NUREG/CR-3810
PNL-5106-2
Vol. 2

Reactor Safety Research Programs

Quarterly Report
April-June 1984

Prepared by S. K. Edler, Ed.

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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NUREG/CR-3810
PNL-5106-2
Vol. 2
R3, 4, 5

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Manuscript Completed: September 1984
Date Published: November 1984

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Washington, DC 20555
NRC FINs: B2043, B2452, B2383, B2084, B2456, B2864, B2455, B2391, B2466, B2041, B2277

ABSTRACT

This document summarizes work performed by Pacific Northwest Laboratory from April 1 through June 30, 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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EXPERIMENTAL SUPPORT AND DEVELOPMENT OF SINGLE-ROD FUEL CODES[a]

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

SUMMARY

The principal objectives of this program are to obtain in-reactor and out-of-reactor data on fuel rod thermal and mechanical performance and to integrate these data into the FRAPCON-2 computer code. Irradiation of test assemblies IFA-518 and IFA-432 continued this quarter. A paper on the retained fission gas data for IFA-432 (Rods 1 and 5) was presented at the annual Enlarged Halden Program Group Meeting; and arrangements were concluded for the postirradiation examination (PIE) of the final set of IFA-432 rods. Programming to include the revised FRACAS-II (trapped-stack) model into FRAPCON-2 was completed, and debugging has begun.

Two deformation/failure tests using iodine were conducted this quarter. A system checkout test using preflawed, nonirradiated Zircaloy cladding was conducted. No failure of the cladding occurred, but experience with the iodine injection system and power supply system was gained. In the second test, an irradiated pressurized water reactor (PWR) cladding was exposed sequentially to three levels of iodine (69, 690, and 70,000 Pa at 300°C) and to powers up to 29 kW/m (9 kW/ft) electrical. Although ridging and slight permanent deformation occurred, the cladding did not fail.

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 up 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 April-June 1984 quarter is described by task in the following sections.

TASK A - IRRADIATION EXPERIMENTS

The fuel thermocouple response data from a preplanned scram of the HBWR in March 1984 were received. The data from Rods 3 and 5 of IFA-432 were normalized, plotted, and compared with those from a similar scram in August 1979. On the basis of this comparison, the decalibration over the intervening 15 MWd/kgM burnup appeared to be less than -5%. Halden analyses put the value at 3% to 4%.

A paper on the retained fission gas data for IFA-432 (Rods 1 and 6) was presented at the Enlarged Halden Program Group Meeting in Stromstad, Sweden (June 4-8). Meetings were held at Harwell to finalize plans for the PIE of the final set of IFA-432 rods. The proposed examinations, listed in Table 1, will be completed in FY 1984.

TABLE 1. Planned Postirradiation Examinations for IFA-432

Rod	As-Fabricated Diametral Gap, μm	Peak End-of-Life Burnup, MWd/kgM	Visual Examination	Profilometry	Axial Gamma Scan	Rod Puncture/ Gas Analysis	Retained Gas Analysis	Optical Ceramography	Burnup Determination
2	380	50	x	x	x	x	x	x	—
3	75	50	x	x	x	x	x	x	x
5	230	50	x	x	x	x	x	x	—
9	180	27	x	x	x	x	—	x	x
4	230	0.1	x	x	—	—	—	—	—

TASK B - DATA QUALIFICATION AND ANALYSIS

In-reactor pressure data from Rod 5 of IFA-432 have now been received through April 1984 (about 38 MWd/kgM rod-average burnup). The assembly was moved in February 1984 to raise the power back to 1979 levels prior to the reactor scram. The resulant 30% increase in power was accompanied by an increase in Rod 5 low-power gas pressure from 2.3 MPa (December 1983) to 2.6 to 2.9 MPa (April 1984). The estimated fission gas release fraction corresponding to 2.9 MPa pressure at the current burn-up level is 14% to 17%.

TASK C - FUEL CODE MAINTENANCE AND IMPROVEMENT

Most of the effort this quarter involved linking FRAPCON-2 to the revised version of the FRACAS-II mechanics subcode, which contains a trapped-stack model and a revised fuel relocation model. This version of FRACAS-II was developed within the FRAPT-6 code, and it has proven to be difficult to replicate the linkage to FRAPCON because of the differences in variable names and common block structures between the two host codes. The FORTRAN coding changes are now nearly complete in FRAPCON-2, and debugging has begun.

