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SUBASSEMBLY TEST PROGRAM
OUTLINE FOR FY 1969 AND 1970

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SUBASSEMBLY TEST PROGRAM OUTLINE FOR FY 1969 AND 1970

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FOREWORD

The purpose of this report is to describe the experimental program planned for the CDC during fiscal years 1969 and 1970. Although needs and rationale for the studies are briefly reviewed, this report is not intended to provide a justification for sub-assembly testing. The major emphasis has been placed on discussions of program method and specific experiments.

This document was originally issued as an internal Phillips Petroleum Company report in August, 1968. Since that time, comments from the AEC and industry have generally supported the as-planned program, and have indicated the desirability for publication of the report with external distribution in order that it may be generally available to the nuclear community. The report is being published essentially in its original form. Revisions include the several additions and modifications in the planned test program that have been made during the past year. The current status and schedule are defined for each test series, reflecting delays, revisions and additions to the test program. Test series that have been completed are identified and reference is made to the reports describing the results.

ABSTRACT

The Subassembly Test Program in the Spert IV Capsule Driver Core for FY 1969 and 1970 is outlined. Basic program objective is to provide the AEC and reactor vendors with experimental information on reactor fuel behavior under transient overpower conditions. Test plans for unirradiated and irradiated (burnups to 30,000 MWd/T) UO_2 fuels, plutonium recycle fuels, and LMFBR fuels are presented. Data of particular interest include failure modes and thresholds, conversion efficiencies of nuclear-to-mechanical energy, transient pressures, and metal-water reaction extents, all as functions of transient energy deposition for a variety of fuel design parameters, burnup, and environmental conditions.

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I. INTRODUCTION

With the increased commercial use of nuclear power and the mounting desire to site nuclear plants in or near population centers, emphasis has been and will continue to be placed on more accurate evaluations of reactor safety. Some areas of reactor safety analysis are currently much better understood and amenable to realistic, quantitative prediction than are others. For example, the analysis of nondestructive excursion behavior of reactors, particularly water-moderated, UO_2 -fueled reactors, appears to be on a relatively sound experimental and theoretical basis. For accidents in which severe damage or destruction of the system would occur, however, relatively little information is available and the methods of analysis are only approximate. In the interests of safety the severe accident is, therefore, generally calculated such that the assumed consequences are many times larger than what might be realistically anticipated.

In order to provide information necessary for an accurate appraisal of reactor safety, the AEC is sponsoring a comprehensive program of nuclear research and development. An appreciable portion of the experimental research on accidents performed under the overall program has either been small scale, out-of-pile laboratory-type testing which has supplied an understanding of basic phenomena, or larger scale engineering tests that supplied information on total integrated system response. The in-pile subassembly program is designed to complement both the laboratory and integral system tests by testing samples of fuel under actual reactor conditions in an effort to eliminate the artificialities of laboratory tests without the time and inordinate expense of integral core tests. In many ways, the role of the in-pile subassembly tests within the safety program is analogous to that of in-pile test loops in steady-state test reactors in power reactor design.

The Subassembly Test Program for the CDC is designed to experimentally investigate the accident conditions that would be expected to exist in operating nuclear reactors. These investigations are done with small samples in a controlled environmental reactor test space to allow inclusion of heating rates, power distributions and radiation levels typical of a reactor accident. Specific areas of in-pile subassembly investigation

include such subjects as fuel-failure thresholds, modes of failure, damaging consequences, fission product release fractions, and the rate and extent of metal-water reactions. Because water-moderated systems with oxide fuels in the form of rods are predominant in usage now, initial emphasis has been placed on these systems in the development of a test program. Consideration has also been given to the needs of other reactor types, such as gas-cooled converters and fast breeders; but experiments specifically applicable to such systems presently constitute only a minor part of the immediate program.

The Subassembly Test Program as outlined here will be conducted in the CDC, which is a UO_2 -fueled system with a central sample-irradiation space. The CDC sample-irradiation space is approximately $3\frac{1}{2}$ inches in diameter and has a length of 6 feet. The CDC is a pulse source (no appreciable steady-state power capabilities) and can repeatedly produce power pulses with minimum initial periods of approximately 3 msec. (A detailed description of the CDC is given in Appendix A.)

The capabilities of the CDC permit an extension of the type of experiments performed in the TREAT reactor, which has a minimum reactor period capability of ≈ 40 msec, to reactor periods more typical of those to be expected in severe reactivity accidents. The Power Burst Facility, when it is completed, will permit extension of the program described here for the CDC to reactor periods in the 1-2 msec range, and to initial flow, temperature, pressure, and power conditions typical of operating conditions in power reactors.

The program plan presented in this document is sufficiently flexible to allow for the addition, deletion, or extension of any particular type or group of tests. Series of tests to investigate a particular fuel type or parametric variation are designed to first scope the behavior over the range of interest. Scoping test results are then evaluated to determine if additional tests are needed to provide an adequate understanding of the situation in question. The program is designed to be responsive to the changing needs of the AEC and the reactor industry in order that any new problem areas can receive early attention.

II. PROGRAM OBJECTIVES

The basic objective of the Subassembly Test Program in CDC is to provide the AEC and the reactor vendors with useful and necessary experimental information on reactor fuel behavior under transient overpower conditions.

The criteria for determining if particular subassembly tests will provide necessary information are related to the usefulness of the data from such tests in designing, licensing and operating commercial power reactors. In areas where problems in reactor safety are not well defined and where little or no experimental data presently exist, reactor design must be conservative and therefore more costly and/or less efficient. Limits imposed by licensing and regulatory agencies must be more restrictive, and operating procedures must provide adequate safety margins to allow for uncertainties in present analyses of credible accidents. The Subassembly Test Program is thus designed to produce needed information on transient-induced accidents in order to provide a better understanding of accident phenomena, thus allowing more confidence to be placed in safety analyses and, therefore, improved design and operating procedures for commercial power-producing plants.

The philosophy of the CDC program is basically one of scoping the problem, evaluating the results of these scoping tests, and then testing in more detail if the observed behavior warrants a more quantitative description. The need for a detailed understanding is best exemplified by examination of the consequences of subjecting a particular fuel design to a scoping series of overpower tests for a specific set of environmental conditions. If the consequences are found to be insignificant, the particular test series will have served its purpose and need not be continued. On the other hand, if the consequences are of sufficient severity to have significant implications in safety analysis, then more detailed testing will be pursued to further identify and quantify the course of events.

This approach to testing in the CDC will provide experimental fuel performance information which will be of immediate value to both the AEC and the vendors as base-point data in support and evaluation of analytical models. As the program progresses, the development of correlations relating test results to fuel design, burnup and environmental conditions

will provide useful information continuously.

In planning the details of the test program described in Section V, it has been assumed, based on engineering judgement and prior experience, that significant consequences will be indicated by certain of the scoping tests. Where significant consequences are observed, detailed testing will be required in order to provide an adequate understanding of the thresholds, modes, and consequences of fuel failure. Because the nature and extent of the detailed testing will be determined by the observed consequences of the scoping tests, the test program outlined in this report is subject to continuous modification as test results become available.

The current problem areas in the analysis of severe reactivity accident situations center around a lack of experimental information on such things as fuel meltdown, heat transfer, metal-water reactions, pressure generation, and fission product release and transport. For investigative purposes, these may be grouped into three broad categories: (a) the physical consequences of fuel failure, i.e., those interactions and reactions between the fuel element and its environment which are important in determining the severity of an accident, (b) the thresholds and causalities of fuel element failure, and (c) the radiological consequences possible following fuel element failure. Each of these categories can be subdivided into an exhaustive listing of specific problems for which little or no information presently exists. It is neither reasonable nor practical to attempt investigation of all of these specific problems in the CDC. Many are related to test conditions outside the capabilities of the CDC, e.g., environmental combinations of high temperature, pressure, flow, and initial power level, large clusters of active fuel rods, and detailed studies of the radiological consequences associated with failure. A major portion of such tests will be amenable to study in the PBF. Other of the specific problems are more basic in nature and are being studied on a laboratory scale in other AEC sponsored programs. Efforts in the CDC will thus be concentrated on acquiring needed data in those areas for which it is uniquely suited and capable. Primary emphasis will be placed on experiments to study physical consequences of fuel failure, and then, where warranted, to obtain detailed information on the thresholds and causalities of failure.

The objective of the CDC test program for fiscal years 1969 and 1970

is to provide information in each of the following areas:

A. Physical Consequences of Fuel Failure

- (1) Effects of energy deposition and heating rate
- (2) Magnitude and nature of pressure pulses generated
- (3) Fraction of thermal energy converted to mechanical energy
- (4) Extent of metal-water reaction
- (5) Extent of fuel sample fragmentation
- (6) Effects of fuel and cladding materials, design, and prior operational history
- (7) Effects of selected environmental variations (pressure, temperature, hydrodynamics)
- (8) Fuel rod interaction effects.

B. Threshold and Causality of Fuel Failure

- (1) Energy deposition and heating rate required for cladding failure
- (2) Modes of cladding failure
- (3) Effects of fuel melting and/or vaporization
- (4) Modes of heat transfer from fuel samples
- (5) Effects of fuel design, materials and burnup
- (6) Effects of selected environmental variations
- (7) Effects of increased metal-to-water ratio
- (8) Fuel rod interaction effects.

C. Radiological Consequences

- (1) Although not of primary programmatic interest, routine measurements of fission product concentrations, locations and types will be made following tests, particularly on fuels with high burnups, and related to the test conditions.

III. EXPERIMENTAL METHOD

To achieve the objectives outlined in Section II of this document, the CDC Subassembly Test Program will include experiments of basically two types which are categorized as "exploratory" and "analytical". The exploratory tests consist of experiments which are intended to define both "what happens" and "how important are the consequences" when a given fuel element under particular conditions is subjected to transient heating. These experiments provide preliminary and immediate information needed by both industry and the AEC for safety analyses of commercial power reactors. They also provide better direction for other experiments in the program by providing a broad overview and identification of the possible problems to be encountered. The second category of experiments is more analytical and quantitative in that the experiments attempt to define not only "what and how important is what happens" but also to explain the "why" and the "how" of what happens. Testing in this latter category is directed both toward the responses of a fuel element in an accident situation and toward the relationship of these responses to variations in system parameters (such as fuel design, temperature, and pressure). Both types of tests, exploratory and analytical, use similar experimental methods.

Generalized Method

Usually one, but possibly several, fuel rods are tested in each experiment. The fuel rods are situated in an environment of coolant, hardware, other fuel rods, flow constraints, pressure, temperature, etc., designed to replicate one of many initial environmental conditions which may exist in a power reactor at the onset of a reactivity accident. This entire assemblage is contained in a capsule (see Figure A-13) which is placed in the central flux trap of the CDC. In the flux trap, final environment factors of neutron flux and power generation representative of severe accidents are created by power transients induced in the CDC.

During a power transient, measurements such as clad temperature, water temperature, flow velocity, pressure, strain, and others which may be useful and feasible are recorded on magnetic tape. After a test, when the capsule is removed and opened, several other measurements and observations are made, including hydrogen gas concentration, characterization of fuel and cladding, composition of chemical by-products, concentrations, types and locations of fission products, and others which may be diagnostic

of the fuel and environment behavior during the test. Photographs are taken to document the test.

Each test is a variation of the above approach, usually designed either to vary one or more of the initial or neutronic conditions, to enhance the measurement (accuracy or statistics) of a test result, or to improve knowledge of relationships between initial conditions and consequences. For example, a given fuel rod type under given initial conditions will be tested several times under varying amounts of energy deposition in the fuel to establish thresholds for certain consequences and to relate severity (or extent) of the consequences with the severity of the power transient.

Capabilities

Capabilities of the equipment to be used in the execution of the Subassembly Test Program are uniquely suited to the tasks posed by the experimental method described above. These capabilities determine to a great extent the types of tests and measurements which can be undertaken and therefore are discussed here as a part of the experimental method.

Power Burst - The CDC, described in detail in Appendix A, was designed within limits imposed by the pre-existing fuel to serve as a pulse source of neutrons for fuel testing. The shape of the power bursts (see Figure A-10) is controlled by the initial reactivity insertion and the dominant feedback mechanism, Doppler broadening, and thus should approximate well the uncontrolled power shape of a reactivity accident in a conventional UO_2 -fueled power reactor. Presently the reactor can sustain transients with periods as low as 3 msec, and improvement to 2.5 msec periods is expected.

Space Accommodation - The flux trap can accommodate capsules as large as 3.4 inches in diameter, although the present capsule, which is designed to prevent damage to the core from a TNT-energy-equivalent explosion of fuel rods, can accommodate actual experiments up to 3.0 inches in diameter. The 3.0-inch diameter will allow tests of single rods, and possibly small clusters of fuel rods for study of rod interactions.

Energy Deposition - Experiments have shown that with the present 3 msec period limit the following typical energy depositions in UO_2 fuels are possible with single rods.

Clad Material	Clad O.D. (in.)	Enrichment (at. % U^{235})	Energy Deposition (cal/g UO_2)
SS	0.5	3	140
SS	0.466	4.8	300
SS	0.250	3	275
SS	0.250	5	360
SS	0.250	10	580

Thus melting of UO_2 , along with resultant melting and chemical reaction of cladding materials is within the capability of the equipment. With the higher enrichments (5% to 10%) tests should experience complete melting of the UO_2 as well as significant vaporization.

Initial Conditions - The present capsule design was conceived for tests incorporating ambient conditions only, viz., pressures at atmospheric or slightly above, ambient temperature, and stagnant water with a head of about 3-feet. New designs for future capsules to be used in the program can incorporate pressure to 2200 psi, at ambient temperature, and will eventually include higher temperature so as to cover the range of these variables in current reactor design. Flow capability, although not expected for the CDC part of the program, is incorporated in the forthcoming PRF loop design and will complete the replication of the three variables, pressure, temperature, and flow. Capsules with extensions for up to 20 feet of water head are also expected. Another initial condition which is adjustable within the present experimental method is burnup. Burnup, including all the consequences - viz., fission deposition, migration, accumulation in defects, fission gas release, breakup of pellets, melting of UO_2 , sintering, and the effects upon the cladding such as embrittlement, hydriding, and corrosion - is believed to bear importantly both upon consequences and thresholds of consequences in reactivity accidents. The present program initiates a study of these effects by pre-irradiating samples of fuel in a test reactor to various levels of burnup up to and exceeding 20,000 MWd/tonne.

IV. STATUS OF THE CDC PROGRAM^(a)

This section briefly reviews accomplishments of the CDC program from early testing and proving of the core itself and its capabilities through the initial phases of fuel element testing in the Subassembly Program up to the present time. Most of the information contained in this section has been reported in much greater detail both in Monthly Reports and Quarterly Technical Reports of the Spert Project.

A. Core Status

A large part of the testing program conducted to date was proof-testing of the CDC. Beginning with long-period, low-energy transients, the CDC was progressively tested to shorter periods and correspondingly higher enthalpies until an initial Operational Limit was established. During most of these tests, an encapsulated CDC-type rod was irradiated in the central flux trap. Since the fuel rod in the flux trap received a much greater nvt than any fuel rod in the core, it achieved higher energy releases and temperatures and was, therefore, a valuable source of advance information about response of the CDC fuel at higher enthalpies.

Early in the testing of the CDC, fuel rods occasionally ruptured in the core. The hypothesis that these ruptures were due to waterlogging was substantiated by the discovery of leak-paths in the end-closures of some of the ruptured rods. Investigations of the end-closures indicated that they had apparently been damaged in fabrication during swaging. Consequently, the CDC fuel rods were refitted with new end-closures and leak-tested. This action allowed the virtual elimination of waterlogged rods from the core as the transient period was gradually decreased.

Currently the core has a capability of repeated operation at minimum periods of 3.0 msec without deleterious effects. Shorter periods may eventually be attained as the future program demands, but such upgrading of the CDC is being delayed pending the completion of a significant part of the program described herein.

B. Fuel Types Tested

Experiments have been performed exclusively on UO_2 rod-type, metal-clad fuels. These fuel types are referred to as CDC, F, PBF, and SPX. The CDC rods are the same type used in the CDC itself. They are clad with 1/2-inch

(a) This section covers work performed through FY 1968. Results obtained during FY 1969 are reported in References 1 through 10.

OD stainless-steel containing swage-compressed powder enriched to either 3% or 4%. The F rods are stainless-steel clad, 0.466-inch OD, containing pelleted fuel. PBF rods, which are prototypes for the Power Burst Facility, are 3/4-inch in diameter, have an insulator separating the fuel from the clad, and contain a ternary $\text{UO}_2\text{-CaO-ZrO}_2$ fuel mixture. The SPX rods obtained specifically for the CDC testing program, are 1/4-inch OD, stainless steel clad, with enrichments of 3% and 10%.

The following table summarizes physical characteristics of the rods in detail:

TABLE I
PHYSICAL PROPERTIES OF FUEL RODS

Title	CDC	F	PBF	SPX
Clad diameter (inches)	0.500	0.466	0.750	0.250
Clad thickness (inches)	0.028	0.020	0.028	0.014
Clad material	304 SS	304 SS	304 SS	304 SS
Clad condition	35% Cold Worked	Annealed	3% Cold Worked	Annealed
Active length (inches)	28 or 55	36	18 or 36	18
Diametral gap (inches)	0	0.006	0.039	0.002
Fuel material/form	UO_2 /Powder	UO_2 /Pellet	$\text{UO}_2\text{-CaO-}$ ZrO_2 /Pellet	UO_2 /Pellet
Fuel density (g/cc)	9.28	10.08	5.97*	10.4
Fuel burnup	0	0	0	0
Fuel enrichment (%)	3 or 4	4.8	21	3 or 10

* Originally 6.54 g/cc.

C. Summary of Test Conditions and Results

CDC tests were performed on each of the fuel rod types described above to investigate their response to transient energy depositions. The tests are separated into several types for purposes of discussion. Within each type, the purpose, scope and primary results of tests performed, are summarized. It should be recognized that the tests performed to date are primarily of the "exploratory" type, rather than of the "analytical" type. A complete listing of all tests performed is presented in Appendix C.

With only a few exceptions, single test rods were radially centered in capsules and positioned in the region of the axial flux peak in the CDC flux trap. The capsules used were either two or three inch diameter stainless steel cylinders approximately five feet long, filled to within a few inches of the top with ambient-temperature demineralized water. The upper void space contained air at atmospheric pressure. Pre-test examination of test rods included radiography and dimensional measurements to establish the initial geometry of the fuel and clad. During tests, measurements included cladding surface temperature, capsule pressure, capsule strain, reactor power, reactor period and energy deposition. Post-test analyses included measurements of hydrogen evolution, cladding growth, radiography, photography, fuel and cladding characterization, and metallurgical examination.

(a) Single Tests on Normal Fuel Rods

Purpose: If a reactivity accident should occur in a power reactor, the most common condition of the fuel rods undergoing this experience is assumed to be normal and intact, that is, without cracks, waterlogging, or other defects. Tests of intact fuel in the CDC likewise comprise the greatest effort of the program. These tests incorporate fuel rods which are conventional except for enrichment and physical dimensions. They are subjected to tests of known energy release, and observations are made to establish whether damage of any type has occurred, to what extent, and consequences, if any, of the damage.

Results: The tested fuel rods indicate a progressively more damaging response as the energy deposition is increased. These responses are listed in Table II along with the energy deposition levels at which the responses were first observed for each of the pertinent fuel rod types.

At energy levels sufficient to cause breakup of the fuel, SPX-type rods yield pressure pulses of ~ 100 psi at 375 cal/g, ~ 300 psi at 475 cal/g, and ~ 600 psi at 570 cal/g. These higher pressures are accompanied by progressively greater degrees of fragmentation of the fuel rod and a greater fractional conversion of thermal energy into mechanical energy. Metal-water reaction extent also increases from about 1% at 275 cal/g to 36% at 570 cal/g.

(b) Sequential Tests of Normal Fuel Rods

Purpose: Particularly in the cases of the CDC fuel and the PBF fuel, it must be ascertained whether the fuel can withstand repeated tests without deterioration. More specifically, we seek to find the highest

Table II

FUEL ROD RESPONSES

(Unirradiated, Stainless Steel Clad)

Responses	Lowest Energy Deposition at which indicated response was observed* (cal/g of UO ₂)		
	CDC	F	SPX
Slight Discoloration	125	---	---
Slight distortion (bowing and darker discoloration)	127	151	---
Diametral and axial growth	N.D.	100	200*
Circumferential ridging (pellet fuel)	---	215	N.D.
Dulling of surface texture	N.D.	263	200
Loss of clad integrity	N.D.	278	275*
Melting of clad	N.D.	300	275*
Evidence of metal-water reaction	N.D.	300	275*
Fuel rod breakup	N.D.	N.D.	375*
Fragmentation prior to clad melt	N.D.	N.D.	475*

N.D. - No data taken to establish this level.

* - Tests have as yet been widely spaced in energy depositions, so that the responses listed could occur at lower energy depositions.

energy deposition (per transient) which can be repeated a large number of times before the rod becomes damaged beyond further use. Thus, the program has included many sequential tests on single fuel rods. Also, to determine whether rods which appear to have survived a test without damage are actually altered in an unobserved way, or whether they are in any way more vulnerable to failure in subsequent transients, sequence tests on F-type rods have been conducted.

Results: Sequential tests have in all cases been run at enthalpies below the single-test failure point. For CDC rods, the sequence tests never exceeded about 135 cal/g and the major response noted as a result of these tests was thin-layer oxidation of the clad surface combined with slight bowing. Similar results were obtained with single exposures at near the same energy level. Sequence tests of F-rods respectively at 100, 150 and 200 cal/g demonstrated axial growth of the cladding in all cases. The amount of growth per test was approximately constant within each series but increased with the energy release. For example, following 14 tests of an F-rod at 200 cal/g in each test, the rod (originally 40" in length) had grown over 2 inches in length but had not failed. Post-test radiography of these F-rods showed that only the cladding had grown; the fuel stack height remaining unchanged.

Most extensive of the sequential tests has been for the PBF rods. Two prototypes have been tested with the most recent having achieved 100 successive tests at enthalpy levels up to 280 cal/g. Results of these tests are to be published when complete.

(c) Single Tests on Waterlogged Fuel Rods

Purpose: If a fault should exist in any fuel rod which would allow coolant to enter the fuel volume (probably during the cooldown period of a power reactor), then the possibility exists of a violent rupture and dispersal of fuel during either normal reactor startup or a startup power excursion accident. To determine the threshold energy density required for failure and test the consequences of such waterlogging, several fuel rods were deliberately waterlogged and subjected to transient power excursions. Generally, the water was introduced through small holes at either end of the fuel rod and allowed to migrate toward the other end. In some cases, simulated cracks in the cladding were made and the water introduced through the side of the fuel rod. In the cases of CDC-3 and CDC-4 type rods wherein the fuel was swage-compressed powder, both partially and completely waterlogged fuel rods were tested.

Results: A wide separation has been found between the responses of powder rods (with cold-worked cladding) and pellet rods (with annealed cladding). Powder fuel rods of the CDC type display a low threshold for failure and little sensitivity to the concentration of water in the fuel. Most water-logged powder rods have ruptured violently whenever the energy deposition exceeded only about 40 cal/g (equivalent to $\sim 600^{\circ}\text{C}$ UO_2 temperature) producing sharp pressure pulses as high as 13,000 psi. Several of the rods in the CDC core itself also ruptured apparently due to waterlogging, caused pressure pulses sufficient to deform several nearby fuel rods, and in one case caused a neighboring rod to be sheared. In pellet-type fuel rods, waterlogging mostly consists of filling the interstices between pellets and between pellets and clad. The pellets themselves apparently absorb little water. The F-rods described above display an ability to withstand energy depositions up to about 250 cal/g without failure. One rod ruptured at ~ 250 cal/g but another similar rod attained 257 cal/g without rupturing. In non-destructive tests on waterlogged F-rods, the annealed cladding grew diametrically as much as 17% in response to the internal pressures. The single rupture encountered in these pelleted fuel tests yielded a pressure pulse of 1300 psi, much lower than the powder rod failures.

(d) Tests of Other Effects

It is known that the response of fuel rods to transients is affected by environment, ie, temperature of the coolant, constraints against coolant flow, presence of nearby fuel rods or other hardware, pressure, flow aeration of the coolant, and others. Although the arsenal of available equipment is not yet capable of investigating all of the important environmental variations, some tests have been aimed at these effects. Particularly, a few tests of coolant aeration, flow restriction, fuel rod interaction, and head effects have been made. Temperature and pressure effects are expected to be tested in the near future. Also, a few tests have been conducted on CDC and SPX fuel rods to determine possible effects of period where energy release is held constant.

D. Discussion of Test Environment Effects

The CDC tests to date have closely approximated the accident conditions expected to exist in actual power reactors. These tests have achieved a high degree of correspondence to the actual power reactor fuel rods undergoing a reactivity accident particularly in aspects such as rod construction,

rod dimensions, fuel condition, properties of cladding, rate of temperature rise, power distribution, and peak temperature external to the fuel rod, the ability of the CDC test capsules to approximate conditions is dependent upon the reactions displayed by the test rod itself. For example, if a fuel rod undergoing transient heating reacts by simply transferring heat to the coolant without rapid or extensive physical disturbance, its environment need only include a thin annulus of coolant to accommodate heat transfer. However, if the tested fuel rod produces disturbances which travel beyond its own domain, viz, to neighboring fuel rods, then, to the degree that these disturbances do not produce "normal" feedback to the test rod, the CDC test conditions do not accurately simulate a power reactor. With some exceptions, the capsule environment in the CDC does not appear to have had much effect upon the results of most CDC tests. Also, tests which have not produced rod failure have not caused disturbances which could reasonably cause deleterious external effects upon other fuel rods or upon a reactor vessel.

On the other hand, tests in the program involving waterlogged fuel rods have resulted in rods which burst during transient heating producing sharp pressure pulses, rapid thrusts of coolant, and deformation of the rods. In these tests, the most significant results observed are the rupture itself, the threshold energy levels, and the fact that environmental disturbances occur which may be significant with regard to other neighboring fuel rods. Tests which demonstrate the effect of these disturbances on neighboring fuel rods have not been made and are difficult to do in the CDC owing to the size constraint of the capsule walls. It is significant to note in this context, that during pre-testing of the CDC, several fuel rods in the CDC itself were apparently waterlogged and eventually ruptured. Environmental effects of these ruptures ranged from bending of neighbor rods, to in one case, complete shearing of a neighboring rod in the region of an intermediate grid. Even in the case of the sheared rod, however, no indications of the potential for a cascading failure phenomenon were observed.

Several of the tests in the CDC program on normal fuel rods have involved sufficient energy release to cause failure of the cladding, melting, metal-water reaction, loss of fuel, and pressure pulses. The primary results of these tests are that failure of a certain character occurs at certain energy levels, together with disturbances which may affect other rods. Unlike the waterlogged failures, which may be expected to occur isolated in a field of

otherwise intact fuel rods, failures arising from excessive heat generation in reactor cores will occur in a field composed of like fuel rods, all suffering from excessive heat and loss of cladding strength. Thus, even the slightest environmental disturbance of one rod can produce failure of its neighbors. Here again, environmental effects have not been studied but future plans include tests of clusters of rods in an attempt to go one step closer to the full scale reactor conditions.

