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Quarterly Report
January-March 1984

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
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Commission

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ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory from January 1 through March 31, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada.

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ACOUSTIC EMISSION/FLAW RELATIONSHIP FOR IN-SERVICE MONITORING OF NUCLEAR PRESSURE VESSELS[a]

P. H. Hutton, Project Manager
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SUMMARY

Progress for this program will be reported in a separate quarterly report.

(a) FIN: B2088; NRC Contact: J. Muscara.

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

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SUMMARY

Progress for this program will be reported in a separate quarterly report.

(a) FIN: B2289-0; NRC Contact: J. Muscara.

EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES[a]

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M. E. Cunningham, Task B Leader
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J. O. Barner, Task D Leader

SUMMARY

The principal objectives of this program are to obtain in-reactor and out-of-reactor data on thermal and mechanical fuel rod performance and to integrate these data into the FRAPCON-2 computer code. This quarter, Rod A-7 from IFA-518 was inspected and checked for a suspected leak. Sipping test results proved negative, and the anomalous instrumentation behavior in Rod A-7 is tentatively attributed to fission gas release rather than cladding failure. The assembly was restarted in mid-March 1984. A preplanned scram of the Halden reactor was conducted in early March, during which the transient response of fuel centerline thermocouples was recorded, including those for IFA-432.

The postirradiation examination (PIE) data report for IFA-527 was published, and the IFA-492 interim PIE report was sent to the U.S. Nuclear Regulatory Commission (NRC) for printing and distribution. The IFA-432 document was amended after submittal to include results of whole pellet retained fission gas analyses.

The current version of the FRAPCON-2 computer code (V1M4) was further assessed in the process of preparing an improved version (V1M5). Samples from irradiated Zircaloy cladding tested in pellet-cladding interaction (PCI) simulations were examined at high magnification by scanning electron microscopy (SEM).

INTRODUCTION

The objectives of the Experimental Support and Development of Single-Rod Fuel Codes Program at Pacific Northwest Laboratory (PNL) are fourfold:

- Task A - collect and correlate in-reactor and PIE data on fuel rod thermal/mechanical behavior, especially as a function of rod design and burnup
- Task B - qualify, organize, and analyze the fuel performance data and report the data, trends, and conclusions
- Task C - integrate the above information into the FRAPCON series of computer codes
- Task D - study the occurrence and mechanisms of cladding deformation and failure using controlled experiments with centrally heated simulated fuel rods in a pressurized water loop at PNL.

The Halden Boiling Water Reactor (HBWR), Halden, Norway, is currently the sole site used by this program for irradiation tests. PIE is being conducted at the AERE-Harwell[b] laboratories in the

(a) FIN: B2043; NRC Contact: H. H. Scott.

(b) Atomic Energy Research Establishment.

United Kingdom. The in-reactor test matrix now spans the full range of normal BWR conditions for pelletized UO_2 fuel, including:

- powers up to 50 kW/m (16 kW/ft)
- diametral gap sizes of 50 to 380 μm (0.002 to 0.015 in.)
- initial gas compositions ranging from pure helium to pure xenon
- fuel densities of 95% and 92% of theoretical density (TD), the latter both stable and unstable regarding in-reactor densification
- burnups to 52 MWd/kgM
- alternate fuel designs (annular fuel pellets, coated cladding, and sphere-pac fuel).

Five instrumented test assemblies have been irradiated thus far in the program. IFA-431 was removed and examined (after 5.5 MWd/kgM peak burnup) in 1977-1978. IFA-527 was removed in April 1981 (at 1 MWd/kgM) after all six rods were suspected of having pressure leaks at the thermocouple seals. IFA-513 was similarly removed from the reactor after two rod failures in April 1981; peak rod burnups were 12 MWd/kgM. It has remained inactive but operable to date, and its four remaining rods may be restarted prior to the end of the program in fiscal 1985. IFA-432 and IFA-518 are scheduled to continue operation until June 1984, when they will be discharged and destructively examined. It is estimated that the peak burnups in IFA-432 and IFA-518 will be 52 and 27 MWd/kgM, respectively.

TECHNICAL PROGRESS

Work that was completed during the January-March 1984 quarter is discussed by task in the following sections.

TASK A - IRRADIATION EXPERIMENTS

The IFA-518 assembly was restarted in mid-March. The assembly had been removed in January for inspection because it was suspected that Rod A-7 (annular pellet fuel) was leaking. The suspicion arose from the anomalous behavior of an elongation sensor. Sipping tests proved negative, and the anomalous behavior is tentatively attributed to fission gas release rather than cladding failure.(1)

A preplanned Halden reactor scram was conducted in early March. During the scram, the transient response of fuel centerline thermocouples was recorded at a frequency of 2 Hz for IFA-432 and several other test assemblies. The prescram power/temperature conditions for the IFA-432 rods were adjusted to match those for a similar scram conducted in August 1979. The thermocouple transient responses will be compared to assess the possible decalibration over the rod-average burnup interval (15 MWd/kgM).

TASK B - DATA QUALIFICATION AND ANALYSIS

Additional whole pellet analyses for retained fission gas were provided by Harwell from the top and bottom positions of Rods 1 and 6 of IFA-432. The following procedure was used: A 3- to 5-mm thick slice was removed from each pellet; the slice was broken up and thoroughly mixed. Replicate samples were then taken (5% to 15% of mass per sample) and fused in a salt. The gas that was released was then quantitatively analyzed (see Table 1). From these results, the estimated fission gas release was 20% ±2% for Rod 1 and 25% ±4% for Rod 6. These results do not compare well with the rod puncture/gas recovery results for Rods 1 and 6 (5% and 19%, respectively). The retained gas results indicate that significant gas was probably slowly lost through the thermocouple seals during irradiation. In particular, the indicated loss of fission gas from Rod 1 is 75%, which compares closely with the observed 80% loss of helium for this rod. This estimate should be tempered by the fact that some undefined loss of intergranular gas probably occurred during the crushing of the retained gas samples. That loss would bias the retention results low and the gas release results high. The IFA-432 interim PIE report, which had been sent to the NRC for printing, was amended to include this information.

