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

First Edition

Chemical and Fuel Cycle Technologies Division
Nuclear Science and Engineering Division

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Executive Summary

This report provides summary information on a selected set of key historic experiment series which were performed in the Transient Reactor Test Facility (TREAT) during its original operation from 1959-1994. The selected experiments are intended to illustrate the diversity in experiments which have been performed, and the range in capabilities of the TREAT reactor. Knowledge of those prior experiments provides a foundation for qualification and safety evaluation of numerous fuel forms already tested in TREAT, planning and designing new future tests, and for future TREAT operation.

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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 sucessfully 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].

It is the intent of this report to provide summary information on selected sets (series) of historic experiments to illustrate the range of experiments (and TREAT's capabilities) that characterized TREAT operations during the time period between the late 1960s and 1994. The experiment sets included in this report were chosen from among the several dozen experiment types or experiment series that were performed. 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 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 B_4C-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

This report is intended to summarize a considerable number of experiment series that were performed in TREAT and archived in the TREAT Experimental Relational (TREXR) Database [6] in a prioritized manner. The prioritization scheme is based on giving higher priority to the later test series (i.e., those performed in the 1970s and 1980s) and also higher priority to series that addressed issues of relatively-broad interest in the technical community. Consideration is also given to adding test series in groups that have some inter-relatedness among the several series in each group (e.g., the type of issues addressed, the experimental approach taken, the type of fuel tested,). In defining each series group, consideration is also given to the total number of tests in the group and how much information is available regarding the tests, with a view to roughly equalizing the effort it will take to properly address all of the tests in the each group, as described below.

In the following sections of this report, an overview is given for each series, using a common format for each overview, providing information on the testing purpose, approach, limitations, experiment vehicle, test fuel, test conditions, results, and applications. Additional information is 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. They include descriptions of the test sample(s), reactor operating conditions, measurements performed during and post-irradation, 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 first five experiment series (designated with prefix names CDT, CO, LO, M, and TS) were selected because they were among the most recent tests performed before TREAT was put on non-operational standby status in 1994, and they thus tested some of the most up-to-date types of irradiated fuels that were available at that time for transient testing. In addition, they illustrate a significant part of the range of types of tests that have been conducted in TREAT regarding test vehicle type, coolant environment of the test sample, test-fuel material and geometry, and transient severity.

The five series and their associated key parameters are summarized in Table 4-1. All tests were performed on fast-reactor fuels using shaped transients in the pre-upgraded TREAT core.

Table 4-1. Key Characteristics of the First Group of Test Series Addressed in this Report

Test series	Test fuel geometry	Coolant	Test-fuel type	Test-fuel damage
CDT	Single pin, or 3 pins in separate flowtubes	Flowing sodium	Oxide	No cladding breach, or breach with extensive pin disruption
CO	Single pin	Static NaK or flowing sodium	Oxide	Cladding breach with moderate or extensive pin disruption
LO	7-pin bundle	Flowing sodium	Oxide	Cladding breach with extensive pin disruption
M	2 or 3 pins in separate flowtubes	Flowing sodium	Metallic	No cladding breach, or breach with moderate pin disruption
TS	Single pin	Flowing sodium	Oxide	Slight cladding breach, or breach with extensive pin disruption

The second group of experiment series that is included here comprises the tests in the EBT, L, PINEX, RFT, and STEP series.

Table 4-2. Key Characteristics of the Second Group of Test Series Addressed in this Report

Test series	Test fuel geometry	Coolant	Test-fuel type	Test-fuel damage
EBT	Single pin, or 3 pins in separate flowtubes	Static NaK or flowing sodium	Oxide	No cladding breach, or breach with extensive pin disruption
L	Single pin, 3-pin bundle, 7-pin bundle	Flowing sodium	Oxide	No cladding breach, or breach with extensive pin disruption
PINEX	Single pin	Static NaK	Oxide	No cladding breach, or breach with extensive pin disruption
RFT	3 pins in separate flowtubes	Flowing sodium	Oxide	No cladding breach, or breach with little pin disruption
STEP	4-pin bundle	Flowing steam	Oxide	Cladding breach with considerable pin disruption