TASK D - PELLET-CLADDING INTERACTION EXPERIMENTS

Two electrically heated PCI simulation tests were conducted at 300°C in a pressurized water loop, using iodine injection to promote stress corrosion cracking. The system checkout test used nonirradiated cladding with 38-mm long machined defects on the inner cladding surfaces that penetrated 19%, 42%, and 60% of the wall thickness. This test was conducted using two sequential levels of iodine: approximately 690 Pa and 70,000 Pa at 300°C. At each iodine level, the simulated fuel rod was raised to a linear heat generation rate (LHGR) of 21 kW/m (6.5 kW/ft) at a rate of about 1.5 kW/m-min and the power was maintained for 1 h. When no failure occurred at the highest iodine concentration, the power was raised at about the same rate to 35 kW/m (10.7 kW/ft). At that point, the electrical heater failed by short-circuiting through the fuel to the cladding.

The second test was conducted using irradiated PWR cladding in a similar manner to the nonirradiated test, except that three levels of iodine were used: 69, 690, and 70,000 Pa at 300°C. For all three levels, the iodine was introduced and then the power was increased. At the highest iodine level, the internal fuel rod pressure dropped nearly exponentially with time and leveled out about 3 to 4 min after it was introduced. This observation indicates that the iodine reacted with metals in the system and that there was only a minor amount of free molecular iodine present when the power and stress were imposed on the cladding. The irradiated cladding was sequentially exposed to the three iodine levels at a LHGR of 21 kW/m for 1 h at each iodine level. When no cladding failure occurred after the third period, the LHGR was raised to 26 kW/m (8 kW/ft) for an additional hour. Again, when no cladding failure occurred, the power was raised to 29 kW/m (9 kW/ft). After 27 min, the electrical heater failed by short-circuiting. Cladding stresses were high during the test as evidenced by relatively large pellet-pellet ridge heights and plastic deformation of the ridges after the first power cycle.

Due to experimental system reasons (for example, low molecular iodine levels when the cladding was stressed) or due to some other experimental reason (for example, the neutron fluence was not high enough to embrittle the cladding), stress corrosion cracking cladding failures could not be induced by simulated PCI conditions in this test series. The test series has included a total of six irradiated cladding tests: five without iodine and one with iodine.

FUTURE V. ORK

Next quarter the linkage between FRAPCON and FRACAS and between FRAPCON and FRAPT-6 should be completed. V1M5 of FRAPCON-2 should then be completed and issued.

ACCELERATED PELLET-CLADDING INTERACTION MODELING[a]

R. E. Williford, Project Manager

D. D. Lanning

SUMMARY

Improvements were made in the cladding deformation equations of the mechanical models, and benchmarking efforts showed good results. A series of power transient simulations for boiling water reactor (BWR) and pressurized water reactor (PWR) rods showed that failure probabilities were less than 5% up to 20 MWd/kgM, depending on cladding creepdown and pellet design.

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) 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 (CC) 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 (CCSCC) 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 April-June 1984 quarter is described below.

The cladding deformation equations in the mechanical model were improved by implementing an incremental form of the Prandtl-Reuss equations. A linear strain-hardening law was also included in this update, along with a simple model for radiation hardening. A number of programming errors in the transient fission gas release model were also corrected.

A comparison of the fuel centerline temperature, cladding elongation, and ridging predicted by GT2-F to the first ramp data of IFA-508 (Rod 11) showed that the improvements in the mechanical models produced reasonably good results, especially considering that Rod 11 had a small gap and thin-walled, fully annealed cladding. Rod 11 is a severe deformation case that is a good test for the code.

The axial extent of radial cracks in the fuel had to be considered to produce these results. The same power and burnup dependence was assumed as for the purely radial cracking model, and the axial extent of the cracks was assumed to vary between 25% and 75% of the pellet length.

The code is apparently very sensitive to pellet design parameters such as end-dishing. The pellet shoulder width of the Rod 11 fuel was reported as "approximately" 2 mm (35% of the fuel radius). A 10% larger effective shoulder width was required because of the very sensitive effect on fuel axial thermal expansion. Slight variations from this value resulted in severe under- or overpredictions of cladding deformations. This may be a source of the apparent stochastic behavior of many fuel rods when the effects of fuel cladding eccentricity are considered; i.e., the effective shoulder width may vary axially in the rod.

