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PROGRESS IN ZIRCALOY CLADDING FAILURE MODES RESEARCH

P. L. Rittenhouse

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PROGRESS IN ZIRCALOY CLADDING FAILURE MODES RESEARCH

P. L. Rittenhouse

DECEMBER 1970

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PROGRESS IN ZIRCALOY CLADDING FAILURE MODES RESEARCH

P. L. Rittenhouse

ABSTRACT

Both the United States Atomic Energy Commission and industry sponsor research and testing programs on the behavior of Zircaloy-clad fuel rods during the loss-of-coolant accident postulated for light-water reactors. This report describes some of the more recent results obtained in studies of cladding failure behavior (deformation, rupture, and embrittlement), the resulting blockage of the core coolant channels, and the effect of blockage on emergency cooling. Continuing programs in these areas of research are outlined. Recommendations are given for the extension of some existing programs and the initiation of new work.

INTRODUCTION

The fuel for modern light-water reactors (LWR's) is contained in zirconium alloy tubes (Zircaloy) nominally 12 ft long × 1/2 in. in diameter. Fission products generated during the operation of these reactors are retained either within the body of the UO_2 fuel or in the gap and plenum inside the fuel cladding. Further protection against release of fission products to the environment is provided both by the reactor primary system and the containment building. However, one can postulate accidents that will compromise the integrity of one or more of the barriers mentioned above. One such accident, the most serious generally considered, is the loss-of-coolant accident (LOCA) in which the coolant and pressure in the primary system are assumed to be lost because of a rupture in the system piping. The fuel cladding would undergo a severe thermal transient during this accident. The initial driving force for the transient would be the heat stored in the fuel, but the decay heat of the fission products and the exothermic

reaction of zirconium with residual steam would predominate at later times and higher temperatures, respectively. Were this transient allowed to progress unhindered, it would result in gross damage to the core and to other reactor components.

Although the LOCA is considered highly unlikely to occur in an LWR, its consequences would be severe, and emergency core-cooling systems (ECCS) are provided to stop the transient before serious core damage could occur. Nevertheless, should such an accident occur, many of the fuel rods in the core might reach temperatures sufficient to result in rupture of the fuel cladding even with the ECCS performing as designed. This rupture will occur when the hoop stress provided by the pressure of fission gas exceeds the strength of the cladding. Considerable swelling (plastic deformation) of the cladding may occur before rupture and reduce the passageway for flow of the emergency coolant through portions of the core. Some critical but unknown degree of flow restriction may seriously delay stopping the transient. This, in turn, would increase the probability of cladding embrittlement and subsequent disintegration or melting of the fuel rods.

Because of these possibilities, a concerted effort is under way to assess the ways in which cladding fails in terms of accident conditions and reactor operating history and to determine to what extent deformation and rupture may affect the performance of emergency cooling systems. In addition, it is necessary to determine the margins of safety for these conditions.

The Oak Ridge National Laboratory (ORNL) coordinates the work sponsored by the U.S. Atomic Energy Commission on the failure behavior of Zircaloy-clad fuel rods for LWR's during an LOCA and the effects of such failure on the performance of emergency cooling systems and also studies the high-temperature deformation and other behavior of Zircaloy cladding. Other research on the failure of fuel rods is sponsored by the AEC at Idaho Nuclear Corporation and Battelle Memorial Institute (Columbus). Related activities conducted by the

manufacturers of LWR's are factored into the direction and scope of these studies to obtain a balanced and timely approach to the questions about the failure of fuel cladding and its effects.

The results described in this report are divided into five areas: (1) High-Temperature Deformation of Zircaloy Cladding Tubes; (2) Multi-Rod Tests and Coolant Channel Blockage; (3) Embrittlement of Zircaloy Cladding; (4) In-Reactor Evaluation of Cladding Expansion, Channel Blockage, and Embrittlement; and (5) Effect of Cladding Failure on Emergency Cooling Effectiveness. Continuing and future research in each of these areas is discussed in the last section of this report. Review of earlier work on the failure modes of Zircaloy cladding and description and discussion of possible problems resulting from the failure of fuel rods, research tasks, and research priorities and scheduling can be found in previous publications.[1–3]

RESUME OF RECENT RESEARCH

High-Temperature Deformation of Zircaloy Cladding Tubes

The deformation and rupture behavior of Zircaloy cladding under conditions associated with the LOCA is an important factor in the analysis of the outcome of the accident. This aspect of the

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[1]P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, ORNL-TM-2347 (September 1968).

[2]P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, Part 2: Program Revisions, ORNL-TM-2548 (May 1969).

[3]P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, Part 3: Description of the ORNL Program, ORNL-TM-2742 (January 1970).

overall program has been studied at a number of sites. Studies at ORNL[4–7] are determining the effects that certain variables related to irradiation history and reactor operation may have on this behavior. Transient-temperature burst tests were performed on both irradiated and unirradiated cladding. Tests of single tubes of unirradiated Zircaloy were designed to cover as wide a range of pertinent variables as possible within a workable framework by limiting the tests to two controlled variables (internal pressure and heating rate) and one monitored variable (variation in wall thickness). Rupture temperature, rupture strength, time to rupture, circumferential strain, maximum internal pressure, and the physical appearance of the ruptured tubing were measured and evaluated.

In the environment of an LOCA, most of the fuel rods would undergo a thermal transient and would be surrounded by similar fuel rods undergoing almost identical transients. Under such conditions, there should be little or no net transfer of heat from any one rod to its nearest neighbors. This condition was simulated by a novel method of radiant heating with a furnace, power supply, and controller designed for rapid thermal-transient work. Details of the experimental procedure have been reported.[4–6]

The results of the transient burst tests of single tubes are summarized in Figs. 1 and 2. Immediately obvious is the ductility

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[4]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for January-February 1970, ORNL-TM-2919.

[5]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1970, ORNL-TM-2984.

[6]D. O. Hobson *et al.*, "Comparison of Rupture Data from Irradiated Fuel Rods and Unirradiated Cladding," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[7]M. F. Osborne and G. W. Parker, Design, Equipment, and Program for Fuel Rod Burst Experiments, ORNL-TM-2850 (February 1970).

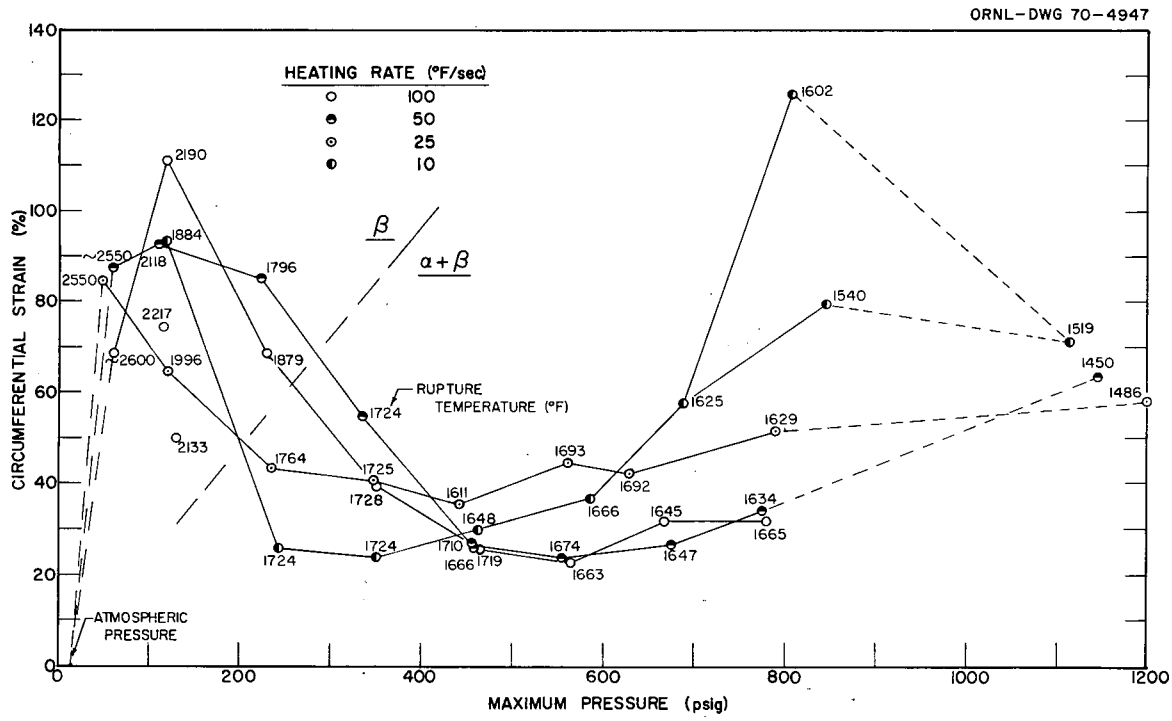

Fig. 1. Circumferential Strain at Rupture as a Function of Maximum Internal Pressure. Results shown are for transient burst tests of single tubes (of the size used for boiling-water reactors) run at four heating rates (10, 25, 50, and 100°F/sec) with initial internal pressures (600°F) from 50 to 1000 psig. The rupture temperature for each test is also shown.

minimum between 400 and 600 psig. Here, strains of 30% are common; at both lower and higher internal pressure, larger strains, as high as 120%, are observed. This behavior is directly related to the crystallographic structure[5,6] of the material at failure. Tubes that rupture in the $\alpha + \beta$ phase region exhibit relatively low ductility, while those that fail at higher temperatures (all β phase) or lower temperatures (all α phase) have relatively higher ductilities. The rupture temperature is related to both internal pressure and heating rate, but is a stronger function of the internal pressure.

