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Reactor Safety Research Programs

Quarterly Report
October-December 1983

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
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U.S. Nuclear Regulatory
Commission

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Prepared by
S. K. Edler, Ed.

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
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ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1983, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation include investigating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems and examining NDE reliability and probabilistic fracture mechanics. 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. Experimental data and validated models are being used to determine a method for evaluating the acceptance of welded or weld-repaired stainless steel piping. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; an instrumented fuel assembly irradiation program is being performed at Halden, Norway; and fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

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

P. H. Hutton, Project Manager
R. J. Kurtz, Assistant Project Manager

SUMMARY

Progress made during this quarter will be presented in a topical report.

INTRODUCTION

The purpose of this Pacific Northwest Laboratory (PNL) program is to provide an experimental evaluation of the feasibility of detecting and analyzing flaw growth in reactor pressure boundaries on a continuous basis using acoustic emission (AE). Type A533B, Class 1 pressure vessel steel, and SA351-CF-8A cast stainless, Type 304 wrought, and A106 ferritic piping steels are being used in experimental testing. Objectives of this program are to:

- develop a method to identify crack growth AE signals in the presence of other acoustic signals
- develop a relationship to estimate flaw significance from AE data
- develop an instrument system to implement these techniques
- demonstrate the total concept off-reactor and on-reactor.

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

INTEGRATION OF NONDESTRUCTIVE EXAMINATION RELIABILITY AND FRACTURE MECHANICS[a]

S. R. Doctor, Program Manager
F. A. Simonen, Project Manager
T. T. Taylor, Project Manager

L. A. Charlot
H. R. Hartzog
P. G. Heasler
G. A. Mart

SUMMARY

Progress made during this quarter will be reported in a topical report.

INTRODUCTION

The primary pressure boundaries (pressure vessels and piping) of nuclear power plants are inspected in-service according to the rules of the ASME Boiler and Pressure Vessel Code, Section XI (Rules for In-Service Inspection of Nuclear Power Plant Components). Ultrasonic techniques are normally used for these inspections, which are periodically performed on a sampling of welds. The Integration of Nondestructive Examination (NDE) Reliability and Fracture Mechanics Program at Pacific Northwest Laboratory (PNL) was established to determine the reliability of current in-service inspection (ISI) techniques and to develop recommendations that will insure a suitably high inspection reliability. The objectives of this U.S. Nuclear Regulatory Commission (NRC) program are to:

- determine the reliability of ultrasonic ISI performed on commercial light-water reactor primary systems
- using probabilistic fracture mechanics analysis, determine the impact of NDE unreliability on system safety and determine the level of inspection reliability required to insure a suitably low failure probability
- evaluate the degree of reliability improvement that could be achieved using improved and advanced NDE techniques
- based on material properties, service conditions, and NDE uncertainties, formulate recommended revisions to ASME Code, Section XI, and Regulatory Requirements needed to insure suitably low failure probabilities.

The scope of this program is limited to ISI of primary systems, and the results and recommendations are also applicable to Class II piping systems.

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

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

D. D. Lanning, Program Manager

D. D. Lanning, Task A Leader
M. E. Cunningham, Task B Leader
W. N. Rausch, Task C Leader
J. O. Barner, Task D Leader

E. R. Bradley
R. J. Guenther
R. E. Williford

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 a leak was suspected in Rod A-7 of IFA-518 from a routine survey of elongation sensor responses to power level changes. IFA-518 will be temporarily removed from the reactor for inspection and sipping. The IFA-527 postirradiation examination (PIE) data report[1] was submitted to the U.S. Nuclear Regulatory Commission (NRC) for publication, and the final sections of the IFA-432 PIE report were completed. Plans were made for the final pellet-cladding interaction (PCI) deformation/failure tests on irradiated pressurized water reactor (PWR) Zircaloy cladding using fuel rod simulators. These tests will involve iodine injection during the deformation, and a cladding PCI-type failure is anticipated.

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 United

(a) FIN: B2043; NRC Contact: H. H. Scott.
(b) Atomic Energy Research Establishment.

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 has been completed during the October-December 1983 quarter is discussed by task in the following sections.

TASK A - IRRADIATION EXPERIMENTS

The irradiation of IFA-432 and IFA-518 continued through the quarter. Rod elongation sensor data revealed anomalous behavior in Rod A-7 of IFA-518 that could be a result of a cladding/sensor leak and pressure communication with the reactor coolant system. A pressurized bellows is attached to the elongation linear variable differential transformer (LVDT) core inside the rod. This arrangement was designed to act as a failure indicator; if the bellows collapse (due to abnormally high rod internal pressure), the LVDT will register a signal opposite in sign to normal rod elongation. The elongation readings for Rod A-7 indicated a slow collapse of the bellows over a period of two months, implying that the internal rod pressure exceeded 2.1 MPa (300 psi), which is the bellows internal pressure at 513K. Such a high pressure could be caused by cladding failure and pressure communication with the coolant system (which is at a pressure of 3.4 MPa or 500 psi). However, that pressure could also occur due to fission gas release within the rod. Hand calculations and computer code simulations indicated that 15% to 25% release of produced gas would cause a pressure of 2.1 MPa or greater in Rod A-7 at its current burnup (18 to 20 MWd/kgM). This estimate is corroborated by the behavior of sibling rods that were ramped to high power in IFA-517 and experienced significant fission gas release that caused the bellows to collapse.

TASK B - DATA QUALIFICATION AND ANALYSIS

IFA-518 and IFA-432 in-reactor performance data tapes for March 14 through October 22, 1983, have been received and processed. The tapes will be analyzed next quarter. This quarter, the IFA-527 PIE data report was submitted to NRC for publication. The final sections of the IFA-432 interim PIE report (covering Rods 1, 6, and 8) were completed; the report is being reviewed and will be submitted to NRC next quarter.

TASK C - FUEL CODE MAINTENANCE AND IMPROVEMENT

Commitments in other areas slowed progress on FRAPCON (V1M5) development this quarter. Continued analysis of FASTGRASS/PARAGRASS fission gas release models as compared with in-reactor data has revealed several possible deficiencies in these models. A grain growth model should definitely be added to FRAPCON to run in conjunction with these models (and with the ANS 5.4 gas release model). Not accounting for grain growth generally makes these models overpredict local gas release because grain size is underestimated; therefore, gas diffusion from grains may be overestimated for certain cases.

FRAPCON should also include a grain boundary gas sweeping model to apply to high grain growth regions and columnar grain formation/sweeping. The lack of such a model means that a potentially important mechanism (especially in overpower conditions) is being ignored, leading to probable underprediction of the total gas release for certain cases.

Other gas release phenomena, including the development of intergranular retained gas and the mechanisms leading to its direct release, should be examined more thoroughly.

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS

Preparations are under way to perform a final PCI deformation/failure test on irradiated PWR Zircaloy cladding, using the fuel rod simulators and the traveling on-line diameter measurement system described earlier.[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. A PCI-type cladding failure is anticipated. 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 checked out thoroughly.

The five irradiated cladding samples that were tested in fiscal 1983 are being examined. The examinations include radiography, ceramography, and limited scanning electron microscopy to locate and characterize any incipient cracks in the tubing.

FUTURE WORK

Next quarter, IFA-518 will undergo a sipping test at Halden and will be returned intact to the reactor if possible. V1M5 of FRAPCON-2 will near completion, and the destructive examinations should be completed on PCI Tests 1-5. The preparations for PCII-6 (iodine injection) and the companion proof test will be nearly complete.

REFERENCES

1. Cunningham, M. E., and D. D. Lanning. 1984. *Measured In-Reactor Data and Postirradiation Observations for IFA-527.* NUREG/CR-3070, PNL-4542, 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/CP-0048, Vol. 3, pp. 504-507.

ACCELERATED PELLET-CLADDING INTERACTION MODELING[(a)]

R. E. Williford, Project Manager

D. D. Lanning

SUMMARY

The development and documentation of a cladding damage/microcrack nucleation model was completed. This model is being implemented in the GT2-F fuel failure code.

INTRODUCTION

This Pacific Northwest Laboratory (PNL) program is divided into two tasks with the following objectives:

- To complete a pellet-cladding interaction (PCI)-related fuel failure model for U.S. Nuclear Regulatory Commission (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, ZrI_4) 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 October-December 1983 quarter is described below.

Most models of the SCC mechanism that have been proposed to explain fuel rod failures address the crack propagation and cladding rupture stages and neglect the nucleation stage for intergranular microcracks small enough to violate the continuum assumptions of fracture mechanics. Microcrack nucleation is necessary to provide sites for subsequent crack processes.

