6.1 s into the transient, the entire fuel column expanded upward ~1.5 cm. Significant slumping of fuel involving 10 g of fuel from the midplane region began 12.7 s into the transient (1.4 s before reactor scram). Posttest examination showed that the heated wall contained the test fuel. The cladding had melted over the entire length of the fuel column. A single fuel pellet had moved to just above the top of the original fuel region. Otherwise no fuel moved beyond the ends of the original fuel region. In the upper and lower 10% of the fuel column (elevations where pre-irradiation retained fission gas concentration was highest), the posttest condition of the fuel pellets showed radial swelling that varied gradually from little swelling at the colder end to swelling at the hotter end extending radially (up to 1.5 times the initial radius) to the heated wall. Most of the upper half of the original axial fuel region (x/L=0.45-0.89) was voided of fuel. The region x/L=0.09-0.44 was filled with once-molten fuel which had flowed down to the top of the partially-melted swelled pellets at the bottom. A large ingot of melted cladding was found near the bottom of the original fuel region. Metallic phases were present in the once-molten fuel.			

Summary of TREAT Test F2

Performed 22 November 1974

Transient No. 1610

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant. Capsule was filled with Ar-3%He at ~1 atm pressure pretest.
Form	Fast reactor gas-bonded MOX	Transient conditions	Steady power preheat for 6.5 s at 41 kW/m in the test fuel (28 kW/m in the heated wall), then a ~450 ms full-width-at-half-maximum power burst to ~11 times the preheat level.
Composition	75% UO ₂ - 25% PuO ₂ ; 80% enr. U	Fuel sample configuration in the test	Single pin
Geometry	Pellet stack in cladding	Test vehicle & experiment system	Dry, sealed capsule, filtered with B ₂ Si to harden the neutron spectrum. Dysprosium collars provided an EBR-II-like axial fuel power profile. Surrounding the test pin was tungsten-UO ₂ (enriched U) tube, 7.87 mm ID, 0.76 mm thickness, wrapped in 0.05-mm-thick tungsten foil to provide a heated wall around the test fuel. Radially outward was a 0.5-mm thick polished Mo reflector and then a stainless steel heat sink.
Fissile height	34.3 cm		
Smear density	(not available)	Measurements /diagnostics during test	Thermocouple above and below the test fuel region; capsule pressure transducer, hodoscope (fuel motion detection)
CLADDING		Posttest coolant flow condition	(not applicable)
Material	316 stainless steel	Posttest condition of test sample(s)	Highly disrupted
OD	5.84 mm	Posttest measurements & information	Neutron radiography, gamma scanning, macro- and microphotography
Thickness	0.38 mm	Posttest analyses	Thermal, fuel motion
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	0.35 at%		
Peak linear power	38 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) flux		
OTHER FUEL SAMPLE INFORMATION			
Pre-irradiated fuel contained a central void ($r/R=0.26$, columnar region to $r/R=0.81$, equiaxed region to $r/R=0.87$, and unstructured region to $r/R=1.00$. The wire wrap on the pin was removed pretest.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test F2 investigated the effect of fission gas on fuel motion resulting from a power burst following cladding melting and drainage during a hypothetical LMFBR loss-of-flow accident simulation. Just before the power burst began, the hodoscope data indicated that the fuel column was still erect, and the lower thermocouple indicated a substantial amount of molten steel (confirmed posttest) had drained to the bottom of the initial fuel region. The major fuel motion event was upward motion of the top ~10 cm of the fuel column and occurred when about one-third of the TREAT burst energy had been generated. A second, upward fuel motion event occurred 0.14 s later (around peak reactor power) involving an additional 5 to 8 cm of the fuel column. During the following 0.7 s, much of the fuel that had moved upward fell back down and accumulated in the region around the bottom half of the pin. Posttest it was determined that fuel was located over a 16-cm-high region extending as high as 19 cm above the original top of the fuel column, in which was a 10-cm-high annular region filled with once-molten fuel. Progressively below that mass were a 14-cm-high region ($x/L=0.7-1.1$) voided of fuel, a 10-cm-high region ($x/L=0.4-0.7$) partially voided of fuel, a 9-cm-high region ($x/L=0.1-0.4$) region of once-molten fuel extending radially outward to the heated wall, and a 3-cm-high region ($x/L=0-0.1$) containing some fuel. Below the original fuel region was a large mass of solidified cladding steel. At an unknown time during the burst, the heated wall had deformed along almost the entire length of the original fuel region, its UO₂ phase had melted and mostly relocated (thereby becoming indistinguishable from fuel), and a considerable amount of test fuel had leaked radially through the wall.

Summary of TREAT Test F3

Performed 3 July 1980

Transient No. 2216

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant
Form	Fast reactor MOX	Transient conditions	Steady TREAT power generating (prior to test-fuel relocation) 82 kW/m in the test fuel.
Composition	75%UO ₂ -25%PuO ₂ , 45% enr. U O/M = 1.97 +/- 0.02	Fuel sample configuration in the test	Single pin
Geometry	Solid-pellet stack in cladding	Test vehicle & experiment system	Dry capsule with a window adjacent to the fuel column to allow optical diagnostics during the transient. Included as CdS thermal neutron filter.
Fissile height	5.1 cm	Measurements /diagnostics during test	High speed photography at 2000 frames/s; fast-neutron hodoscope
Smear density	88% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Highly disrupted
Material	316 stainless steel	Posttest measurements & information	Neutron radiography, macro-examination
OD	6.35 mm	Posttest analyses	Thermal
Thickness	0.41 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9 at%		
Peak linear power	26.2 - 29.5 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) spectrum.		
OTHER FUEL SAMPLE INFORMATION			
The fuel pin was made from an 8.2-cm length of the 34.3 cm fuel region of a pre-irradiated pin. 1.6 cm of fuel at each cut end was removed, tantalum pellets were placed against the ends of the remaining fuel stack, and end plugs were welded on. The 10-11 g of fuel (and adjacent cladding) in the finished pin was from an axial location x/L = 0.67-0.83 along the original fuel column. The fuel sample likely had large equiaxed grains at the center of the fuel column.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test F3 was performed to determine the timing and mode of disruption of fuel pins in the lead power-to-flow subassemblies during an unprotected loss-of-flow situation in an LMFBR with a moderately positive sodium void reactivity. The fuel power level was about 2.7 times nominal power in lead power-to-flow subassemblies in CRBR. Fuel failure occurred when approximately 870 J/g of energy had been generated in the test fuel. The initial fuel motion caused fuel to move laterally 25 mm to the mirror in about 1.5 ms. Calculations indicate the fuel was not molten when the cladding first failed. Posttest examination showed that the mid-section of the fuel region was destroyed, the upper segment of the pin had moved down about 4.5 cm and was tilted, the fuel had partially melted, and there was some fuel dispersal. There was no fuel in either the upper or lower fuel pin segment. The bottom 0.5 mm of the upper tantalum pellet showed signs of having been heat-affected by the transient.			

Summary of TREAT Test F4

Performed 12 June 1980

Transient No. 2214

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant
Form	Fast reactor MOX	Transient conditions	Steady TREAT power generating (prior to test-fuel relocation) 216 kW/m in the test fuel.
Composition	75%UO ₂ -25%PuO ₂ , 45% enr. U O/M = 1.97 +/- 0.02UO ₂ -PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Solid-pellet stack in cladding	Test vehicle & experiment system	Dry capsule with a window adjacent to the fuel column to allow optical diagnostics during the transient. Included a CdS neutron filter.
Fissile height	5.1 cm	Measurements /diagnostics during test	High speed photography at 2000 frames/s; fast-neutron hodoscope
Smear density	88% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Highly disrupted
Material	316 stainless steel	Posttest measurements & information	Neutron radiography, macro-examination
OD	6.35 mm	Posttest analyses	Thermal
Thickness	0.41 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9 at%		
Peak linear power	26.2 - 29.5 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) spectrum.		
OTHER FUEL SAMPLE INFORMATION			
The fuel pin was made from an 8.2-cm length of the 34.3 cm fuel region of a pre-irradiated pin. 1.6 cm of fuel at each cut end was removed, tantalum pellets were placed against the ends of the remaining fuel stack, and end plugs were welded on. The 10-11 g of fuel (and adjacent cladding) in the finished pin was from an axial location x/L = 0.67-0.83 along the original fuel column. The fuel sample likely had large equiaxed grains at the center of the fuel column.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test F4 was performed to determine the timing and mode of disruption of fuel pins in the lead power-to-flow subassemblies during an unprotected loss-of-flow situation in an LMFBR with a moderately positive sodium void reactivity. The fuel power level was about seven times nominal power in lead power-to-flow subassemblies in CRBR. Fuel failure occurred when approximately 810 J/g of energy had been generated in the test fuel. Calculations indicate that the fuel was not molten when the cladding first failed, and the cladding may not have been molten either. Posttest examination showed that the mid-section of the fuel region was destroyed, the upper part of the pin had moved down about 2 cm, the fuel had partially melted, and there was some fuel dispersal. The only fuel remaining inside intact cladding was a small amount of apparently unmelted fuel located against the upper tantalum pellet which was still in place. Neither the upper nor the lower tantalum pellet showed any evidence of having been affected by the transient.			

Summary of TREAT Test F6

Performed 9 April 1984

Transient No. 2554

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	No coolant
Form	Fast reactor MOX	Transient conditions	Constant TREAT power producing a temperature rise rate of ~600 K/s in the test fuel, simulating a reactor power of 2 to 4 times nominal (roughly 90 kW/m in the test fuel).
Composition	75%UO ₂ -25%PuO ₂ , 45% enr. U O/M = 1.97 +/- 0.02UO ₂ -PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Pellet stack in cladding	Test vehicle & experiment system	Dry capsule similar to the capsule used in tests F3 and F4 but designed for laser illumination of the sample. CdS neutron filters were included.
Fissile height	5.1 cm	Measurements /diagnostics during test	High-speed photography at 8000 frames/s; fuel motion monitoring using the hodoscope
Smear density	88% of theoretical	Posttest coolant flow condition	(not applicable)
CLADDING		Posttest condition of test sample(s)	Highly disrupted
Material	316 stainless steel	Posttest measurements & information	(not available)
OD	6.35 mm	Posttest analyses	(not available)
Thickness	0.41 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	9 at%		
Peak linear power	29.5 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) spectrum.		
OTHER FUEL SAMPLE INFORMATION			
The fuel sample was made from a segment of a pre-irradiated fuel pin, probably as in tests F3 and F4.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test F6 investigated the behavior of a short segment of mixed-oxide fuel pin heated through pin disruption in a simulation of the thermal conditions in an LMFBR loss-of-flow accident following sodium voiding. It was intended to repeat test F4 except with an improved optical fuel-motion diagnostics system. A clear sequence of photographs of the failing fuel pin was obtained. An initial examination of the photographs showed a local break or tear in the still-solid cladding and a release of a bright (extremely hot) substance that quickly (within ~1 ms) filled the viewing area.			

Appendix Q: One-page Summaries of D-series Tests

One-page summaries of each of the D-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test D-1

Performed 22 June 1971

Transient No. 1394

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium flow at 850 cm ³ /s; inlet temperature increasing from 415 °C to 570 °C during the transient
Form	UO ₂ pellets helium bonded to cladding	Transient conditions	Constant power of 40 kW/m (82 kW/m in the over-enriched section) in the central pin for ~13 s. The power generated in the outer pins was 10-20% less than in the center pin.
Composition	UO ₂ ; 26% enriched U in central pin (except 3.8 cm of fuel at midplane 93% enriched), 20% enriched in outer pins; O/M = 2.00	Fuel sample configuration in the test	Seven-pin bundle, with center pin higher enriched than edge pins
Geometry	Stack of dished pellets	Test vehicle & experiment system	Mark-IIA integral sodium loop; 7-pin test train with flowing sodium area = 1.86 cm ² surrounded by a 0.9-mm-thick stainless steel wall
Fissile height	34.3 cm	Measurements /diagnostics during test	Coolant pressure, flow rate, and temperature; structure temperatures.
Smear density	9.2 g/cm ³	Posttest coolant flow condition	No flow blockage
CLADDING		Posttest condition of test sample(s)	All pins remained intact.
Material	316 stainless steel	Posttest measurements & information	Neutron radiography; macro- and micro-examinations; gamma scanning; X-radiography; gas analysis; SEM
OD	5.84 mm	Posttest analyses	Thermal-hydraulic and cladding mechanics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	None		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
All pins were wire-wrapped.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test D1 was intended to obtain information on the conditions required to achieve the release of a small amount of molten fuel from a failed fresh UO₂ pin into flowing sodium and, if failure occurred, to provide information useful in determining the mechanism and cause of the release, the extent of any molten-fuel/coolant interaction, the nature of any flow impediment caused by the fuel release, and whether any damage to adjacent pins tends to propagate additional pin failure. Pin failure did not occur in the test, however. Posttest visual examination revealed no anomalies on the pin exteriors. There was no statistically-significant change in length or bow of any of the pins before their wire wraps were removed, and there was little or no change in cladding diameters. The pellets of the fully-enriched fuel in the central pin were extensively cracked, especially the top pellet which was badly shattered. Melting of the fully-enriched pellets (up to 55% of the fuel area) created central voids in those pellets. Fractured pellets of 26% enrichment just below the fully-enriched pellets showed no evidence of melting or central void formation. No melting occurred in any of the edge pins.			

Summary of TREAT Test D-2

Performed 28 October 1971

Transient No. 1414

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Sodium flow at 870 cm ³ /s; inlet temperature increasing from 425 °C to 555 °C during the transient
Form	UO ₂ pellets, helium bonded to cladding	Transient conditions	Constant power of ~70 kW/m (~140 kW/m in the fully-enriched section) in the central pin for ~9 s. The power generated in the outer pins was 10-20% less than in the center pin.
Composition	UO ₂ ; 26% enriched U in central pin (except 3.8 cm of fuel at midplane 93% enriched), 20% enriched in outer pins; O/M = 2.00	Fuel sample configuration in the test	Seven-pin bundle, with center pin higher enriched than edge pins
Geometry	Stack of dish-d pellets	Test vehicle & experiment system	Mark-IIA integral sodium loop; 7-pin test train with flowing sodium area = 1.86 cm ² surrounded by a 0.9-mm-thick stainless steel wall
Fissile height	34.3 cm	Measurements /diagnostics during test	Coolant pressure, flow rate, and temperature; structure temperatures.
Smear density	9.2 g/cm ³	Posttest coolant flow condition	No flow blockage
CLADDING		Posttest condition of test sample(s)	All pins remained intact.
Material	316 stainless steel	Posttest measurements & information	Neutron radiography; macro- and micro-examinations; eddy-current examination of the cladding; gamma scanning; X-radiography; gas analysis; alpha-radiography
OD	5.84 mm	Posttest analyses	Thermal-hydraulic and cladding mechanics
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	None		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(not applicable)		
OTHER FUEL SAMPLE INFORMATION			
All pins were wire-wrapped.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test D2 was a higher-power version of test D1 to cause pin failure and obtain information on the conditions required to achieve the release of a small amount of molten fuel from a failed fresh UO₂ pin into flowing sodium and, if failure occurred, to provide information useful in determining the mechanism and cause of the release, the extent of any molten-fuel/coolant interaction, the nature of any flow impediment caused by the fuel release, and whether any damage to adjacent pins tends to propagate additional pin failure. The test caused more fuel to melt than in D1, but pin failure did not occur in D2 either. The outlet sodium temperature at the end of the power transient was 690 °C. Posttest visual examination revealed no anomalies on the pin exteriors. There was no statistically-significant change in length or bow of any of the pins before their wire wraps were removed, and there was little or no change in cladding diameter. Destructive examination showed that up to 75% of the area of the fully-enriched section of the central pin had melted, and small amounts of molten fuel flowed through fissures in the solid fuel and contacted the cladding, where the fuel was immediately quenched. Along the 26%-enriched regions of the central pin, the maximum amount of melting was 40%. In that pin, ~6 cm length of central void immediately below the fully-enriched pellets was filled with fully-enriched fuel, and the central void immediately above the fully-enriched pellets had a layer of fully-enriched fuel. In many axial locations, melted fuel had penetrated to the cladding through pellet cracks or spaces between pellets, but the fuel-cladding contact had practically no effect on the cladding. Molten fuel significantly flowed downward into available central void spaces of pellets below. Central voids had formed in all of the fuel pins.			