V. TEST PROGRAM

The planned test program for FY 1969 and FY 1970 is presented in this section. This program, as stated in Section III, proceeds initially with exploratory tests to determine consequences, viz., "what happens" and "the importance of what happens" to fuel elements under particular conditions. These tests will be followed, when necessary, by tests which attempt a better definition of the consequences and the relationship of consequences to certain parametric variables. Tests with parametric variations (usually clad and fuel characteristics, irradiation level and environmental conditions) will provide information on their influence on fuel performance and provide insight into reasons for the observed behavior.

The program will immediately explore a variety of situations before much effort is made toward basic study of mechanisms. In this way, it is felt that valuable information will evolve early to aid the AEC in current safety reviews.

The basic test fuel specimen for the CDC program is the SPX-type fuel rod. This type fuel rod is manufactured specifically for the subassembly test program to a set of precise specifications and subjected to rigorous pre-test inspections. These test specimens include fuel rods with 3, 5, and 10 percent enriched UO_2 , stainless steel and Zircaloy-2 clad, and annealed and cold-worked clad. Data on these rods are given in Table III. This set of test specimens allows the program to establish base-point information with a basic rod design that is well defined, and then, by testing parametric variations, to determine the influence of variations in properties on transient response of fuel rods.

In addition to the SPX-type fuel rods, fuel samples which are representative of commercial, BWR production standards have been obtained from the General Electric Company. These rods (see Table III) will include parametric variations of fuel type (pellet and powder with both UO_2 and $\text{PuO}_2\text{-UO}_2$ fuel) and rod size (0.3125 in. OD and 0.562 in. OD). As a consequence of these variations, information can be obtained on the influence of fabrication procedures, fuel type and rod size on transient fuel response.

TABLE III

DESCRIPTIONS OF TEST FUELS FOR CDC PROGRAM
IN FY-69 AND FY-70

Title	SPX	GEP	GEX	GEXPR	LMFBR
Clad Material	Both Zr-2 and 304 SS	Zr-2	Zr-2	Zr-2	<div>To Be Determined</div>
" Condition	Both Annealed and Cold Worked	~ 10% Cold Worked			
" O.D. (in.)	0.250	0.562	0.3125	0.3125	
" Thickness (in.)	0.014	0.032	0.020	0.020	
Active length (in.)	18 and 5	5.1*	5.1	5.1	
Diametral Gap (in.)	0.002	0.010	0.006	0.006	
Fuel Material	UO ₂	UO ₂	UO ₂	PuO ₂ in	
" Form	Pellet	Both Pellet and Powder		Nat. UO ₂	
" Density (<u>grams</u> cc)	10.4	Pellet - 10.2, Powder - 9.3		Pellet	
				10.2	
Enrichment (at. %)	3%, 5%, & 10% U ²³⁵	7% U ²³⁵	7% U ²³⁵	5.5% Pu ²³⁹ 0.3% Pu ²⁴¹	

* A few GEP rods have an active length of 28 inches (Test Series #115).

The fuel specimens will be tested in a stagnant water environment in a test capsule in the CDC test space. The initial portion of the testing will be conducted in a Type B capsule (see Table IV and Figure A-13). The Type B capsule is capable of providing an atmospheric pressure and ambient temperature water environment for testing of various fuel rods. Additional capsule designs of essentially the same physical dimensions as the Type B capsule are being developed so that initial pressures to 2000 psig, initial temperatures to 620°F and static water heads of up to 20 feet can be incorporated as environmental parameters in the test program.

During each test in the program, many variables will be measured to determine the behavior of the fuel and the consequences of its failure. The exact nature of the measurements for each test will depend on the specific objectives of the individual test, but the capability exists for measuring numerous variables on all tests. Among these are dynamic measurements of cladding surface temperature, capsule pressure, fuel rod internal pressure, capsule wall strain, axial fuel rod growth, fuel rod power density, capsule water velocity and axial fuel stack growth. The types of transducers used for these measurements and their capabilities are summarized in Table V. From these measurements it is possible to determine the time of failure of the fuel rod, its energy density at time of failure, and the conversion ratio of nuclear energy to mechanical energy. In addition to the present capability for dynamic measurements, instrumentation development work is going on to develop techniques for measuring fuel temperatures, radial clad growth and incremental fuel rod growth.

Measurements made after the tests include hydrogen concentration to determine metal-water reaction extent, analyses of fission products in the water and air space, and particle sizing of fuel rod residue. The metal-water reaction extent is measurable to an accuracy of about 10% by measurement of post-test hydrogen evolution and will be independently checked by metallurgical investigation of the oxide content of fuel cladding residue. The fission product analysis will be primarily directed at determination of iodine release and deposition.

Individual tests are grouped together into test series in the following, and each test series is described together with its specific

TABLE IV

TYPE B CAPSULE CHARACTERISTICS

	Type-B Thin Wall
Material	304 SS
Outside Diameter (in.)	3.000
Thickness (in.)	0.065
Inside Cross Section (in. ²)	6.470
Length (in.)	61.375
Volume (empty, in. ³)	397.1
Hydrostatic Pressure Test (psi)	1000

TABLE V
MEASUREMENT CAPABILITY

Measurement	Detector	Response	Range	Accuracy
Nuclear Power	Ion Chamber	200 μ sec	100 GW	2-5%
Clad Temperature	CR/Al Thermocouple	3-5 msec	1400°C	2-5%
	Ir-60 Ro/Ir	< 1 msec	2000°C	2-5%
Capsule Strain	Strain Gauges	10-20 μ sec	10,000 μ in/in	10-20%
Capsule Pressure	Pressure Transducer	10-20 μ sec	20,000 psi	10-20%
Water Velocity	Velocity Transducer	50-100 μ sec	75 ft/sec	10-20%
Internal Fuel Rod Pressure	Pressure Transducer	$\sim$ 3 msec	3,000 psi	10-20%
Axial Fuel Rod Growth	Linear Variable Differential Transformer	1 msec	1 inch	2-5%
Axial Fuel Stack Growth	Linear Variable Differential Transformer	1 msec	0.75 inch	2-5%

objectives. These test summaries are presented for: unirradiated UO_2 (100 series of tests), irradiated UO_2 (200 series of tests), plutonium recycle fuels, (300 series of tests), LMFBR fuels (500 series of tests), and other tests such as instrumentation tests, calibrations, PBF tests and core performance tests (900 series of tests). The order of presentation does not imply an ordering of performance but only a categorization of types of tests. A tentative schedule for performance of these series is given following the summaries of the tests.

A. UNIRRADIATED UO₂ FUELS TESTING

The specific tests planned for the CDC on unirradiated UO₂ fuels (cold, clean, unburned) are presented in this section. The tests are basically directed toward obtaining experimental data as a function of transient energy deposition on the failure thresholds, modes and mechanisms of failure, and failure consequences of UO₂ fuel rods representative of those in boiling and pressurized water reactors. To accomplish this goal, a wide range of fuel design variables and initial conditions will be investigated. The results from these tests will allow evaluation and further development of analytical models used to predict the course and consequences of reactivity accidents in power reactors. In addition, baseline data on response will be established to which test results on irradiated, plutonium recycle and LMFBR fuels can be compared.

Most of the tests will be performed on single rods encapsulated in the flux peak region of the CDC test space. Within the capsule, a test rod is normally surrounded by a water annulus slightly greater than one-inch in thickness. This amount of water is somewhat greater than that associated with a fuel rod in a core lattice, and may affect the magnitude and nature of the consequences observed following fuel rod failure in CDC tests. To evaluate the effect of water volume on test consequences, scoping tests will be performed early in the program with the amount of water surrounding the test rod reduced to that more representative of a core lattice. The results of these preliminary tests will provide guidance in the experimental design and interpretation of results from subsequent tests, some of which will be performed on clusters of three or five fuel rods.

The effects of variations in cladding material (stainless steel and Zr-2) and cladding heat treatment (annealed and cold-worked) on transient response will be scoped during the initial stages of the test program. Subsequently, most tests will be performed on rods clad with cold-worked Zr-2, as this is the type of clad predominantly used in water reactors. By the same token, most tests will be performed on rods containing pelletized UO₂. The performance of this type fuel under operating conditions has proven satisfactory, however, fuel technology studies indicate that powder fuel has comparable operational characteristics in addition to being more economic to fabricate and reprocess. Because of this economic advantage, it is expected that the pressures to use powder fuel in power reactors will increase in the near future. To determine if any significant differences in failure thresholds and consequences exist between pellet and powder fuels, tests on powder fuels are included in the CDC program.

Tests will be conducted to determine the failure thresholds and consequences of fuel rods containing more than normal amounts of water in the fuel. These rods will be representative of those in reactor cores which have cladding defects sufficient to allow coolant water to leak into the internal fuel bearing portion, thus resulting in what are commonly termed "waterlogged" rods. Investigations will be directed towards obtaining information for what is considered to be the worst condition, ie, complete waterlogging with sealed defects located in low temperature regions, which offer no expansion relief during the course of a power excursion.

Exploratory tests will be performed to investigate the effects of hydrodynamic variations within the capsule, initial pressure and initial coolant temperature on transient response. The hydrodynamic variations will include high static heads over the fuel samples and isolation of the water surface to simulate a liquid full system. Initial pressures and temperatures will cover the range

of operating boiling and pressurized reactors. Other tests will be performed to investigate the effects of radial flux depression and axial length on response.

The test series currently planned in this portion of the program are listed below with a brief summary of each series on the following pages.

<u>Series</u>	<u>Title</u>
CDC-102	Scoping Series - Annealed Zr-2 Clad SPX Rods
CDC-103	Scoping Series - Cold-worked Zr-2 Clad SPX Rods
CDC-104	Scoping Series - GEX Pellet and Powder Rods
CDC-105	Scoping Series - GEP Pellet and Powder Rods
CDC-106	Detailed and Length Effect Series - Cold-worked Zr-2 Clad SPX Rods
CDC-107	Detailed Series - GEX Pellet Fuel Rods
CDC-108	Detailed Series - GEX Powder Fuel Rods
CDC-109	Waterlogged Series - Stainless Steel and Zr-2 Clad SPX Rods
CDC-110	Waterlogged Series - GEP Pellet and Powder Fuel Rods
CDC-111	Cluster Tests - Cold-worked Zr-2 Clad SPX Rods
CDC-112	Cluster Tests - GEX and GEP Pellet Fuel Rods
CDC-113	Flux Depression Effects
CDC-115	Length Effect Series - GE Pellet and Powder Fuel Rods
CDC-116	Hydrodynamic Effects - SPX Rods
CDC-117	Pressure Effects - SPX Rods
CDC-118	Pressure and Temperature Effects - SPX Rods

1. TITLE: Scoping Series - Annealed Zr-2 Clad SPX Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to provide scoping information on the failure threshold, modes and mechanisms of failure, and failure consequences of annealed Zr-2 clad fuel as a function of transient energy depositions in the range from 200 to 550 cal/g of UO_2 , (2) to scope the effect on failure consequences of reducing the volume of water surrounding the test sample to that more representative of core lattice configurations, and (3) to evaluate the performance of capsule instrumentation, capsule hardware and experimental techniques over a broad range of dynamic conditions.

3. EXPERIMENT:

a. Test Fuel

Eighteen inch long, three and ten percent enriched SPX rods clad with annealed Zr-2 (see Table III for details).

b. Test Conditions

Single SPX rods will be positioned in the flux peak in Type B CDC capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure. Tests will be performed on the following two configurations.

- (1) Active rod surrounded by normal amount of water in capsule.
- (2) Active rod surrounded by water-filled aluminum tubes. Tubes will be arranged to provide a metal-water ratio in the region of the test rod more nearly representative of that in a core lattice.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy deposition

(3) Post-test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

The following seven tests will be performed with the normal amount of water in the capsule.

- (1) Three percent enriched rod @ 200 cal/g of UO_2
- (2) Three percent enriched rod @ 250 cal/g of UO_2
- (3) Three percent enriched rod @ 275 cal/g of UO_2
- (4) Ten percent enriched rod @ 275 cal/g of UO_2
- (5) Ten percent enriched rod @ 375 cal/g of UO_2
- (6) Ten percent enriched rod @ 475 cal/g of UO_2
- (7) Ten percent enriched rod @ 550 cal/g of UO_2

The following two tests will be performed with aluminum tubes surrounding the test rod.

- (1) Ten percent enriched rod @ 375 cal/g of UO_2
- (2) Ten percent enriched rod @ 475 cal/g of UO_2

4. DISCUSSION:

The results from the tests with a normal amount of water around the test sample will provide the initial CDC fuel performance information on the response of Zr-2 clad oxide fuel over a wide range of transient energy depositions. Baseline information will be obtained on the thresholds, modes, mechanisms, and consequences of fuel failure. The results will also provide guidance for establishing details of subsequent tests in CDC designed to obtain added precision in energy regions of interest and to investigate the effects of fuel design parameters such as clad material, clad heat treatment, flux depression and fuel composition on the phenomena of interest, e.g. CDC-103, 104, 106 and 107. Direct comparison of these test results with those of other tests will provide an indication of the sensitivity of fuel failure and consequences to parametric design variations in the range commonly encountered in boiling and pressurized water power reactors.

Tests with the water volume about the fuel sample reduced will provide preliminary information early in the program on the sensitivity of measured test consequences to metal-water ratio. If significant differences are observed, subsequent test series designed to investigate variations in fuel design parameters, burnup, and environmental conditions may require hardware modifications to reduce the water volume. The tests will also provide guidance in establishing

the detailed design features of cluster tests (CDC-111 and 112).

Pre- and post-test measurements in conjunction with calculations based upon these measurements will be used to evaluate and further develop analytical models designed to predict the course and consequences of potential reactivity accidents in water reactors. Of particular value will be the measurements of cladding temperature, capsule pressure, metal-water reaction extent, and computation of the conversion efficiency of thermal to mechanical energy, all as functions of fuel enthalpy. Availability and use of these data by the reactor industry will allow prediction of the damage potential to reactor hardware of postulated full-scale reactivity accidents with improved confidence. The data will also provide a yardstick by which licensing applications can be more realistically evaluated by the AEC.

5. STATUS:

This test series was completed early in FY 1969 and the preliminary results published in IDO-ITR-102 (Reference 3).

1. TITLE: Scoping Series - Cold Worked Zr-2 Clad SPX Rods.

2. OBJECTIVES:

The objective of this test series is to provide scoping information on the failure threshold, modes and mechanisms of failure, and failure consequences of cold worked Zr-2 clad fuel as a function of transient energy depositions in the range from 275 to 550 cal/g of UO₂.

3. EXPERIMENT:

a. Test Fuel

Eighteen inch long, ten percent enriched SPX rods clad with cold worked Zr-2 (see Table III for details).

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Cold worked rod @ 275 cal/g of UO_2 .
- (2) Cold worked rod @ 375 cal/g of UO_2 .
- (3) Cold worked rod @ 475 cal/g of UO_2 .
- (4) Cold worked rod @ 550 cal/g of UO_2 .

4. DISCUSSION:

The results from a comparable test series on fully annealed Zr-2 clad rods (CDC-102) will have been obtained by the time this series is run. Information from CDC-102 and 103 will be used within the subassembly program for comparison and definition of the relative differences of cold worked and annealed cladding with regard to thresholds, modes, mechanisms, and consequences of fuel failure. Direct comparison of these test results with those of other tests, e.g. CDC-104, 106, and 107, will provide both the AEC and industry with an indication of the sensitivity of fuel failure and consequences to parametric design variations in the range commonly encountered in boiling and pressurized water power reactors, and on a fuel type, i.e., with cold worked clad, representative of that in current use in power reactors.

5. STATUS:

This test series was completed early in FY 1969 and the results reported in IDO-ITR-102 (Reference 3).

1. TITLE: Scoping Series - GEX Pellet and Powder Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to provide scoping information on the failure thresholds, modes and mechanisms of failure, and failure consequences of pellet and powder fuel rods fabricated to commercial standards as a function of transient energy depositions in the range from 200 to 450 cal/g of UO_2 , and (2) to evaluate the performance of capsule instrumentation, capsule hardware and experimental techniques specifically designed for the GE fuel rods.

3. EXPERIMENT:

a. Test Fuel

GEX pellet and powder fuel rods (see Table III for details).

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) cladding surface temperature
- (d) capsule pressure
- (e) internal fuel rod pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power
- (i) energy deposition

(3) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Pellet fuel rod @ 200 cal/g of UO_2
- (2) Powder fuel rod @ 200 cal/g of UO_2
- (3) Pellet fuel rod @ 275 cal/g of UO_2
- (4) Powder fuel rod @ 275 cal/g of UO_2
- (5) Pellet fuel rod @ 375 cal/g of UO_2
- (6) Powder fuel rod @ 375 cal/g of UO_2
- (7) Pellet fuel rod @ 450 cal/g of UO_2
- (8) Powder fuel rod @ 450 cal/g of UO_2

4. DISCUSSION:

These scoping tests will provide AEC and industry with the initial short-period fuel performance information on the comparative behavior of pellet and powder oxide fuel rods over a wide range of transient energy depositions. Identical cladding (10% cold worked Zr-2) is used on both the pellet and powder rods, thus the effect of fuel composition on the thresholds, modes, mechanisms, and consequences of failure can be ascertained by direct comparison of test results at the planned energy deposition levels. The GEX fuel rods are scaled down versions (0.3125 inch diameter of the conventional size fuel rods (0.562 inch diameter) currently being used in boiling water power reactors. The rods were fabricated to commercial standards, the smaller size being selected to allow the attainment of higher energy depositions in the CDC than could be achieved with the larger size rods. Subsequent test series, CDC-105, will include experiments on the large size rods at energies to about 275 cal/g of UO_2 to obtain comparative information regarding the effect of rod diameter on test results. Results from these scoping tests on GEX rods will also establish those energy regions in which additional tests, CDC 107 and 108 are required to reduce statistical uncertainties in the performance data.

5. STATUS:

This test series was completed in the second quarter of FY 1969 and the results reported in IDO-ITR-103 (Reference 4).

1. TITLE: Scoping Series - GEP Pellet and Powder Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to obtain scoping data on the response of conventional diameter (0.562 inch) BWR-type pellet and powder fuel fabricated to commercial standards for energy depositions in the range from 200 to 275 cal/g of UO_2 , and (2) to determine the effects of rod diameter on response by direct comparison with similar tests (CDC-104) performed on the smaller diameter (0.3125 inch) GEX fuel rods.

3. EXPERIMENT:

a. Test Fuel

GEP pellet and powder fuel rods (see Table III for details)

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) cladding surface temperature
- (d) capsule pressure
- (e) internal fuel rod pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power
- (i) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-

test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Pellet fuel rod @ 200 cal/g of UO_2
- (2) Powder fuel rod @ 200 cal/g of UO_2
- (3) Pellet fuel rod @ 275 cal/g of UO_2
- (4) Powder fuel rod @ 275 cal/g of UO_2

4. DISCUSSION:

The fuel rod samples to be used during this test series have a diameter (0.562 inch) typical of that conventionally used in boiling water power reactors. Rods of this size and enrichment (7%) can be subjected to maximum energy depositions of about 275 cal/g of UO_2 with the present operational capability of the CDC. An energy deposition of 275 cal/g is expected to be in the region of the initial failure threshold for rods of this type. By comparing the results from these tests to similar tests on the smaller diameter GEX fuel rods, (CDC-104), information will be obtained on the effect of rod diameter on test results. This information will be of use within the subassembly program and to industrial groups in the proper interpretation and evaluation of higher energy tests (in excess of 275 cal/g) on the GEX rods.

5. STATUS:

This test series was completed in the second quarter of FY-1969 and the results published in IDO-ITR-104 (Reference 5).

1. TITLE: Detailed and Length Effect Series - Cold-Worked Zr-2
Clad SPX Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to evaluate the feasibility of using five-inch long SPX rods in subsequent test series (eg, CDC-111, -113 and -201), (2) to provide more precise definition of the failure threshold, modes and mechanisms of failure, and failure consequences of cold-worked Zr-2 clad fuel as a function of transient energy depositions in the range of 200 to 500 cal/g of UO_2 , and (3) to obtain a measure of the reproducibility of test results by repeating several tests performed during the scoping series on this type fuel rod (CDC-103).

3. EXPERIMENT:

- a. Test Fuel

Five inch long, ten percent enriched SPX rods clad with cold worked Zr-2 (see Table III for details).

- b. Test Conditions

Single rods will be positioned in the flux peak in Type-B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

- c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

- (1) Pre-Test Examinations

- (a) radiographs
 - (b) photographs
 - (c) axial length
 - (d) diameter

- (2) Dynamic Measurements

- (a) axial fuel rod growth
 - (b) cladding surface temperature
 - (c) capsule pressure
 - (d) capsule wall strain
 - (e) water column velocity
 - (f) power
 - (g) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) metallographic investigation

d. Tests

- (1) Cold-worked rod @ 200 cal/g of UO_2
- (2) Cold-worked rod @ 250 cal/g of UO_2
- (3) Cold-worked rod @ 275 cal/g of UO_2
- (4) Cold-worked rod @ 300 cal/g of UO_2
- (5) Cold-worked rod @ 325 cal/g of UO_2
- (6) Cold-worked rod @ 350 cal/g of UO_2
- (7) Cold-worked rod @ 375 cal/g of UO_2
- (8) Cold-worked rod @ 425 cal/g of UO_2
- (9) Cold-worked rod @ 475 cal/g of UO_2
- (10) Cold-worked rod @ 500 cal/g of UO_2

4. DISCUSSION:

The scoping series on 18 inch long, cold-worked Zr-2 clad rods (CDC-103) will be completed prior to this time, thus providing scoping information on the transient response of these fuel rods. Tests in series CDC-106 will provide data needed to evaluate the effects of reducing the active length of SPX rods from 18 to 5 inches. If the response is determined to be relatively insensitive to the reduction in length, then subsequent test series (CDC-111, -113, -116, -117, -118, and -201) will be performed using the 5-inch long fuel rods. The short rods will have an axial flux profile which is very nearly flat, thus allowing metal-water reaction extent and nuclear to mechanical energy conversion efficiency to be determined more accurately. In addition, tests in this series will allow the threshold and consequences of failure to be defined more precisely as a function of energy deposition. Repetition of several tests previously performed will provide a measure of the reproducibility of test results.

The SPX rods to be used in the irradiated fuels tests (CDC-201) will also be of the five-inch length. Several of the tests in this series (CDC-106) are being performed at the same energy depositions planned for the irradiated tests. Thus, differences in response due to fuel burnup can be detected by direct comparison of experimental data. The results from these tests will thus be useful for planning

and comparative purposes within the subassembly program, and will provide the AEC and industry with more accurate information on conversion efficiency and metal reaction extent.

5. STATUS:

Tests in this series were completed in the fourth quarter of FY 1969 and the results are published in IN-ITR-107 (Reference 8)..

1. TITLE: Detailed Series - GEX Pellet Fuel Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to provide more precise definition of the failure threshold, modes and mechanisms of failure, and failure consequences of GEX pellet fuel rods as a function of energy depositions in the range from 275 to 450 cal/g of UO_2 ; and (2) to obtain a measure of the reproducibility of test results by repeating several tests performed during the scoping series on this type fuel rod (CDC-104).

3. EXPERIMENT:

a. Test Fuel

GEX pellet fuel rods (See Table III for details)

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) cladding surface temperature
- (d) capsule pressure
- (e) internal fuel rod pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power
- (i) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Pellet fuel rod @ 275 cal/g of UO_2
- (2) Pellet fuel rod @ 325 cal/g of UO_2
- (3) Pellet fuel rod @ 375 cal/g of UO_2
- (4) Pellet fuel rod @ 425 cal/g of UO_2
- (5) Pellet fuel rod @ 450 cal/g of UO_2

4. DISCUSSION:

The intent of this test series is to further define the dependence of fuel rod response on energy deposition and to investigate the reproducibility of test results. As such, the tests represent an extension of the scoping series (CDC-104) on this type of fuel rod. Energy depositions sufficient to cause cladding failure are concentrated on, beginning with a test for which delayed failure (failure from clad melting) is expected and continuing to the highest energies possible for which prompt failure (failure from internal pressure) is expected. Repetition of several tests performed previously will provide insight into the reproducibility of tests results. The particular energies chosen for these tests may be modified slightly on the bases of the results obtained during the scoping series. These data will be used in the same way as those obtained during CDC-106.

5. STATUS:

Tests in this series will be initiated during the first quarter of FY 1970.

1. TITLE: Detailed Series - GEX Powder Fuel Rods

2. OBJECTIVES:

The objectives of this test series are: (1) to provide more precise definition of the failure threshold, modes and mechanisms of failure, and failure consequences of GEX powder fuel rods as a function of transient energy depositions in the range from 275 to 450 cal/g of UO₂, and (2) to obtain a measure of the reproducibility of test results by repeating several tests performed earlier during the scoping series on this type fuel rod (CDC-104).

3. EXPERIMENT:

a. Test Fuel

GEX powder fuel rods (see Table III for details)

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (e) energy deposition

(3) Post-test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Powder fuel rod @ 275 cal/g of UO_2
- (2) Powder fuel rod @ 325 cal/g of UO_2
- (3) Powder fuel rod @ 375 cal/g of UO_2
- (4) Powder fuel rod @ 425 cal/g of UO_2
- (5) Powder fuel rod @ 450 cal/g of UO_2

4. DISCUSSION:

As in the tests on GEX pellet rods (CDC-107) the intent of this test series, CDC-108, is to further define the dependence of fuel rod response on energy deposition and to investigate the reproducibility of test results. The tests therefore represent an extension of the previously performed scoping series on GEX powder fuel rods (CDC-104). No CDC experience on the destructive behavior of powder fuel will exist until the scoping tests of CDC-104 are completed; therefore, the specific tests to be performed in this series are subject to change pending the results of the scoping series. The detailed series will begin at the lowest energy density at which clad failure was observed in the scoping series and continue to the highest energy possible. Data will thus be obtained in the transition region from delayed to prompt failure.

Comparison of the results from these tests on powder fuel to those obtained previously on pellet fuel in CDC 104 and 107 will allow further evaluation of the behavioral differences between powder and pellet type fuels under accident conditions. Repetition of several tests performed previously on the powder fuel will also provide additional insight into the reproducibility of test results.

5. STATUS:

Tests in this series were completed during the third quarter of FY 1969 and the results are reported in IDO-ITR-106 (Reference 7).

1. TITLE: Waterlogged Series - Stainless Steel and Zr-2 Clad SPX Rods

2. OBJECTIVES:

The objective of this test series is to determine the effect of clad material (stainless steel and Zr-2) and clad heat treatment (cold worked and annealed) on failure thresholds and consequences of waterlogged SPX fuel rods as a function of energy depositions in the range from 200 to 300 cal/g of UO_2 .