A series of power transient calculations were performed for BWR and PWR rods at high burnups. Demo Ramp II simulations were performed for BWR rod designs. The results showed that the probability of failure is less than 1% for burnups less than 32 MWd/kgM, transient powers less than 13.1 kW/ft, and hold times less than 5 min, assuming that the CCSCC model provides the best estimate of fracture behavior.

Uncontrolled Control Rod Withdrawal from Low Power (UCRW-LP) situations were simulated for PWR designs. For relatively low cladding creepdown (0.3% in 600 days), results indicated that the failure probability is less than 1% for dish-ended pellets at burnups up to 32 MWd/kgM. PWR designs with flat-ended pellets have a failure probability of 1% or greater at this burnup. For higher cladding creepdown (1.1% in 300 days), the dish-ended PWR design has a failure probability between 1% and 5% at 20 MWd/kgM.

FUTURE WORK

Another series of transient scenarios will be simulated to further define failure probabilities as a function of rod design and burnup. Documentation of the fuel failure code will be completed in the final quarter of FY 1984.

PIPE-TO-PIPE IMPACT[a]

M.C.C. Bampton, Project Manager

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

SUMMARY

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

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.

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

SEVERE CORE DAMAGE SUBASSEMBLY PROCUREMENT PROGRAM

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

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A. J. Anthony
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 a future quarterly report.

INTRODUCTION

The Severe Core Damage Subassembly Procurement Program at Pacific Northwest Laboratory addresses the design, development, and fabrication of fully instrumented test train assemblies for the U.S. Nuclear Regulatory Commission sponsored test program at the Power Burst Facility (PBF), Idaho Falls, Idaho. The objective of this program is to provide test train assemblies used to study the behavior of light-water reactor fuel under severe high-temperature, flow-starvation conditions. In Phase 1, design peak cladding temperatures were 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. Other tests designed for peak test assembly temperatures of 3100K (the melting temperature of UO_2) were considered. Many features of the PBF SFD tests should directly benefit the coolant boil-away and damage progession experiments being performed in the NRU reactor because of the similarities in the experimental objectives and test train functional requirements for the two programs.

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

SEVERE CORE DAMAGE MATERIALS PROPERTY TESTS[a]

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

SUMMARY

During this quarter, viscosity measurements on Zr/UO_2 mixtures with up to 30 mol% UO_2 were performed. Equipment modifications are being made for conducting Zircaloy oxidation experiments in steam/hydrogen gas mixtures and for measuring the heat of reaction of $Zr/UO_2/H_2O$.

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 reaction products to model rod oxidation behavior and to properly account for the melting and refreezing of the cladding. Zircaloy/UO_2/H_2O 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 April-June 1984 quarter is described below.

Viscosity measurements to 2100°C were performed on Zr/UO_2 mixtures containing 15, 20, 25, and 30 mol% UO_2. An oscillating cup viscometer was used, and the molten material was contained in ThO_2 crucibles. Decay constants were measured experimentally at several temperatures during heatup and cooldown and then used to calculate viscosities.

The mixture of Zr with 15 mol% UO_2 appeared to melt near 1925°C. At temperatures above 2000°C, this mixture behaved like a true liquid with a viscosity of 12 to 13 centipoise at 2100°C. In comparison, pure zirconium melted near 1850°C and had a viscosity of 3 centipoise above 1925°C.

As the concentration of UO_2 in the Zr/UO_2 mixture increased above 15 mol%, the apparent melting temperature rose. The Zr mixtures with 20, 25, and 30 mol% UO_2 showed substantial hysteresis between heating and cooling conditions, which is probably a result of nonhomogeneous melting and crystallization. This phenomena was not observed in the Zr-15 mol% UO_2 mixture and will be more closely characterized in future experiments.

Modifications are being made to the Zircaloy oxidation apparatus to permit experiments with H_2O/H_2 mixtures. These experiments are extremely important for predicting the kinetics of the cladding oxidation process in a loss-of-coolant accident (LOCA) where large quantities of H_2 are evolved. The H_2 dilutes the steam and may significantly alter the rate of Zircaloy oxidation. To date, a design has been approved and modifications are under way.