TABLE 1. Estimate of Retained Gas Fractions from Whole Pellet Data

Rod	Pellet	Total Measured Retained Fission Gas,(a) STP $cm^3/g\ UO_2$	Estimated Produced Fission Gas, STP $cm^3/g\ UO_2$	Fraction Retained Fission Gas, %	Fraction Released Fission Gas, %
1	19	672	925	73	27
	19	683	925	74	26
1	39	630	691	91	9
	39	628	691	91	9
	39	569	691	82	18
6	19	580	936	62	38
	19	614	936	66	34
	19	654	936	70	30
6	39	589	707	83	17
	39	621	707	88	12
	39	522	707	74	26

(a) Xenon and krypton.

TASK C - FUEL CODE MAINTENANCE AND IMPROVEMENT

The current version of the FRAPCON-2 steady-state fuel performance computer code (V1M4) was assessed in preparation for issuing an improved version (V1M5). Major areas of improvement included:

- The cladding creep models available to the code (MATPRO-11, Rev. 1, for FRACAS-II and Pankaskie et al. for PELET) appear to be geared to pressurized water reactor (PWR) data on cold-worked tubing. In any case, they definitely overpredict the creepdown for annealed tubing relative to that observed in commercial BWRs using such tubing. The rate and magnitude of the predicted creepdown from the MATPRO model seem to be excessive relative to commercial light-water reactor experience. The call to the model has to be deleted completely to perform successful simulations of HBWR test rods.
- The MATPRO fuel swelling model accessed by FRAPCON does not properly account for accommodation of gaseous fission product swelling by porosity in the high-temperature regions. Mechanical elastic/plastic response to fuel swelling also seems to be poorly modeled, resulting in unrealistic predicted cladding stresses/strains at high burnup. The 1% radial strain limit imposed in V1M3 and V1M4 is only a temporary resolution to this problem.
- A correction for thermal gap sizes for FRACAS-II was made in V1M4; the thermal gaps now match the physical gaps, rather than being zero for all conditions. However, the crack factor on the thermal conductivity used in conjunction with FRACAS-II was apparently developed or accepted in conjunction with the zero gap condition. Now that the gap size has been corrected, FRAPCON-2 with the FRACAS-II option overpredicts fuel temperatures relative to Halden thermocouple data. The interrelated fuel relocation/gap size/effective fuel thermal conductivity models used in conjunction with FRACAS-II need to be reassessed.

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS

Preparations are under way for the final PCI deformation/failure test on irradiated PWR Zircaloy cladding. The fuel rod simulators and the traveling on-line diameter measurement system described earlier will be used.[2] This final test will feature preflawed cladding and iodine injection during the deformation to simulate a rod undergoing thermal feedback and fission product release late in life.

Prior to the test with irradiated tubing, a full-scale "proof test" will be conducted using unirradiated tubing in the fuel rod simulator. During this proof test, the iodine injection equipment/procedures and the rod failure detection equipment will be thoroughly checked out. A preflawed nonirradiated tube has been fabricated with defects 1.5 in. long and 0.010 in. wide; defect depths are equivalent to 20%, 40%, and 60% of the tubing wall thickness.

Three of the five irradiated cladding tubes that were tested in fiscal 1983 were examined ceramographically using SEM. No evidence of PCI-type cracks was observed on the inner cladding surfaces.

FUTURE WORK

Next quarter the scram data from the remaining IFA-432 rods will be analyzed and compared with previous scram data to assess thermocouple decalibration. Work will continue on V1M5 of the FRAPCON-2 computer code. The PCI simulation test using irradiated cladding and iodine (PCII-6) and the companion proof test using nonirradiated preflawed cladding will be conducted.

REFERENCES

1. Edler, S. K., ed. 1984. *Reactor Safety Research Programs Quarterly Report, October-December 1983.* NUREG/CR-3307, Vol. 4, PNL-4705-4, Pacific Northwest Laboratory, Richland, Washington.

2. Barner, J. O., et al. 1983. "Ex-Reactor PCI Experiments." In *Proceedings of Eleventh Water Reactor Safety Research Information Meeting,* NUREG/CR-0048, Vol. 3, pp. 504-507.

ACCELERATED PELLET-CLADDING INTERACTION MODELING[(a)]

R. E. Williford, Project Manager

D. D. Lanning

SUMMARY

Developmental efforts for the U.S. Nuclear Regulatory Commission (NRC)/Pacific Northwest Laboratory (PNL) fuel failure code were completed with the implementation of new submodels for statistical predictions, cladding radiation hardening, and chemically assisted cladding microcrack nucleation. Code benchmarking is now in progress. Simulation of pressurized water reactor (PWR) fuel rod transients at low burnup showed no propensity for fuel failure.

INTRODUCTION

This PNL program is divided into two tasks with the following objectives:

- To complete a pellet-cladding interaction (PCI)-related fuel failure model for NRC policy use, to assess the model, and to report the results.
- To coordinate efforts with the PCI fuel failure experiments in Task D of FIN B2043.