Appendix R: One-page Summaries of S-series Tests

One-page summaries of each of the S-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test S-2

Performed 11 September 1968

Transient No. unknown

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 190 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	Three power-burst transients, with cooldowns in between, generating average fission energies of 900, 1380, and 1420 J/g UO ₂ , respectively. The full-width-at-half-maximum of the last transient was about 220 ms.
Composition	UO ₂ – 10% enriched U	Fuel sample configuration in the test	Square (3 x 3) array of nine pins: 5 fuel rods and 4 hollow dummy pins. The dummy pins contained helium at 1 bar pressure inside Type-304 stainless-steel tubing 30.5 cm long, 0.737 cm OD and 0.305 mm wall.)
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave with pin bundle submerged in sodium located inside a Zircaloy sleeve.
Fissile height	14.9 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer.
Smear density	84-90% of theoretical density	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Central pin had failed with about 60% of the fuel having left the cladding. The cladding had split, but there was no sign of melting or cladding expansion at that location. The other pins remained intact.
Material	Type 304 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	7.37 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellet, with an unenriched pellet at each end. Total of 41 g of fuel in each fueled pin (205 g total in the test). Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 200 g of sodium in the test. The fuel in the pins was helium bonded to the cladding at normal temperature and pressure. A 15.5 cm long, unfueled section, filled with helium at 0.1 MPa, was located in the cladding above the fuel but sealed off from the fueled section. Ratio of sodium/fueled pin cross-sectional areas = 2.35.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-2, in conjunction with tests S-3 and S-4, investigated the effect of increasing fuel energy on the consequences of fuel pin failure. In Test S-2, the pins were subjected to three power burst transients of increasingly-greater energy. The cladding of the central fuel pin failed at 980 kJ/g during the third transient by a tear in the cladding, causing an initial 0.55 MPa pressure pulse, followed by a 1.2 MPa pulse (the highest pressure recorded during the test) when the pin energy reached 1170 kJ/g (the latter probably after fuel-sodium contact). About 59% of the fuel in the central pin escaped from the cladding. Diametral measurements showed no dimensional changes in the vicinity of the tear, but they showed about 3% increase in diameter at the fuel midplane. The other pins remained intact during the test and showed no dimensional changes. No signs of cladding melting were found.			

Summary of TREAT Test S-3

Performed 1 November 1968

Transient No. unknown

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 165 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	Two power-burst transients, generating average fission energies of 0.99 kJ/g and 1.92 kJ/g, respectively, in the test fuel. The full-width-at-half-maximum of the last transient was about 220 ms.
Composition	UO ₂ – 10% enriched U	Fuel sample configuration in the test	Square (3 x 3) open array of nine pins: 5 fuel rods and 4 hollow dummy pins. The dummy pins contained helium at 1 bar pressure inside Type-304 stainless-steel tubing 30.5 cm long, 0.737 cm OD and 0.305 mm wall.)
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave with pin bundle submerged in sodium located inside a Zircaloy sleeve.
Fissile height	14.9 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer.
Smear density	84-90% of theoretical density	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	All five fueled pins were completely disrupted and melted throughout the fuel region. Much of the fuel was in the form of hollow cylinders. All dummy pins were also breached.
Material	Type 304 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	7.37 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellet, with an unenriched pellet at each end. Total of 41 g of fuel in each fueled pin (205 g total in the test). Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 200 g of sodium in the test. The fuel was helium bonded to the cladding at normal temperature and pressure. A 15.5 cm long, unfueled section, filled with helium at 0.1 MPa, was located in the cladding above the fuel but sealed off from the fueled section. Ratio of sodium/fueled pin cross-sectional areas = 2.35.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-3, in conjunction with tests S-2 and S-4, investigated the effect of increasing fuel energy on the consequences of fuel pin failure. In test S-3, there were five separate pressure pulses: two (3.1 and 2.8 MPa) occurring at 1.86 kJ/g fuel energy, and the third pulse (largest observed in the test, having 3.7 MPa and 2.7 ms width) occurring about 10 ms later. The acceleration of the piston occurred simultaneously with the generation of the pressure pulses. A peak sodium temperature of 1100 °C was recorded by a thermocouple located in the upper part of the fueled section of the fuel pin about 0.8 s after peak power.			
Only small (~1-cm long) hollow, cylindrical sections (4.8 mm ID, 6.7 mm OD) of fused UO ₂ remained after the test. The cylindrical sections were very friable and readily broke into fine particles upon application of slight force. The lower portions of the dummy pins failed by melting. The mass-mean fuel particulate diameter was 2.7 mm. The calculated energy conversion was 0.002%.			

Summary of TREAT Test S-4

Performed 19 August 1969

Transient No. unknown

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 165 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	One power burst transient with 160 ms full-width-at-half-maximum generating 3.0 kJ/g in the test fuel.
Composition	UO ₂ – 10% enriched U	Fuel sample configuration in the test	Square (3 x 3) open array of nine pins: 5 fuel rods and 4 hollow dummy pins. The dummy pins contained helium at 1 bar pressure inside Type-304 stainless-steel tubing 30.5 cm long, 0.737 cm OD and 0.305 mm wall.)
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave with pin bundle submerged in sodium located inside a Zircaloy sleeve.
Fissile height	14.9 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer.
Smear density	84-90% of theoretical density	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Gross destruction of all five fuel pins and the four dummy pins.
Material	Type 304 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	7.37 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellet, with an unenriched pellet at each end. Total of 41 g of fuel in each fueled pin (205 g total in the test). Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 200 g of sodium in the test. The fuel was helium bonded to the cladding at normal temperature and pressure. A 15.5 cm long, unfueled section, filled with helium at 0.1 MPa, was located in the cladding above the fuel but sealed off from the fueled section. Ratio of sodium/fueled pin cross-sectional areas = 2.35.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-4, in conjunction with tests S-2 and S-3, investigated the effect of increasing fuel energy on the consequences of fuel pin failure. In Test S-4 there were five distinct pressure pulses. The first two occurred 30 ms after peak power, at the time 2.0 kJ/g had been generated in the test fuel. At about 34 ms after peak power, a pressure pulse with two peaks occurred, one of 7 MPa and the other of about 12 MPa. At about 42 ms after peak power, a 9 MPa peak pressure pulse occurred. The acceleration of the piston occurred simultaneously with the generation of the pressure pulses. Hot fuel and fuel-pin debris were propelled upward during the test. Cladding melting was much more extensive than in test S-3. The mass-mean diameter of the fuel residue was about 1.2 mm. The calculated conversion from nuclear energy to mechanical energy was estimated to be 0.025%, ten times higher than in test S-3.			

Summary of TREAT Test S-5

Performed 3 June 1970

Transient No. unknown

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 170 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	The full-width-at-half-maximum of the last transient was about 195 ms. The transient generated an average of 2.13 kJ/g within the fuel mass in the test.
Composition	UO ₂ – 10% enriched U	Fuel sample configuration in the test	Square (3 x 3) open array of nine pins: 5 fuel rods and 4 hollow dummy pins. The dummy pins contained helium at 1 bar pressure inside Type-304 stainless-steel tubing 30.5 cm long, 0.737 cm OD and 0.305 mm wall.)
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave with pin bundle submerged in sodium located inside a Zircaloy sleeve.
Fissile height	14.9 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer.
Smear density	84-90% of theoretical density	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Gross disruption of the fuel pins, with the fuel residue being highly fragmented.
Material	Type 304 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	7.37 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellet, with an unenriched pellet at each end. Total of 41 g of fuel in each fueled pin (205 g total fuel in the test). Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 200 g of sodium in the test. The fuel pins (including upper blanket sections) and dummy pins had been sealed under vacuum of 5×10^{-5} torr. A 15.5 cm long, unfueled section, filled with helium at 0.1 MPa, was located in the cladding above the fuel but sealed off from the fueled section. Ratio of sodium/fueled pin cross-sectional areas = 2.35.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-5 had the objective of investigating whether large pressures and work energies would be produced in the absence of bond gas within the pins. (Release of a gas bond from failed cladding could blanket the fuel and reduce fuel-sodium contact.) However, because the fuel pellets were 93% (not 100%) of theoretical density, some gas within the fuel pellet could have had an effect. The initial pressure pulse (0.5 MPa) occurred when the fuel energy generated had reached 1.80 kJ/g, simultaneous with the acceleration of the piston. Sixteen pressure pulses higher than 0.3 MPa were recorded, the highest being 20 MPa. There was more fuel damage than in Test S-3, and more fuel fragmentation and particulate. The mass-mean residue diameter was about 0.2 mm. A conservative maximum calculated nuclear to mechanical energy conversion during the test was estimated to be 0.04-0.2% depending upon assumptions.			

Summary of TREAT Test S-6

Performed 24 Nov 1970

Transient No. 1348

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 210 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	Power burst with 170 ms full-width-at-half-maximum, 2.06 kJ/g in the fuel
Composition	UO ₂ - uranium enrichment 13% (max.) in central pin, 10% (max.) in edge pins	Fuel sample configuration in the test	Bundle of 7 wire-wrapped fuel pins.
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave with pin bundle submerged in sodium located inside a Zircaloy sleeve.
Fissile height	12.4 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer. The TREAT Fast Neutron Hodoscope was used to measure fuel motion.
Smear density	84-90 % of theoretical density	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Central pin had failed, and about 60% of its fuel had left the cladding. The cladding had split, but there was no sign of melting or cladding expansion at that location. The other pins remained intact.
Material	Type 304 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	7.37 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellets, with an unenriched pellet at each end. Total of 286 g of fuel in the test. Pins were wrapped with 1.6 mm diameter wire. Central pin received 9% more power than each edge pin. There was 200 g of sodium in the test. Ratio of sodium/fueled pin cross-sectional areas = 0.94. A 15.5 cm long, unfueled section, filled with helium at 0.1 MPa, was located in the cladding above the fuel but sealed off from the fueled section.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-6 investigated the effect of a more-closely-packed fuel-pin geometry on fuel-coolant interaction effects, compared to the earlier tests S-2 through S-5. In S-6, cladding failure occurred when the fuel energy had reached 1.59 kJ/g. The largest pressure pulse reached 14 MPa. Based on the average fuel energy at initial cladding failure, the calculated energy conversion ratio was 0.15% to 0.037% to 0.2% depending on whether damped pressure oscillations are included in the calculation. The fuel debris particle sizes were overall much like those in tests S-3 and S-4, but they were not as small as the sizes in test S-5 which used evacuated pins.			

Summary of TREAT Test S-7

Performed 21 May 1971

Transient No. 1376

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 500-510 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	Power burst with 175 ms full-width-at-half-maximum, generating 2.15 kJ/g in the fuel.
Composition	UO ₂ – uranium enrichment 13% in central pin, 10% in edge pins	Fuel sample configuration in the test	Bundle of 7 wire-wrapped fuel pins.
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave vehicle (much different from the vehicle used in tests M-2 through M-6) with pin bundle submerged in sodium inside a Zircaloy sleeve.
Fissile height	~ 13 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer. The TREAT Fast Neutron Hodoscope monitored fuel motion.
Smear density	85% of theoretical	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Highly disrupted.
Material	Type 316 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	5.84 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellets, with an unenriched pellet at each end. In the central pin, a 10%-enriched pellet was located next to the unenriched pellet. Total of 203 g of enriched fuel in the test (~29 g/pin). Pins were wrapped with 1.6 mm diameter wire. Central pin received 9% more power than each edge pin. There was 186 g of sodium in the test-. Ratio of sodium/fueled pin cross-sectional areas = 0.97			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-7, together with test S-8, investigated fuel-coolant interaction effects with FFTF-pin geometry and sodium preheated to ~500 °C (compared to ~200 °C in the previous S-series tests. Test S-7 was run with a fuel fission energy generation similar to that in test S-3. In S-7, the initial sodium temperature was selected so that sodium boiling would occur before cladding failure. The cladding failed at 1.51 kJ/g fuel energy. The largest pressure pulse reached 2.9 MPa. The mass-mean fuel debris diameter was about 0.3 mm, similar to that in test S-6. Based on the average fuel energy at initial cladding failure, the calculated energy conversion ratio was ~0.0006%.			

Summary of TREAT Test S-8

Performed 1 October 1971

Transient No. 1403

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 500-510 °C before the power burst.
Form	He-bonded UO ₂	Transient conditions	Power burst with 140 ms full-width-at-half-maximum, generating 3.28 kJ/g in the fuel.
Composition	UO ₂ – uranium enrichment 13% in central pin, 10% in edge pins	Fuel sample configuration in the test	Bundle of 7 wire-wrapped fuel pins.
Geometry	Solid pellet	Test vehicle & experiment system	Piston autoclave vehicle (much different from the vehicle used in tests M-2 through M-6) with pin bundle submerged in sodium inside a Zircaloy sleeve.
Fissile height	~ 13 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. (The TCs had time constant = 15 ms.) Pressure transducer measured autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer. The TREAT Fast Neutron Hodoscope monitored fuel motion.
Smear density	85% of theoretical	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Highly disrupted, with evidence of high mobility of molten steel-UO ₂ debris having penetrated into small spaces.
Material	Type 316 stainless steel	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	5.84 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellets, with an unenriched pellet at each end. In the central pin, a 10%-enriched pellet was located next to the unenriched pellet. Total of 203 g of fuel in the test. Pins were wrapped with 1.6 mm diameter wire. Central pin received 9% more power than each edge pin. There was 138 g of sodium in the test. Ratio of sodium/fueled pin cross-sectional areas = 1.02			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test S-8, together with test S-7, investigated fuel-coolant interaction effects with FFTF-pin geometry and sodium preheated to ~500 °C (compared to ~200 °C in the previous S-series tests.) Test S-8 was run with a fuel fission energy generation similar to that in test S-4. In Test S-8, the initial sodium temperature was selected so that sodium boiling would occur before cladding failure. The cladding failed at 1.47 kJ/g fuel energy. The largest pressure pulse reached 5.1 MPa. Based on the average fuel energy at initial cladding failure, the calculated energy conversion ratio was 0.0006%. The mass-mean fuel debris diameter was about 0.9 mm.			