In the first quarter of fiscal 1984, diffusion-controlled grain boundary cavitation concepts were used to approximate this microcrack nucleation process. The effects of protective oxide films, film rupture criteria, adsorption phenomena, the Zircaloy-iodine chemical reaction, and cladding embrittlement by ZrI_4 in a process similar to grain boundary segregation theories are modeled. The model simulates an apparent threshold behavior by rapid changes in process rates. Documentation of this model has been completed with the following results:

- Chemically assisted microcrack nucleation in Zircaloy fuel rod cladding is determined by a complex interrelationship between grain boundary void growth rates, stress, chemical reaction rates, and the protective/rupture characteristics of the oxide film.
- Rapid changes in microcrack nucleation process rates result in an apparent threshold behavior over a narrow range of iodine concentrations that are in acceptable agreement with the experimentally determined threshold concentration. Microcracks can also nucleate at stresses well below the experimentally determined apparent threshold stress for other SCC fracture processes.
- Cladding microcrack nucleation times depend on fuel rod stresses and fission gas release histories, and nucleation may occur after about 20 MWd/kgM (440 to 660 days of operation), which is consistent with recent fuel rod power transient test results.
- Much of the scatter in the computed microcrack nucleation times is caused by uncertainties in the Zircaloy-iodine reaction rate, uncertainties that originate from differences in materials and conditions. Lesser uncertainty contributions result from rod design (total internal free volume).
- The presence of oxygen appears to retard the Zircaloy-iodine reaction rate, and iodine may participate in a dissolution process that assists the assumed embrittlement mechanisms under some conditions.

FUTURE WORK

The microcrack nucleation model will be implemented in the fuel failure code, and documentation and benchmarking efforts will continue.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

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

SUMMARY

During the last quarter, 16 crush tests were performed to provide input to the crush/bend modeling effort.

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 September-December 1983 quarter is summarized below.

A 200,000-lb capacity test cell was designed and built to perform static crush tests on pipe specimens. The load cell currently being used to measure loading force limits the device to 100,000-lb capacity, which is adequate for the small-scale (2-in.) specimens being tested.

A total of 16 crush tests were performed. All specimens were 2-in. nominal pipe made from 106 Grade B carbon steel. The 16 tests consisted of four replications of a set of four thickness-to-diameter (t/d) ratios: 2-in. Schedule 40, 80, and 160 and a thinned size simulating the t/d ratio of a 6-in. Schedule 40 pipe. Loading was performed with a round solid platen, the axis of which was perpendicular to that of the specimen, thus simulating the condition of pipes impacting at right angles.

Two types of data were taken: crush load/deflection data and crush volume/deflection data. The crush volume/deflection data compensate, in an approximate sense, for the effects of pipe contents on load/deflection data. Typical test results are shown in Figures 1 and 2.

FUTURE WORK

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

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

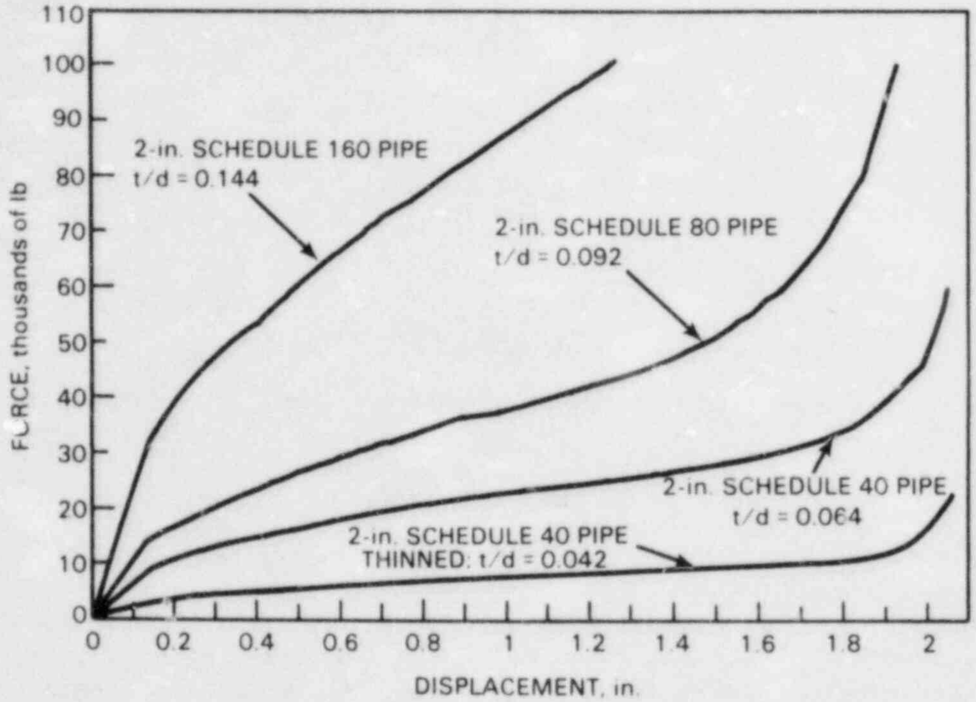

Figure 1. Crush Load/Deflection Data for 2-in. Pipes

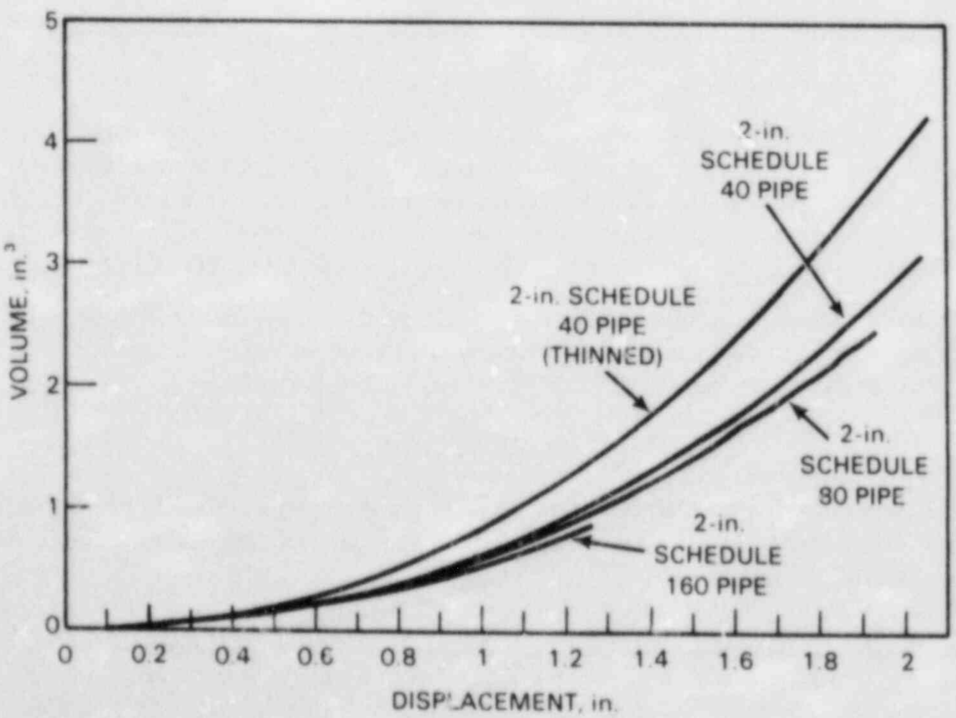

Figure 2. Crush Volume/Deflection Data for 2-in. Pipes

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

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

R. L. Goodman, Program Manager

G. S. Allison
A. J. Anthony
L. R. Bunnell
N. C. Davis
L. L. King
L. J. Parchen
J. O. Vining
G. D. White

SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) placed a hold on all further design and development work on the Power Burst Facility (PBF) Phase 2 test train assemblies effective November 1, 1983, anticipating possible termination of the Phase 2 experimental program. Work at both Pacific Northwest Laboratory (PNL) and EG&G Idaho was stopped. Most of the material and instrument components procured in support of the Phase 2 test train assemblies have been delivered to PNL. They are being stored for possible future use on either the Phase 2 PBF SFD program or other NRC-sponsored experimental programs; many of the instrument components will be used in support of the coolant boilaway and damage progression experiments in the National Research Universal (NRU) reactor.

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 progression experiments in the NRU reactor due to similarities in the experimental objectives and for materials, instrumentation, and fabrication development.

FUTURE WORK

A very limited generic thoria crucible development program will be continued in support of SFD high-temperature tests to 3100K.

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

SEVERE CORE DAMAGE MATERIALS PROPERTY TESTS[a]

E. L. Courtright, Project Manager

L. R. Bunnell
J. E. Garnier
C. W. Griffin
J. T. Prater

SUMMARY

During this quarter, isothermal and transient oxidation measurements were conducted on Zircaloy in steam at 1600 to 1800°C. Metallography was performed on zirconium with 5 to 10 mol% UO_2 heated to temperatures of 2000°C.