The resulting code will predict the probability of fuel rod failure under normal reactor operating conditions and for events described by Chapter 15 of the Safety Analysis Review. Four major components were developed and implemented in the fuel failure code (GT2-F) in fiscal 1983: a transient temperature calculator; a mechanical model to describe the cladding stress concentrations caused by cracked fuel pellets; a submodel for corrodent (iodine) release and/or inventory during steady-state and transient conditions; and three cladding fracture process submodels.

The transient temperature submodel was based on work conducted under FIN B2043. A new constitutive equation was developed for the mechanical model to properly account for the effects of the nonlinear mechanical behavior of cracked fuel on cladding ridge formation. Results of this submodel show that the largest cladding stress concentrations are not always associated with the smallest gap size for a given fuel rod power rating. The steady-state corrodent (iodine) gas release model was developed from the ANS 5.4 fission gas release model and accounts for the decay of unstable isotopes. The transient iodine release model is based on the direct electrical heating experiments performed at Argonne National Laboratory.

Two cladding fracture process submodels were developed and implemented; these submodels describe 1) nonchemically assisted (slower) creep cracking and 2) chemically assisted (faster) stress corrosion cracking (SCC). These two submodels represent the lower and upper bounds of possible cladding fracture mechanisms. The third fracture submodel produces a best-estimate calculation and describes the transition between the other two fracture submodels as the corrodent concentration increases.

At NRC's request, the first working version of the fuel failure code was completed in late July 1983. The results indicated that the fracture submodels can adequately bound ramp test data and simulate failure events given the proper initial flaw size.

(a) FIN: B2452; NRC Contact: H. H. Scott.

TECHNICAL PROGRESS

Progress made during the January-March 1984 quarter is described below.

Three new submodels were implemented into the fuel failure code. The statistical submodel is based on open literature data for pre-existing flaws in Zircaloy cladding and generally shows that failure probabilities of less than 5% agree with the results of fuel rod power ramping experiments. The submodel for the burnup dependence of the cladding yield stress is also based on open literature data and predicts the saturation of radiation hardening effects at about 5.5 MWd/kgm, given an initial yield stress for unirradiated cladding. The cladding microcrack nucleation submodel was implemented to assess the effects of corrodents (iodine) on the formation of new cracks as a function of burnup. This submodel generally predicts that microcracks will probably not complete nucleation until after about 20 to 30 MWd/kgm if protective oxides are present on the inner surface of the cladding. Implementation of these submodels completes the development stage of the code, and efforts are now being directed toward code benchmarking and verification.

Two test cases were run to simulate a PWR Uncontrolled Control Rod Withdrawal (UCRW) situation from low power at low burnup. The first case reached a peak power of 76.6 kW/m at 19 s into the transient, and the second case reached a peak power of 50 kW/m at the same time. Preliminary results showed that the peak fuel temperatures for the two cases (2250°C and 1530°C, respectively) occurred about 24 s into the transient. Peak cladding hoop stresses for the low-power case momentarily reached yield, and yield was obtained for about 16 s in the high-power case. However, both cases predicted that no cladding cracks would initiate at 3 MWd/kgm; and, consequently, there is no propensity for fuel rod failure.

FUTURE WORK

Code benchmarking will continue with the simulation of the low-power PWR UCRW at higher burnups. Parametric cases will be run to benchmark the code against the Demo Ramp II results for higher burnup rods.

PIPE-TO-PIPE IMPACT(a)

M.C.C. Bampton, Project Manager

J. M. Alzheimer
J. R. Friley
F. A. Simonen

SUMMARY

General algebraic expressions for load and crush volume as functions of crush depth were calculated based on last quarter's testing.

INTRODUCTION

The objective of the Pipe-to-Pipe Impact Program is to provide the U.S. Nuclear Regulatory Commission (NRC) with experimental data and analytical models for making licensing decisions regarding pipe-to-pipe impact following postulated breaks in high-energy fluid system piping. Current licensing criteria—as contained in Standard Review Plan 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with Postulated Rupture of Piping"—will be evaluated. Data will be obtained from a series of tests in which selected pipe specimens with appropriate energies will be impacted against stationary specimens to achieve required damage levels.

TECHNICAL PROGRESS

Progress made during the January-March 1984 quarter is summarized below.

The static crush data collected last quarter on 2-in. pipe with various wall thicknesses were used to obtain general algebraic expressions for load and crush volume as functions of crush depth. These general expressions will be utilized in the crush/bend model to predict pipe damage in future pipe-to-pipe impact experiments.

Initially, several types of curve fits were tried for fitting load and crush volume data. The following expressions were chosen:

$$P = \sigma_y \; d^2 \left\{ 0.43 \left(\frac{t}{d}\right) + 19.77 \left(\frac{t}{d}\right)^2 \right\} \sqrt{\frac{\delta}{d}}$$

$$\frac{V}{d^3} = \left[0.869 + 0.5052 \left(\frac{t}{d}\right)\right] \left(\frac{\delta}{d}\right)^2 + \left[1.2495 - 11.09 \left(\frac{t}{d}\right)\right]\left(\frac{\delta}{d}\right)$$

where P = pipe crush load
σ_y = yield stress of piping material
t = pipe wall thickness
d = pipe outside diameter
δ = crush depth
V = crush volume.

(a) FIN: B2383; NRC Contact: G. Weidenhamer.

The numerical coefficients were chosen using an integral least squares fit technique. Since accuracy is necessary for deflections less than about 60% of the pipe outside diameter, experimental data corresponding to deflections greater than 60% were not used, which explains the rather poor fit for the load displacement relationship for large displacements. Fitted data versus experimental data for a typical pipe crush test are plotted in Figures 1 and 2.