3. EXPERIMENT:

a. Test Fuel

Five inch long, five percent enriched SPX rods clad with annealed and cold-worked Zr-2 and annealed and cold-worked stainless steel. (See Table III for details.) Waterlogging will be accomplished by filling the internal void space of each rod with water through small holes drilled in the end plug regions. The holes will be sealed prior to testing.

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements will include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity

- (f) power
- (e) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Three tests on annealed stainless steel clad rods; 200, 250, and 300 cal/g of UO_2 .
- (2) Three tests on cold worked stainless steel clad rods; 200, 250, and 300 cal/g of UO_2 .
- (3) Three tests on annealed Zr-2 clad rods; 200, 250, and 300 cal/g of UO_2 .
- (4) Three tests on cold worked Zr-2 clad rods; 200, 250, and 300 cal/g of UO_2 .

4. DISCUSSION:

Waterlogged fuel, ie, fuel rods which contain coolant water from cladding faults, is generally not considered to be a hazardous condition for reactors operating at steady state power levels. It is recognized, however, that during a reactivity accident, such waterlogged fuel can rupture, sometimes violently, and with potentially serious consequences.

This test series will provide information on the effects of variations in clad material (stainless steel and Zr-2) and clad heat treatment (annealed and 10% cold worked) on the failure thresholds and consequences of waterlogged UO_2 pellet fuel rods. These data will allow both industry and the AEC to more realistically assess the potential hazard of waterlogged fuel rods in power reactors.

5. STATUS:

Tests in this series were completed during the fourth quarter of FY 1969.

1. TITLE: Waterlogged Series - GEP Pellet and Powder Fuel Rods

2. OBJECTIVES:

The objective of this test series is to determine the failure threshold and consequences for waterlogged GEP pellet and powder fuel rods.

3. EXPERIMENT:

a. Test Fuel

GEP pellet and powder fuel rods (see Table III for details). Waterlogging will be accomplished by filling the internal void space of each rod with water through small holes drilled in the end plug regions. The holes will be sealed prior to testing.

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) cladding surface temperature
- (d) capsule pressure
- (e) internal fuel rod pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power
- (i) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (i) metallographic investigation

d.. Tests

- (1) Waterlogged Pellet rod @ 250 cal/g of UO_2 .
- (2) Waterlogged Pellet rod @ 250 cal/g of UO_2 .
- (3) Waterlogged Powder rod @ 250 cal/g of UO_2 .
- (4) Waterlogged Powder rod @ 250 cal/g of UO_2 .

4.. DISCUSSION:

The tests performed in this series will provide data on the failure thresholds and consequences of commercially fabricated fuel rods of the BWR-type containing pellet and powder fuel. Previous experience indicates that the failure threshold of pellet rods is the order of 250 cal/g of UO_2 , whereas the failure threshold of powder rods is the order of 40 cal/g of UO_2 . These data were obtained for fuel rods having somewhat different cladding characteristics, and may not directly apply to the GEP rods.

The energy deposition required to fail the rods will be determined by observing the time at which the pressure pulse occurs and computing the energy in the fuel at this time. Previous experiments have shown that the pressure pulse is a reliable indicator of cladding failure. Peak pressure and calculations of the conversion efficiency of thermal to mechanical energy will provide information of value to both AEC and industry in predicting possible interaction effects and potential hardware damage resulting from failure of waterlogged powder fuel rods under accident conditions.

5.. STATUS:

Tests in this series were completed during the third quarter of FY 1969 and the results reported in IDO-ITR-105 (Reference 6).

1. TITLE: Cluster Tests - Cold Worked Zr-2 Clad SPX Rods

2. OBJECTIVES:

The objectives of these tests are: (1) to determine the effects of increasing the metal-to-water ratio to that representative of core lattice configurations on the response of fuel rods subjected to energy depositions sufficient to result in damage and/or failure of one or more rods, and (2) to obtain scoping information on the extent of interactions between adjacent fuel rods.

3. EXPERIMENT:

a. Test Fuel

Five inch long, three and ten percent enriched SPX fuel rods clad with cold-worked Zr-2 (see Table III for details).

b. Test Conditions

Three and five rod clusters will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure. Tests will be performed on the following four configurations:

- (1) Active rod surrounded by dummy rods - A ten percent enriched rod radially centered in the capsule will be surrounded by four dummy fuel rods to reduce the amount of water around the central rod.
- (2) Five rod cluster of ten percent enriched rods - A central rod will be radially centered in the capsule surrounded by four rods spaced equidistantly.
- (3) Three rod cluster of ten percent enriched rods - Three rods will be radially positioned on a triangular pitch in the center of the capsule.
- (4) Five rod cluster with higher enriched center rod - A ten percent enriched rod will be radially centered in the capsule surrounded by four, three percent enriched rods spaced equidistantly.

Conceptual design of the hardware and materials for these tests is in the preliminary stages.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) cladding surface temperature
- (b) capsule pressure
- (c) capsule wall strain
- (d) water column velocity
- (e) power
- (f) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Five rod cluster, central rod 10% enriched, surrounded by dummy rods @ 275, 375, and 475 cal/g of UO_2 .
- (2) Five rod cluster of 10% SPX rods @ 275, 375, and 475 cal/g of UO_2 .
- (3) Three rod cluster of 10% SPX rods @ 275, 375, and 475 cal/g of UO_2 respectively.
- (4) Five rod cluster of SPX rods, central rod 10% enriched, surrounding rods 3% enriched, @ 275, 375, and 475 cal/g of UO_2 .

4. DISCUSSION:

These cluster tests are intended to provide gross information on the response of several fuel rods spaced in a lattice configuration representative of that which exists in power reactors. The configurations will be designed to simulate the thermal and hydraulic coupling between adjacent fuel rods. To accomplish this coupling, the surface-to-surface distance between fuel rods and the total water volume for the assembly will be chosen to match as nearly as possible those which exist in integral cores.

Reduction in water volume and thermal rod coupling are expected to result in higher clad temperatures, lower failure thresholds, and increases in the conversion efficiency of thermal to mechanical energy for the cluster tests as compared to single rods exposed to the same energy depositions. To determine the effect of reduction of water volume, a set of tests in this series will be conducted on a cluster of five rods, the four outside rods being dummies and the central rod being fueled.

In order to determine both water volume reduction effects and thermal coupling, two sets of tests are planned: one containing a five rod cluster, all fueled, and one containing a three rod cluster, all fueled.

To explore interaction effects after failure of one rod in a cluster, one set of tests will be conducted with a five rod cluster containing a central rod of 10% enrichment and peripheral rods of 3% enrichment. For a given period, the central rod will experience greater energy release than the peripheral rods, and thus tests can be run in which failure induced directly by energy deposition is achieved in only the central rod. Consequently, these tests will provide scoping information on the effects a rod failure has on adjacent rods, and may thus help answer the question of whether failure cascading is likely in core lattice configurations.

The CDC test space size, environment, and energy deposition capability preclude the investigation of clusters containing large numbers of fuel rods with the many attendant parametric variations which may be of importance. Notwithstanding these limitations, results from the tests planned in this series are expected to provide valuable guidance in the planning and execution of subsequent cluster tests in both the CDC and PBF. In addition, the results are expected to be of some use to AEC and industry in the extrapolation via calculational models, of single rod test data to the behavior expected in reactor core lattices.

5. STATUS:

Tests in this series will be initiated during the first quarter of FY 1970.

1. TITLE: Cluster Tests - GEX and GEP Pellet Fuel Rods

2. OBJECTIVES:

The objectives of these tests are: (1) to determine the effects of increasing the metal-to-water ratio to that representative of BWR core lattice configurations on the response of fuel rods subjected to energy depositions sufficient to result in damage and/or failure of one or more rods, and (2) to obtain scoping information on the extent of interaction between adjacent fuel rods.

3. EXPERIMENT:

a. Test Fuel

Cluster of five GEX fuel rods and cluster of three GEP fuel rods (see Table III for fuel rod details).

b. Test Conditions

Rod clusters will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure. Fuel rods will be radially positioned in the capsule to simulate as near as possible the thermal and geometrical conditions of BWR core lattice configurations. Conceptual design of the hardware and materials for these tests is in the preliminary stage.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) cladding surface temperature
- (b) capsule pressure
- (c) capsule wall strain
- (d) water column velocity
- (e) power
- (f) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Five rod cluster of GEX pellet rods @ 350 cal/g of UO_2 .
- (2) Three rod cluster of GEP pellet rods @ 250 cal/g of UO_2 .

4. DISCUSSION:

The GEX and GEP cluster tests are included in the testing program to obtain scoping information on the response of several rods spaced in a lattice configuration representative of that in boiling water power reactors. The purpose, expected results, and limitations of these tests are similar to those discussed previously for the SPX cluster tests (CDC-111). Comparison of the data from the GE cluster tests with data from single GE rod tests (CDC-104, 105, 107, and 108) will be useful in extrapolating the single rod test results to reactor core lattice geometries. Information on interaction effects between rods may also be obtained.

5. STATUS:

Tests in this series will begin during the first quarter of FY 1970.

1. TITLE: Flux Depression Effects

2. OBJECTIVES:

The objective of this test series is to determine the effect radial flux depression has on the failure threshold, modes and mechanisms of failure, and failure consequences of fuel rods.

3. EXPERIMENT:

a. Test Fuel

Five inch long, three and ten percent enriched SPX rods clad with cold-worked Zr-2 (see Table III for details).

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure. The capsule water will be poisoned, either with boric acid or a cadmium sleeve during tests on ten percent enriched rods, to obtain the same figure-of-merit as for three percent enriched rods.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (e) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Three percent enriched rod @ 250 cal/g of UO_2 .
- (2) Ten percent enriched rod @ 250 cal/g of UO_2 .
- (3) Three percent enriched rod @ 275 cal/g of UO_2 .
- (4) Ten percent enriched rod @ 275 cal/g of UO_2 .
- (5) Three percent enriched rod @ 350 cal/g of UO_2 .
- (6) Ten percent enriched rod @ 350 cal/g of UO_2 .

4. DISCUSSION:

A major portion of the fuel performance experiments to be conducted in the CDC will use fuel samples having enrichments in the 5 to 10% range, whereas boiling and pressurized water power reactors normally use fuel enriched the order of 2 to 3%. The higher enrichment is required for CDC tests in order to attain fuel enthalpies sufficient to cause complete melting and partial vaporization of fuel. The primary effect of increased enrichment in the test samples is to cause a higher radial flux and power depression during transient testing. A greater fraction of the energy in the fuel is deposited near the cladding, which could affect the thresholds and modes of failure. The tests in this series will provide data on the effects of variations in flux depression for rods subjected to the same energy deposition rates and magnitudes.

The three and ten percent enriched SPX fuel rods have flux depressions of about 7 and 24%, respectively. In order to keep the energy deposition rate constant for each energy level tested, it will be necessary to increase the absorption cross section in the capsule for the ten percent rods. This will be done either with a water soluble poison, eg, boric acid, or with a solid poison sleeve, eg, cadmium, which lines the inside wall of the capsule. The planned tests include a pair slightly below the expected failure threshold, a pair slightly above the failure threshold, and a pair at the maximum energy

possible. The results will provide experimental evidence of the significance of variations in flux depression, and thus allow further insight into the proper interpretation and applicability of the fuel performance data from the overall program. As such, the data will be primarily of use for analysis purposes within the Subassembly Program.

5. STATUS:

This test series will be initiated during the third quarter of FY 1970.

1. TITLE: Length Effect Series - GE Pellet and Powder Fuel Rods

2. OBJECTIVES:

The objective of the test series is to determine the effects of increasing the length of GEP and GEX fuel rod samples from about 8-inches to 28-inches on the dynamic response of the rods to transient energy depositions.

3. EXPERIMENT:

a. Test Fuel

28-inch length GEP pellet and powder fuel rods (see Table III for details)

b. Test Conditions

Single rods will be positioned in the flux peak in Type B capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (e) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-

test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) 28-inch GEP pellet fuel rod @ 250 cal/g of UO_2
- (2) 28-inch GEP powder fuel rod @ 250 cal/g of UO_2
- (3) 28-inch GEX pellet fuel rod @ 400 cal/g of UO_2
- (4) 28-inch GEX powder fuel rod @ 400 cal/g of UO_2

4. DISCUSSION:

All GEP and GEX fuel rod samples except the four in this test series are about 8-inches in length (5-inch active length). The short length samples were chosen to minimize the axial peak to average flux ratio over the active length of the rods in the CDC test space. A relatively flat axial flux means all of the fuel in the test sample is subjected to about the same energy density, thus, thermal to mechanical energy conversion and metal-water reaction extent can be computed more accurately than for longer samples. It is assumed that the length of the short samples is sufficient to provide fuel performance information that is representative of that which would be expected for the much longer fuel rods used in power reactors. To evaluate the validity of this assumption, the tests in this series are being performed on 28-inch long GEP pellet and powder fuel rods. The test results will be directly compared with those previously obtained on the shorter fuel samples in series CDC-105. Of particular interest will be a comparison of thermal to mechanical energy conversion efficiency to provide an indication of the appropriate weighting factors which must be applied to reactor cores containing fuel rods at various enthalpies during an accident. The energy depositions possible in this series with the current CDC capability may not be sufficient to provide meaningful energy conversion data; however, information related to axial length effects will also be obtained during tests on GPX fuel analysis purposes within the subassembly program.

5. STATUS:

Tests in this series will be completed during the first quarter of FY 1970.

1. TITLE: Hydrodynamic Effects - SPX Rods

2. OBJECTIVES:

The objective of these tests is to determine the effects of variations in the hydrodynamic characteristics within the capsule on the response and consequences of fuel rods subjected to transient energy depositions from 200 to 550 cal/g of UO_2 .

3. EXPERIMENT:

a. Test Fuel

Five inch long, three and ten percent enriched SPX rods clad with cold-worked Zr-2 (see Table III for details).

b. Test Conditions

Single rods will be positioned in the flux peak in capsules containing ambient temperature demineralized water at atmospheric pressure. The hydrodynamics internal to the capsules will be varied in the following four ways:

- (1) Dynamic loading of the water column - A conventional Type B capsule with a ten-inch air space in the top will be used. The velocity transducer plug which comprises the upper portion of the water column is normally of the same density as water. In this case, the plug weight will be increased to simulate the inertial head loading associated with BWR's.
- (2) Orifice or rupture disc - A conventional Type B capsule with a ten-inch air space in the top will be used. The water surface will be isolated from the air space by either a slow relief orifice valve or a rupture disc to approximate the liquid-full condition of PWR's.
- (3) High static head with free water surface - A special capsule will be developed to provide a static head of up to 20 ft over the fuel sample. The upper ten-inches will contain air and the water surface will be unrestricted.
- (4) High static head with isolated water surface - The same as (3) except that the water surface will be isolated with either a slow relief orifice valve or a rupture disc.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy

(3) Post-test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

The following seven tests will be performed for each of the four test condition variations:

- (1) Three percent enriched rod @ 200 cal/g of UO_2
- (2) Three percent enriched rod @ 250 cal/g of UO_2
- (3) Three percent enriched rod @ 275 cal/g of UO_2
- (4) Ten percent enriched rod @ 275 cal/g of UO_2
- (5) Ten percent enriched rod @ 375 cal/g of UO_2
- (6) Ten percent enriched rod @ 475 cal/g of UO_2
- (7) Ten percent enriched rod @ 550 cal/g of UO_2

4. Discussion

The tests in this series are designed to provide scoping information on the effects of variations in the hydrodynamics of the capsule environment on the transient response of fuel rods. Both BWR's and PWR's operate with a somewhat higher coolant head over the fuel than is normally used in CDC tests. In addition, the primary vessel in PWR's is liquid-full, whereas

tests in CDC are normally conducted with an expansion space in the capsule to minimize the possibility of capsule failure during fuel performance testing. The test conditions for this series of experiments will provide information relative to both the high-head and liquid-full situations. Initially, variations in the hydrodynamics of Type B capsules will be made by increasing the inertial load of the water column and by placing a slow relief orifice valve or rupture disc at the water surface. Tests under these conditions will provide limited information because the time required for pressure relief is shorter than for the actual high head case. Later tests will be done in a specially designed capsule which has provision for static heads to 20 ft. The results from these tests should display the differences in response due to the high head and liquid-full conditions. Tests in this series will be useful in planning subsequent experiments in CDC, and later in PBF, designed to further investigate the effects of environmental conditions on fuel rod response and failure consequences.

5. STATUS:

Tests in this series will be initiated during the fourth quarter of FY 1970.

1. TITLE: Pressure Effects - SPX Rods

2. OBJECTIVES:

The objective of this test series is to determine the effect of 2000 psi internal capsule pressure on the response of fuel rods as a function of transient energy deposition from 200 to 550 cal/g of UO₂.

3. EXPERIMENT:

a. Test Fuel

Five inch long, three and ten percent enriched SPX rods clad with cold-worked Zr-2 (see Table III for details).

b. Test Conditions

Single rods will be positioned in the flux peak in capsules filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at 2000 psi.

c. Measurements

To obtain information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy deposition

(3) Post-test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

- (1) Three percent fuel rod @ 200 cal/g of UO_2
- (2) Three percent fuel rod @ 250 cal/g of UO_2
- (3) Three percent fuel rod @ 275 cal/g of UO_2
- (4) Ten percent fuel rod @ 275 cal/g of UO_2
- (5) Ten percent fuel rod @ 375 cal/g of UO_2
- (6) Ten percent fuel rod @ 475 cal/g of UO_2
- (7) Ten percent fuel rod @ 550 cal/g of UO_2

4. DISCUSSION:

The intent of this test series is to obtain scoping information on the effect of 2000 psi internal capsule pressure on the failure threshold, modes and mechanisms of failure, and failure consequences of SPX fuel rods as a function of transient energy depositions in the range from 200 to 550 cal/g of UO_2 . Internally pressurizing the capsule will increase the saturation temperature of the water to about 335°C and will externally pre-stress the fuel rod cladding. The higher saturation temperature will delay the onset of boiling heat transfer, and will thus affect the temporal behavior of the cladding surface temperature and the fuel enthalpy throughout the tests. This may affect the magnitude of the energy deposition required to cause failure from clad melting, commonly referred to as delayed failure, compared to that observed at atmospheric pressure. The external pressure on the cladding is also expected to affect the failure point at higher energies, referred to as prompt failure, where internal pressure causes cladding rupture with the clad below the melting temperature. The consequences associated with those failures which disperse hot fuel into the water may also be modified as a result of the higher saturation temperature.

These tests will investigate the effect of environmental pressure on the response of fuel samples. The results will provide scoping data at pressure high enough to encompass those encountered in operating boiling and pressurized water reactors. The data will be useful for analysis and planning purposes within the subassembly program, and will provide operational information needed prior to performance of series CDC-118.

5. STATUS:

Tests in this series will be initiated during the first quarter of FY 1971.

1. TITLE: Pressure and Temperature Effects - SPX Rods

2. OBJECTIVES:

The objective of this test series is to determine the effect of elevated temperatures and pressures representative of those in operating boiling and pressurized water reactors on the response of fuel rods as a function of transient energy depositions from 200 to 550 cal/g of UO₂.

3. EXPERIMENT:

a. Test Fuel

Five inch long, three and ten percent enriched SPX rods clad with cold-worked Zr-2 (see Table 111 for details).

b. Test Conditions

Single rods will be positioned in the flux peak in capsules filled to within ten inches of the top with demineralized water. The top ten inches will contain pressurized air. Two operating conditions will be investigated; 1000 psi pressure and ~ 540°F representative of BWR conditions, and 2000 psi pressure and ~ 620°F representative of PWR conditions.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements will include:

(1) Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) cladding surface temperature
- (c) coolant temperature
- (d) capsule pressure
- (e) capsule wall strain
- (f) water column velocity
- (g) power
- (h) energy deposition

(3) Post-test Measurements and Examination

The extent and nature of the post test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

The following seven tests will be performed first at 1000 psi and $\sim 540^{\circ}\text{F}$; then repeated at 2000 psi and $\sim 620^{\circ}\text{F}$. (Energy depositions listed are with reference to ambient temperature, ie, the enthalpy change owing to the initial coolant temperature is included).

- (1) Three percent enriched rod @ 200 cal/g of UO_2
- (2) Three percent enriched rod @ 250 cal/g of UO_2
- (3) Three percent enriched rod @ 275 cal/g of UO_2
- (4) Ten percent enriched rod @ 275 cal/g of UO_2
- (5) Ten percent enriched rod @ 375 cal/g of UO_2
- (6) Ten percent enriched rod @ 475 cal/g of UO_2
- (7) Ten percent enriched rod @ 550 cal/g of UO_2

4. DISCUSSION:

This series of tests will represent the first CDC fuel performance experiments conducted with the capsule water at both high temperature and high pressure. The coolant conditions will approximate the operational temperature-pressure combinations in current boiling and pressurized water reactors, ie, $\sim 540^{\circ}\text{F}$ - 1000 psi and $\sim 620^{\circ}\text{F}$ - 2000 psi, respectively. The effect of these conditions on the failure thresholds, modes and mechanisms of failure and failure consequences will be scoped as a function of energy deposition. SPX fuel rods clad with cold-worked Zr-2 will be used in the experiments primarily because the response of these type rods at ambient conditions (series CDC-103 and 106) will have previously been well established. Direct comparison of ambient with elevated temperature-pressure test results will provide AEC and industry with a gross measure of the differences in failure thresholds and consequences brought about by the changes in environment. This information will also allow further evaluation of calculational models and provide program guidance needed to effectively plan and execute subsequent investigations to better understand environmental effects.

5. STATUS:

Tests in this series will be initiated during the first quarter of FY 1971.

B. Irradiated Fuels Testing

The reactivity accident is theoretically possible at any time during the operating cycle between fuel recharges. As a consequence, the levels of burnup existing in the reactor fuel can vary from zero up to the discharge level normally expected for that class of reactor. At the present time, water reactors expect to discharge fuel at levels ranging roughly between 15,000 and 20,000 MWd/tonne. Expectations for near-future plants go higher - to 30,000 MWd/tonne - and probably will go higher still as the technology develops. For these reasons, safety studies which aim to establish what happens to fuel rods and the fuel rod environment as the reactivity accident proceeds must include burnup as a parameter since it may affect fuel rod responses. Effects of burnup on the cladding include both increase of strength and loss of ductility. In the UO_2 fuel itself, breakup of pellets, grain growth, sintering, "re-healing" of cracks, deposition of fission products, migration and accumulation of gases in preferred sites, and release of volatiles - all are experienced during burnup.

Many of the effects of burnup in the fuel and cladding are dependent upon the operating history of the rod - viz, operating time, temperature of the clad, corrosion, UO_2 peak temperature, and others. Since it is not economically feasible to undertake a study of the effects of each variable in the burnup history of a fuel rod, a judicious selection must be made of the variables to be studied and the levels to be set for those that are not studied. In the present program, burnup has been selected as the primary variable. Other factors, such as cladding surface temperature, UO_2 temperature, heat flux and rod fabrication, are set to levels or standards which closely represent working reactors. Some deviations, however, have been found necessary to accommodate needs of the program and these include rod diameter and UO_2 enrichment.

The energy which is released in a reactivity accident is, of course, also unknown and constitutes a second essential variable, making the program two dimensional. Considered as a function of increasing energy release, the reactivity accident passes first through a region of "low" but increasing consequence and then into a region of "low" and decreasing probability. At the present time greatest attention is given to that intermediate energy region which may have significant levels of both probability and consequence. This range is perceived by some to be between 200 cal/g and 500 cal/g with most emphasis between 200 and 350 cal/g - a region well represented in this test program.

Measurements such as fuel growth, fuel enthalpy and cladding growth, and inspections of the fuel rod after the test will establish the thresholds at which reactivity accidents first become severe and may point out modifications to reactor hardware which could ameliorate the consequences of such accidents. Measures of the amount of thermal energy converted into the form of kinetic energy and associated pressures will be weighted to predict the potential impact upon reactor hardware of full-scale reactivity accidents. Predictions of vessel strain, vessel rupture, control system damage, fuel damage, piping strain and damage may be made with increased credibility after acquisition of this data; and, should rupture of the vessel be predicted to occur, information relative to fission product release obtained in these tests should prove useful in predicting off-site hazard levels.

The test series currently included in this portion of the program are listed below with a detailed summary of each series on the following pages.

<u>Series</u>	<u>Title</u>
CDC-201	Variable Burnup Effects, SPX Pellet Rods
CDC-202	End-of-Core-Life Effects, GEX Pellet Rods
CDC-203	End-of-Core-Life Effects, GEX Powder Rods
CDC-204	Thermal and Neutron Sintering Effects, GEX Powder Rods
CDC-205	End-of-Core-Life Effects, GEP Pellet Rods
CDC-206	Intermediate Burnup Effects, GEX Pellet Rods
CDC-207	Intermediate Burnup Effects, GEP Powder Rods
CDC-208	Low Burnup Effects, GEX Pellet Rods
CDC-209	Low Burnup Effects, GEP Pellet Rods

1. TITLE: Variable Burnup Effects, SPX Pellet Rods2. OBJECTIVES:

The objectives of these tests are to determine, (1) the response of fuel rods with prior burnup extending from 2000 MWd/tonne to 33,000 MWd/tonne to transient energy depositions extending from 150 cal/g UO_2 to about 600 cal/g UO_2 , (2) base point data for comparison with fuel rods of differing size and other characteristics, (3) the extent of metal-water reaction, (4) thermal-to-mechanical energy conversion, and (5) the type and extent of fission product releases as a function of burnup and energy release.

3. EXPERIMENT:a. Test Fuel

Type - Five-inch active length, 10% enriched SPX rods clad with cold-worked Zr-2. (See Table III for details.)

These fuel rods will be irradiated in the ETR to various burnups up to 33,000 MWd/tonne. Irradiation will take place in a loop facility with operating conditions representative of levels found in proposed BWR plants.