Summary of TREAT Test S-11

Performed 31 October 1972

Transient No. 1469

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 140 °C and atmospheric pressure before the power burst.
Form	He-bonded UO ₂	Transient conditions	Power burst with 23 ms initial period and 104 ms full-width-at-half-maximum, generating 7.1 kJ/g average in the test fuel.
Composition	UO ₂ – 13% enriched U	Fuel sample configuration in the test	Single fuel pin
Geometry	Solid pellet (4.93 mm dia.)	Test vehicle & experiment system	Piston autoclave. Fuel pin surrounded by 2.5-mm-thick sodium annulus bounded by a large molybdenum heat sink.
Fissile height	15.2 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. A pressure transducer measured the autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer. The TREAT Fast Neutron Hodoscope monitored fuel motion.
Smear density	g/cm ³ (92% of theoretical)	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Highly disrupted.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	5.84 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.381 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellets, with an unenriched pellet at each end. Fuel mass = 28.4 g. Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 150 g of sodium in the test.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test S-11 was performed to investigate fuel-coolant interactions under extreme conditions of fuel-vapor-driven fragmentation and mixing of molten fuel and sodium. The heat sink and low initial temperature were used to match prompt-burst conditions of fuel, cladding, and coolant temperature at the time of cladding breach thus compensating for the power rise rate being much slower than for a prompt burst. The cladding failed when the fuel energy reached 2.22 kJ/g, about 15.6 ms before peak power, generating a ~5 ms wide pressure pulse reaching 15 MPa followed (about 50 ms after peak power) by a broad pressure pulse of about 130 ms full width at half maximum with peak value about 13.6 MPa. The UO ₂ vapor pressure at cladding failure was estimated to be roughly 6-12 MPa. The estimated energy conversion ratio was 0.1%. Analysis suggested that the posttest events observed were due primarily to the action of fuel vapor pressure, and that there was little or no sodium vaporization (presumably because fuel vapor prevented fuel-sodium contact). The observed pressure events indicated that fuel-vapor-driven fuel fragmentation and mixing with sodium (as a possible mechanism for generating an energetic fuel-coolant interaction) did not occur.

Summary of TREAT Test S-12

Performed 2 November 1972

Transient No. 1470

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant sodium at 140 °C and atmospheric pressure before the power burst.
Form	He-bonded UO ₂	Transient conditions	Power burst with 23 ms initial period and 104 ms full-width-at-half-maximum, generating 7.1 kJ/g average in the test fuel.
Composition	UO ₂ – 13% enriched U	Fuel sample configuration in the test	Single fuel pin
Geometry	Solid pellet (4.93 mm dia.)	Test vehicle & experiment system	Piston autodave. Fuel pin surrounded by a 2.4-mm-thick sodium annulus bounded by a large molybdenum heat sink.
Fissile height	15.2 cm	Measurements /diagnostics during test	Temperatures were measured by four tungsten-rhenium TCs. A pressure transducer measured the autoclave pressure. Upward motion of a 260 g piston located immediately above the fuel-pin bundle was measured by a linear motion transducer. The TREAT Fast Neutron Hodoscope monitored fuel motion.
Smear density	g/cm ³ (92% of theoretical)	Posttest coolant flow condition	(Not relevant.)
CLADDING		Posttest condition of test sample(s)	Highly disrupted.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography of the fuel sample after each transient.
OD	6.35 mm	Posttest analyses	Thermal-mechanical and heat transfer analyses.
Thickness	0.635 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	Zero		
Peak linear power	(not applicable)		
Peak internal clad temperature	(not applicable)		
Fluence (spectrum)	(zero)		
OTHER FUEL SAMPLE INFORMATION			
Fueled section in each pin was a stack of pellets, with an unenriched pellet at each end. Fuel mass = 28.2 g. Top and bottom enriched pellets received 14% more energy than the other enriched pellets. There was 150 g of sodium in the test.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test S-12 was performed to investigate fuel-coolant interactions under extreme conditions of fuel-vapor-driven fragmentation and mixing of molten fuel and sodium. The heat sink and low initial temperature were used to match prompt-burst conditions of fuel, cladding, and coolant temperature at the time of cladding breach, thus compensating for the power rise rate being much slower than for a prompt burst. The cladding failed when the fuel energy reached 2.26 kJ/g, about 14.6 ms before peak power, indicated by a rapid temperature rise of the sodium and rapid acceleration of the piston, but without a significant pressure rise. The piston movement indicated that the impulse imparted to it was about half as strong as in test S-11. A pressure pulse peaking at ~7 MPa occurred at about peak reactor power. This was followed about 0.36 s later by a set of three 1-ms-wide pulses, the largest reaching ~20 MPa. (similar to the delayed pressure event in test S-11). The UO ₂ vapor pressure at cladding failure was estimated to be roughly 7-14 MPa. The estimated energy conversion ratio was 0.1%. The observed pressure events indicated that fuel-vapor-driven fuel fragmentation and mixing with sodium (as a possible mechanism for generating an energetic fuel-coolant interaction) did not occur.

Appendix S: One-page Summaries of RX-series Tests

One-page summaries of each of the RX-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test RX1

Performed 17 December 1982

Transient No. 2410

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Hot-pressed urania/steel
Composition	87.6 wt% UO ₂ / 12.4 wt% SS316, 93% or 50% enriched U
Geometry	Stack of ten 2.33-cm diameter, 1.27-cm high cylindrical slugs
Fissile height	12.7 cm
Smear density	Smear density is not applicable. The slug density was 9.68 g/cm ³
CLADDING	
Material	No cladding
OD	(not applicable)
Thickness	(not applicable)
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(not applicable)
OTHER FUEL SAMPLE INFORMATION	
The top and bottom fuel slugs had 50% enriched U. The middle eight slugs had 93% enriched U. Vertical off-center holes in the slugs (4 in the top three slugs, 3 in the fourth slug, 2 in the fifth slug, and 1 in the sixth and seventh slugs, each accommodated one thermocouple. A hole down the center of the top seven slugs accommodated a boron nitride (BN) sheath for a tungsten ultrasonic thermometer probe. The total UO ₂ fuel mass was 460 g.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	No coolant was present.
Transient conditions	TREAT power rose from 0 to 12.8 MW (~19 W/g UO ₂ in the test fuel) in 4.0 s, then was adjusted to maintain that same power density in the test fuel during the next 156 s, then scram.
Fuel sample configuration in the test	Cylindrical stack of fuel slugs 2.33 cm diameter, 12.7 cm high.
Test vehicle & experiment system	The urania/steel fuel stack was supported on a depleted urania base below which was a BN pedestal. The fuel stack was radially surrounded by a 9% enriched UO ₂ cylinder (nuclear wall, 24.6 mm inner diameter, 4.8 mm thick) extending from the bottom of the fuel stack to 30.5 cm above the top of the stack. The nuclear wall was radially surrounded by heat shields of tungsten, molybdenum, graphite, and stainless steel. A void volume of ~5 liters was present above the fuel. Outside the primary containment was a dysprosium neutron filter. The test vehicle was positioned so that the top of the fuel was at the axial midheight of the TREAT core.
Measurements /diagnostics during test	Temperatures within and above the fuel (by various types of thermocouples including a BN-sheathed ultrasonic thermocouple), system pressure, fuel motion (hodoscope)
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Melted and axially dispersed
Posttest measurements & information	(apparently no additional posttest examination was performed)
Posttest analyses	Thermal, thermal-hydraulic (particularly regarding heating of the BN above and below the pool)

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test RX1 investigated the behavior of a molten mixture of UO ₂ and steel subjected to fission heating in a simulation of the decay-heating stage of the "transition phase" of a hypothetical unprotected LMFBR accident. It was a scoping test designed specifically to focus on the nature of a heat-generating "open pool" of molten fuel and boiling steel. The hodoscope indicated fuel motion began at ~119 s into the test. Steel vaporization had been predicted to begin at about that time, but the BN sheaths used on the ultrasonic thermometer apparently began decomposing and generating considerable nitrogen gas at that time, also. The fuel initially moved upward ~4 cm, where it temporarily contacted a 0.25-mm-thick tantalum foil. At ~131 s, the fuel moved ~8.5 cm higher where it contacted a similar tantalum foil. Simultaneously, at that time, the BN began a vigorous, constant decomposition producing nitrogen gas apparently over its outer surface area extending upward at least 15 cm above the original elevation of the top of the fuel. Between 150-155 s, the fuel reached an elevation ~11 cm above the initial top-of-fuel location, and by 160 s it had moved upward an additional ~11 cm. How much of the BN decomposed above the pool versus within the pool, and how much effect the generated nitrogen gas may have had on the overall pool behavior, remained unknown.	

Summary of TREAT Test RX2

Performed 27 May 1983

Transient No. 2450

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Hot-pressed urania/steel
Composition	87.6 wt% UO ₂ / 12.4 wt% SS316 93% or 50% enriched U
Geometry	Stack of ten 2.33-cm diameter, 1.27-cm high cylindrical slugs
Fissile height	12.7 cm
Smear density	Smear density is not applicable. The slug density was 9.68 g/cm ³
CLADDING	
Material	No cladding
OD	(not applicable)
Thickness	(not applicable)
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(not applicable)
OTHER FUEL SAMPLE INFORMATION	
The bottom fuel slugs had 50% enriched U. The top nine slugs had 93% enriched U. Vertical holes in the top seven slugs accommodated a total of three thermocouples. The total UO ₂ fuel mass was ~463 g.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	No coolant was present.
Transient conditions	TREAT power rose from 0 to 12.8 MW (~21 W/g UO ₂ in the test fuel) in 4.0 s, then was adjusted to maintain that power density in the test fuel during the next 154 s, then scram.
Fuel sample configuration in the test	Cylindrical stack of fuel slugs 2.33 cm diameter, 12.7 cm high.
Test vehicle & experiment system	The urania/steel fuel stack was supported on a depleted urania base below which was a BN pedestal. The fuel stack was radially surrounded by a 9% enriched UO ₂ cylinder (nuclear wall, 24.6 mm inner diameter, 4.8 mm thick) extending from the bottom of the fuel stack to 17.8 cm above the top of the stack. The nuclear wall was radially surrounded by heat shields of tungsten, molybdenum, graphite, and stainless steel. A void volume of ~0.5 liter was present above the fuel. Outside the primary containment was a dysprosium neutron filter. The test vehicle was positioned so that the top of the fuel was at the axial midheight of the TREAT core.
Measurements /diagnostics during test	Temperatures at various locations in and above the fuel, system pressure, fuel motion (hodoscope) [No BN-sheathed ultrasonic thermometer was present.]
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Melted; radially and axially dispersed
Posttest measurements & information	Neutron radiography (apparently no additional posttest examination was performed)
Posttest analyses	Thermal, thermal-hydraulic

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test RX2 was designed to examine the behavior of the "debris pool" of molten urania fuel and boiling steel in a confined volume having void volume about ten times the pool volume. The hodoscope indicated gradual upward movement of the top of the fuel by 1.6 cm during the first 90 s of the test. The largest mass of fuel movement occurred between 109 and 120 s into the test, with the upper quarter of the fuel column relocating radially outward and downward. (This movement was confirmed by posttest radiography as having been molten fuel and steel breaching the surrounding nuclear heated wall and lodging in the annulus between the heated wall and the tungsten heat shield.) During the next 50 s, the hodoscope data indicated that fuel continued to redistribute to the lower annular region between the nuclear-heated wall and the tungsten heat shield, and a small amount of fuel accumulated about 12 cm above the top of the original fuel column.	

Appendix T: One-page Summaries of J-series Tests

One-page summaries of each of the J-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test J1

Performed 21 September 1979

Transient No. 2159

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium set at 610 cm/s (940 cm ³ /s) and 475 °C
Form	MOX fuel helium-bonded to cladding	Transient conditions	Fast power rise to 180 MW (75 kW/m peak-axial in the edge pins), then linear decrease for 1.3 s to 169 MW, then linear increase for 3.5 s to 255 MW peak power (at t=9.12 s), then linear decrease for 0.2 s to 30 MW, then scram 0.9 s later. Ratio of central pin power to edge pin power was 81%.
Composition	75%UO ₂ -25%PuO ₂ , 79 at% enriched U; O/M=1.96	Fuel sample configuration in the test	Seven-pin bundle of pre-irradiated pins
Geometry	Dished pellets in stack	Test vehicle & experiment system	Mark-IIIC sodium loop with neutron filtering and axial-flux-shaping collars; fluted stainless steel flowtube surrounded by inert-gas annulus.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures, inlet and outlet flow rates, inlet and outlet pressures, flowtube strains
Smear density	9.34 g/cm ³	Posttest coolant flow condition	Variable, increasing after scram from a low of 20% of the initial flow rate to a final value of 70% of the initial flow rate.
CLADDING		Posttest condition of test sample(s)	Three of the four pins had mildly failed.
Material	316 stainless steel, 20% cold-worked	Posttest measurements & information	Neutron radiography, macro- and micro-examinations.
OD	5.84 mm	Posttest analyses	Thermal-hydraulic; fuel transient behavior
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~ 7 at%		
Peak linear power	26 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available. Fast (EBR-II) spectrum.		
OTHER FUEL SAMPLE INFORMATION			
Fuel mass per pin = 64.3 g. Fuel pins were wire wrapped with 1.1-mm diameter wire. Above the fuel was 12.3 cm of UO ₂ insulator, then a 14.5-cm-long 4.8-mm-diameter Inconel reflector.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test J1 showed that a slow-period LMFBR transient overpower simulation causing cladding failure in an FFTF-like fuel pin could be performed within the limitations of the TREAT reactor. Fuel pin failure occurred at 9.035 s (252 MW TREAT power), causing the inlet flow rate to suddenly decrease by ~25% (not enough to trigger the programmed TREAT power drop). A second, stronger flow event occurred 40 ms later, causing (within 50 ms) the inlet flow rate to decrease to about 44% of its original value (triggering the power drop) and the outlet flow rate to triple. During the next ~100 ms, the fuel region began refilling with sodium, and the inlet and outlet flow rates both reached 80-90% of their initial values. Then the flow rates began dropping to a temporary hold at ~60% of initial value, then dropping steadily to ~20% of initial as the reactor was scrambling at t=10.26 s. Beginning at t~15 s, the inlet and outlet flow rates began a steady, linear increase reaching 70-75% at t=120 s. (The cause of this late flow behavior was apparently never determined.) Posttest examination showed that the fuel columns of some fuel pins had elongated, from 0.2 to 3.2 cm in the edge pins (higher values corresponding to higher relative pin power during the test) and 0.2 cm in the center pin. The largest expansions almost fully compressed the plenum spring. Up to about 30% areal fuel melting occurred in the pins, and central voids had formed in all of the edge pins. Major failures occurred in three pins (two of the hotter edge pins and one of the cooler edge pins); these were the only pins to release fuel. The breaches were all at the top of the fuel column and extended up along the insulator pellet stacks. The breaches were up to ~3 cm long and involved up to half the circumference of the cladding. Practically all of the fuel that had melted in the failed pins was released from those pins. Much of the released fuel (estimated to be 53 g total) stayed near the failure sites. No significant blockage to coolant flow was evident in the region occupied by the fuel pins. Only a small amount of the fuel in the center pin had melted, and there was a small crack (adjacent to an insulator pellet) in the pin's cladding. The flowtube had remained intact.			

Appendix U: One-page Summaries of HUT-series Tests

One-page summaries of each of the HUT-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test HUT1-6A

Performed 05-Apr-1978

Transient No. 2001?

