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AEC RESEARCH AND
DEVELOPMENT REPORT
SEPTEMBER 1969

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TRANSIENT OVERPOWER IRRADIATION
OF AXIALLY-RESTRAINED,
ZERO-BURNUP,
FAST REACTOR FUEL SPECIMENS

G. R. THOMAS
J. H. FIELD

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT (04-3)-189
PROJECT AGREEMENT 10

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ABSTRACT

This report presents the results of an experimental program for evaluating the transient overpower limits of Liquid Metal Fast Breeder Reactor (LMFBR) fuel specimens as a function of fuel column axial restraint, fuel density, and fuel form (pellet versus powder fuel). The test specimens contained zero-burnup ($Pu_{0.2}U_{0.8}$)O₂ fuel. In size, they were one-quarter of an inch in diameter with a fuel column length of ≈ 14 inches. The specimens were clad by type 316 stainless steel.

The results indicate that the transient failure threshold for zero-burnup fuel is a strong function of the combination of fuel density and molten fuel volume. Following cladding failure, molten fuel can exhibit high mobility, and under the conditions of the experiment, the resultant interaction between molten fuel, cladding, and sodium coolant is not violent.

1. SUMMARY AND CONCLUSIONS

A series of transient overpower irradiations was conducted involving four pellet and two powder fuel specimens. These $\frac{1}{4}$ -inch diameter fuel specimens contained ≈ 14 inches of zero-burnup, mixed-oxide fuel under axial restraint.

The specimens underwent transient irradiation in the Transient Reactor Test Facility (TREAT) while encapsulated in instrumented, sodium-filled capsules. Failure occurred, in the form of breached cladding, in both pelleted specimens (C4A and C4B) having a 90% smeared density, but failure did not occur in either of the two similar specimens (C4C and C4D) having an 80% smeared density. Cladding failure also occurred in one of the two 83.5% dense powder fuel specimens (C4J failed, C4I did not).

Based on comparison of the data resulting from transient irradiation of these six specimens and on comparison with a previously reported test series involving short lengths of unrestrained fuel¹ (see table 1-1), the following conclusions were reached for zero-burnup fuel exposed to transients that caused fuel in 50% or more of the fuel cross section to attain conditions in or above the heat of fusion (i.e., 50% of the fuel cross section attaining molten conditions):

- Fuel pin failure (breach of cladding) appears to be a strong function of the amount of fuel which becomes molten during the transient irradiation, rather than total or peak energy release.
- Axial restraint of the fuel column, and/or increased fuel column length, significantly lowers the failure threshold for fuel pins containing fuel of $\geq 90.4\%$ smeared density.

- On a $\frac{1}{4}$ -inch diameter fuel specimen, a fuel column length of ≈ 14 inches is apparently sufficient to result in inherent axial restraint of the fuel column during severe transient heating.
- In pellet fuel specimens, decreasing the smeared density markedly increases the transient failure threshold.
- Following cladding failure, molten mixed-oxide fuel can exhibit a high degree of mobility, leading to axial relocation of fuel in both directions.
- Cladding failure can apparently occur at cladding temperatures significantly below conditions that could be associated with overheating of the cladding due to boiling of the sodium coolant.
- Coherent formations of fuel in the failure regions indicate that the post-failure interaction between molten fuel, cladding, and sodium was not violent.

The transient overpower conditions simulated in these experiments correspond to hypothetical accident conditions in an LMFBR. The high volume fractions of molten fuel attained in these tests are equivalent to typical values calculated for a 1.50\$ reactivity insertion on a 15\$/sec ramp.⁹ Since such extreme conditions represent highly unlikely events, these tests indicate a very substantial margin between any credible accident situation and the failure threshold for zero burnup fuel. The continuation of this test series will provide a comparison to failure thresholds for high burnup specimens irradiated in EBR-II.

Table 1-1
SUMMARY OF TREAT IRRADIATIONS OF ZERO-BURNUP FUEL

	Specimen								
	1	2	3*	4*	5	6	7	8	9
Smeared Density (% TD)	80	80	83	84	90	90	91	91	92
Fuel Column Length to Diameter Ratio	≈66	≈66	≈61	≈61	≈28	≈66	≈28	≈66	≈28
Fuel Volume Melt (%)	≈50	67-70	≤30	49-60	≈0	57-63	≈72	76-79	≈80
Axial Restraint	yes	yes	yes	yes	no	yes	no	yes	no
Cladding Failure	no	no	no	yes	no	yes	no	yes	yes

NOTE:

PARAMETERS AFFECTING FAILURE THRESHOLD

COMPARE:

- (1) Axial Restraint and/or Fuel Column Length 6 and 8 with 7
 (2) Smeared Fuel Density 6 and 8 with 2

* Vipac Power Fuel

2. INTRODUCTION

2.1 PROGRAM DESCRIPTION

Over the past several years, General Electric Company has been conducting experiments under the sponsorship of the USAEC,* to determine the response of fast reactor fuel to severe transient overpower conditions. These tests, which were begun in 1961, have been performed in the Transient Reactor Test Facility (TREAT) at the National Reactor Testing Station, Idaho Falls, Idaho. In general, the objective of these tests has been the identification of failure thresholds and mechanisms for fast reactor oxide fuel under simulated overpower accident conditions. Such realistic experimental information on the behavior of fuel and cladding and the definition of the transient failure limits of typical LMFBR fuel geometries are vital to the orderly design and analysis of this reactor concept in order to reach the required goal of safety and licensability.

This report describes a transient irradiation test series, designated series IV: Axially-Restrained Fuel, in which comparatively long (≈ 14 -inch fuel-column length) axially-restrained fuel specimens were tested. These 0.250-inch-diameter pins contained mixed oxide (Pu-U) O_2 fuel, clad with type 316 stainless steel, and they had not received irradiation prior to the transient test, i.e., they had zero-burnup.

2.2 PURPOSE OF THE TEST SERIES

The basic purpose of these tests was to generate empirical data and resulting analytical correlations for use in evaluating the transient overpower limits of LMFBR fuel specimens as a function of fuel column axial restraint, fuel density, and fuel form (coprecipitated pellet versus Vipac powder fuel). A continuation of this series (now in progress) will evaluate the effect of prior steady-state, fast-flux irradiation on the transient overpower limits of fuel specimens with identical parameters. The series IV tests represent a continuation of transient overpower experiments (series I, II, and III¹⁻⁴) that have been of an increasingly complex nature, approaching conditions prototypic of those calculated for an actual fuel specimen, in the core of an LMFBR during severe transient overpower conditions.

The actual magnitude of each of the transient irradiations of the series IV fuel specimens was chosen with the intention of producing test results that would bracket the cladding-breach failure threshold. Based on the experience gained in previous tests,¹⁻⁴ it was expected that this threshold would be in the realm of high fuel volume melt fractions (greater than 50% of the fuel cross section becoming molten) and would be primarily due to differential thermal volumes of the fuel, entrapped gas, and cladding.

2.3 EXPERIMENTAL PROCEDURE

All of the zero-burnup fuel specimens of this test series were subjected to high-power, short-term transient irradiation (pulse half-widths from $\approx 3/4$ to 1 second, peak fuel specimen power from ≈ 100 to 200 kW/ft; table 4-1) at the TREAT reactor⁵ while encapsulated in instrumented, sodium-filled capsules. During the transient irradiation, reactor power and the output of capsule thermocouples were recorded as a function of time on high-speed oscillographs and a magnetic tape unit.

The transient irradiation of each fuel specimen occurred in two phases. The first phase consisted of a comparatively low-power transient irradiation which raised peak fuel temperature to the range of $\approx 2000^\circ\text{F}$ to 3000°F . This irradiation, called the calibration transient, was used to check the calculated ratio of test specimen power to the TREAT reactor power [the power factor (watts/cm³ fuel)/TREAT watt*] and to determine the thermal response of the test capsule. Following the calibration transient and subsequent calculations, each specimen was irradiated a second time at a much higher power, resulting in peak fuel temperatures reaching or exceeding the melting temperature of the fuel (assumed $\approx 5200^\circ\text{F}$ for calculations in this report). If this second transient failed to attain test objectives, additional irradiations were performed. Typically, the accumulated burnup resulting from these series of transient irradiations is <0.05 MWd/Te per specimen. This amount of burnup is insufficient to disturb the assumption of zero fission-product inventory, i.e., zero-burnup.

Following completion of the transient irradiations, each fuel specimen was examined in detail at the hot-cell laboratories of the Vallecitos Nuclear Center (GE-VNC). This examination consisted of x-rays, Neutrography,** and dimensional measurements (collectively termed nondestructive examination). This was followed by destructive examination, which typically included sectioning of the fuel specimen, detailed metallographic examination, and autoradiography of fuel and cladding, plus qualitative and quantitative chemical analysis of the fuel, cladding, fission products, and specimen cover gas.

2.4 FUEL SPECIMEN DESCRIPTION

A total of 12 primary fuel specimens and 2 auxiliary fuel specimens (containing in-fuel thermocouples) were

* The power factor is a ratio representing the fraction of energy produced in the test specimen in comparison to the total energy produced by the TREAT core. The actual power factors used in the analysis of the series IV transient irradiations are presented in table B-1.

** General Electric Company service trademark for neutron radiographs (appendix C).

* Contract Number AT(04-3)-189, Project Agreement 10.

fabricated for the series IV tests. This report deals with the six primary specimens and two auxiliary specimens which contained zero-burnup mixed-oxide fuel (fuel with no prior irradiation history) of $(\text{Pu}_{0.2}, \text{U}_{0.8})\text{O}_{1.97-1.98}$ composition. The remaining six specimens will undergo fast-flux, steady-state irradiation in EBR-II to a burnup of $\approx 50,000$ MWd/Te. They will then be reencapsulated and irradiated in the TREAT reactor.

The six primary specimens covered in this report are divided into three specific fuel densities: pelleted fuel of $\approx 80\%$ and $\approx 90\%$ smeared density (fractional theoretical fuel density based on the interior cladding volume rather than the actual fuel volume) and Vipac powdered fuel of $\approx 84\%$ smeared density. The six EBR-II specimens duplicate these densities to permit comparisons of transient overpower behavior as a function of burnup when the tests on them are completed.

The two small auxiliary fuel specimens contained tungsten-rhenium thermocouples inserted into annular fuel pellets. These two specimens were mounted independently in the same capsules as two of the full-length zero-burnup pins to give direct indication of fuel temperature during the TREAT pulse.

The fuel columns of the primary specimens were restrained axially by a mechanical locking device (extensometer) with a limited collapse capability. As a result, the planned axial thermal expansion of the fuel column was restricted to a given distance established during specimen assembly. The purpose of this axial restraint was to simulate the potential condition in an operating fuel specimen as a result of fuel-cladding interaction (e.g., due to irradiation-induced fuel swelling). As a result of this axial restraint, volume changes resulting from thermal expansion and phase changes in the fuel must be absorbed by the fabricated internal porosity voids plus the volume contained in the fuel-cladding gap. Mixed-oxide fuel (with a high weight fraction of UO_2) should exhibit a volume increase, solidus to liquidus, of $\approx 10\%$ during melting.⁶

The basic designations and parameters for all of the series IV fuel specimens are presented in table 2-1. A schematic of the zero-burnup fuel pin appears in the lower portion of figure 2-1. Fuel column length is 13.5 to 14.2 inches with a UO_2 insulator pellet located at the base of the fuel column, and $\approx 4\text{--}3/4$ inches of UO_2 blanket located at the top of the fuel column. The limited-travel extensometer is mounted on top of the blanket, and, depending on initial compression, it is designed to act as a rigid axial restraint for the fuel column above a predetermined axial expansion of the fuel column (see table 4-1 in section 4). The remaining portion of the fuel pin forms the gas plenum.

A more detailed as-fabricated fuel description appears as table A-2 of appendix A. In addition, the assembly drawings for the zero-burnup fuel specimens, the auxiliary fuel specimens, and the extensometer appear as figures A-1 through A-4 of appendix A.

2.5 TREAT CAPSULE DESCRIPTION

As mentioned above, the series IV fuel specimens were irradiated in the TREAT reactor in an instrumented, sodium-filled capsule. For the purposes of the test series, the primary design criterion for this TREAT capsule was to provide the proper transient thermal resistance and effective energy storage capacity to enable the TREAT test specimen to simulate the reaction of an actual fuel specimen in an LMFBR core during a transient overpower accident. The purpose of this design criterion was to ensure that the transient temperatures or energy densities in the fuel, cladding, and the adjacent sodium coolant, plus the extent of fuel melting in the test fuel specimen, would closely match in magnitude, spatial distribution, and time scale, the comparable parameters expected in an LMFBR fuel specimen undergoing transient overpower. Comparative calculations based on these series IV TREAT tests, using the THTD transient heat transfer code,⁷ a hypothetical ramp excursion accident in an LMFBR core, and the FORE-II heat transfer-neutronics coupled code,⁸ indicate this matching criterion was satisfied by the TREAT capsule design.⁹

Several imposed design considerations were accounted for in selecting the final design:

- The TREAT reactor is graphite moderated with a highly thermal neutron spectrum. As a result, the series IV fuel specimens have a significant radial neutron flux depression through the fuel (a surface to centerline flux ratio of ≈ 2.8 ; see figures B-1 and B-2 in appendix B).
- Power transients in the TREAT reactor must be initiated from approximately zero power (isothermal conditions).
- The entire experimental capsule must be compatible with the physical size limitations of the central test hole of the TREAT reactor core (≈ 4 inches square by ≈ 9 feet in length).
- Satisfaction of TREAT hazards analysis requirements for experimental tests¹⁰ required double containment of the test fuel specimens, with the outer containment serving as a pressure vessel.