Burnup histories of the rods are summarized as follows:

Burnup -----	variable up to $\approx 33,000$
	MWd/tonne
nvt. (thermal)-----	$< 2.3 \times 10^{21}$
nvt. (fast)-----	$< 3.5 \times 10^{18}$
Clad Temperature-----	550°F
Power (max)-----	< 16 kW/foot
Heat Flux-----	$< 10^6$ Btu/hr-ft ²
Coolant-----	Water
Loop Pressure-----	1000 psi
Decay Time-----	< 30 days

b. Test Conditions

Following irradiation at the ETR to the preselected burnup level, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches will contain air at atmospheric pressure. The capsule, its contents, and instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and

post-test measurements and/or inspection; during each test phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Post-irradiation and Pre-test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) radial growth
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy deposition

(4) Post-test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation

d. Tests

This series will investigate the effects of both burnup and energy level with tests at the following conditions:

ENERGY RELEASE (cal/g)	Burnup (MWd/tonne)				
	≈ 2000	≈ 8000	≈ 15,000	≈ 28,000	≈ 33,000
200	1	1	1	1	1
275	1	1	1	1	1
350	1	1	1	1	1
425	1	1	1	1	1
500		1	1	1	1

4. DISCUSSION:

The tests in this series will provide experimental data on the performance of representative nuclear fuels to transient energy generation as a function of fuel burnup. By including burnup among the initial conditions of these tests, results are expected to be of immediate value to the industry in accident evaluation. The data from these tests, together with that from tests on similar unirradiated fuel rods, eg, CDC-106, will allow an evaluation to be made at several burnup levels of any changes in performance that result from irradiation. Of major importance in the evaluation of this data will be determination of any change in the nature of or lowering of the thresholds for fuel failure and/or an increase in the failure consequences as a function of prior fuel burnup. The amount and type of fission products released to the water and air plenum in the capsule will prove useful in analysis of the radiological hazards associated with a reactivity accident.

The data from these tests will also be utilized for comparison with data from tests on different rod types and fuel rods irradiated under differing conditions, eg, CDC-202 and CDC-203. These comparisons will provide data on the effects of irradiation and fuel rod characteristics on transient performance which are not obtainable from this series of tests.

5. STATUS:

Irradiation of SPX rods for this test series is scheduled to begin in the third quarter of FY 1970. Tests are scheduled in the CDC through the first quarter of FY 1972.

1. TITLE: End-of-Core-Life Effects, GEX Pellet Rods2. OBJECTIVES:

The objectives of this test series are to determine; (1) the response of scaled BWR-type fuel fabricated to commercial standards and irradiated to 20,000 MWd/tonne burnup when subjected to transient energy releases of 150 to 400 cal/g of UO_2 ; (2) energy thresholds at which damage to the rod occurs; (3) the extent of metal-water reactions; (4) thermal-to-mechanical energy conversion; and (5) the type and extent of fission product release.

3. EXPERIMENT:a. Test Fuel

Type - GEX Pellet (See Table III for details). These fuel rods will be irradiated in the ETR to about 20,000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

Burnup history of the rods is summarized as follows:

Burnup -----	~ 20,000 MWd/tonne
nvt (thermal) -----	~ 10^{21}
nvt (fast) -----	~ 2×10^{18}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 15 kW/foot
Heat Flux (max)-----	~ 640,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of ~ 20,000 MWd/tonne, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and at ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection. During each test, several reactions of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack length
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (f) power

(4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test examination will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack length
- (k) weight

d. Tests

The series is planned as the four tests listed below:

<u>Test No.</u>	<u>Energy Release (Cal/gm of UO₂)</u>
1	200
2	260
3	340
4	400 or 150

4. DISCUSSION:

This test series uses fuel rods obtained in a cooperative program with General Electric which enables the CDC program to include in the irradiated fuel test program fuel samples fabricated to commercial standards. Test results will be of immediate value to BWR accident analysis for both vendors and the AEC. The fuel group obtained from General Electric includes four types of fuel: conventional size (0.562 inch diameter) rods with either pellets or powder fuel, and scaled, smaller rods (0.3125 inch diameter) with either pellets or powder fuel. The larger rods are tested in the CDC-205 series of tests and represent the closest approximation of the CDC tests to actual BWR fuel. The scaled rods used in this series of tests are identical in fabrication techniques to the large rods but allow higher test enthalpies to be achieved. The results of this test series together with those of CDC-203 will allow comparison of pellet fuel with powder fuel in the 0.3125 inch diameter and also allow a comparison to be made of the relative response characteristics of large versus small diameter rods at one energy level. It is important that this final comparison be made since the CDC must use scaled rods together with relatively high enrichments to obtain enthalpy levels of interest above 300 cal/gram. The Irradiated Fuel Tests also include 0.250 inch-diameter clad fuel of pellet type in tests series CDC-201 and these tests will provide additional information regarding the effect of fuel rod diameter as well as accident responses.

5. STATUS:

Irradiation of fuel rods for this test series was initiated in May, 1969, and is expected to be completed in January, 1970, at which time CDC testing will commence.

1. TITLE: End-of-Core-Life Effects, GEX Powder Rods2. OBJECTIVES:

The objectives of these tests are to determine; (1) the response of scaled BWR-type fuels with an alternate fuel type (powder) that have been irradiated to 20,000 MWd/tonne and subsequently subjected to transient energy releases of 200 to 400 cal/g of UO_2 ; (2) energy thresholds at which damage to the rod occurs; (3) the extent of metal-water reactions; (4) thermal-to-mechanical energy conversion; and (5) the type and extent of fission product release.

3. EXPERIMENT:a. Test Fuel

Type - GEX Powder (See Table III for details). These fuel rods will be irradiated in the ETR to about 20,000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

Burnup history of the rods is summarized as follows:

Burnup -----	~ 20,000 MWd/tonne
nvt. (thermal) -----	~ 10^{21}
nvt. (fast) -----	~ 2×10^{18}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 15 kW/ft
Heat Flux (max) -----	~ 640,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of 20,000 MWd/tonne, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena

of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (f) power

(4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test examination will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests

The series is planned as the three tests listed below:

Test No.	Energy Release (Cal/gm of UO ₂)
1	200
2	260
3	340

4. DISCUSSION:

This test series, CDC-203, uses fuel rods obtained in a cooperative program with General Electric which was set up to enable the CDC program to include in the irradiated fuel test program fuel samples fabricated to the same commercial standards as used in power producing plants. In the case of the rods used in this series, the powder fuel fabrication procedures represent those to be used if powder fuel becomes acceptable for forthcoming BWR's. Test series CDC-203 will obtain direct information about possible reactions of powdered BWR fuel to conditions expected in reactivity accidents. Should the results of these tests appear favorable, the eventual use of powder fuel with its attendant savings in fabrication costs may be hastened, and thus the test results will be of value initially to the BWR vendors as well as the AEC. The fuel group obtained from General Electric includes four types of fuel: conventional size (0.562 inch diameter) rods with either pellets or powder fuel and scaled, smaller rods (0.3125 inch diameter) with either pellets or powder fuel. The larger rods are tested in the CDC-205 series of tests and represent the closest approximation of the CDC tests to actual BWR fuel. The scaled rods used in this series of tests are identical in fabrication techniques to the large rods but allow higher test enthalpies to be achieved. The results of this test series together with those of CDC-202 will allow comparison of pellet fuel with powder fuel in the 0.3125 inch diameter and also allow a comparison to be made of the relative response characteristics of large versus small diameter rods at one energy level. It is important that this final comparison be made since the CDC must use scaled rods together with relatively high enrichments to obtain enthalpy levels of interest above 300 cal/gram. The Irradiated Fuel tests also include 0.250 inch-diameter clad fuel of pellet type in test series CDC-201 and these tests will also provide additional information regarding the effect of fuel rod diameter as well as accident responses.

5. STATUS:

Irradiation of fuel rods for this test series is scheduled to begin in the third quarter of FY 1970 and be completed in the second quarter of FY 1971.

1. TITLE: Thermal and Neutron Sintering Effects, GEX Powder Rods

2. OBJECTIVES:

The objective of these tests will be to determine the response of powder fuel that has been thermal- and neutron-sintered when subjected to transient energy depositions up to about 350 cal/g of UO_2 .

3. EXPERIMENT:

a. Test Fuel

Type GEX powder (See Table III for details). These fuel rods will be irradiated in the ETR to achieve about 2000 MWd/tonne burnup - a level sufficient to establish agglomeration of fuel due to irradiation. Operating temperature of the fuel will be above 3000°F and is expected to achieve thermal sintering. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

The burnup history of the rods will be as follows:

Burnup-----	~ 2000 MWd/tonne
nvt (thermal)-----	~ 1×10^{20}
nvt (fast)-----	~ 2×10^{17}
Clad Temperature-----	~ 550°F
Power (max)-----	~ 15 kW/foot
Heat Flux (max)-----	~ 640,000 Btu/hr-ft ²
Coolant-----	Water
Loop Pressure-----	1000 psig
Decay Time-----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of ~2000 MWd/tonne, the rods will be transported to the TRA Hot Shop where they will be positioned in Type B capsules filled to within ten inches of the top with ambient temperature, demineralized water. The top ten inches of the capsules will contain air at atmospheric pressure. The capsules, contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. The tests will be initiated from conditions of ambient temperature and atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, the fuel rods will undergo extensive pre- and

post-test measurements and/or inspection. During the tests, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power

(4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test samples has occurred. The post-test measurements will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests

The series consists of the three tests listed below:

<u>Test No.</u>	<u>Energy Release (cal/g of UO₂)</u>
1	200
2	260
3	340

4. DISCUSSION:

A deterrent to the use of powder fuel in power reactors is the opinion that in the event of an accident powder fuel will disperse in the coolant to a much greater degree than pelletized fuel. If this is true, it may have the possible consequence of more rapid and more complete heat transfer to the coolant leading to pressures or explosions which are more energetic and damaging. Counter to this argument is another which asserts that shortly after service is begun with powder fuel, agglomeration of the fuel particles takes place due both to the thermal sintering effect and to fissioning itself. Thus, powder fuel may soon assume a physical condition similar to pelleted fuel. Differences in behavior between pelleted and powder fuel during a reactivity accident would thus be minimal.

The present test, CDC-204 will determine if short-term thermal and fission sintering obtained under conditions typical of an operating BWR produce significantly altered accident behavior as compared to fresh, unirradiated powder fuel (Series CDC-104 and 108). Results will also be compared with the results of Test Series CDC-203 which fuel has a full 20,000 MWd/tonne burnup, and should have value in accident analysis for both the AEC and BWR vendors should powder fuel be utilized.

5. STATUS:

Irradiation of fuel rods for this test series is expected to begin and end during the third quarter of FY 1970.

1. TITLE: End-of-Core-Life Effects, GEP Pellet Rods2. OBJECTIVES:

The objectives of these tests are to determine: (1) the effect of high burnup on the performance of standard diameter BWR oxide fuel rods, containing pelletized fuel, when subjected to transient energy depositions of about 275 cal/g of UO_2 ; (2) the response of the standard diameter rods for comparison with the response of the scaled rods; (3) metal-water reaction; (4) thermal-to-mechanical energy conversion; and (5) the relative fission product release between pelletized and powder fuel.

3. EXPERIMENT:a. Test Fuel

Type - GEP-pellet (see Table III for details).

These fuel rods will be irradiated in the ETR to either about 10,000 or to about 20,000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

The burnup history of the rods is:

Burnup-----	~ 10,000 or ~20,000 MWd/tonne
nvt (thermal)-----	~ 5×10^{20} to ~ 10^{21}
nvt (fast)-----	~ 10^{18} to ~ 2×10^{18}
Clad Temperature-----	~ 550°F
Power (max)-----	~ 16 kW/foot
Heat Flux (max)-----	~ 400,000 Btu/hr-ft ²
Coolant-----	Water
Loop Pressure-----	1000 psig
Decay Time-----	< 30 days

b. Test Conditions

Following irradiation at the ETR, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power

(4) Post-test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test inspections will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests:

The series is planned as the two tests listed below:

<u>Test No.</u>	<u>Energy Release (cal/g of UO₂)</u>
1	275
2	275

4. DISCUSSION:

This series of tests, CDC-205, will provide data on the relative performance of standard sized BWR fuel rods fueled with conventional pelletized fuel. The use of the standard sized rods in this series will provide experimental information that may be compared with that from series CDC-202, to establish the effects of rod diameter. Comparison of these data with those obtained in series CDC-105 will also allow experimental verification of irradiation effects on fuel performance. The burnup level will be established at either 10,000 MWd/tonne or 20,000 MWd/tonne depending upon results found in test series CDC-202 and -206.

Results of these tests will be of immediate value to BWR vendors and the AEC for accident analyses.

5. STATUS:

Irradiation of fuel rods for this test series began in May, 1969, and will reach completion of $\approx 20,000$ MWd/tonne in the first quarter of FY 1972.

1. TITLE: Intermediate Burnup Effects, GEX-Pellet Rods2. OBJECTIVES:

The objectives of this test series are to determine; (1) the response of scaled BWR-type fuel fabricated to commercial standards and irradiated to 10,000 MWd/tonne burnup when subjected to transient energy releases of 200 to 400 cal/g of UO_2 ; (2) energy thresholds at which damage to the rod occurs; (3) the extent of metal-water reactions; (4) thermal-to-mechanical energy conversion; and (5) the type and extent of fission product release.

3. EXPERIMENT:a. Test Fuel

Type - GEX Pellet (See Table III for details). These fuel rods will be irradiated in the ETR to about 10,000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

Burnup history of the rods is summarized as follows:

Burnup -----	~ 10,000 MWd/tonne
nvt (thermal) -----	~ 5×10^{20}
nvt (fast) -----	~ 1×10^{18}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 15 kW/foot
Heat Flux (max) -----	~ 640,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of ~ 10,000 MWd/tonne the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and at ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection. During each test, several reactions of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power

(4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test examination will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests

The series is planned as the four tests listed below:

<u>Test Number</u>	<u>Energy Release (cal/g of UO_2)</u>
1	200
2	260
3	340
4	400 or 150

4. DISCUSSION:

This test series uses fuel rods obtained in a cooperative program with General Electric which enables the CDC program to include in the irradiated fuel test program fuel samples fabricated to commercial standards. Test results will be of immediate value to BWR accident analysis for both vendors and the AEC. The fuel group obtained from General Electric includes four types of fuel: conventional size (0.562 inch diameter) rods with either pellet or powder fuel, and scaled, smaller rods (0.3125 inch diameter) with either pellet or powder fuel. The larger rods are tested in the CDC-205 and CDC-207 series of tests and represent the closest approximation of the CDC tests to actual BWR fuel. The scaled rods used in this series of tests are identical in fabrication techniques to the large rods; but allow higher test energies to be achieved. Results from tests in this series on fuel burned to 10,000 MWd/tonne will be compared with those of CDC-202 on fuel burned to 20,000 MWd/tonne. This comparison will be important in determining whether the larger rods of the GEP type in series CDC-205 need be irradiated to 20,000 MWd/tonne, or whether the burnup of the GEP rods can be terminated at 10,000 MWd/tonne for earlier testing.

5. STATUS:

Irradiation of two of the four fuel rods for this test series began in May, 1969, and will be completed in September, 1969. At this time the last two rods will begin irradiation and be completed in the third quarter of FY 1970.

1. TITLE: Intermediate Burnup Effects, GEP Powder Rods2. OBJECTIVES:

The objectives of these tests are to determine; (1) the effect of high burnup on the performance of standard diameter BWR oxide fuel rods containing powder fuel, when subjected to transient energy depositions of about 275 cal/g of UO_2 ; (2) data on the response of the standard diameter rods for comparison with data from scaled rods; (3) metal-water reaction; (4) thermal-to-mechanical energy conversion; and (5) the relative fission product release between pelletized and powder fuel.

3. EXPERIMENT:a. Test Fuel

Type - GEP-powder (see Table III for details).

These fuel rods will be irradiated in the ETR to about 10,000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

The burnup history of the rods is:

Burnup -----	~ 10,000 MWd/tonne
nvt (thermal) -----	~ 5×10^{20}
nvt (fast) -----	~ 1×10^{18}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 16 kW/ft
Heat Flux (max) -----	~ 400,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of ~ 10,000 MWd/tonne, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. The CDC tests will take place within 30 days after removal of the fuel rods from the ETR. At the time of the test, the capsule and contents will be at atmospheric pressure and ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel column height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel column growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power

(4) Post-test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test inspections will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel column height
- (k) weight

d. Tests

The series is planned as the two tests listed below:

<u>Test Number</u>	<u>Energy Release (cal/g of UO₂)</u>
1	275
2	275

4. DISCUSSION:

This series of tests, CDC-207, will provide data for comparison of the performance of standard size BWR fuel rods fueled with conventional pelletized fuel and the alternate powder fuel. The use of the standard sized rods in this series will also provide experimental information that may be compared with that from series CDC-203 to establish the effects of rod diameter. Comparison of these data with those obtained in series CDC-105 will also allow experimental verification of irradiation effects on fuel performance. Performance of this test series will take place only if series CDC-205 uses rods burned to 10,000 MWd/tonne.

Results of these tests will be of immediate value to BWR vendors and the AEC for accident analyses.

5. STATUS:

If it is decided to conduct this test series, irradiation of the two fuel rods needed will begin in the first quarter of FY 1971 and be completed in the first quarter of FY 1972.

1. TITLE: Low Burnup Effects, GEX-Pellet Rods

2. OBJECTIVES:

The objectives of this test series are to determine; (1) the response of scaled BWR-type fuel fabricated to commercial standards and irradiated to about 3000 MWd/tonne burnup when subjected to transient energy releases in the range of 150 to 400 cal/g of UO_2 , (2) energy thresholds at which damage to the rod occurs; (3) the extent of metal-water reactions; (4) thermal-to-mechanical energy conversion; and (5) the type and extent of fission product release.

3. EXPERIMENT:

a. Test Fuel

Type - GEX Pellet (See Table III for details). These fuel rods will be irradiated in the ETR to about 3000 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

Burnup history of the rods is summarized as follows:

Burnup -----	~ 3000 MWd/tonne
nvt (thermal) -----	~ 1.5×10^{19}
nvt (fast) -----	~ 3×10^{17}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 15 kW/ft
Heat Flux (max) -----	~ 640,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR to a burnup level of ~ 3000 MWd/tonne the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. At the time of the test, the capsule and contents will be at atmospheric pressure and at ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection. During each test, several reactions of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

- (1) Pre-Irradiation Examinations
 - (a) radiographs
 - (b) photographs
 - (c) axial length
 - (d) diameter
 - (e) weight
- (2) Post-Irradiation and Pre-Test Examinations
 - (a) radiographs
 - (b) photographs
 - (c) axial length
 - (d) diameter
 - (e) internal pressure
 - (f) fuel stack height
 - (g) weight
- (3) Dynamic Measurements
 - (a) axial fuel rod growth
 - (b) axial fuel stack growth
 - (c) radial fuel rod growth
 - (d) internal fuel rod pressure
 - (e) capsule pressure
 - (f) capsule wall stain
 - (g) water column velocity
 - (h) power
- (4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test examination will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests

The series is planned as the four tests listed below:

<u>Test Number</u>	<u>Energy Release (cal/g of UO₂)</u>
1	150
2	200
3	260
4	340

4. DISCUSSION:

This test series uses fuel rods obtained in a cooperative program with General Electric which enables the CDC program to include in the irradiated fuel test program fuel samples fabricated to commercial standards. Test results will be of immediate value to BWR accident analysis for both vendors and the AEC. The fuel group obtained from General Electric includes four types of fuel: conventional size (0.562 inch diameter) rods with either pellet or powder fuel, and scaled, smaller rods (0.3125 inch diameter) with either pellet or powder fuel. The larger rods are tested in the CDC-205 and CDC-207 series of tests and represent the closest approximation of the CDC tests to actual BWR fuel. The scaled rods used in this series of tests are identical in fabrication techniques to the large rods; but allow higher test energies to be achieved. Results from tests in this series on fuel burned to 3000 MWd/tonne will be compared with those of CDC-206 on fuel burned to ~ 10,000 MWd/tonne and to CDC-202 with fuel burned to 20,000 MWd/tonne. These comparisons will be important in determining whether the larger rods of the GEP type in series CDC-205 need be irradiated to >20,000 MWd/tonne, or whether the burnup of the GEP rods can be terminated at 10,000 MWd/tonne for earlier testing.

5. STATUS:

Irradiation of the four fuel rods for this test series was completed in December, 1968. Testing of the fuel rods in the CDC was completed in July, 1969.

1. TITLE: Low Burnup Effects, GEP Pellet Rods2. OBJECTIVES:

The objective of these tests is to determine: (1) the effects of low burnup on the performance of standard diameter BWR oxide fuel rods, containing pelletized fuel, when subjected to transient energy depositions between 200 and 275 cal/g of UO_2 ; (2) the response of the standard diameter rods for comparison with the response of the scaled rods; (3) metal-water reaction; and (4) thermal-to-mechanical energy conversion.

3. EXPERIMENT:a. Test Fuel

Type - GEP-pellet (see Table III for details).

These fuel rods will be irradiated in the ETR to about 1200 MWd/tonne burnup. Irradiation will take place in a loop facility with operating conditions representative of the highest levels found in proposed BWR plants.

The burnup history of the rods is:

Burnup -----	~ 1200 MWd/tonne
nvt (thermal) -----	~ 6×10^{19}
nvt (fast) -----	~ 10^{16}
Clad Temperature -----	~ 550°F
Power (max) -----	~ 16 kW/ft
Heat Flux (max) -----	~ 400,000 Btu/hr-ft ²
Coolant -----	Water
Loop Pressure -----	1000 psig
Decay Time -----	< 30 days

b. Test Conditions

Following irradiation at the ETR, the rods will be transported to the TRA Hot Shop where each rod will be positioned in a Type B capsule filled to within ten inches of the top with demineralized water. The top ten inches of the capsule will contain air at atmospheric pressure. The capsule, its contents, and attached instrumentation will then be transported to the CDC for testing. At the time of the test, the capsule and contents will be at atmospheric pressure and ambient temperature.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Irradiation Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) weight

(2) Post-Irradiation and Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) internal pressure
- (f) fuel stack height
- (g) weight

(3) Dynamic Measurements

- (a) axial fuel rod growth
- (b) axial fuel stack growth
- (c) radial fuel rod growth
- (d) internal fuel rod pressure
- (e) capsule pressure
- (f) capsule wall strain
- (g) water column velocity
- (h) power

(4) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test inspections will include those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) fuel stack height
- (k) weight

d. Tests

The series is planned as the two tests listed below:

<u>Test Number</u>	<u>Energy Release (cal/g of UO₂)</u>
1	200
2	270

4. DISCUSSION:

This series of tests, CDC-205, will provide data on the relative performance of standard sized BWR fuel rods fueled with conventional pelletized fuel. Comparison of these data with those obtained in series CDC-105 will also allow experimental verification of irradiation effects on fuel performance.

Results of these tests will be of immediate value to BWR vendors and the AEC for accident analyses.

5. STATUS:

Irradiation of the fuel rods for this test series was completed in December, 1968. Testing of the two fuel rods in the CDC is scheduled during the first quarter of FY 1970.

C. PLUTONIUM RECYCLE FUELS TESTING

The use of plutonium recycled fuel in the form of mixed oxides is now feasible in boiling and pressurized water power reactors. Current indications are that the mixed oxide fuel will probably be fabricated in the form of pellets (similar to UO_2 fabrication technology) in the immediate future. Powder fuel technology, which may be significantly more economic in the case of mixed oxides, is not considered to be sufficiently well developed to warrant CDC tests at this time.

When using plutonium in thermal recycle fuels, the potential exists for heterogeneities in the fissile content not normally present in uranium fuels. Mixed oxide fuel in which the fissile material (PuO_2) is physically blended with the fertile material (natural or depleted UO_2) will have a certain degree of heterogeneity, depending upon the plutonium particle size and the details of the fabrication process. When fuel of this type undergoes a fast transient, the plutonium particles will attain relatively high temperatures compared to the average oxide temperature. From the reactor safety viewpoint, there is some concern that this effect could lower the energy density failure threshold and increase the thermal-to-mechanical energy conversion efficiency compared with that associated with conventional UO_2 fuel. Scoping tests in the CDC will be performed to determine if mixed oxide fuel fabricated to current technology standards exhibits any significant differences in response compared with UO_2 fuel of similar design.

<u>Series</u>	<u>Title</u>
CDC-301	Scoping Series - Plutonium Recycle Fuels

1. TITLE: Scoping Series - Plutonium Recycle Fuels

2. OBJECTIVES:

The primary objective of this series is to determine the differences in failure thresholds, modes and mechanisms of failure, and failure consequences that may exist between mixed oxide fuel samples representative of plutonium recycle fuel proposed for use in boiling and pressurized water power reactors and the conventional UO_2 fuel currently used. A secondary objective is to determine the reactivity effect in the fuel brought about by a nondestructive test slightly below the failure threshold.

3. EXPERIMENT:

a. Test Fuel

GEXPR fuel rods (see Table III for details). The fuel is comprised of a physical blend of PuO_2 with natural UO_2 in the form of pellets. The PuO_2 has an average particle size of 40 to 50 microns. The cladding and design dimensions are similar to the GEX rods.

b. Test Conditions

Single rods will be positioned in the flux peak in a Type B capsule filled to within ten inches of the top with ambient temperature demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter
- (e) reactivity worth

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) radial growth
- (c) capsule pressure
- (d) capsule wall strain
- (e) water column velocity
- (f) power
- (g) energy deposition
- (h) internal fuel rod pressure
- (i) fuel clad temperature

(3) Post-Test Measurements and Examinations

The extent and nature of the post-test work will depend on whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in air space
- (g) fission product analysis of water
- (h) fission product analysis of air sample
- (i) metallographic investigation
- (j) reactivity worth

d. Tests

- (1) Pellet fuel rod @ 250 cal/g of fuel
- (2) Pellet fuel rod @ 200 cal/g of fuel
- (3) Pellet fuel rod @ 275 cal/g of fuel
- (4) Pellet fuel rod @ 325 cal/g of fuel
- (5) Pellet fuel rod @ 375 cal/g of fuel
- (6) Pellet fuel rod @ 425 cal/g of fuel
- (7) Pellet fuel rod @ 500 cal/g of fuel

4. DISCUSSION:

A total of seven mixed oxide fuel rod samples are available for testing. Except for fuel composition, the samples are identical in design to the UO₂ GEX pellet fuel rods used in series CDC-104 and 107, thus allowing for a direct comparison of experimental data. The information obtained from these tests will provide a basis for both industry and AEC to more objectively evaluate the expected behavior of mixed oxide fuel under accident conditions. The results will also provide program guidance with regard to the need for further subassembly testing of mixed oxide fuels.

5. STATUS:

Tests in this series were initiated in May, 1969, and will be performed periodically over a period of several months.

D. LMFBR Fuel Testing

The design of safe, economic and reliable Liquid Metal Fast Breeder Reactors requires consideration of fuel behavior under a variety of design and off-design conditions. Confidence in present predictions of this behavior is limited due to a lack of experimental data. There are no available test facilities which provide for fast-flux testing either at design conditions or off-design conditions. Studies of effects such as gaseous fission product release and damage thresholds in fast fuels can be attempted in thermal reactor test facilities but the difference in radial energy distribution within a fuel pin makes extrapolation of thermal data to the actual fast-flux case difficult.

There is an area, however, in which a fast neutron flux is not as essential to the testing of fast fuel behavior. This is in testing the energy exchange from heated fuel to the surrounding sodium coolant when a fuel rod fails. The amount and form of mechanical energy produced in the sodium for a given energy deposition in the fuel must be known, if valid analytical models are to be constructed. This can only be tested, for the case of a power transient, by expelling heated fuel particles into sodium. Electric resistance heating or oven heating of fuel and subsequent insertion into sodium present problems of heat leakage and time lapse and make particle size and velocity simulation arbitrary variables.