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

Calorimetry equipment is also being modified to permit heat of reaction measurements for $Zr/UO_2/H_2O$ up to 1700°C and for Zr/UO_2 mixtures up to 2200°C. These data are important in refining the predictions on fuel melting above 1800°C.

FUTURE WORK

- Viscosity studies on Zr/UO_2 mixtures with 0 to 30 mol% UO_2 at temperatures up to 2200°C will be completed.
- Zircaloy oxidation experiments in H_2O/H_2 gaseous environments will be performed.
- Oxidation rates of Zr-U-O compositions will be measured at temperatures above the melting point of Zircaloy.
- Physical properties such as specific heats, thermal conductivities of liquid and solid phases, and melting/freezing temperatures as a function of composition will be measured.
- Thermodynamic properties such as heats of formation, reaction, solution, and the solubility of hydrogen in Zr-U-O compositions will be investigated.

COBRA APPLICATIONS[a]

D. S. Trent, Project Manager

J. M. Kelly
R. J. Kohrt

SUMMARY

The COBRA/TRAC input deck for the Full Integral Simulation Test (FIST) facility Test GDBA1B is complete and is being verified. An input deck for the Surry severe core damage transient is being created, and necessary code modifications are being done. COBRA-NC post-test calculations of HDR Tests V45 and V21.3 and a pretest calculation of HDR Test T31.2 will be submitted. Work is continuing on a three-dimensional (3-D) model of the HDR containment. A COBRA-NC model of a generic pressurized water reactor (PWR) containment to calculate distribution of core debris after a high-pressure failure of the vessel is being developed. Assessment of this COBRA-TF code for analysis of FLECHT-SEASET program data is complete.

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 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 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.

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

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.

TECHNICAL PROGRESS

The technical progress that was completed during the April-June 1984 quarter is described in the following sections.

COBRA/TRAC APPLICATIONS

Analyses are currently being done for two thermal-hydraulic transients: a design basis accident in the FIST facility and a hypothetical severe core damage transient in the Surry reactor. COBRA/TRAC input to model FIST test GDBA1B is complete. The input deck is currently being verified by simulating steady-state operation of the facility.

Test GDBA1B is a 100% suction line break in one of two recirculation loops. The calculation is being done to model the variety of heater rod conditions that exist in the FIST core. Experimental measurements show that some rods remain quenched, some rods heat up and quench intermittently, and other rods continue to heat up until the entire core is quenched. The COBRA/TRAC model is sufficiently detailed to model all three types of rod behavior.

An input model of the Surry vessel was set up and debugged during this quarter. COBRA-NC will be used to analyze the natural convection coding that occurs during a station blackout transient as the core boils dry. The purpose of the calculation is to examine the temperature distribution in the upper plenum and core as well as the temperatures at possible failure locations in the vessel (for example, the hot leg nozzles). To analyze this transient, the following code modifications are being implemented:

- addition of steam/hydrogen mass source terms
- incorporation of a heat transfer coefficient for natural convection based on THTF boiloff data
- revision of the Cathcart Zircaloy oxidation equation at high temperature
- addition of a hydrogen blanketing model
- addition of gas mixture transport properties.

Figure 1 shows the region of the vessel that is being modeled. It includes the core, downcomer, upper plenum, guide tubes, and upper head. The figure also shows the internals that are included in the heat transfer model. The corresponding mesh (shown in Figure 2) uses 284 cells in 50 channels and 4 sections. Flow paths from the downcomer to the upper head through the coding jets and from the upper plenum to the upper head through the guide tubes are modeled. The model assumes an axisymmetric flow since it uses a 2-D (r-z) mesh.

A report[1] describing the COBRA/TRAC prediction of Semiscale Test S-UT-5 was prepared and published as NUREG/CR-3748. This experiment simulated a 2.5% cold leg break in a system with UHI.