The effect of variation in wall thickness on ductility was quite noticeable in these tests. Variation of up to 0.004 in. on any particular cross section of a specimen was seen. This is illustrated in Fig. 3. Strain increased rather dramatically with increasing

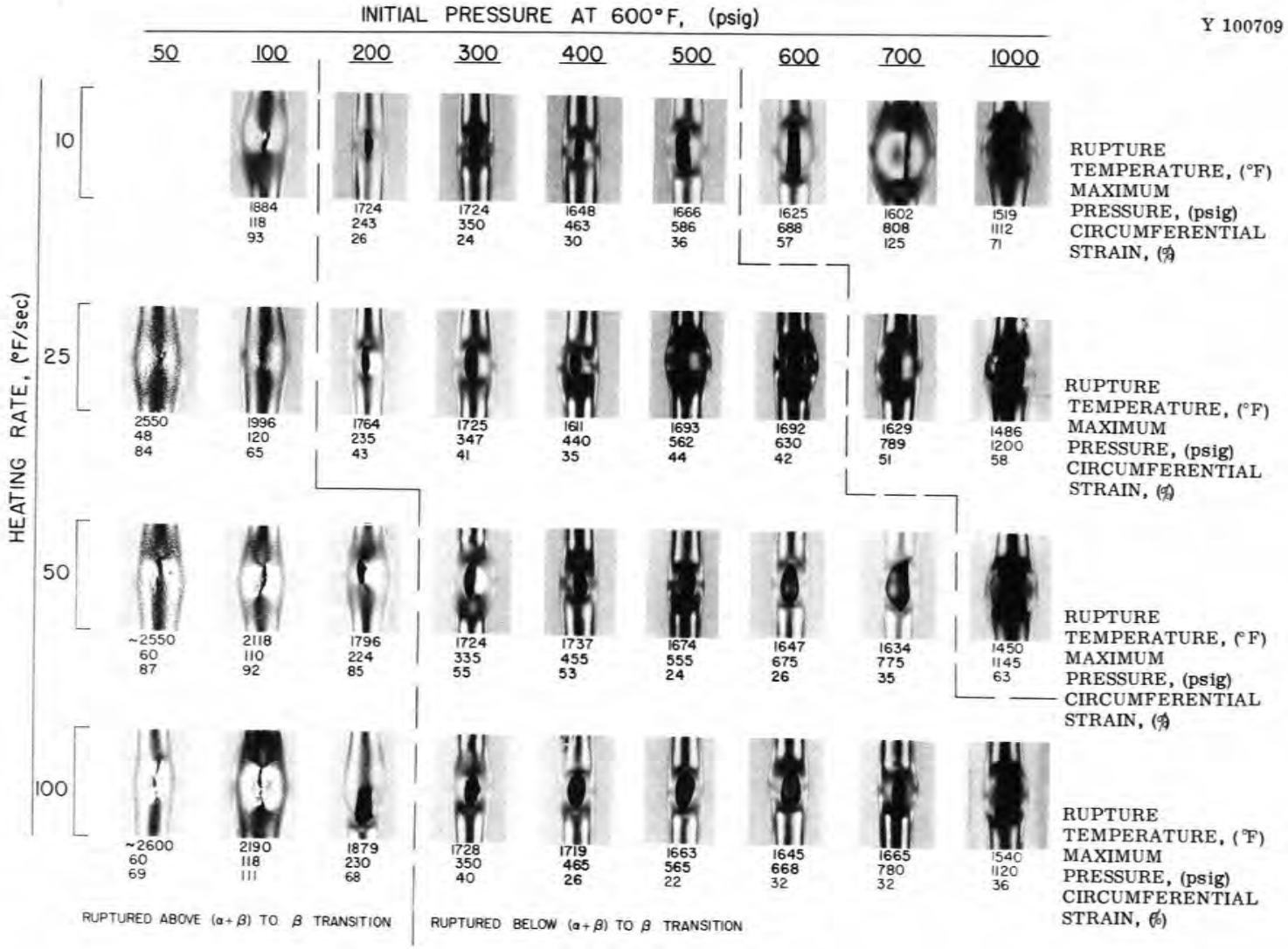

Fig. 2. Rupture Areas on Transient Tube-Burst Specimens. The rupture openings and deformations before rupture are shown for the specimens as functions of initial internal pressure and heating rate. Also shown are the rupture temperature, maximum pressure, and circumferential strain.

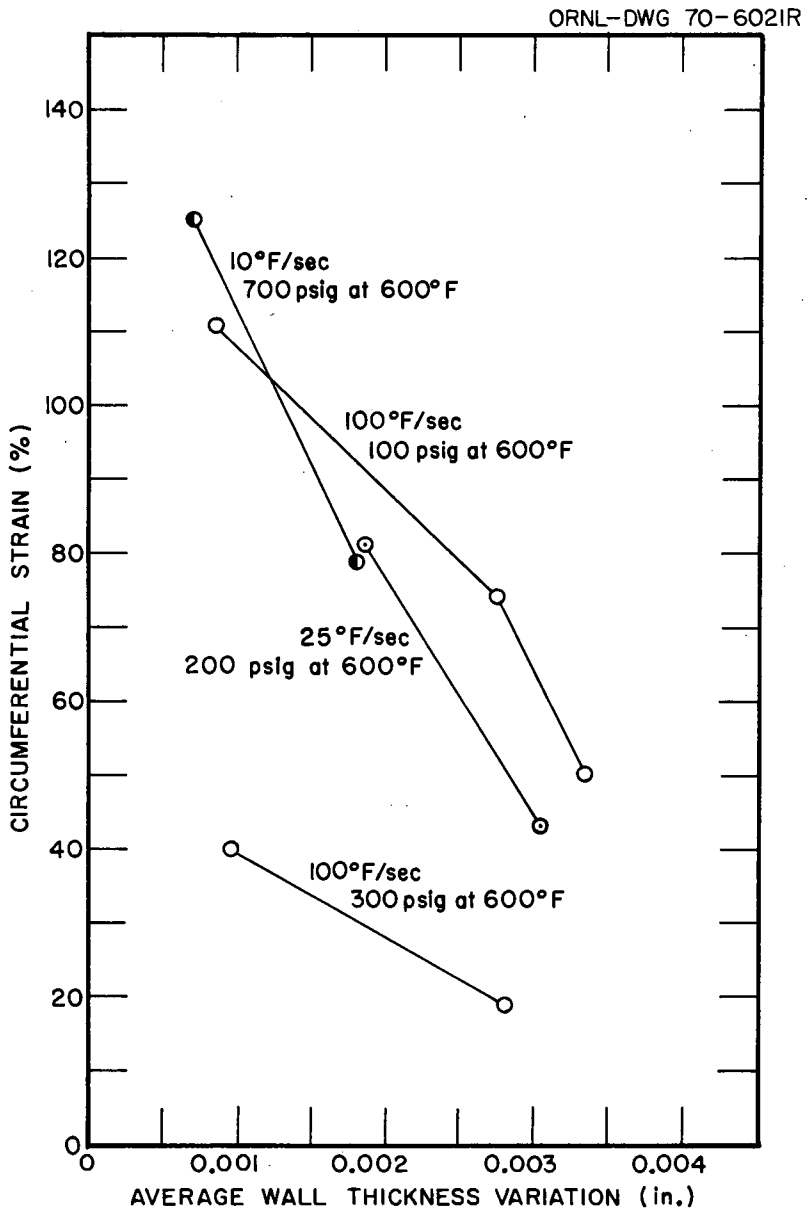

Fig. 3. Effect of Variation in Wall Thickness on Ductility. Results shown are for duplicate or triplicate tests run under the conditions shown. Circumferential strain at rupture increased rapidly as the variation in wall thickness around the circumference at any particular cross section decreased.

dimensional perfection. A direct correlation between the position of the thinnest point in the circumference of the tubes and the rupture opening was also found.