On the other hand, a thermal transient test reactor such as CDC can provide such fuel heating at a high rate, directly in the sodium environment, and in the actual geometry involved. For this reason, the test series listed below are expected to provide useful insight into this energy exchange and allow the fitting of analytical methods to actual experimental data. A brief summary of these test series is given in the following pages.

<u>Series</u>	<u>Title</u>
CDC-501	Simulated Sodium Capsule Figure-of-Merit Tests
CDC-502	Fast Reactor Fuel Transient Scoping Tests

1. TITLE: Simulated Sodium Capsule Figure-of-Merit Tests

2. OBJECTIVES:

The objective of these tests for figure-of-merit is to obtain data to which predictive calculational methods may be fitted. This will then provide an analytic tool to accurately describe energy deposition for the fast reactor fuel test program to follow.

3. EXPERIMENT:

a. Test Assembly

The assembly for this test will simulate the design of a test capsule to be used in transient tests of fast reactor fuel pins in a stagnant sodium environment. The schematic design is shown in Figure 1 as a double-walled container. The main difference between the mock-up and the working capsule to be used in transient tests later is in the substitution of aluminum or magnesium for sodium in the annulus immediately surrounding the fuel pin. This substitution can be corrected for in the calculational checks, and will eliminate sodium handling problems during this preliminary phase. These tests will not involve high pressures, but wall thicknesses will be similar to the high-pressure design. Pressure relief for the outer annulus will be provided, and adequate arrangement made to duct the effluent safely away.

A heating element will be wound on the outside of the inner capsule in the same way as for the actual test capsule. This will allow temperature levels in the sample to be varied and controlled.

The fuel pins will be representative of those to be used during the actual testing program. They will, therefore, be UO_2 , PuO_2 - UO_2 mixed, and PuC - UC types.

SCHEMATIC OF
TEST CAPSULE

Figure 1.

b. Test Conditions

The assembly will be placed in the experiment tube in CDC and irradiated at low power for a sufficient time to develop adequate fission products for analysis in the test sample. Water in the experiment tube will be free to convect to remove leakage heat during elevated temperature runs.

c. Measurements

Integrated power in the driver core will be measured with flux wires. Integrated power in the test sample will be measured by radiochemical analysis of the fuel element after the test. If available, neutron energy spectrum detectors will be used to measure energy ranges. Temperature history will be recorded for the fuel-cladding-aluminum system and for at least two radial locations in the $ZrH_{1.8}$. The temperature of the outer capsule must be monitored during the test to assure that no boiling occurs in the water during a test. Pressure in the outer annulus must be monitored.

d. Tests

A minimum of seven cases will be run. The enrichment of the fissionable compound in the fuel will be varied to identify variations due to this parameter. As mentioned above, PuO_2-UO_2 and $PuC-UC$ fuel will be compared in one case with UO_2 fuel. And also, for one fuel type, the sample temperature will be varied. This results in the following test series:

<u>Fuel Material</u>	<u>Percent Enrichment</u>	<u>Sample Temperature</u>
UO_2	5%	ambient
UO_2	10%	ambient
UO_2	10%	300°F
UO_2	10%	500°F
UO_2	10%	700°F
UO_2	25%	ambient
PuO_2-UO_2	10%	ambient
$PuC-UC$	10%	ambient

4. DISCUSSION

Prior to a program of transient tests on fast reactor fuels in a sodium environment, prediction of energy deposition must be made. This will allow safety analysis on the containment capsule and valid explosive proof-testing. Instrumentation planning also requires such information.

The present calculational methods for test samples in CDC are found to predict figure-of-merit with good accuracy for an established geometry and fuel type. But it has been found that adjustments in the calculational method are required when fuel type or geometry is varied significantly. These are empirically established by controlled, steady state tests which are calibrated by radiochemical analysis.

Since the presently planned fast fuel testing in CDC involves not only a new design of capsule, but different materials and test temperatures, it is necessary to establish an analytical method which can predict figures-of-merit with confidence.

The behavior of expelled fuel in the event of pin failure can be expected to vary with the temperature of the surrounding sodium. Since fast reactors will operate with Na at elevated temperatures, transient fast reactor fuel tests should include a study of this effect. Figure-of-merit calculations must, therefore, take sample temperatures into account.

5. STATUS:

Design of a capsule for use in this test series is underway. Tests are expected to begin in the fourth quarter of FY 1970.

1. TITLE: Fast Reactor Fuel Transient Scoping Tests
2. OBJECTIVES:

The general objective of this fast reactor fuel transient program is the study of energy transfer from hot fuel to the surrounding sodium coolant in the event of fuel failure during a power excursion. This will be done on a scoping basis for this first series to establish the effects, indicated in 3c., below, both qualitatively and to some degree quantitatively.

3. EXPERIMENT:

- a. Test Assembly and Fuel

Testing in this series will be of single fuel elements in the form of pelletized, stainless steel clad rods. This series will use 25% enriched UO_2 fuel of the 1/4" diameter now typical for fast fuel design.

The sample will be contained in a sodium-filled stainless steel capsule which is proof-tested against the maximum pressures possible based on energy conversion (from nuclear to mechanical) during a burst. To maintain the sodium at required temperatures, an electrical heating element will be wound around the outside of this capsule. As an added precaution against sodium escaping and contacting the water in the core, a second capsule will be placed around the first. The space between the two capsule walls will contain a moderator ($\text{ZrH}_{1.8}$) and an inert gas. Pressure relief for the outer annulus will be provided, and adequate arrangement made to duct any effluent away. A magnet-carrying plunger at the sodium surface will be used to measure sodium motion as is done at present for water-filled thermal fuel capsules.

SCHEMATIC OF
TEST CAPSULE

Figure 2.

b. Test Conditions

The assembly will be placed in the experiment tube in CDC for transient testing as is presently done for the water reactor fuel experiments. The sodium will be held at elevated temperatures by the heater until just before the transient. Therefore, the core water surrounding the outer capsule must be free to convect to remove the leakage heat.

c. Measurements

To obtain the information necessary to satisfy the objectives of this test series, each fuel rod will undergo extensive pre- and post-test measurements and/or inspection; during each test, phenomena of importance will be measured as a function of time (referred to as dynamic measurements). The examinations and measurements include:

(1) Pre-Test Examinations

- (a) radiographs
- (b) photographs
- (c) axial length
- (d) diameter

(2) Dynamic Measurements

- (a) axial fuel rod growth
- (b) fuel motion
- (c) cladding surface temperature
- (d) internal fuel rod pressure
- (e) inner capsule pressure
- (f) outer annulus pressure
- (g) sodium displacement
- (h) sodium temperature
- (i) outer capsule wall temperature
- (j) neutron energy spectrum
- (k) power
- (l) energy deposition

(3) Post-Test Measurements and Examination

The extent and nature of the post-test work will depend upon whether failure of the test sample has occurred. The post-test work will include only those of the following which are appropriate:

- (a) residual particle sizing
- (b) radiographs
- (c) photographs
- (d) axial length
- (e) diameter
- (f) hydrogen concentration in outer annulus
- (g) fission product analysis of sodium
- (h) fission product analysis of gas space
- (i) metallographic investigation

d. Tests

For the first in this series of tests, 25% enriched UO_2 pelletized fuel samples will be used. A single sample will be subjected to a transient producing 150 calories per gram, and then visually inspected. The next transients will be successively increased in energy deposition in 50 calorie per gram increments. This process will continue until damage thresholds are identified. The decision to use the same sample again at a higher energy deposition will depend on capsule-handling turnaround time and sample supply as well as condition of the rod.

Further refinement in energy increment will be done if it appears useful to explore the nature of an effects-threshold around the energy deposition level at which it was observed in the first set.

Since this series is a scoping test, the specific direction which experiments will take is not predictable. However, a typical series of tests is postulated below in order to present information for planning and scheduling.

	<u>Test Number</u>	<u>Sample Enthalpy</u>	<u>Initial Sample Temperature</u>
A	1	150 cal/g	300°F
	2	200 cal/g	300°F
	3	250 cal/g	300°F
	4	300 cal/g	300°F
B	5	240 cal/g	300°F
	6	260 cal/g	300°F
	7	280 cal/g	300°F
C	8	240 cal/g	600°F
	9	240 cal/g	800°F
	10	260 cal/g	600°F
	11	260 cal/g	800°F

Set A in this table will be the original scoping set described above. Set B will center around the point of fuel expulsion, for more definition of threshold. Set C will provide

insight into the effect of sodium temperature as fuel is expelled into it, and will, therefore, be done at the energy deposition range where failure occurred for the 300° temperature as defined by Set B*. Approximately ten fuel rod samples will be required for this series.

4. DISCUSSION:

The tests planned in this series are a first step in a program being developed to define the modes and degree of energy exchange between hot fuel expelled from a ruptured rod, and the surrounding sodium. Present efforts to predict such behavior analytically are limited by lack of knowledge based on actual performance. And present out-of-pile experiments cannot simulate well the fuel condition as it emerges from a break caused by a transient.

To investigate mechanical results of fuel failure in sodium, it is not mandatory that the nuclear energy generated in the fuel be produced by a fast flux. It is desirable that the energy be produced by a short-period transient and that flux depression is not excessive. Both of these can be accomplished by sodium capsule tests in the CDC test hole, using enriched UO₂ to begin with. The data thus generated will allow analytical models to be produced to predict such behavior in a wider range of specific enthalpies and designs. Fuel-to-sodium ratio may be varied, as well as initial sodium pressure and temperature and other variables. Various types of fuel materials will be tested as fast reactor fuel design progresses.

While the Spert CDC reactor is thermal, it is capable of producing relatively high energy in a test sample at short periods. It is available now, when no similar fast facilities exist, and, therefore, can be used to provide early information on some aspects of fast fuel behavior. It is expected that similar tests, with the added effect of steady state initial power, will be performed in PBF when it becomes available.

It is to be noted that all details are not presently developed for these tests. More information will be available as hardware design progresses and further studies to predict test results are made.

Information expected from these tests will be of value to industry and to the regulatory effort as they analyse off-design behavior for safety studies.

5. STATUS:

Tests in this series are expected to begin early in FY 1971.

*the 240-280 cal/gm range for Set B and the 240-260 for Set C are only estimates here--and will be determined by results from Set A.

E. OTHER TESTS

Tests and test series with designation numbers above 900 generally have singular purposes unrelated to the specific objectives of other test series, but which are necessary in order to properly carry out or interpret the results of other tests series. Additional tests may be added at any time, but at this writing the following titles are present:

<u>Series</u>	<u>Title</u>
CDC-901	Radiation Sensitivity Tests
CDC-902	Fuel Temperature Measurement Development
CDC-903	Neutron Detector Performance Evaluation
CDC-904	Diamctrical Fuel Rod Growth Detector Development
CDC-905	Incremental Axial Rod Growth Detector Development
CDC-906	Flux Spectrum Detector Development
CDC-907	PBF Waterlog Tests
CDC-908	PBF Fuel Operation Trials
CDC-909	Energy Calibration Tests
CDC-910	Upgrade Driver Core Capability
CDC-911	Urania Equation of State
CDC-912	Scoping Tests of $\text{UO}_2\text{-H}_2\text{O}$ Reaction
CDC-913	Detailed Tests of $\text{UO}_2\text{-H}_2\text{O}$ Reaction
CDC-914	Cladding Temperature Measurement Development

1. TITLE: Radiation Sensitivity Tests

2. OBJECTIVES:

The objective of these tests will be to determine the radiation sensitivity of new transducers which are developed or purchased for use in the various safety programs.

3. EXPERIMENTS:

a. Test Specimen

The test specimen will vary from time to time. Representative types of detectors to be tested will be pressure transducers, strain gages, LVDT's, and displacement transducers.

b. Test Conditions

The test specimens will usually be situated in the experiment tube at the axial flux peak.

c. Measurements

The measurements required to satisfy the objective of the test series are:

1. Depending upon the specific needs, period, power, energy and possibly other variables will be measured. If detectors are available which will provide information on the time-history of the neutron and gamma flux levels at the test specimen, these measurements will also be made.

2. Test specimen output.

d. Tests

To be determined according to specific needs.

4. DISCUSSION:

Radiation sensitivity tests will be performed as necessary to: (1) evaluate the feasibility of using a specific transducer in the CDC, LCD, or PBF test environment, (2) determine the time history of the radiation induced signal of those transducers being used, so that, if necessary, the radiation-induced signal can be subtracted from resultant transducer output.

5. STATUS:

Tests are performed periodically as needed.

1. TITLE: Fuel Temperature Measurement Development

2. OBJECTIVE:

The objective of these tests is to develop and/or evaluate instrumentation techniques to measure the temperature time-history of uranium-oxide fuels during short period, high-energy nuclear excursions.

3. EXPERIMENT:

a. Test Specimens

A total of eight 3-inch long samples are planned to be tested, four each with 4% powdered fuel and 4.8% pelleted fuel. Thermocouples will be embedded in the fuel at the samples geometrical centerline. Two samples with a grounded hot-tip thermocouple junction will be tested, both on a powdered and a pelleted fuel rod. The remainder of the samples will use bare-wire junctions. "Conax" pressure fittings will be seal-welded to the end of the samples and the thermocouple leads will exit through these fittings. Other specimens may be used as developed.

b. Test Conditions

The test samples will be positioned in the type B capsules which are filled to within a few inches of the top with demineralized water. Normally, two test samples will be tested simultaneously.

c. Measurements

The measurements required to satisfy the objectives of the test series are:

1. Period, power, and energy - these parameters will be measured using the standard core instrumentation, and will provide a reference for the amount and rate of energy deposition in the fuel rod.
2. Exterior clad temperature - chromel-alumel thermocouples will be attached to the cladding surface of the test samples.
3. Interior clad temperature - on the pelleted rods, chromel thermocouple (forming a chromel-stainless steel thermocouple junction) will be attached to the clad interior at the vertical centerline.
4. Fuel temperature - this data will be provided by the thermocouples previously described in section 3-a.

d. Tests

One each of the powdered and pelleted fuel rods samples will be used for test sample energy calibration by fission product analysis. Listed below are the test series to be performed on each of the remaining six rods.

Test Number	Period (msec)	Maximum Fuel Temperature (°C)	
		Powder	Pellet
1	4.9	1970	2260
2	4.0	2260	2550
3	4.9	1970	2260
4	3.6	2410	2730
5	4.9	1970	2260
6	3.1	2620	2830

Additional tests will be made as the program develops.

4. DISCUSSION:

The results of this test series will be evaluated on: (1) the basis of repeatability, (2) characteristic shape of the data (viz, the initial temperature rise should be proportional to the nuclear energy input), and (3) comparison of the data to existing analytical models. The clad interior temperature data, in conjunction with the centerline fuel temperature data and exterior clad temperature data, will provide end points for the temperature distribution across the fuel rod.

The primary purpose for measuring fuel temperature is to improve our knowledge of fuel rod behavior and to assist in development of models currently used to determine the fuel temperature from measurements of nuclear energy and surface temperature. The measurement is of sufficient complexity that it would not be used for most of the fuel rods tested in the CDC, but rather, special tests would be performed on each of the different types of fuel rods to evaluate the accuracy of the model being applied to a specific fuel rod.

5. STATUS:

Development will be limited in the foreseeable future due to low priority and limited manpower.

1. TITLE: Neutron Detector Performance Evaluation Test

2. OBJECTIVE:

The objective of this test series is to evaluate instrument capability of dynamic neutron flux measurements especially in the region of 10^{16} nv to 10^{18} nv thermal. Two of the types of measurements to be evaluated include one which will provide an output which is proportional to the average of the neutron flux distribution of a 48-inch length, and a second which will provide a "point-source" measurement.

3. EXPERIMENT:

a. Test Specimens

Test specimens utilizing two different detection methods will be used, specifically semi-rad and prompt self-powered detectors. Similar types of detectors will be purchased from at least three different manufacturers to evaluate the effect of construction techniques.

b. Test Conditions

Two to three of the experimental chambers will be located in the CDC experiment tube at one time and subjected to a series of nuclear excursions with the peak fluxes listed in part "d" below.

c. Measurements

1. Power - nuclear power will be measured using the standard core instrumentation, and this data will be used as a reference for evaluation of experimental detector data.

2. Experimental detector data.

d. Tests

A series of four tests will be performed on each experimental detector. Three of these tests will be at periods selected to provide peak fluxes of approximately 3×10^{14} nv, 3×10^{15} nv, and 3×10^{16} nv, respectively. This will provide detector linearity data over two decades of neutron flux. For the 4th test, the detectors will be wrapped in a neutron absorbing material and tested to provide information on the neutron-to-gamma sensitivity ratio. Six to eight detectors will be tested and normally two detectors will be tested simultaneously, resulting in a total of about fifteen tests.

4. DISCUSSION:

Required in the Safety Programs are in-core flux measurements in the range from 10^{16} to 10^{18} nv. Currently, little information is available on neutron detectors capable of providing usable data in these flux regions. The data from the experimental detectors will be compared to data from a conventional ion chamber located exterior to the core to determine the linearity characteristics of the test detectors. Evaluation of the test results may require further developmental work by the manufacturers resulting in testing of a group of modified detectors. In that case, a second test series will be required 6 to 8 months after the completion of this test series. The second test series would be similar to this series.

5. STATUS:

Tests to meet the first objective will be completed by the middle of FY 1970. Development schedule for a "point-source" measurement is indefinite.

1. TITLE: Diametrical Fuel Rod Growth Detector Development

2. OBJECTIVES:

The objectives of this test series are to evaluate a detector, which utilizes the eddy current principle, for measurement of diametrical fuel rod growth during short-period, high-energy nuclear excursions.

3. EXPERIMENTS:

a. Test Specimen

The experimental detector will be used on two different test specimens: (1) a piece of stainless steel tubing, and (2) a short length of type-F fuel rod.

b. Test Conditions

The experimental detector and associated hardware will be mounted in a type-B capsule. The environmental conditions will be standard, ie, stagnant water at room temperature.

c. Measurements

The measurements required to satisfy the objectives of this test series are:

1. Period, power and energy - these will be measured using the standard core instrumentation and will provide a reference for the amount and rate of energy deposition in the stainless steel rod and/or fuel rod.
2. Growth detector data.
3. Surface temperature of the stainless steel tubing - the gamma heating induced diametrical growth will be analytically determined from the data.

d. Tests

This test series is divided into two parts. The first part will be the measurement of the gamma heating induced growth of a piece of stainless steel tubing, this portion of the test series will consist of a 4.6 msec period test and a 3.2 msec period test. After comparison of the results from these tests, the detector will be mounted on a type-F fuel rod. This portion of the test series will consist of a 4.6, 4.0 and a 3.2 msec period test.

4. DISCUSSION:

The diametrical fuel rod growth data is needed to provide a time-history of stress in the fuel rod cladding up to the time of failure, to evaluate analytical models. The data will provide added insight into the modes or mechanisms of fuel rod failure. Strain gages attached to the cladding cannot be used to provide this data because of the large magnitude (1000°C or greater) transient temperature of the cladding which will either destroy the gage, or make the data invalid due to the transient thermal effects on the strain gage.

5. STATUS:

Development of this device was completed in FY 1969 and reported in Reference 10.

1. TITLE: Incremental Axial Rod Growth Detector Development

2. OBJECTIVES:

Develop and evaluate a detector to measure incremental axial growth of the fuel rod cladding during power excursions.

3. EXPERIMENT:

a. Test Specimen

The coils which comprise the exciting and sensing mechanism of a linear variable differential transformer (LVDT) will be wound on the outside wall of a type-B capsule. The LVDT's will be located approximately three inches apart, over the length of the fuel rod. The LVDT cores will be mounted on the fuel rod. In addition to the external LVDT's, one LVDT will be mounted inside the capsule to provide the total longitudinal growth of the fuel rod.

b. Test Conditions

The fuel rod used will be a SPX-SS clad rod. The environmental conditions will be standard, ie, stagnant water at room temperature.

c. Measurements

The measurements required to satisfy the objectives of this test series are:

1. Period, power and energy - these will be measured using the standard core instrumentation, and will provide a reference for the amount and rate of energy deposition in the fuel rod.
2. LVDT data - the incremental growth data will be totaled and then compared to the internal LVDT data which directly measures the total longitudinal fuel rod growth.

d. Tests

Three tests will be performed in this test series, corresponding to energy densities in the fuel rod of 175, 225, and 275 cal/gm, respectively.

4. DISCUSSION:

Measurement of the total longitudinal growth, both dynamic and after-the-fact, has shown that longitudinal or axial growth is an important parameter which can effect the modes or mechanisms of fuel rod failure. Measurement of the total and incremental longitudinal growth is needed to evaluate analytical models which are being developed. Because of the large magnitude (1000°C or greater) transient temperature variations, this measurement cannot be performed by attaching strain gages to the cladding surface.

5. STATUS:

Plans for this series are indefinite.

1. TITLE: Flux Spectrum Detector Development

2. OBJECTIVES:

The objectives of this test series are to: (1) evaluate detectors for measurement of the neutron energy spectrum inside a sodium filled capsule, and (2) evaluate detectors for measurement of the neutron energy spectrum in water, which will provide at least two energy groups, specifically above and below 0.625 Mev.

3. EXPERIMENTS:

a. Test Specimens

The detectors currently in the developmental stages, consist of thermocouples attached to small spheres of material, either fissionable or poison, which have resonant absorption peaks at various neutron energy levels.

b. Test Conditions

The experimental detectors will be mounted in a type-B capsule. For the first group of detectors described, the capsule will be filled with a material that has a similar neutron cross-section to sodium. A water-filled capsule will be used to test the second group of detectors.

c. Measurements

The measurements required to satisfy the objectives of this test series are:

1. Period, power and energy - these will be measured using the standard core instrumentation and will provide a reference for comparison of the detector data.
2. Flux Spectrum - detector data.
3. Static flux spectrum data - activation techniques which will provide the integral of the flux spectrum are being investigated.

d. Tests

This series of tests is planned to consist of six 4-msec period tests, three on each group of detectors.

4. DISCUSSION:

Testing of LMFBR fuels in the CDC will require measurement of the neutron energy spectrum to determine energy density in the CDC at which the various types of fast-fuels can be tested, and to relate the energy spectrum in the CDC tests to that in LMFBR's.

The development of such flux spectrum measurement capabilities is important to the post-test analysis of the fast-fuel testing in the CDC. Since fuel cross-sections are energy-dependent, behavior of fuel samples during transient tests will vary with the experimental flux spectrum. Generally, the flux depression at the center of fuel pin is greater at lower neutron energies. This produces higher energy deposition densities at the surface of the fuel than would exist in a fast reactor. Definition of the thermal fluxes experienced in CDC by the fast fuel samples is therefore necessary to allow useful evaluation of test results and their implications for fuel behavior in fast reactors. Also, the Large Core program will require measurement of the flux spectrum since the distribution of the flux spectrum is a required input into the analytical models.

5. STATUS:

Development of these devices is expected to begin in the second quarter of FY 1970.

1. TITLE: PBF Waterlog Tests

2. OBJECTIVES:

It is the objective of these tests to establish the threshold for failure and the consequences of waterlog failure with PBF fuel.

3. EXPERIMENTS:

a. Test Fuel

PBF (see Table I)

b. Test Conditions

One or more normal PBF fuel rods will be subjected to transients in the CDC to effect initial breakup of pellets. The rods will subsequently be waterlogged and tested in the CDC again at various energy releases to establish the objectives. All tests will be conducted in type-B capsules with ambient temperature water and atmospheric pressure.

c. Measurements

Will consist of pre- and post-test inspections including length and diameter measurements, dynamic pressure measurements, possibly water velocity measurement and clad surface temperature, as well as the power.

d. Tests

Actual test energy releases will be determined as the program develops but will progress to the limit of 3-msec periods.

4. DISCUSSION:

It is possible that in the course of PBF operation, one or more rods may become defective and take on water. Since it is known that such waterlogging can produce severe pressures in the rod and cause cladding rupture, it is necessary to establish for these rods the potential thresholds and consequences of this type of accident, particularly to learn if such defects can cause damage to other nearby rods. The tests planned in this series are intended to replicate this condition and will allow a more knowledgeable analysis of the accident in the PBF.

5. STATUS:

Several tests in this series were completed during the first two quarters of FY 1969. Further testing is expected during the first quarter of FY 1970.

1. TITLE: PBF Fuel Operation Trials

2. OBJECTIVES:

The objective of these tests is to verify that final design PBF fuel (Core #1) will have satisfactory performance over the expected lifetime of the fuel and to learn of any unanticipated responses of the fuel to long term repeated usage in the transient mode.

3. EXPERIMENTS:

a. Fuel

PBF (see Table I)

b. Test Conditions

One or more rods will be positioned in a Type B capsule nearly filled with ambient temperature water at atmospheric pressure.

c. Measurements

Dynamic measurements will include power, energy release, and possibly water displacement, clad temperature, and water pressure. Interim inspections will be made to determine if visible changes are occurring with the test rod(s). Also, pre-test, interim-test, and post-test measurements will be made of all important physical dimensions. Radiography probably will be used for non-destructive analysis of fuel and insulator behavior.

d. Tests

Test periods and energy releases are not yet established but will probably reach the operational limit of the CDC (for most) in order to closely approach the PBF Design Base Transient. The total number of transients will be established as testing proceeds.

4. DISCUSSION:

This test series is part of an overall fuel testing effort required to satisfy the stringent demands of the PBF. It is the intent of this test series (as well as others preceding) to prove current fuel designs as completely as possible prior to actual PBF operation.

5. STATUS:

Tests in this series will be performed during the first two quarters of FY 1970.

1. TITLE: Energy Calibration Tests

2. OBJECTIVES:

The objective of these tests is to evaluate and further develop fission product and calorimetric methods used to measure the energy deposited in test samples during CDC power excursions.

3. EXPERIMENTS:

a. Test Fuel

Fuel rod samples of the types used in the various fuel performance tests will be used.

b. Test Conditions

One or more fuel rods will be encapsulated in conventional CDC capsules and/or a calorimeter type capsule. The capsules will contain water at the temperature for which the calibration is desired. The driver core will be operated in a steady-state mode for most calibrations, although transient operation may be used in certain cases.

c. Measurements

During calibration tests, reactor power and energy release, capsule water temperature, and temperature distribution within the capsule will normally be measured. Flux wires and/or foils will be activated in both the core and capsule. Some experiments will require post-test analysis of test sample and hot-spot core fuel for fission product concentration.

d. Tests

(1) Low power steady-state tests and long period transients on fuel samples encapsulated in operational capsules to activate passive flux detectors and to produce fission product concentrations of sufficient magnitude to accurately determine the fission density.

(2) Low power steady-state tests on fuel samples encapsulated in a calorimeter capsule to determine the fuel sample energy release by measuring the temperature rise of the water in the capsule.