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 October-December 1983 quarter is described below.

A second series of isothermal oxidation experiments for Zircaloy in steam at 1597, 1687, and 1779°C has been completed. The results are in excellent agreement with those obtained in previous experiments.[1] The ZrO_2 oxide and α-Zr layer growth kinetics are parabolic.

Near isothermal control was achieved by heating the specimen in argon and then introducing steam through a fast-acting valve. The ensuing oxidation produced a rapid heating of the sample to the desired temperature; this temperature was maintained by controlling the laser heat input to the back side of the sample. Temperature control during the initial few seconds of the transient was reproducible to ±35°C for the 1597°C and 1687°C experiments and to ±20°C at 1779°C. Subsequent temperature control during the isothermal portion of the experiment was ±7°C at 1597°C, ±10°C at 1687°C, and ±18°C at 1779°C. The small amount of oxidation that occurred during the initial exothermic reaction that brought the sample to temperature was measured independently and subtracted to obtain the isothermal growth kinetics.

The isothermal oxidation results are in good agreement with previous work by Urbanic, although they are slightly higher. The difference is believed to be due to the unique experimental configuration that was adopted to closely simulate the actual conditions experienced by a fuel rod. Further oxidation experiments will be conducted to quantify the differences. Isothermal oxidation experiments are also planned to extend the work to temperatures above the melting point of Zircaloy.

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

Transient oxidation experiments have been conducted to examine the exothermic reaction and relate its kinetics to the isothermal data. Initially, the effect of heating rate in steam on subsequent thermal transients was examined. Rapid heating to temperatures above 1400°C resulted in a violent oxidation reaction that can drive sample surface temperatures to 2100°C. Slower heating rates, which permitted some oxide growth prior to the transient, moderated the oxidation reaction and the ensuing temperature rise. To model this phenomena, the TRUMP computer code—a finite difference code for predicting transient and steady-state temperature distributions in multidimensional systems—has been modified to permit calculation of the thermal distribution of the oxidation samples as a function of time. Both the isothermal and transient experiments will be modeled.

The oxidation apparatus has been modified to permit gas mixing so that H_2O/Ar and H_2O/H_2 mixtures can be prepared and studies on steam starvation can be completed. Preliminary Raman spectroscopy experiments are in progress using the recently installed Nd:Yag laser to determine whether this high-powered laser can stimulate sufficient Raman scattering from gaseous species to permit study of hydrogen blanketing effects during the rapid transient oxidation experiments.

The Zr-7.5 mol% UO_2 mixture that was tested in the viscometer to 2000°C has been sectioned for metallographic examination. Complete melting does not appear to have occurred. Recent work by the Germans[2] suggests that this mixture should melt between 1950°C and 2000°C. However, the metallography suggests that there are still undissolved phases present, which may indicate an error in the temperature measurement, too much contamination by the ThO_2 crucible, or inaccurate German data. The compositions of the various phases are being analyzed.

ThO_2 crucibles are being machined in preparation for the next series of viscosity measurements. Measurements will be extended to 2300°C to assure melting, and Zr/UO_2 mixtures with up to 30 mol% UO_2 will be measured. The composition range was expanded to assure that our measurements include all the melt compositions that are likely to occur during fuel melting.

FUTURE WORK

- Kinetic studies of the Zircaloy reaction in Ar/H_2O and H_2/H_2O environments will be initiated at pressures up to 150 psi.
- Laser Raman spectroscopy will be added to the oxidation experiment to study hydrogen blanketing.
- The oxidation kinetics of molten Zircaloy in steam will be determined.
- The viscosity of Zr/UO_2 mixtures will be measured at higher temperatures and at greater UO_2 concentrations.
- The extent of reaction of ThO_2 crucibles with molten Zr/UO_2 mixtures will be determined, and its effect on viscosity measurements will be established.

REFERENCES

1. Edler, S. K., ed. 1984. *Reactor Safety Research Programs Quarterly Report. July-September 1983.* NUREG/CR-3307, Vol. 3, PNL-4705-3. Pacific Northwest Laboratory, Richland, Washington.

2. Skokan, A., et al. 1983. "Investigation of the Phase Equilibria in the Ternary System U-Zr-O." KFK-3350, PNS Jahresbericht, Federal Republic of Germany.

COBRA APPLICATIONS[a]

M. J. Thurgood, Project Manager

T. E. Guidotti
J. M. Kelly
R. J. Kohrt

SUMMARY

The gap conductance model in COBRA/TRAC was assessed against the FRAPCON-2 code, and the most significant heat transfer improvements from COBRA-TF were implemented into COBRA/TRAC. A COBRA-NC simulation of International Standard Problem No. 16 was completed. Significant progress was made on a fully implicit multidimensional two-fluid solution scheme. The assessment of COBRA-TF against both blocked and unblocked bundle data is continuing, and possible fuel rod deformation models are being evaluated.

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

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

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.

TECHNICAL PROGRESS

COBRA/TRAC APPLICATIONS

Three NRC documents were prepared this quarter.[1,2,3] A comprehensive report on the 3-D best-estimate PWR/UHI calculation will be published as NUREG/CR-3642. The COBRA/TRAC prediction for Semiscale Test S-UT-2 was documented in NUREG/CR-3563. This test simulated a small-break (10%) LOCA in a PWR. In addition, a report describing the TEMPEST analysis of pressurized thermal shock will be published as NUREG/CR-3564.

As a result of comparing the TRAC-PF1 and COBRA/TRAC calculations,[4] a discrepancy in "best-estimate" gap conductance values was found. This discrepancy led to discussions with the FRAPCON-2 code developers[a] about how fuel performance code results can be input into COBRA/TRAC to represent a rod at a given burnup. From these meetings it was discovered that the previous gap conductance assessment used an "old" FRAPCON-2 mechanics package that is no longer considered best-estimate. This discovery led to three questions that need to be resolved:

- Which FRAPCON-2 models should be used to obtain best-estimate values of gap conductance?
- How should the COBRA/TRAC input be specified to give the same stored energy as FRAPCON-2 for rods at a given burnup?
- Since the previous assessment was not best-estimate, how good or bad was the gap conductance used in previous PWR/UHI LBLOCA calculations?

The first two questions were resolved. To assess this new methodology, the COBRA/TRAC-calculated results were compared with the "best-estimate" FRAPCON-2 results at three burnup values. Figures 1, 2, and 3 show the stored energy, gap conductance, and fill gas pressure as calculated by both codes and indicate the good comparison that was obtained.

Using the gap conductance calculated by FRAPCON-2 (Figure 2), the answer to the third question can be resolved. The gap conductance (3700 Btu/h-ft^2-°F) and the stored energy used in the UHI calculations were acceptable best-estimate values for a burnup of 9000 MWd/MTU. COBRA/TRAC used an average gap conductance for its best-estimate value. The TRAC-PF1 analysis had assumed a burnup of 16,000 MWd/MTU but used a gap conductance of 1000 Btu/h-ft^2-°F, which is inconsistent with the results shown in Figure 2. A gap conductance of 1000 Btu/h-ft^2-°F corresponds to the lifetime minimum and is too conservative to be best-estimate.

The most significant heat transfer improvements in COBRA-TF were added to COBRA/TRAC. These improvements include:

- increased vapor superheat in the inverted annular film boiling regime
- Westinghouse drop size correlation

(a) PNL staff members M. E. Cunningham, D. D. Lanning, and W. N. Rausch.