FUTURE WORK

Next quarter the new crush data will be incorporated into the crush/bend model. Numerous pipe-to-pipe impact events will be simulated experimentally, and the damage to the struck pipe will be studied as a function of various input parameters.

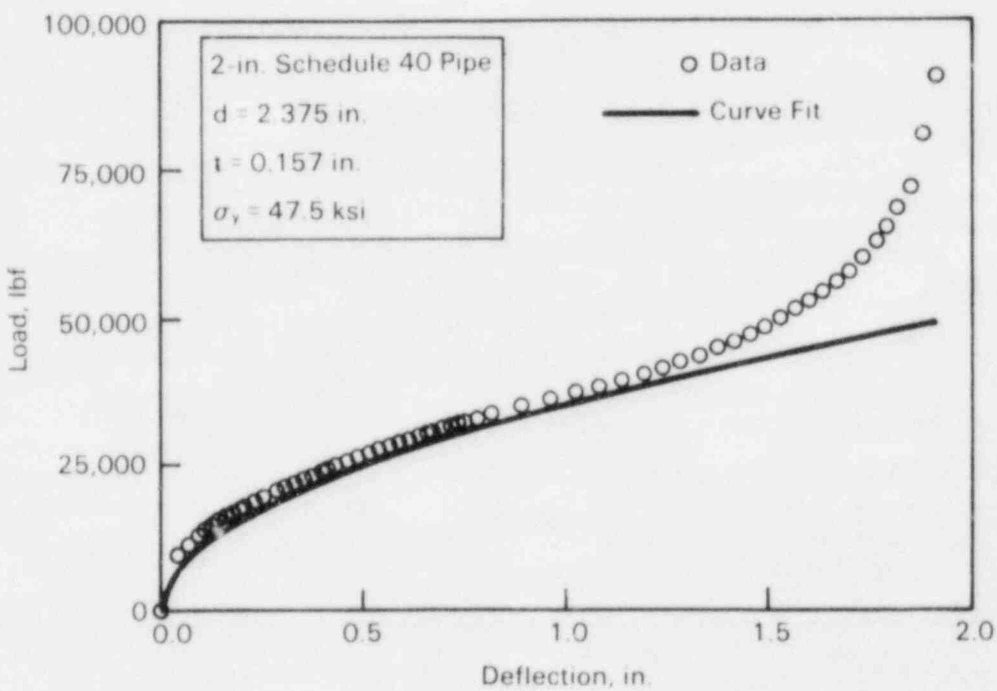

FIGURE 1. Load/Deflection Data for Crush Test No. 13

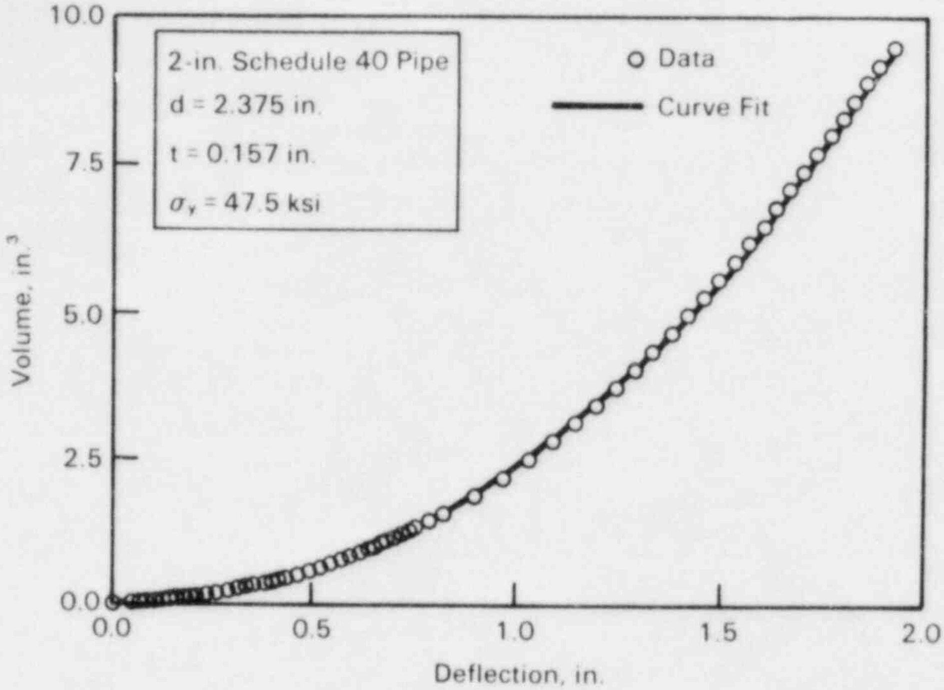

FIGURE 2. Volume/Displacement Data for Crush Test No . 13

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

POWER BURST FACILITY SEVERE FUEL DAMAGE (SFD) TEST PROJECT[a]

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L. R. Bunnell
N. C. Davis
L. L. King
L. J. Parchen
J. O. Vining
G. D. White

SUMMARY

Due to the reduced scope of this project, the limited technical work that was completed this quarter will be included in the next quarterly report.

INTRODUCTION

The Severe Core Damage Subassembly Procurement Program includes the design, development of appropriate materials and supporting fabrication processes, and complete fabrication of fully instrumented test train assemblies for the NRC-sponsored test program at the PBF, Idaho Falls, Idaho. The objective of this PNL program is to study the behavior of light-water reactor fuel under severe high-temperature, flow-starvation conditions. In Phase 1, peak cladding temperatures were limited to 2400K, which included conditions ranging from those anticipated in a design-basis loss-of-coolant accident to those anticipated through the melting point of Zircaloy. Phase 2 tests will run to peak test assembly temperatures of 3100K, the melting temperature of UO_2. Many portions of the PBF Phase 1 and Phase 2 SFD tests should directly benefit the coolant boilaway and damage progession experiments in the NRU reactor due to similarities in the experimental objectives and for materials, instrumentation, and fabrication development.