The ductility data presented above were compared with those obtained by Combustion Engineering, Inc.,[8] and Babcock & Wilcox Company.[9]

In the tests run at Combustion Engineering, Inc., at pressures of 200, 800, 1225, 1350, 1800, 1900, and 2500 psig and heating rates of 10 to 170°F/sec, the maximum increase in circumference observed was 50%. In the tests run with initial internal pressures of 200 and 800 psig (maximum pressure unknown), the averages of circumferential expansions found were 36 and 21%, respectively. If the 200-psig pressure increased to as much as 300 psig during the test, the 36% value of expansion agrees fairly well with the ORNL data shown in Fig. 1. However, the 21% value obtained for tests starting at 800 psig was lower than can be reconciled with the ORNL results. Unique to the tests at Combustion Engineering, Inc., was cutoff of the transient at 30 sec, regardless of heating rate and whether rupture had occurred. The transient was provided by induction heating.

Data obtained on single tubes by the Babcock & Wilcox Company,[9] with initial internal pressures of 150, 1000, 2000, and 3000 psig and heating rates of nominally 15 and 75°F/sec, are more in agreement with those obtained at ORNL than are those obtained at Combustion Engineering, Inc. Diameter increases up to 120% were observed. In the tests at 150 psig (maximum 210 psig), the average measured increase in diameter was 76%; that in the tests at 1000 psig (maximum 1200 psig) was 38%. These results are not much different from those in Fig. 1. A welding-machine

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[8] A. D. Emery *et al.*, "Effects of Heating Rate and Pressure on Expansion of Zircaloy Tubing During Sudden Heating Conditions," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in *Nuclear Applications & Technology*.

[9] J. M. Brett and T. P. Papazoglou, "Zircaloy-4 Single- and Multiple-Tube Temperature Excursion Tests," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in *Nuclear Applications & Technology*.

power supply provided current for direct resistance heating of the specimens. The Babcock & Wilcox Company concluded that lower internal gas pressures produced more swelling and that the slower heating rate generally did the same. The hydrogen content of the cladding, up to 500 ppm, had no observable effect on swelling.

Results of burst tests of fuel cladding, conducted with the ORNL hot-cell equipment[10,11] on both irradiated and unirradiated materials are summarized in Fig. 4. The hot-cell data for unirradiated samples showed a saddle similar to that of the ORNL tests outside hot cells. Variations measured in wall thickness in the tubes tested in the hot cells were somewhat larger than the average measured in specimens outside the cells, and, if these data are normalized by reference to Fig. 3, the ductility saddle of the tests of single tubes outside the cells is even more closely approximated. The irradiated cladding varied rather greatly in the amount of circumferential strain, and this strain was, in general, slightly lower than that measured in tests of unirradiated cladding tubes in the hot cells. It cannot be stated whether this decreased ductility was a function of irradiation history, contamination acquired during reactor exposure, variation in wall thickness, or a combination of all these. Wall thickness was not measured.

Our reluctance to attribute the observed decrease in ductility to irradiation is due, in part, to results previously reported by General Electric Company[12] and Westinghouse Electric Corporation.[13] These

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[10]D. O. Hobson *et al.*, "Comparison of Rupture Data from Irradiated Fuel Rods and Unirradiated Cladding," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[11]M. F. Osborne and G. W. Parker, Design, Equipment, and Program for Fuel Road Burst Experiments, ORNL-TM-2850 (February 1970).

[12]E. F. Juenke and J. F. White, Physico-Chemical Studies of Clad UO_2 Under Reactor Accident Conditions, GEMP-731 (April 1970).

[13]J. B. Roll, Performance of Zircaloy-Clad Fuel Rods During a Simulated Loss-of-Coolant Accident--Single-Rod Tests, WCAP-7379 (September 1969).

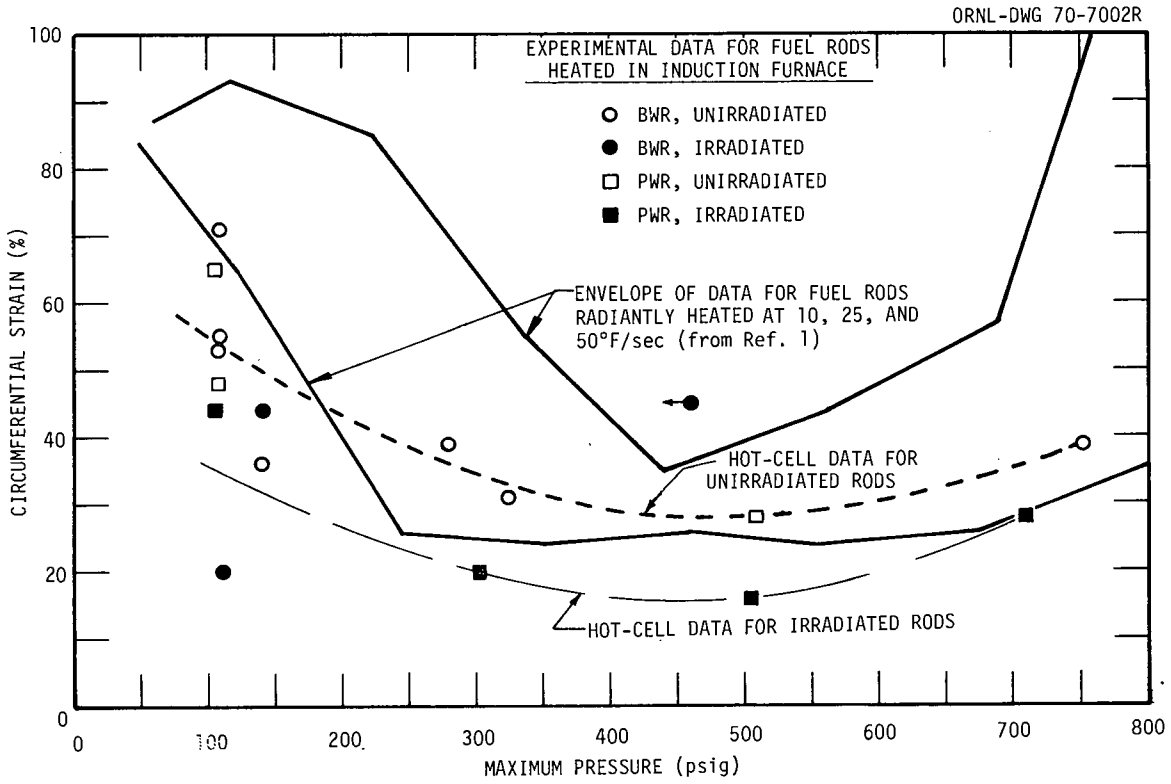

Fig. 4. Comparison of Ductility of Irradiated and Unirradiated Fuel Cladding. The two solid curves denote the data for the envelop of 10, 25, and 50°F/sec obtained in ORNL tests of single tubes of unirradiated Zircaloy (see Fig. 1). The dashed curve and associated open data points denote unirradiated cladding tested in the hot-cell equipment. The filled data points denote cladding from fuel rods irradiated to burnups of up to 7000 Mwd/metric ton.

results on Zircaloy cladding exposed to higher burnup under more modern reactor operating conditions indicated a lesser effect of irradiation on ductility. In all three sets of data, however, there is some indication that irradiation increased rupture temperature and rupture strength.

Multi-Rod Tests and Coolant Channel Blockage

A series of transient burst tests[14–18] of internally heated tubes is being conducted at ORNL to study the interaction of

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[14]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for January-February 1970, ORNL-TM-2919.

Zircaloy tubes as they deform simultaneously and to measure how much the coolant channel area will be restricted by these deformations. Also inherent in these tests is a demonstration of the degree of axial randomness of the position of rupture or maximum swelling deformation. Tests to date (see Table 1) have been concerned with bundles of 13 tubes of the size used for boiling-water reactors (BWR's); bundles of 32 tubes of the size used for pressurized-water reactors (PWR's) are to be tested soon. Details of the bundles, testing procedure, and blockage measurements are given elsewhere.[14–18]

The average blockage over the cross section of the coolant channel ranged from about 20% for test 3 to more than 80% for test 6. However, even in tests with results in the lower portion of this range, some individual coolant channels were almost totally blocked. The greatest blockage observed is illustrated in Fig. 5. This result mirrors the low internal pressure and, perhaps, the presence of the grid spacer at the axial center of the bundle. In general, low pressures and low heating rates seemed to increase blockage. However, tests have not yet been run with pressures above 400 psig, and it might be expected that blockage will again increase at pressures corresponding to the ductility maximum shown in Fig. 1 at 800 to 1000 psig.

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[15] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1970, ORNL-TM-2984.

[16] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for Nov.-Dec. 1969, ORNL-TM-2829.

[17] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1970, ORNL-TM-3061.