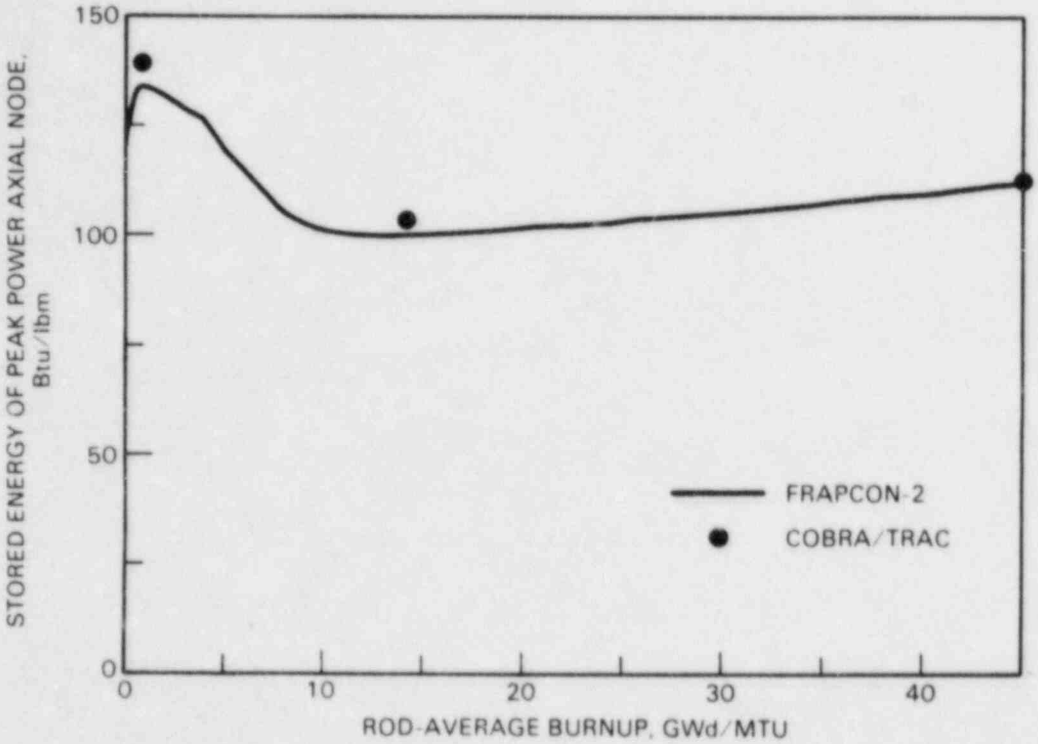

Figure 1. COBRA/TRAC and FRAPCON-2 Values for Stored Energy Versus Rod-Average Burnup

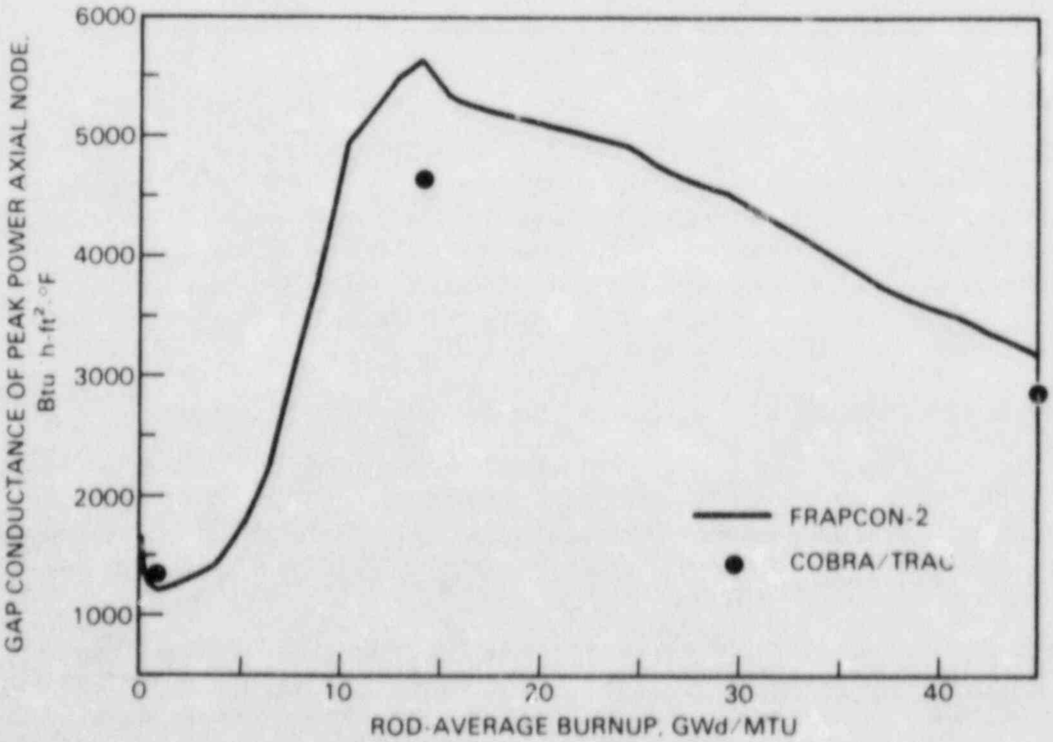

Figure 2. COBRA/TRAC and FRAPCON-2 Values for Gap Conductance Versus Rod-Average Burnup

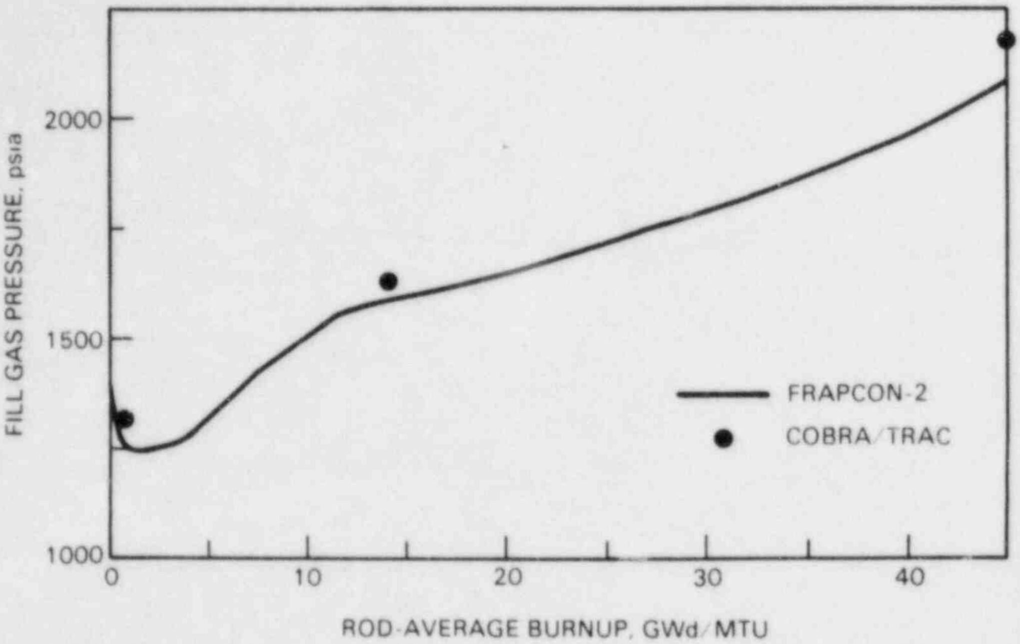

Figure 3. COBRA/TRAC and FRAPCON-2 Values for Fiss Gas Presure Versus Rod-Average Burnup

- removed wet grid effects
- two-phase enhancement of dispersed flow vapor heat transfer coefficient
- FLECHT SEASET vapor heat transfer coefficient
- drop deposition heat transfer rather than Forslund-Rohsenow.

FLECHT-SEASET Test 31805 is being used to compare COBRA/TRAC results with COBRA-TF results. The mesh used in these calculations models a central channel within the bundle; it neglects the region near the cold wall on the outside of the bundle. COBRA/TRAC now gives a good temperature prediction of the 6-ft level and overpredicts the temperature at the 10-ft level instead of underpredicting these temperatures as in the previous assessment calculation. To improve the prediction at the 10-ft level, the grid cooling effects will be modeled by adding an interfacial heat transfer correlation to represent the "small" drops in COBRA-TF.

COBRA-NC CONTAINMENT APPLICATIONS

A post-test calculation has been performed of the HDR steam blowdown Test V44 and will be submitted as the COBRA-NC prediction for International Standard Problem No. 16. A sensitivity study of various input variables was conducted in the process of performing this simulation. The blowdown source liquid content and drop size, loss coefficients for various flow openings, and the amount of drop carryover between rooms were studied.

The short and medium time period predictions of the absolute pressure in the blowdown room are shown in Figures 4 and 5. The two dashed lines in Figure 4 represent the range of oscillations in the experimental data. The COBRA-NC prediction represents a high average between the two limits of the measured data. The medium time period pressure response is slightly overpredicted (Figure 5). The Uchida correlation was used to calculate the condensation heat transfer coefficient.