The basic configuration of the series IV TREAT capsule is shown schematically in figure 2-1. The outer containment is a heavy-walled, carbon steel vessel, with a length and basic outer envelope identical to that of a TREAT fuel assembly. Inside of this vessel is a sealed stainless-steel inner capsule, surrounded by a heater can with an insulating jacket. This sodium-filled inner capsule contains the fuel pin surrounded by an annular aluminum heat sink, which results in an inner sodium annulus, adjacent to the fuel pin, and an outer sodium annulus, adjacent to the wall of the inner capsule. Both of these annuli are instrumented with chromel-alumel thermocouples (three sets equally spaced axially in the fueled region). The response of these thermocouples, during the transient irradiation, is used to

Table 2-1
TASK C SERIES IV
FUEL SPECIMEN DESIGNATION AND BASIC DESIGN PARAMETERS

	SPECIMEN†	FUEL COMPOSITION	FUEL FORM	SMEARED DENSITY (% TD)	FUEL COLUMN LENGTH (inches)	COMMENTS
ZERO-BURNUP FUEL SPECIMENS*	C4A	$(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{1.98}$	solid pellet	90.4	14.2	Transient irradiation has been completed ↓
	C4B			91.2	14.2	
	C4C			80.4	14.2	
	C4D	$(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{1.97}$	Vipac powder	80.1	14.2	
	C4I			83.5	13.5	
	C4J			83.5	13.5	
PRE-IRRADIATED FUEL SPECIMENS	C4E	$(\text{Pu}_{0.28}\text{U}_{0.72})\text{O}_{1.97}$	solid pellet	80.5	13.5	Transient irradiation scheduled in 1970-72 ↓
	C4F			80.5	13.5	
	C4G			90.5	13.5	
	C4H		Vipac powder	90.4	13.5	
	C4K			84.1	13.5	
	C4L			84.2	13.5	
INSTRUMENTED FUEL SPECIMENS*	C4A-II	$(\text{Pu}_{0.2}\text{U}_{0.8})\text{O}_{1.98}$	solid-annular pellet	90.1	3.0	Auxiliary fuel specimens, each of which contained a central in-fuel thermocouple (W5Re/W26Re). Transient irradiation has been completed.
	C4C-II			nominal 80.4 nominal	3.0	

* These fuel specimens are covered in this report.

† All of the series IV fuel specimens were clad with type 316 stainless steel welded tubing, 0.250-inch outer diameter by 0.015-inch wall thickness.

Figure 2-1. TREAT Transient Test Assembly (Task C Series IV)

determine the time-dependent heat flux through the heat sink. This information is then used to reconstruct the radial temperature history of the fuel during the transient.

A typical cross section through the fueled region of the TREAT capsule is shown in figure 2-2. The complete series IV TREAT capsule is shown in assembled views in figure A-5 of appendix A. There are a total of eleven thermocouples in the complete capsule that contains both the main and auxiliary fuel specimens. The designation and location of these thermocouples (with respect to the TREAT core midplane) is shown in table A-1 in appendix A.

2.6 SCOPE OF THIS REPORT

This report covers that portion of the series IV transient tests involving irradiation of the zero-burnup fuel

specimens. The report will concentrate on the observed experimental results through the destructive examination, including as-fabricated specifications for the fuel specimens and actual transient irradiation data from both the TREAT reactor and the test capsule, and the basic post-transient examination data, such as radial location of the fuel-melt boundary, fuel movement, and absence or extent of cladding damage.

Subsequent reports will be issued as appropriate to cover detailed failure mechanisms and the results of the transient irradiation of the pre-irradiated specimens following their steady-state irradiation to $\approx 50,000$ MWd/Te in EBR-II. Two of the six specimens have completed EBR-II steady-state irradiation ($\approx 57,600$ MWd/Te burnup), and are being prepared for TREAT irradiation.

SECTION		MATERIAL	INNER RADIUS (in.)	OUTER RADIUS (in.)
①	FUEL	MIXED-OXIDE	0	0.110
②	CLADDING	316 STAINLESS STEEL	0.110	0.125
③	INNER ANNULUS	SODIUM	0.125	0.1625*
④	HEAT SINK	6061 ALUMINUM	0.1625*	0.4625
⑤	OUTER ANNULUS	SODIUM	0.4625	0.5135
⑥	CAPSULE WALL	304 STAINLESS STEEL	0.5135	0.5625
⑦	GAP	AIR	0.5625	0.590
⑧	HEATER CAN	304 STAINLESS STEEL	0.590	0.625
⑨	INSULATOR	THERMOFLEX	0.625	1.072
⑩	INSULATOR CAN	304 STAINLESS STEEL	1.072	1.110
⑪	GAP	AIR	1.110	1.125
⑫	VESSEL LINER	GRAPHITE	1.125	1.230
⑬	GAP	AIR	1.230	1.250
⑭	PRESSURE VESSEL WALL	4130 CARBON STEEL	1.250	1.500

* 0.200 inch FOR CAPSULES CONTAINING FUEL SPECIMENS C4I AND C4J

Figure 2-2. Capsule Cross Section (Task C Series IV)

3. EXPERIMENTAL AND CALCULATED RESULTS

3.1 TRANSIENT DATA

During the transient irradiation of a typical series IV fuel specimen, two high-speed oscillographs and a magnetic tape unit were used to record instantaneous TREAT core power (in megawatts) and integrated power (in megawatt-seconds),^{5 11} as well as the output of the thermocouples contained in the experimental capsule itself. These data represent the basic transient information used for the post-transient analysis of the events which occurred within the capsule and fuel specimen during and shortly after the reactor power pulse. The TREAT core instantaneous and integrated power are converted to values applicable to the test specimen through use of the specimen power factor (subsection 2.3).

3.2 UNCERTAINTIES IN TREAT TEST PARAMETERS

Several uncertainties can have a significant effect on the planning and performing of TREAT transient irradiation experiments. For example, prior to each test, the fuel specimen power factors are calculated from a theoretical treatment of the TREAT neutron spectrum and its interaction with the constituents of the test capsule. In addition, there are inherent uncertainties in the effective TREAT scram time following scram initiation signal and pre-transient uncertainties in the actual reactivity addition that is inserted into the TREAT core during a transient irradiation. These uncertainties and the ensuing corrective measures are discussed in appendix D.

3.3 DATA UTILIZATION

In order to maximize the usefulness of the experimental data, the series IV capsule was designed with an insulating jacket which permits the inner capsule to act as a calorimeter during the transient portion of the experiment. Ten to thirteen seconds after initiation of the transient, more than 95% of the total energy generated within the fuel has been absorbed by the inner capsule in reaching a new quasi-isothermal condition. This temperature rise of the inner capsule, acting as a calorimeter, provides a direct verification of the total energy generated by the test specimen during the transient. The combination of this indicated energy release with the reactor power pulse and reactor total energy release results in an experimental determination of the effective power factor for the test specimen (subsection 2.3) and the resultant time dependent energy generation rate in the fuel of the specimen.

Through use of the insulating jacket and a layer of spiral-wrapped resistance heaters (see figure A-5 in appendix A), the inner capsule can be stabilized at a pre-transient isothermal condition (with a typical $\pm 1.5^\circ\text{F}$ variation radially and axially in the fueled region of the inner capsule). The resistance heaters are shut off by the signal

which actuates the TREAT power pulse. For the series IV irradiations, this preheat temperature (table 3-1) ranged from 300° to 800°F . The choice of the initial temperature for a test transient was primarily dependent upon the combined requirements of a given peak cladding temperature range and a desired maximum fuel melt volume during a specific planned TREAT power pulse.

As mentioned in subsection 2.5, each pair of capsule thermocouples were located radially in the inner and outer sodium annuli, separated by the annular aluminum heat sink. During the transient the differential response of the thermocouples in the inner and outer annuli effectively represented the transient heat flux through the heat sink. By simultaneously matching the response of both of these thermocouples with computer calculations, it was possible to reconstruct the radial temperature history of the fuel and cladding throughout the transient (see subsection 3.4 and appendix B).

3.4 MATCHING EXPERIMENTAL DATA WITH CALCULATIONS

3.4.1 Calculational Model

In order to arrive at a calculational fit of the experimental data, it was necessary to put into the THTD heat transfer code⁷ the following information:

- the radial capsule geometry: Because of the effectiveness of the series IV inner capsule as a calorimeter, the geometry used in the THTD calculations included only this inner capsule (figure 2-1), with the outer surface of the inner capsule assumed to be ideally insulated. The basic THTD geometry is shown in figure 3-1.
- the thermal-physical properties for the constituents of the inner capsule: This included density effects on fuel energy storage capacity and thermal conductivity and the heat of fusion for each solid-phase constituent (see appendix B for actual THTD input data).
- the radial power profile in the fuel (figures B-1 and B-2 in appendix B)
- the TREAT reactor power versus time curve (in megawatts)
- the fuel specimen power factor (subsection 2.3)
- the heat transfer coefficient of the fuel-cladding gap (h_{gap}): This is the only heat transfer coefficient required since the sodium coolant exterior to the cladding is virtually stagnant, and hence there is only conductive heat transfer from the cladding to the sodium. For the calculations appearing in this report, constant values for h_{gap} were used for each individual case (h_{gap} was constant, but optimized for each case; see table B-1).

Figure 3-1. Typical Series IV TREAT Capsule THTD Heat-Transfer Code Geometry

3.4.2 Available Experimental Data for Calculational Correlations

As indicated in subsections 3.1 and 3.2, there are four basic data outputs (see figure 3-2) monitored during the actual transient test that are of concern to the transient analysis. They are the following:

- reactor power versus time
- integrated fuel specimen energy output (indicated by overall temperature rise of the inner capsule)
- response of the capsule inner sodium annulus thermocouples
- response of the capsule outer sodium annulus thermocouples.

In addition, an important fifth experimental output is the radial extent of fuel melting (the fuel melt radius), which cannot be determined until the destructive examination of the fuel specimen. This information must be obtained from measurements of sections of the fuel specimen that have been metallurgically mounted, polished, and etched.

It was determined from the thermocouple responses, plus the lack of axial variations in the fuel melt radius, that there was no significant variation in the axial power profile over the length of the series IV fuel column. Therefore, for calculational purposes, the axial power profile is assumed flat and the problem is reduced to one dimension (axial and circumferential conduction ignored).

3.4.3 Criteria for Heat Transfer Correlations

A computer-generated fit to the experimental data was considered sufficient when the calculations fell within the following variations with respect to the experimental data:

- (a) The calculated total temperature rise of the inner capsule (i.e., the indicated total energy input), must match the experimental temperature rise to within $\pm 2\%$. This temperature rise was effectively completed between 12 to 15 seconds following initiation of the transient.
- (b) The calculated transient temperature response of both the inner and the outer sodium annuli must agree with the comparable experimental thermocouple responses to within $\approx 10\%$ (relative to the total transient temperature rise of the capsule, typically $200-300^\circ\text{F}$) throughout the effective transient response time of the capsule. The calculated fit can be adjusted by varying either the fuel specimen power factor or the fuel-cladding heat-transfer coefficient.
- (c) The calculated fuel melt radius must agree with the measured melt radius to within $\pm 2\%$.

When these three criteria were met by varying the calculational inputs for specimen power factor and h_{gap} , the calculated simulation was considered to adequately represent the true radial temperature history of the fuel and cladding throughout the transient test.

Figure 3-2. Typical Transient Response of the Series IV Capsule to a TREAT Transient Irradiation

3.4.4 Complicating Factors in Correlations

Complicating factors entered into the ability to correctly calculate the response of the thermocouples located in the inner and outer sodium annuli [item (b) of subsection 3.4.3]. As an example, radial temperature gradients through the inner capsule could introduce convective currents in the inner sodium annulus. This would upset the assumption of a pure conduction solution form. Also the physical size (1/16-inch diameter) and the exact radial location of the junction for the inner annulus thermocouples can have perturbative effects on the idealized solution form.

In both of these example effects, the outer sodium annulus thermocouple responses are far less likely to be significantly perturbed, since these thermocouples reside in a region which rarely sees significant radial temperature gradients. This is a result of being effectively shielded from the fuel specimen by the time constant of the intervening aluminum heat sink (figure 3-1 and figures 2-1 and 2-2 in section 2). Therefore, greater reliance was given to a calculated match of the experimental response of the thermocouples in the outer sodium annulus than to those in the inner sodium annulus.

Additionally, some error in calculated response arose from the assumption of a constant fuel-cladding gap coefficient throughout the transient irradiation. The actual h_{gap} would initially be quite low, due to the existence of a fuel-cladding diametral gap in this zero-burnup fuel at the initial isothermal conditions. This gap would close, probably at a quasi-exponential rate, following the reactor power rise, with a resultant high h_{gap} for the latter portion of the transient.