(3) Steady-state activation of flux wires and/or foils in operational and calorimeter capsules to obtain a correlation between the calorimeter and the operational capsule results.

4. DISCUSSION:

The proper interpretation and correlation of fuel performance data collected in the CDC depends heavily on an accurate knowledge of the energy deposition as a function of time in the test samples. The calibration method presently used in the CDC is based on fission product analysis to determine the fission density in

the encapsulated test sample relative to that in the core hot spot. Concurrently, flux wires are activated in the core to obtain a calibration constant relating activation to fission density. Flux wires activated during transient tests can then be used to compute the energy deposited in the test sample. The core energy release, as measured by ion chambers, provides the time-history of the energy deposition.

The tests proposed in this series are part of a continuing effort to refine the uncertainty in energy deposition data by further refinement of existing methods and by evaluation and development of new methods. Preliminary investigation of calorimetric methods indicate that the use of a calorimeter capsule to determine energy release may be feasible. The calorimetric method has the potential for reducing the uncertainty in energy deposition data to less than 10% and is more amenable for calibration of samples in limited supply, such as pre-irradiated or mixed oxide fuels, as destructive analysis is not required.

5. STATUS:

Tests in this series are performed periodically throughout the program.

1. TITLE: Upgrade Driver Core Capability

2. OBJECTIVES:

The objective of this test series is to establish the shortest period at which the CDC may be repeatedly pulsed without incurring fuel rod damage of such magnitude as to significantly affect subsequent operation; this will provide a higher energy deposition capability in the test space needed to accomplish the objectives of the fuel testing program in CDC.

3. EXPERIMENT:

a. Test Fuel

The fuel being tested is primarily that in the driver core. No fuel will necessarily be included in the test space; developmental instrumentation may be included in the test space for evaluation purposes.

b. Test Conditions

Core conditions representative of those normally in effect during fuel sample testing will prevail. A Type B capsule filled to within ten inches of the top with water may be located in the test space.

c. Measurements

Core measurements will include those normally monitored during excursion testing, eg, core power, energy release, period, clad surface temperature and pressure. In addition, certain fuel rods located in high flux regions will be periodically inspected to determine the onset and magnitude of damage in the form of clad deformation and rod bowing.

d. Tests

The driver core has previously been shown to perform satisfactorily at reactor periods down to 3.0 msec. Tests in this series will be performed at progressively shorter periods in about 0.1 msec increments, eg, 2.9, 2.8, 2.7, ... msec. It is expected that some repeat tests and some longer period tests will be included. The total number of tests cannot be specified but will be in the range of 10 to 20 or more.

4. DISCUSSION:

The CDC is presently capable of reactor periods down to 3.0 msec. This provides sufficient energy deposition capability in the test space to

accomplish a major part of the program objectives set forth for low-burnup fuels testing. Additional capability is required to meet the objectives of the pre-irradiated and LMFBR test programs in CDC. The goal of the core performance tests is to reach a period of 2.2 msec, which would increase the existing energy deposition capability at 3.0 msec by about one-third. It is recognized that the total desired gain in performance may not be attainable because of the possibility of core damage. Any added capability over that presently available will, however, be reflected in an increased information output of the subsequent fuel testing program.

5. STATUS:

Tests in this series may be initiated by the middle of FY 1970.

1. TITLE: Urania Equation of State

2. OBJECTIVES:

The objective of these tests is to establish for urania an empirical relationship between the enthalpy, h , and vapor pressure, P_v , in the enthalpy region between 270 cal/gram and 600 cal/gram.

3. EXPERIMENT:

a. Test Fuel

The test fuel will consist of either pellet or powder urania with the stoichiometric ratio well established and an enrichment of about 10%.

b. Test Conditions

A known quantity of urania will be placed in a dry capsule chamber and positioned in the CDC central flux peak. The chamber will be constructed with rigid walls to contain expected pressure rises and fitted with a fast-acting pressure transducer. These and other conditions such as fill-gas, chamber size, and initial temperature will be more firmly established at a later time as analysis of the problem and preparations proceed.

c. Measurements

Prior to testing any particular fuel samples, a similar fuel sample will be exposed in the CDC in a calorimeter designed to duplicate neutronic conditions of the actual test chamber. From this exposure, a measurement will be taken of the energy deposited in the fuel. This measure will then be used in conjunction with CDC energy measurements to obtain calculations of the energy deposited in actual fuel samples used for the h vs P_v determinations. Wall temperature of the chamber will also be measured to aid in accounting mathematically for urania condensation on the walls; chamber pressure will also be measured dynamically.

d. Tests

Details of test energy releases will be established as the program develops.

4. DISCUSSION:

A well known problem in the safety analysis of both thermal and fast reactors is the absence of good information on the equation-of-state of urania at enthalpies above the melting point. It is not possible at present to reliably predict urania pressures in accidents which deposits in the urania more than about 270 calories per gram. In the case of LMFR's,

the problem is particularly severe because of the extremely short neutron lifetime. LMFBR accidents can theoretically release heat at a rate which is comparable with chemical explosives and, for this reason, the vapor pressure of uranium takes on a significance of major import in determining the theoretical consequences of such accidents.

Controlled laboratory measurements of the $h-P_v$ relationship have not as yet been successful due to such problems as the management of materials at high temperature and difficult energy measurement methods. Even in the study planned here, difficult problems exist. However, this study has the distinct advantages of simplified materials-temperature methods, simplified (albeit coarse) energy measurement, and finally, the closer approximation of materials, geometries, heating rates, and initial conditions which would exist in a reactor accident.

Results of these tests are expected to have immediate AEC application in the development and safety analysis of LMFBR reactors.

5. STATUS:

Tests in this series are expected to begin in the third quarter of FY 1970.

1. TITLE: Scoping Tests of $\text{UO}_2\text{-H}_2\text{O}$ Reaction.

2. OBJECTIVES:

The objective of these tests is to determine the extent of reaction between UO_2 and water for high-energy tests by the measurement of hydrogen evolution.

3. EXPERIMENT:

a. Test Fuel

A stack of unclad, 10% enriched SPX fuel pellets approximately five inches high.

b. Test Conditions

The fuel stack will be positioned at the flux peak of the CDC in a Type-B capsule filled to within ten inches of the top with ambient temperature, demineralized water. The top ten inches will contain air at atmospheric pressure. Except for the capsule itself, there will be no exposed metal near the fuel stack. The stack will be contained in a cage of Al_2O_3 rods and exposed directly to the water (no cladding). A ceramic crucible about two-inches in diameter will protect the capsule from direct impact of molten urania.

c. Measurements

- (1) A pre-test measurement of the fuel weight.
- (2) Dynamic capsule pressure and water column velocity.
- (3) Post-test measurements of hydrogen evolution, particle size distribution, and fuel residue weight.

d. Tests

One or more, at periods of ~ 3 msec, which will produce energy depositions of approximately 600 cal/g of UO_2 .

4. DISCUSSION:

Measurements of metal-water reaction extent during fuel performance tests in the CDC are normally based on the quantity of hydrogen evolved by the cladding oxidation. Production of hydrogen by other processes, such as UO_2 -water reaction, could introduce a significant error into the data if not accounted for. The tests in this series are intended to provide scoping data on the UO_2 -water reaction extent for high energy depositions. If the hydrogen evolution is sufficient to significantly affect measurements of metal-water reaction extent, or if breakup of the fuel is atypical of the breakup of clad fuel, then a more detailed study will be undertaken as described in CDC-913. Data from these tests will be used primarily for planning and analysis purposes within the subassembly program.

5. STATUS:

Tests in this series were completed in the second quarter of FY 1969 and reported in IDO-ITR-108 (Reference 9).

1. TITLE: Detailed Tests of $\text{UO}_2\text{-H}_2\text{O}$ Reaction

2. OBJECTIVES:

The objective of this series of tests is to determine the extent of reaction between UO_2 and water as a function of transient energy deposition.

3. EXPERIMENT:

a. Test Fuel

Oxide fuel clad with non-active material such as glass or ceramic.

b. Test Conditions

The fuel will be positioned at the flux peak of the CDC in a Type-B capsule filled to within ten inches of the top with ambient temperature, demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

- (1) A pre-test measurement of the fuel weight.
- (2) Dynamic capsule pressure and water column velocity.
- (3) Post-test measurements of hydrogen evolution, particle size distribution, and fuel residue weight, oxygen-uranium ratio.

d. Tests

Several at energy depositions in the range from about 350 to 600 cal/g of UO_2 .

4. DISCUSSION:

The nature and extent of the tests in this series will depend on the results of CDC-912. Assuming significant UO_2 -water reaction is observed in CDC-912, this series will be carried out with the intent of obtaining sufficient data as a function of energy deposition to provide a correction to metal-water reaction data obtained during fuel performance tests on oxide fuel rods. The data will be used primarily for analysis purposes within the subassembly program.

5. STATUS:

Tests in this series were completed in the fourth quarter of FY 1969 and reported in IDO-ITR-108 (Reference 9).

1. TITLE: Cladding Temperature Measurement Development.

2. OBJECTIVE:

The objective of these tests is to develop and/or evaluate instrumentation techniques to measure the surface temperature time-history of Zircaloy cladding during transient fuel performance tests.

3. EXPERIMENT:

a. Test Specimens

Various types of thermocouple junctions attached to the surface of Zircaloy clad SPX fuel rods.

b. Test Conditions

Samples will be positioned in the flux peak in Type-B capsules filled to within ten inches of the top with ambient temperature, demineralized water. The top ten inches will contain air at atmospheric pressure.

c. Measurements

Of primary interest will be the output of the thermocouples.

d. Tests

Several Tests with energy deposition in the 150 to 250 cal/g of UO_2 range are planned.

4. DISCUSSION:

The in-pile tests in this series will be performed following extensive out-of-pile development. Two primary problems exist; (1) the tendency of most conventional thermocouple materials to form a eutectic when welded to Zircaloy which melts at a lower temperature than Zircaloy, and (2) no reliable method exists for attaching the thermocouples to the cladding which will provide a good bond between the materials and not excessively damage the Zircaloy.

5. STATUS:

Tests in this series were completed in the third quarter of FY 1969.

F. SCHEDULE

A schedule for performance of tests in the CDC is presented in this section. Because of delays, revisions, and additions to the test program during FY 1969, some of the tests originally scheduled during the FY 69-70 period are now planned for FY 1971, as shown on the schedule. It should be pointed out, however, that the major part of the FY 1971 program is in the planning stage and is not identified in this document.

It is not always possible to schedule CDC tests in the order of their importance because in many cases the schedule is controlled by logistics problems, viz, obtaining such things as materials, fuels, or even prior data. As shown, the schedule represents our current best effort to do the most important jobs first; but changes, consisting of advances for certain tests, delays for others, elimination of others, and the addition of new tests are expected to continually modify the planned program.

SUBASSEMBLY TEST PROGRAM
TENTATIVE SCHEDULE

	FY-69				FY-70				FY-71				
Unirradiated UO ₂ Fuels													
102	X												
103	X												
104	X	X											
105		X											
106		X	X	X									
107					X	X							
108			X										
109			X	X									
110		X	X										
111					X	X	X	X					
112						X							
113							X						
115		X			X								
116								X	X	X	X	X	X
117									X	X	X	X	X
118									X	X	X	X	X
Irradiated UO ₂ Fuels													
201							X		X	X	X	X	X
202							X						
203										X			
204								X					
205													
206					X	X							
207													
208				X	X								
209					X								
Plutonium Recycle													
301				X		X	X						
LMFBR													
501								X	X				
502										X	X	X	
Other Fuels													
901	X	X	X	X	X		X		X		X		
902							X		X				
903		X	X	X	X		X						
904		X	X										
905								X					
906		X				X							
907	X	X			X								
908					X	X							
909		X	X	X	X			X	X				
910						X	X		X				
911							X	X					
912		X											
913			X	X									
914		X	X										

VI. REFERENCES

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APPENDIX A

Description of the CDC

Facility Description

The CDC is located in the Spert IV facility which is approximately $\frac{1}{2}$ mile from the Control Center building and approximately $\frac{3}{4}$ mile from the nearest other reactor facility (Spert III). Nuclear operation of the CDC is carried out by remote control from the control console located in the Control Center building.

The reactor building, shown in Figures A-1 and A-2, houses the reactor tank, storage tank, the necessary auxiliary equipment and machinery, mechanical setup and working areas, and personnel convenience areas. It consists of a reactor bay flanked by low wings on either side above grade and a basement level below grade. The core is located within the open-top, north tank which is 20 feet in diameter and 25 feet deep. The south tank is used for fuel and irradiated test material storage. The two tanks are connected at the top by a 6- by 6-foot removable gate shown in Figure A-3.

The reactor control drives are located on a control bridge which spans the width of the tank and which is movable, on rails, for the full length of both tanks.

Core Description

Figure A-4 is a cutaway view of the CDC, and Figure A-5 is a plan view of the core. A photograph of a top view of the CDC is shown in Figure A-6. The core is divided by 12 aluminum rod-guide crosses which house the 8 cruciform-shaped control rod blades and the four cruciform-shaped transient rod blades. A cylindrical, stainless steel experiment tube in which test samples will be placed for kinetic testing lies along the vertical center line of the core.

The fuel rods are six feet long with welded-seam, stainless steel tube clad. They contain low-enrichment UO_2 powder, swage-compressed to an effective density of approximately 85 percent of the theoretical density of UO_2 . The rods contain nominally 1600 grams of UO_2 (three percent enriched) and are clad with a nominal 28 mils of Type 304 stainless steel. Reactor control is accomplished by the use of four control rod units, each consisting of two cruciform-shaped, neutron-absorbing blades connected by a yoke assembly and shaft to the armature of a coupling magnet. The four transient rods, which are ganged and used for step insertions of reactivity, are similar to the control rods except that their poison sections are normally below the active core region. The poison sections of the control and transient rods are 12 weight percent natural boron as boron carbide in an aluminum matrix and have follower sections which are constructed of aluminum.

The experiment tube, which extends the length of the core, consists of a six-foot-long, Type 304 stainless steel cylinder with inner and outer diameters of 3.438 and 4.500 inches, respectively. The tube, which has a welded end-plate on the bottom with a two-inch hole to allow for possible flow service to cool the capsule, is attached to the lower grid support.

A summary of CDC characteristics is found in Table A-1.

Figure A-1. The CDC (Spert IV) Reactor Building - Exterior, Cutaway View.

Figure A-2. Floor Plan of the CDC (Spert IV) Reactor Building.

Figure A-3. Elevation Drawing of the CDC (Spert IV) Reactor Building.

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Cutaway View Of Capsule Driver Core.

Figure A-4. Cutaway View of the CDC and Core Support Structure.

Figure A-5. Cross-section Plan View of the CDC Core and Experimental Tube.

Figure A-6. Top View of the CDC Installed in Spert IV Tank.

TABLE A-1

CHARACTERISTICS OF THE CDC REACTOR

Reactor Type	open pool
Reactor Vessel	
Inside diameter	20 ft
Depth	25 ft
Wall thickness	0.31 in.
Material	Type 304 stainless steel
Design pressure	
above gate	hydrostatic head plus 10-psi sur-charge
below gate	hydrostatic head plus 50-psi sur-charge
Design temperature	130°F
Primary Coolant	
Type	light water
Pressure	atmospheric
Volumetric flow	convection
Temperature	ambient ($\approx 70^{\circ}\text{F}$)
Fuel	
Type	compressed UO_2 powder in rod form
Rod OD	0.500 in.
Clad thickness	0.028 in.
Clad material	cold-worked, Type 304 stainless steel
Fuel enrichment, U-235	3 weight percent
Fuel density	9.28 g/cm ³
Length of fuel section	67 $\pm$ 1 in.
Control Rods	
Number	8
Type	cruciform blades
Width	4.75 in.
Thickness	0.245 in.
Poison	12 weight percent as a sintered dispersion of B_4C in aluminum
Poison length	61 in.
Rod drive	electric withdraw or insert; compressed air accelerated, gravity scram
Rods per drive	2
Follower	aluminum

TABLE A-1

CHARACTERISTICS OF THE CDC REACTOR (Contd.)

Transient Rods

Number	4
Type	cruciform blades
Width	2.125 in. by 3.375 in.
Thickness	0.245 in.
Poison	12 weight percent boron as a sintered dispersion of B_4C in aluminum
Poison length	28 in.
Rod drive	compressed air positioning and fast ejection
Rods per drive	4
Follower	aluminum

Core

Moderator/coolant	light water
Reflector	light water
Lattice spacing	0.663 in. square pitch
Nonmoderator-to-moderator ratio	0.807
Core diameter (overall)	28 in.
Active core height	67 ± 1 in.
Operational core loading	
number of fuel rods	1660
kg of UO_2	2.66×10^3
flux ratio (peak/avg)	3.6
Heat transfer active surface area	1221 ft^2
Reduced prompt neutron lifetime	4.18 msec
Reactivity coefficients	
control rod worth	12.2\$
excess reactivity	2.9\$
shutdown margin	9.3\$
isothermal temperature coefficient (at 20°C)	$-0.57\phi/^\circ\text{C}$
uniform void coefficient of core	$-1.7 \times 10^{-3}\phi/\text{cc}$
transient rod worth	$> 3\phi$

Experiment Tube

Tube material	Type 304 stainless steel
Length	6 ft
OD	4.500 in.
ID	3.438 in.
Wall thickness	0.531 in.

Flux

nv/watt (average over core)	$\approx 3.2 \times 10^5$
-----------------------------	---------------------------

TABLE A-1

CHARACTERISTICS OF THE CDC REACTOR (Contd.)

nv/watt (peak nv in test space -- with no fuel and no capsule present per watt of reactor power)
 nvt (peak in test space as above and for 3-msec transient)

$$\approx 3.5 \times 10^6$$

$$\approx 10^{15}$$

Although the CDC has an active core length of six feet, control rod poisoning has reduced the useful length of the core to about 40 inches. This is shown in the vertical flux profile (cobalt) of Figure A-7 taken at a fuel rod position (3-3) adjacent to the experiment tube. The depression in flux at a vertical position of 18 inches is due to the presence of an aluminum support grid. This "intermediate" grid serves to prevent fuel rod movement which arises due to skewed temperature distributions in the fuel and cladding.

Figure A-8 shows the radial flux profile (cobalt) taken at a vertical height near the intermediate grid and with no capsule present in the experiment tube. The effect of the experiment tube as a flux suppressor is clearly evident.

More important to capsule experiments is the flux profile in the test space itself. Figure A-9 shows the vertical flux (cobalt) profile along the center line of the experiment space. Within a typical capsule, but with no fuel present, there is no obvious depression in this profile arising from the intermediate grid.

FIG. A-7 Vertical Flux Profile at Position N3, E3 - adjacent to the Experiment Tube.

Irradiation Performance of the CDC

Typical burst shapes of the CDC are shown in Figure A-10 for several initial asymptotic periods. The upper decade of each of these power bursts is quite symmetric around the peak; but as the power falls after peak into the region of about 100 MW, the CDC typically begins a long, relatively flat power runout. Conventional operation calls for scram shortly after the main power burst in order to eliminate the power tail.

Figures A-11 and A-12 show the experimental relationships of peak power and of the energy release in the CDC to the reciprocal of the asymptotic period. In Figure A-12, the "total energy release" is defined arbitrarily as the energy release up to a time of 0.5 second after peak power.

Figure A-8. CDC Radial Flux Profile.

Figure A-9. Axial (vertical) Flux Profiles in the CDC Test Space - Comparison of Two Experiments.

Figure A-10. Experimental CDC Power Plots for Various Period Transients.

FIG. A-11 Experimental CDC Peak Power Versus Reciprocal Period.

FIG. A-12 Experimental CDC Energy Release Versus Reciprocal Period.

Performance of the CDC as an irradiation facility is related to the number of neutrons supplied to the test space during a power burst. The major limitations here lie almost entirely upon the heat capacity of the core material and the maximum enthalpy to which the core may be raised without damage.

From experience with the CDC, it has been shown that a hot-spot enthalpy of at least 100 cal/g in the CDC oxide material should be possible. Higher enthalpies may be possible, but scorching, and thermal expansion problems are expected to arise above 120 cal/g. Presently, the CDC is limited to a peak enthalpy of 75 cal/g until a portion of the experimental program is completed.

As a measure of the performance of the CDC, a figure-of-merit, F_m , has been defined which is simply the ratio of the hot-spot energy releases in the test fuel to the core fuel.* Thus, if a sample fuel rod achieves a maximum energy release of 200 cal/g of UO_2 while the core hot-spot achieves only 100 cal/g of UO_2 , the figure-of-merit is 2.

Representative of the CDC performance is the figure-of-merit possible when a standard CDC fuel rod (stainless steel clad, 0.5-inch-OD, 3 percent enriched UO_2) is placed in a capsule, see Figure A-13, in the experiment tube. For this configuration, the presently measured figure-of-merit is about 1.9. If the same fuel rod were 4 percent enriched, the figure-of-merit would be 2.7.

* The "hot-spot" as used here is defined as the radial average energy release at the axial (vertical) flux peak of the fuel rod in question.

Figure A-13. Typical Capsule Designed for Fueled Irradiation Experiments in the CDC. The Body of the Capsule is 3-inch-OD by 0.065-inch Wall Thickness, Type 304 Stainless Steel.

Higher figures-of-merit are possible with smaller diameter fuels and with higher enrichments. Lower figures-of-merit arise when clusters of fuel rods are placed in the capsules.

Table A-2 illustrates figures-of-merit possible with various fuel samples. From the numbers shown, it is apparent that the CDC should be able to produce vaporization of UO_2 in the highly enriched samples.

TABLE A-2

FIGURES-OF-MERIT FOR CLUSTERS OF FUEL RODS WHICH ARE 0.25 IN. OD, 0.014 IN. STAINLESS CLAD, 95 PERCENT THEORETICAL DENSITY AND WITH ENRICHMENT AND CLUSTER SIZE AS SPECIFIED

<u>Cluster (No. of Rods)</u>	<u>Enrichment (percent)</u>	<u>F_m</u>
1	3	3.7
1	5	4.8
1	10	7.8
3	3	~ 2.9
3	5	~ 3.9
3	10	~ 5.9

APPENDIX B

Analysis Codes (Through FY 1968)

The following are brief descriptions of efforts being made in the Safety Division of INC to improve analytical capability for predicting reactor behavior and fuel failure under power transient conditions. Two of these deal with thermo-mechanical behavior of fuel pins and another with the reactor kinetics as the fuel experiences a power transient. HEAT-1-JUNK and CYGRO-2 can be applied to analysis of fuel pin behavior during a power burst up to the point of cladding failure. HYDRAX deals with reactor performance through pin failure and beyond, but requires input on assumed heat exchange from expelled fuel to the coolant. No analysis has been attempted for the mechanism of this heat transfer and it appears that an experimental approach will be more successful in yielding information on this phase.

In addition, some presently available neutronic codes are briefly listed.

HEAT-1-JUNK

Introduction

The purpose for this model is to predict thermal-mechanical behavior of fuel rods and to make a sensitivity analysis to determine the important parameters affecting and causing fuel rod failure during reactor power excursions. The model has been developed for computer solution using finite difference techniques. The model has provision for study of effects caused by fission flux distribution, fuel-clad gap, temperature dependent thermal properties, system operating conditions, fuel and cladding mechanical properties, and fuel rod design. The model will calculate the time dependent temperature distribution in the rod and from this predict behavior such as fuel and cladding thermal expansion, fuel melting, fuel-cladding interfacial pressure and cladding stress and strain.

Model Description

The model consists of a one-dimensional heat conduction code coupled with subroutines that compute the fuel-clad mechanical interaction and the effect of this on the fuel-clad thermal resistance. The heat conduction code had previously been developed and is called HEAT-1*.

The computer code HEAT-1 provides a numerical procedure for obtaining a solution, that is the temperature distribution, to the one-dimensional time dependent or steady state heat conduction equation in slab or cylindrical geometry. Provision is made for regions of different

* IDO-16867, "HEAT-1, A One Dimensional Time Dependent or Steady State Heat Conduction Code for the IBM-650", R. J. Wagner.

HEAT-1-JUNK (Continued)

materials, ie, fuel, fuel-clad gap, cladding, and coolant. The thermal properties necessary for solution within the fuel rod, ie, thermal conductivity and volumetric specific heat are described as temperature dependent functions over an applicable temperature range. Changes in phase may be accounted for by the heat of fusion or heat of vaporization.

The energy generated within the fuel rod may be distributed as a function of space, material, and time. A spatial distribution based on fission flux depression must be determined from diffusion or transport theory. The time dependent power level to which the model rod is exposed is found from kinetics calculations or power measurement of a reactor during an excursion.

The boundary conditions necessary to complete the solution for the temperature distribution may be fixed constant or expressed as functions of time.

Although the HEAT-1 code is general in nature, special subroutines are needed to calculate changes in thermal properties and boundary conditions as a function of parameters other than time and temperature. These routines also permit calculation of the rod's mechanical behavior. These routines are grouped together under one name called "JUNK". Thus the combined program has been called HEAT-1-JUNK.

The procedures and assumptions used in JUNK are very flexible as they may be modified or enlarged to suit the users own special needs. The routines as they have been developed so far are not fixed but are constantly being improved and modified.

At the present the following routines are included in JUNK:

The thermal resistance of the gap between the fuel and cladding is a function of gap size or contact pressure as the gap closes during an excursion. Thus, one job of JUNK is to determine the size of the gap and the interfacial pressure once contact between the fuel and cladding is made. Gap size is determined by calculating the time-wise differential thermal expansion of the fuel and cladding and comparing this to the initial specified gap. The thermal expansion of the fuel is a function of temperature until contact with the clad occurs. Then it becomes a function of the restraint imposed by the clad also. Once contact is made the theory developed for shrink-fitting cylinders is applied to determine interfacial pressure and fuel-clad deformation until the cladding elastic limit is reached. Then deformation is based on fuel expansion.

To determine the cladding elastic limit the stress and strain are computed including the effects of temperature on the pertinent mechanical properties, temperature gradient, system pressure and fuel-clad interfacial pressure.

The thermal boundary condition between the clad and water coolant is also not a simple function of temperature. It is a function of the heat flux, coolant flow rate and temperature difference as well. These

HEAT-1-JUNK (Continued)

factors influence the heat transfer mechanism, ie, conduction, convection, nucleate boiling or film boiling. Thus the user must select the mechanism criteria which JUNK can use to predict a clad-coolant thermal resistance which feeds back into the main program as information necessary to derive the needed temperature distribution.

Assumptions and Limitations

The main assumption inherent in JUNK is that the thermal and mechanical behavior can be predicted using a one-dimensional model, ie, radius for cylindrical rods. Axial effects are therefore difficult to include or are omitted.

The mechanical behavior is based on elastic theory which is not good beyond the yield points of either fuel or cladding. The inclusion of plastic stress-strain equations has been deemed too complex for the initial purpose of the code, ie, sensitivity analysis, however the effect of plasticity should be included for more sophisticated analysis.