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static sodium (was intended to be flowing), initially at 370 °C
Form	Mixed U-Pu oxide fuel pin	Transient conditions	Only a preliminary check-out power transient was performed. The TREAT power increased from 2 MW at t=2.7 s to ~40 MW at t=3.7 s, then remained at ~40 MW until 8.9 s, then dropped to 1 MW by 9.2 s. The planned overpower transient was not performed.
Composition	75 wt% UO ₂ - 25 wt%PuO ₂ ; 93% enriched U; O/M=1.98	Fuel sample configuration in the test	3 fuel pins, each in its own flow tube
Geometry	Stack of solid, sintered pellets	Test vehicle & experiment system	Boron-filtered Mark-IIIC sodium loop with axial flux-shaping collars. Flowtubes were 9.52 mm OD x 0.71 mm wall thickness, surrounded by inert gas
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures, sodium flow monitoring
Smear density	87% of theoretical (W3-14 pin) 90% of theoretical (W3-30 pin) 85% of theoretical (W3-31 pin)	Posttest coolant flow condition	Zero flow (same as pre-test)
CLADDING		Posttest condition of test sample(s)	All pins remained intact. Minor damage.
Material	Type 316 stainless steel, 20% cold-worked	Posttest measurements & information	Cladding profilometry; gamma scanning; neutron radiography. Destructive examination was not performed.
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	112-113 MWd/kg		
Peak linear power	~28.2 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	~7.1 x 10 ²² n/cm ² (fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
EBR-II-irradiated pins W3-14, W3-30, and W3-31 were used.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT1-6A was intended to thermally simulate the transient performance of high burnup, medium-power FFTF fuel pins to conditions representative of a 3 cents/s reactivity insertion terminated by the FFTF plant protection system. During the preliminary, constant-power heat-balance run, sodium flow was intended but did not occur, resulting in overheating of the fuel pins. High fuel-pin cladding temperatures occurred and the cladding remained intact, but the mechanical properties had become altered to the extent that the objectives of the planned test would be compromised. Therefore, the planned overpower transients on those fuel pins were cancelled. Posttest measurements showed no transient-induced cladding strain in pins W3-30 and W3-31, and 0.2% in W3-14. (The latter pin experienced the highest power among the three pins.) No fuel relocation occurred within the pins. Comparison of pre- and post-transient gamma scans indicated no fuel relocation occurred. Posttest neutron radiography showed no observable transient-induced changes in the fuel. Post-transient calculations indicated that cladding recrystallization probably occurred in the hottest region of the fuel pins and that the cladding probably would have failed if the peak cladding temperatures had been ~100 K higher.			

Summary of TREAT Test HUT1-8A

Performed (date not available)

Transient No. (not available)

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium
Form	Mixed U-Pu oxide fuel pin	Transient conditions	Five preliminary runs were performed, starting with a constant-power preheat followed by a mild overpower transient. The last preliminary run included a continuous sodium flow reduction. The final transient, run without the flow reduction, had a similar initial constant-power segment but was followed by a 19.7-s rise from 43 MW to 68 MW TREAT power. The total TREAT energy generated during the run was 1330 MJ.
Composition	Information not available; probably 75 wt% UO ₂ – 25 wt% PuO ₂		
Geometry	Stack of solid, sintered pellets		
Fissile height	34.3 cm		
Smear density	(information not available)		
CLADDING		Fuel sample configuration in the test	Information not available; probably the three pins were in separate, parallel flowtubes
Material	Type 316 stainless steel, 20% cold-worked	Test vehicle & experiment system	Integral sodium loop
OD	5.84 mm	Measurements /diagnostics during test	Information not available; probably temperatures and sodium flow rates.
Thickness	0.38 mm	Posttest coolant flow condition	(not available)
PRE-IRRADIATION CHARACTERISTICS		Posttest condition of test sample(s)	The fuel sample remained intact.
BU (peak axial)	48-52 MWd/kg	Posttest measurements & information	Gas sampling of the loop plenum.
Peak linear power	18 kW/m	Posttest analyses	LIFE-4 thermal-mechanical analysis.
Peak internal clad temperature	(not available)		
Fluence (spectrum)	~11 x 10 ²² n/cm ² (fast spectrum, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pins P-14A-32, -24, and -64 were tested.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test HUT1-8A thermally simulated the performance of high-burnup medium-power FTR fuel pins under conditions representative of an FFTF continuous flow reduction and a 3 cents/s unterminated reactivity insertion. The final run was preceded by five transients. The first four of those preliminary transients began with a 5.5 s constant power segment at 43 MW TREAT power followed by a 3 cents/s overpower increase lasting 9.2 s and terminated by the plant protection system, with peak power of 54 MW and a total TREAT energy of 698 MJ. The fifth preliminary transient began with a 5.7 s constant power segment at 43 MW, after which the power was reduced to 40 MW over 6 s and then to 35 s during the next 8.8 s, followed by reactor scram, with 828 MJ TREAT energy generated. The flow rate was continuously reduced during the power decrease. After those preliminary transients, gas sampling of the loop plenum verified that the fuel pins had remained intact. The final transient, simulating the 3 cents/s unterminated reactivity insertion was then performed (as described above).

Summary of TREAT HUT3-2A

(test was originally named HUT3-1A)

Performed 14-March-1975

Transient No. 1642

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK with temperature varying linearly from 170 °C at fuel bottom to 340 °C at fuel top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 51 MW in 0.5 s, held at 51 MW for 5.4 s, rose to 790 MW during next 0.50 s, then scram. Total of 556 MJ. Test fuel midplane power coupling factor was 4.2 to 4.5 W/g per TREAT MW. Radial power profile in test fuel was about 1.7 edge/center.
Composition	75wt% UO ₂ - 25wt%PuO ₂ , O/M=1.98	Fuel sample configuration in the test	Single pin
Geometry	Stack of solid sintered pellets	Test vehicle & experiment system	Static NaK capsule with boron filtering and shaping collars. Fuel pin surrounded by 1.3-mm NaK annulus inside a 7.8-mm thick nickel heat sink.
Fissile height	34.4 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	85% of theoretical	Posttest coolant flow condition	Static coolant (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	Significant cladding failure and fuel dispersal
Material	Type 316 stainless steel	Posttest measurements & information	Neutron radiography, gas sampling and analysis, autoradiography.
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of fuel and cladding.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	50 MWd/kg		
Peak linear power	41 kW/m		
Peak internal clad temperature	~510 °C (midplane)		
Fluence (spectrum)	~4.0 x 10 ²² n/cm ²		
OTHER FUEL SAMPLE INFORMATION			
Pin tested was PNL 11-47. Based on sibling pin examination, the fuel zone radii (r/R) were: central void 0.19, columnar zone 0.70, equiaxed zone 0.80.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Thermocouple data indicated that cladding failed 0.70 s after the start of the power burst (soon after peak power), in the upper part of the fuel column (x/L=0.8). Radiography indicated that most of the fuel dispersal from the cladding was above the fuel midheight location. The fuel melt radius r/R was ~60% at height x/L=0.4. At the fuel bottom (x/L=0.03) the melt radius was r/R=0.6, and about 40% of the cladding circumference had melted away. Near the fuel top (x/L= 0.8) the cladding was still intact but filled with once-molten fuel and surrounded by once-molten fuel that had filled the coolant annulus. An estimated 35% of the as-irradiated fission gas inventory had been released from the fuel. No results from the hodoscope are available.

Summary of TREAT Test HUT3-2B

(test was originally named HOP3-3C)

Performed 25 July 1973

Transient No. 1518

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt%UO ₂ -25wt% PuO ₂
Geometry	Stack of solid, sintered pellets
Fissile height	34.3 cm
Smear density	89% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	30 MWd/kg
Peak linear power	~30 kW/m
Peak internal clad temperature	465 °C (midplane)
Fluence (spectrum)	2.4 x 10 ²² n/cm ² (Fast, EBR-II)

OTHER FUEL SAMPLE INFORMATION	
Pin tested was PNL-17-34. It had been previously been exposed to a TREAT transient (#1452 – referred to as either test HOP 3-2A or HOP 3-3A) which did not cause the fuel temperatures to exceed usual steady-state operation and or the cladding strain to increase). The estimated as-irradiated fuel region radii (relative to fuel outer radius R) were no central void, columnar grain region r/R=0.38, equiaxed grain region r/R=0.56.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK, initial temperature varying approximately linearly from ~150 °C at the fuel bottom to ~245 °C at the fuel top
Transient conditions	TREAT power rose to 70 MW within 0.5 s, was held at 70 MW for 3.5 s, then rose rapidly, reaching peak power (640 MW) 0.5 s later. The power burst full width at half maximum was ~ 0.33 s. Total TREAT energy was 509 MJ. The power coupling factor was roughly 3 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule; 1.6 mm thick NaK annulus around the fuel pin; surrounded radially by a 7.5 mm thick nickel heat sink
Measurements /diagnostics during test	Temperatures
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	Slight cladding failure.
Posttest measurements & information	Macro-photography, destructive examination and micro-photography
Posttest analyses	Thermal-mechanical analysis of the fuel pin

KEY SUMMARY INFORMATION ABOUT THE TEST

In Test HUT3-2B, the FFTF-like test pin was subjected to a simulated 3 \$/s untermated FFTF reactivity insertion event. The cladding failed as the power was dropping after peak power, more or less circumferentially at the fuel axial midplane. The cladding breach was small and localized. Based on the orientation of the fractures above the breach, it was concluded that significant axial stresses must have been present in the cladding. Data from thermocouples suggested that fuel motion began about 0.15 s after cladding failure. The fuel areal melt fraction at the midplane was ~40%.

Summary of TREAT Test HUT3-5A

(test was originally named HUT3-3A)

Performed 06-December-1974

Transient No. 1614

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt%UO ₂ – 25 wt%PuO ₂ , uranium enriched to 65%; O/M = 1.98.
Geometry	Stack of sintered, solid pellets
Fissile height	34.2 cm
Smear density	85.3% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	54 MWd/kg
Peak linear power	30 kW/m
Peak internal clad temperature	493 °C (midplane)
Fuence (spectrum)	4.1 to 4.2 x 10 ²² n/cm ² (Fast; EBR-II)

OTHER FUEL SAMPLE INFORMATION	
Fuel pin PNL 10-20 was used. The as-irradiated fuel had an estimated central void radius $r/R = 0.06$, a columnar grain region radius = 0.5, and an equiaxed grain region radius = 0.7 (relative to the fuel outer radius).	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK. Initial temperature varied fairly linearly from 187 °C at the fuel bottom to 337 °C at the fuel top.
Transient conditions	TREAT power rose to 40 MW in the first 0.5 s, then held at 40 MW for 5.4 s, then rose on a period of ~0.16 s for 0.47 s (reaching 650 MW), then rounding off to a peak of 710 MW during the next 0.10 s, then scram. The burst full-width-at-half-maximum was 0.22 s. The total TREAT energy was 440 MJ. The power coupling factor was 4.3 to 4.5 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	Gamma scan, profilometry, microphotography, autoradiography, fission gas analysis
Posttest analyses	Thermal-mechanical analyses of the fuel pin

KEY SUMMARY INFORMATION ABOUT THE TEST

Test HUT3-5A subjected an FFTF-like fuel pin to a simulated 3\$/s untermiated reactivity insertion event. The test was complementary to HUT3-5B. Posttest examination showed that no central void had formed and that fuel did not reach the liquidus temperature at $x/L=0.2$ and 0.8. At the midplane, however, the liquidus radius (r/R , relative to the fuel outer radius) was 0.14. The peak cladding strain was 0.71%. No results from the hodoscope are available.

Summary of TREAT Test HUT3-5B

Performed 18-March-1977

Transient No. 1898

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK; (initial temperatures not available)
Form	Mixed U-Pu oxide in cladding	Transient conditions	Probably similar to those in test HUT3-5A. Peak TREAT power was approximately 710 MW. Final TREAT energy was 429 MJ. Power coupling factor was 4.3 to 4.4 W/g-MW.
Composition	75 wt% UO ₂ - 25 wt% PuO ₂ , uranium enriched to 65%, O/M=1.98	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered solid pellets	Test vehicle & experiment system	Static NaK capsule.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures, fuel motion
Smear density	85.7% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	Cladding remained intact.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Cladding profilometry
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	48 MWd/kg		
Peak linear power	26 kW/m		
Peak internal clad temperature	481 °C (midplane)		
Fluence (spectrum)	Fluence unknown (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin PNL 10-42 was used.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
In Test HUT3-5B, an FFTF-like fuel pin was subjected to a simulated 3 \$/s unterminated FFTF reactivity insertion event. The test produced little or no fuel above the fuel liquidus. The measured solidus radius was 1.8+/-0.3 mm at x/L=0.19, 2.05 +/- 0.3 mm at the midplane, and 2.07 +/- 0.3 mm at x/L=0.81. No results from the hodoscope are available.			

Summary of TREAT Test HUT3-6A

Performed 11-August-1977

Transient No. 1924

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK; initial temperature varied from 220 °C at the fuel bottom to 360 °C at the fuel top
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 32 MW within 0.6 s, held at 32 MW for 5.7 s, then rose nearly linearly to 555 MW during the next 0.43 s, then peaked at 585 MW 0.07 s later, then scram. The power burst full width at half maximum was about 0.3 s. The total TREAT energy generated was 378 MJ. The test fuel power coupling factor was about 3.85 W/g-MW at the fuel midplane.
Composition	75 wt%UO ₂ - 25 wt% PuO ₂ , uranium enriched to 93%, O/M=1.98		
Geometry	Stack of solid, sintered pellets		
Fissile height	34.3 cm		
Smear density	91.5% of theoretical	Fuel sample configuration in the test	Single pin
CLADDING		Test vehicle & experiment system	Static NaK capsule, 1.33 mm thick NaK annulus between fuel pin and the surrounding 7 mm thick nickel heat sink.
Material	Type 316 stainless steel, 20% CW	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
OD	5.84 mm	Posttest coolant flow condition	Static (same as initial conditions)
Thickness	0.38 mm	Posttest condition of test sample(s)	Cladding breached.
PRE-IRRADIATION CHARACTERISTICS		Posttest measurements & information	Macro- and microphotography
BU (peak axial)	112 MWd/kg	Posttest analyses	Thermal-mechanical analyses of fuel pin.
Peak linear power	27 to 29 kW/m		
Peak internal clad temperature	483 °C (midplane)		
Fluence (spectrum)	About 6×10^{22} n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin WSA-3-35 was used. Based on sibling pin examination, the fuel equiaxed grain region radius was $r/R=0.4$.			

KEY SUMMARY INFORMATION ABOUT THE TEST

Test HUT3-6A subjected an FFTF-like fuel pin to conditions simulating an FFTF untermated 3 \$/s reactivity insertion event. The cladding failed ~0.72 s into the overpower burst initially at a location $x/L = 0.4$ to 0.6 when the TREAT power was 186 MW. Posttest, six failure sites were found, in the range $x/L=0.25$ to 0.80 , all of which were judged to have occurred between 186 MW on the power rise (when only about 12% of the power burst energy had been generated) and 274 MW as the power was coming back down. No results from the hodoscope are available.