The effect of this increasing h_{gap} was apparent in a comparison of the experimental thermocouple response to the calculated history (where an intermediate value of h_{gap} was used). The calculated results for the inner annulus temperatures initially lead the measured response, indicating a true h_{gap} lower than that assumed in the calculations. As the transient progressed, the slope ($\Delta T/\Delta t$) of the measured response exceeded that of the calculated response, indicating that the true h_{gap} had exceeded the value in the calculations. Finally, near the initial peak in the inner annulus temperature, the measured response would often cross over and lead the calculated response. This result is shown schematically in figure 3-2 together with the corresponding responses for the outer annulus. This general pattern can be seen throughout the graphical presentations of the thermocouple data for the series IV zero-burnup (figures 3-3 through 3-20, beginning on page 3-8).

3.5 TRANSIENT IRRADIATION RESULTS

3.5.1 TREAT Reactor Parameters

The parameters of the TREAT power pulses used to irradiate the series IV zero-burnup fuel specimens are given in table 3-1. The last column of table 3-1 refers to natural and clipped transients. The natural transient power pulse has a self-limited peak power due primarily to temperature dependent reactivity feedback mechanisms of the TREAT core and containment. In the clipped transient power pulse, the reactor is scrammed prior to reaching this self-limited power level.

3.5.2 Temperature Histories

The sodium annuli temperature histories for the transient irradiation of the capsules containing the series IV zero-burnup fuel specimens are presented in graphical form in figures 3-3 through 3-20. The TREAT data are presented as a series of individual datum points for the thermocouple readings in the inner and outer sodium annuli. The calculated data for the two sodium annuli are presented as solid lines overlaying the TREAT data. Generally, only two experimental datum sets (one for each annulus) are shown on each graph, although there were six thermocouples available. Normally, the thermocouple pair located at the fuel column midplane was chosen as representative of the transient data. If one of these two thermocouples failed, the output of another pair, or the inner and outer annulus averages for the remaining thermocouples, was used for data interpretation. Some of the calibration transient correlations (figures 3-3 to 3-8) were calculated by the ARGUS transient-heat-transfer code;¹² otherwise, all of the correlations were attained using the THTD code.

As mentioned earlier, fuel specimens C4A, C4B, and C4J failed during their test transient irradiation. As a result of these failures, there were severe distortions in the corresponding sodium annuli temperature histories. In the case of both specimens C4B and C4J, uncertainties were introduced in the measurement of the effective power factor as a result of post-failure movement of fuel (which distorts the axial linearity of the fuel energy release). As a result, the THTD correlations for the failed specimens depended heavily upon matching the measured fuel melt radius, plus the power factor determined from calibration transients and/or that corresponding to the companion specimen (e.g., for specimen C4J, C4I is an identical companion). As a result, data correlation for the failed specimens [particularly specimens C4B and C4J (see figures 3-11, -12, -19, and -20)] was less effective than for the remaining transient analyses, but the resulting radial fuel and cladding temperature profiles do provide a fairly accurate representation of the intact geometry conditions up to the time of failure.

Table 3-1
TREAT REACTOR PARAMETERS FOR THE SERIES IV
ZERO-BURNUP TRANSIENT IRRADIATIONS

Fuel Specimen	Transient Purpose	Treat Transient Number	Specimen Preheat Temperature (°F)	Initial Period (seconds)	Reactivity Addition (% Δk)	Peak Reactor Power (MW)	Integrated Reactor Power (MW-sec)	Type of Transient
C4A	0*	1088	404	0.608	0.76	28	78	natural
	1**	1093	585	0.176	1.18	246	202	natural
C4B	0	1051	300	0.465	0.82	38	56	natural
	1	1057	800	0.155	1.26	339	207	natural
C4C	0	1079	412	0.565	0.78	29	79	natural
	1	1087	585	0.161	1.24	300	115	clipped before peak power
	2***	1094	604	0.190	1.14	231	186	natural
C4D	0	1058	418	0.549	0.78	36	78	natural
C4C-II	1	1078	606	0.226	1.06	164	159	natural
C4I	0	1195	413	0.634	0.75	28	80	natural
C4A-II	1	1199	617	0.582	0.76	36	88	unintentional undershoot
	2	1203	589	0.252	1.02	128	156	natural
C4J	0	1198	395	0.590	0.76	29	78	natural
	1	1204	606	0.197	1.13	206	177	natural

* 0 = calibration transient

** 1 = first test transient

*** 2 = second test transient (if needed)

3.5.3 Calculated Radial Temperature Profiles

Table 3-2 presents the calculated peak fuel and cladding temperatures for the zero-burnup tests. In addition, the actual fuel melt radius is compared to the calculated fuel melt radius. The fractional fuel volume melt also

appears in this table.

The summarized development of the calculated radial temperature profiles of the fuel and cladding for the test transients are presented in graphical form in figures 3-21 through 3-27.

Table 3-2
CALCULATED MAXIMUM FUEL AND CLADDING TEMPERATURES
AND COMPARISON OF CALCULATED AND ACTUAL FUEL MELT RADII

	Transient	Calculated Peak Fuel Temp (°F)	Calculated Peak Cladding Temp (°F)	Calculated Fuel Melt Radius ¹ (mils)	Measured Fuel Melt Radius ² (mils)	Measured Fuel Volume Melt ³ (%)	Comments
High Density Pellet Fuel	C4A-0*	≈2150	-	-	-	-	
	C4A-1**	≈5500	1400	≈86	82-87	56-63	specimen failed, moderate fuel movement
	C4B-0	≈2020	-	-	-	-	
	C4B-1	≈5800	1580	96-97	96-98	76-79	specimen failed, extensive fuel movement
Low Density Pellet Fuel	C4C-0	≈2900	-	-	-	-	
	C4C-1	≈4900	1110	-	-	-	
	C4C-2***	≈5600	1260	90-91	90-92	67-70	unfailed, significant porosity movement in the molten region
	C4D-0	≈3000	-	-	-	-	
	C4D-1	5200 (melt temp)	1180	77-78	77-78	≈50	
Vipac Powder Fuel	C4I-0	≈2500	-	-	-	-	
	C4I-1					(accidental undershoot, not correlated)	
	C4I-2	5200 (melt temp)	1120	61-62	≤60	≤30	localized melting and porosity movement accompanied by grain growth across 60 mils radius
	C4J-0	≈2600	-	-	-	-	
	C4J-1	≈5480	1280	83-85	78-86	49-60	specimen failed, very extensive fuel movement

NOTES:

¹ Defined as the maximum radius that the fuel reaches the thermal arrest associated with the melt temperature (assumed 5200°F). This region may include fuel which has not completely passed through its heat of fusion, even though that fuel has reached 5200°F.

² Defined as the extent of significant restructuring (usually dendritic substructure indicative of resolidification) and porosity movement as measured during destructive examination. The indicated ranges of melt radius are a result of multiple measurements at different axial and/or angular locations along the length of the fuel column.

³ The fuel volume melt is based on the ratio of the cross-sectional area of the molten region of the fuel to the cross-sectional area of the interior cladding (calculated from the cladding as-fabricated inner diameter).

* 0 = calibration transient

** 1 = first test transient

*** 2 = second test transient (if needed)

3.6 POST-TRANSIENT NONDESTRUCTIVE EXAMINATION

3.6.1 X-ray and Neutrography

After completion of the TREAT irradiation, the test capsules were returned to the GE Vallecitos Nuclear Center for disassembly and examination.

The first examination performed was x-raying of the fueled portion of the TREAT capsule, which indicated the occurrence or absence of cladding failure and provided assurance that the inner capsule had maintained integrity. An example of capsule x-radiography is presented in figure 3-28 which shows the post-transient condition of the failed specimen C4A.

Having obtained assurance of inner capsule integrity, the inner capsule was removed from the pressure vessel and transferred to the Nuclear Test Reactor (NTR) at Vallecitos, for Neutrography. The advantages of Neutrography over x-radiography are discussed in appendix D.

The post-transient Neutrographs for the six fuel specimens appear in figures 3-29 through 3-34 (the view of specimen C4A in figure 3-29 may be compared to that taken by x-rays in figure 3-28).

3.6.2 Dimensional Measurements

Table 3-3 presents a summary of the dimensional changes noted as a result of transient irradiation of the series IV zero-burnup fuel specimens. It is significant to note that dimensional changes in cladding diameter and fuel column length were minor or nonexistent (with exception

of the localized region of specimen C4A referred to in the footnotes of table 3-3).

The dimensional changes noted in table 3-3 include both diametral and axial changes. The diametral deformations are based on a comparison of pre- and post-transient profilometer measurements (resolution <0.1 mil) over the entire length of the fuel specimen cladding. For the failed specimens, any obtainable sections of undamaged cladding were used in the profilometer measurements.

The fuel column length changes were also determined by comparative pre- and post-transient measurements. The retained fuel column length changes were directly measurable as a permanent change in the length of the extensometer located on top of the UO₂ blanket (figure 2-1). It was originally planned to monitor transient length changes in the fuel column using a follower located in each extensometer (figures A-3 and A-4 in appendix A). Difficulties were encountered, however, in properly positioning these followers during fabrication. In those instances where proper positioning was achieved, there was no evidence of fuel column extension during the transient.

The lack of evidence of either transient or retained axial growth of the series IV zero-burnup fuel columns indicates that the frictional forces induced between the fuel and cladding, due to differential thermal expansion, were sufficient to impose a self-restraint on the fuel columns. This effective self-restraint has not been apparent in previous tests of zero-burnup specimens with smaller length-to-diameter ratios.^{1 2}

Table 3-3
SUMMARY OF TRANSIENT INDUCED DIMENSIONAL CHANGES,
SERIES IV ZERO-BURNUP FUEL SPECIMENS

Fuel Specimen	Cladding Diametral Deformation				Permanent Change in Fuel Column Length	
	Average		Peak		inch	%
	mils	%	mils	%		
C4A	1.15	0.46	8-9	3.2-3.6*	0.28	≈2.0
C4B	—	—	≈1.0	≈0.4**	≈0.08	≈0.55
C4C	<0.5	<0.2	≈1.0	≈0.4	negligible	
C4D	negligible		—	—	negligible	
C4I	≤0.3	≤0.1	≈1.0	≈0.4	negligible	
C4J	—	—	1-2	0.4-0.8**	negligible	

* This peak deformation occurred locally (≈1-1.5 inches long) at the upper end of the fuel column. See discussion on specimen C4A, subsection 4.5.1.

** Location and extent of cladding failure and resultant fuel displacement precluded conclusive measurement.

Figure 3-3. Temperature Histories: Inner and Outer Sodium Annuli--Specimen C4A Transient Irradiation Results (Calibration Transient No. 1088)

Figure 3-4. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4B Transient Irradiation Results (Calibration Transient No. 1051)

Figure 3-5. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4C Transient Irradiation Results (Calibration Transient No. 1079)

Figure 3-6. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4D Transient Irradiation Results (Calibration Transient No. 1058)

Figure 3-7. Temperature Histories: Inner and Outer Sodium Annuli-Specimen C4I Transient Irradiation Results (Calibration Transient No. 1195)

Figure 3-8. Temperature Histories: Inner and Outer Sodium Annuli-Specimen C4J Transient Irradiation Results
(Calibration Transient No. 1198)

Figure 3-9. Temperature Histories: Inner and Outer Sodium Annuli--Specimen CAA Transient Irradiation Results (Test Transient No. 1093)

GEAP-13662

Figure 3-10. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4A Transient Irradiation Results
(Test Transient No. 1093)

Figure 3-11. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4B Transient Irradiation Results (Test Transient No. 1057)

GEAP-13562

Figure 3-12. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4B Transient Irradiation Results (Test Transient No. 1057)

Figure 3-13. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4C Transient Irradiation Results, First Test Transient (No. 1087)

Figure 3-14. Temperature Histories: Inner and Outer Sodium Annuli--Specimen C4C Transient Irradiation Results, Second Test Transient (No. 1094)

Figure 3-15. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4D Transient Irradiation Results (Test Transient No. 1078)

Figure 3-16. In-Fuel W5Re/W26Re Thermocouple Compared to Calculated Central Fuel Temperature—Specimens C4D and C4C-II Transient Irradiation Results (Test Transient No. 1078)

Figure 3-17. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4I Transient Irradiation Results, Second Test Transient (No. 1203)

Figure 3-18. In-Fuel W5Re/W26Re Thermocouple Compared to Calculated Central Fuel Temperature—Specimens C4I and C4A-II Transient Irradiation Results, Second Test Transient (No. 1203)

Figure 3-19. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4J Transient Irradiation Results (Test Transient No. 1204)

Figure 3-20. Temperature Histories: Inner and Outer Sodium Annuli—Specimen C4J Transient Irradiation Results, (Test Transient No. 1204)

Figure 3-21. Calculated Radial Temperature Profiles Specimen C41 Transient Irradiation Results
(Test Transient No. 1093)

Figure 3-22. Calculated Radial Temperature Profiles—Specimen C4B Transient Irradiation Results (Test Transient No. 1057)

Figure 3-23. Calculated Radial Temperature Profiles- Specimen C4C Transient Irradiation Results, First Test Transient (No. 1087)

Figure 3-24. Calculated Radial Temperature Profiles—Specimen CAC Transient Irradiation Results, Second Test Transient (No. 1094)

Figure 3-25. Calculated Radial Temperature Profiles- Specimen C4D Transient Irradiation Results (Test Transient No. 1078)

Figure 3-26. Calculated Radial Temperature Profiles Specimen CA1 Transient Irradiation Results, Second Test Transient (No. 1203)

Figure 3-27. Calculated Radial Temperature Profiles - Specimen CAJ Transient Irradiation Results (Test Transient No. 1204)

Figure 3-28 Fuel Specimen C4A Post-Transient X-Radiograph

Figure 3-29 Fuel Specimen C4A Post-Transient Neutrograph

Figure 3-30. Fuel Specimen C4B Post-Transient
Neutrograph

Figure 3-31 Fuel Specimen CAC Post-Transient Neutrograph

Figure 3-32 Fuel Specimens C4D and C4C-II
Post-Transient Neutrograph
3-41/3-42

Figure 3-33 Fuel Specimens C4I and C4A-II
Post-Transient Neutrograph

Figure 3-34 Fuel Specimen C4J Post-Transient Neutrograph

4. DISCUSSION OF RESULTS AND CONCLUSIONS

4.1 COMPARATIVE TEST RESULTS

Table 4-1 presents a summary of the transient irradiation results for the six zero-burnup, axially-restrained fuel specimens of series IV. For comparison, the same transient summary is included for the three series II zero-burnup fuel specimens that were axially unrestrained.¹ Of the previous task C tests,¹⁻⁴ these three series II specimens provide the best comparison to the series IV transient conditions. The following important features are comparatively tabulated in table 4-1:

- (a) The primary as-fabricated variables encompassed by these specimens were:
 - Fuel column length Series II $\approx$ 6 inches
Series IV $\approx$ 14 inches
 - Presence of a device for mechanically limiting the axial movement of the series IV fuel columns.
 - Fuel form, pellet versus powder.
 - Fuel smeared density
- (b) The peak instantaneous power generated by several of the specimens was greater than ten times the probable peak steady-state power level of a specimen in an LMFBR 1000 MW core.
- (c) Cladding failure occurred in three pellet fuel specimens [C4A, B, and C2C (all had $\geq 90.4\%$ smeared density fuel)] and in powder fuel specimen C4J (83.5% smeared density).
- (d) The fuel failure threshold is apparently a function of both fuel smeared density and volume melt fraction,

but, in addition, may also involve axial restraint and fuel form.