The JUNK routine was not intended to compute the mechanical behavior as a function of long-term burnup during quasi steady state operation. However, the effects of burnup on mechanical properties, thermal properties, and internal fission gas pressure can be included as initial conditions.

Forced flow steady state correlations are used for all clad-coolant heat transfer predictions.

Problems

Several problem areas exist in the development of models for the prediction of the thermal-mechanical behavior of fuel rods subjected to power excursions. During a transient the rate of strain has a significant effect on the mechanical properties, ie, yield stress, modulus of elasticity, and elongation of the clad prior to rupture.

Because of a lack of applicable experimental data and calculational techniques, it is difficult to predict the elongation or rupture point for a tube exposed to a biaxial stress as would occur during steady state or transient operation. Most elongation data are obtained from uniaxial tensile tests. However, the allowable elongation is considerably reduced during biaxial loading.

The phenomenon of clad ridging is a problem. A cause has been postulated, ie, fuel pellet hourglassing, but no analysis to date can predict its magnitude. This effect could be included in the model if a valid theory existed.

The plastic behavior of the fuel once clad contact is made is not well known or described. To calculate how the ceramic fuel will continue to expand, radially or axially when restrained by the clad, the axial restraint imposed by the fuel and the frictional force between the UO₂ and clad are needed.

HEAT-1-JUNK (Continued)

The effect of pellet cracking on thermal-mechanical behavior is not readily amenable to solution. Nor is the effect of repeated transients and the axial growth observed by F- and PBF-rod transient testing conducted in the CDC.

CYGRO-2

Introduction

This digital computer program, originated at Bettis for use on the CDC-6600, is being converted for use by Phillips on the UNIVAC-1108. It provides a method of structural analysis for an oxide-fueled, metal clad rod-type fuel element in a pressurized environment. Major loading conditions include fuel swelling, fission gas and coolant pressures, clad growth and differential thermal expansion. The application for which the program has been developed is Zircaloy tubes containing bulk oxide fuel, but the basic techniques may be extended to other materials.* It is intended to modify the code for use at Phillips, to include behavior of fuel under rapid power excursion conditions.

Physical Problem

The problem of interest is the quasi-static transient stress analysis of a nuclear reactor fuel element in the form of a long metal tube which is the clad for cylindrical pellets of porous fuel and is closed at each end. The pellets may be in radial contact with the clad, but are never in contact with the top of the tube. There is axial symmetry, and plane cross sections perpendicular to the axis remain plane, so that all variables change only with radius and time.

Each fuel pellet may be solid or may have a cylindrical hole, the hole may contain a rigid plug, and the plug may be bonded to the pellet. The axial restraint of the plug on the pellet is neglected. Each pellet is assumed to be a homogeneous material containing spherical holes. Surface tension and gas generation by irradiation effect the volume of the holes.

The fuel element is subjected to the following time dependent forcing functions: coolant temperature, coolant pressure, gap pressure between pellets and tube, fission gas pressure in the fuel pores, and power level. The power level is a measure of irradiation rate, which generates fission gas in the pellet pores, generates fission solids in the homogeneous fuel, and generates heat in the plug, fuel and clad.

Heat generation causes a temperature rise in the fuel element above coolant temperature. Specific heat and axial heat transfer are neglected, so that the temperature distribution at any time is the solution of the steady-state equation of radial heat transfer. Film coefficients are used at the outer radial surfaces of both fuel and clad. Thermal conductivity of the fuel is a function of temperature and porosity, and the conductivity of the clad depends only on temperature. The heat

* Descriptions here are taken from WAPD-TM-547.

generated in the plug determines the radial temperature gradient at the inner surface of the fuel.

During a change in time, a change in fuel volume is caused by thermal expansion, solid fission products, fission gas pressure, surface tension of the holes, gap pressure between fuel and clad, and contact forces on the fuel by the plug and the clad. The change in clad volume is due to thermal expansion, gap pressure, coolant pressure, and contact forces on the clad by the fuel. Usually, the variation in fuel volume is much larger than that of clad volume.

Deformation of the fuel element is affected by the elastic-plastic creep properties of the fuel and clad materials. These properties are functions of temperature and irradiation.

If the fuel pellet is solid, there is no plug and the inner radius of the pellet is zero for all time. If the pellet is bonded to a plug, the inner radius of the pellet equals the rigid plug radius for all time. If the plug is not bonded to the pellet, there may be separation or contact. If there is a radial gap between fuel and plug, the pressure on the inner radial surface of the fuel is the gap pressure between fuel and clad. Otherwise, there is an additional radial contact force between plug and fuel, with the axial restraint of the plug on the fuel being neglected.

If there is a radial gap between fuel and clad, the pressure on the outer radial surface of the fuel and the inner radial surface of the clad is the gap pressure, the axial load per unit area of the fuel is the gap pressure, and the axial load on the clad is produced by the tensile effect of the gap pressure and the compressive effect of the coolant pressure. Otherwise, when the fuel is in contact with the clad, there are radial and axial contact forces between the fuel and the clad, in addition to the pressure loads.

Method of Solution

The initial forces, dimensions, stresses and strains are assumed known. Consider an infinitesimal height of fuel and clad, with both fuel and clad partitioned into concentric rings. Each ring is considered separately as a free body, with stresses, strains, and material properties assumed uniform throughout the ring. For a small increment of time, the stress rates and strain rates are assumed to be constant and uniform throughout each ring.

The radial, circumferential, and axial strain rates are expressed in terms of the inner radial, outer radial, and axial deformation rates of the ring. Strain rates are related to stress rates by the Prandtl-Reuss equations, modified to include thermal expansion, fuel growth, plasticity, and creep. The equations are solved for the radial, circumferential, and axial stress rates in terms of the ring deformation rates. The internal strain energy rate in a ring is equated to the

CYGRO-2 (Continued)

external work rate on the ring to obtain equilibrium equations relating ring boundary loads to stresses. Differentiating the equilibrium equations with respect to time gives the ring boundary load rates in terms of the stress rates. Since the stress rates have been found in terms of the ring deformation rates, the three ring boundary load rates can be expressed in terms of the three ring deformation rates. These are the flexibility equations of the rings.

The fuel rings are joined together into one free body of fuel by the following process. The radial forces between adjacent rings are set equal but opposite in sign by Newton's third law of action and reaction, and the radial deformation rates at adjacent boundaries are set equal for continuity. The axial deformation rates are also set equal and the sum of the ring axial force rates is set equal to the total axial load rate on the fuel. The boundary condition equations are combined with the flexibility equations of the rings to determine the three flexibility equations of the fuel free body; that is, the inner fuel radial load rate, outer radial load rate, and total axial load rate are expressed in terms of the inner radial deformation rate, outer radial deformation rate, and axial deformation rate. The same process is used to obtain the three flexibility equations for the clad.

The boundary conditions of the fuel and clad have been given in the preceding section describing the physical problem. These boundary conditions are used with the flexibility equations of the fuel and clad to determine the load and deformation rates of the fuel and clad. The ring flexibility and boundary condition equations are then solved to give the load and deformation rates of each ring. Since stress and strain rates have already been expressed in terms of ring deformation rates, the stress and strain rates of each ring are now determined. The size of the time increment is restricted so that the maximum changes in stress and strain during the time step do not exceed specified limits.

All rates are multiplied by the time increment and added to the corresponding variables at the start of the time step to determine their values at the end of the time step. This procedure is repeated until the end of the problem.

Provision is made for up to five types of pores which may contain gas. Generation of the gas is determined as a function of irradiation rate, and diffusion of the gas from the holes is calculated in terms of irradiation and temperature. The pore model is a spherical hole centered in an incompressible sphere with plastic-creep properties. Pores are subjected to surface tension forces (sintering) and hydrostatic pressure (hot pressing) tending to reduce the hole size, densifying the ceramic. Fission gas within the pores tends to increase hole size resisted by surface tension, hydrostatic pressure and ceramic strength (fuel swelling).

Thermal conductivity is a function of both temperature and porosity, and elastic-plastic-creep data are given in a table of properties for four temperatures.

CYGRO-2 (Continued)

A more realistic representation of large deflection theory is used to relate strains to deflections, and stresses to forces, in terms of the geometry of the element.

Major Assumptions

Major assumptions used in the development of the program include:

1. Axial and azimuthal symmetry of temperature and stresses.
2. Steady-state temperature calculations.
3. Fuel pellet cracking is not allowed.
4. Fuel-clad contact always operates without slip.
5. Mass transfer from ring to ring is not considered.
6. Fission gas release from the pellet must be specified for each type of hot pore.
7. The gas pressure in the plenum is specified directly as the amount of gas in the plenum is not considered.

HYDRAX

Introduction

The HYDRAX digital computer code for the UNIVAC-1108 is designed to calculate the total fission energy that would be released in a water-moderated power reactor in the event of a large accidental reactivity insertion. It solves the system of coupled differential equations which represents the kinetic behavior of a water-moderated reactor in a short-period excursion. A number of coupled neutronic/hydrodynamic codes are in existence, of which the AX-1 code, which was developed to predict the consequences of short period excursions in a fast reactor "meltdown and reassembly" situation and the RAC code, which was used with some success to predict the energy release in the graphite core KIWI-TNT test are perhaps the best known. None of these codes, however, are directly applicable to the analysis of a large reactivity accident in a water-moderated reactor for reasons which will be set out in some detail below.

In a water-moderated reactor, unlike a fast reactor, physical core disassembly is not an important shutdown effect since there exist strong prompt feedback mechanisms which will act to terminate the nuclear portion of the excursion before disassembly begins. These feedback effects include Doppler broadening of fuel resonances with increasing fuel temperature, and decreasing moderator density due to prompt n and γ heating of water in the coolant channels. In addition, if the transient is so severe that the reactor starts to "run away" from the prompt Doppler and moderator density feedback, and fuel vaporization and pin rupture occur, hot fuel will be injected into the moderator and the resultant rapid heating and

HYDRAX (Continued)

voiding of the coolant will act to terminate the transient almost immediately.

In the HYDRAX code, the neutronic behavior of the reactor is represented by a space-independent "point-kinetics" calculation. Fission energy is distributed into the fuel and moderator according to a thermal flux profile which may be constant or time-varying. The kinetic behavior of the coolant is represented by a 1-dimensional, inviscid, compressible-fluid hydrodynamic model which includes a tabular, multiphase water equation of state. The reactor fuel is assumed to remain in position until after the nuclear portion of the excursion is terminated. Once the fuel energy density in a region reaches a specified threshold level, the fuel is assumed to fragment and disperse into the moderator. Energy transfer from the failed fuel to the moderator then proceeds according to a specified volumetric heat transfer coefficient. Reactivity feedback contributions are calculated locally on the basis of local fuel and moderator conditions and then are weighted by an importance function before being summed into the total system reactivity and fed back into the neutronics equations. Feedback mechanisms considered include fuel Doppler, moderator density and moderator temperature effects.

Neutronics Model

Physical disassembly of the core is not considered in HYDRAX, since it was felt that the additional complexity of a space-dependent neutronics calculation could not be justified for the first version of the HYDRAX program. Consequently, a much simpler space-independent "point-kinetics" formulation with a constant or time-varying spacial distribution of fission energy was used.

The neutron kinetics equation in HYDRAX are solved in the form proposed by H. Soodak.

$$\frac{dN(t + \Delta t)}{dt} = \frac{\bar{\beta}}{\Lambda} \left\{ [\$(t) - 1] N(t + \Delta t) + \frac{\bar{\beta}}{\Lambda} [$(t + \Delta t) - $(t)] N(t + \Delta t) + \sum_{i=1}^n f_i W_i(t + \Delta t) \right\} + S(t) \quad (1)$$

$$\frac{dW_i(t + \Delta t)}{dt} = \lambda_i N(t + \Delta t) - \lambda_i W_i(t + \Delta t) \quad (2)$$

This method of solution is a modified Runge-Kutta scheme suggested by A. H. Cohen.

HYDRAX (Continued)

The fission energy calculated in equation (1) is distributed into the fuel and moderator regions according to a spacial flux profile which may be constant or may vary with time. In the latter case, new flux profiles are read in at designated times and then the code linearly interpolates between two appropriate profiles to determine the proper spacial energy distribution to use at a specific time.

The reactivity, $\rho(t)$, in equation (1) is obtained by summing the local values of the various reactivity feedback quantities in each spacial region weighted in each case by an importance factor, the flux times the adjoint flux times the region volume, which is normalized such that the total importance is one.

Initiation of nuclear excursions in HYDRAX can be accomplished in a variety of ways allowing almost any sort of a reactivity accident to be simulated. Provision is also made for threshold reactivity insertion mechanisms dependent on power or energy level for their initiation.

Fuel Model

In the HYDRAX code, the fueled reactor core is divided up into 1 or more unequally-spaced fueled regions. The fuel is assumed to remain in its original position until after the neutronic portion of the excursion is over. Fragmentation of the fuel into small particles may occur in a fueled region once the fuel energy density reaches a threshold value; however, the fuel fragments are assumed to be retained within the region and the relative power distribution is assumed to be unchanged.

Since the duration of the neutronic portion of a severe reactivity transient is so short and the conductivity of oxide-fuels is so low as to preclude any significant prompt energy transfer through convective heating, fuel heating in the HYDRAX code is adiabatic with a constant fraction of the fission energy distributed to a fueled region being retained in the fuel. Once the fuel energy density threshold for failure is reached in a fueled region and the fuel ruptures, heat transfer between the fuel fragments and the moderator is assumed to proceed according to a constant volumetric heat transfer coefficient.

Since the power distribution, and, hence, the fraction of fission energy distributed into any fueled region may vary with time, the fuel energy density in a fueled region is obtained by integrating the instantaneous rate of fuel heating in that region. The method of integration is discussed in Appendix A.

Moderator Model

Moderator density reactivity feedback is an important factor in determining the total fission energy release during a short period excursion in a water-moderated reactor.

If fuel integrity is maintained, prompt neutron and gamma heating of the moderator tends to rapidly reduce the moderator density in the high-flux core regions, and since the void coefficient is strongly negative, this amounts to a sizeable prompt negative reactivity feedback.

HYDRAX (Continued)

If fuel integrity is lost, droplets of hot molten fuel are rapidly expelled into the moderator in the vicinity of the ruptured elements. The resultant rapid heating of the moderator by the molten fuel droplets gives rise to a large pressure pulse which acts to expell water from the core and terminates the nuclear portion of the excursion almost immediately.

Thus, moderator density reactivity feedback from prompt moderator heating or from fuel element rupture is a shut-down mechanism that may act to limit the energy release in a severe reactivity accident far short of the value which might have been predicted on the basis of Doppler feedback alone.

The driving force behind the reduction in core moderator density in a short-period transient is the pressure pulse generated by rapid moderator heating. Since this pressure pulse has a very short rise time and can propagate quickly through the core, and since the moderator enthalpies and densities may vary rapidly over a wide range as heating and vaporization of the moderator occurs, the "incompressible" and "semi-compressible" models often used to describe the fluid dynamics of the moderator in a nuclear reactor are not sufficient in this case and a "full compressible" model must be used.

HYDRAX solves the balance equations of mass, momentum and energy which describe the dynamic behavior of a compressible, inviscid fluid

$$\frac{D\vec{u}}{Dt} = -v\vec{\nabla}P - \vec{u}|\vec{u}| \frac{2f_F}{D_e} \quad (3)$$

$$\frac{De}{Dt} = vQ - Pv(\vec{\nabla} \cdot \vec{u}) + \vec{u} \cdot \vec{u}|\vec{u}| \frac{2f_F}{D_e} \quad (4)$$

where

$$P = P(e,v) \quad (5)$$

$$T = T(e,v) \quad (6)$$

and the differential operator $\frac{D}{Dt}$ is defined as

$$\frac{D}{Dt} \equiv \frac{\partial}{\partial t} + \vec{u} \cdot \vec{\nabla}. \quad (7)$$

in one-dimensional slab, cylindrical-r or spherical-r geometry.

The hydrodynamic pressure calculated from equations (3), (4) and (5) may be augmented by the addition of a term, $\Delta P_v = C_{vp}/v [(\Delta r/v) (Dv/Dt)]^2$,

HYDRAX (Continued)

when the fluid undergoes a compression. This term represents a viscous dissipative mechanism in the fluid in the region of a shock.

Equations (5) and (6) represent the multiphase equation of state of water. The calculation of the moderator pressure and temperature from the energy and specific volume is accomplished by interpolating in a large "steam-table" constructed from experimental data. The method of interpolation is based on successive quadratic fits on the energy and then the specific volume.

A constant fraction of the fission energy is assigned to prompt moderator heating at a given moderator density with the fraction being reduced in proportion to any reduction in moderator density. The allocation of energy to individual moderator regions is accomplished by means of a flux or power profile which may be constant or time-varying. If a time-varying flux profile is used, two appropriate profiles corresponding to times preceding and succeeding the current problem times are used in an interpolation scheme to determine the current flux profile.

The frictional term in equations (3) and (4) may be used to represent the frictional drag of the coolant channel walls (slab geometry) or the resistance to radial outflow of the fuel rod matrix (cylindrical or spherical geometry).

The method used to integrate equations (3) and (4) is similar to that used for the neutronic equations.

Fuel-Pin Failure Model

Once a significant amount of fuel is vaporized within a fuel pin, rupture of the fuel clad and ejection of hot fuel into the adjacent moderator will occur. The actual mechanism of heat transfer between the fuel particles and the moderator may be fairly complex involving a number of distinct phases. However, it seems unlikely that it will be possible for the present to do more than to assign a single overall heat transfer coefficient to this process. Approximate values for this coefficient may then be determined on the basis of experimental tests such as those presently being carried out in the Capsule Driver Core installed in the Spert IV reactor at the NRTS.

In the HYDRAX program, once the fuel energy density in a fueled region reaches a specified threshold value, the fuel in the region is assumed to fragment instantaneously while remaining within the region for purposes of heat transfer calculations. Heat transfer between the fuel fragments and the adjacent moderator proceeds according to a specified constant volumetric heat transfer coefficient which is assigned on the basis of an estimated heat transfer area per unit core volume and an estimated overall surface heat transfer coefficient. The flux distribution is assumed to be unchanged by the fragmentation of the fuel and so the fuel and prompt moderator heating rates are calculated on the same basis as before the failures occurred.

HYDRAX (Continued)

Reactivity Feedback Models

In the HYDRAX Code, the various reactivity feedback quantities are calculated in each fuel or moderator region on the basis of local values of the fuel or moderator variables in that region. Then, these feedback quantities are weighted by an importance function, the normalized product of the flux, adjoint flux and the region volume, and summed into the total reactivity which is used in the neutronic calculations. The local values of the flux and adjoint flux are calculated from profiles that are read in as input data.

Fuel Doppler feedback is calculated on the basis of the local value of the fuel enthalpy in each fuel region. The relationship may be expressed either as a table of values or as an analytical function with specified coefficients. When fuel failure occurs in a region, the Doppler feedback in that region generally increases due primarily to loss of self-shielding in the fuel as the lump size decreases. HYDRAX makes provision for specifying a certain fraction of the fuel in a failed region to have its Doppler contribution multiplied by a specified factor to account for this change. Since Doppler feedback is generally expressed as a function of the core-average fuel enthalpy, provision is made in the HYDRAX program to normalize the importance-weighted Doppler feedback to the core-average value at a specified core energy level.

Reactivity feedback effects due to changes in the moderator are considered in two parts: first, a pure density coefficient, \$ per percent change in specific volume at constant temperature; and secondly, a pure temperature coefficient, \$ per Fahrenheit degree change in temperature at constant specific volume. These moderator reactivity coefficients are assumed to be constants in the present model.

PARET

A Program for the Analysis of Reactor Transients

PARET is a digital computer code which predicts the course of non-destructive accidents in small water-moderated reactor cores. Its operation is based on an evaluation of the coupled thermal, hydrodynamic, and nuclear effects in the core.

Basically, the program is an extension of the CHIC-KIN code. Whereas CHIC-KIN assumes the reactor to be represented by a single region, PARET possesses the advantage of providing for up to four regions. The several further advantages that PARET possesses over CHIC-KIN lie primarily in the realm of boiling.

The reactor model assumed in PARET consists of a water-cooled core represented by an arbitrary number of fuel elements and associated concentric moderator channels, up to a maximum of four. The fuel

PARET (Continued)

elements may be either cylinders or plates. Each of the channels included in the calculations is assumed to represent the average behavior of some selected region of the core. Provision is made for weighting these average channels for purposes of the reactivity feedback calculations.

A transient problem is forced through specification of externally inserted reactivity versus time or average core power versus time. Either the pressure drop across a given channel of the core or the moderator inlet mass flow rate to that channel must be specified with time. The program embodies continuous reactivity feedback from the known major feedback mechanisms, detailed heat transfer calculations including boiling, as well as nonboiling, convective transfer, and hydrodynamic calculations allowing for moderator flow reversal.

Heat transfer in each fuel element is determined on the basis of a one-dimensional conduction solution in each of up to twenty axial sections. In turn, each axial section can be subdivided into a maximum of fifty radial sections for purposes of these calculations. The moderator in each axial section is not subdivided radially; it is assumed to be adequately represented by an average bulk temperature and average fluid properties. A maximum of three regions (eg, fuel, insulating region or gas gap, and clad) is allowed in each fuel element.

IREKIN

The IREKIN computer code is a numerical solution of the ordinary, space-independent, differential equations describing the power-time behavior of a reactor in response to an arbitrary reactivity input. This code has been used to calculate reactor response at short reactor periods where heat transfer effects are not significant.

The purpose of this code is to obtain a numerical solution of the ordinary, space-independent, differential equations which describe the power behavior of a reactor in response to an arbitrary function of reactivity. In addition, the code yields numerical solutions to other ordinary differential equations which might be used to describe a particular reactor system. Some, or all, or even none of these additional differential equations may describe the feedback mechanisms representing the dependence of the reactivity of the system upon the reactor power and/or energy release. Thus, the reactivity can be represented as the sum of an arbitrary function of time and a function of the power, energy, and other variables which may be specified in the differential equations.

In methods and purpose, this code is similar to the AIREK code. Neither is a complete code in the sense that subprograms must be added in order to define the particular model for the reactor feedback and to specify the desired variables and the form for the output. The IREKIN code differs from the AIREK code mainly in programming details in order to permit easier inclusion of a description of the reactor

IREKIN (Continued)

model of interest and also to reduce the computation time to some extent. The IREKIN code represents an attempt to write a flexible code, readily adaptable to any reactor kinetics problem, but yet remaining easy to use. It is hoped that through the use of the subprograms, a wide range of reactor kinetics problems with special input and output requirements can be easily accommodated.

Reactor Neutronics Codes

As input to analyses for prediction of fuel failure and consequences, the following reactor neutronic codes are in use at Phillips, AED. However, performance of the CDC facility is now well established and empirical data for input to thermodynamic and mechanical studies is available in most cases. The brief listing here is included for general information.

A. PDQ

PDQ solves the neutron diffusion equations in XY or RZ geometry for up to 20,000 mesh points, and for one to four neutron energy groups. The boundary conditions allow zero flux, reflective symmetry, and 90° or 180° rotational symmetry.

Reference: W. R. Cadwell, PDQ-4
(PHILCO-2000), WAPD-TM-230
(June 1961)

B. IMP

IMP is the preliminary version of the CORE code. The code solves the general one-dimensional neutron diffusion problem with up to four neutron energy groups, forty regions, and two hundred and fifty space points.

Reference: E. C. Anderson and G. E. Putnam,
CORE, IDO-17199 (To be published)

C. TEMPEST

TEMPEST is a neutron thermalization code and it computes the thermal neutron flux spectrum based upon one of the following approximations:

1. The Wigner-Wilkins light moderator equation
2. The Wilkins heavy moderator equation
3. The Maxwellian distribution

It provides macroscopic and microscopic cross-section averages over the thermal neutron spectrum. It provides effective diffusion constants for highly absorbing, or moderating regions such as control rods.

Reference: R. H. Shudde and J. Dyer,
TEMPEST II, TID-18284 (May 1961)

D. PHROG

PHROG is Phillips Hanford-Revision of GAM-I and is a multigroup code for calculation of fast neutron spectra and multigroup constants.

Reactor Neutronics Codes (Continued)

It provides effective diffusion constants for highly absorbing, non-moderating regions such as control rods.

Reference: G. D. Joanou and J. S. Dudek,
GAM-I, GA-1850 (June 1961)

E. ZUT, TUZ

These codes calculate resonance integrals from resonance parameters for a wide variety of temperatures, compositions, and geometries. The resolved resonances are treated by the ZUT program and the unresolved resonances are treated by the TUZ program.

Reference: G. F. Kuncir, GA-2525
(August 1961)

F. DANCOFF JR

This code calculates the Dancoff-Ginsburg correction which accounts for the reduction of surface resonance absorption in a cylindrical absorber lump due to the shadowing effect of nearby lumps.

Reference: H. Gellings and A. Sauer,
DANCOFF JR, KEA-114 (April 1964)

G. TOPIC

TOPIC solves the one-dimensional Boltzmann equation in cylindrical geometry with up to six energy groups, two hundred and forty space points, forty regions, and anisotropic (P_1) scattering.

Reference: G. E. Putnam, TOPIC, IDO-16968
(April 1964)

H. TOTEM

TOTEM links the TOPIC and the TEMPEST codes into a single package. TOTEM iterates between a thermal group cell flux calculation using TOPIC and a neutron spectrum calculation made with TEMPEST. The code has not been documented.

Reference: See TOPIC and TEMPEST

APPENDIX C

Listing of CDC Fuel Tests Through FY 1969

Table C-1 contains a list of all transients conducted through FY 1969 in the CDC for the purpose of testing samples of fuel. Brief comments on the results of the tests are also given but are not necessarily complete. Arrows indicate sequence testing with one fuel rod, dashed lines join interrupted sequences. Detailed results of tests performed during FY 1969 are available in References 1 through 10.