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 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 17-18 x10^{22} n/cm^2 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 9.4-9.9 x 10^{22} n/cm^2 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 $/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_o) 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_o 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_o in slow-ramp test CDT-1 and 17 P_o in test CDT-2. The medium-burnup annular-pellet pin in slow-ramp CDT-3 survived 4.5 P_o despite experiencing 20% more fuel enthalpy and 90 ^oC 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 $P_{o,}$ 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_o. 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 ¼ height of the fuel column, 50% at midheight, and 56% at ¾ 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 ¼ fuel height, 75% at the midplane, and 68% at ¾ 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 ¼, half, and ¾ 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 ~7% near the midplane, ~4% near the fuel top, and ~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 $/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 [11] [12] [13] [14] [15] [16] [17] [10]. 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 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 CO1, LO1, and LO6; pins of 9 at% burnup were used in CO3 and CO5; 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 (CO1, CO2, and CO3 on fuel of 0%, 4% and 9% burnup, respectively) and in the seven-pin loop tests (LO1 and LO2 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 (CO4 and CO5 on fuel of 4% and 9% burnup, respectively) and one seven-pin loop test (LO3 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 CO6R and four seven-pin loop tests LO4 through LO7 (using 0% burnup fuel in LO6 and 4% burnup fuel in CO6R, LO4, LO5, and LO7). 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 CO4 in which the worth decrease was about 30%,

and LO5 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 (LO4, LO5, and LO7) 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 LO6 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 CO1, CO2, and CO3 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 [18] [19] [10]. 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 U-5Fs fuel of burnups3.5 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] [20] [10] [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 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 sodlum 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 HUT5-1B; 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-IIC 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 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 ~6x1022 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 1x1023 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.

10 L-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 the 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 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 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 UO_2 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.)

12 RFT-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 [40] [41] [10] [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 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.

13 STEP-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 $^{\circ}$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.

14 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 provide 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)
APED	Fresh Na-bonded UO_2; single pin	Stagnant Na; undercooling and overpower	Limited damage; in-pin fuel motion
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
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
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
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
FC	(TBD)	(TBD)	(TBD)
FCT	(TBD)	(TBD)	(TBD)

FRF	(TBD)	(TBD)	(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	
HOP	Fast reactor MOX, medium burnup	Sodium (stagnant or flowing); transient overpower	Mild cladding failure; fission gas release
HUCPTO	Fast reactor MOX, medium burnup	Flowing sodium; undercooling	(TBD)
HUT	Fast reactor MOX, low, medium and high burnup	Stagnant sodium; overpower	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	(TBD)	(TBD)	(TBD)
NRPA	(TBD)	(TBD)	(TBD)
ORNL	(TBD)	(TBD)	(TBD)
PBF	(TBD)	(TBD)	(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,	Pre-failure fuel motions; cladding failure threshold;

		undercooling+ overpower	post-failure material interactions and motions
RFT	Fast reactor MOX; low and medium burnup; single pin	Flowing sodium; overpower	Limited fuel damage; pre-failure fuel motion
RP	(TBD)	(TBD)	(TBD)
RX	Fast reactor UO_2+steel; fresh fuel; monolithic mass	No coolant; heating to simulate melt-down accident transition phase	(TBD)
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**	(TBD)	(TBD)	(TBD)
TR	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
23nT[1]	LWR UO_2	Water coolant; overpower	Cladding failure threshold; 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

FUEL	
Form	He-bonded MOX
Composition	UO_2(Natural) – 24.6% PuO_2
Geometry	Solid pellet
Fissile height	91.4 cm
Smear density	84.3% of theoretical

CLADDING	
Material	Thermo-mechanical treated HT9 ferritic alloy
OD	0.686 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium
Transient conditions	10-s steady-state flattop to 33.6 kW/m, 28-s 5¢/s TOP to peak power of 160.9 kW/m, scram
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Modified Mark IIIC Loop
Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Posttest coolant flow condition	No blockage
Posttest condition of test sample(s)	No cladding failure, severe fuel motion/voiding, max. cladding strain of 2.5%
Posttest measurements & information	Neutron radiography, ceramography, macro and metallographic examination, gamma scanning, full plenum spring compression
Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)

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	
FUEL	
Form	He-bonded MOX
Composition	ACO-1: UO_2 (natural)– 24.6w/o PuO_2 FO-2: UO_2(depleted) – 26w/o PuO_2
Geometry	Pellet (ACO-1 solid; FO-2 solid or annular)
Fissile height	91.4 cm
Smear density	ACO-1: 84.3% of theoretical FO-2 Solid: 84.3% of theoretical FO-2 Annular: 80% of theoretical

CLADDING	
Material	Thermo-mechanically treated HT9 ferritic alloy
OD	0.686 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	Three pins in separate flow tubes
Test vehicle & experiment system	Modified Mark IIIC Loop
Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Posttest coolant flow condition	No blockage
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
Posttest measurements & information	Neutron radiography, ceramography, macro & metallographic examination, gamma scanning
Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)

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

FUEL	
Form	He-bonded MOX
Composition	UO_2(natural) – 26.0w/o PuO_2
Geometry	Annular pellet
Fissile height	91.4 cm
Smear density	80.0% of theoretical