- (e) Significant fuel movement was associated with every case of cladding failure.
- (f) No significant cladding deformation was detected in either the unfailed specimens or in the accessible unfailed regions of cladding in the failed specimens (with the exception of a local region in specimen C4A where there was contact of molten fuel against the inner surface of the cladding; see subsection 4.5.1).
- (g) No central void was formed in any fuel column in which fuel did not pass completely through the heat of fusion associated with melting of the fuel.

The total energy release for each of the specimens has been included in the table for completeness, but intercomparison of these numbers is only valid for specific cases. Differences in the power factor between groups of specimens (i.e., C4A and B; C and D; I and J; and C2A-C) can result in similar peak powers and total energy releases, but the effective energy input rate and hence, net instantaneous energy density of the fuel (energy input to fuel minus energy lost from the fuel to the cladding and inner capsule constituents), can vary significantly due to variations in initial TREAT reactor periods (table 3-1). For example, the highly effective heat absorption rate of the capsule sodium and heat sink results in substantial heat transfer from the specimen during the main portion of the TREAT power pulse. Typically, 60% to 80% of the integrated energy production would be effectively residing in the fuel itself at any given instant during the transient power pulse.

Table 4-1
SUMMARY OF DESIGN PARAMETERS AND TRANSIENT IRRADIATION RESULTS

	Series IV Axially Restrained Zero-Burnup Pelleted Fuel Specimens				Series IV Axially Restrained Zero-Burnup Powder Fuel Specimens		Series II Unrestrained Zero-Burnup Pelleted Fuel Specimens*		
Specimen	C4A	C4B	C4C	C4D	C4I	C4J	C2A	C2B	C2C
Smeared Density (% TD)	90.4	91.2	80.4	80.1	83.4	83.5	90.4	90.6	91.8
Fuel Column Length (in.)	14.2	14.2	14.2	14.2	13.5	13.5	6	6	6
Cladding	316 stainless steel, 0.250-in. o.d. X 0.015-in. wall thickness				316 stainless steel, 0.250-in. o.d. X 0.015-in. wall thickness		347 stainless steel, 0.250-in. o.d. X 0.015-in. wall thickness		

SOURCE:

- * J. E. Hanson, J. H. Field, and S. A. Rabin, "Experimental Studies of Transient Effects in Fast Reactor Fuels Series II—Mixed Oxide (PuO₂-UO₂) Irradiations," GE-BRDO, June 1965, (GEAP-4804).

Table 4-1 (Continued)

	Series IV Axially Restrained Zero-Burnup Pelleted Fuel Specimens				Series IV Axially Restrained Zero-Burnup Powder Fuel Specimens		Series II Unrestrained Zero-Burnup Pelleted Fuel Specimens*		
Average Fuel- Cladding Diametral Gap (mils)	4.3	4.5	3.6	4.6	0	0	5	5	4.5
Designed Axial ^a Restraint (°F)	>5200 (≈melt)	5200	>5200	>3000	>5200	>5200	none	none	none
Transient Peak Specimen Power (kW/ft)	148	204	139	93	87	138	134	211	200
Total Specimen ^b Energy Release (kW-sec/ft)	122 ±10	124 ±20	112 ±10	90 ±8	106 ±10	119 ±20	118	146	198
Calculated Maximum Fuel Temperature (°F)	5500 ±600	5800 ±600	5600 ±600	5200 ±300	5200 ±300	5500 ±600	4900 ^c	5000 ^c	7500 ^d
Fuel Volume ^e Melt (%)	56-63	76-79	67-70	≈50 ^f	≤30 ^f	49-60	≈0 ^f	≈72	≈80 ^a
Central Void Formed	yes	yes	yes	no	no	yes	no	yes	yes
Fuel Movement	mod- erate ^g	exten- sive ^h	mod- erate	no	no	exten- sive ⁱ	no	mod- erate	exten- sive
Average Cladding Deformation (mils) ^e	1-15	>1 ^j	<0.5	no	<0.3	1-2 ^j	no	no	no
Cladding Failure	yes ^g	yes ^h	no	no	no	yes ⁱ	no	no	yes

NOTES

- ^a Calculated
^b A fuel specimen of 100% effective smeared density would generate ≈290 cal/gm for every 100 kW-sec/ft of energy production
^c Fuel melt temperature in the original series II analysis was assumed ≈4900°F
^d Probably an excessive temperature prediction as a result of calculational techniques used in the series II analysis
^e Measured during destructive examination
^f Calculations indicate fuel reaching melt temperature, but not penetrating the associated heat of fusion
^g Cladding failed during transient irradiation, <7% of fuel relocated outside of cladding **
^h Cladding failed during transient irradiation, ≈30% of fuel relocated outside of cladding
ⁱ Cladding failed during transient irradiation, >50% of fuel relocated outside of cladding ***
^j Location and extent of cladding failure precluded conclusive measurement

SOURCES

- * J E Hanson, J H Field, and S A Rabin, "Experimental Studies of Transient Effects in Fast Reactor Fuels Series—Mixed Oxide (PuO₂ UO₂) Irradiations," GE-BRDO, June 1965, (GEAP-4804)
** G R Thomas and J H Field, "Destructive Examination of an Axially Restrained Pelleted Mixed-Oxide Specimen that Failed During Transient Overpower Conditions," Trans Am Nucl Soc, 11, 697, (1969)
*** G R Thomas and J H Field, "Transient Performance of Mixed Oxide Powder Fuel with Up to 60% Melting," Trans Am Nucl Soc, 12, 343, (1969)

4.2 CONCLUSIONS

Based on the transient test results and thermal analysis of the six zero-burnup specimens (as summarized in tables 3-3 and 4-1), the following specific conclusions are indicated:

- (a) Comparison of the failed specimens C4A and C4B with C2B indicates that axial restraint of the fuel column and/or increased fuel column length significantly lowers the failure threshold for fuel pins of $\geq 90.4\%$ smeared density.
- (b) No significant changes in fuel column length were noted (permanent or transient), except those associated with molten fuel extrusion. This result indicated that the length of the fuel column, length-to-diameter (L/D) ratio of 61 to 66, was sufficient to provide self-imposed axial restraint during severe transient heating (despite the lack of prior irradiation history and resultant fuel swelling).
- (c) Comparison of specimen C4C with C4A and C4B indicates that, at zero-burnup, a lower smeared density will provide an increased failure threshold for pelleted fuel undergoing a severe accident (50% to 80% molten fuel volume), despite the inherent axial restraint of the fuel column.
- (d) The exact conditions leading to initial cladding failure cannot be observed directly, and verification from post-transient examination was impossible due to additional damage in the original failure region caused by the post-failure movement of molten fuel.
- (e) It is evident from the calculated temperatures of the inner capsule that boiling of the sodium coolant was not a cause of initial cladding failure. The peak cladding temperatures (at the inner cladding surface; table 3-2) were below the saturation temperature of sodium at one atmosphere, and the inner capsule cover gas pressure was two to three atmospheres during transient irradiation.
- (f) The cause of initial cladding failure in the series IV specimens cannot be attributed to contact of molten fuel with the cladding, although such contact can be a potential contributor to the failure mechanism. The evidence indicates that no significant molten fuel-cladding contact occurred in the failure regions prior to failure; this is based on the following:
 - the absence of significant increases in fuel column length (except that region of C4A mentioned above)
 - the post-transient accountability of every fuel pellet in both failed pelleted specimens C4A and C4B (see the neutrographs of these specimens; figures 3-29 and 3-30), even in the failure regions.
- (g) The displacement of molten fuel toward the failure region, from undamaged regions both above and below the failure region, indicates that the molten fuel has a high degree of mobility. The driving force for this extensive movement could be entrapped porosity gas, adsorbed gases contained in the fuel, generation of substantial vapor pressures in the fuel as a result of high temperatures, or a combination of these and other causes.
- (h) Since the fuel of both the series II and series IV tests had no prior irradiation history, and the burnup as a result of the transient irradiation amounts to < 0.05 MWd/Te, the movement of molten fuel cannot be attributed to fission gases released during the transient.
- (i) The forced separation of fuel pellets and resultant lifting of the blanket above the fuel column indicates substantial forces internal to the fuel column when molten fuel is present.
- (j) The fuel volume increase of $\approx 10\%$, as it passes through the heat of fusion, can be a substantial contributor to internal pressure and to the resultant movement of molten fuel. The internal pressure is caused by the volume demand of this sudden fuel expansion and the resulting compression of gases entrapped in the molten fuel region.
- (k) In all three failed specimens, the fuel, whether molten or solid, that was exposed to the sodium coolant showed a high degree of coherency despite evidence of contact under extreme temperature situations. This coherency, as typified by a free-standing fuel wall in the melt-through region of the powder fuel specimen C4J (subsection 4.10.1 and references 13 and 14), indicates a lack of violent molten fuel-cladding-sodium reactions following cladding failure.
- (l) The indicated thermal conductivity of powder fuel, with no prior in-situ sintering history, was significantly above that predicted from UO_2 powder fuel data.^{1,5} Based on the calculated correlations for specimens C4I and C4J, the effective powder fuel conductivity was $\approx 33\%$ higher than that predicted by the powder UO_2 data (or $\approx 15\%$ lower than that predicted for zero-burnup pelleted fuel of comparable density).

4.3 CALIBRATION TRANSIENT RESULTS

The TREAT power histories, transient sodium temperature data, and corresponding computer correlations for the calibration transients are presented in figures 3-3 through 3-8. The calculated fits show very good agreement with the experimental data for all six specimens. The peak calculated fuel temperatures for the calibration transients are presented in table 3-2. The differences in the peak fuel temperatures between the two high fuel density specimens, C4A and C4B, and the remaining four specimens can be mainly attributed to the differences in fuel thermal conductivity due to density effects (see figure B-3 in appendix B).

The ability to fit calibration transients to a high degree of accuracy is primarily because calibration transients are significantly slower and lower in power than are test transients (calibration transient: ≈ 30 MW peak by ≈ 80 MW-sec integrated power; versus test transient: ≈ 300 MW peak by ≈ 200 MW-sec integrated power). As a result, the transient radial temperature gradients through the inner capsule during a calibration transient are much less severe than is the case for the test transients. As indicated in subsection 3.4.4, these reduced temperature gradients should result in a close approach to the ideal calculated solution form for transient heat transfer through the inner capsule.

4.4 TEST TRANSIENT RESULTS, IN GENERAL

The TREAT power histories, transient sodium temperature data, and corresponding computer correlations for the test transients are presented in figures 3-9 through 3-20. For explanation of the important characteristics represented in these curves, see subsections 3.4.2 through 3.4.4 and figure 3.2.

The calculated radial fuel and cladding temperature profiles that correspond to the above calculated fit to the test transient sodium temperature data are presented in figures 3-21 through 3-27. The calculated maximum radial extent of the fuel-melt boundary shows excellent agreement with the physically measured values of melt radius for specimens C4A-D and J, and only slightly overpredicts the melt radius in specimen C4I (see table 3-2).

The characteristic shape of the radial temperature profiles, with peak fuel temperatures appearing at $\approx 3/4$ of the fuel radius, is a result of the large radial flux depression in the mixed oxide fuel (figures B-1 and B-2 in appendix B), combined with the necessity of initiating transients from isothermal zero-power conditions in TREAT. As mentioned in subsection 2.5, a comparison of the transient irradiation of these series IV zero-burnup specimens with a hypothetical ramp excursion accident in an LMFBR⁹ indicates that this characteristic TREAT-generated radial temperature profile can produce a reasonable simulation of hypothetical LMFBR accidents for zero-burnup or low-burnup fuel (i.e., fuel with little radial variation in the porosity or fission gas distribution).