(a) FINs: B2084, B2456, and B2864; NRC Contact: R. Van Houten.

SEVERE CORE DAMAGE MATERIALS PROPERTY TESTS(a)

J. T. Prater, Project Manager

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E. L. Courtright
J. E. Garnier
C. W. Griffin

SUMMARY

During this quarter, isothermal and transient oxidation experiments were conducted above 1600°C on Zircaloy in steam/helium gas mixtures. Work continued on using TRUMP-II, a two-dimensional finite difference computer code, to model earlier oxidation results.

INTRODUCTION

The objective of this Pacific Northwest Laboratory (PNL) program is to perform high-temperature materials property tests and to provide data that will assist in the planning and analysis of U.S. Nuclear Regulatory Commission (NRC) severe core damage fuel behavior irradiation tests. High-temperature (>1600°C) ex-reactor physical property data and reactor kinetics data are needed on cladding and cladding/fuel/structure reaction products to model rod oxidation behavior and to properly account for the melting and refreezing of the cladding. Zircaloy/H_2O/UO_2 reaction kinetics will be studied, and the viscosities of liquefied fuel for several Zr/UO_2 compositions will be determined.

TECHNICAL PROGRESS

Progress made during the January-March 1984 quarter is described below.

Two sets of Zircaloy oxidation experiments were performed in steam/helium gas mixtures to determine the effects of steam dilution on oxidation kinetics. The results suggest that steam dilution has no effect on Zircaloy oxidation until steam concentrations drop below 30%.

Isothermal oxidation experiments at 1600°C were conducted in steam/helium mixtures ranging from 100 to 20 vol% H_2O and in pure helium. Samples were heated in helium to 1400°C, and then the steam/helium mixture was introduced through a fast-acting valve. The sample temperature was then raised to 1600°C in 8 s and maintained at that temperature for 40 s by controlling the laser heat input to the back of the specimen. Temperature control was ±7°C at 1600°C. The experiment was terminated by closing the shutter to the laser. The thickness of the oxide layer that formed in each gas mixture was then determined by metallography. All oxide layers were 220 ±20 μm except for the specimen tested in helium on which no oxide had formed. Thus, for this experiment, there was no evidence of steam starvation in gas mixtures containing 20% or more steam.

A set of transient experiments was also conducted in steam/helium gas mixtures. Virgin samples were placed in mixtures ranging from 0 to 100% steam and were then rapidly heated to temperature (1600°C) by introducing a constant laser heat input to the back of the sample. The ensuing violent oxidation reaction resulted in sample surface temperatures approaching 2000°C. The peak temperature reached in each gas mixture was recorded. For mixtures ranging from pure steam to 33% steam/67%

(a) FIN: B2455; NRC Contact: R. Van Houten.

helium, the peak excursion temperature was directly related to the vol% H_2O. There was a gradual linear decrease in peak temperature as the helium content increased. This decrease was attributed to the greater thermal conductivity of helium-rich gas mixtures. In gas mixtures with H_2O concentrations below 30%, the maximum temperature began to decrease much more rapidly, suggesting that the onset of steam-limiting conditions may have been reached.

The transient experiments provided an extremely sensitive measure of the kinetics that occur during the most rapid stage of oxidation: the first 3 to 5 s of oxidation of a virgin surface. During this time, steam consumption is at a maximum and steam starvation effects should be the most pronounced. Thus, it is reasonable that the onset of steam starvation occurred at slightly higher H_2O/He ratios in the transient experiments than in the longer term isothermal experiments, where the temperature excursions were limited by controlling the laser input power.

Efforts to model the oxidation reaction are continuing. TRUMP—a finite difference code for predicting transient and steady-state temperature distributions in multidimensional systems—has been modified to calculate the oxidation and thermal distribution behavior of the Zircaloy samples for the geometry used in the oxidation experiments. Oxide layer growth and sample temperature distributions have been successfully predicted for the 1400°C isothermal oxidation experiments. This modified code will now be extended to the isothermal and transient experiments already performed in this program.

Preparations for conducting viscosity measurements on Zr-UO_2 mixtures up to 30 mol% UO_2 have been completed. The experiments will be performed by heating the mixtures to 2100°C and measuring the viscosity of the melts as they heat and as they cool.

FUTURE WORK

- Kinetic studies of the Zircaloy reaction in H_2/H_2O environments will be initiated at pressures up to 150 psi.
- The oxidation kinetics of molten Zircaloy in steam will be determined.
- Laser Raman spectroscopy will be added to the oxidation experiment to study hydrogen blanketing.
- The viscosity of Zr/UO_2 mixtures will be measured at higher temperatures and at greater UO_2 concentrations.
- The capability to perform real-time viscosity data analysis will be developed.

COBRA APPLICATIONS(a)

M. J. Thurgood, Project Manager

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SUMMARY

The technical progress that was completed during this quarter will be reported in the next quarterly report.

INTRODUCTION

The COBRA computer code is being developed for the U.S. Nuclear Regulatory Commission (NRC) to provide better digital computer codes for assessing the behavior of full-scale reactor systems under postulated accident conditions. This Pacific Northwest Laboratory (PNL) project has three main objectives:

- Develop a water reactor primary system simulation capability that can model complex internal vessel geometries such as those encountered in upper head injection (UHI)-equipped pressurized water reactors (PWRs).
- Develop a hot bundle/hot channel analysis capability to evaluate the thermal-hydraulic performance of light-water reactor (LWR) fuel bundles during postulated accidents.
- Develop a containment code capable of simulating the steam/water blowdown and hydrogen distribution phases of an accident.