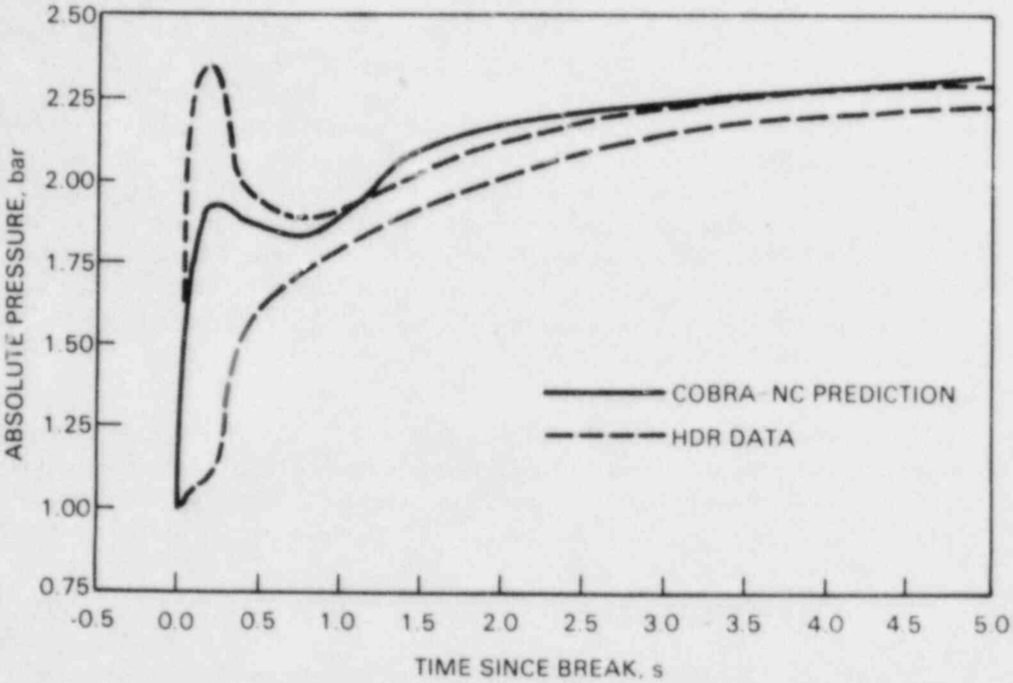

Figure 4. Short Time Period Pressure Response of Blowdown Room

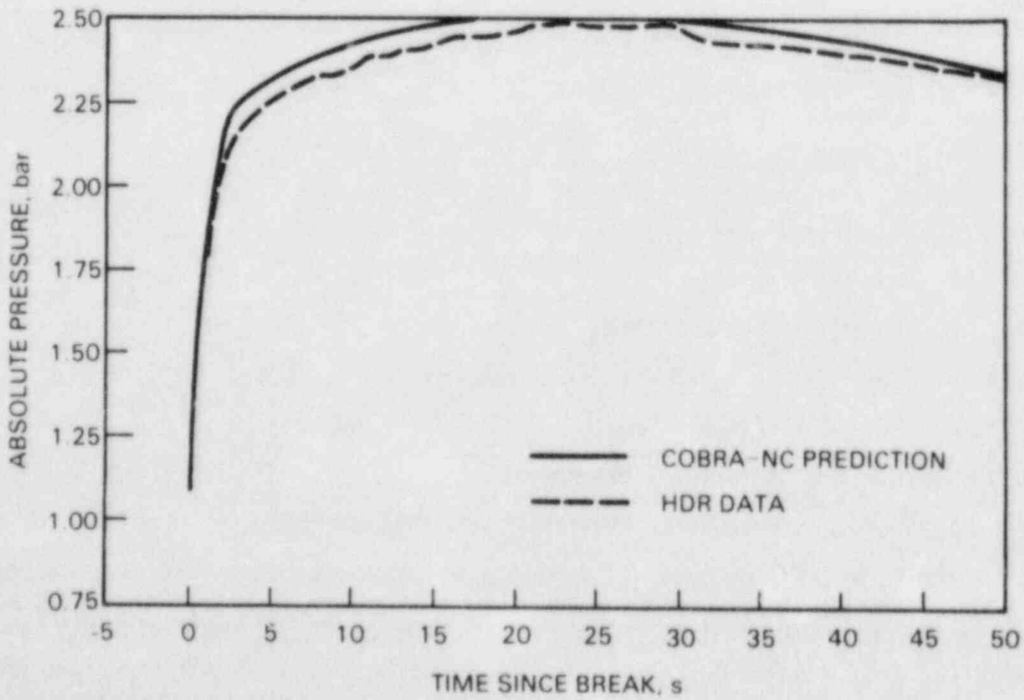

Figure 5. Medium Time Period Pressure Response of Blowdown Room

Through the sensitivity studies, it was found that the results were most sensitive to the liquid content of the blowdown flow, the value of the loss coefficient between rooms, and the amount of liquid carryover between rooms. The drop size was not an overly sensitive parameter when varied within a reasonable physical range. The initial dip in the pressure between 0.5 and 1.0 s seems to be related to the liquid content of the blowdown flow and the drop carryover between rooms. Inaccuracies in the measured blowdown flow rate and enthalpy during this time period have a significant impact on the dip in the pressure response. The amount of liquid carryover determines the increase in the pressure after 1.0 s. The prediction was most sensitive to the value of the loss coefficient between the blowdown room and two of the adjacent rooms. Unphysically high values had to be used for the flow opening containing the blowdown nozzle. It is believed that the high energy of flow exiting the blowdown pipe offers an additional resistance to flow out this opening that cannot be represented by the normal geometric loss coefficient.

A 2-D input deck has been set up to model the DEMONA Test X2. Initial runs indicate that the steam jet velocities are sufficiently high to require small time step sizes because of the Courant limitations. These time steps would require too much computer time for the 55-h transient. Therefore, a method has been devised to obtain an implicit thermal solution for the structures. An explicit transient will then be run for the period of air injection only.

A 3-D input deck for the HDR Test V44 blowdown room is nearly complete. This model will be used to study the flow patterns within the room, the droplet deposition, and the effect of the blowdown jet on the pressure distribution and flow in front of the vent containing the blowdown nozzle.

HOT BUNDLE CODE DEVELOPMENT

Work performed this quarter concentrated on three tasks:

- flow blockage heat transfer
- rod deformation modeling
- implicit fluid solution.

A brief description of the activities for each task follows.

Flow Blockage Heat Transfer

Five subtasks are included in this task:

- Subtask 1 - simulate FLECHT-SEASET 21-rod unblocked bundle reflood data
- Subtask 2 - simulate FEBA 5 x 5 grid effects reflood tests
- Subtask 3 - simulate FLECHT-SEASET 21-rod bundle flow blockage tests
- Subtask 4 - simulate FEBA 5 x 5 flow blockage tests
- Subtask 5 - perform blind post-test predictions for the FLECHT-SEASET 163-rod blocked bundle.

Work was performed in Subtasks 1 through 4 this quarter, and Subtask 1 is now complete. Two 21-rod unblocked bundle simulations were run: 42606A (40 psi; 0.9 in./s) and 43208A (40 psi; 1.5 in./s). Both simulations resulted in good data comparisons as illustrated in Figures 6 and 7. The overprediction of the rod temperatures at 84 in. and 96 in. for the 21-rod bundle (discussed in the July-September quarterly report)[4] was traced to the quenching of the steam probes attached to the 82-in. grid spacer. This overprediction can be seen in Figure 7 just downstream of the 82-in. grid. Instead of attempting to model this phenomena, only data downstream of this grid prior to the probe quench time were used.

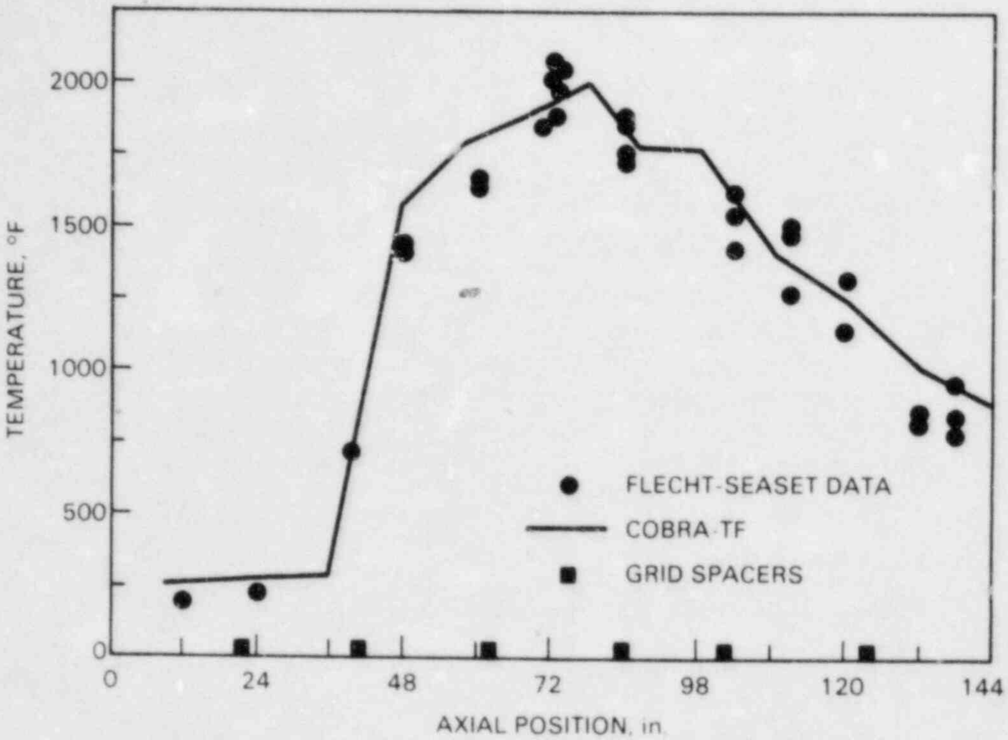

Figure 6. Axial Profile of Rod Temperatures at 100 s for FLECHT-SEASET 21-Rod Unblocked Bundle Test 42606A