CLADDING	
Material	Thermo-mechanically treated HT9 ferritic alloy
OD	0.686 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	Three pins in separate flow tubes
Test vehicle & experiment system	Modified Mark IIIC Loop
Measurements /diagnostics during test	Thermocouples, flowmeters, fuel motion (hodoscope)
Posttest coolant flow condition	No blockage
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
Posttest measurements & information	Neutron radiography, ceramography, macro and metallographic examination, gamma scanning
Posttest analyses	Profilometry, TEMECH and SIEX-4P code comparison (pre-failure and end-of-life fuel condition)

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

FUEL	
Form	Helium-bonded MOX
Composition	UO_2 (natural U) – 31.5 wt% PuO_2
Geometry	Annular pellet, flat ends
Fissile height	91.4 cm
Smear density	80.0% of theoretical

CLADDING	
Material	316 stainless steel
OD	0.584 cm
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 UO2 (10 cm upper/solid, 46 cm breeder lower/annular); bottom plenum 15.4 cm^3

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	$5/s ramp insertion
Fuel sample configuration in the test	Single full-size fuel pin
Test vehicle & experiment system	Single-pin capsule with a massive nickel heat sink at the periphery of the NaK coolant.
Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movement (by hodoscope and PINEX camera)
Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
Posttest condition of test sample(s)	Gross disruption
Posttest measurements & information	Location of fuel failure, fuel and cladding distribution, neutron radiography, optical microscopy
Posttest analyses	Thermal-mechanical analyses

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

FUEL	
Form	Helium-bonded MOX
Composition	UO_2 (natural U) – 31.5 wt% PuO_2 Oxygen-to-metal ratio = 1.985
Geometry	Annular pellet, flat ends
Fissile height	91.4 cm
Smear density	80.0% of theoretical

CLADDING	
Material	316 stainless steel
OD	0.584 cm
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 1.61x 10^{-3} moles = 29%.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	$5/s reactivity ramp insertion
Fuel sample configuration in the test	Single full size fuel pin
Test vehicle & experiment system	PFR/TREAT Test Vehicle Assembly (stagnant NaK capsule)
Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movements (by hodoscope and PINEX camera)
Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
Posttest condition of test sample(s)	Gross disruption
Posttest measurements & information	Neutron radiography, optical microscopy, location of fuel failure, fuel and cladding redistribution,
Posttest analyses	Thermal-mechanical analyses

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

FUEL	
Form	Helium-bonded MOX
Composition	73 wt% UO2 (23.5% enriched), 27 wt% PuO_2
Geometry	Annular pellet, flat ends
Fissile height	91.44 cm
Smear density	81.5% of theoretical

CLADDING	
Material	316 stainless steel
OD	0.584 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	$5/s reactivity ramp insertion
Fuel sample configuration in the test	Single full-size fuel pin
Test vehicle & experiment system	PFR/TREAT Test Vehicle Assembly (static NaK capsule)
Measurements /diagnostics during test	Coolant and heat-sink temperature, fuel movements (by hodoscope and PINEX camera)
Posttest coolant flow condition	Cladding and fuel debris around pin; no flow data
Posttest condition of test sample(s)	Gross disruption
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
Posttest analyses	Thermal-mechanical analyses

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 ¼ 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

FUEL	
Form	Helium-bonded MOX
Composition	UO_2 (natural U) – 31.5 wt% PuO_2 Oxygen-to-metal ratio = 1.985
Geometry	Annular pellets with flat ends
Fissile height	91.4 cm
Smear density	79% of theoretical

CLADDING	
Material	316SS, 20% CW
OD	0.584 cm
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 1.61x 10^{-3} moles=29%.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium at 120 g/s; 0.47 MPa system pressure, 670 K
Transient conditions	Fast rise to ~44 kW/m immediately followed by a 15-s period power increase; scram immediately upon pin failure
Fuel sample configuration in the test	Single fuel pin
Test vehicle & experiment system	Mark IIIA Mod 1 loop with Single-Pin Test Train
Measurements /diagnostics during test	Coolant temperature, flow rate, fuel movement (by hodoscope)
Posttest coolant flow condition	Nearly the same as the pre-test flow rate.
Posttest condition of test sample(s)	Cladding intact up to fuel height x/L~0.98. Local breach at fuel top. Fuel dispersal from cladding. Unmelted fuel on interior wall of cladding along most of fuel region.
Posttest measurements & information	Optical microscopy, neutron radiography, photography, x- radiography, autoradiography
Posttest analyses	Thermal-mechanical analysis of fuel pin

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

FUEL	
Form	Helium-bonded MOX
Composition	73 wt% UO_2 (23.5% enriched), 27 wt% PuO_2
Geometry	Annular pellets with flat ends
Fissile height	91.4 cm
Smear density	81% of theoretical

