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DESIGN, EQUIPMENT, AND PROGRAM FOR FUEL ROD BURST EXPERIMENTS

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ABSTRACT

Irradiated Zircaloy-clad fuel rods will be rupture-tested to study the effects of irradiation on the failure characteristics of the cladding. A system has been constructed in a hot cell, and burst experiments on both unirradiated and irradiated fuel rods are being conducted under conditions predicted for a Loss-of-Coolant Accident in a water cooled power reactor.

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CONTENTS

		Page
Abstract		1
I.	Introduction	5
II.	Experimental Equipment and Conditions	6
III.	Post-Test Analyses	18
IV.	Rupture Test Program	19
V.	Summary and Conclusions	22
References		24

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I. INTRODUCTION

Studies relating to the safety of current nuclear reactor power plants are being conducted at the Oak Ridge National Laboratory (ORNL) under the auspices of the ORNL Nuclear Safety Research and Development Program. A Loss-of-Coolant Accident (LOCA) has been postulated, wherein the water coolant would be lost as a result of a major pipe break. Although it is assumed that the reactor would be shut down, the stored energy and fission product decay heat in the fuel rods would be sufficient to cause rapid temperature increase and, in the absence of emergency cooling, severe damage to the core. While all reactors are equipped with Emergency Core Cooling Systems (ECCS), the capacity and effectiveness of these systems to control the thermal transient in the larger reactors now being constructed has been questioned. Consequently, several different experiments have been devised to investigate the behavior of Zircaloy clad UO_2 fuel rods during a rapid thermal transient in a steam atmosphere.[1,2] This report discusses the experimental equipment and techniques used (1) to burst test irradiated fuel rods in a hot cell, and (2) to determine the mode of cladding failure and the extent of fission product release.

II. EXPERIMENTAL EQUIPMENT AND CONDITIONS

The experimental equipment was designed to accommodate segments of fuel rods from current power reactors, and also short test capsules. The test rods will be either 29 or 41 inches long, and will be 0.565 or 0.424 inches in diameter, thus representing both Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) dimensions. Zircaloy clad rods of both diameters[3] which have been irradiated to 3,000 to 10,000 Mwd/T[4] have been acquired. The construction and dimensions of a typical rod are illustrated in Fig. 1.

These older rods (vintage about 1960) have two obsolete characteristics which will require considerable remote modification to assure meaningful test results. The rods have very small plenums compared to newer designs (~0.5 in. for 27 in. UO_2 vs about 12 in. for 144 in. UO_2), and the irradiation conditions (temperature and burnup) experienced by these rods were too low to cause appreciable gas release from the UO_2.[5] The modification procedure includes three remote operations.

1. A small hole (~0.05 in.) is drilled through one end plug, and where desired, the contained gas may be collected and analyzed for ^{85}Kr (see Fig. 2).
2. A pressure lead of 1/8 in. diameter tubing is attached to the fuel rod, either by welding or with a compression fitting. In some cases

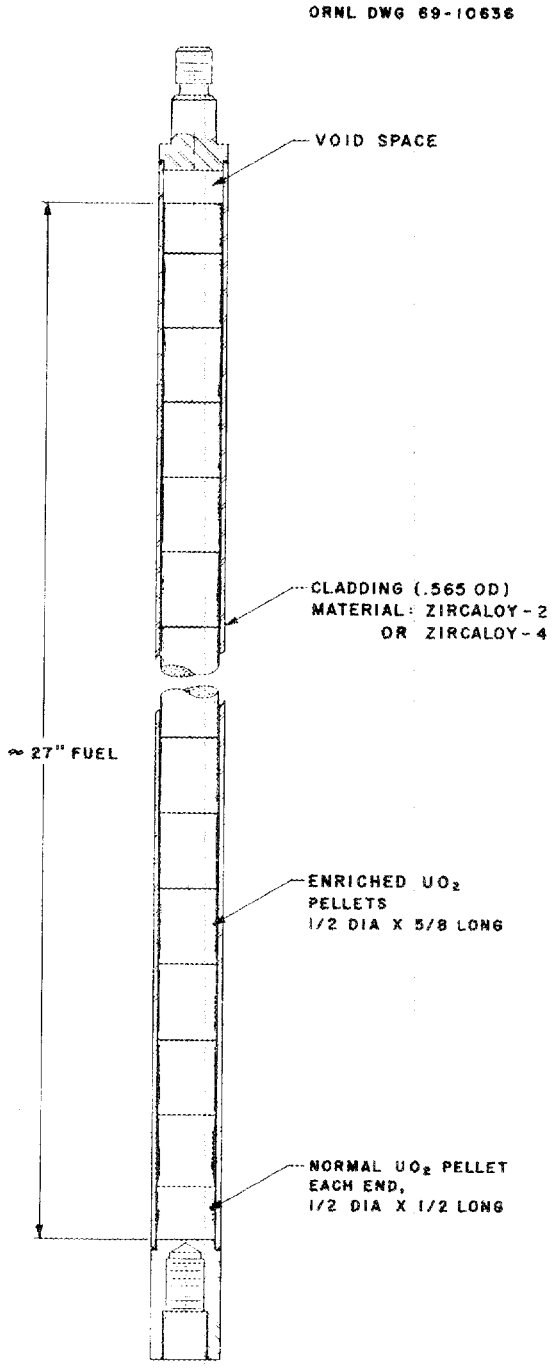

Fig. 1. Typical Fuel Rod.