4.5 FUEL SPECIMEN C4A TRANSIENT IRRADIATION

The planned test objective of the C4A irradiation was to obtain a fuel volume melt fraction greater than 50%, yet significantly below the melt fraction of the previously irradiated specimen C4B (76% to 79% volume melt). This test objective was attained with an average fuel volume melt fraction of $\approx 60\%$.

4.5.1 Post-Transient Condition

Specimen C4A suffered cladding failure during a transient irradiation (C4A-1; tables 3-1 and 3-2) which produced $\approx 56\%$ to 63% molten fuel volume. The failure occurred above the fuel-column midplane and resulted in only $\approx 5\%$ of the fuel being dispersed outside of the cladding, over a region 3 inches in length. Additional melting of the cladding that resulted from this fuel relocation prevented positive verification of the primary failure mode. Shown in figure 3-29 is a post-transient neutron radiograph of the fuel and blanket region of specimen C4A with the fuel dispersed outside of the cladding. Nine separate void regions can be identified along the fuel column length.

A comparison of the pre- and post-transient profilometer readings for the lower $8\frac{1}{4}$ inches of the fuel column, representing 58% of the overall fuel column length, indicated an average permanent cladding deformation slightly over 1.1 mils ($\approx 0.46\%$; table 3-3). In addition, there were additional 0.2 to 0.6 mil deformations at the fuel pellet interfaces, caused by pellet ridging. Profilometer readings for the portion of the fuel column above the failure were dominated by a region $\approx 1\frac{1}{2}$ inches long, where permanent cladding deformation was $>3\%$ (a maximum of 8 to 9 mils, with a negative 4 mils at one orientation, indicating a complex ovality).

Post-transient measurement of the fuel column indicated a retained increase in length of ≈ 0.28 inch ($\approx 2.0\%$), which was by far the greatest increase in length of all six specimens tested (table 3-3). The follower located in the extensometer (figures 3-29, A-1 and A-5), indicated that transient expansion was less than an additional 0.05 inch. The permanent fuel column growth was primarily caused by a two-part phenomenon associated with the highly deformed cladding region at the top of the fuel column. At this elevation, the large volume change associated with a fuel-melt change of phase ($\approx 10\% \Delta V/V$) apparently forced a separation of the upper two fuel pellets from the remaining portion of the fuel column. This separation, ≈ 0.18 inch in length, was filled by molten fuel that quenched on the inner surface of the cladding. A second smaller molten fuel-cladding contact occurred when the molten fuel broke through the end of the uppermost fuel pellet and lifted the blanket and additional ≈ 0.03 inch upward from the top fuel pellet.

An examination of the cladding in this region revealed no notable change in the cladding microstructure; however, as mentioned previously, the local permanent cladding deformation was $>3\%$ versus an average of $\approx 0.5\%$ for the unfailed portion of the fuel column.^{16 17} It is believed that the section of cladding contacted by the molten fuel would have failed had not pressure relief been offered as a result

of the actual cladding failure, which occurred below this zone. As a result, this region probably represents a potential failure mechanism which was interrupted at an intermediate stage of development.

4.5.2 Thermal Analysis

Excellent correlation for both the inner and outer sodium annuli temperature histories were obtained for the C4A-1 transient (see figure 3-9). From the response of thermocouples No. 6 and No. 10, the cladding apparently failed ≈ 3.1 seconds after initiation of the transient (thermocouple No. 6 was located directly adjacent to the cladding failure region; see figure 3-10. For thermocouple No. 10; see figure 3-9). An examination of the radial fuel and cladding temperature histories (see figure 3-21) indicates that the failure occurred relatively early in the development of the molten fuel core and well before attainment of peak fuel temperatures at ≈ 3.6 seconds. However, the time of failure does coincide with the maximum radial extent of the fuel-melt boundary and the peak cladding temperature.

4.6 FUEL SPECIMEN C4B TRANSIENT IRRADIATION

Specimen C4B was the first test performed in the series and transient irradiation was planned to closely match the fuel volume melt fraction obtained in the irradiation of specimen C2C (table 4-1 and reference 1). This test result was attained with slightly less than 80% of the fuel molten during the transient.

4.6.1 Post-Transient Condition

Specimen C4B failed during the transient irradiation,^{18 19} which produced a measured fuel volume melt of $\approx 76\%$ to 79% (table 3-2). The cladding was breached and fuel dispersed outside of the pin over a region extending from the fuel column midplane downward for ≈ 5 inches. This expulsion of fuel from the pin was more extensive than occurred in specimen C4A, with $\approx 30\%$ of the total fuel volume relocated outside of the cladding. Most of the expelled fuel apparently came from the central molten core located above the upper limit of the cladding breach. The extent of the fuel movement can be seen in the post-transient neutrographs of the fuel specimen in its inner capsule (figure 3-30). Almost 80% of the fuel volume was missing from the uppermost pellets in the fuel column leaving behind a thin, tapering chimney of fuel with an ≈ 12 mil minimum thickness.

Measurement of transient-induced cladding deformation was seriously hampered as a result of the location and extent of the cladding failure, fuel dispersal, and ensuing heat sink damage. The indication of a slight diametral increase over a short section of cladding was questionable because of the proximity of an examination sectioning cut and the rough handling required to remove the section of cladding from its heat-warped aluminum heat sink.

The retained increase in fuel column length (≈ 0.08 inch; table 3-3) was primarily a result of molten fuel again breaking through the upper end of the top fuel pellet (as was the case for specimen C4A), raising the blanket, and forming an ≈ 0.038 -inch gap filled with molten fuel. The remainder of the length change is accounted for by several small fuel pellet interface gaps.

A resolidified fuel formation, indicating an upward-downward sequence of molten fuel movement, was found in the lower portion of specimen C4B. A similar formation was also seen in the lower portion of the fuel column of specimen C4A. In both cases, the formations were well below the failure zones, indicating the possibility that molten zero-burnup fuel can move upwards towards the failure zone, once cladding failure has occurred. There was negligible fission gas present to cause such movement. These formations are visible near the lower end of the C4A and the C4B fuel columns in the neutrographs shown in figures 3-29 and 3-30.

4.6.2 Thermal Analysis

The larger and more centralized failure region in specimen C4B had a much greater disruptive effect on the response of the sodium annuli thermocouples than was the case for the C4A failure. Most of the C4A thermocouple responses were sufficiently undisturbed that it was possible to determine the power factor from the inner capsule temperature rise (thermocouples No. 8 through No. 11 of figure 3-9). The significant axial realignment of fuel in the C4B failure, however, prohibited an axially independent measure of the inner capsule temperature rise, and hence C4B power factor.

Since fuel specimen C4A was virtually identical to specimen C4B, the two specimens were assumed to have similar power factors, and the C4A power factor was applied to the C4B-1 calculations with excellent results. The final calculations showed a good correlation with the early portion of the temperature rise (prior to failure) in both the inner and the outer sodium annuli of C4B (figures 3-11 and 3-12), and the calculated radial temperature profiles in the fuel showed an excellent match with the measured fuel melt radius (figure 3-22).

The C4B cladding failure apparently occurred shortly after the maximum power, 2.75 to 2.9 seconds (figures 3-11 and 3-12). Beyond this point, the data for thermocouples Nos. 6 and 7 (located well above the region of failure) and thermocouples Nos. 8, 9, and 11 (located in the region of the failure) show considerable deviation from the calculated values. The equilibrium temperatures of thermocouples Nos. 6 and 7 were noticeably below both those predicted from intact geometry calculations and those measured on thermocouples Nos. 8, 9, and 11. This reduction in overall temperature rise was apparently caused by the loss of local heat energy as the molten fuel flowed from the area of thermocouples Nos. 6 and 7, following cladding

failure. Thermocouples Nos. 8, 9, and 11 ranged much higher than the calculated results, reflecting the redistributed excess fuel into their local region. The temperature of thermocouple No. 11 (outer annulus, figure 3-12) indicated complete melting of the aluminum heat sink in this region, which was confirmed during destructive examination. A very similar thermocouple response pattern was noted in the thermal analysis of the failed powder fuel specimen C4J (subsection 4.10.2).

From examination of the calculated radial fuel and cladding temperature profiles (figure 3-22), it appeared that the C4B failure also occurred relatively early in the development of the molten fuel core (as was the case for the C4A failure) and again prior to the time of maximum fuel temperatures. The cladding was again at peak temperature during this estimated failure time.

4.7 FUEL SPECIMEN C4C TRANSIENT IRRADIATION

The planned test objective of the C4C irradiation was to match the fuel volume melt achieved in specimen C4B to determine the effect of fuel density. A fuel melt volume interim to those obtained in the C4A and C4B irradiations, i.e., $\approx 70\%$ volume, was attained after two test transient irradiations.

The first test transient was inadvertently scrambled prior to reaching the normal peak power, i.e., the transient was clipped (subsection 3.5.1). Calculations matching the transient data indicate that the peak fuel temperature reached $\approx 4900^\circ\text{F}$ (figures 3-13 and 3-23). Following this irradiation, the specimen was x-rayed at TREAT. In view of the short time that the fuel was at high temperature and of the lack of any visible evidence of fuel restructuring in the x-ray negatives, it was assumed that no significant change in the fuel structure resulted from this first test transient.

4.7.1 Post-Transient Condition

As a result of the second test transient (C4C-2), significant movement of porosity and fuel occurred in the interior of the fuel column while the cladding retained its integrity. The primary results of the irradiation follow:

- Measurement of the fuel melt radius indicated that 67% to 70% of the fuel became molten (determined from radial extent of porosity movement and melt structure).
- A large central void was formed. The diameter varied up to ≈ 137 mils, with considerable axial variation in size (see neutrograph of the fuel column in figure 3-31). An ≈ 80 mil average diameter was predicted from fuel-densification calculations.
- Fuel movement resulted in the formation of fuel "bridges" which separated the central void into ten primary regions (figure 3-31).
- The majority of the void regions tapered upwards, in contrast to the characteristic downward tape associated with previously transient-irradiated high density fuel specimens^{1 2 4} and with shrinkage voids found in castings. In addition, the bottom of many of these voids was convex upward, resembling a positive meniscus (e.g., similar to mercury versus the negative meniscus associated with water).
- Two fuel plugs, found in the central void, were apparently not attached to the wall of the void. The largest of these plugs $\approx \frac{1}{4}$ -inch long by ≈ 120 -mil diameter, is located at approximately one-third of the fuel column height in the C4C view shown in figure 3-31. The second plug appeared to be spherical, ≈ 75 mils in diameter, located $\approx \frac{1}{4}$ of the distance up the fuel column (figure 3-31).²⁰
- There was no detectable permanent length change in the fuel column, and the cladding deformation was quite small, < 0.5 -mil average (table 3-3).

Apparently, the extensive internal fuel and porosity movement was a result of the high fuel melt volume and the low fuel smeared density. As a result of the comparatively high porosity fraction (19.3% compared to the $< 9.6\%$ smeared porosity in specimens C4A and C4B), the $\approx 10\%$ fuel volume increase upon melting was readily absorbed, leaving free porosity present.

4.7.2 Thermal Analysis

The transient response of the sodium annuli thermocouples to the second test transient are presented in figure 3-14. The calculated responses show a relatively good agreement with the data. The radial fuel temperature profiles (figure 3-24) show an excellent match of fuel melt radius to the measured value of 90 mils, resulting in a calculated peak fuel temperature of $\approx 5600^\circ\text{F}$. These results place specimen C4C fuel conditions interim to those of the failed specimens C4A and C4B (table 3-2), although cladding peak temperature was somewhat less than specimen C4A ($\approx 1260^\circ\text{F}$ versus $\approx 1400^\circ\text{F}$) and significantly less than specimen C4B ($\approx 1260^\circ\text{F}$ versus $\approx 1580^\circ\text{F}$).

4.8 FUEL SPECIMENS C4D AND C4C-II TRANSIENT IRRADIATION

The low density pellet fuel specimen C4D ($\approx 80.1\%$ smeared density), virtually identical to specimen C4C, was irradiated in conjunction with the small auxiliary fuel specimen C4C-II (also low density pellet fuel), which contained a W5Re/W26Re in-fuel thermocouple. The specimens were mounted on the same axial centerline with specimen C4D centered about the TREAT core midplane and specimen C4C-II located directly below (subsection 2.4 and figures A-3 and A-5).

The test transient irradiation of specimen C4D was planned to result in a portion of the fuel entering, but not passing through, the thermal arrest associated with heat of fusion.* Based on the post-transient examination, the objectives were met with $\approx 50\%$ of the fuel showing evidence of at least partial phase change.

4.8.1 Post-Transient Condition

Neutrographs of specimens C4D and C4C-II (figure 3-32) show no indication of large continuous voids in the fuel column, but pellet cracking is visible. In specimen C4C-II, it is possible to see a central hole in the bottom half of the fuel column, containing the W-Re thermocouple.

During the destructive examination of specimen C4D, it was discovered that there was significant initiating void movement in the form of pore aggregation. Bubbles as large as 44-mils diameter were observed during examination and it was noted that there was general pore movement to a radial region approximately 40 to 70 mils from the fuel center. Equiaxed grain growth, of almost uniform size (≈ 2 -mil diameter), occurred across the region to which a portion of the heat of fusion was supplied (out to a radius of 77 to 78 mils; table 3-2).