The resulting codes—COBRA/TRAC, COBRA-TF, and COBRA-NC—are being used to perform pre- and post-test analysis of LWR components and system effects experiments.

COBRA-TF is formulated to model three-dimensional (3-D), two-phase flow using a three-field representation: the vapor field, the continuous liquid field, and the droplet field. The model allows thermal nonequilibrium between the liquid and vapor phases and allows each of the three fields to move at a different velocity. Thus, one can mechanistically treat a continuous liquid core or film moving at a low or possibly negative velocity from which liquid drops are stripped off and carried away by the vapor phase. This feature is essential in the treatment of the hydrodynamics encountered during the reflooding phase of a loss-of-coolant accident (LOCA). The treatment of the droplet field is also essential in predicting other phenomena such as countercurrent flow limiting (CCFL), upper plenum deentrainment and fallback, and two-phase jet impingement.

The code features flexible noding, which allows modeling of complex geometries such as slotted control rod guide tubes, jet pumps, and core bypass regions. These geometries cannot be easily modeled in regular Cartesian or cylindrical mesh coordinates; however, since they have significant impact on the thermal-hydraulic response of the system, these geometries must be modeled with reasonable accuracy.

The fuel rod heat transfer model uses a rezoning mesh to reduce the rod heat transfer mesh size automatically in regions of high heat flux or steep temperature gradients and to increase the mesh size in regions of low heat flux. This model has proven very effective in resolving the boiling curve in the region of the quench front.

(a) FINs: B2391, B2466, and B2041; NRC Contacts: R. Lee and T. Lee.

EVALUATION OF WELDED AND WELD-REPAIRED STAINLESS STEEL FOR LWR SERVICE(a)

D. G. Atteridge Project Manager

S. M. Bruemmer
B. Norton
R. E. Page

SUMMARY

Progress for this program will be reported in a separate quarterly report.

(a) FIN: B2449; NRC Contact: J. Muscara.

COOLANT BOILAWAY AND DAMAGE PROGRESSION EXPERIMENTS IN THE NRU REACTOR(a)

F. E. Panisko, Program Manager
J. P. Pilger, Deputy Project Manager

SUMMARY

Work is continuing in preparation for the MT-6A, MT-6B, FLHT-1, and FLHT-2 tests. Major activities this quarter involved preparing for the MT-6A test, which is scheduled for late May, and for the MT-6B test, which is scheduled for mid-June 1984.

INTRODUCTION

The Coolant Boilaway and Damage Progression (CBDP) Program is being conducted by Pacific Northwest Laboratory (PNL). The program is an extension of light-water reactor (LWR) large-break loss-of-coolant accident (LOCA) simulation testing to evaluate the advanced stages of LOCA scenarios. The coolant boilaway series of experiments will evaluate fuel behavior during a simulated small-break LOCA that results in a partially coolant-uncovered reactor core. Fission heating is used to simulate decay heat generation to boil the coolant and cause damage progression including fission product release in the uncovered fuel rods. Previous experiments were conducted to evaluate the thermal-hydraulic characteristics and fuel rod rupture characteristics of the heatup and quenching phases of a simulated large-break LOCA.

A four-test program is planned that will cover fuel bundle damage behavior from 1090 to 2500K in a series of progressively more severe tests operating at prototypic power densities, thermal gradients, and steam mass fluxes. The four tests currently planned are:

- Materials Test 6A (MT-6A)
- Materials Test 6B (MT-6B)
- Full-Length High-Temperature Test 1 (FLHT-1)
- Full-Length High-Temperature Test 2 (FLHT-2).

The experiments will use full-length pressurized water reactor (PWR) fuel bundle test assemblies and will be performed in the National Research Universal (NRU) reactor at Chalk River, Ontario. Highlights of the test conditions are given in Table 1.

TABLE 1. CBDP Program Test Matrix

Test	Peak Temperature, K	Type of Shroud Insulation	Hydrogen Measurement	Planned Test Date
MT-6A	1090	ZrO_2	No	5/84
MT-6B	1670	ZrO_2	No	6/84
FLHT-1	2150	ZrO_2	Yes	11/84
FLHT-2	2500	ZrO_2/ThO_2	Yes	4/85

(a) FIN: B2277; NRC Contact: R. Van Houten.

The program will develop a well-characterized data set for evaluating the effects of coolant boilaway and core damage progression in an LWR. Coolant boilaway will be achieved using low-level fission heat to simulate the system enthalpy and decay heat expected to drive a postulated coolant boilaway accident. These data will provide a basis for development of accident mitigation strategy, for evaluation of postulated coolant boilaway accidents, for development of concepts for accident prevention and quantifying safety margins, and for development of computer codes using data from separate effects tests.

The following data will be obtained from the CBDP tests:

- fission product release and transport
- temperature distribution for the full-length fuel bundles as a function of liquid level
- fuel bundle damage progression (core degradation) behavior
- cladding melt progression (dissolution and solidification of UO_2)
- core debris and grid spacer interaction
- coolant boilaway behavior
- debris bed formation and coolability
- hydrogen evolution
- flow channel blockage behavior
- inner and outer diameter cladding oxidation and embrittlement
- test train design verification for subsequent tests.