CLADDING	
Material	316SS, 20% CW
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium at 120 g/s; 0.33 MPa system pressure
Transient conditions	Fast rise to ~44 kW/m immediately followed by a 15-s period power increase; scram immediately upon pin failure
Fuel sample configuration in the test	Single fuel pin
Test vehicle & experiment system	Mark IIIA Mod 1 Loop with Single-Pin Test Train
Measurements /diagnostics during test	Coolant temperature, flow rate, fuel movement (by hodoscope)
Posttest coolant flow condition	Fully-blocked flow channel
Posttest condition of test sample(s)	Cladding intact up to fuel height x/L~0.94. Localized cladding breach. Fuel dispersal from cladding. Unmelted fuel on interior wall of cladding along most of fuel region.
Posttest measurements & information	Optical microscopy, neutron radiography, photography, x-radiography, autoradiography
Posttest analyses	Thermal-mechanical analysis of fuel pin.

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	
FUEL	
Form	Helium-bonded MOX
Composition	UO_2 (natural U) – 31.5wt% PuO_2 Oxygen-to-metal ratio = 1.985
Geometry	Annular pellets with flat ends
Fissile height	91.4 cm
Smear density	81% of theoretical
CLADDING	
Material	M316 stainless steel
OD	0.585 cm
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 1.61x 10^{-3} moles=29%.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium, inlet temperature 670 K, 0.184 MPa system pressure
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.
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Single-Pin Test Loop
Measurements /diagnostics during test	Coolant temperature, coolant flow rate, coolant pressure, fuel distribution
Posttest coolant flow condition	(not relevant to the accident scenario due to the large amount of excess energy generated)
Posttest condition of test sample(s)	Gross disruption
Posttest measurements & information	Neutron radiography
Posttest analyses	Fuel-pin thermal-mechanics, coolant thermal-hydraulics, fuel motion

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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; Pu/(U+Pu) = 0.289 Oxygen-to-metal ratio 1.965
Geometry	Annular pellet, flat ends
Fissile height	91.4 cm
Smear density	78.5% of theoretical

CLADDING	
Material	SS316 20% CW
OD	0.584 cm
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 UO2 (9.4 cm upper, 46 cm lower);
15.4 cm3 bottom plenum at 4.19 atm (cold)

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	7-pin, grid-spaced bundle
Test vehicle & experiment system	Mark IIIA loop, dysprosium axial shaping collar, B_6Si & dysprosium neutron filters
Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion (hodoscope)
Posttest coolant flow condition	Complete upper blockage of once-molten fuel; metal blockage at the bottom of the initial fuel region
Posttest condition of test sample(s)	Severe cladding failure and dispersive fuel motion, complete pin disruption in upper two-thirds of fuel zone
Posttest measurements & information	Macro- and micro examination, neutron radiography, metallographic examination, sibling pin analysis,
Posttest analyses	Thermal hydraulic, cladding failure, fuel and cladding motion, code validation

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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; 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

CLADDING	
Material	SS316 20% CW
OD	0.584 cm
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).

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	7-pin bundle; grid-spaced pins
Test vehicle & experiment system	Mark IIIA Loop, dysprosium axial shaping collar, B_6Si and dysprosium neutron filters
Measurements /diagnostics during test	Flow rate, coolant pressure, temperatures, fuel distribution
Posttest coolant flow condition	Complete upper blockage of once-molten fuel/cladding material
Posttest condition of test sample(s)	Severe cladding failure, net upward dispersive fuel motion, complete pin disruption in upper 3/4 of fuel zone
Posttest measurements & information	Macro and micro examination, neutron radiography, metallographic examination
Posttest analyses	Thermo-hydraulic analysis, fuel and cladding motion analysis, code validation

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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; Pu/(U+Pu) = 0.315 Oxygen-to-metal ratio 1.980
Geometry	Annular pellets with flat ends
Fissile height	91.4 cm
Smear density	80% of theoretical

CLADDING	
Material	SS316 20% CW
OD	0.584 cm
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).

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
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.
Fuel sample configuration in the test	7-pin hexagonal bundle; grid-spaced pins
Test vehicle & experiment system	Modified Mark IIIA loop with dual ALIPs;, dysprosium shaping collars
Measurements /diagnostics during test	Flowmeters, pressure transducers, thermocouples, fuel motion (hodoscope)
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.
Posttest condition of test sample(s)	Extremely severe cladding and fuel relocation (gross disruption), molten fuel and steel relocation throughout loop components
Posttest measurements & information	Macro and micro examination, neutron radiography, metallographic examination, fuel and cladding distribution
Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation

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	UO2 (natural U)-PuO2; 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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; 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; flow rate held at 36% of nominal
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.
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	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Total blockage of once-molten fuel and steel around upper breeder region, lower blockage of once-molten steel
Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), , flowtube melt-through, pellet stack collapse
Posttest measurements & information	Neutron radiography and tomography, 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 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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; 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)	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).