The shift of the porosity to a specific radial region probably resulted from the specific temperature profiles generated in TREAT test specimens (subsection 4.4). As can be seen in figures 3-21 to 3-27, the radial flux depression leads to some distortion in the instantaneous temperature profile, with the maximum fuel temperature often occurring in the region of mid radius. Since the pore movement should be governed by temperature and energy gradients, the initial pore migration should be to this radial region of maximum temperature or energy density.

There was no measurable retained increase in length of the specimen C4D fuel column. The extensometer follower was undisturbed, indicating that the transient change in fuel column length was less than 0.10 inch ($< 0.7\%$). Also, there was no detectable diametral cladding deformation induced by the transient irradiation (table 3-3).

The most notable features in the examination of the auxiliary specimen C4C-II were the concentration of porosity in the radial region 40 to 70 mils from the fuel center and the fact that pellet interfaces remained visible in longitudinal sections of the fuel column. The retention of these fuel interfaces is evidence that the fuel did not become completely molten. This would indicate a maximum limit on the fuel temperature at or below fuel melting, tending to verify the calculated results.

* It is recognized that mixed-oxide fuel is a two-phase solid solution, and as such, does not melt congruently. Melting of 20-80 mixed oxide fuel actually occurs in company with a temperature rise of $\approx 20^\circ\text{F}$,²¹ but for calculational purposes, it was assumed that the fuel does melt congruently at 5200°F .

4.8.2 Thermal Analysis

The calculated and measured transient values for the inner and outer sodium annuli thermocouples resulting from the TREAT irradiation are plotted in figure 3-15. The excellent fit obtained also applied to the predictions of the radial fuel temperature profiles (figure 3-25). From the calculations, the following were predicted:

- The melt radius (the radius within which fuel reached the melt temperature) corresponded quite well with the measured extent of the grain growth region.
- The maximum fuel temperature reached, but did not exceed, the melt temperature of the fuel (i.e., none of the fuel became 100% molten).
- The maximum fuel temperatures, as the fuel approaches its melting point, are in the radial region of ≈ 60 to 70 mils. After reaching the melting temperature, this same region continues to be the region of highest energy density, with the greatest penetration into the fuel heat of fusion. This region corresponds to the region of porosity movement noted previously.

Data from the in-fuel thermocouple are compared to the calculated central fuel temperature of specimen C4D in figure 3-16. The thermocouple registered a peak temperature of $\approx 4400^\circ\text{F}$. Basic agreement between measured and calculated temperature is indicated by the similarity of slopes for the cooldown portion of the traces. The temperature plateau during the thermocouple temperature rise (at $\approx 3800^\circ\text{F}$) was apparently due to a thermal arrest resulting from a phase change in the BeO insulator used in the thermocouple. The lack of thermal arrest upon cooldown reflects the fact that the BeO insulator melted and slumped downward and away from the junction at the grounded head of the thermocouple.

The initial differences in the preheat temperatures ($\approx 606^\circ\text{F}$ for specimen C4D and $\approx 430^\circ\text{F}$ for specimen C4C-II) were a result of end effects from the combination preheat heater can and insulator can. As mentioned in subsection 3.3, these end effects were insignificant over the length of the test specimen fuel column.

4.9 FUEL SPECIMENS C4I AND C4A-II TRANSIENT IRRADIATION

As was the case for the low density pellet specimen C4D, the powder fuel specimen C4I was irradiated in conjunction with a small secondary fuel specimen C4A-II (high density pellet fuel), which contained a W5Re/W26Re in-fuel thermocouple. The geometry within the C4I TREAT capsule was identical to that described for specimens C4D and C4C-II.

The test purpose for the C4I irradiation was to match the C4D test, i.e., to have a portion of the fuel reach but not exceed the fuel melt temperature. But, because of predicted low thermal conductivity for powder fuel (based on powder UO_2 data¹⁵), a less severe test transient was planned in order to avoid the passage of any of the fuel

completely through the heat of fusion. Because of operational difficulties, however, the initial transient power pulse was significantly smaller than planned, with peak fuel temperatures of only $\approx 3000^\circ\text{F}$. Since this was well below the required test fuel conditions, a second larger test transient was performed to reach the desired conditions.

4.9.1 Post-Transient Condition

The second test transient apparently satisfied the test objective of just reaching the melting point, although there was some local evidence that fuel had passed completely through heat of fusion. The central region of the fuel, <60 mils in radius, showed some equiaxed grain growth, a significant amount of intraparticle sintering, and some evidence of porosity movement and agglomeration. This was particularly true in the regions at both ends of the fuel column, apparently a result of end-flux peaking. There also was evidence of some localized regions of fuel (20-25 mils in diameter) where complete melting had occurred.

The cladding diameter of specimen C4I was slightly deformed (<0.3 mil average) along the length of the fuel column. There was no detectable permanent or transient increase in the fuel column length.

The fuel of the auxiliary specimen C4A-II showed little effect from the irradiations except for some thermally-induced pellet cracking. This fuel apparently operated at a significantly lower temperature than did the primary specimen because of end-effect cooling of the capsule assembly (subsection 4.8.2) and the higher thermal conductivity of the pellet fuel.

4.9.2 Thermal Analysis

The comparison between calculated and measured temperatures for the inner and outer sodium annuli during the final test transient is shown in figure 3-17. Excellent agreement with the measured data is shown by the calculations, particularly in the outer annulus. The calculated radial fuel temperature profiles are shown in figure 3-26. The calculations predicted a slightly greater radial extent of melting (fuel that had reached the melt temperature) than was measured.

In order to simultaneously match the power factor (determined in the previous calibration transient analysis), to fit the thermocouple data satisfactorily, and to predict the approximate melt radius, it was necessary to use values for thermal conductivity of the powder fuel significantly above the predicted values based on powder UO_2 data.¹⁵ The correlation presented in figures 3-17 and 3-26 was obtained by increasing the predicted thermal conductivity by $\approx 33\%$. This assumption results in a powder fuel conductivity $\approx 15\%$ below pellet thermal conductivity for an equivalent density. The specific temperature-dependent thermal conductivities for fuel used in the analysis of the series IV irradiations are compared in figure B-3.

The in-fuel thermocouple in specimen C4A-II gave excellent results throughout the test transient irradiation. These data are presented in figure 3-18, together with the calculated center fuel temperature for the primary specimen. The difference between the measured temperature of the C4A-II thermocouple, and the calculated temperature in the C4I fuel, was greater than in the C4C-II thermocouple and the C4D fuel (figure 3-16). As mentioned previously, this difference was primarily due to the variation in thermal conductivity.

4.10 FUEL SPECIMEN C4J TRANSIENT IRRADIATION

The transient irradiation of specimen C4J (83.5% smeared density powder fuel) was planned to produce a fuel volume melt comparable to those previously attained in the irradiation of pellet specimens C4A, C4B, and C4C. The actual result was 49% to 60% fuel volume melt, which was comparable to the results of C4A. This result was somewhat lower than expected from pre-transient calculations, apparently again due to differences in the calculated values used for powder fuel thermal conductivity.

4.10.1 Post-Transient Condition

The primary features noted during examination of specimen C4J follow:^{13 14}

- (a) The molten fuel volume of 49% to 60%, as measured by porosity movement and extent of fuel-melt structure, was comparable to the results for the previously failed pelleted fuel specimen C4A (subsection 4.5 and references 16 and 17).
- (b) Fuel was expelled from the cladding envelope over approximately 63% of the $13\frac{1}{2}$ -inch fuel-column length (figure 3-34). This fuel displacement apparently involved driving forces in excess of gravity, e.g., gas pressure and molten fuel expansion. The extent of the failure precluded positive identification of the primary failure mechanism.
- (c) Below the failure zone, about 32% of the local fuel volume was expelled upward to the region of breached cladding. This fuel movement apparently confirmed the observation of upward movement of molten fuel previously indicated in the test results of zero-burnup specimens C4A and C4B.^{16 17} In addition, about 40% of the local fuel volume above the failure zone had moved downward to the cladding breach.
- (d) The unmelted powder fuel (with no prior in-situ sintering history) retained a significant amount of coherency, even in the regions of cladding failure. For example, in areas where $>90\%$ of the cladding circumference was removed due to melting, the particulate fuel appeared to have sintered to the outside of a hollow, molten-fuel chimney (additionally, see first item in (e) below).

(e) Apparent gas or vapor blanketing of the 0.075-inch sodium annulus surrounding the fuel specimen extended the effective cladding failure region both upward and downward from the primary failure zone.

- In the upward direction, the cladding breach extended approximately 1 inch above the expelled fuel zone. In this region, the melted cladding had curled back, exposing the fuel and forming a bead of resolidified cladding around the circumference of the breach, which was 100 to 130 mils in width. The powder fuel exposed at this breach retained its geometry despite the removal of the cladding and, presumably, the contact of the hot fuel by sodium. The fuel spanning the breach had a minimum thickness of only 12 to 14 mils. This thin wall of oxide resulted from the large central void formed by loss of molten fuel to the failure region, plus some fuel washout from the fuel surface (which probably occurred during examination of the specimen).
- The cladding damage extended an additional one-half inch above the cladding breach, where a blister (approximately 100-mils in diameter) was formed from cladding which had become partially molten during the apparent gas or vapor blanketing.
- Evidence of downward coolant expulsion was found in a partial cladding melt, extending through the inner third of the cladding wall thickness, in the unfailed portion of the fuel specimen located below the primary failure zone. This observation indicated that this lower portion of the cladding was also temporarily insulated by gas or vapor in the sodium annulus.

(f) Short sections of cladding which were measured, indicated cladding deformation in the fuel region of one to two mils ($\approx 0.4\%$ to 0.8% ; table 3-3). The extent of the cladding failure precluded any conclusions about average or peak deformation. Measurements indicated

no permanent change in fuel column length, and a maximum transient change of less than 0.02 inch.

4.10.2 Thermal Analysis

The thermal analysis of C4J was conducted assuming intact geometry and using the C4I power factor. The C4J calibration transient analysis indicated the validity of such a power factor assumption. This method of analysis was required because of the extensive failure with the resulting disruptive effect on the sodium annuli thermocouples.

The actual responses of the sodium annuli thermocouples are shown in figures 3-19 and 3-20, together with the calculated results, assuming intact geometry. It appears from the response of thermocouple No. 9 (figure 3-19) that failure occurred at 3.1 to 3.2 seconds into the transient. The response of thermocouple No. 10 indicated that the failure reached its location at ≈ 3.6 seconds.

In general, thermocouple response was similar to that associated with the C4B post-failure fuel movement (subsection 4.6.2). Once again, the increased linear density of fuel in the failed region, as a result of axial movement, caused thermocouples Nos. 8 through 11 (located in the failure region) to have a significantly higher quasi-equilibrium value than predicted by the intact geometry calculations (figure 3-19). Conversely, thermocouple No. 6 of specimen C4J was located above the failure zone, in the region where $\approx 40\%$ of the fuel moved downward and away to the failure zone. As a result of this decrease in local energy source, thermocouple No. 6 registered significantly below the equilibrium level predicted by the intact geometry calculations (figure 3-20). These results are similar to the response of corresponding thermocouples in specimen C4B (figures 3-11 and 3-12).

The radial fuel temperature profiles calculated for the intact geometry assumption are shown in figure 3-27. The fuel melt radius was adequately predicted by the calculations. Comparison of these calculated fuel temperatures with the thermocouple traces indicates that failure may have occurred prior to, or concurrent with, fuel melting.