The CBDP test data will be used to confirm the validity of results obtained from separate effects tests that are being sponsored by the NRC. The CBDP experiments are being designed and conducted to utilize the advantages of the NRU such as 1) the capability for testing highly instrumented, multirod 12-ft long fuel bundles under thermal-hydraulic conditions representative of contemporary LWRs, 2) the ability to achieve requisite power densities and axial distributions typical of Three Mile Island-2 (TMI-2) accident conditions using preirradiated fuel rods with commercial enrichment, and 3) the ability to provide prototypic coolant mass fluxes at the fluid/vapor interface typical of a TMI boildown condition. These unique capabilities will reduce uncertainties in length and power distribution scaling factors and the interpretation of the experimental results from small-scale separate effects tests.

The CBDP tests will be the first full-length nuclear heated PWR multirod boilaway tests ever performed. The deformation, rupture, fission product release, and debris bed data can be used to evaluate LWR accident code models and help quantify the conservatism in safety limits used in the nuclear industry. The results of this program will provide the critical length-dependent information for the severe fuel damage data base and expand the data to be obtained from tests at the Annular Core Research Reactor, Albuquerque, New Mexico; the Power Burst Facility, Idaho Falls, Idaho; and other domestic and foreign laboratories.

TECHNICAL PROGRESS

The major accomplishments during the January-March 1984 quarter were the completion of the MT-6A and MT-6B test assembly designs, hardware procurement, fabrication and assembly of the MT-6A 21-rod instrumented fuel bundle, safety analyses for the MT-6A and MT-6B tests, and checkout of the data acquisition and control system (DACS).

The design of the MT-6A and MT-6B test assemblies was completed, including internal design reviews. A key design feature is a composite shroud consisting of thermal insulation sandwiched between an outer stainless steel component and an inner Zircaloy liner. The ZrO_2 insulation, which is characterized by low density and high thermal resistance and strength, is fabricated and assembled as interlocking tiles. The MT-6A shroud contains 29 Type K thermocouples with hot junctions distributed along the axial region at the stainless steel/ZrO_2 interface and at the ZrO_2/Zircaloy interface.

The main purpose of the shroud is to mechanically support the test bundle while minimizing radial heat losses, which permits the fuel bundle to operate at high temperatures (simulating various LWR accident conditions) while maintaining the main reactor components at normal operating temperatures.

Other important insulated shroud features include:

- two time domain reflectometer (TDR) probes for measuring liquid level in the bundle and outside the shroud above the bundle regions
- fuel rod pressure sensors on each of the 21 MT-6A fuel rods
- independent bundle water feedlines for MT-6B
- large-diameter Zircaloy-to-stainless steel transition pieces at the top and bottom ends that seal weld the Zircaloy inner shroud liner to the outer stainless steel component.

Transverse cross sections of the MT-6A and MT-6B test assemblies are shown schematically in Figures 1 and 2.