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.
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	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Upper blockage of once-molten fuel and steel, complete lower blockage of once-molten steel
Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), flowtube melt-through
Posttest measurements & information	Neutron radiography and tomography, macro examination, fuel and cladding distribution
Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation

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

FUEL	
Form	He-bonded MOX
Composition	UO2 (natural U)-PuO2; 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

CLADDING	
Material	SS316 20% CW
OD	0.584 cm
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 UO2 (10 cm upper, 46 cm lower); 15.4 cm3 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; flow rate held at 43% of nominal
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.
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	Coolant channel flow rate and pressure, temperatures, fuel motion (hodoscope)
Posttest coolant flow condition	Total blockage of once-molten fuel and steel around spacer grids, lower blockage of once-molten steel
Posttest condition of test sample(s)	Gross fuel and cladding disruption (severe melting and movement), flowtube melt-through, pellet stack collapse
Posttest measurements & information	Neutron radiography and tomography, macro examination, fuel and cladding distribution
Posttest analyses	Pin failure, thermal-hydraulics, post-failure fuel motion, code validation

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	
FUEL	
Form	Lateral segments of clad fuel cut from fuel pin
Composition	U-5Fs
Geometry	Two fuel segments, separated
Fissile height	1.27 cm, each segment
Smear density	~75% of theoretical (17.6 g/cc as-cast)
CLADDING	
Material	316 stainless steel, SA
OD	0.442 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	No coolant; inert gas environment
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.
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
Test vehicle & experiment system	F-series type of dry capsule
Measurements /diagnostics during test	High-speed photography with laser illumination; thermocouples spot-welded to each sample; pressure transducers; fuel motion by hodoscope
Posttest coolant flow condition	Not applicable
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.
Posttest measurements & information	Neutron radiography
Posttest analyses	Not available

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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-5Fs
Geometry	Cast pin
Fissile height	34.3 cm, as fabricated
Smear density	~75% of theoretical (17.6 g/cc as-cast)

CLADDING	
Material	316 stainless steel, SA
OD	0.442 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium; same flow rate for each pin
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.
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
Test vehicle & experiment system	Mark-IIIC sodium loop
Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
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.
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.
Posttest measurements & information	Neutron radiograph; detailed posttest metallurgical examinations
Posttest analyses	Cladding failure analysis; pre- and post-failure fuel motion analysis; thermal-hydraulic analysis

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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-5Fs
Geometry	Cast pin
Fissile height	34.3 cm, as fabricated
Smear density	~75% of theoretical (17.6 g/cc as-cast)

CLADDING	
Material	316 stainless steel, SA
OD	0.442 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium; same flow rate for each pin
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.
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
Test vehicle & experiment system	Mark-IIIC sodium loop
Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
Posttest coolant flow condition	No significant changes
Posttest condition of test sample(s)	All pins intact. Significant axial fuel elongation occurred in the lowest burnup pin.
Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
Posttest analyses	Cladding failure predictions, pre-failure fuel motion analysis, thermal-hydraulic analysis

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 flow rates 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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-5Fs
Geometry	Cast pin
Fissile height	34.3 cm, as fabricated
Smear density	~75% of theoretical (17.6 g/cc as-cast)

CLADDING	
Material	316 stainless steel, SA
OD	0.442 cm
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)	Fuence 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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium
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.
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
Test vehicle & experiment system	Mark-IIIC sodium loop
Measurements /diagnostics during test	Coolant pressure, temperature, and flow rate; flowtube outer surface temperature; fuel motion monitoring by hodoscope
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.
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.
Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations.
Posttest analyses	Cladding failure analysis, pre- and post-failure fuel motion analysis, thermal-hydraulic analysis

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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-19Pu-10Zr
Geometry	Cast fuel slug
Fissile height	34.3 cm (as-fabricated)
Smear density	72.5 % of theoretical 11.2 g/cc

CLADDING	
Material	D9 Steel
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium; 0.4 MPa system pressure
Transient conditions	Power increased quickly to nominal power, then increased exponentially on an 8-s period.
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.
Test vehicle & experiment system	Mark-IIIC sodium loop
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
Posttest coolant flow condition	Coolant boiled at end of test
Posttest condition of test sample(s)	Both fuel pins remained intact
Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
Posttest analyses	Cladding failure, pre- and post-failure fuel motion, thermal-hydraulics