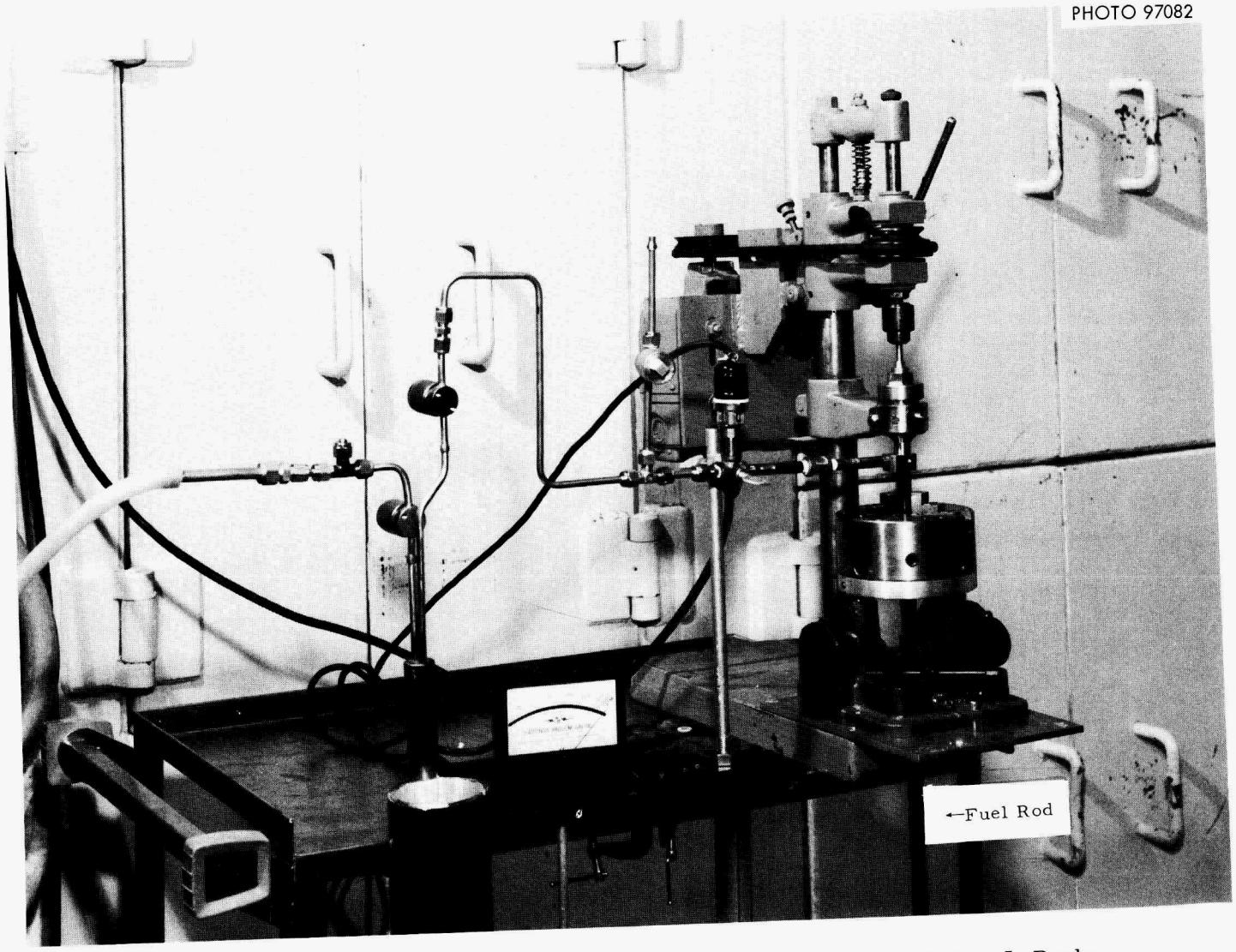

Fig. 2. Apparatus For Drilling Irradiated Fuel Rods and Collecting the Contained Gas.

additional plenum volume is attached; such an assembly is illustrated in Fig. 3. The remote welder, which is used to weld highly radioactive rods in an inert atmosphere is shown in Fig. 4.