[18] R. D. Waddell, Jr., "Measurement of Light Water Reactor Coolant Channel Reduction Arising from Cladding Deformation During a Loss-of-Coolant Accident," paper presented at the American Nuclear Society 1970 Annual Meeting June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

Table 1. Test Conditions and Results for Bundles of 13 Tubes of Size for Boiling-Water Reactors

Test Number	Internal Pressure (psig)	Heating Rate (°F/sec)	Center Grid Spacer	Maximum Average Blockage of Coolant Channel (%)	Average Rupture Temperature (°F)
1[a]	300	100	No	61	1740
2	200	100	No	57	1850
3	400	100	No	22	1720
4	400	25	No	30	1780
5	200	25	No	46	1790
6[b]	100	50	Yes	80	2400[c]

[a]Only ten tubes were pressurized and contributed to blockage.

[b]Two tubes experienced a partial loss of pressure and, therefore, did not rupture and contributed only slightly to blockage.

[c]The average rupture temperature in test 6 was more than 200°F higher than that measured in tests on single tubes. The rupture temperatures in tests 1 through 5 agreed well with data for single tubes. Although not certain, the result of test 6 may have been caused by the presence of the spacer.

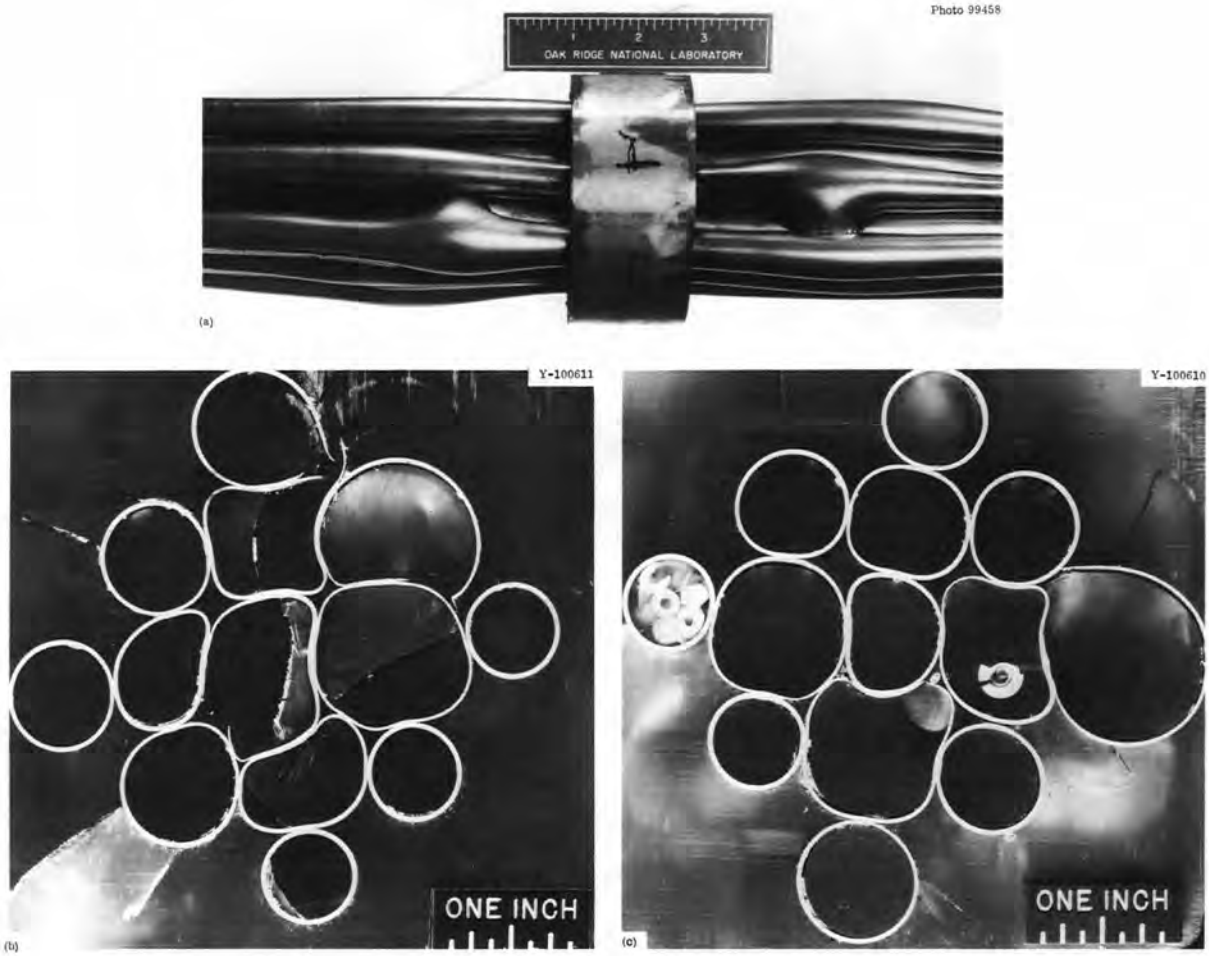

Fig. 5. Bundle 6 After Test. Internal pressure was 100 psig and heating rate was 50°F/sec. Failure at approximately 2400°F. Two of the thirteen tubes lost pressure and did not fail or deform significantly. (a) Overall view showing deformed regions and grid spacer; (b) cross section 4.37 in. below axial center of bundle, 80% blockage; and (c) cross section 2.75 in. above center, 75% blockage.

The distribution of axial positions of maximum expansion on individual tubes is shown in Fig. 6 for three of the tests. Expansions of 10% or greater occurred over some 10 to 12 in. of length, but

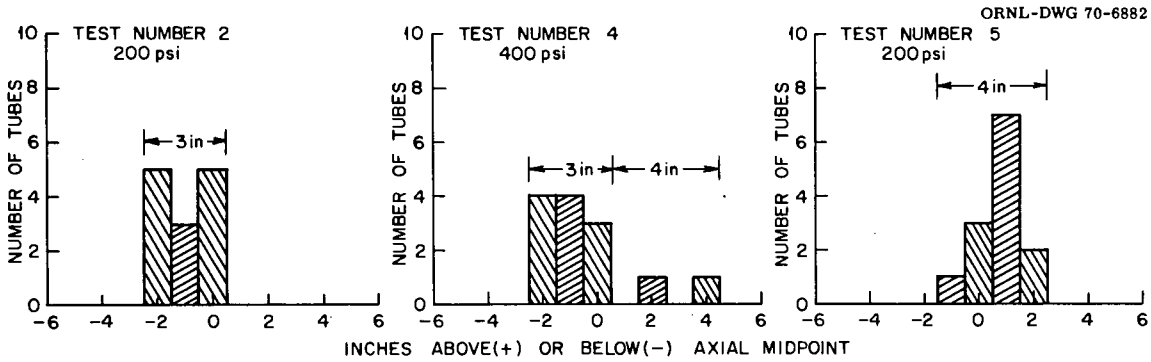

Fig. 6. Histograms of Axial Position of Maximum Expansion. The axial midpoint is the center of the heated zone. The temperature difference between the 0 position (center of heated zone) and positions either 4 in. above or below is much less than 10°F during the transient.

maximum expansions were normally restricted to some 3 to 4 in. Only in the test at 400 psig and 100°F/sec (test 3) can the positions of the failures and deformations be described as random. Therefore, either failure point is extremely sensitive to temperature or some other, unknown factor is operative. Furthermore, the presence of the grid spacer in test 6 seemed to reduce axial randomness and provide, instead, two concentrated locations of severe blockage.

Multi-rod tests were conducted by Babcock & Wilcox Company to determine whether cladding failures occur randomly in the axial direction of a typical core configuration.[19] Indeed, even though a larger axial temperature gradient existed in these tests, the randomness shown was greater than that in the ORNL tests. The heating method employed, immersion in a preheated furnace, resulted in a large radial temperature gradient across the bundle. Therefore, no conclusions on blockage of the coolant channels can be drawn from these tests. This heating and test method may also have affected the degree of randomness of failure and swelling.

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[19]J. M. Brett and T. P. Papazoglou, "Zircaloy-4 Single- and Multiple-Tube Temperature Excursion Tests," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal.

Although Westinghouse Electric Corporation performed a series of experiments on multiple tubes to measure channel blockage, their results have not been made generally public, and no other recent, comparable data are available for comparison with the ORNL results.

Babcock & Wilcox Company developed a model[20] to predict the distribution (randomness) of cladding failures. The model, in its present stage of development, does not take into account the length of rod affected by failure and swelling, the swelling and subsequent mechanical interactions of the rods, the influence of ovality (see "High-Temperature Deformation of Zircaloy Cladding Tubes," pp. 3–10 this report), or scatter in mechanical properties.

Embrittlement of Zircaloy Cladding

Early work on the embrittlement of Zircaloy cladding by oxygen under LOCA conditions in a steam environment at Argonne National Laboratory has recently been presented in a topical report.[21] In this study, the criteria for failure of the Zircaloy cladding was its breakup or brittle fracture when quenched from high temperature. Heating rates to the quench temperature were relatively slow (about 10°F/sec), and the quench temperatures ranged from 1900 to 3800°F. The minimum quench temperature to avoid failure under these conditions was about 2900°F. Holding at 2200°F for more than 6 min also resulted in failure. No other measure of embrittlement, such as impact properties or hardness, was reported.