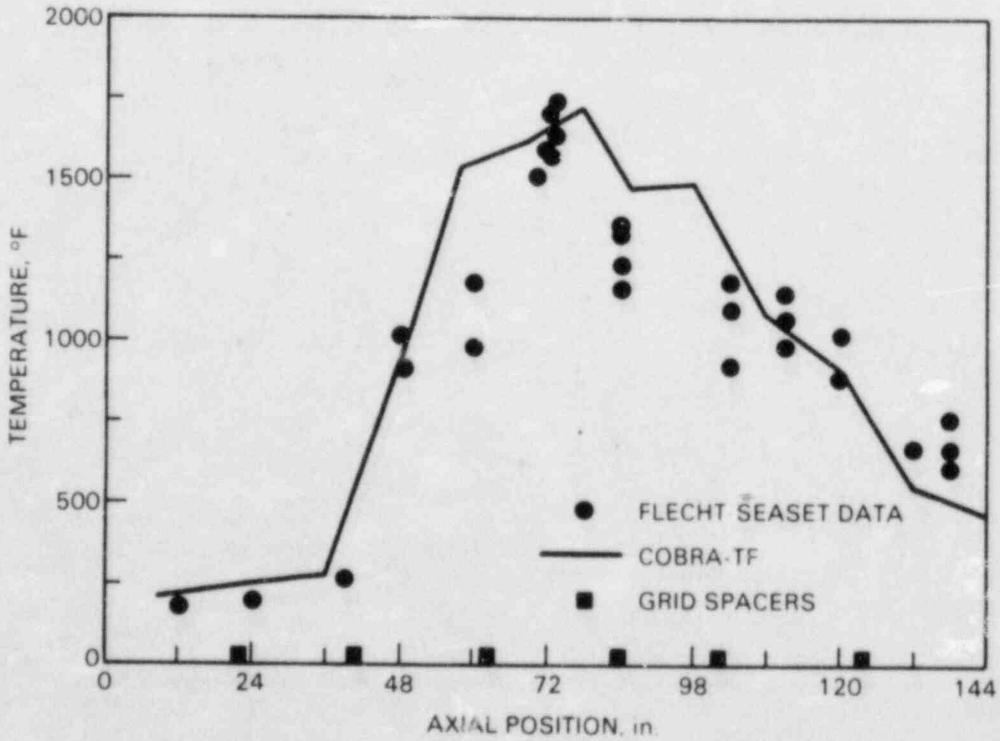

Figure 7. Axial Profile of Rod Temperatures at 100 s for FLECHT-SEASET 21-Rod Unblocked Bundle Test 43208A

The flow blockage heat transfer models were successfully incorporated into COBRA-TF this quarter. Initial simulations for four blockage tests with concentric coplanar sleeves on all rods were conducted:

- FLECHT-SEASET 21-rod Bundle C (62% blockage)
  - 42506C (40 psi; 0.9 in./s)
  - 42008C (40 psi; 1.5 in./s)
- FEBA Series VII (62% blockage) - Run 324 (60 psi; 1.5 in./s)
- FEBA Series VIII (90% blockage) - Run 337 (60 psi; 1.5 in./s).

Both of the 21-rod Bundle C simulations predicted the blockage heat transfer enhancement reasonably well (see Figures 8 and 9). The results of the FEBA tests were not as promising. In Run 324 (62% blockage), the heat transfer enhancement was slightly underpredicted; and in Run 337 (90% blockage), the observed enhancement was vastly larger than the predicted effect. The large blockage effect seen in Run 337 is probably due to atomization of the drops as the flow accelerated through the 90% flow blockage. This hypothesis will be tested by simulating the FEBA 90% blockage with flow bypass tests.

Rod Deformation Modeling

Seven rod deformation models have been identified as possible candidates for the COBRA-TF ballooning model:

- CANSWELL-2 (Winfrith)
- Rowe and Associates (EPRI)

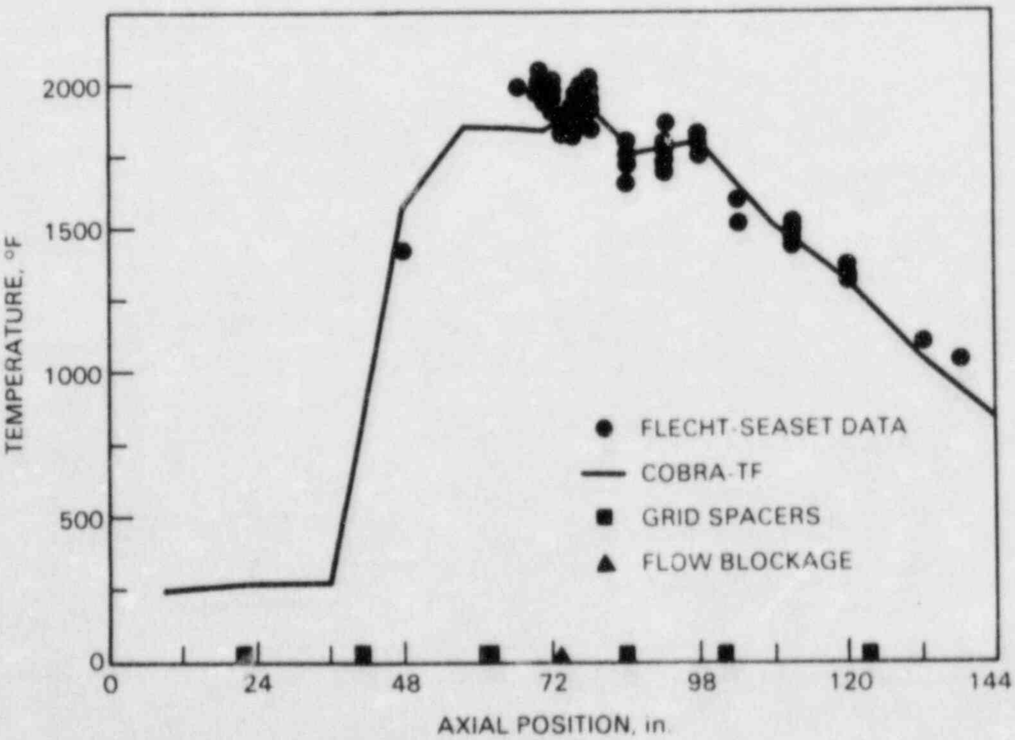

Figure 8. Axial Profile of Rod Temperatures at 110 s for FLECHT-SEASET 21-Rod Bundle Test 42506C

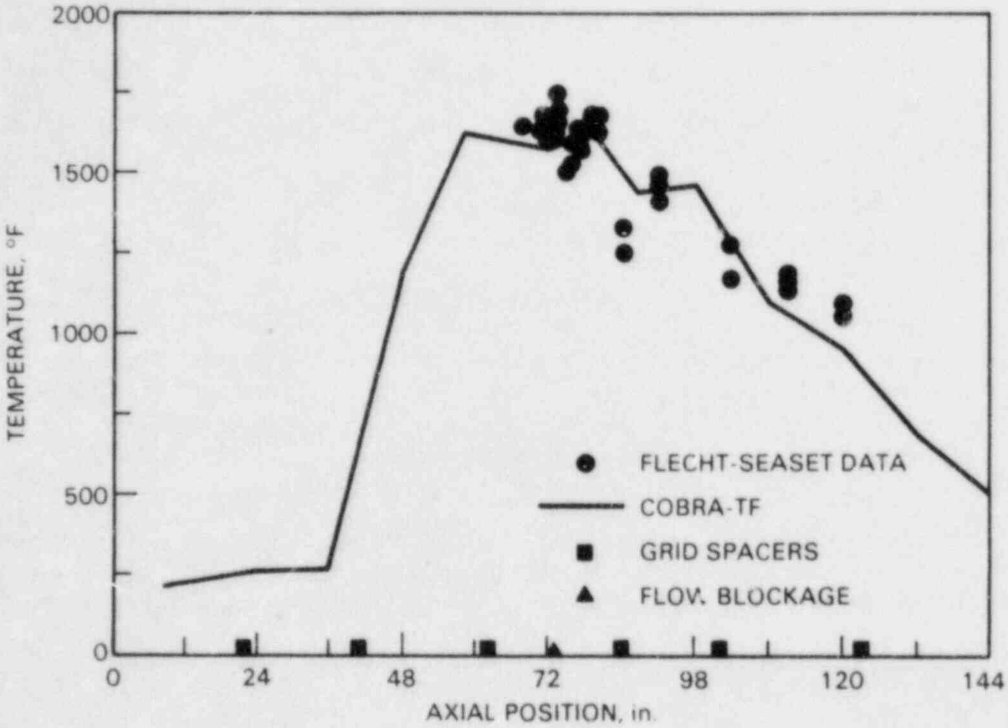

Figure 9. Axial Profile of Rod Temperatures at 80 s for FLECHT-SEASET 21-Rod Bundle Test 42008C

- FRAP-T6 (INEL)
- SYSST-2 (KFK)
- FREY (Anatech International Corporation)
- NUREG-0630 (NRC)
- CARATE (KWU).