3. The entire assembly is pressurized with helium and sealed via a previously prepared 1/8 in. diameter tube, which is shown in Fig. 3. Pressures representative of current high performance fuel are installed; these are 50 and 200 psia for BWR, and up to 500 psia for PWR type rods.[6]

We feel that the presence of a plenum containing an appreciable reservoir of gas (several times that volume dispersed along the length of the rod) could significantly affect both the rupture characteristics of the cladding and the gross release of fission products. For instance, a small failure will quickly relieve the pressure from a rod with a small gas volume, whereas a larger volume might cause the failure to propagate and become much larger, possibly to the point of dumping out fuel. Also, the larger volume of gas rushing along the fuel rod from the plenum to the rupture is likely to sweep out particulate matter (fuel and fission products) proportionate to the volume flow. We plan to investigate the resistance to gas flow along irradiated rods by making flow vs pressure measurements both before and after rupture testing.

The modified fuel rod is mounted vertically in concentric quartz tubes within an induction coil, as shown schematically in Fig. 5. The lengths of the interchangeable quartz

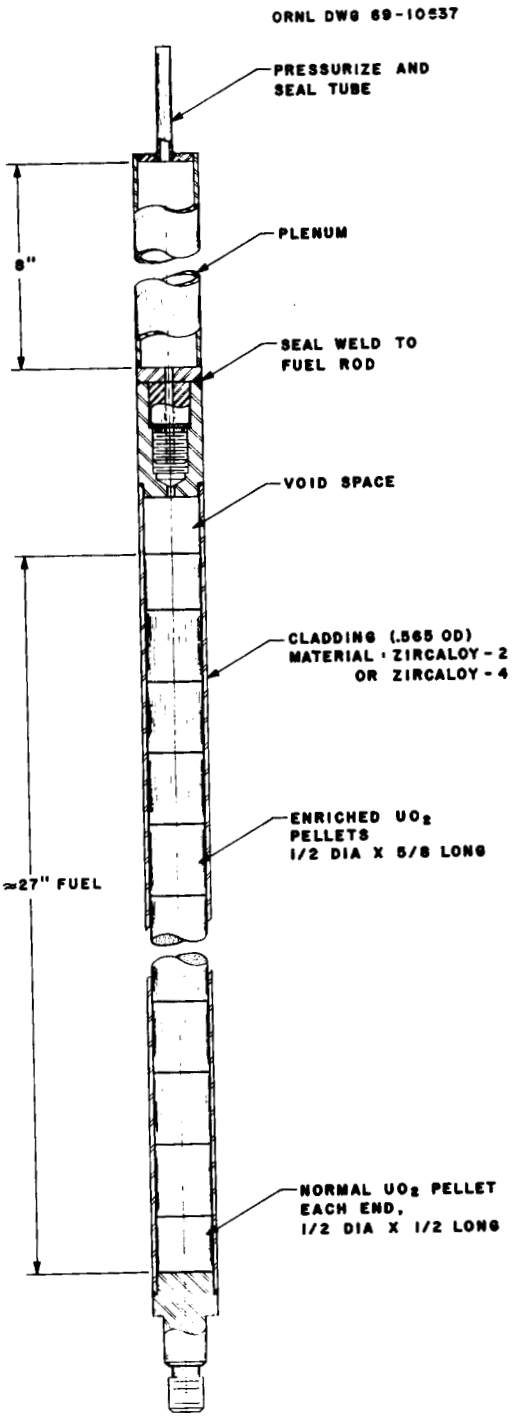

Fig. 3. Fuel Rod With Plenum Added.