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Design Engineering at IPO
N. C. Shirley

Post-Irradiation Examinations
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Fuel Fabrication
E. F. Kurtz

Computer Calculations
R. C. Thompson

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REFERENCES

1. J. E. Hanson, J. H. Field, S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series II—Mixed Oxide (PuO₂-UO₂) Irradiations*, GE-BRDO, June 1965, (GEAP-4804).
2. J. H. Field, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series I UO₂ Irradiations*, GE-BRDO, November 15, 1962, (GEAP-4130).
3. J. E. Hanson and S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series III Pre-Irradiated Mixed Oxide (PuO₂-UO₂) Irradiations Interim Report Steady-State Irradiations*, GE-BRDO, September 1965, (GEAP-4819).
4. J. E. Hanson and J. H. Field, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series III Pre-Irradiated Mixed Oxide (PuO₂-UO₂) Irradiations Final Report Transient Irradiations*, GE-BRDO, July 1967, (GEAP-4469).
5. D. R. MacFarlane, G. A. Freund, and J. F. Boland, *Hazards Summary Report on the Transient Reactor Test Facility (TREAT)*, Argonne National Laboratory, October 1958, (ANL-5923).
6. J. A. Christensen, *Thermal Expansion and Change in Volume of Uranium Dioxide on Melting*, J. Am. Ceram. Soc., **46**:607-8, (December 21, 1963).
7. S. C. Skirvin, *User's Manual for the Standard THTD Computer Program on GE-600 Series Computers*, General Electric Research and Development Center, May 1969, (69-C-205).
8. J. N. Fox, B. E. Lawler, and H. R. Butz, *FORE-II A Computational Program for the Analysis of Steady-State and Transient Reactor Performance*, GE-BRDO, September 1966, (GEAP-5273).
9. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Thirtieth Quarterly Report, February-April 1969*, May 1969, (GEAP-10028).
10. *Requirements for Experiments in the Transient Reactor Test Facility*, Argonne National Laboratory, December 1960.
11. F. Kirn et al., *Reactor Physics Measurements in TREAT*, Argonne National Laboratory, October 1960, (ANL-6173).
12. D. F. Schoeberle et al., *A Method of Calculating Transient Temperatures in a Multiregion Axisymmetric, Cylindrical Configuration. The ARGUS Program*, Argonne National Laboratory, November 1963, (ANL-6654).
13. G. R. Thomas and J. H. Field, *Transient Performance of Mixed-Oxide Powder Fuel with Up to 60% Melting*, Trans. Am. Nucl. Soc., **12**, 343, (1969).
14. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-eighth Quarterly Report, August-October 1968*, November 1968, (GEAP-5700).
15. M. F. Lyons et al., *UO₂ Powder and Pellet Thermal Conductivity During Irradiation*, GE-BRDO, March 1966, (GEAP-5100-1).
16. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-sixth Quarterly Report, February-April 1968*, May 1968, (GEAP-5631).
17. G. R. Thomas and J. H. Field, *Destructive Examination of an Axially Restrained Pelleted Mixed-Oxide Specimen that Failed During Transient Overpower Conditions*, Trans. Am. Nucl. Soc., **11**, 697, (1968).
18. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-first Quarterly Report, November 1966-January 1967*, February 1967, (GEAP-5433).
19. G. R. Thomas and J. H. Field, *Response of Axially Restrained Pellet Fuel to Transient Overpower*, Trans. Am. Nucl. Soc., **10**, 707, (1967).
20. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-fourth Quarterly Report, August-October 1967*, November 1967, (GEAP-5541).
21. J. L. Krankota and C. N. Craig, *The Melting Point of Plutonia-Urania Mixed Oxides Irradiated to High Burnup*, GE-BRDO, July 1969, (GEAP-13515).

APPENDIX A

DESIGN DETAILS AND AS-FABRICATED FUEL SPECIFICATIONS FOR THE SERIES IV ZERO-BURNUP FUEL SPECIMENS AND CAPSULES

This appendix contains the final design drawings and the as-fabricated fuel specifications for fuel specimens and capsules used in the series IV zero-burnup tests. The design-

nations and locations of the thermocouples in the TREAT capsule are given in table A-1. Also, the as-fabricated fuel and blanket specifications are given in table A-2.

Table A-1
SERIES IV ZERO-BURNUP CAPSULE THERMOCOUPLE LOCATIONS

Thermocouple Number	Material	Location		Comments
		Radial	Axial [above/below core Q] (inches)	
1	Tungsten-Rhenium	secondary fuel specimen	- 12-7/8	In-fuel thermocouple.
2	Chromel-Alumel	inner Na annulus	- 12-7/8	
3	Chromel-Alumel	inner Na annulus	- 12-7/8	
4	Chromel-Alumel	outer Na annulus	- 12-7/8	
5	Chromel-Alumel	outer Na annulus	- 12-7/8	
6	Chromel-Alumel	inner Na annulus	+ 3-7/16	Thermocouples 1 to 5 appear only in the secondary fuel capsule.
7	Chromel-Alumel	outer Na annulus	+ 3-7/16	
8	Chromel-Alumel	outer Na annulus	0	
9	Chromel-Alumel	inner Na annulus	0	
10	Chromel-Alumel	inner Na annulus	- 3-7/16	
11	Chromel-Alumel	outer Na annulus	- 3-7/16	

Table A-2
TASK C SERIES IV AS-FABRICATED FUEL AND BLANKET SPECIFICATIONS

Specimen	EBR-II Capsule	Average Cladding Inner Diameter* (in.)	Average Pellet Diameter (in.)	Fuel Col. Length (in.)	Average Pellet Density (% TD)	Smeared Density (% TD)	O/M	Fuel Composition (%)						Blanket/Insulator		
								Pu/Pu+U	U ₂₃₅ /U	Pu ₂₃₉ /Pu	Pu ₂₄₀ /Pu	Pu ₂₄₁ /Pu	Pu ₂₄₂ /Pu	Length (in.)	Smeared Density (% TD)	O/M
I. TASK C SERIES IV ZERO-BURNUP FUEL SPECIMENS																
C4A	none	0.2202	0.2159	14.23	93.8	90.4	1.983	19.1	0.57	91.04	8.22	0.69	0.04	4.81/0.252	93.57	2.00
C4B	none	0.2202	0.2155	14.21	94.5	91.2	1.983	19.4	0.57	91.04	8.22	0.69	0.04	4.81/0.250	93.37	2.00
C4C	none	0.2202	0.2164	14.19	83.1	80.4	1.983	20.9	0.91	91.04	8.23	0.69	0.04	4.89/0.250	92.86	2.00
C4D	none	0.2202	0.2154	14.22	83.7	80.1	1.983	20.9	0.91	91.04	8.23	0.69	0.04	4.89/0.254	90.78	2.00
C4H	none	0.2204	powder	13.48	powder	83.5	1.973	20.2	0.894	90.84	8.35	0.768	0.04	4.787/0.682	62.02	2.00
C4J	none	0.2205	powder	13.45	powder	83.5	1.973	20.2	0.894	90.84	8.35	0.768	0.04	4.810/0.669	62.12	2.00
II. TASK C SERIES IV PRE-IRRADIATED FUEL SPECIMENS																
C4E	F7A	0.2202	0.2159	13.46	83.2	80.5	1.974	27.6	92.7	91.11	8.18	0.68	0.04	4.55/0.25	91.95	2.00
C4F	F7B	0.2205	0.2160	13.51	84.0	80.5	1.975	28.3	92.8	90.88	8.31	0.76	0.04	4.55/0.253	91.91	2.00
C4G	F7C	0.2202	0.2154	13.50	94.4	90.5	1.973	27.6	89.3	91.04	8.23	0.69	0.04	4.53/0.25	91.69	2.00
C4H	F7D	0.2202	0.2152	13.45	94.4	90.4	1.976	27.6	89.3	91.04	8.23	0.69	0.04	4.538/0.251	92.12	2.00
C4K	F7E	0.2205	powder	13.50	powder	84.1	1.972	27.5	93.0	90.98	8.20	0.79	0.03	4.750/0.351	63.88	2.00
C4L	F7F	0.2206	powder	13.47	powder	84.2	1.972	27.5	93.0	90.98	8.20	0.79	0.03	4.78/0.375	60.92	2.00

* Cladding is 316 stainless steel: 0.250-inch o.d. by 0.015-inch-wall-thickness welded tubing, ballized to 0.2200 ^{+0.0005}/_{-0.0000} inch i.d.

Pt No	DIM A	MATERIAL
4	092	300 SERIES STN STL
5	048	
6	018	
7	010	

[illegible]

Figure A-1. Series IV Zero-Burnup Pellet Fuel Specimens

UNLESS OTHERWISE SPECIFIED, USE THE FOLLOWING:				GENERAL ELECTRIC	
APPLIES PRACTICES	SURFACES	FINISHES	UNIT OF MEASURE	SYMBOL	DESCRIPTION
7EAS481	63	52	556	9190707	UPPER FUEL ASSEMBLY
FIRST READ FOR P&H TASK C SERIES 4				NEXT ASSEMBLY	
				NAME	DESCRIPTION
				1 TOP BIND	SEE DETAIL
				2 CLAD	
				3 BOND	
				4 BOND	SEE DETAIL
				5 EXTENSOMETER	SEE DETAIL
				6 PLUS = E	SEE DETAIL
				7 PLASTER = E	SEE DETAIL
				8 BATTERY	SEE DETAIL
				9 INSULATION	SEE DETAIL
				10 SPACE

[illegible]

1. STAINING AND DYES INTRODUCED IN A2 NOT TO BE USED IN A3, A4, A5, A6, A7, A8, A9, A10, A11, A12, A13, A14, A15, A16, A17, A18, A19, A20, A21, A22, A23, A24, A25, A26, A27, A28, A29, A30, A31, A32, A33, A34, A35, A36, A37, A38, A39, A40, A41, A42, A43, A44, A45, A46, A47, A48, A49, A50, A51, A52, A53, A54, A55, A56, A57, A58, A59, A60, A61, A62, A63, A64, A65, A66, A67, A68, A69, A70, A71, A72, A73, A74, A75, A76, A77, A78, A79, A80, A81, A82, A83, A84, A85, A86, A87, A88, A89, A90, A91, A92, A93, A94, A95, A96, A97, A98, A99, A100, A101, A102, A103, A104, A105, A106, A107, A108, A109, A110, A111, A112, A113, A114, A115, A116, A117, A118, A119, A120, A121, A122, A123, A124, A125, A126, A127, A128, A129, A130, A131, A132, A133, A134, A135, A136, A137, A138, A139, A140, A141, A142, A143, A144, A145, A146, A147, A148, A149, A150, A151, A152, A153, A154, A155, A156, A157, A158, A159, A160, A161, A162, A163, A164, A165, A166, A167, A168, A169, A170, A171, A172, A173, A174, A175, A176, A177, A178, A179, A180, A181, A182, A183, A184, A185, A186, A187, A188, A189, A190, A191, A192, A193, A194, A195, A196, A197, A198, A199, A200, A201, A202, A203, A204, A205, A206, A207, A208, A209, A210, A211, A212, A213, A214, A215, A216, A217, A218, A219, A220, A221, A222, A223, A224, A225, A226, A227, A228, A229, A230, A231, A232, A233, A234, A235, A236, A237, A238, A239, A240, A241, A242, A243, A244, A245, A246, A247, A248, A249, A250, A251, A252, A253, A254, A255, A256, A257, A258, A259, A260, A261, A262, A263, A264, A265, A266, A267, A268, A269, A270, A271, A272, A273, A274, A275, A276, A277, A278, A279, A280, A281, A282, A283, A284, A285, A286, A287, A288, A289, A290, A291, A292, A293, A294, A295, A296, A297, A298, A299, A300, A301, A302, A303, A304, A305, A306, A307, A308, A309, A310, A311, A312, A313, A314, A315, A316, A317, A318, A319, A320, A321, A322, A323, A324, A325, A326, A327, A328, A329, A330, A331, A332, A333, A334, A335, A336, A337, A338, A339, A340, A341, A342, A343, A344, A345, A346, A347, A348, A349, A350, A351, A352, A353, A354, A355, A356, A357, A358, A359, A360, A361, A362, A363, A364, A365, A366, A367, A368, A369, A370, A371, A372, A373, A374, A375, A376, A377, A378, A379, A380, A381, A382, A383, A384, A385, A386, A387, A388, A389, A390, A391, A392, A393, A394, A395, A396, A397, A398, A399, A400, A401, A402, A403, A404, A405, A406, A407, A408, A409, A410, A411, A412, A413, A414, A415, A416, A417, A418, A419, A420, A421, A422, A423, A424, A425, A426, A427, A428, A429, A430, A431, A432, A433, A434, A435, A436, A437, A438, A439, A440, A441, A442, A443, A444, A445, A446, A447, A448, A449, A450, A451, A452, A453, A454, A455, A456, A457, A458, A459, A460, A461, A462, A463, A464, A465, A466, A467, A468, A469, A470, A471, A472, A473, A474, A475, A476, A477, A478, A479, A480, A481, A482, A483, A484, A485, A486, A487, A488, A489, A490, A491, A492, A493, A494, A495, A496, A497, A498, A499, A500, A501, A502, A503, A504, A505, A506, A507, A508, A509, A510, A511, A512, A513, A514, A515, A516, A517, A518, A519, A520, A521, A522, A523, A524, A525, A526, A527, A528, A529, A530, A531, A532, A533, A534, A535, A536, A537, A538, A539, A540, A541, A542, A543, A544, A545, A546, A547, A548, A549, A550, A551, A552, A553, A554, A555, A556, A557, A558, A559, A560, A561, A562, A563, A564, A565, A566, A567, A568, A569, A570, A571, A572, A573, A574, A575, A576, A577, A578, A579, A580, A581, A582, A583, A584, A585, A586, A587, A588, A589, A590, A591, A592, A593, A594, A595, A596, A597, A598, A599, A600, A601, A602, A603, A604, A605, A606, A607, A608, A609, A610, A611, A612, A613, A614, A615, A616, A617, A618, A619, A620, A621, A622, A623, A624, A625, A626, A627, A628, A629, A630, A631, A632, A633, A634, A635, A636, A637, A638, A639, A640, A641, A642, A643, A644, A645, A646, A647, A648, A649, A650, A651, A652, A653, A654, A655, A656, A657, A658, A659, A660, A661, A662, A663, A664, A665, A666, A667, A668, A669, A670, A671, A672, A673, A674, A675, A676, A677, A678, A679, A680, A681, A682, A683, A684, A685, A686, A687, A688, A689, A690, A691, A692, A693, A694, A695, A696, A697, A698, A699, A700, A701, A702, A703, A704, A705, A706, A707, A708, A709, A710, A711, A712, A713, A714, A715, A716, A717, A718, A719, A720, A721, A722, A723, A724, A725, A726, A727, A728, A729, A730, A731, A732, A733, A734, A735, A736, A737, A738, A739, A740, A741, A742, A743, A744, A745, A746, A747, A748, A749, A750, A751, A752, A753, A754, A755, A756, A757, A758, A759, A760, A761, A762, A763, A764, A765, A766, A767, A768, A769, A770, A771, A772, A773, A774, A775, A776, A777, A778, A779, A780, A781, A782, A783, A784, A785, A786, A787, A788, A789, A790, A791, A792, A793, A794, A795, A796, A797, A798, A799, A800, A801, A802, A803, A804, A805, A806, A807, A808, A809, A810, A811, A812, A813, A814, A815, A816, A817, A818, A819, A820, A821, A822, A823, A824, A825, A826, A827, A828, A829, A830, A831, A832, A833, A834, A835, A836