A rod deformation model will be selected based on 1) ease or difficulty of incorporating the model into COBRA-TF, 2) relative cost in computer time, and 3) accuracy of predictions and size of data base.

Implicit Fluid Solution

The objective of this task is to develop a fully implicit fluid solution algorithm for COBRA-TF. If successful, this improvement would allow the material Courant time step limitation to be exceeded. Two important benefits would result:

- significantly reduced run times for reflood transients (factor of 3 to 5)
- capability to employ a fine mesh hydraulic grid in the vicinity of a flow blockage (2 in. versus 12 in. node).

Although not part of our original scope for fiscal 1984, this task has been given high priority as a consequence of the large potential benefits. The three man-month effort that is expected will impact the available funding for other tasks.

Significant progress has been made on this task:

- The fluid solution subroutines have been rewritten (approximately 3000 lines of coding).
- A parallel-channel steady-state boiling test—similar to boiling water reactor (BWR) operating conditions—was run successfully.
- A two-channel dispersed-flow quasi-steady simulation was run with a factor of 20 reduction in computer time versus the explicit solution.

FUTURE WORK

During the next quarter, the flow blockage heat transfer models will be further assessed. Both FLECHT-SEASET and FEBA blocked bundles with bypass and noncoplanar blockage geometries will be simulated. The rod deformation modeling work will continue at a low level of effort until the FLECHT-SEASET project is completed. Work will continue on the implicit fluid solution through January; after that time, this effort will be reduced (or postponed) due to budget constraints.

Detailed geometric data were obtained from Westinghouse for the Surry Plant. The input deck for natural convection studies during severe core damage accidents will be set up during the next quarter. The BWR-FIST input deck for Test 6DBA1B will be completed, and initial runs will be made.

COBRA-NC simulations of the DEMONA Test X2 and the 3-D flow in the HDR blowdown room will be completed next quarter. The fully implicit solution scheme will be implemented into COBRA-NC.

REFERENCES

1. Guidotti, T. E., and M. J. Thurgood. 1984. *A COBRA/TRAC, Best-Estimate Analysis of a Large-Break Accident in a PWR Equipped with Upper Head Injection.* NUREG/CR-3642, PNL-4971, Pacific Northwest Laboratory, Richland, Washington.

2. Bian, S. H., M. J. Thurgood, and J. M. Kelly. 1984. *COBRA/TRAC Simulation of Semiscale S-UT-2 Test.* NUREG/CR-3563, PNL-4908, Pacific Northwest Laboratory, Richland, Washington.

3. Eyler, L. L., and D. S. Trent. 1984. *Pressurized Thermal Shock: TEMPEST Computer Code Simulation of Thermal Mixing in the Downcomer of a Pressurized Water Reactor.* NUREG/CR-3564, PNL-4909, Pacific Northwest Laboratory, Richland, Washington

4. Edler, S. K., ed. 1983. *Reactor Safety Research Programs Quarterly Report, July-September 1983.* NUREG/CR-3307, Vol. 3, PNL-4705-3, Pacific Northwest Laboratory, Richland, Washington.

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

The fiscal 1983 progress report was completed and is ready for submission to the U.S. Nuclear Regulatory Commission (NRC). The program's data retrieval and analysis system (DRAS) is ready to collect welding/repairing data. Work towards the welding of a 24-in. diameter Type 304 stainless steel (SS) pipe was initiated. Work began on a literature review entitled "Compositional Effects on the Sensitization of Austenitic Stainless Steels."

INTRODUCTION

The objective of this Pacific Northwest Laboratory (PNL) program is to determine a method for evaluating the acceptance of welded and/or weld-repaired SS piping for light-water reactor (LWR) service. Validated models, based on experimental data, will be developed to predict the degree of sensitization (DOS) and the intergranular stress corrosion cracking (IGSCC) susceptibility in the heat-affected zones (HAZs) of SS weldments. The cumulative effects of material composition, past fabrication procedures, past service exposure, weldment thermomechanical (TM) history, and projected component postrepair life will be considered.

TECHNICAL PROGRESS

During the October-December 1983 quarter, the major effort was expended in the preparation of a literature review report entitled "Compositional Effects on the Sensitization of Austenitic Stainless Steels." Preparation of a 24-in. diameter Type 304 SS Schedule 80 pipe for TM history determination during welding/repairing was initiated. Data analysis continued in two areas: 1) TM data generated during the monitoring of a 1.25-in. thick flat plate Type 304 SS feasibility weld and 2) data generated using the electrochemical potentiokinetic reactivation (EPR) DOS measurement technique.

WELD TM HISTORY DETERMINATION AND PREDICTION

The TM weld/repair history determination analysis system was brought on-line. The DRAS can collect 64 channels of data at a rate of 640 data points per second. Strain and temperature data collected during welding are placed on magnetic tape for post-test data analysis and long-term storage. The system is capable of displaying up to 16 channels during welding and plotting test data as a function of test time once the test weld has been completed.

Work was initiated on preparing a 24-in. diameter Type 304 SS Schedule 80 pipe for welding/repairing. The pipe was previously sent out for weld preparation, including a 2T counterbore region on the

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

inside surface of the pipe. A section of the counterbored surface was prepared for subsequent DOS measurements in the HAZ after selected weld/repair passes. Installation of the instrument domain strain and temperature sensors in the counterbore region was initiated.

INFLUENCE OF COMPOSITION AND TM HISTORY ON DOS AND SCC

Standard-cell EPR measurements were completed on five heats of Type 304 SS and were initiated on two heats of Type 316 SS specimens heat-treated to different DOS levels as the initial effort towards establishing the programmatic DOS history data base. The test results are being plotted as time-temperature-sensitization curves and will be included in the literature review report.

Licensing personnel who attended the Program Review Meeting in September 1983 requested that a review of current knowledge of the change in resistance to SCC as a function of alloy content and type be compiled in fiscal 1984. Work was initiated on a literature review entitled "Compositional Effects on the Sensitization of Austenitic Stainless Steels." The open literature was combed for relevant articles and individuals were contacted to improve the comparatively limited data base on NG316 SS. The main basis for the comparison between materials will be isothermal sensitization. More than 100 time-temperature-sensitization curves on different heats of Type 304 or 316 SS have been collected. The data were analyzed to yield time-to-sensitize information at 600°C, 650°C, and 700°C as a function of compositional parameters. Statistical analysis calculations are under way that will yield factors of improvement as a function of alloy content.

SCC PREDICTION FROM TM HISTORIES

The initial steps needed to predict changes in DOS/IGSCC as a function of TM history will be presented in the literature review. The prediction trends of existing models are being compared with measured DOS trends from experimental data.

FUTURE WORK

The literature review will be completed and sent to NRC for review. Welding will be initiated on the 24-in. diameter Type 304 SS pipe. The fiscal 1983 progress report will be sent to the NRC for publication. Eleven additional heats of 304, 304L, NG304, 316, 316L, and NG316 will be purchased to fill out the EPR and IGSCC composition matrix required for this program. All of the material will be purchased as 4-in. diameter pipe.

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 progressing on the design, procurement, and safety analyses for the MT-6A and MT-6B experiments. These tests were recently added to the program and will be performed in the National Research Universal (NRU) reactor. Major efforts this quarter were in the areas of test train design, preparation for hardware procurement, and test safety analyses.

INTRODUCTION

The Pacific Northwest Laboratory (PNL) Coolant Boilaway and Damage Progression (CBDP) Program is an extension of light-water reactor (LWR) large-break loss-of-coolant accident (LOCA) simulations using fission heating to evaluate the advanced stages of LOCA scenarios. The coolant boilaway series of experiments will evaluate the fuel behavior and characteristics during a prototypic simulated small-break LOCA that results in a partially coolant-uncovered reactor core. Decay heat generation boils the coolant and causes damage progression including fission product release in the uncovered fuel rods. Previous experiments evaluated the thermal-hydraulic characteristics and fuel rod rupture characteristics of heatup and quenching phases of a simulated large-break LOCA using the NRU reactor.