TABLE C-1

LISTING OF CDC FUEL TESTS AND RESULTS

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
8	20.3	CDC-3	26	No damage
9	16.6	CDC-3	28	"
10	14.4	sequence ↓	36	"
11	9.6		54	"
12	7.6		59	"
13	7.6		77	"
14	8.0	sequence ↓	77	Waterlogged-rod burst and expelled all fuel
15	8.4		78	Waterlogged-rod burst and expelled most fuel
16	7.9		76	Waterlogged - no rupture
17	7.9		73	Waterlogged-rod burst and expelled all fuel
18	7.5	sequence ↓	80	Two rods - 1 dry, 1 waterlogged. Waterlogged rod ruptured, caused 5000 psi pressure pulse and expelled fuel but did no damage to dry rod.
19	6.9		65	No damage
20	6.3		71	"
21	5.7		70	"
22	5.7	CDC-3	91	55 inch rod - no damage
23	5.1	sequence ↓	75	No damage
24	7.9		60	"
25	4.75		84	"
26	8.00		63	"
27	4.59	sequence ↓	93	"

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
29	4.3		97	No damage
30	6.87	CDC-3	73	"
31	4.04	sequence	103	"
32	6.16		84	"
33	5.74		90	"
34	~ 500		24	No damage (power calibration)
35	~ 500		72	" "
36	4.67		98	No damage
37	4.14	CDC-3	112	"
38	5.12	CDC-3	91	55 inch rod - No damage
39	4.59	sequence	100	"
40	4.54		99	"
41	4.59		98	"
42	4.61		99	"
43	4.75		101	"
44	4.65		98	"
45	4.65	CDC-3	100	No damage
46	4.73	sequence	97	"
47	4.70		101	"
48	4.65	CDC-3	96	55 inch rod - No damage
49	4.6	sequence	96	"
50	4.6		100	"
51	4.6		95	"
52	4.6		95	"
53	4.6		95	"
54	4.6		95	"
55	4.6	CDC-3	95	No damage
56	4.6	sequence	95	"
57	4.6		95	"
58	4.6		95	"
59	4.6		95	"
60	4.6		100	"

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)				Remarks
61	4.6	sequence ↓ CDC-3 ↓ sequence ↓ CDC-3 ↓ CDC-4 ↓ sequence ↓ CDC-3 ↓		95			No damage
62	4.6			95			"
63	4.6			95			"
64	4.6			95			"
65	4.6			95			"
66	4.6			95			"
67	4.6			95			"
68	4.6			95			"
69	4.6			95			"
70	4.6			95			"
71	4.6			95			"
72	4.6			95			"
73	4.6			95			"
74	4.6			95			"
75	4.6			95			"
76	4.6			120			"
77	4.6			120			"
78	4.6			120			"
79	5.0			115			"
80	5.0			115			"
81	5.0			112			"
82	4.6			92			"
83	4.6			92			"
84	8.1	NO	TEST	ROD	IN	CAPSULE	
85	4.6	"	"	"	"	"	
86	5.3	CDC-4		105			No damage
87	5.3	sequence ↓		105			"
88	5.3			105			"
89	5.3			101			"
90	4.64			126			Waterlogged rod failed
108	4.54	CDC-3		108			Waterlogged rod failed and fuel was expelled
109	4.6	CDC-3		97			No damage
110	4.6	↓		105			"

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
111	4.6	CDC-3	100	No damage
112	4.6	CDC-3	100	"
118	4.6	CDC-3	100	"
119	4.65	CDC-3	101	Waterlogged 6-inch rod failed and fuel was expelled
123	4.41	CDC-3	112	" " "
124	8.23	CDC-3	64	" " "
125	4.52	sequence	95	" " "
126	4.6		100	" " "
127	4.6		100	" " "
128	3.98	CDC-3	104	No damage
137	4.6	CDC-3	100	"
142	4.6	CDC-4	120	"
143	4.6	sequence	120	"
144	4.6		130	"
145	4.6		125	"
146	4.6		125	"
147	4.6	CDC-4	125	"
156	4.45	CDC-4	127	Rod bowed and colored light brown
157	4.37	sequence	157	Rod bowed and increased discoloration
158	4.44		125	Rod bowed and color became brown to dark blue
159	4.39		118	Run 159 through 166 rod bowed 3/8 inch and color darkened with each run, going from a light brown to dark blue
160	4.28		123	
161	3.93	sequence	134	
162	3.93		134	
163	3.9		130	
165	4.0		130	
166	3.96	F	134	
182	25		57	No damage
183	15		71.4	"
184	11.7		94	"
185	6.5		152.7	Slight bowing and discoloration

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
186	5.2	sequence	184.4	Magnitude of bowing increased
187	4.1		219.7	Bowing increased and a slight increase in diameter was measured
188	3.5		257.2	Bowing increased and integrity of cladding was lost
189	6.8	F	151	Slight bow and discoloration
190	5.2	F	182.4	3/16 inch bow and deep blue color
191	4.0	F	215.6	Little bow, rod showed ridges and discoloration
192	3.4	F	263	1/4-inch bow, surface texture was dulled
193	3.2	F	278.2	Little bow. Rod felt more lumpy than ridged. Clad integrity lost
194	32	F	53.7	No damage
195	20	sequence	69.9	"
196	15.5		70.7	0.004 inch diameter growth
197	12.4		88.5	0.005 " " "
198	11.3		100.8	0.007 " " " and rod slightly bowed
199	9.1		118.9	0.018 inch growth and bow increased
200	7.7		135.7	0.001 inch growth and bow increased
201	6.7	sequence	155	0.029 inch growth and bow increased significantly
202	6.8		152.7	Waterlogged - slight bow, 0.020 inch growth
203	2.9	F	298.8	Waterlogged rod discolored and a portion appeared to have metal-water reaction. Rod failed
205	4.9	F	180.6	Waterlogged rod bowed and discolored. 0.031 inch growth
206	4.2	F	22.5	Waterlogged - no bowing. 0.029 inch growth
207	3.4	PBF	215	Tests 207 through 216. These ten tests were a sequential series on a PBF prototype fuel rod
208		sequence		
209				

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
210				(#T-18) which had previously been subjected to a ten transient series in the TREAT Reactor. Energy release for each of the tests was about ~ 215 cal/gram fuel (1480 cal/cc). The rod showed as a result of these ten tests: (1) an axial growth of 70 mils, (2) diametrical growth of ~ 4 mils, (3) slightly increased bowing, (4) separation between insulator sleeves, but (5) no indication of sleeve breakdown, fuel displacement into cracks, or development of hot spots
211				
212				
213				
214				
215				
216				
		sequence		
217	3.45	F	256.8	Waterlogged-rod bowed and discolored. 0.055 inch growth
218	3.2	F	283.3	Waterlogged-rod ruptured, 75% fuel expelled. 0.062 inch growth
229	9.1	F	103	No damage
230	9.1		103	"
231	3.6	F	251	Large bow (about 1.25 inches)
232	3.6	F	264	
233	3.6		251	Test 232 to 236 showed a slight bow after first test and progressed to 3/4 inch after fifth test
234	3.6		251	
235	3.6		251	
236	3.6		251	
237	3.8		237	
238	3.8		237	Test 237 to 240 - No bowing after first test bowing occurred on succeeding tests, reaching a maximum of about 1/2 inch
239	3.8		237	
240	3.8		237	
241	4.35		208	
242	4.35		208	Test 241 to 245 - Rod was bowed and discolored in first test and these conditions became progressively worse, reaching a maximum of about 1 inch offset after fifth test
243	4.35		208	
244	4.35		208	
245	4.35		208	
246	4.3	F	214	Very slight bow - cumulative axial growth 0.129
247	4.3		214	Slight bow - 0.389

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
248	4.3	↓	214	1/8 inch bow - 0.577
249	4.3		214	1/4 inch bow - 0.861
250	4.3		214	3/4 inch bow - 0.970
251	4.3		214	1/4 inch bow - 1.142
252	3.4	PBF	250	<p>Tests 252 through 280 (29 tests) - A prototype PBF rod (T-18) - previously subjected to ten transients in the TREAT reactor and ten transients in the CDC (transient #'s 207 through 216) - was subjected to 29 additional transients each with a period of 3.4 msec to establish if any deleterious effects in this rod type result from repeated short-period series such as it would experience in the PBF itself. The average radial energy deposition at the axial flux peak was ~250 cal/g of UO₂ for an average radial fuel temperature of ~1920°C. As a result of these transients, there appeared to be: (1) no change in the gap size, (2) no deterioration of the insulator sleeves, and (3) a total growth in rod length of 0.156".</p>
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TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
281	4.3	F	200	Increased bowing - cumulative axial growth 1.269
282	4.3		200	" " " 1.442
283	4.3		200	" " " 1.584
284	4.3		200	" " " 1.761
285	4.3		200	" " " 1.893
286	4.3		200	" " " 2.017
287	4.3		200	" " " 2.155
288	11.0	F	100	Tests 288 to 301 - No discoloration and a slight bow was noted during these 14 tests. There was 0.260 inches of axial growth
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300				
301				
302	6.8	F	150	Slight coloring, axial growth to 152 mils, slight bowing
303	6.8		150	
304	6.8		150	
307	-	SPX-3%-SS	-	
308	-	SPX-10%-SS	-	Figure-of-merit measurements
309	3.4	PBF(T-18)	250	
310	3.4		250	A continuation of testing on the PBF (T-18) rod. During this series of 9 tests, the rod grew about 102 mils in length to a total growth of 446 mils. No apparent damage occurred
311	3.4		250	
312	3.4		250	
313	3.4		250	

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
314	3.4	↓	250	
315	3.4		250	
316	3.4		250	
317	3.4		250	
318		PBF(6 5/8" stack)		200 KW for 5 minutes
319	4.2	SPX-3%-SS	200	Axial growth of 177 mils. Badly discolored, slight bow
320	6.8	↓	150	Axial growth continued to 263 mils, bowing slight
321	6.8		150	
322	3.4	SPX-3%-SS	250	axial growth of 258 mils, badly discolored, slight bow
323	6.2	SPX-10%-SS	275	Clad melted but no fragmentation, no pressure, 1% M-W reaction
324	6.8	↓	150	Axial growth continued to 365 mils, bowing slight to about 1/8-inch off-center
325	6.8		150	
326	4.75	SPX-10%-SS	375	Clad melted, fuel rod breakup 100 psi pressure pulse, 6% M-W reaction.
327	3.75	SPX-10%-SS	475	Rod fragmented, clad melted, 300 psi pressure pulse, 14% M-W reaction.
328	3.15	SPX-10%-SS	570	Rod fragmented before clad melting, 600 psi pressure pulse, 36% M-W reaction
329	3.0	SPX-3%-SS	275	Clad melted, no fragmentation, no pressure, 1% M-W reaction

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
330	Periods range between 3.0 and 5.6 msec with most between 3.0 and 3.4 msec.	PBF (T-1-F)	Peak energy disposition for the series was about 280 cal/g of fuel mixture.	Fuel mixture is 30.6 wt.% urania, 62.3 wt.% zirconia, and 7.1 wt.% calcia. Pellet breakup observed. Dynamic axial growth during each test was between 60 and 80 mils. Total residual rod growth was about 140 mils.
429				
430	----	SPX-10%-Zr	---	Figure-of-merit measurement.
431	4.30	SPX-3%-Zr	197	Axial growth of 323 mils. Cladding discolored and rod bowed. 1.8% M-W reaction.
432	3.60	SPX-3%-Zr	239	Rod fractured, cladding discolored and rod bowed. 5.3% M-W reaction.
433	3.15	SPX-3%-Zr	269	Rod separated, melting and deformation of the cladding occurred. 8.1% M-W reaction.
434	6.73	SPX-10%-Zr	296	Rod separated in two locations. Melting and extensive damage of cladding. 12.1% M-W reaction.
435	----	SPX-10%-Zr in center of 5x5 Array of Aluminum Tubes	---	Figure-of-merit measurement.
436	----	SPX-3%-Zr	---	Figure-of-merit measurement.
437	5.02	SPX-10%-Zr	388	Extensive fragmentation into large pieces. 28.7% M-W reaction.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
438	4.09	SPX-10%-Zr	469	Extensive fragmentation. 60.4% M-W reaction.
439	3.43	SPX-10%-Zr	569	Extensive fragmentation. Cladding temperature $\approx 580^{\circ}\text{C}$ at time of cladding failure. 100% M-W reaction. 120 psi pressure pulse.
440	3.27	PBF-T24	245	Test used to crack fuel pellets prior to waterlogging.
443	3.13	Waterlogged PBF-T24	266	Waterlogged rod failed at fuel temperature of 2180°C . Maximum capsule pressure was 2350 psi. About 2/3 of fuel expelled.
446	3.33	PBF-T25	240	Test used to crack fuel pellets prior to waterlogging.
447	3.41	Waterlogged PBF-T25	245	Waterlogged rod failed at fuel temperature of 1920°C . Maximum pressure was 2675 psi. About 2/3 of fuel expelled.
448	4.09	PBF-T26	205	Test used to crack fuel pellets prior to waterlogging.
449	3.31	PBF-T26	240	Test used to further crack fuel pellets prior to waterlogging.
451	5.10	Waterlogged PBF-T26	160	No cladding rupture of waterlogged rod. Fuel temperature reached 1280°C . 52 mil maximum diametral growth.
452	4.18	SPX-10%-Zr in 5x5 Array in Aluminum Tubes	391	Active fuel length fragmented. 48.9% M-W reaction.
453	4.13	Waterlogged PBF-T26	205	Rod failed at fuel temperature of 1600°C . Maximum capsule pressure of about 2000 psi. About 2/3 of fuel expelled.
454	6.91	SPX-10%-Zr	286	Rod separated, partial melting of the cladding over about $\frac{1}{2}$ of active fuel length. 10.9% M-W reaction.
455	5.09	SPX-10%-Zr	388	Extensive fragmentation. 30.4% M-W reaction.
456	4.00	SPX-10%-Zr	498	Extensive fragmentation. 57.6% M-W reaction.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
457	3.30	SPX-10%-Zr	590	Extensive fragmentation. Cladding surface temperature of 280°C at time of failure. 400 psi pressure pulse. 96.7% M-W reaction.
458	3.35	SPX-10%-Zr in center of 5x5 Array of Aluminum Tubes.	471	Extensive fragmentation. 500°C cladding temperature at time of failure. 67.2% M-W reaction. 215 psi pressure pulse.
459	6.0	PBF-T27	145	Test used to crack fuel pellets prior to waterlogging.
460	4.05	PBF-T27	205	Test used to further crack fuel pellets prior to waterlogging.
461	3.05	PBF-T27	275	Test used to further crack fuel pellets prior to waterlogging.
462	6.11	PBF-T27	140	Test used to further crack fuel pellets prior to waterlogging.
463	4.04	PBF-T27	205	Test used to further crack fuel pellets prior to waterlogging.
464	3.01	PBF-T27	280	Test used to further crack fuel pellets prior to waterlogging.
465	4.24	Waterlogged PBF-T27	195	The waterlogged rod did not fail. Cladding exhibited maximum diametral growth of 63 mils.
466	----	GEX-7%-Pellet	---	Figure-of-merit measurement.
467	----	GEP-7%-Pellet	---	Figure-of-merit measurement.
468	4.09	Waterlogged PBF-T27	205	Waterlogged rod failed. Maximum capsule pressure was 3275 psi. About 2/3 of fuel was expelled.
471	6.44	GEX-7%-Pellet	194	18 mils permanent axial growth. Cladding surface blackened and slightly bowed. 1.3% M-W reaction.
472	6.65	GEX-7%-Powder	200	11 mils permanent axial growth. Cladding discolored and slightly bowed. 0.6% M-W reaction.
476	4.80	GEX-7%-Pellet	257	Rod separated and several regions of melted cladding. 8.8% M-W reaction.
477	5.02	GEX-7%-Powder	255	23 mils permanent axial growth. Only slight discoloration of cladding. 3.2% M-W reaction.
478	3.66	GEX-7%-Pellet	342	Gross melting of fuel and cladding. 1.5 inches of the rod fragmented. 19% M-W reaction. 375 psi pressure pulse measured.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
479	2.95	GEX-7%-Powder	414	Three inch length of rod fragmented. 720 psi pressure pulse measured. 25.9% M-W reaction.
480	3.88	GEX-7%-Powder	342	Four inch length of rod melted and fragmented. 50 psi pressure pulse. 23% M-W reaction.
481	2.97	Unclad 10% SPX Pellets in Alumina Crucible.	605	Maximum capsule pressure was 410 psi. 13.1 cc of hydrogen gas released per gram of UO_2 .
483	2.98	GEX-7%-Powder	440	Entire active length melted and fragmented. 400 psi pressure pulse. 27.1% M-W reaction.
484	3.76	GEP-7%-Pellet	194	22% increase in diameter. Permanent axial growth of 10 mils. 0.4% M-W reaction.
485	3.99	GEP-7%-Powder	193	Rod sustained an S-shaped bow. Cladding discolored. 1.0% M-W reaction.
487	3.00	GEP-7%-Pellet	243	Cladding failed at 205 cal/g. 39 psi capsule pressure pulse. Rod remained intact. 6.0% M-W reaction.
488	2.98	GEP-7%-Powder	254	Permanent axial growth of 46 mils. Several hairline cracks in cladding. Mean diametral growth of 19 mils was measured. 7.0% M-W reaction.
489	3.84	GEP-7%-Pellet	201	Mean diametral growth of 17 mils. Permanent axial growth of 19 mils. Rod slightly discolored and slightly bowed. 0.7% M-W reaction.
490	5.26	GEP-5%-Pellet	123	Fuel pellets cracked. Permanent axial growth of 7 mils.
491	2.96	GEX-7%-Pellet 24 inch active fuel length	427	Rod failed at 333 cal/g. Maximum capsule pressure was 715 psi. Extensive fragmentation occurred. 22% M-W reaction.
492	3.05	GEX-7%-Powder 24 inch active fuel length	460	Rod failed at 391 cal/g. Maximum capsule pressure was 575 psi. Rod was extensively fragmented. 24% M-W reaction.
494	3.07	Waterlogged GEP-7%-Powder	265	Waterlogged rod failed at 66 cal/g of UO_2 . All fuel was expelled from the rod.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
496	2.96	Waterlogged GEP-5%-Pellet	220	Waterlogged rod failed at 60 cal/g of UO_2 . About 90% of the fuel was expelled from the rod.
501	5.17	GEP-5%-Pellet	120	Fuel pellets cracked. Cladding surface did not sustain any noticeable damage.
503	7.60	SPXM-10%-Zr	257	Permanent axial growth of 73 mils. Very little cladding strength. Cladding discolored and deformed. 15.5% M-W reaction.
504	6.86	SPXM-10%-Zr	287	Regions of melted fuel and cladding. Several holes formed in cladding. Rod fractured in four locations. 21.5% M-W reaction.
505	3.27	SPXM-10%-Zr	613	Cladding failed at 470 cal/g. Extensive fragmentation. 1.35% energy conversion. All of the cladding reacted with water.
506	4.93	SPXM-10%-Zr	378	Cladding failed at 365 cal/g. 65 psi capsule pressure pulse. Most of active fuel region was melted and fragmented. 37% M-W reaction.
507	4.10	SPXM-10%-Zr	490	Cladding failed at 400 cal/g. 180 psi pressure pulse. Extensive melting and fragmentation. 51.7% M-W reaction.
509	3.02	GEP-7%-Powder 24 inch active fuel length	251	Most of energy was deposited at time of failure. 36 psi maximum pressure pulse. Melted cladding at flux peak. Cladding discolored and deformed. 3% M-W reaction.
510	11.7	SPXM-10%-Zr	168	Permanent axial growth of 38 mils. Cladding surface was blackened and dulled and rod was bowed slightly.
511	3.03	Waterlogged GEP-7%-Powder	265	Waterlogged rod failed at 60 cal/g. 3300 psi capsule pressure pulse. All of the fuel was expelled from rod.
514	2.98	Waterlogged GEP-5%-Pellet	220	Waterlogged rod failed at 33 cal/g. Capsule pressure pulse of 1680 psi. All fuel was expelled.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
517	3.11	SPX-10%-Pellets clad with Flint Glass Tubing	552	Failure at 388 cal/g. Maximum capsule pressure was 310 psi. 14.3 cc of hydrogen gas released per gram of UO_2 .
518	5.13	GEX-7%-Powder	262	Mean diametral growth of 5.9 mils. Permanent axial growth of 20 mils. Slight discoloration and very slight axial bow. 1.5% M-W reaction.
519	4.62	GEX-7%-Powder	292	Cladding temperatures of 2025°C. Melted cladding over active fuel length. Rod fractured. No pressure. 14.6% M-W reaction.
520	4.94	GEX-7%-Powder	266	Permanent axial growth of 10 mils. Cladding surface was discolored. 2% M-W reaction.
521	2.98	Waterlogged F Rod	300	Waterlogged rod failed at 245 cal/g. 1850 psi capsule pressure pulse. About half of the fuel was expelled.
522	8.60	F	100	Fuel pellets cracked.
523	8.53	GEX-7%-Powder	151	Maximum cladding temperature of 900°C. Cladding surface was discolored and rod was slightly bowed.
524	3.93	GEX-7%-Powder	343	Rod failed at 330 cal/g. Maximum cladding temperature of 1775°C. Melted cladding and complete separation of cladding at two points. 18.9% M-W reaction.
525	4.22	GEX-7%-Powder	314	Cladding failed at 290 cal/g. 36 psi pressure pulse. Maximum cladding temperature of 1875°C. Extensive melting. 17.4% M-W reaction.
526	3.84	SPX-10%-Pellets clad with Flint Glass Tubing	452	Failure at 414 cal/g. Maximum capsule pressure was 40 psi. 11.1 cc of hydrogen gas released per gram of UO_2 .
527	2.98	Waterlogged F Rod	304	Waterlogged rod failed at 240 cal/g. 2200 psi capsule pressure pulse. About 67% of the fuel was expelled from the rod.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
529	4.29	SPX-10%-Pellets clad with Flint Glass Tubing	412	Failure at 376 cal/g. Maximum capsule pressure was 185 psi. 10.5 cc of hydrogen gas released per gram of UO_2 .
531	4.56	SPX-10%-Pellets clad with Flint Glass Tubing	389	Failure at 310 cal/g. Maximum capsule pressure was 87 psi. 9.5 cc of hydrogen gas released per gram of UO_2 .
532	5.07	SPX-10%-Pellets clad with Flint Glass Tubing	328	Failure time unknown. No pressure generation. 5.3 cc of hydrogen gas released per gram of UO_2 .
533	3.42	SPX-10%-Pellets clad with Flint Glass Tubing	508	Failure at 420 cal/g. Maximum capsule pressure was 830 psi, 12.5 cc of hydrogen gas released per gram of UO_2 .
534	12.1	SPX-10%-Pellets clad with Flint Glass Tubing	153	Figure-of-merit experiment.
536	3.28	SPXM-10%-Zr	620	Cladding failure at 420 cal/g. 500°C cladding temperature at time of failure. 380 psi pressure pulse. 1.44% energy conversion. Essentially all of the cladding had undergone M-W reaction.
537	8.25	SPXM-10%-Zr	235	102 mil permanent axial growth. Cladding surface was discolored and bowed slightly. 6.6% M-W reaction.
538	3.03	Waterlogged SPXM-5%-SS	389	Waterlogged rod failed at 350 cal/g. Maximum capsule pressure was 1275 psi. All of the fuel was expelled.
539	8.11	SPXM-10%-Zr	240	Cladding temperature of 1700°C. Permanent axial growth of 169 mils. Cladding surface discolored and deformed. 14.7% M-W reaction.
540	3.05	Waterlogged SPXM-5%-SS	393	Waterlogged rod failed at 255 cal/g. Maximum capsule pressure was 485 psi.
541	5.59	SPXM-10%-Zr	338	Cladding temperature of 1880°C. Rod broken into several pieces with extensive melting and slight fragmentation. 22.2% M-W reaction.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
543	5.20	SPXM-10%-Zr	360	Cladding failed at 330 cal/g. Cladding temperature was 1440°C at time of failure. Extensive fragmentation. 23.9% M-W reaction.
544	2.99	Waterlogged SPXM-5%-SS	392	Waterlogged rod failed at 116 cal/g. Maximum capsule pressure was 330 psi. All of the fuel was expelled.
545	7.26	SPXM-10%-Zr	267	Maximum cladding temperature of 1860°C. Partial melting and cracking of cladding. Rod fractured in three places. 17.1% M-W reaction.
546	3.03	Waterlogged SPXM-5%-SS	193	Waterlogged rod failed at 352 cal/g. Maximum capsule pressure was psi. All of the fuel was expelled.
547	2.97	Waterlogged SPXM-5%-Zr	410	Waterlogged rod failed at 85 cal/g. Capsule pressure pulse of 130 psi. All of the fuel was expelled from the rod.
548	----	SPX-10%-Pellets clad with Flint Glass Tubing	---	Figure-of-merit measurement.
549	3.01	SPXM-10%-Zr	655	Cladding failure at 360 cal/g. Cladding temperature of 300°C at time of failure. 1750 psi capsule pressure pulse. 2.75% energy conversion ratio. Essentially all of the cladding had undergone a metal-water reaction.
550	3.00	SPX-10%-Pellets clad with Flint Glass Tubing	509	Failure at 374 cal/g. Maximum capsule pressure was 1510 psi. 13 cc of hydrogen gas released per gram of UO ₂ .
551	2.99	SPXM-10%-Zr	653	Cladding failure at 400 cal/g. Cladding temperature at 300°C at time of failure. 1600 psi capsule pressure. 2.2% energy conversion. Essentially all cladding reacted with water.
552	3.01	Waterlogged SPXM-5%-Zr	412	Waterlogged rod failed at 164 cal/g. Capsule pressure pulse of 920 psi. All of the fuel was expelled.

TABLE C-1 (Continued)

CDC Transient Number	Period (msec)	Test Rod Type	Total Energy Deposition (cal/g)	Remarks
553	5.65	SPXM-5%-SS	207	Transient figure-of-merit measurement. Cladding discolored and rod slightly bowed.
555	5.50	Waterlogged SPXM-5%-Zr	223	Waterlogged rod failed at 159 cal/g. Capsule pressure pulse of 325 psi. Less than 1 gram of fuel was expelled from the rod.
556	5.60	Waterlogged SPXM-5%-Zr	225	Waterlogged rod failed at 150 cal/g. Capsule pressure pulse of 450 psi. Less than 1 gram of fuel was expelled from the rod.
558	5.49	Waterlogged SPXM-5%-Zr	224	Waterlogged rod failed at 157 cal/g. Capsule pressure pulse of 380 psi. Less than 1 gram of fuel was expelled from the rod.
559	5.54	Waterlogged SPXM-5%-Zr	223	Waterlogged rod failed at 121 cal/g. Capsule pressure pulse of 330 psi. Less than 1 gram of fuel expelled.
562	5.69	GEXPR-T2 $\text{PuO}_2\text{-UO}_2$	274	Cladding failure occurred. Cladding was partially melted and rod was separated into 2 major pieces. Maximum cladding temperature was 1910°C. 13% M-W reaction.
566	7.37	GEXPR-T1 $\text{PuO}_2\text{-UO}_2$	225	Maximum cladding temperature was 1400°C. Permanent axial growth of 63 mils measured. Cladding surface discolored.
567	4.53	Irradiated GEX-7%-Pellet	260	Rod irradiated to 3330 MWd/MTM in ETR. Rod failed at 225 cal/g. 300 psi capsule pressure pulse. 5.1% M-W reaction.
568	6.06	Irradiated GEX-7%-Pellet	200	Rod failed at 147 cal/g. Maximum capsule pressure was 180 psi. Cladding blistered and cracked. 1.7% M-W reaction.
569	3.55	Irradiated GEX-7%-Pellet	340	Rod failed at 300 cal/g. Maximum pressure was 2400 psi. Melted cladding and hole formed in small area.