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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-19Pu-10Zr
Geometry	Cast fuel slug
Fissile height	34.3 cm (as-fabricated)
Smear density	72.5 % of theoretical 11.2 g/cc

CLADDING	
Material	D9 Steel
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium; 0.4 MPa system pressure.
Transient conditions	Power was raised first to approximately nominal power, the increased exponentially on an 8-s period until cladding failure was reached, then sharply reduced.
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.
Test vehicle & experiment system	Mark-IIIC sodium loop
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
Posttest coolant flow condition	Temporary reversal of flow in the flowtube of the 5.3 at% burnup pin.
Posttest condition of test sample(s)	5.3 at.% pin failed, releasing fuel into coolant flowtube.
Posttest measurements & information	Neutron radiograph; detailed metallurgical examinations
Posttest analyses	Cladding failure analysis, pre- and post-failure fuel motion analyses, thermal-hydraulic analysis

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

FUEL	
Form	Sodium-bonded metallic fuel
Composition	U-19Pu-10Zr in one pin; U-10Zr in the other
Geometry	Cast fuel slug
Fissile height	34.3 cm (as-fabricated)
Smear density	72.5 % of theoretical 11.2 g/cc

CLADDING	
Material	D9 steel for the U-Pu-Zr fuel; HT9 steel for the U-Zr fuel
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium; 0.4 MPa system pressure
Transient conditions	Power increased exponentially for 8 s until cladding failure was reached, then sharply reduced.
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
Test vehicle & experiment system	Mark-IIIC sodium loop
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
Posttest coolant flow condition	Temporary reversal of flow, followed by restoration of 0.7 times prefailure flow rate in flowtube of failed pin.
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.
Posttest measurements & information	Neutron radiography, isotopic gamma scanning, laser profilometry, plenum gas analysis, macro- and microphotography
Posttest analyses	Cladding failure, pre- and post-failure fuel motion, thermal-hydraulics

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

FUEL	
Form	He-bonded MOX
Composition	UO_2(Natural) – 22.4 w/o PuO_2
Geometry	Solid, dished pellet
Fissile height	91.4 cm
Smear density	85.6% of theoretical

CLADDING	
Material	20% cold-worked type 316 SS
OD	0.584 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium, 103-114 g/s
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
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Single Pin Test Loop (SPTL) – Type B, with dysprosium collars to axially shape the neutron flux reaching the test fuel
Measurements /diagnostics during test	Thermocouples, flow sensors, fuel motion (hodoscope)
Posttest coolant flow condition	Incomplete blockage; 90% of initial flow rate, but the flow tube was breached.
Posttest condition of test sample(s)	Cladding breach was at 79 cm above fuel bottom and less than 1 cm long.
Posttest measurements & information	Neutron radiography, ceramography, cladding strain measurement
Posttest analyses	TEMECH thermal-mechanical analysis of fuel, cladding, and flow tube, including cladding failure time and location.

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

FUEL	
Form	He-bonded MOX
Composition	UO_2(Natural) – 22.4 w/o PuO_2
Geometry	Solid, dished pellet
Fissile height	91.4 cm
Smear density	85.6% of theoretical

CLADDING	
Material	20% cold-worked type 316 SS
OD	0.584 cm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium, 113 to 103 g/s
Transient conditions	7-s hold at 34 kW/m; then rise on 20.8-s exponential period (5 ¢/s TOP simulation) to cladding failure; then immediate reactor scram
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Single Pin Test Loop (SPTL) – Type B, with dysprosium collars to axially shape the neutron flux reaching the test fuel
Measurements /diagnostics during test	Thermocouples, flow sensors, fuel motion (hodoscope)
Posttest coolant flow condition	10% of the initial flow rate, but the flowtube wall had been extensively breached
Posttest condition of test sample(s)	The fuel pin was destroyed in the fuel region above X/L = 0.6.
Posttest measurements & information	Neutron radiography, posttest examination of remains
Posttest analyses	TEMECH thermal-mechanical analysis of fuel, cladding, and flow tube, including cladding failure time and location.