NOTES:

DESCRIPTION OF GROUPS	REVISIONS	POINTS TO
CIA PEN BRIDGE A-13-GT 8 ADVISORY	IPD	SID

Figure A-2. Series IV Zero-Burnup Powder Fuel Specimens

GEAP-13562

Figure A-3. Series IV Secondary Zero-Burnup Fuel Specimen Containing an In-Fuel W5Re/W26Re Thermocouple

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING -		TITLE	
APPLIED PRACTICES	SURFACES	TOLERANCES ON MACHINED DIMENSIONS	GENERAL ELECTRIC
145A5481	63	FRACTIONS DECIMALS ANGLES	112C3230
		$\pm .005$ $\pm .005$ $\pm 5^\circ$	EXTENSOMETER
FIRST MADE FOR FCR TASK C			
NEXT ASSY 886DIS6Q1			
GI PT	NAME	DESCRIPTION	
1	END CAP	SEE DETAIL	
1	PISTON		
1	SPRING		
1	END PLUG	SEE DETAIL	
1	RING	SEE DETAIL

- 3 COMPRESSION SPRING
- .032 WIRE DIA
 - .146 INSIDE DIA
 - 3 5/16 FREE LENGTH
 - 64 ACTIVE COILS
 - 66 TOTAL COILS
 - INCONEL-X-750 WIRE
 - ENDS SQ & GR
 - ⚠ RATE 4.05 $\pm .3$ LBS/INCH

REVISIONS		PRINTS TO
3	ADDED 'RATE' TO SPRING (P3)	
1	3A WAS 1 1/16, 3B WAS 7/32, 3C WAS 1 1/32, 3D WAS .125 DIA, 3E WAS .165 DIA, 3F WAS 304 STN STEEL, 3G WAS .122 DIA, 3H WAS .275 DIA, 3J WAS .130 DIA, 3K WAS .122 DIA, 3L WAS .030 WIRE DIA, .170 INSIDE DIA, 1.500 FREE LENGTH, 36 ACTIVE COILS, 38 TOTAL COILS, RATE .6 TO 1.0 LBS/INCH	
2	GENERAL CHANGES	
DATE 12-19-66		DL
APPROVED		
100 VAL		
112C3230		
CHK'D		

Figure A-4. Extensometer for the Series IV Fuel Specimens

Figure A-5. TREAT Capsule Used in the Transient Irradiation of the Series IV Zero-Burnup Fuel Specimens

APPENDIX B

INPUT PARAMETERS FOR THE THTD COMPUTER CODE ANALYSIS OF THE SERIES IV ZERO-BURNUP TRANSIENT IRRADIATIONS

This appendix presents the various input parameters necessary to perform correlations of the experimental transient data using the THTD transient heat transfer code. The required inputs were discussed previously in subsection 3.4.1. The actual values of these parameters used in the calculational model are presented below.

Table B-1 presents the optimized values for the fuel-cladding gap coefficient and the specimen power factor (which are compared to the pre-transient calculated power factors). Figures B-1 and B-2 present the radial power profile through the fuel, based on calculations using the THERMOS code.^{B1}

Figure B-3 presents the thermal conductivity of the mixed oxide fuel as used in the THTD calculations. The determination of the density and temperature dependence of the fuel thermal conductivity was based on the recommendations for mixed oxide fuel presented in reference B-2, with one assumed exception. Considering the short time required to heat the fuel from the initial preheat temperature of 600° to 800°F (table 3-1) to fuel-melt temperature, <5 seconds, it was assumed that no significant restructuring of the fuel occurred prior to melting. As a result, the initial fuel density was used to calculate the fuel thermal conductivity up to the melting point (in variance with the restructuring-densification corrections recommended in reference B-2). Above the melting temperature, all fuel, regardless of initial density or form, was assumed to be 100% dense, and the corresponding value of conductivity for 100% dense solid fuel just below melting was assigned as a constant value.

The remaining thermal-physical properties required to describe the inner capsule constituents (figures 2-2 and 3-1) are presented in figures B-4 through B-10.

Table B-1
FUEL-CLADDING GAP COEFFICIENTS
AND POWER FACTORS
USED IN THE THTD CORRELATIONS
OF THE TEST TRANSIENTS

Fuel Specimen	Fuel and Cladding Gap Coefficient	Power Factor	Calculated Power Factor
	$\frac{\text{Btu}}{\text{h-ft}^2 \cdot ^\circ\text{F}}$	$\frac{\text{W/cm}^3 \text{ fuel}}{\text{TREAT watt}}$	$\frac{\text{W/cm}^3 \text{ fuel}}{\text{TREAT watt}}$
C4A	2500	0.835×10^{-4}	0.833×10^{-4}
C4B	2300	0.835×10^{-4}	0.833×10^{-4}
C4C	1900	0.837×10^{-4}	0.825×10^{-4}
C4D	1500	0.790×10^{-4}	0.825×10^{-4}
C4I	6000	0.900×10^{-4}	0.826×10^{-4}
C4J	6000	0.890×10^{-4}	0.826×10^{-4}

Figure B-1. Radial Power Profile—Specimens C4A and C4B

Figure B-2. Radial Power Profile—Specimens C4C, D, I, and J

Figure B-3. Thermal Conductivity of Mixed-Oxide Fuel

Figure B-4. Composite Specific Heat Curve for $(U_{0.8}, Pu_{0.2})O_{1.98}$

Figure B-5. Thermal Conductivity of Types 304 and 316 Stainless Steel

Figure B-6. Specific Heat of Types 304 and 316 Stainless Steel

Figure B-7. Thermal Conductivity of Sodium

Figure B-8. Specific Heat of Sodium

Figure B-9. Thermal Conductivity of 6061-T6 Aluminum

Figure B-10. Specific Heat of 6061-T6 Aluminum

APPENDIX B REFERENCES

- B-1. H. C. Honeck, *THERMOS A Thermalization Transport Theory Code for Reactor Lattice Calculations*, Brookhaven National Laboratory, November 5, 1962, (BNL-5826).
- B-2. W. E. Baily et al., *Thermal Conductivity of Uranium-Plutonium Oxide Fuels*, presented at the Nuclear Metallurgy Symposium at Camelback Inn, Phoenix, Arizona, October 4 to 6, 1967.

APPENDIX C

NEUTRON RADIOGRAPHY

Neutrography, or neutron radiography, is a technique that takes advantage of the differential neutron absorption characteristics of materials to view the interior of opaque objects in a manner somewhat similar to x-radiography. The major advantage over x-rays is that neutrons are absorbed as a function of the individual nuclear characteristics of every material, whereas x-rays are absorbed basically as a function of material atomic weight. Fast reactor fuels, very high in molecular weight and containing plutonium, offer a prime example of the advantages offered by Neutrography. An x-ray sees a quarter-inch diameter fuel specimen, containing mixed-oxide fuel, as effectively as a totally black medium. A neutron sees the plutonium selectively (the UO_2 is natural enrichment, therefore fairly invisible to neutrons). As a result, the Neutrograph of the fuel will show such features as void grouping (e.g., central voids) and pellet interfaces or cracks in the fuel. A high degree of resolution

is an additional advantage. This is possible because the parallax in Neutrographs is significantly lower than for x-rays, as a result of the greater focal length used to make Neutrographs, compared to that which is generally feasible with most commercial x-ray installations. As a comparison, the post-transient x-ray of fuel specimen C4A is shown in figure 3-28. Evidence of cladding failure can be seen in the form of fuel displacement outside of the cladding envelope. As can be seen, the x-ray sees the fuel and the blanket as very black. Figure 3-29 presents the post-transient Neutrographs of this same specimen. The large central void at the upper end of the fuel column is readily identifiable (evidence of slumping of molten fuel to the failure region), as are several voids located in the region below the failure. It is even possible to identify every fuel pellet interface along the length of the fuel column. Additionally, the UO_2 blanket located directly above the fuel is virtually invisible.

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APPENDIX D

ATTAINMENT OF THE DESIRED TREAT
REACTOR TEST PARAMETERS

In performing TREAT transient irradiation, it was necessary to attempt to predict the results of a planned transient irradiation (i.e., predict the temperature histories of the fuel, cladding, and sodium annuli) prior to performing the actual experiment. These predictions were accomplished by initially choosing a TREAT power pulse profile from the records of previously performed TREAT transient irradiations. The basics of the choice were: (a) The reactivity addition ($\% \Delta k$; see table 3-1), which also effectively defines the initial reactor period and peak reactor power, and (b) the total reactor energy release (the integrated power, MW-sec). The selected power pulse was then added to a THTD heat-transfer-code deck containing the series IV inner capsule geometry (figure 3-1) and used to calculate the expected sodium annuli and fuel temperature histories. If the computed expectation happened to differ from the result desired, then an appropriate change in the input parameters could be made and the THTD computations repeated.

As mentioned in subsection 2.3, in order to make these calculations meaningful, the correct power factor (ratio of specimen power to reactor power) must be determined. This power factor was initially obtained from calculations based on the resultant interconnected nuclear characteristics of the fuel specimen, capsule, and the TREAT core. The power factor was then confirmed or modified through use of a trial or calibration transient (which is a comparatively mild, slow transient irradiation of a peak power of 25 to 40 megawatts and an integrated power of about 70 to 80 MW-sec; table 3-1), followed by sets of THTD calculations using the actual TREAT parameters for the calibration transient. The calibration transients were of insufficient power to cause fuel melting or significant fuel restructuring (e.g., sintering), even if significant error existed in the calculated power factor. Once the power factor had been experimentally determined, a test transient could be chosen, through THTD calculations, to yield the desired fuel, cladding, and sodium temperatures. The test transient parameters were then

requested, the irradiation performed, and the calculations regenerated using the actual TREAT reactor test transient parameters.

Application of this experimental procedure was complicated by uncertainties inherent in the operation of the TREAT reactor. There are potential uncertainties in the calibration and linearity of core instrumentation (e.g., the neutron flux meters). In addition, the reproducibility of a given TREAT reactor transient pulse is affected by uncertainties in both reactivity addition (about $\pm 0.05\%$ for series IV sized transients; affected by parameters such as initial core ambient temperature and minor differences in core loading) and total integrated power (e.g., there is typically a 150 ± 50 msec delay in power dropoff following scram signal).^{D-1}

As an example, the first series IV test transient, specimen C4B(TT1057, table 3-1), was requested to be a repeat of a previously performed TREAT transient (TT767, capsule C2C). The requested transient parameters were $1.27\% \Delta k$ reactivity addition and ≈ 260 MW peak power. The resulting C4B test transient had experimentally measured parameters of $1.26\% \Delta k$ and 339 MW peak power.^{D-2 3}

Even when the actual measured reactivity input closely matched the requested reactivity input, there could still be a significant variation in peak reactor power (± 40 megawatts for a C4B-1 size transient power pulse). This problem was partially accounted for in the series IV tests by plotting the peak power versus the measured reactivity input from essentially all previous GE-TREAT transient tests (a total of 43 irradiations), which include both task C^{D-3} D-4 D-5 D-6 and SEFORD^{D-7} test programs. The resulting data spread was enveloped, and the test transient parameters chosen so that the expected peak power of a transient was essentially the average power of the envelope for a given reactivity input. This compilation of TREAT transient data is presented as a working design curve in figure D-1.

Figure D-1. TREAT Peak Power Versus Reactivity Input

APPENDIX D REFERENCES

- D-1. J. F. Boland, Argonne National Laboratory, TREAT Manager, personal communication to G. R. Thomas, GE-BRDO, April 20, 1967.
- D-2. GE-BRDO, *Sodium-cooled Reactors, Fast Ceramic Reactor Development Program, Twenty-first Quarterly Report, November 1966-January 1967*, February 1967, (GEAP-5433).
- D-3. J. E. Hanson, J. H. Field, S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series II—Mixed Oxide ($\text{PuO}_2\text{-UO}_2$) Irradiations*, GE-BRDO, June 1965, (GEAP-4804).
- D-4. J. H. Field, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series I UO_2 Irradiation*, GE-BRDO, November 15, 1962, (GEAP-4130).
- D-5. J. E. Hanson and S. A. Rabin, *Experimental Studies of Transient Effects in Fast Reactor Fuels Series III Pre-Irradiated Mixed Oxide ($\text{PuO}_2\text{-UO}_2$) Irradiations Interim Report Steady-State Irradiations*, GE-BRDO, September 1965, (GEAP-4819).
- D-6. G. R. Thomas and J. H. Field, *Response of Axially Restrained Pellet Fuel to Transient Overpower*, Trans. Am. Nucl. Soc., **10**, 707, (1967).
- D-7. J. H. Field et al., *An Evaluation of the Effect of Design and Operating Variables on SEFOR Fuel and Fuel Cladding*, GE-BRDO, December 1967, (GEAP-5309).