The program will develop a well-characterized data set for evaluating the consequences of coolant boilaway and core damage progression in an LWR. Coolant boilaway will be simulated using low-level fission heat as a surrogate for the system enthalpy and decay heat expected to drive a postulated coolant boilaway accident. These data will provide a basis for accident mitigation strategy development and damage assessment for a postulated coolant boilaway accident.

A three-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 three tests in the program are identified as:

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

Near the end of fiscal 1983, the NRC requested that the MT-6 test be included in the current program. The experiment will use full-length pressurized water reactor (PWR) fuel bundle test assemblies and will be performed in the NRU reactor, Chalk River, Ontario. Highlights of the test conditions are given in Table 1.

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

Table 1. CBDP Program Test Matrix

Test	Peak Temperature, K	Type of Shroud Insulation	Hydrogen Measurement	Fission Product Measurement	Includes Control Material	Includes Preirradiated Rods	Planned Test Date
MT-6A	1090	ZrO_2	No	Minimal	No	No	6/84
MT-6B	1670	ZrO_2	No	Minimal	No	No	6/84
FLHT-1	2150	ZrO_2	Yes	Yes	No	No	11/84
FLHT-2	2500	ZrO_2/ThO_2	Yes	Yes	No	No	4/85

The following data will be obtained from the CBDP tests:

- fission product release
- temperature distribution for each full-length bundle as a function of liquid level
- description of fuel bundle damage progression (core degradation)
- 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
- inner and outer diameter cladding oxidation and embrittlement
- test train design verification for subsequent tests.

These data will provide a basis for accident mitigation strategy development, for evaluation of postulated coolant boilaway accidents, and for developing concepts for accident prevention and quantifying safety margins.

The CBDP integrated core damage effects tests will provide data to assess and confirm the validity of results obtained from numerous separate effects tests being sponsored by the NRC. Experiments are being designed and conducted to use NRU advantages such as 1) capabilities for testing highly instrumented multirod 12-ft long bundles under thermal-hydraulic conditions representative of contemporary LWRs, 2) the ability to achieve requisite power densities and 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 advantages will eliminate 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 from these tests can be used to evaluate LWR accident code models and 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 (SFD) data base and expand the data to be obtained from ACRR and Power Burst Facility testing and from tests at other domestic and foreign laboratories.

TECHNICAL PROGRESS

Work completed during the October-December 1983 quarter concentrated on preparing for the MT-6A and MT-6B tests that are scheduled for June 1984. Activities included:

- test planning and scheduling
- preparation of a preliminary safety analysis report
- design of the MT-6A and MT-6B shrouds and fuel bundles
- preparations to procure the hardware for the shrouds and bundles
- initial preparations at CRNL.

A cross section of the MT-6A insulating shroud and 21-rod fuel bundle is shown in Figure 1. The insulating shroud is a sealed composite tube consisting of low-density ZrO_2 insulation sandwiched between an outer layer of stainless steel and a inner layer (liner) of Zircaloy. Several of the nearly 200 test train instrumentation components (thermocouples, liquid level detectors, neutron detectors, etc.) are illustrated in the schematic.

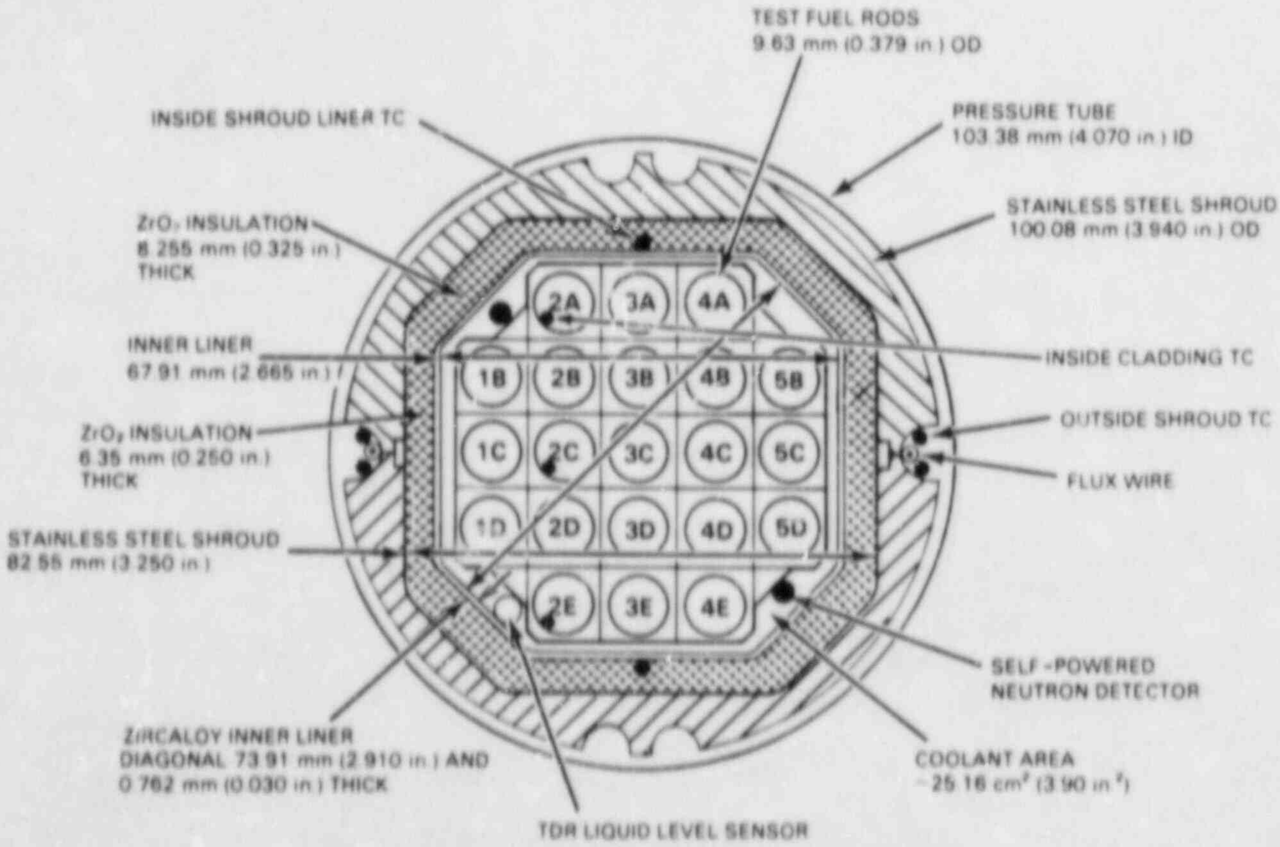

Figure 1. Transverse Cross Section of MT-6A Insulating Shroud and Fuel Bundle

FUTURE WORK

Work will continue in the following areas to meet MT-6A, MT-6B, and FLHT-1 test dates:

- test assembly design
- test train hardware development, fabrication, and assembly
- licensing for shipment to Canada
- thermal-hydraulic calculation for test operations and safety analyses
- neutronics analysis related to safety
- test assembly instrumentation
- effluent control module, including H_2 measurement system design
- data acquisition and control logic development
- test safety analyses
- quality control and assurance
- loop modifications at CRNL.

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P.O. Box 10412
Palo Alto, CA 94304

J. Mundis
Electric Power Research Institute
P.O. Box 10412
Palo Alto, CA 94304

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B. R. Sehgal
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W. L. Pearl
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7015 Realm Drive
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Oak Ridge National Laboratory
Oak Ridge, TN 37830

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P. O. Box 900
Princeton, NJ 08540

J. Vahaviolos
Western Electric, ERC
P. O. Box 900
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U.K.

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16. ABSTRACT (200 words or less)

This document summarizes work performed by Pacific Northwest Laboratory from October 1 through December 31, 1983, for the Division of Accident Evaluation and the Division of Engineering Technology, U.S. Nuclear Regulatory Commission. Evaluations of nondestructive examination (NDE) techniques and instrumentation include investigating the feasibility of detecting and analyzing flaw growth in reactor pressure boundary systems and examining NDE reliability and probabilistic fracture mechanics. 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. Experimental data and validated models are being used to determine a method for evaluating the acceptance of welded or weld-repaired stainless steel piping. Thermal-hydraulic models are being developed to provide better digital codes to compute the behavior of full-scale reactor systems under postulated accident conditions. High-temperature materials property tests are being conducted to provide data on severe core damage fuel behavior. Severe fuel damage accident tests are being conducted at the NRU reactor, Chalk River, Canada; an instrumented fuel assembly irradiation program is being performed at Halden, Norway; and fuel assemblies and analytical support are being provided for experimental programs at the Power Burst Facility, Idaho National Engineering Laboratory, Idaho Falls, Idaho.

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