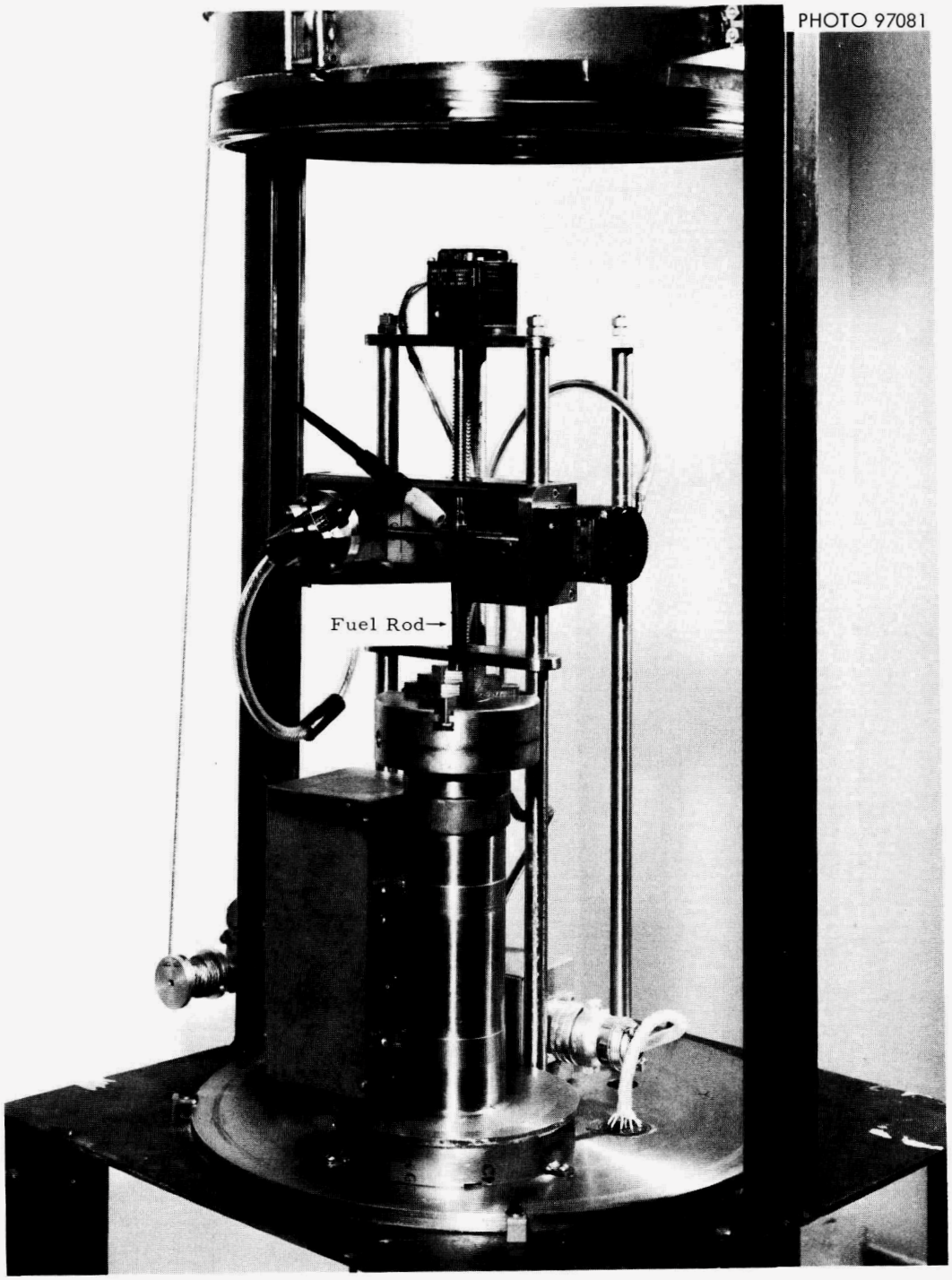

Fig. 4. Remote Inert Gas Welder, With Bell Jar Raised.