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[20] B. E. Bingham *et al.*, "Application of Experimental Data to Analytical Evaluation of Cladding Failure Distribution," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in *Nuclear Applications & Technology*.

[21] J. C. Hesson *et al.*, *Laboratory Simulations of Cladding-Steam Reactions Following Loss-of-Coolant Accidents in Water-Cooled Reactors*, ANL-7609 (January 1970).

The effects of an LOCA on Zircaloy embrittlement have also been studied at Idaho Nuclear Corporation.[22,23] Brittle behavior (determined by compression tests) at room temperature was observed after 6 min at 2150°F or more than 1 min at 2500°F. Those conditions correspond roughly to the conversion of 15% of the thickness of the cladding wall to ZrO_2 or oxygen-stablized α-zirconium. In addition, similarly exposed specimens were compression tested at 900 and 1800°F by INC. Ductile behavior was observed at 1800°F.

The Oak Ridge National Laboratory has undertaken a program to generate time-temperature envelopes for the embrittlement of Zircaloy cladding. These envelopes will be related to bend ductility, hardness, and impact behavior at room temperature and elevated temperatures. Both the inner and outer surfaces of the cladding are being exposed to steam to simulate oxidation after rupture of a fuel rod. To date, we have found that Zircaloy with more than 14 at. % O has zero bend ductility at room temperature and a diamond pyramid hardness (DPH) of about 500 (ref. 24). These facts are being used to assess the oxygen content (and its gradient) and bend ductility of Zircaloy cladding specimens exposed to steam at 1800 to 2400°F. An impact test for use at elevated temperature is being developed as a measure of the embrittlement of the cladding. This will provide a more realistic evaluation of embrittlement than can slow-rate compression or bend tests.

In-Reactor Evaluation of Cladding Expansion, Channel Blockage, and Embrittlement (TREAT Fuel Rod Failure Experiments)

Experiments on failure of fuel rods under LOCA conditions for an LWR are being conducted in the Transient-Reactor Test Facility (TREAT)

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[22] R. Herzel and R. H. Meservey, Trans. Am. Nucl. Soc. 12(1), 355–356 (1969).

[23] M. J. Graber and W. F. Zelezny, Trans. Am. Nucl. Soc. 12(1), 356–357 (1969).

[24] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1970, ORNL-TM-3061.

with clusters of Zircaloy-clad fuel rods to determine whether their swelling and rupture will interfere with the effectiveness of emergency cooling. Two such experiments, FRF-1 and FRF-2, have been performed. In both experiments, the test assembly consisted of a cluster of seven 27-in.-long Zircaloy-clad UO_2 fuel rods and in both cases the center rod had been previously irradiated. All rods were pressurized with helium before testing to simulate fission gas accumulated during reactor operation. The tests were performed in the TREAT by operating the reactor at steady power for about 30 sec so that fission heat in the UO_2 pellets raised the temperature of the cladding to duplicate the temperature behavior of an LOCA immediately after blowdown. An atmosphere of flowing steam simulated residual steam boil-off from the reactor vessel. Descriptions of equipment, fuel rods, experimental procedure, and the data obtained from these tests were reported.[24–28]

Cladding expansion in these tests is compared with that for other tests[26,29,30] in Fig. 7. We consider the degree of correspondence excellent. No effect of the irradiation of the center rod was evident. The extent of channel blockage provided by these cladding expansions is shown in Fig. 8. Nowhere in FRF-1 was more than 50%

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[25]P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, Part 3: Description of the ORNL Program, ORNL-TM-2742 (January 1970).

[26] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1970, ORNL-TM-2984.

[27]R. A. Lorenz et al., Preliminary Evaluation of the First Fuel Rod Failure Transient Test of a Zircaloy-Clad Fuel Rod Cluster in TREAT, ORNL-TM-2774 (November 1969).

[28]R. A. Lorenz et al., "Fuel Rod Failure Under Loss-of-Coolant Conditions in TREAT," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[29]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for January-February 1970, ORNL-TM-2919.

[30]D. O. Hobson et al., "Comparison of Rupture Data from Irradiated Fuel Rods and Unirradiated Cladding," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

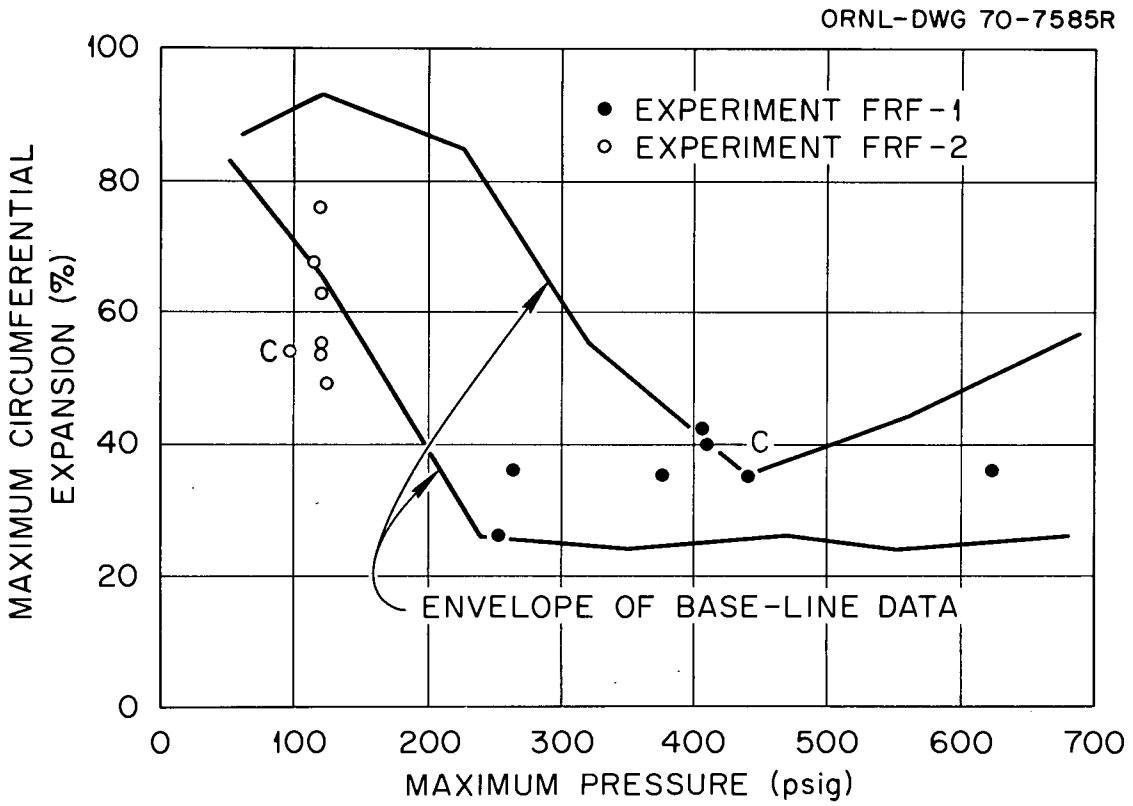
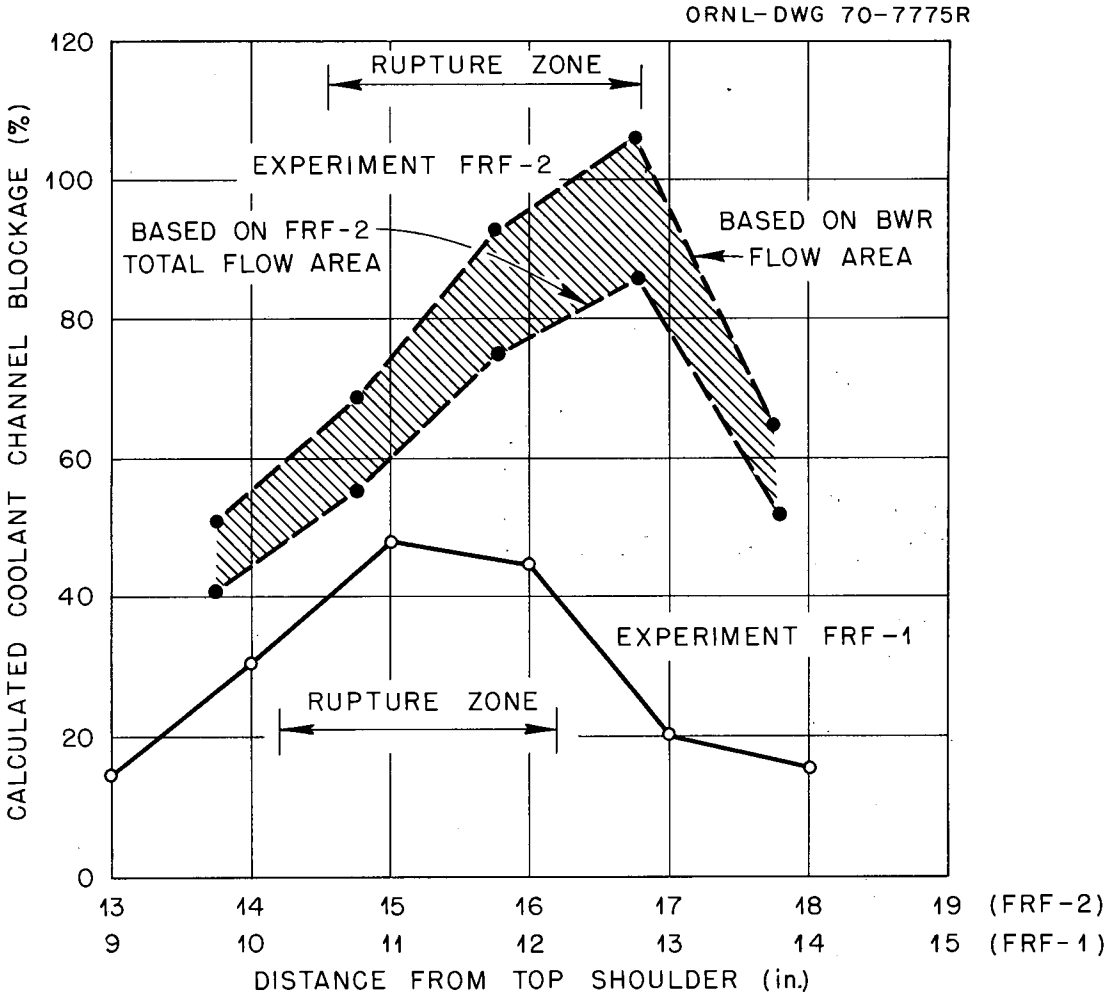