Summary of TREAT Test HUT3-6B

Performed 22-August-1977

Transient No. 1928

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK. Initial temperature varied from 220 °C at the fuel bottom to 360 °C at the fuel top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 32 MW within 0.8 s, held at 32 MW for 5.54 s (to time t=8.94 s), then rose to 80 MW by t=9.1 s, to 147 MW by t=9.2 s, to 255 MW by t=9.35 s, to 585 MW by t=9.52 s, held at 585 MW until t=9.54 s, then dropped to 175 MW by t= 9.62 s, and 0 MW by t=9.69 s. The power burst full width at half maximum was ~0.27 s. Total TREAT energy = 376 MJ. The power coupling factor was ~5.1 W/g-MW.
Composition	75 wt% UO ₂ – 25 wt%PuO ₂ , uranium enriched to 93%, O/M=1.99		
Geometry	Stack of sintered, solid pellets		
Fissile height	34.0 cm		
Smear density	85.6%		
CLADDING		Fuel sample configuration in the test	Single pin
Material	Type 316 stainless steel, 20% CW	Test vehicle & experiment system	Static NaK capsule. The fuel pin was surrounded by a NaK annulus 1.3 mm thick, bounded radially by a 7.8-mm-thick nickel heat sink.
OD	5.84 mm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Thickness	0.38 mm	Posttest coolant flow condition	Static; (same as initial conditions)
PRE-IRRADIATION CHARACTERISTICS		Posttest condition of test sample(s)	Cladding failure
BU (peak axial)	108 MWd/kg	Posttest measurements & information	Examination of the fuel pin
Peak linear power	27 kW/m	Posttest analyses	Thermal-mechanical analysis of fuel pin
Peak internal clad temperature	477 °C (midplane)		
Fluence (spectrum)	About 7×10^{22} n/cm ² (Fast: EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
The WSA3-28 fuel pin had estimated fuel structure radii (relative to the outer fuel radius R) of 0.08 (central void), 0.31 (columnar grain region), and 0.51 (equiaxed grain region.)			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test HUT3-6B subjected the FFTF-like fuel pin to conditions simulating a 3 \$/s untermiated FFTF reactivity insertion event. Cladding failed at t=9.525 s near the bottom of the fuel column (at approximately x/L=0.15). Posttest examination showed the breach extended axially upward to x/L=0.6. No results from the hodoscope are available.

Summary of TREAT Test HUT3-7A

(test was originally named HUT3-5A)

Performed 31-January-1975

Transient No. 1634

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , uranium was enriched to 30%, O/M=1.985
Geometry	Stack of sintered, solid pellets
Fissile height	34.1 cm
Smear density	85.0% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	50 MWd/kg
Peak linear power	19 kW/m
Peak internal clad temperature	463 °C (midplane)
Fluence (spectrum)	5.7×10^{22} n/cm ² (Fast; EBR-II)

OTHER FUEL SAMPLE INFORMATION	
The fuel pin tested was PNL9-45.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK. Initial temperature varied from 225 °C at the fuel bottom to 395 °C at the fuel top
Transient conditions	The TREAT power rose to 34 MW within 0.5 s, held at 34 MW for 5.45 s, then rose almost linearly to 800 MW during the next 0.7 s, then scram. The power burst full width at half maximum was ~0.185 s. The total TREAT energy was 470 MJ, of which 285 MJ was generated during the burst. The power coupling factor was about 3.75 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule; the test pin was within a 1.6 mm thick NaK annulus, surrounded by a 7.5-mm-thick nickel heat sink
Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	Neutron radiography, eddy current measurement of cladding, profilometry, released fission gas analysis, autoradiography, micro-exams.
Posttest analyses	Thermal-mechanical analyses of fuel pin

KEY SUMMARY INFORMATION ABOUT THE TEST

Test HUT3-7A subjected the FFTF-like fuel pin to conditions simulating a 3 \$/s untermiated FFTF reactivity insertion event. Posttest examination of the intact pin resulted in no fuel being identified as having reached the liquidus. The solidus radius at the axial midplane was $r/R = 0.78$ (relative to the fuel outer radius). The diametral strain due to the transient was 0.4 or 0.7%. Eddy current examination showed no evidence of incipient cladding breach. The fission gas collected posttest from the pin was 32% more than collected from a sibling irradiated (but untested) pin. The area of equiaxed grain region in the tested pin was almost the same as in the untested sibling pin. No results from the hodoscope are available.

Summary of TREAT Test HUT3-7B

(test was originally named HUT3-5B)

Performed 04-April-1975

Transient No. 1645

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK. Initial temperature varied from 195 °C at the fuel bottom to 365 °C at the fuel top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 38 MW within 0.5 s, held at 38 MW for 5.65 s, then rose to 126 MW during the next 0.33 s, then to 820 MW during the next 0.47 s, then scram. The power burst full width at half maximum was ~0.73 s. The total TREAT energy generated was 550 MJ. The power coupling factor was ~3.4 W/g-MW.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , uranium enriched to 30%, O/M=1.985	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.35 mm thick NaK annulus, surrounded by a ~7.8-mm-thick nickel heat sink.
Fissile height	34.1 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	85.0% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	The fuel pin cladding failed. Gross failure above the midplane.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Gamma scan of the capsule; macro-examination, micro-examination of the intact section of the pin below the midplane
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	50 MWd/kg		
Peak linear power	19 kW/m		
Peak internal clad temperature	464 °C (midplane)		
Fluence (spectrum)	~5.9 x 10 ²² n/cm ² (Fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin PNL-9-54 was tested.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT3-7B subjected an FFTF-like fuel pin to conditions simulating an FFTF 3 \$/s untermiated reactivity input event. The test was complementary to test HUT3-7A which did not result in cladding breach. In HUT3-7B the cladding failed 0.1 s after scram, in the axial region x/L=0.5 to 0.8. No results from the hodoscope are available.			

Summary of TREAT Test HUT5-1A

(The test name HUT5-1A was first used for a previous test that was later renamed HUT5-2A.)

Performed 13-July-1978

Transient No. 2032

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK (initial temperature not available)
Form	Mixed U-Pu oxide in cladding	Transient conditions	Peak TREAT power approximately 230 MW. Other information not available. Probably similar to power-time history of test HUT5-9A.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. (Details not available but probably similar to those of other HUT tests performed in the late 1970s.)
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	Information not available	Posttest coolant flow condition	Static (same as original condition)
CLADDING		Posttest condition of test sample(s)	Cladding failed
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	(information not available)
OD	5.84 mm	Posttest analyses	(information not available)
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	28 MWd/kg		
Peak linear power	38 kW/m		
Peak internal clad temperature	(information not available)		
Fluence (spectrum)	1.8 X 10 ²² n/cm ² (Fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin tested was P-14-309. Information regarding the dimensions and composition of the fuel and cladding indicated above was not available but are based on the assumption that they were the same as for all of the other HUT tests.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT5-1A subjected the FFTF-like fuel pin to conditions simulating an unterminated 50 ¢/s FFTF reactivity insertion event. Posttest information regarding the state of the fuel pin, and results from the hodoscope, were not available.			

Summary of TREAT Test HUT5-2A

(test was originally named HUT5-1A)

Performed 13-September-1974

Transient No. 1600

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK. Initial temperature varied from 130 °C at the fuel bottom to 270 °C at the fuel top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 52 MW within ~1 s, held at 52 MW for 5.2 s (to 300 MJ energy), then rose to 92 MW during the next 1.3 s, then to 245 MW (peak) during the next 2.2 s (reaching 740 MJ), then scram. Total TREAT energy was 774 MJ. Power coupling factor was ~4.4 W/g-MW.
Composition	75 wt%UO ₂ – 25 wt% PuO ₂ , uranium enriched to 65%, O/M=1.98	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Fissile height	34.4 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	84.9% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	Cladding remained intact.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, fission gas measurement, metallography
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	49 MWd/kg		
Peak linear power	40 kW/m		
Peak internal clad temperature	~510 °C (midplane)		
Fluence (spectrum)	4.1 x 10 ²² n/cm ² (Fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin PNL 11-28 was tested. As-irradiated fuel region outer radii (relative to fuel outer radius R) at the midplane were estimated to be r/R=0.21 for central void, 0.72 for columnar grain region, and 0.79 for equiaxed grain region.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT5-2A subjected the FFTF-like fuel pin to conditions simulating an unterminated 50 ¢/s reactivity insertion event. Test conditions were designed to be similar to those in TREAT Test H4. The posttest cladding strain was 0.2% average and 0.59% peak. The fuel column expanded axially by 10 mm, resulting in a large void near the top of the fuel. The cladding at this elevation showed evidence of overheating and localized deformation. Fuel voids formed along an axial region x/L=0.3 to 0.9 and were especially pronounced in the region x/L=0.4 to 0.7. Fuel melting occurred essentially all the way to the bottom of the fuel column (as far down as the central void extended in the untested sibling pin). At the midplane, the solidus radius was r/R=0.75 (55% of the fuel cross sectional area). The liquidus radius was 0.74 at x/L= 0.2 ; it was 0.73 at the midplane and 0.69 at x/L= 0.8. Based on fission gas release measurements of the as-tested pin and the as-irradiated sibling pin, an estimated 18% of the pretest retained fission gas was released from the fuel during the transient. No results from the hodoscope are available.			

Summary of TREAT Test HUT5-2B

(test was previously named HUT5-1B)

Performed 22-April-1975

Transient No. 1648

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , U enriched to 65%, O/M=1.98
Geometry	Stack of sintered, solid pellets
Fissile height	34.3 cm
Smear density	84.9% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	49 MWd/kg
Peak linear power	39 kW/m
Peak internal clad temperature	510 °C (midplane)
Fluence (spectrum)	3.9 to 4.1 n/cm ² (Fast, EBR-II)

OTHER FUEL SAMPLE INFORMATION	
The as-irradiated pin PNL 11-15 was estimated to have region outer radii of r/R=0.2 for the central void, 0.7 for the columnar grain region, and 0.8 for the equiaxed grain region.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK. The initial temperature varied linearly from 160 °C at the fuel bottom to 320 °C at the fuel top.
Transient conditions	The TREAT power rose to 48 MW within 1 s, was held at 48 MW for ~5.5 s, then rose to 88 MW during the next 1.1 s (reaching 360 MJ energy), then to 175 MW (576 MJ) during the next 1.64 s, then to 285 MW (~800 MJ) about 1.0 s later, then scram. The power coupling factor was ~4.6 to 4.8 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	Cladding breach.
Posttest measurements & information	Neutron radiography, auto-radiography, fission gas collection, macro- and micro-photography
Posttest analyses	Thermal-mechanical analyses of the fuel pin

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HUT5-2B subjected an FFTF-like fuel pin to a simulated FFTF untermiated 50 cents/s reactivity input excursion. The test was a follow-on test to HUT5-2A. In HUT5-2A the test pin came close to incipient cladding breach. Test HUT5-2B was intended to cause breach. The cladding failed when about 3700 J/g had been generated, approximately 6% more energy than had been generated in HUT5-2A. About 12% additional energy was generated after breach. The initial cladding failure occurred in the upper portion of the fuel column. The only fuel pellets that remained intact were those near the bottom of the fuel column. Fuel melted along nearly the entire length of the fuel column. The melt radius r/R was 0.64 just above the fuel bottom (x/L=0.04). Fuel in the top half of the fuel column was extensively dispersed. An estimated 95% of the fission gas present in the as-irradiated fuel was released. No results from the hodoscope are available.	

Summary of TREAT Test HUT5-5A

(test was originally named HUT5-3A)

Performed 03-June-1974

Transient No. 1577

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK; initial temperature varied axially from 165 °C at the fuel bottom to 320 °C at the fuel top
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 50 MW within 0.5 s, held at 50 MW for 4.9 s, then rose to 73 MW during next 1.4 s, then rose to 240 MW during next 1.9 s (attaining 650 MJ TREAT energy), then scram. Power coupling factor was 4.7 to 4.9 W/g-MW.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , U enriched to 65%, O/M=1.98	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Fissile height	34.0 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	85.6% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	Cladding failed
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, autoradiography, fission gas release measurement, metallurgical examination
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	52 MWd/kg		
Peak linear power	29 kW/m		
Peak internal clad temperature	490 °C (midplane)		
Fluence (spectrum)	3.9 to 4.0 x10 ²² n/cm ² (Fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin PNL 10-17 was tested. At the axial midplane of the as-irradiated fuel, relative to the fuel outer radius R, the central void extended to r/R=0.09, the columnar grain region extended to r/R=0.53, and the equiaxed grain region extended to r/R=0.69.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT5-5A subjected the fuel pin to a simulated FFTF untermiated 50 cents/s reactivity excursion. The cladding failed approximately 0.12 s before the TREAT reactor scrammed. Posttest examination showed that the cladding melted along a line from x/L=0.15 to 0.80 with the breach extending azimuthally 90 degrees. Approximately 78% of the fission gas present in the fuel before the transient was released during the transient. No results from the hodoscope are available. The test is related to TREAT Test H6 (also a 50 ¢/s reactivity insertion on irradiated PNL-10 pins), to Test H4 (on similar pins irradiated at 32 kW/m to 41 MWd/kg), and to Test H5 (on similar pins irradiated at 26 kW/m to 31 MWd/kg).			

Summary of TREAT Test HUT5-5B

Performed 18-April-1977

Transient No. 1903

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide, in cladding
Composition	75 wt% UO ₂ - 25 wt% PuO ₂ , 65 % enriched U; O/M=1.96
Geometry	Stack of sintered, solid pellets
Fissile height	34.6 cm
Smear density	84.9% of theoretical
CLADDING	
Material	Type 316 stainless steel, 20%CW
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	49 MWd/kg
Peak linear power	27 kW/m
Peak internal clad temperature	480 °C (midplane)
Fluence (spectrum)	3.7 x10 ²² n/cm ²
OTHER FUEL SAMPLE INFORMATION	
Fuel pin tested was PNL 10-72. As-irradiated fuel zone outer radii (r/R) at the midplane were: central void= 0.09, columnar grain region = 0.50, equiaxed grain region = 0.67.	
TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK. Initial temperature varied linearly from 200 °C at fuel bottom to 345 °C at top.
Transient conditions	TREAT power rose to 40 MW within 1 s, then held at 40 MW for 5.1 s, then rose to 52 MW during the next 0.85 s, then rose linearly to 161 MW during the next 3.0 s, then dropped to ~zero during the next 0.18 s. Based on this power-time history, the total TREAT energy was approximately 580 MJ. The power coupling factor was 4.5 to 4.7 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule probably similar to the others in this test series.
Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	No cladding breach.
Posttest measurements & information	Micro-examination, fission gas measurement
Posttest analyses	Thermal-mechanical analyses of the fuel pin
KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HUT5-5B subjected the FFTF-like fuel pin to a simulated untermiated 50 cents/s FFTF reactivity insertion. The cladding remained intact. Posttest measurements of the fuel central void and liquidus radii (relative to the fuel outer radius R) were zero and 0.48 at x/L= 0.2; they were 0.22 and 0.53 at the midplane, and they were 0.28 and 0.51 at x/L=0.8. An estimated 23% of the pre-test fission gas inventory in the fuel was released during the transient. No results from the hodoscope are available.	