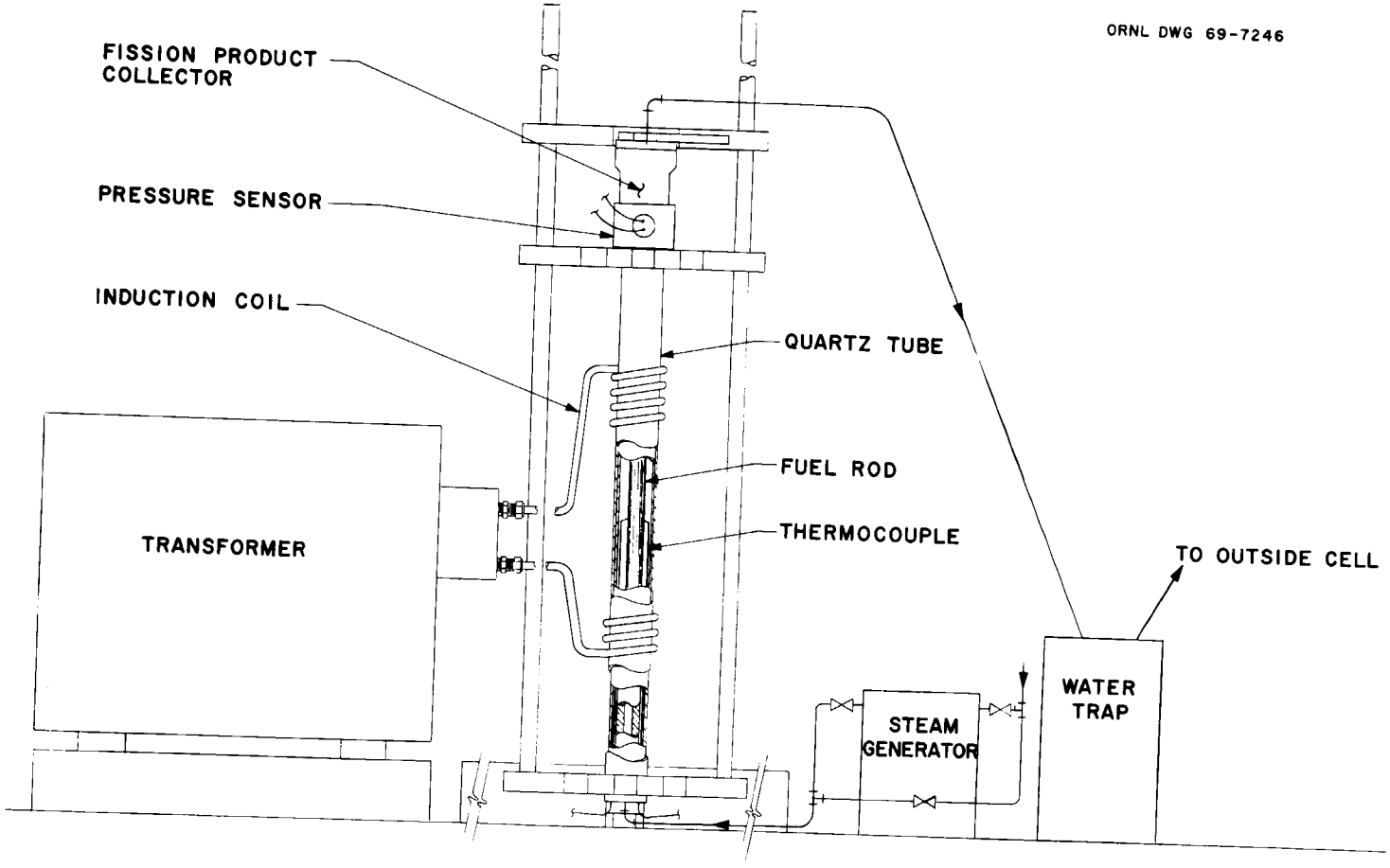

Fig. 5. Fuel Rod Burst Experiment.

containment tubes and copper induction coils are varied to match the fuel rod. The test atmosphere is a flowing mixture of carrier gas (argon + 10% nitrogen) and steam. A small fraction of nitrogen in the carrier gas is necessary to suppress electrical discharge from the induction coil to the fuel rod, a problem which increases with the volatage applied to the induction coil. Although the embrittlement of Zircaloy by nitrogen at high temperature is a well known reaction, the very short time (~30 seconds) at reaction temperatures should minimize any effects of nitrogen reaction in these tests. Steam is generated by passing the carrier gas through water in a controlled temperature reservoir. The steam flow rate is sufficient to allow unlimited reaction with the Zircaloy cladding up to the expected rupture temperature, which will be 1400 to 2300°F, depending on the internal pressure.

The source of high frequency (~400 kc) power is a radio frequency generator (Lepel Model No. T-50-3_4S), which is mounted outside the hot cell. Power is transmitted to the in-cell assembly by 1/2 in. diameter water cooled copper tubes. The system is capable of a wide range of rod heating rates and maximum temperatures, up to ~100°F/sec and >2500°F. Tests at heating rates of 20, 50 and 100°F/sec are currently planned.

Fuel rod temperature during the test is monitored by at least two Pt vs Pt-10% Rh thermocouples attached to the cladding. A pressure sensor is mounted in the fission product

collector immediately downstream from the fuel rod to detect the pressure surge at the instant of cladding rupture. This sensor transmits a signal to a two-pen recorder (Hewlett-Packard No. 7100B) which simultaneously records the temperature signal from a thermocouple, thereby permitting accurate determination of the rupture temperature.

The fission product collector assembly is a Teflon cylinder (Fig. 6) containing high efficiency fiber glass particle filters and activated charcoal for removing such volatile species as iodine, cesium, tellurium, and ruthenium from the gas stream. It is heated to 150°C to prevent moisture condensation. Traps cooled by water ice (0°C) and dry ice (-78°C) remove the water vapor from the gas stream prior to collection of the rare gases — ^{85}Kr and stable Kr and Xe — in a dry ice cooled charcoal trap. (See system flow diagram, Fig. 5.) A photograph of the in-cell equipment is shown in Fig. 7.