Fig. 7. Comparison of Cladding Ductility Observed in Fuel Rod Failure Experiments in the Transient-Reactor Test Facility with That Measured in Tests of Single Tubes. The curves shown are the envelope of data obtained on single tubes at 10, 25, and 50°F/sec. Filled circles are data obtained in FRF-1. Note that the rods were pressurized to different levels before testing. Open circles are for FRF-2 fuel rods. In both cases, the datum point for the center, irradiated rod is indicated by "C".

of the coolant channel blocked, but more than 50% of a 5-in. section and approximately 100% over a 1-in. section of FRF-2 were blocked. These results are consistent with those obtained in our experiments on multiple tubes, as discussed above. (See "Multi-Rod Tests and Coolant Channel Blockage," pp. 10–15, this report). Tubing ruptures were confined to a 2-in. length of the bundle.

A total of 1.1 vol % of the Zircaloy cladding of FRF-2 reacted with steam compared to only 0.2 vol % for FRF-1 (maximum temperature



Coolant Channel Blockage For BWR Rod Spacing.

Fig. 8. Coolant Channel Blockages Observed in TREAT Fuel Rod Failure Experiments. These curves present channel blockages calculated over some 5 in. of the TREAT bundles. The calculations were based on micrometer measurements of the cladding dimensions. Blockage more than 100% is possible as rods are not constrained from bowing in these tests. Also shown for FRF-2 is a curve calculated on the basis of total flow area of the experiment cross section. The maximum blockage calculated using this area is 85%. The true representation of the blockage lies somewhere between the two curves shown for FRF-2.

1800°F). In FRF-2 the transient peaked at 2400°F, but the total times above 2200, 2000, and 1800°F were only 90, 120, and 180 sec, respectively. Figure 9 is a photograph of the FRF-2 bundle after removal from the primary vessel. A heavy layer of white oxide was apparent mainly at the bottom (steam inlet) and on the outer surfaces of the rods.

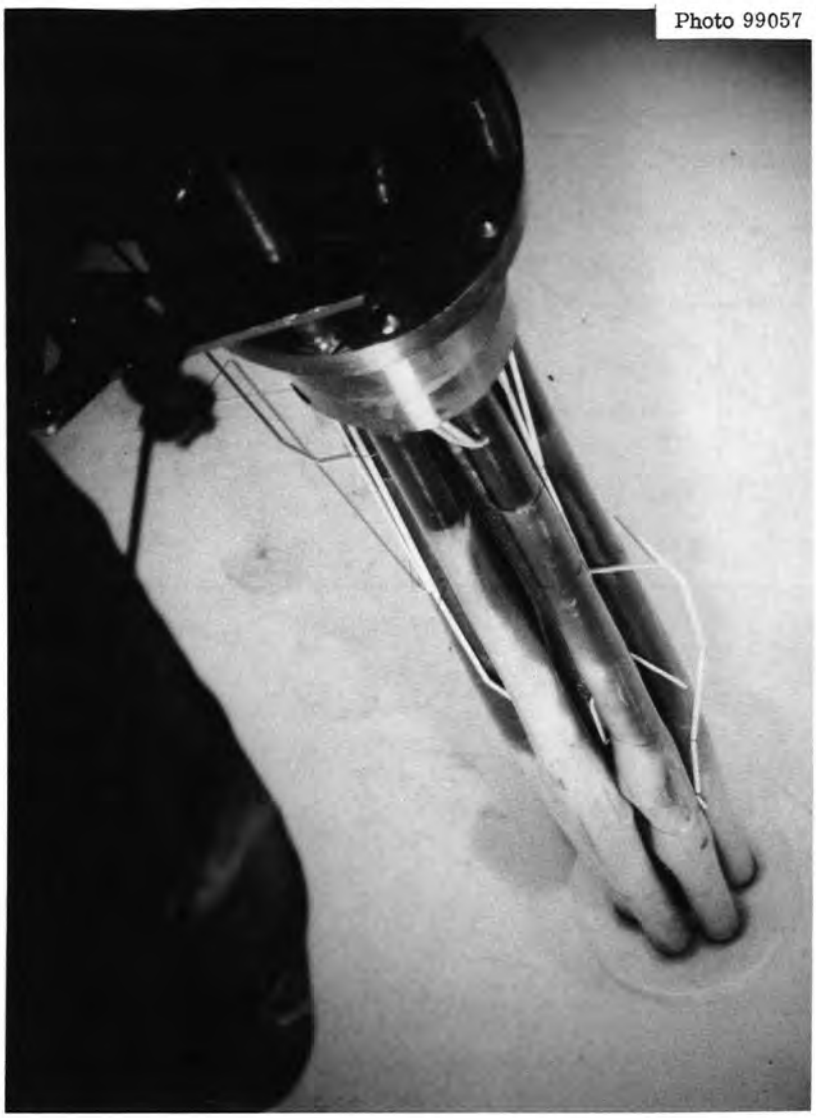


Fig. 9. Photograph of FRF-2 Test Assembly After Test. The bundle of Zircaloy-clad fuel rods shows white oxide toward bottom (steam inlet) and original black oxide around plenums at top. Notice also the deformation on the two nearest rods and the center rod visible between them.

This was formed primarily after the time of rupture and transient peaking. The absence of this oxide on the inner surfaces above the deformed area can lead to speculation about diversion of steam flow by the blockage. The brittleness of the cladding was demonstrated accidently when one of the rods slipped from the hot-cell manipulators, dropped about 12 in. onto a plywood platform covered with blotter paper, and fractured as shown in Fig. 10. Subsequently, several other rods were broken during handling. Metallographic examination and hardness tests of these rods will be performed after some further period of radioactive decay.

Fig. 10. Brittle Fracture of Fuel Rod from Test Assembly FRF-2. This Zircaloy-clad UO_2 fuel rod fractured across the rupture region when dropped in the hot cell. The center UO_2 pellet is also somewhat cracked, and small fragments of fuel, together with fragments of the cladding, can be seen on the blotter paper.

Effect of Cladding Failure on Emergency Cooling Effectiveness

A series of scoping calculations were made at Battelle Memorial Institute (Columbus)[31,32] to estimate the effects of blockage on the performance of the ECCS. It was concluded from these calculations that reductions in flow area up to almost 90%, although they would reduce the efficiency of emergency cooling, would not prevent the core from being cooled but would only slow the cooling rate. One of the major uncertainties in these calculations was the flow resistance provided by various blockage geometries. Flow tests that are planned with distorted 32-rod arrays (see "Multi-Rod Tests and Coolant Channel Blockage," pp. 26–28 this report) should provide experimental data that can be used to characterize flow resistance as a function of blockage geometry.

A single effort aimed at defining not only failure mode but its effect is progressing at ORNL.[33–38] The dimensional and thermal

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[31]W. A. Carbiener and R. L. Ritzman, An Evaluation of the Applicability of Existing Data to the Analytical Description of a Nuclear-Reactor Accident, Quarterly Progress Report for January through March 1970, BMI-1881 (April 1970).