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

FUEL	
Form	He-bonded MOX
Composition	Pu/(U+Pu) = 0.29; 81% enr. U
Geometry	Solid pellet; 4.95 mm diameter
Fissile height	34.3 cm
Smear density	90.2% of theoretical

CLADDING	
Material	D9 alloy, 20% cold worked
OD	5.84 mm
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)	7.0×10^{22} n/cm^2 fast spectrum

OTHER FUEL SAMPLE INFORMATION
65 g of fuel in the pin.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	5.6 s preheat at ~40 kW/m, then 50 ¢/s reactivity ramp overpower to 270 kW/m peak
Fuel sample configuration in the test	Single pin
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
Measurements /diagnostics during test	Thermocouples, fuel motion (hodoscope)
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Massively and extensively disrupted
Posttest measurements & information	Neutron radiography, gas sampling
Posttest analyses	(unknown)

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	
FUEL	
Form	He-bonded MOX;
Composition	Pu/(U+Pu) = 0.29; 81% enr. U
Geometry	Solid pellet; 4.95 mm diameter
Fissile height	34.3 cm
Smear density	90.2% of theoretical
CLADDING	
Material	D9 alloy, 20% cold worked
OD	5.84 mm
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)	3.1×10^{22} n/cm^2 fast spectrum
OTHER FUEL SAMPLE INFORMATION	
65 g of fuel in the pin.	

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
Transient conditions	5.6 s preheat at ~40 kW/m, then 50 ¢/s reactivity ramp overpower to 270 kW/m peak
Fuel sample configuration in the test	Single pin
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
Measurements /diagnostics during test	Thermocouples, fuel motion (hodoscope)
Posttest coolant flow condition	(not applicable)
Posttest condition of test sample(s)	Greatly disrupted over most of the fuel column length
Posttest measurements & information	Neutron radiography, gas sampling
Posttest analyses	(unknown)

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

FUEL	
Form	He-bonded MOX
Composition	1: UO_2 – 25.0 PuO_2 2: UO_2 – 29.3 PuO_2 3: UO_2 – 29.3 PuO_2 All Pu/(Pu+U) = 0.25
Geometry	Solid pellet
Fissile height	34.3 cm
Smear density	~86% of theoretical

CLADDING	
Material	1: 316 SS, 20% cold worked 2: D9 austenitic alloy, 20% CW 3: HT9 ferritic alloy, air quenched, tempered
OD	5.84 mm
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)

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing sodium
Transient conditions	6-s steady state of between 35-38 kW/m, then 10 ¢/s overpower reaching peak of 91-100 kW/m at 10 s from the start of the overpower, then scram
Fuel sample configuration in the test	Three pins, each in a separate flowtube
Test vehicle & experiment system	Mark-IIC sodium loop with axial thermal neutron filters
Measurements /diagnostics during test	Flowmeters, thermocouples, pressure transducers, fuel motion monitor (hodoscope)
Posttest coolant flow condition	No blockages
Posttest condition of test sample(s)	No cladding failures, fuel melting in all pins, maximum strain 0.7% (in the HT9-clad pin)
Posttest measurements & information	Neutron radiography, macro examination, profilometry, ceramography
Posttest analyses	TMECH and SIEX code analyses

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)	1.0×10^{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-IIC 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.

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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 buldging 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 flattop 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 overshot 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 eructation 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 ~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 UO_2 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 UO_2 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

FUEL	
Form	He-bonded MOX
Composition	UO_2-75%PuO_2, with the UO_2 being 30% enriched
Geometry	Solid pellet
Fissile height	86.4 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)	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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

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,
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.
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-IIC integral sodium loop
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)
Posttest coolant flow condition	No sodium flow could be achieved posttest.
Posttest condition of test sample(s)	Gross failure of all pins over nearly the entire length of the fuel region.
Posttest measurements & information	Neutron radiography, gamma-ray scans, macro- and micro-exainations of the fuel remains
Posttest analyses	Analyses with SAS3A

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 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	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-IIC 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-IIC 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 UO_2 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 cm3/s, then coasting down at ~20 cm3/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-thick16-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-IICB 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

FUEL	
Form	He-bonded MOX
Composition	25 wt % PuO2, 75 wt % UO2 (natural U)
Geometry	Annular pellets (0.81 mm ID)
Fissile height	86.4 cm
Smear density	86.1% of theoretical density

CLADDING	
Material	316 SS, 20% CW
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Static NaK
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.
Fuel sample configuration in the test	Single pin
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
Measurements /diagnostics during test	PINEX pinhole imaging system; fuel motion (hodoscope), temperatures near fuel pin
Posttest coolant flow condition	(not applicable)
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.
Posttest measurements & information	Neutron radiography, x-radiography, ceramographic microstructural examination
Posttest analyses	Code comparison among the ORIGEN, TEMP, LAFM, HOTPIM, FUMO-T, FRAX-2, and PINEX-AR codes.

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

FUEL	
Form	He-bonded MOX
Composition	25 wt % PuO2, 75 wt % UO2 (natural U)
Geometry	Annular pellets (0.81 mm ID)
Fissile height	86.36 cm
Smear density	86.1% of theoretical density

CLADDING	
Material	316 SS, 20% CW
OD	0.584 cm
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.