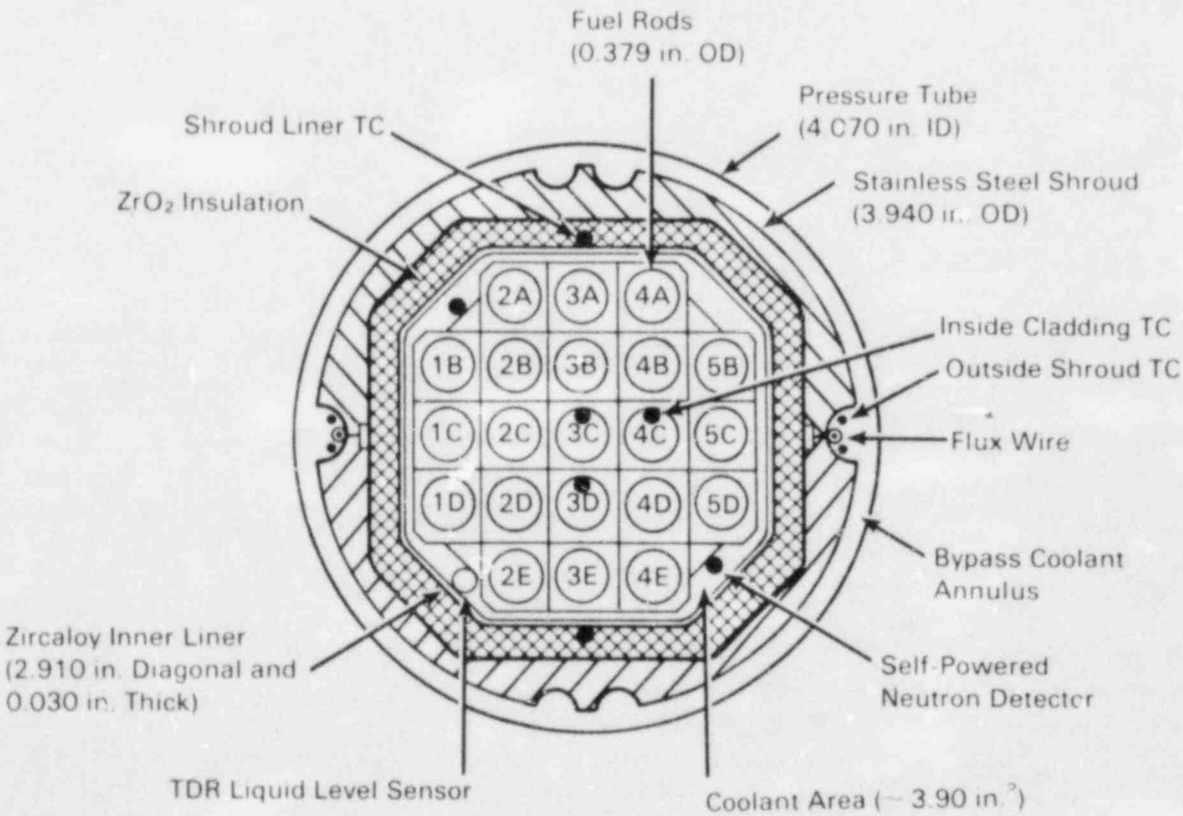

FIGURE 1. Cross Section of MT-6A Test Assembly

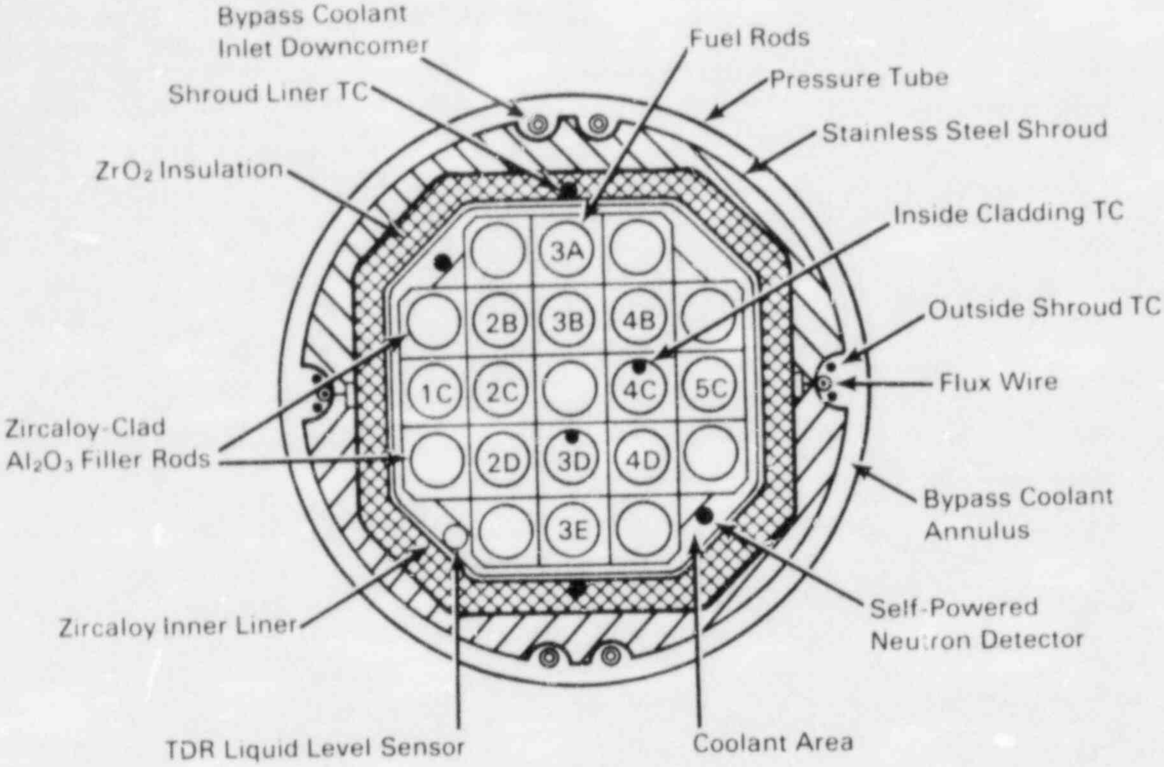

FIGURE 2. Cross Section of MT-6B Test Assembly

All hardware procurement orders for the MT-6A and MT-6B test trains were placed, and most of the hardware was received for test assembly fabrication. All 21 MT-6A fuel rods were fabricated and assembled as shown in Figure 1. The MT-6B fuel bundle is about 50% complete. The fabrication of the MT-6A shroud is about 90% complete and the MT-6B shroud fabrication was started. The fabrication of four instrumented fuel rods for MT-6B required weld development to attach the long Zircaloy/tantalum sheath Type C (W-5Re/W-26Re) thermocouples to the inner surfaces of the fuel rod cladding.

Safety analyses for the MT-6 tests addressed fission product release, transport, deposition, and associated radiation levels; hydrogen generation; loss of coolant from various parts of the test assembly during critical test times; and possible fuel oxidation, fragmentation, relocation, and associated thermal-hydraulic and neutronic effects. These safety analyses are used by Chalk River Nuclear Laboratories (CRNL) safety engineers to obtain test approval from the Canadian Nuclear Safety Analysis Committee (NSAC). Approval was obtained for the MT-6A test, but additional analyses are needed before approval for MT-6B will be granted. These analyses are in progress at PNL and CRNL.

The hardware and software for the DACS were checked out and brought to operational status. This computer system stores, analyzes, displays, and checks as many as 244 signals from thermocouples, TDRs, and pressure and flow transducers at rates as high as 10 times per second during the test. This hardware/software package will be upgraded after the MT-6B test to provide expanded capabilities for the upcoming FLHT tests.

FUTURE WORK

During the next quarter, the following work will be completed:

- MT-6A test train fabrication and irradiation test
- MT-6B test train fabrication and irradiation test
- FLHT-1 safety analyses
- system configurational design for the new DACS (overall software/hardware interfacing and control logic).

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16. ABSTRACT (200 words or less)

This document summarizes work performed by Pacific Northwest Laboratory from January 1 through March 31, 1984, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Results from an instrumented fuel assembly irradiation program being performed at Halden, Norway, are reported. Accelerated pellet-cladding interaction modeling is being conducted to predict the probability of fuel rod failure under normal operating conditions. Experimental data and analytical models are being provided to aid in decision making regarding pipe-to-pipe impacts following postulated breaks in high-energy fluid system piping. Fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada.

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