[32]W. A. Carbiener, "The Effects of Clad Swelling on Emergency Core Cooling Performance," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[33]P. L. Rittenhouse, Failure Modes of Zircaloy-Clad Fuel Rods, Part 3: Description of the ORNL Program, ORNL-TM-2742 (January 1970).

[34]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for Jan.-Feb. 1970, ORNL-TM-2919.

[35]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1970, ORNL-TM-2984.

[36]W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for Nov.-Dec. 1969, ORNL-TM-2829.

thermal responses of simulated fuel rods are being investigated during the depressurization, transient heatup, and emergency cooling stages of the LOCA. Heater rods clad with Zircaloy and operable with internal pressure at heat fluxes and stored energies comparable to those of fuel rods were developed and tested, and the closed-cycle loop in which the experiments are being conducted was constructed and operated.

Seven of these depressurization tests of single heater rods were completed. Coolant discharge times were varied from 12 to 60 sec and rod power from 13 to 17.5 kw/ft (342,000 to 450,000 Btu/hr-ft^2) at steady state. Steady state power was maintained for 4 to 8 sec after the start of depressurization and then decreased to 10% of the original level. In all of these tests, the temperature of the cladding and the water followed each other closely until the loop pressure approached atmospheric pressure; then the cladding temperature increased very rapidly. In none of these tests was the heater rod internally pressurized.

Although the data obtained in the depressurization tests of single heater rods is encouraging in terms of cooling of fuel rods during the blowdown portion of the LOCA, it cannot be extrapolated directly to reactor situations because the rods were not pressurized and because of lack of simulation of geometry, coolant inventory, and coolant enthalpy rise. Most of these difficulties and objections will be overcome in the planned series of depressurization experiments with clusters of seven heater rods.

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[37] W. B. Cottrell, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May-June 1970, ORNL-TM-3061.

[38] C. G. Lawson, "Simulation of Fuel Rod Behavior in a Light Water Reactor Loss-of-Coolant and Pressure Accident (LOCA)," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

The Full Length Emergency Cooling Heat Transfer Project[39,40] (FLECHT) conducted by Idaho Nuclear Corporation is directed toward experimental investigation of thermal and hydraulic aspects of termination of the thermal transient of the LOCA by emergency core cooling. The BWR and PWR portions of this project are subcontracted to General Electric Company and Westinghouse Electric Corporation, respectively. The FLECHT tests employ 49- and 100-rod bundles of full-length, electrically heated rods. Stainless-steel-clad rods are used to obtain parametric heat transfer information and Zircaloy-clad rods are tested to determine the behavior of Zircaloy during the LOCA and to demonstrate effectiveness of the ECCS.

In the PWR-FLECHT Project, tests of flow blockage were performed with 7 × 7 arrays (49-rod bundles) of stainless-steel-clad rods.[41] Flow was blocked by a 3/8-in.-thick plate at the middle of the bundle length. Blockages of 50 and 75% of the flow area in the 5 × 5 center array of the 7 × 7 bundle were not found detrimental to the effectiveness of bottom-flooding heat transfer. In fact, these blockages seemed to improve cooling capability. Much the same thing was observed by Idaho Nuclear Corporation in smaller scale experiments.[42] This effect is explained in terms of atomization of the coolant as it is

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[39]J. C. Haire and G. F. Brockett, Boiling Water Reactor-Full Length Emergency Cooling Heat Transfer (BWR-FLECHT) Tests Project, IN-1385 (June 1970).

[40]J. C. Haire and G. F. Brockett, Pressurized Water Reactor-Full Length Emergency Cooling Heat Transfer (PWR-FLECHT) Tests Project, IN-1386 (April 1970).

[41]J. O. Chermak *et al.*, "Effect of Flow Blockage on Bottom Cooling Heat Transfer Effectiveness (FLECHT)," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[42]P. R. Davis, "Experimental Studies of the Effect of Flow Restrictions in a Small Rod Bundle Under Emergency Core Coolant Injection Conditions," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

forced through the openings in the blockage plate. Real blockages, however, will probably not provide the simple, uniform orifice configuration that gives rise to this mechanism (coolant atomization) for improved cooling. Also, it seems likely that in a reactor core the coolant would bypass the blockage zones instead of being forced through them. Tests with more severe blockages were recently performed, but the results of these experiments are not yet available in the literature.

A BWR-FLECHT experiment of particular interest is one in which a bundle of 49 Zircaloy-clad rods was internally pressurized and then spray cooled under LOCA conditions.[40–44] The maximum reported reduction in the area for coolant flow around a single rod caused by swelling of the cladding was 50%. This did not seem to seriously affect the effectiveness of spray cooling. However, interpretation of results was complicated by the electrical failure of ten of the rods. Cooling was started at a maximum cladding temperature of 1920°F, and the highest temperature reached by any rod in the bundle was 2250°F. Most of the rods, however, were much cooler than 1900°F (as low as 1700°F) at the time when spray cooling was started. Also, since the maximum temperature reached by some of the rods was less than 1800°F, the cladding did not fragment during the cooling. In subsequent tests (unpressurized) in which spray cooling was started at higher temperatures, problems with breakup of the cladding have been encountered.

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[43] R. G. Bock *et al.*, "The Effect of Fuel Rod Failures on BWR Core-Spray Standby Cooling Performance," paper presented at the American Nuclear Society 1970 Annual Meeting, June 29–July 2, 1970, Los Angeles, Cal. To be published in Nuclear Applications & Technology.

[44] J. E. Leonard and J. D. Duncan, Preliminary Results of the BWR-FLECHT Internally Pressurized Zircaloy-Clad Bundle, Spray Cooling Test, NEDG-13064 (January 1970).

CONTINUING AND FUTURE WORK

High-Temperature Deformation of Zircaloy Cladding Tubes

The ORNL tests of single tubes described earlier in this report (see "High-Temperature Deformation of Zircaloy Cladding Tubes," pp. 3–10, this report) are being continued with Zircaloy tubes of the size used for cladding in PWR's. The range of internal pressures studied will be broadened to include the higher pressures of interest in the PWR systems. All other variables and testing conditions will be as previously described. These studies will be completed in fiscal year 1971.

The ORNL tests of single tubes with irradiated cladding will also continue. We plan to use internal heating in at least eight tests with short sections (8 in.) of Zircaloy fuel cladding, previously emptied of fuel and cleaned, that has undergone a fuel burnup above 20,000 Mwd/metric ton. In addition, a few fueled rods (about 40 in. long) with burnups to 10,000 Mwd/metric ton will be tested using the in-cell method of induction heating.

Multi-Rod Tests and Coolant Channel Blockage

Transient tube burst experiments on multiple tubes scheduled at ORNL during fiscal year 1971 are shown in Table 2. All of the tests shown will, of course, be internally pressurized and internally heated and will be examined after test to determine the randomness of the failure position and blockage of the coolant channels. In addition, all of the deformed bundles will undergo flow tests (measurements of pressure drop) in a water loop to provide experimental information on flow resistance as related to the extent and geometry of blockage. (Combustion Engineering, Inc., will perform the flow test of test 8.) Heating rates and internal pressures used in the transient burst tests will be chosen by consultation with the reactor vendor that supplies the cladding for each particular test.

Table 2. Transient Tube Burst Tests Scheduled for Fiscal Year 1971

Test Number	Number of Rods	Zircaloy Cladding Source
7	32	ORNL tests of single tubes
8	16	Combustion Engineering, Inc.[a]
9	32	Westinghouse Electric Corporation
10	32	Babcock & Wilcox Company
11	32	Combustion Engineering, Inc.
12	32	Westinghouse Electric Corporation
13	32	Babcock & Wilcox Company
14	32	Combustion Engineering, Inc.
15	24	General Electric Company

[a]Test 8 is a cooperative effort between ORNL and Combustion Engineering, Inc.

Also planned for fiscal year 1971 at ORNL is a test of a bundle of 13 internally heated irradiated cladding tubes (about 7000 Mwd/metric ton burnup in fuel rods). This bundle will not be tested in flow after completion of the transient. Test conditions selected will duplicate those in one of the tests with unirradiated cladding. Depending upon the success and results of this first experiment with irradiated cladding, a similar test with high-burnup fuel rods (above 20,000 Mwd/metric ton) will be proposed for fiscal year 1972.

Embrittlement of Zircaloy Cladding

Investigations of the ductility of Zircaloy cladding exposed to simulated LOCA transients will continue and be essentially completed at ORNL in fiscal year 1971. This program, as noted earlier, will evaluate embrittlement in terms of oxygen content and gradient, microstructure, bend ductility, and resistance to impact at elevated temperatures. Both the ORNL fuel rod failure experiments run in the TREAT reactor and the depressurization experiments on the bundles of seven Zircaloy-clad heater rods will provide additional information on embrittlement. Also PWR- and BWR-FLECHT bundles of Zircaloy cladding should be useful in this regard.