A series of preliminary tests were run to determine (1) the optimum coil configuration, (2) the optimum generator settings, and (3) the effects of various operating conditions on the accuracy of the cladding thermocouples. A single coil, 18 turns over a nine inch length, connected to 4 to 1 step-down transformer proved to have adequate capacity for heating an eight inch long capsule at rates up to 200°F/sec. Comparison of the output from thermocouples on both the inside and the outside surfaces of the cladding with an optical

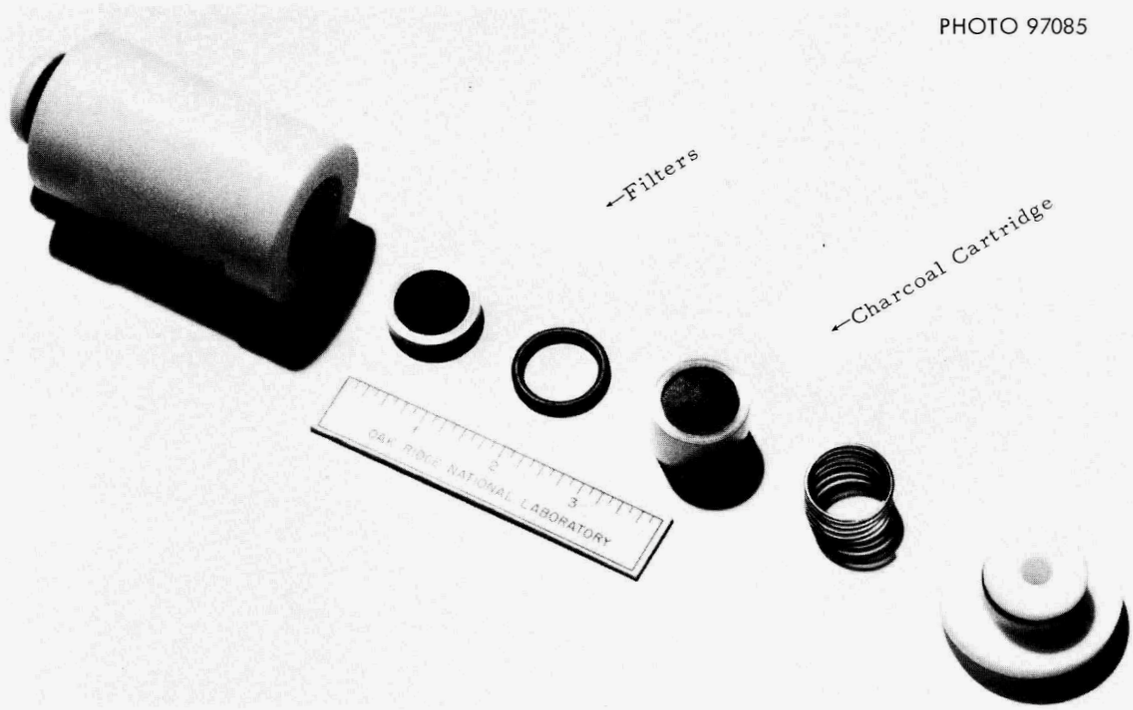

Fig. 6. Components of Fission Product Collector.