Summary of TREAT Test HUT5-7A

(test was originally named HUT5-5A)

Performed 19-July-1974

Transient No. 1592

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Stagnant NaK with initial axial temperature profile increasing linearly from 200 °C at fuel bottom to 335 °C at fuel top
Form	Mixed U-Pu oxide pellets in cladding	Transient conditions	TREAT power rose to 46 MW in 0.5 s, then held at 46 MW for 5.8 s, then accelerated to 73 MW during next 1.1 s, then to 212 MW during next 2.4 s (reaching 690 MJ TREAT energy), then scram. Power coupling factor = 3.7 to 4.0 W/g-MW.
Composition	75 wt%UO ₂ – 25 wt%PuO ₂ , U enriched to 30%, O/M=1.97	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Fissile height	34.4 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	85.2% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	No cladding breach.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Neutron radiography, total gamma scan, profilometry, micro-exam
OD	5.84 mm	Posttest analyses	Thermal-mechanical analyses of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	47 MWd/kg		
Peak linear power	18 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	5.2 to 5.6 x 10 ²² n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
The pin tested was PNL 9-25. Based on sibling pin measurements, the as-irradiated test pin fuel had no central void or columnar grain regions at the midplane and an equiaxed grain region comprising ~32% of the cross sectional area.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test HUT5-7A, conditions were designed to be similar to those of TREAT Test H3. The measured peak cladding strain was 0.1-0.2% The posttest measured central void and liquidus radii (relative to fuel outer radius R) were 0.13 and 0.61 at x/L = 0.2; they were 0.15 and 0.55 at x/L=0.5, and they were zero and 0.59 at x/L=0.8. An estimated 35% of the fission gas present after the pre-irradiation in EBR-II was released during the transient. No results from the hodoscope are available.

Summary of TREAT Test HUT5-7B

(test was originally named HUT5-5B)

Performed 04-Feb-1975

Transient No. 1637

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt%PuO ₂ , 30% enriched U, O/M=1.97
Geometry	Stack of sintered, solid pellets
Fissile height	34.2 cm
Smear density	85.2% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS

BU (peak axial)	47 MWd/kg
Peak linear power	18 kW/m
Peak internal clad temperature	456 °C (midplane)
Fluence (spectrum)	5.3 to 5.6 x10 ²² n/cm ² (Fast spectrum; EBR-II)

OTHER FUEL SAMPLE INFORMATION

Pin tested was PNL9-34. Sibling pin had no central void or columnar grain region but had an equiaxed grain region extending to r/R=0.56.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK with initial temperature varying linearly from 235 °C at fuel bottom to 395 °C at fuel top.
Transient conditions	Power rose to 35 MW (TREAT) within 0.5 s, held at 35 MW for 5.5 s, gradually rose to 87 MW during next 2.0 s, then ~ linear rise to 290 MW during next 2.9 s (870 MJ generated by that time), then scram. Power coupling factor was 3.5-3.8 W/g-MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule. The test pin was within a 1.33 mm thick NaK annulus, surrounded by a 7.8-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	The fuel pin was breached.
Posttest measurements & information	Neutron radiography, autoradiography, fission gas analysis, microexamination
Posttest analyses	Fuel pin thermal-mechanical analysis

KEY SUMMARY INFORMATION ABOUT THE TEST

Test HUT5-7B subjected a single FFTF-like fuel pin to a simulated FFTF untermiated 50 cent/s overpower excursion. The cladding failed at about 4.7 to 4.8 seconds into the overpower rise at an axial location of x/L= 0.4 to 0.8. About 26% of the fission gas present in the fuel prior to the transient was released during the transient. No results from the hodoscope are available.

Summary of TREAT Test HUT5-9A

Performed 12-June-1979

Transient No. 2133

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK (initial temperature not available)
Form	Mixed U-Pu oxide in cladding	Transient conditions	Actual conditions not available. Planned transient was rise to 47 MW within ~0.5 s, hold at 47 MW for 6.45 s, then rise to 324 MW peak during next 3.85 s, then scram, generating 930 MJ total TREAT energy. Power coupling factor not available.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Stack of sintered, solid pellets	Test vehicle & experiment system	Static NaK capsule. (Details not available but probably similar to those of other HUT tests performed in the late 1970s.)
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures, fuel motion (hodoscope)
Smear density	(Information not available)	Posttest coolant flow condition	Static (same as original condition)
CLADDING		Posttest condition of test sample(s)	(information not available)
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	(information not available)
OD	5.84 mm	Posttest analyses	(information not available)
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	6 MWd/kg		
Peak linear power	14 kW/m		
Peak internal clad temperature	(information not available)		
Fluence (spectrum)	1.2 X 10 ²² n/cm ² (Fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin tested was P-15-77D or P-15-74. Information regarding the dimensions and composition of the fuel and cladding indicated above was not available but are based on the assumption that they were the same as for all of the other HUT tests. Based on the test naming system, the cladding may have breached and the fission gas released prior to being tested in TREAT.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HUT5-9A subjected the FFTF-like fuel pin to conditions simulating an unterminated 50 ¢/s FFTF reactivity insertion event. Posttest information regarding the state of the fuel pin, and results from the hodoscope, are not available.			

Summary of TREAT Test HUT-L2

Performed 28-October-1976

Transient No. 1859

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , uranium enriched to 65% O/M=1.97 for pins HUT-L2-1 and L2-2; O/M=1.94 for pin HUT-L2-4
Geometry	Stack of sintered, solid pellets
Fissile height	91.5 cm
Smear density	85.8% of theoretical for HUT-L2-1 85.8% of theoretical for HUT-L2-2 88.7% of theoretical for HUT-L2-4
CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	Zero
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(not applicable)
OTHER FUEL SAMPLE INFORMATION	
(none)	
TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium initially at 313 °C. The initial velocity in the flowtube occupied by pin HUT-L2-4 was 4.6 m/s.
Transient conditions	TREAT power rose to 102 MW within ~1 s, then held at ~100 MW for 4.6 s (reaching 466 MJ), then rose nearly linearly to peak power of 540 MW in 2.3 s (reaching ~1200 MJ), then dropped to zero power during final 0.2 s. Total TREAT energy was 1260 MJ. The power coupling factor was ~1.9 to 2.0 W/g-MW in pin HUT-L2-1, 1.7 to 1.8 W/g-MW in pin HUT-L2-2, and ~1.8 in pin HUT-L2-4.
Fuel sample configuration in the test	Three fuel pins in separate, parallel flow tubes.
Test vehicle & experiment system	Mark-II C sodium loop, with boron filtering and axial flux-shaping collars. Flow tubes were type 304 stainless steel, 8.1 mm ID, 0.71 mm thick
Measurements /diagnostics during test	Flowtube temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Same as pre-test
Posttest condition of test sample(s)	No cladding failures. Fuel melting and small cladding strain in one pin.
Posttest measurements & information	Cladding profilometry, examination of fuel cracks, fuel porosity, radial gaps, and fuel areal melt fractions
Posttest analyses	Thermal-mechanical analyses of the fuel pins.
KEY SUMMARY INFORMATION ABOUT THE TEST	
The objectives of Test HUT-L2 were to obtain measurable cladding deformation and fuel melting in all three FFTF prototypic-length pins in a simulated untermated FFTF 50 cent/s reactivity insertion event. Fuel melting did occur in each of the pins. Cladding deformation was ~0.2% in the high-smear-density pin HUT-L2-4 and only ~0.03% in the other two pins. The maximum fuel areal fraction bounded by the solidus temperature was 35% in pin HUT-L2-1 and 26% in both HUT-L2-2 and HUT-L2-4. There was a porosity decrease in the fuel at about the fuel centerline of the three pins. At the axial locations of minimum post-test radial fuel-cladding gaps, the transient caused the gap to decrease from 65 microns to 28 microns at x/L= 0.55 in pin HUT-L2-1; from 64 to 39 microns at x/L= 0.5 in HUT-L2-2, and from 30 to 11 microns at x/L=0.75 in HUT-L2-4. Results from the hodoscope are not available.	

Appendix V: One-page Summaries of HOP/HUC-series Tests

One-page summaries of each of the HOP/HUC-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test HOP 1-6A

Performed 6-October-1977 (1st transient) and 10-October-1977 (2nd transient)
Transient Nos. 1943 and 1944

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ - 25 wt% PuO ₂ , 93% enriched U, and O/M= 1.98-1.99
Geometry	Stack of solid pellets in cladding
Fissile height	34.3 cm
Smear density	85 to 90% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 cm
Thickness	0.38 cm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	110 to 114 MWd/kg
Peak linear power	28 to 34 kW/m
Peak internal clad temperature	(information not available)
Fluence (spectrum)	6.7 x 10 ²² n/cm ² (Fast; EBR-II)

OTHER FUEL SAMPLE INFORMATION	
Pins WSA-3-27, WSA-3-25, and WSA-3-8 were used in the test. The as-fabricated fuel-cladding gap in pin WSA-3-27 was ~15% less than the lower bound of the FFTF cladding gap design envelope. Also pin WSA-3-27 had experienced an unusually-long out-of-reactor period prior to being transient tested.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Sodium flow of about 104 g/s during the first transient and about 118 g/s during the second transient. The initial temperature was about 370 °C in both tests.
Transient conditions	Two transients were run. In each, the TREAT power rose to 45 MW within ~1 s, held at 45 MW for 5.7 s, then rose approximately linearly to 57 MW during the next 8.2 s, then scram. The total TREAT energy was 704 MJ for each transient. The TREAT to test fuel power coupling was in the range 3.8 to 4.2 W/g-MW for pin WSA-3-27 and 3.4 to 3.7 W/g-MW for the other two pins.
Fuel sample configuration in the test	Three pins in parallel, each pin in its separate flow tube. The flowtubes were 8.14 mm in ID and 0.7 mm thick and surrounded by inert gas.
Test vehicle & experiment system	Mark-II C integral sodium loop with boron neutron filtering.
Measurements /diagnostics during test	Sodium flow rate, pressure, and temperatures.
Posttest coolant flow condition	Flow rate was essentially unchanged from initial value.
Posttest condition of test sample(s)	The cladding of pin WSA-3-27 was breached. The other two pins remained intact.
Posttest measurements & information	Cladding profilometry, gamma scanning, and microphotography of samples.
Posttest analyses	Thermal analysis of test, and thermal-mechanical analysis of fuel pins

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HOP 1-6A subjected a high-burnup FFTF-like fuel pin to two transients, each of which simulated conditions of an FFTF/CRBR 3 c/s reactivity insertion event terminated by the plant protective system (PPS). The actual coolant flow rate during the transients was about 10 to 20% lower than desired. The power level in pin WSA-3-27 during the constant-power portion of the TREAT transient was 12% higher than the pin experienced during its pre-irradiation in EBR-II. Pin WSA-3-27 incurred a pin-hole size cladding breach at the fuel midplane. It is unknown whether the breach occurred during the first transient or during the second. The cladding breach in that pin was the only observed change in any of the test pins. Examination indicated that no measurable cladding swelling, fuel melting, or fuel relocation occurred. A slow pin failure as occurred in this test had not been observed in any previous TREAT test and indicated that high fuel burnup can significantly reduce the cladding failure threshold, causing the cladding to be at or near failure during this 3 c/s PPS-terminated event.	

Summary of TREAT Test HOP 3-1A

Performed 24-May-1973

Transient No. 1513

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂
Geometry	Stack of solid pellet in cladding
Fissile height	34.3 cm
Smear density	85.75% of theoretical
CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	7.5 MWd/kg
Peak linear power	34 kW/m
Peak internal clad temperature	(information not available)
Fluence (spectrum)	0.8 x 10 ²² n/cm ² (Fast; EBR-II)
OTHER FUEL SAMPLE INFORMATION	
Fuel pin ANL-A-EW-13 was used in the test.	
TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK. Initial temperatures ~150 °C at fuel bottom, ~225 °C at midplane, and ~305 °C at fuel top.
Transient conditions	TREAT power rose to ~68 MW within ~1 s, held at ~68 MW for 3.2 s, then rose sharply to 225 MW during the next 0.3 s, then at 255 MW for 0.04 s, then scram. The total TREAT energy was 347 MJ. The TREAT to test fuel power coupling was ~4.3 W/g per MW.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule with boron neutron filtering. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures near the NaK annulus and in the surrounding nickel heat sink. Fuel motion monitoring using the TREAT hodoscope.
Posttest coolant flow condition	Static (same as initial conditions)
Posttest condition of test sample(s)	The cladding remained intact.
Posttest measurements & information	Limited microexamination
Posttest analyses	Thermal-mechanical analysis of the fuel pin.

KEY SUMMARY INFORMATION ABOUT THE TEST
Test HOP 3-1A subjected an FFTF-like fuel pin to conditions representing a 3 \$/s reactivity insertion event in FFTF/CRBR terminated by the plant protection system. Posttest examination of a midplane sample showed no fuel had reached the liquidus, the central void radius was 0.57 mm, and the fuel-cladding radial gap was 0.05 mm. No measurable transient-induced cladding strain occurred.

Summary of TREAT Test HOP 3-1B

(test was originally named HOP 3-2A)

Performed 31-January-1974

Transient No. 1542

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK initially at ~160 °C at the fuel bottom, 245 °C at the midplane, and 400 °C at the top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 72 MW within 0.7 s, held at 72 MW for 5.6 s, then rose sharply to 355 MW during the next 0.32 s, held at 355 MW for 0.06 s, then scram. The total TREAT energy was 545 MJ. The TREAT to test fuel power coupling was ~4.6 W/g-MW.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Stack of solid pellet in cladding	Test vehicle & experiment system	Static NaK capsule with boron neutron filters. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures near the NaK annulus and surrounding nickel heat sink. Fuel motion monitoring using the TREAT hodoscope.
Smear density	87% of theoretical	Posttest coolant flow condition	Static (same as pre-test)
CLADDING		Posttest condition of test sample(s)	No cladding breach.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Micro-examination
OD	5.84 mm	Posttest analyses	Thermal-mechanical analysis of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	24.5 MWd/kg		
Peak linear power	42 kW/m		
Peak internal clad temperature	580 °C		
Fluence (spectrum)	1.9 x 10 ²² n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
The pin tested was P-23A-27.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HOP 3-1B was intended to subject an FFTF-like fuel pin to conditions simulating an FFTF/CRBR 3 \$/s reactivity insertion event terminated by the plant protection system, but apparently 25% more energy was generated in the test fuel than was planned. Posttest examination shows that, at the fuel axial midplane, the fuel-cladding gap was 0.02 mm and a central void of radius 0.4 mm had formed in the fuel but none of the fuel had reached the liquidus temperature, although there was incipient fuel melting. The peak cladding strain was 0.3 to 0.4%.			

Summary of TREAT Test HOP 3-2A

(test was originally named HOP 3-3A)

Performed 25-May-1972

Transient No. 1452

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂
Geometry	Stack of solid pellet in cladding
Fissile height	34.3 cm
Smear density	88.8% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	30 MWd/kg
Peak linear power	33 kW/m
Peak internal clad temperature	(information not available)
Fluence (spectrum)	2.4 x 10 ²² n/cm ² (Fast; EBR-II)

OTHER FUEL SAMPLE INFORMATION	
The fuel pin tested was PNL-17-34.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK, initially at 130 °C along the entire fuel column length
Transient conditions	TREAT reactor power rose to 27 MW within 2 s, was held at 27 MW for 5.85 s, then rose to 69 MW peak power during the next 0.54 s, then scram. TREAT energy generated was 160 MJ during the power hold, then an additional 24 MJ during the power rise and 6 MJ more after peak power.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule without neutron filtering. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures near the NaK annulus and in the surrounding nickel heat sink. Fuel motion monitoring using the TREAT hodoscope.
Posttest coolant flow condition	No flow (same as pre-test)
Posttest condition of test sample(s)	Essentially no change from pre-test conditions.
Posttest measurements & information	None
Posttest analyses	Thermal-mechanical analysis of the fuel pin.