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Static NaK
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
Fuel sample configuration in the test	Single pin
Test vehicle & experiment system	Nickel heat sink static capsule in containment vessel, B_6Si axial flux shaping collars
Measurements /diagnostics during test	Fuel motion (hodoscope), photography (pinhole imaging system), thermocouples
Posttest coolant flow condition	(not applicable)
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
Posttest measurements & information	X-radiography, neutron radiography, destructive examination, microstructure analysis
Posttest analyses	Analyses with TEMP, LAFM, FUMO-E codes.

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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
Transient conditions	Short (~6 s) preheat at slightly more than nominal power, then FFTF 50 ¢/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 ¢/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

FUEL	
Form	He-bonded MOX
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)
Geometry	Solid pellet
Fissile height	91.4 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: 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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
Transient conditions	Short (~7 s) preheat at pin powers between 1x and 2x nominal, then 5 ¢/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)
Fuel sample configuration in the test	Three pins, each in a separate flowtube
Test vehicle & experiment system	Mark-IIIA sodium loop, with axial flux-shaping collars
Measurements /diagnostics during test	Thermocouples, flowmeters
Posttest coolant flow condition	Essentially the same as pretest.
Posttest condition of test sample(s)	All pins remained intact.
Posttest measurements & information	Macro-examination, neutron radiography, gamma scanning, fission gas sampling, ceramographic examination
Posttest analyses	Analyses using TEMECH, SAS4A, FPIN2, and LIFE codes

KEY SUMMARY INFORMATION ABOUT THE TEST

Test RFT-L3 was conducted to investigate the effects of a 50 ¢/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

FUEL	
Form	He-bonded MOX
Composition	UO_2 – 23% PuO_2 (Natural UO_2, O/M = 1.96)
Geometry	Solid pellet
Fissile height	91.4 cm
Smear density	85% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.82 mm
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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	Three pins, each in a separate flowtube
Test vehicle & experiment system	Mark-IIIA sodium loop, with axial flux-shaping collars
Measurements /diagnostics during test	Thermocouples, flowmeters
Posttest coolant flow condition	Essentially the same as pretest.
Posttest condition of test sample(s)	No cladding failures.
Posttest measurements & information	Neutron radiography, gamma scanning, profilometry, destruction examination
Posttest analyses	Analyses with SAS4A and TEMECH

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

FUEL	
Form	He-bonded MOX
Composition	UO_2 – 22% PuO_2
Geometry	Solid pellet
Fissile height	91.4 cm
Smear density	85% of theoretical

CLADDING	
Material	SS 316 (20% CW)
OD	5.82 mm
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: 0.423×10^{23} n/cm^2 Pin 2: 0.877×10^{23} n/cm^2 Pin 3: 0.872×10^{23} n/cm^2 Fast spectrum

OTHER FUEL SAMPLE INFORMATION

FFTF driver fuel pins; 145 mm long Inconel reflector above upper insulator pellets

TEST CONDITIONS, MEASUREMENTS, & ANALYSES

Coolant environment	Flowing sodium
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
Fuel sample configuration in the test	Three pins in separate flowtubes
Test vehicle & experiment system	Mark-IIIA sodium loop, with axial flux-shaping collars
Measurements /diagnostics during test	Thermocouples, flowmeters
Posttest coolant flow condition	Essentially the same as pretest.
Posttest condition of test sample(s)	No cladding failures.
Posttest measurements & information	Gamma scanning, neutron radiography, profilometry, destructive examinations
Posttest analyses	Analysis with SAS4A and TEMECH codes.

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

FUEL	
Form	He-bonded cm 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

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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing steam
Transient conditions	20.6 min. at low pressure with an initial heatup rate of 1.8 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, temporary blockage
Posttest condition of test sample(s)	Severe fuel damage over 90% of the fuel column (in central & upper part)
Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry, chemical composition analysis, 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-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

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	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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing steam
Transient conditions	21.4 min. at high pressure with heatup rate of 1.1 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	No blockage
Posttest condition of test sample(s)	Major fission product release; severe fuel damage limited to upper third of 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-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

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	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

TEST CONDITIONS, MEASUREMENTS, & ANALYSES	
Coolant environment	Flowing steam
Transient conditions	23.8 min. at high pressure (8 MPa) with a fast heatup rate (initially 1.0 K/s)
Fuel sample configuration in the test	4-pin bundle; control rod material present (to simulate PWR control rod)
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, f fuel motion (hodoscope)
Posttest coolant flow condition	No blockage
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.
Posttest measurements & info	SEM examination, electron microprobe, neutron radiography, gamma ray scanning, X-ray diffraction, mass spectrometry.
Posttest analyses	SOAP computer code validation, fission product analysis, thermodynamic analysis, thermal-hydraulics, hydrogen generation and release

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.

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