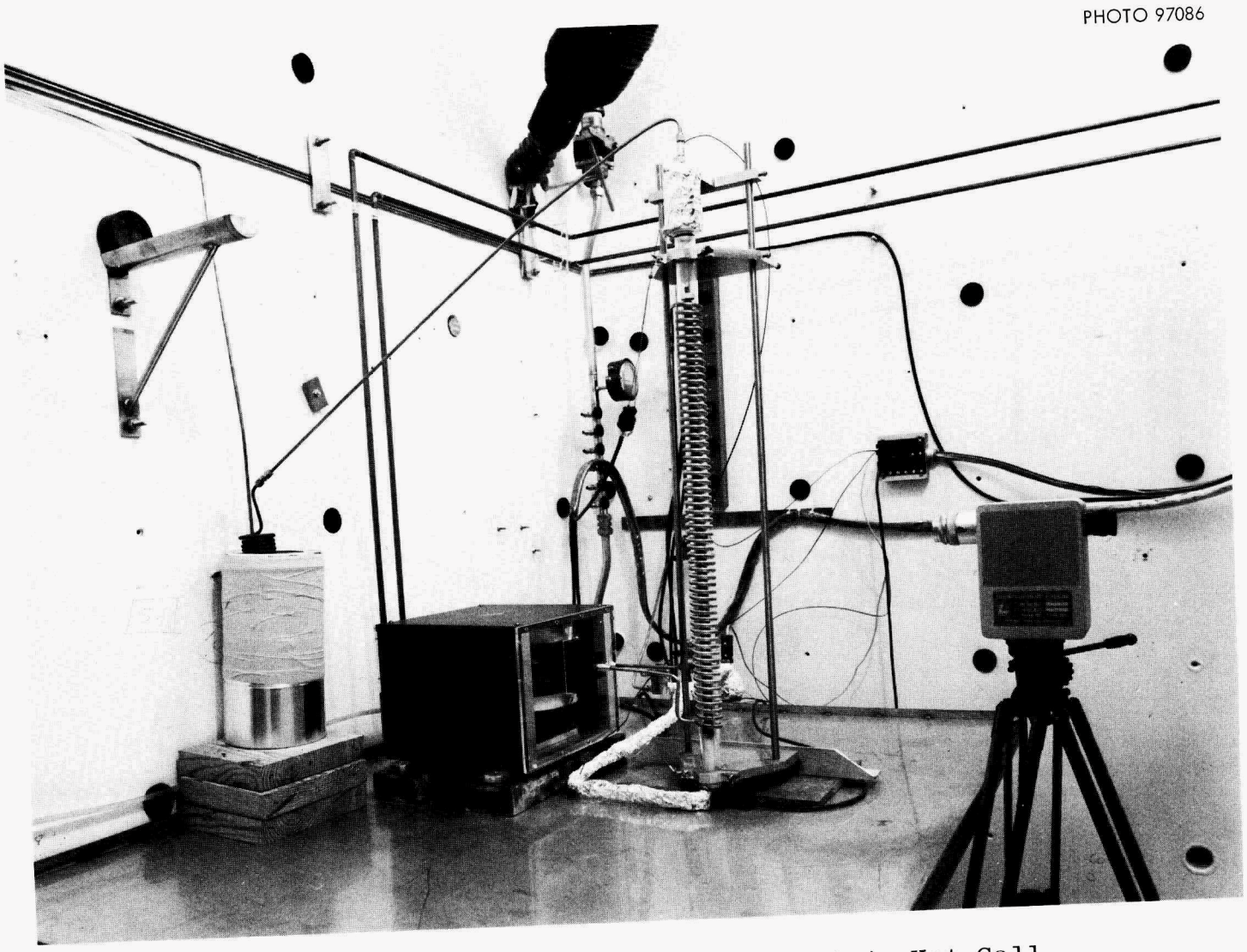

Fig. 7. View of Rod Burst Equipment in Hot Cell.

pyrometer showed induction heating had no detectable effect on the inside thermocouple and negligible effect on the outside thermocouple at the lower voltages, i.e., with the stepdown transformer.

Rapid, uniform heating of a 29-inch long rod proved to be somewhat more difficult. Several coil configurations, at both high and low voltages, were tested. As noted previously, ionization of the inert carrier gas limited the heating rates in argon and helium, so a mixture of argon and nitrogen was tested[7] and found to allow more rapid heating. In addition, the use of high coil voltages caused intolerable heating in the thermocouples welded to the cladding surface. A three section coil, connected in parallel to the stepdown transformer, proved to be the best combination; typical generator settings and heating rates are shown in Table 1.

Table 1. Approximate RF Generator Settings for Fuel Rod Rupture Tests
(Three Section Coil, Stepdown Transformer, in Hot Cell)

Heat Rate (°F/sec)	Saturable Reactor (%)	Thyratron (%)	RF Current (amps)
20	0	45	150
50	15	50	230
100	25	55	280

III. POST-TEST ANALYSES

After fuel rod rupture, steam flow is terminated and the system is swept with argon until it has cooled to ambient temperature. The fission product collector is removed and placed in a plastic sack to guard against cross-contamination. Then the fuel rod may be removed from the system and inspected visually. Areas of particular interest are photographed and rod diameters are measured in two perpendicular planes at regular intervals, so that swelling can be charted. Depending on the general appearance, samples may be cut from the rupture areas for metallographic examination to study the cladding microstructure. The mode of failure, extent of swelling, and microstructural effects are compared as functions of irradiation exposure (burnup) and rupture test conditions.

The filters and hot and cold charcoal collectors are analyzed radiochemically to determine the amount and species of fission products released. The fission product inventory in the fuel-cladding gap is estimated from the irradiation history and previous release from UO_2 work;[5] thus the fractional release of the fission products residing in the fuel-cladding gap can be determined.

IV. RUPTURE TEST PROGRAM

The aim of this program is to obtain consistent results over a broad range of both irradiation and test conditions. In the immediate future, we shall be limited to test specimens with considerably less than ideal characteristics. The readily available fuel rods were fabricated almost ten years ago; thus they do not reflect current fabrication technology. (Presumably, significant improvement in the quality and reproducibility of the fuel and cladding material, and in welding, has been achieved during this period.) Secondly, the irradiation history of these rods is both different from that of rods in commercial power reactors (which have larger, flatter flux cores and subject the fuel to fewer thermal cycles) and is not well known. (I.e., while the total exposure is known to reasonable accuracy, the various time-heat rating combinations are not well known.) These uncertainties will undoubtedly reduce the consistency of, and confidence in, the results of this series of rupture tests. Therefore, we shall attempt to obtain more modern, better characterized fuel rods for future testing.