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HOP 3-2A simulated a 3\$/s reactivity insertion terminated by the plant protection system, relevant to the FFTF and CRBR reactors. Because the transient was intended to cause at most limited fuel-pin damage, and because the power coupling between the fuel pin and the reactor was determined after the test to have been about 7% below the anticipated value, the posttest computations indicated that the maximum fuel pin centerline temperature during the transient did not exceed the pin's steady-state operating conditions during its pre-irradiation in EBR-II, and that no additional cladding strain resulted from the test.	

Summary of TREAT Test HOP 3-2B (test was originally named HOP 3-3B)

Performed 22-May-1973

Transient No. 1511

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK; initially at 105 °C at fuel bottom, 165 °C at midplane, and 235 °C at fuel top
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 75 MW within 1 s, then held at 75 MW for 4.0 s, then sharp rise to 220 MW during the next 0.29 s, held at 220 MW for 0.05 s, then scram. Total of 401 MJ of TREAT energy. Test fuel power coupling was ~3.8 W/g-MW of TREAT power.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ , O/M = 1.96	Fuel sample configuration in the test	Single pin
Geometry	Stack of solid pellets	Test vehicle & experiment system	Static NaK capsule with boron neutron filters. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Fissile height	34.2 cm	Measurements /diagnostics during test	Temperatures near the NaK annulus and in the surrounding nickel heat sink. Fuel motion monitoring using the TREAT hodoscope.
Smear density	88.8% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	No cladding breach and no cladding strain
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Gamma scan, profilometry, limited micro-examination
OD	5.84 mm	Posttest analyses	(information not available)
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	33 MWd/kg		
Peak linear power	31 kW/m		
Peak internal clad temperature	470 °C		
Fluence (spectrum)	2.6 x 10 ²² n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Fuel pin tested was PNL-17-21.			

KEY SUMMARY INFORMATION ABOUT THE TEST
Test HOP 3-2B subjected an FFTF-like pre-irradiated fuel pin to conditions representing a 3 \$/s reactivity insertion event in FFTF/CRBR terminated by the plant protection system. There was little or no change in the fuel pin due to the test. At the midplane, there was no fuel melting, and the fuel-cladding gap was 0.018 mm.

Summary of TREAT Test HOP 3-2C

(test was originally named HOP 3-2B)

Performed 01-February-1974

Transient No. 1543

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK initially at ~180 °C at the fuel bottom, 270 °C at the midplane, and 315 °C at the top.
Form	Mixed U-Pu oxide in cladding	Transient conditions	TREAT power rose to 68 MW within 0.6 s, held at 68 MW for 5.1 s, then rose sharply to 290 MW during the next 0.34 s, dropped to 250 MW during the next 0.10 s, then scram. The total TREAT energy was 465 MJ. The TREAT to test fuel power coupling was ~4.6 W/g-MW.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Stack of solid pellet in cladding	Test vehicle & experiment system	Static NaK capsule with boron neutron filtering. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures near the NaK annulus and surrounding nickel heat sink. Fuel motion monitoring using the TREAT hodoscope.
Smear density	87% of theoretical	Posttest coolant flow condition	Static (same as initial conditions)
CLADDING		Posttest condition of test sample(s)	No cladding breach.
Material	Type 316 stainless steel, 20% CW	Posttest measurements & information	Micro-examination
OD	5.84 mm	Posttest analyses	Thermal-mechanical analysis of the fuel pin.
Thickness	0.38 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	45 to 49 MWd/kg		
Peak linear power	41 kW/m		
Peak internal clad temperature	(information not available)		
Fluence (spectrum)	3.8 x 10 ²² n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
The pin tested was P-23A-30.			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test HOP 3-2-C was intended to subject an FFTF-like fuel pin to conditions simulating an FFTF/CRBR 3 \$/s reactivity insertion event terminated by the plant protection system, but apparently 25% more energy was generated in the test fuel than was planned. Posttest examination shows that none of the fuel had reached the liquidus. The fuel central void radius and fuel-cladding gap were 0.46 mm and 0.024 mm, respectively, at x/L= 0.2; they were 0.56 mm and 0.014 mm at the midplane (x/L=0.5), and they were 0.38 mm and 0.024 mm at the fuel top.			

Summary of TREAT Test HOP 3-2D

(test was originally named HOP 3-3C)

Performed 25-July-1972
Transient No. 1518

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂
Geometry	Stack of solid pellet in cladding
Fissile height	34.3 cm
Smear density	88% of theoretical

CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	30 MWd/kg
Peak linear power	33 kW/m
Peak internal clad temperature	(information not available)
Fluence (spectrum)	2.4 x 10 ²² n/cm ² (Fast; EBR-II)

OTHER FUEL SAMPLE INFORMATION	
The fuel pin tested was PNL-17-34. It had been previously exposed to a mild transient in TREAT (test HOP 3-2A, transient #1452) that did not cause peak fuel pin temperature to exceed the temperature during the pin's pre-irradiation and did not cause any additional cladding strain.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK initially with an axial temperature gradient from ~140 °C at the fuel bottom to ~230 °C at the fuel top
Transient conditions	TREAT reactor power rose to 66 MW within 0.6 s, was held at 66 MW for ~3.55 s, then rose to 640 MW peak power during the next 0.7 s, then scram. TREAT energy generated was ~250 MJ at the end of "flattop" power hold, then an additional 190 MJ during the power rise, and 70 MJ more after peak power.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Static NaK capsule with neutron filtering. The pin was surrounded by a 1.6-mm NaK annulus bounded by a 15-mm-thick nickel heat sink.
Measurements /diagnostics during test	Temperatures near the NaK annulus and in the surrounding nickel heat sink. (Fuel motion data were collected but were not able to be analyzed posttest.)
Posttest coolant flow condition	No flow (same as pre-test)
Posttest condition of test sample(s)	Localized cladding failures with some fuel escape from the cladding.
Posttest measurements & information	Neutron radiography, gamma scanning, posttest destructive examination including SEM.
Posttest analyses	Thermal-mechanical analysis of the fuel pin.

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HOP 3-2D simulated a 3 \$/s reactivity insertion not terminated by the plant protection system, relevant to the FFTF and CRBR reactors. The fuel pin cladding failed, and some fuel was expelled beyond both ends of the fuel column. A high concentration of fuel was found near the axial center of the pin. The pin had become severed at that location, but most of the cladding was still intact elsewhere. Along the cladding from that point to about 4 cm below that elevation were localized breaches where the cladding had melted through but fuel was still present inside the cladding. The posttest fuel condition was consistent with rapid liberation of fission gas in the hottest columnar and equiaxed grain regions and significant central void formation by molten fuel having been expelled from that location into the NaK coolant (with subsequent local voiding of the coolant). The void areal fraction was about 40%. The SEM examination indicated that the central region of the sample failed by intergranular fracture of the cladding.	

Summary of TREAT Test HOP-PTO 1-2A

Performed (9-11)-June-1976
Transient Nos. 1794 through 1797

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Flowing sodium, initially at 375 °C, held constant during the transient.
Form	Mixed U-Pu oxide in cladding	Transient conditions	Four nearly-identical transients were performed. In each transient, the TREAT power was quickly raised to ~60 MW, held at ~60 MW for ~5 s and then rose linearly for ~9 s peaking at ~77 MW and then scram. The total TREAT energy release was ~950 MJ. The test fuel to TREAT power coupling was 3.8 W/g-MW for the hottest pin and 3.4 W/g-MW for the two cooler pins.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂		
Geometry	Stack of solid pellet in cladding	Fuel sample configuration in the test	Three pins in separate, parallel Type 304 stainless steel flow tubes of 9.3 mm ID and 0.9 mm thickness
Fissile height	34.3 cm	Test vehicle & experiment system	Mark-II C sodium loop with boron neutron filtering
Smear density	~87% of theoretical	Measurements /diagnostics during test	Temperatures of the flowtubes and sodium, flow rate of the sodium
CLADDING		Posttest coolant flow condition	Same as pre-test.
Material	Type 316 stainless steel, 20% CW	Posttest condition of test sample(s)	There was no discernable change in any of the pins relative to their pre-transient condition..
OD	5.84 mm	Posttest measurements & information	Neutron radiography, gamma scanning, cladding strain testing.
Thickness	0.38 mm	Posttest analyses	Thermal-mechanical analysis of the fuel pins.
PRE-IRRADIATION CHARACTERISTICS		KEY SUMMARY INFORMATION ABOUT THE TEST	
BU (peak axial)	31 MWd/kg	Test HOP-PTO 1-2A subjected the three pins to four nearly identical transients, each simulating a 3 ¢/s FFTF reactivity insertion limited by the FFTF plant protective system. After the test, samples from a fuel pin were subjected out-of-pile to internal pressure and temperature transients similar to conditions the pin had experienced during the TREAT test. The observed sample behavior of the TREAT-tested cladding was found to be generally similar to results from as-irradiated specimens that had not been subjected to the TREAT test.	
Peak linear power	~35 kW/m		
Peak internal clad temperature	~565 °C	After the TREAT test, one or two of the fuel pins were further irradiated to ~120 MWd/kg in EBR-II without a cladding breach.	
Fluence (spectrum)	2.6 to 2.9 x 10 ²² n/cm ² (Fast; EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
Pin #1 was P-23C-52A; pin #2 was P-23B-22A; pin #3 was P-23C-17C. The predicted as-irradiated fuel structure radii (r/R) were ~0.20 for the central void, 0.75 for the columnar grain region, and 0.79 for the equiaxed grain region.			

Summary of TREAT HOP-PTO 3-2E

(The test may also have been designated HOP-PTO 3-2C)

Performed 24-June-1976
Transient Nos. 1798 through 1800

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂
Geometry	Stack of solid pellet in cladding
Fissile height	34.3 cm
Smear density	~87% of theoretical
CLADDING	
Material	Type 316 stainless steel, 20% CW
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	~31 MWd/kg
Peak linear power	~34-35 kW/m
Peak internal clad temperature	~565 °C
Fluence (spectrum)	2.6 to 2.9 x 10 ²² n/cm ² (Fast; EBR-II)
OTHER FUEL SAMPLE INFORMATION	
Pin #1 was P-23C-53C, pin #2 was P-23B-44B, and pin #3 was P-23C-20A. The predicted as-irradiated fuel structure radii (r/R) were 0.20 for the central void, 0.74 for the columnar region, and 0.80 for the equiaxed grain region.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Sodium flowing at a constant rate.
Transient conditions	Three transients were performed. During each of the first two transients, the TREAT power was quickly raised to ~60 MW, held at ~60 MW for ~5 s and then rose linearly for ~9 s peaking at ~77 MW and then scram. The total TREAT energy release was ~950 MJ. During the third transient, the power rose quickly to 50 MW, held at 50 MW for 5 s, then raised to a peak power of ~430 MW during the next 0.4 s, then scram. The total TREAT energy during the last transient was 460 MJ. The test fuel to TREAT power coupling was 3.8 W/g-MW for the hottest pin and 3.4 W/g-MW for the two cooler pins.
Fuel sample configuration in the test	Three pins in separate, parallel Type 304 stainless steel flow tubes of 9.3 mm ID and 0.9 mm thickness
Test vehicle & experiment system	Mark-II C sodium loop with boron neutron filtering
Measurements /diagnostics during test	Temperatures of the flowtubes and sodium, flow rate of the sodium
Posttest coolant flow condition	Same as pretest.
Posttest condition of test sample(s)	There was no discernable change in any of the pins relative to their pre-transient condition.
Posttest measurements & information	Gamma scanning, non-destructive examination, destructive examination of selected pin(s)
Posttest analyses	Thermal-mechanical analysis of the fuel pin.

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HOP-PTO 3-2E subjected three pre-irradiated FFTF-like fuel pins to three transients. The first two transients simulated FFTF 3 ¢/s reactivity insertions limited by the FFTF plant protective system. The third transient simulated conditions representative of the maximum allowed by the FFTF plant protective system during a 3 \$/s initiated overpower transient. (After the TREAT test, fuel pin(s) from the test were further irradiated to ~120 MWd/kg in EBR-II without a cladding breach.)	

Summary of TREAT Test HUC-PTO 2-2A

(The test was also called HOP-PTO 2-2A.)

Performed (8-12)-July-1976

Transient Nos. 1806 through 1809

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed U-Pu oxide in cladding
Composition	75 wt% UO ₂ – 25 wt% PuO ₂
Geometry	Stack of solid pellet in cladding
Fissile height	34.3 cm
Smear density	~87% of theoretical
CLADDING	
Material	Type 316 stainless steel
OD	5.84 mm
Thickness	0.38 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	~31 MWd/kg
Peak linear power	~35 kW/m
Peak internal clad temperature	~560 °C
Fluence (spectrum)	2.6 to 2.9 x 10 ²² n/cm ² (Fast; EBR-II)
OTHER FUEL SAMPLE INFORMATION	
Pin #1 was P-23C-51C; pin #2 was P-23B-23A; pin #3 was P-23C-18A. The predicted as-irradiated fuel structure radii (r/R) were 0.20 for the central void, 0.75 for the columnar region, and 0.79 for the equiaxed grain region.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium initially at 370 °C. The flow rate was steady during the first three power transients. During the fourth power transient, the flow rate was continuously reduced from 100% to 58% of the initial flow rate, beginning when the TREAT energy release had reached about 350 MJ.
Transient conditions	The fuel pins were subjected to four transients. During each of the first three transients, the TREAT power rose quickly to ~60 MW, was held at ~60 MW for ~5 s, then rose linearly for ~9 s peaking at ~77 MW, then scram. The total TREAT energy release was ~950 MJ. During the fourth transient, the power rose to 63 MW, then gradually decreased to 61 MW during the following 18 s, and then a reactor scram. The total TREAT energy in the final transient was 1210 MJ. The test fuel to TREAT power coupling was 3.8 W/g-MW for the hottest pin and 3.4 W/g-MW for the two cooler pins..
Fuel sample configuration in the test	Three pins in separate, parallel Type 304 stainless steel flow tubes of 9.3 mm ID and 0.9 mm thickness
Test vehicle & experiment system	Mark-II C sodium loop with boron neutron filtering
Measurements /diagnostics during test	Temperatures of the flowtubes and sodium, flow rate of the sodium
Posttest coolant flow condition	Same as pre-test.
Posttest condition of test sample(s)	There was no discernable change in any of the pins relative to their pre-transient condition.
Posttest measurements & information	Gamma scanning, non-destructive examination, destructive examination of selected pin(s)
Posttest analyses	Thermal-mechanical analysis of the fuel pins.

KEY SUMMARY INFORMATION ABOUT THE TEST	
Test HUC-PTO 2-2A subjected three pre-irradiated fuel pins to four transients. The first three transients were simulations of 3 ¢/s FFTF reactivity insertion events terminated by the plant protection system. The fourth transient was a simulated FFTF loss-of-flow continuous-flow-reduction design transient event. After the TREAT test, fuel pin(s) from the test were further irradiated to ~120 MWd/kg in EBR-II without a cladding breach.	