In-Reactor Evaluation of Cladding Expansion, Channel Blockage, and Embrittlement (TREAT Fuel Rod Failure Experiments)

Only one ORNL experiment in TREAT on fuel rod failure is planned for fiscal year 1971. In this experiment, however, all fuel rods will have been irradiated to more than 20,000 Mwd/metric ton burnup. An array of 13 heater rods on a square pitch will replace the array of 7 heater rods (triangular spacing) used in the previous experiments. This experiment will provide data on the expansion and rupture of highly irradiated Zircaloy cladding, the randomness of failure, blockage of the coolant channel, degree of reaction between Zircaloy and steam, and embrittlement.

Effect of Cladding Failure on Emergency Cooling Effectiveness

Clusters of seven Zircaloy-clad heater rods will be subjected to the depressurization and thermal transient of an LOCA and then spray cooled during fiscal year 1971 at ORNL. These experiments will provide, in addition to cladding data on rupture and blockage, information on

heat transfer (cooling effectiveness) during depressurization and the spray-cooling period. About six of these experiments are planned for the fiscal year. In most cases the heater rods will be internally pressurized.

Both the BWR- and PWR-FLECHT Projects are scheduled for completion in fiscal year 1971. Although a few more bundles of Zircaloy cladding are scheduled for FLECHT, none are to be tested with internal pressure. However, at least one FLECHT test with rods internally pressurized so as to promote a maximum of cladding deformation and blockage either before or after the start of cooling, seems more realistic, meaningful, and potentially rewarding than those of the recent and planned series of Zircaloy FLECHT experiments.

New Programs

A program[45] proposed by Idaho Nuclear Corporation is Simulated Emergency Flow Effects Tests (SEFET). This program is designed to determine if radial flow occurs during emergency core cooling and, if so, how the resulting coolant starvation influences cooling requirements. The two driving forces of particular concern with regard to radial flow are the blockage provided by deformation of the cladding and the large radial temperature gradient that exists in the reactor core. These or similar experiments seem worthwhile and meaningful. No funding is available at present.

The Power Burst Facility (PBF) at the National Reactor Testing Station also has applicability to fuel rod failure associated with the LOCA. In the PBF, large clusters of up to 76 fuel rods (about 36-in. long) can be tested within a driver core. The reactor may be operated in steady state mode, power burst mode, or a combination of both steady

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[45] J. C. Haire and G. F. Brockett, Simulated Emergency Flow Effects Tests (SEFET) Project, IN-1387 (June 1970).

state and power burst modes. A pressurized water loop permits control of coolant conditions within the experiment container. Combination of loop and reactor conditions should make it possible to simulate conditions appropriate to the LOCA. The PBF test plan[46] includes the running of clusters of LWR fuel rods, but no experiment has been designed specifically for failure of fuel rods under LOCA conditions. Evaluation and design of an integral test of this type for the PBF should be initiated.

SUMMARY

Experiments conducted over the past year have led to a much improved understanding of the high-temperature deformation (swelling) and rupture behavior of Zircaloy-clad fuel rods. This work will be extended to cover the range of fission-gas pressures expected in PWR fuel rods of high burnup. Still not completely answered are the questions of axial randomness of fuel rod swelling and rupture and the degree and geometry of core blockage that result from cladding swelling, rupture, and interactions between fuel rods. These questions and the effects of reactor exposure (burnup, etc.) on all of the foregoing are being investigated. Studies conducted to date on cladding reaction with steam have been neither systematic nor realistic in their approach to evaluation of embrittlement. Further work is needed and is planned.

The results of tests aimed at demonstrating the effectiveness of emergency core cooling are highly encouraging, but still not conclusive. For instance, it is unreasonable to expect that blockages of the coolant channels experienced in an LOCA will have the fortunate orifice geometry of blockages simulated in tests to date. Also, more flow diversion can be expected. Therefore, extrapolation of these data to

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[46] E. Feinauer *et al.*, PBF Test Program Outline. To be published as Idaho Nuclear Corporation technical report. IDO-17298.

ECCS performance should proceed with great care and caution. There are several continuing programs that will provide additional information for this evaluation. New programs, as yet only proposed, would aid greatly in answering questions about both cooling effectiveness and failure of fuel rods.

A summary of the programs discussed in this report is given in Fig. 11. Both continuing and future work, planned and proposed, are shown. The organization responsible for each of the projects and whether these projects are funded by the AEC or privately funded are also identified.

ACKNOWLEDGMENTS

This report would not have been possible without the contributions of many individuals of the ORNL Nuclear Safety Program. I am also grateful to the authors of the various reports and papers I studied in preparing this document. Finally, thanks are due Jane Morrison for drafting this report and to Vicki Sise, of the Metals and Ceramics Division Reports Office, for preparing the final report.

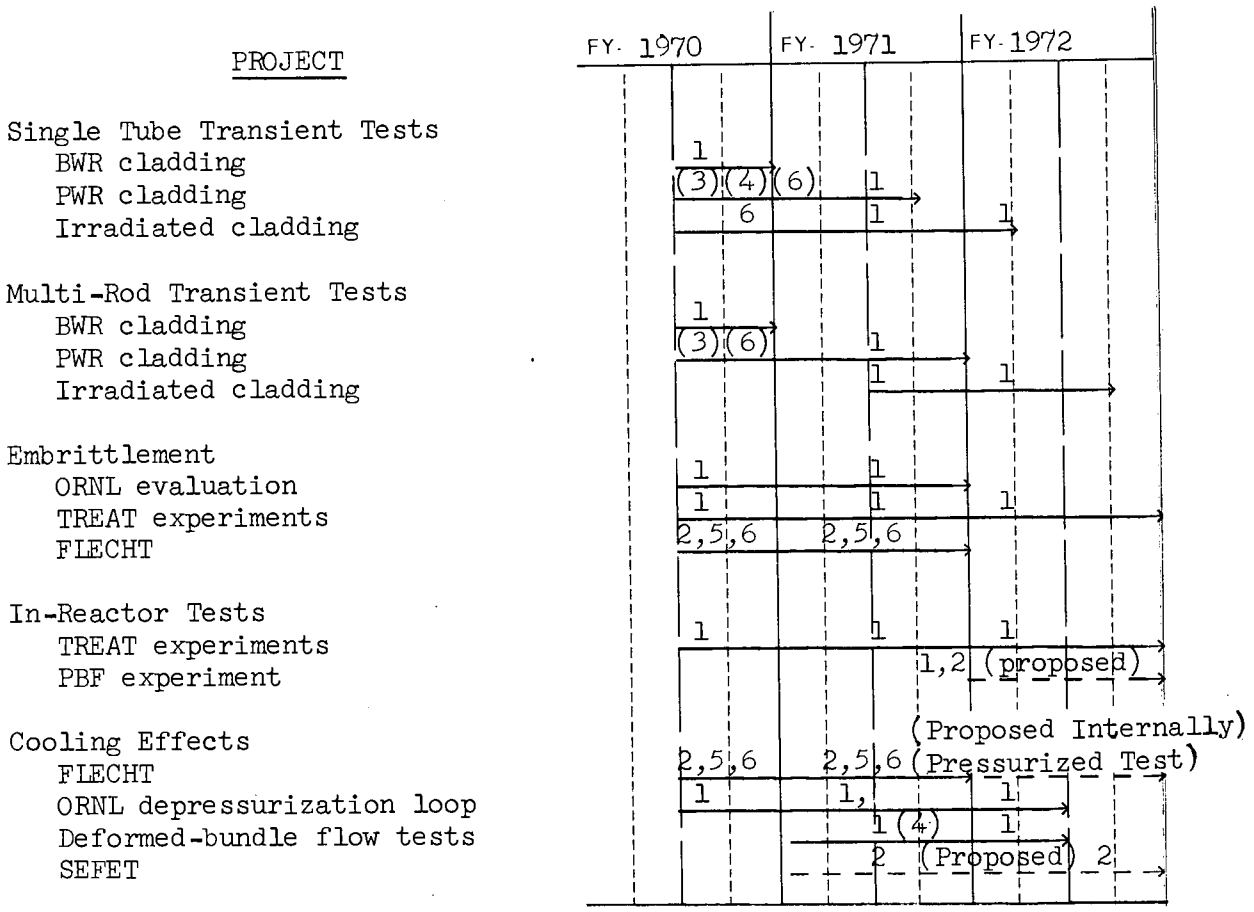

Fig. 11. Work Chart for Experiments on Failure of Fuel Rods. Organizations active in each project are identified as (1) ORNL, (2) Idaho Nuclear Corporation, (3) Babcock & Company, (4) Combustion Engineering, Inc., (5) General Electric Company, and (6) Westinghouse Electric Corporation. Where work is privately funded, the identifying numeral is shown in parentheses.

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