Several unirradiated Dresden I type rods have been ruptured in system and procedure testing, and to serve as controls for the irradiated rods. The results of some of these tests are presented in Table 2, and the appearance of the ruptured rods is illustrated in Fig. 8. The preliminary

Table 2. Rupture Test Data For Unirradiated Fuel Rods

Rod No.	Internal Pressure (psig)	Heat Rate (°F/sec)	Rupture Temp. (°F)	Length of Rupture (in.)	Maximum Diameter[a] (% increase)	Maximum Swelling[b] (%)
R-1	200	14	1560	0.15	44	34
R-13	200	18	1565	0.15	36	25
R-14	50	29	1805	0.08	39	36
R-15	50	19	1700	0.25	55	47
R-11	100	24	1645	0.13	39	33

[a]Nominal pre-test diameter: 0.567 ± 0.001 in.

[b]Based on circumference.

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Fig. 8. Appearance of Rupture-Tested, Unirradiated Fuel Rods. a) R-1, 200 psig, and b) R-14, 50 psig.

tests showed that unpressurized rods (i.e., about one atmosphere internal pressure at room temperature) swelled but did not rupture up to test temperatures of >2500°F. Consequently, no tests of unpressurized, irradiated rods are planned. Tests of irradiated rods which are currently planned are listed in Table 3. Some of the PWR type rods (0.424 in. O.D.) will be pressurized to at least 500 psia for testing, since PWR rods normally build up considerably higher fission gas pressures than BWR rods.

V. SUMMARY AND CONCLUSIONS

Since irradiation effects may influence the mechanical properties of the Zircaloy clad-UO_2 fuel rods used in water cooled power reactors, an experimental program to rupture test irradiated fuel rods is being conducted. The equipment and procedure for rapid heating of irradiated rods in a steam atmosphere was developed for remote operation. Techniques for modifying the rods to provide realistic internal pressure and gas volume, and for remote attachment of cladding thermocouples were required. The induction heating system proved to be adequate to perform tests with uniform heating rates of 10 to 100°F/sec. Rupture tests of unirradiated fuel rods have been performed successfully, and the results — rupture temperature and extent of swelling — compare reasonably well with other experiments.[8,9]

Table 3. Planned Fuel Rod Rupture Tests

	Rod No.	Length (in.)	Burnup (Mwd/T)	Internal Pressure (psig)	Heating Rate (°F/sec)
I.	BWR Type Rods: 0.565 in. O.D.				
	F6-I	29	7,000	50	20
	F6-IV	29	7,000	50	50
	A-II	29	7,000	200	20
	S-1	41	2,700	50	20
	S-2	41	3,000	50	50
	S-4	41	3,000	50	100
	S-6	41	3,000	200	20
	S-11	41	2,700	200	50
	S-25	41	3,000	200	100
II.	PWR Type Rods: 0.424 in. O.D.				
	R4D-21	41	7,400	50	20
	R4D-22	41	7,400	50	50
	R4D-25	41	7,400	50	100
	R4D-6	41	8,200	500	20
	R4D-10	41	8,200	500	50
	R2D-90	41	8,400	500	100

REFERENCES

1. P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, USAEC Report ORNL-TM-2347, Oak Ridge National Laboratory, September 1968.

2. P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, Part 2: Program Revisions, USAEC Report ORNL-TM-2548, Oak Ridge National Laboratory, May 1969.

3. C. J. Baroch, J. P. Hoffman, and W. C. Rous, AEC Fuel Cycle Program Design and Fabrication of the Basic Fuel Assemblies, USAEC Research and Development Report GEAP-3653 (General Electric Company) 1963.

4. Private correspondence, F. H. Megerth of General Electric Company to R. A. Lorenz of Oak Ridge National Laboratory, July 26, 1967.

5. J. P. Hoffman and D. H. Coplin, The Release of Fission Gases from Uranium Dioxide Pellet Fuel Operated at High Temperatures, GEAP-4596, September 1964.

6. G. W. Parker, et al., Fission Gas Accumulation in Normally Operating Fuel Rods, pp. 84-88, Nuclear Safety Program Ann. Prog. Rept. For Period Ending Dec. 31, 1968, USAEC Report ORNL-4374, Oak Ridge National Laboratory.

7. High Frequency Heating Review, Vol. 1, No. 10, Lepel High Frequency Laboratories, Inc.

8. R. A. Lorenz, G. W. Parker, and D. O. Hobson, Transient Tests of Zircaloy-Clad Fuel Rod Clusters in TREAT, pp. 29-56, ORNL Nuclear Safety Research and Development Program Bimonthly Report for Sept.-Oct. 1969, USAEC Report ORNL-TM-2777, Oak Ridge National Laboratory.

9. R. D. Waddell, Jr., High-Temperature Properties of Zircaloy Cladding, pp. 15-21, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1969, USAEC Report ORNL-TM-2663, Oak Ridge National Laboratory.

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