Appendix W: One-page Summaries of PNL-series Tests

One-page summaries of each of the PNL-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test PNL 1-1

Performed (probably between 1970 and 1974)

Transient No. (unknown)

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Cladded mixed-oxide fuel pin
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U
Geometry	Pellet column, solid pellets
Fissile height	34.3 cm
Smear density	86.6% of theoretical

CLADDING	
Material	Type 304 stainless steel, solution annealed
OD	6.35 mm
Thickness	0.406 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	8.20 MWd/kg
Peak linear power	28 kW/m
Peak internal clad temperature	(unavailable)
Fluence (spectrum)	Fluence unavailable; Fast (EBR-II) spectrum

OTHER FUEL SAMPLE INFORMATION
The fuel pin was PNL 1-1. The as-irradiated fuel had a central void ~0.4 mm diameter and a columnar grain region of ~2.0 mm outer diameter. About 8% of the fission gas generated during the steady-state irradiation was released during that irradiation.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	Transient overpower, probably using a natural TREAT transient
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Posttest coolant flow condition	None (same as initially)
Posttest condition of test sample(s)	Gross failure
Posttest measurements & information	(unavailable)
Posttest analyses	Fuel pin thermal-mechanical analysis.

KEY SUMMARY INFORMATION ABOUT THE TEST

80% of the fuel melted during the transient.

Summary of TREAT Test PNL-1-2

Performed (probably in 1970 or 1971)

Transient No. (unknown)

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Mixed oxide pellets
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U.
Geometry	Stack of solid pellets in cladding
Fissile height	34.3 cm
Smear density	87.5% of theoretical

CLADDING	
Material	Type 304 stainless steel
OD	6.35 mm
Thickness	0.406 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	~8 MWd/kg
Peak linear power	~ 29 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fast spectrum (EBR-II)

OTHER FUEL SAMPLE INFORMATION

The fuel pin was PNL 1-2. The as-irradiated fuel had a central void ~0.4 mm diameter, a columnar grain region of 2.0 mm outer diameter. About 8% of the fission gas generated during the steady-state irradiation was released during that irradiation.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK initially at 150 °C
Transient conditions	"Natural" TREAT transient reaching 124 MW peak power and having fuel-width-at-half-maximum of 0.97 s. The peak power and total energy generated in the test fuel were ~250 kW/m and about 1.23 kJ/g. The radial power profile in the test fuel was strongly peaked toward the fuel surface (surface to center ratio was ~4 to 1).
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Measurements /diagnostics during test	Temperatures near the NaK annulus were measured by thermocouples.
Posttest coolant flow condition	Static (same as initially)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	Radiography; cladding profilometry, axial gamma scanning, fission gas collection, ceramography
Posttest analyses	Fuel pin thermal-mechanical analysis.

KEY SUMMARY INFORMATION ABOUT THE TEST

The total fission gas release from the fuel (during the pre-irradiation and as a result of the transient) was ~24% of the fission gas generated during the pre-irradiation. As a result of the transient, the fuel column had lengthened by 0.4 cm, fuel pellets had fused together, and the entire length of the cladding increased in diameter by about 25 microns. No melting occurred in the fuel, but the fuel cracking pattern was much different from the un-tested irradiated sibling pin. (There were no large radial cracks but many smaller radial cracks, plus a circumferential crack at mid-radius.)

Summary of TREAT Test PNL 1-7

Performed (probably in 1970 or 1971)

Transient No. (unknown)

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Cladded mixed-oxide fuel pin	Transient conditions	Transient overpower, probably using a natural TREAT transient.
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U	Fuel sample configuration in the test	Single pin
Geometry	Pellet column; annular pellets with 1.37 mm diameter hole	Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Smear density	84.2% of theoretical	Posttest coolant flow condition	None (same as initially)
CLADDING		Posttest condition of test sample(s)	Cladding remained intact.
Material	Type 304 stainless steel, solution annealed	Posttest measurements & information	(unavailable)
OD	6.35 mm	Posttest analyses	Fuel pin thermal-mechanical analysis
Thickness	0.406 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~9 MWd/kg		
Peak linear power	29 kW/m		
Peak internal clad temperature	(not available)		
Fluence (spectrum)	Fluence not available; Fast (EBR-II) spectrum		
OTHER FUEL SAMPLE INFORMATION			
KEY SUMMARY INFORMATION ABOUT THE TEST			
No fuel melted. There was ~0.4% cladding strain due to the transient. Post-transient the total fission gas release from the fuel (including steady-state release during the pre-irradiation) was 18% of the gas generated.			

Summary of TREAT Test PNL-1-8

Performed (probably between 1970 and 1974)

Transient No. (unknown)

PRE-TEST SAMPLE CHARACTERIZATION	
FUEL	
Form	Mixed oxide pellets
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U
Geometry	Annular pellets with 1.37 mm diameter hole
Fissile height	~34.3 cm
Smear density	83.9% of theoretical
CLADDING	
Material	Type 304 stainless steel
OD	6.35 mm
Thickness	0.406 mm
PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	~8.5 MWd/kg
Peak linear power	~ 31 kW/m
Peak internal clad temperature	(not available)
Fluence (spectrum)	Fluence not available; Fast (EBR-II) spectrum
OTHER FUEL SAMPLE INFORMATION	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK, probably initially at ~150 °C
Transient conditions	"Natural" TREAT transient reaching 139 MW peak power and having fuel-width-at-half-maximum of ~0.95 s. The peak power and total energy generated in the test fuel were ~280 kW/m and about 1.3 kJ/g. The radial power profile in the test fuel was strongly peaked toward the fuel surface (surface to center ratio was ~4 to 1).
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Posttest coolant flow condition	Static (same as initially)
Posttest condition of test sample(s)	Localized cladding failure
Posttest measurements & information	Fission gas collection
Posttest analyses	Fuel pin thermal-mechanical analysis

KEY SUMMARY INFORMATION ABOUT THE TEST	
By the end of the transient, the total fission gas release (during the pre-irradiation and during the transient) from the fuel was ~45% of the fission gas generated during the pre-irradiation. No fuel melted. Cladding failure was at a hot spot.	

Summary of TREAT Test PNL 2-1

Performed (probably between 1970 and 1974)

Transient No. (unavailable)

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Cladded mixed-oxide fuel pin	Transient conditions	Transient overpower, probably using a natural TREAT transient
Composition	75 wt% UO ₂ – 25 wt% PuO ₂	Fuel sample configuration in the test	Single pin
Geometry	Pellet column, solid pellets	Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Smear density	87.6% of theoretical	Posttest coolant flow condition	None (same as initially)
CLADDING		Posttest condition of test sample(s)	Cladding failed
Material	Type 304 stainless steel, solution annealed	Posttest measurements & information	(unavailable)
OD	6.35 mm	Posttest analyses	Fuel pin therman-mechanical analysis
Thickness	0.406 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~41 MWd/kg		
Peak linear power	~30 kW/m		
Peak internal clad temperature	504 °C		
Fluence (spectrum)	~ 2.2 x 10 ²² n/cm ² (fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test PNL 2-1 simulated a 3 \$/s overpower ramp reactivity accident. There was a large uncertainty in the fuel pin failure time. The fuel pin suddenly broke completely into two pieces. The cladding failure extended from 13 to 18 cm above the bottom of the fuel column (fuel x/L = 0.37 to 0.52). The calculated cladding temperature rise rate near the time of cladding failure was ~930 °C/s.			

Summary of TREAT Test PNL 2-10

Performed (probably between 1970 and 1974)

Transient No. (unknown)

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Cladded mixed-oxide fuel pin
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U
Geometry	Pellet column, annular pellets with 1.37 mm diameter hole
Fissile height	34.3 cm
Smear density	79.2% of theoretical

CLADDING	
Material	Type 304 stainless steel, solution annealed
OD	6.35 mm
Thickness	0.406 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	~43.5 MWd/kg
Peak linear power	28 kW/m
Peak internal clad temperature	(unavailable)
Fluence (spectrum)	Fluence unavailable; Fast (EBR-II) spectrum

OTHER FUEL SAMPLE INFORMATION	
~50% of the generated fission gas was released during the pre-test, steady-state irradiation.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK
Transient conditions	Overpower transient, probably using a natural TREAT transient
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Posttest coolant flow condition	None (same as initially)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	(unavailable)
Posttest analyses	Fuel pin thermal-mechanical analysis

KEY SUMMARY INFORMATION ABOUT THE TEST	
There was no fuel melting and only slight strain of the cladding due to the transient. Post-transient the total fission gas release (during the pre-irradiation and the transient) was 78% of the gas generated during the pre-irradiation.	

Summary of TREAT Test PNL 2-11

Performed (probably between 1970 and 1974)

Transient No. (unavailable)

PRE-TEST SAMPLE CHARACTERIZATION		TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
FUEL		Coolant environment	Static NaK
Form	Cladded mixed-oxide fuel pin	Transient conditions	Overpower transient, probably using a natural TREAT transient
Composition	75 wt% UO ₂ – 25 wt% PuO ₂ 93% enriched U	Fuel sample configuration in the test	Single pin
Geometry	Pellet column, annular pellets with 1.37 mm diameter hole	Test vehicle & experiment system	Test pin was thermally bonded by a NaK annulus to a thick annular nickel heat sink located in a capsule with no neutron filtering.
Fissile height	34.3 cm	Measurements /diagnostics during test	Temperatures in or near the NaK annulus were measured by thermocouples.
Smear density	78.9% of theoretical	Posttest coolant flow condition	None (same as initially)
CLADDING		Posttest condition of test sample(s)	Gross cladding failure
Material	Type 304 stainless steel, solution annealed	Posttest measurements & information	(unavailable)
OD	6.35 mm	Posttest analyses	Fuel pin thermal-mechanical analysis.
Thickness	0.406 mm		
PRE-IRRADIATION CHARACTERISTICS			
BU (peak axial)	~41 MWd/kg		
Peak linear power	~27 kW/m		
Peak internal clad temperature	494 °C		
Fluence (spectrum)	~ 1.75 x 10 ²² n/cm ² (fast, EBR-II)		
OTHER FUEL SAMPLE INFORMATION			
KEY SUMMARY INFORMATION ABOUT THE TEST			
Test PNL 2-11 simulated a 3 \$/s overpower ramp reactivity accident. The cladding failure extended from 28 to 32 cm above the bottom of the fuel column (fuel x/L = 0.82 to 0.93).			

Appendix X: One-page Summaries of ORNL-TR-series Tests

One-page summaries of each of the ORNL-TR-series tests are included here. These summaries can also be found in TREXR [6].

Summary of TREAT Test ORNL-TR-1

Performed November 1969

Transient No. 1294

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Cladded mixed-oxide fuel pins
Composition	(80 wt%U - 20 wt%Pu) - O _{1.98}
Geometry	Column of solid pellets in Pins A and C; Sphere-Pac fuel in Pin B
Fissile height	20.3 cm in each pin
Smear density	Pin A: 88.9%; Pin B: 80.9%, Pin C: 79.7% (relative to theoretical density)

CLADDING	
Material	Type 304 stainless steel
OD	6.45 mm
Thickness	0.41 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(not applicable)

OTHER FUEL SAMPLE INFORMATION	
The fuel was produced by the sol-gel process.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static sodium; initial temperature before transient was 410 °C in Pin A, 449 °C in Pin B, and 414 °C in Pin C.
Transient conditions	Probably the power transient was basically a natural-burst transient. Estimated peak (midplane) linear power (in kW/m) and energy release (in kJ/g) were: 577 and 1.63 in Pin A, 505 and 1.57 in Pin B, and 386 and 1.22 in Pin C. The reactor peak power was 202 MW and integrated power was 140 MJ. The transient duration was < 4 s.
Fuel sample configuration in the test	Three pins in tandem (vertically), with Pin A at bottom and Pin C at top.
Test vehicle & experiment system	Static sodium capsule
Measurements /diagnostics during test	Fuel extensometer; probably also temperature measurement(s)
Posttest coolant flow condition	Zero (same as initially)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	Neutron radiography, cladding ovality, gamma scanning
Posttest analyses	(unknown)

KEY SUMMARY INFORMATION ABOUT THE TEST	
No fuel melting occurred in any of the pins. As a result of the transient, the fuel column length increased roughly 2.2 mm in Pin A and decreased roughly 0.5 mm in Pin B and 0.8 mm in Pin C. A central void 2.5 mm in diameter formed in Pin B. The reported radius of equiaxed grain growth at the fuel midplane (r/R , where R is the fuel radius) was 0.85 in Pin A, 1.00 in Pin B, and zero in Pin C. No regions of low density developed within the fuel pellets. There was no cladding deformation except for an ovality of 25-125 microns along Pin C. The fuel columns of Pin B shortened due to the transient.	

Summary of TREAT Test ORNL-TR-2

Performed November 1969

Transient No. 1295

PRE-TEST SAMPLE CHARACTERIZATION

FUEL	
Form	Cladded mixed-oxide fuel pins
Composition	(80 wt%U - 20 wt%Pu) - O _{1.98}
Geometry	Sphere-Pac fuel in Pins A and B. Column of solid pellets in Pin C.
Fissile height	20.3 cm in each pin
Smear density	Pin A: 82.0%; Pin B: 80.9%, Pin C: 80.2% (relative to theoretical density)

CLADDING	
Material	Type 304 stainless steel
OD	6.45 mm
Thickness	0.41 mm

PRE-IRRADIATION CHARACTERISTICS	
BU (peak axial)	None
Peak linear power	(not applicable)
Peak internal clad temperature	(not applicable)
Fluence (spectrum)	(not applicable)

OTHER FUEL SAMPLE INFORMATION	
The fuel was produced by the sol-gel process.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static sodium; initial temperature before transient was 400 °C in Pin A, 436 °C in Pin B, and 410 °C in Pin C.
Transient conditions	Probably the power transient was basically a natural-burst transient. Estimated peak (midplane) linear power (in kW/m) and energy release (in kJ/g) were: 576 and 2.17 in Pin A, 533 and 2.04 in Pin B, and 343 and 1.32 in Pin C. The reactor peak power was 202 MW and integrated power was 172 MJ. The transient duration was < 4 s.
Fuel sample configuration in the test	Three pins in tandem (vertically) with Pin A at bottom and Pin C at top.
Test vehicle & experiment system	Static sodium capsule
Measurements /diagnostics during test	Fuel extensometer; probably also temperature measurement(s)
Posttest coolant flow condition	Zero (same as initially)
Posttest condition of test sample(s)	Cladding remained intact.
Posttest measurements & information	Neutron radiography, cladding ovality, gamma scanning, metallurgical examination
Posttest analyses	(unknown)

KEY SUMMARY INFORMATION ABOUT THE TEST	
As a result of the transient, a central void formed in the fuel in Pins A and B (of diameter 3.0 mm in Pin A and 2.5 mm in Pin B). No central void formed in the fuel in Pin C. At the midplane of Sphere-Pac Pins A and B fuel melting was 50% by volume, and the equiaxed grain growth extended to the fuel outer radius. No fuel melting occurred in pelletized Pin C, and the equiaxed grain growth extended to 70% of the fuel outer radius. No regions of low density developed within the fuel pellets. There was no cladding deformation except for a localized ovality of 40 microns at the bottom of Pin C. The posttest appearances of Pins A and B were similar; each contained several separated, conical voids. In addition, Pin B had two band-type separations in the fuel column, each about 1.5 mm wide. The fuel columns of both Pins A and B shortened due to the transient.	



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