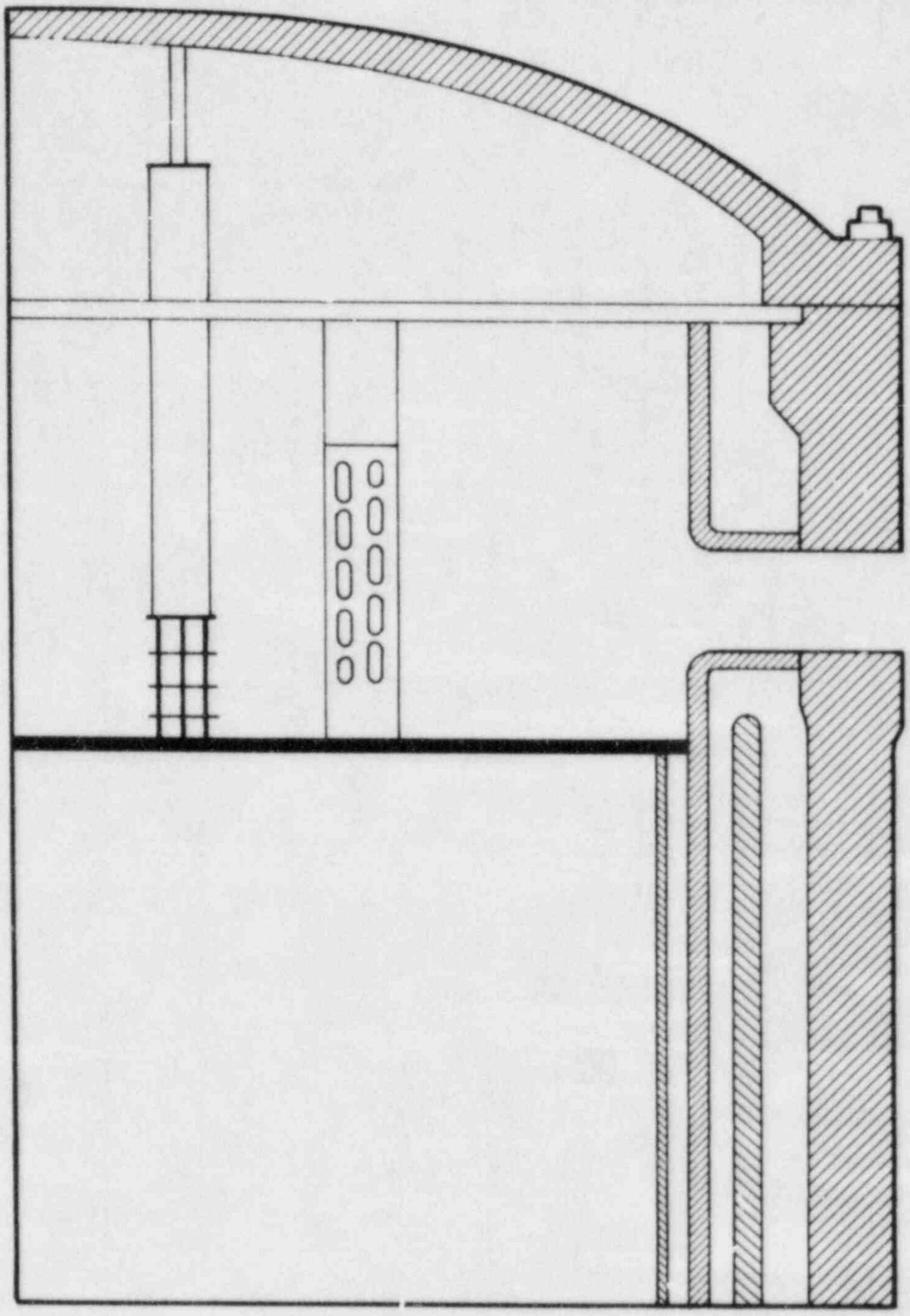

FIGURE 1. Region of the Surry Reactor Vessel Being Modeled

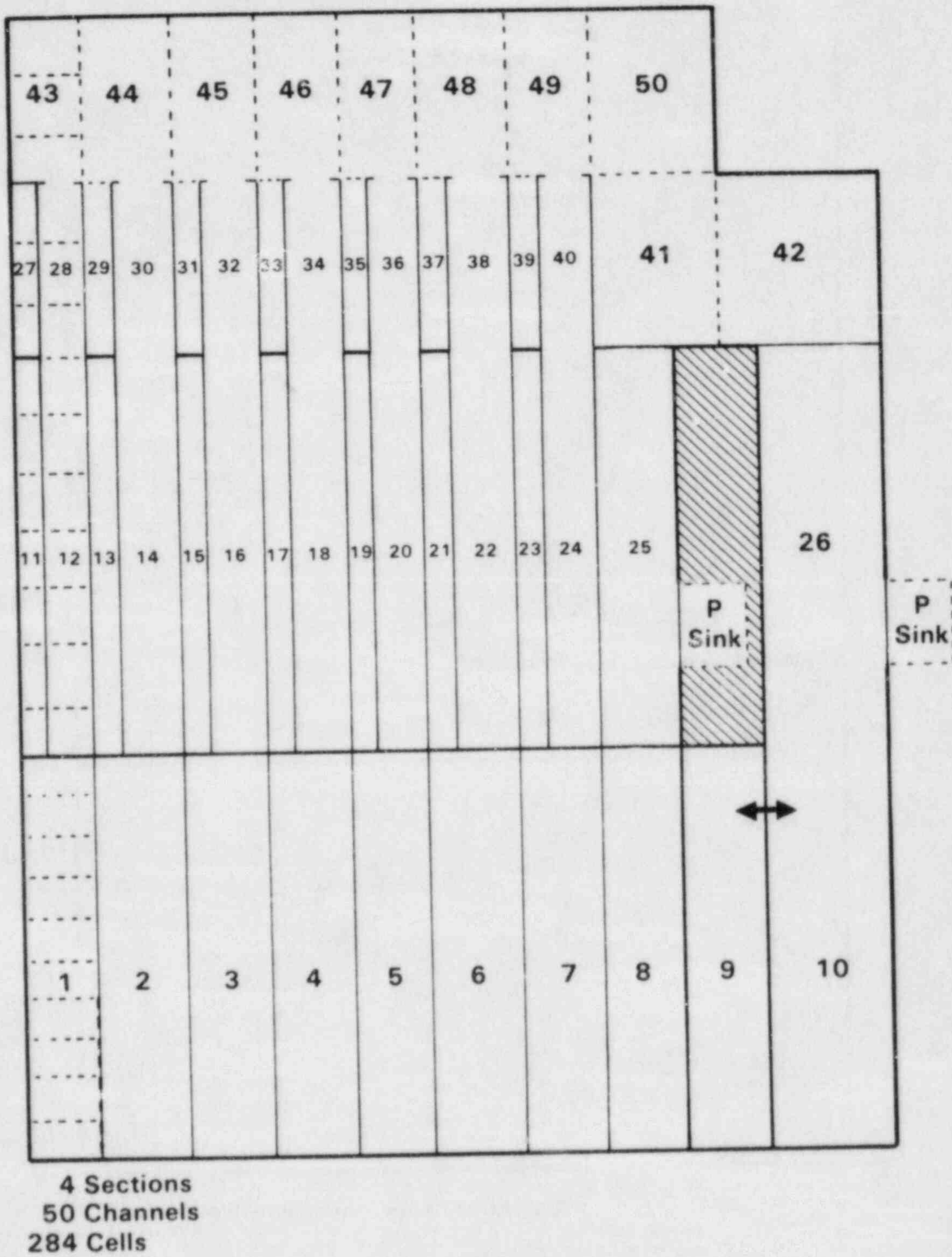

FIGURE 2. Code Nodalization

HOT BUNDLE CODE DEVELOPMENT

Work concentrated on completing the flow blockage heat transfer models for analysis of the FLECHT-SEASET program tests. Development of a rod deformation model and an implicit solution scheme have been postponed due to budget constraints.

Assessment of the hot bundle code is complete. The assessment includes comparison with experimental data from FLECHT SEASET 21-rod blocked and unblocked bundles, FEBA 5x5 grid effects reflood tests, FEBA 5x5 flow blockage tests, and FLECHT-SEASET 161-rod bundle tests. All assessment calculations compared favorably with experimental data signaling that the code is ready to perform blind post-test calculations for the FLECHT-SEASET 163-rod blocked bundle. Participation in the FLECHT-SEASET program will conclude with the blind post-test calculations and a final report on FLECHT-SEASET analysis.

COBRA-NC CONTAINMENT APPLICATIONS

A parametric study was performed for the HDR Test V44 to determine the sensitivity of the calculated pressures to the major input variables. The results of this study are summarized in Table 1.

It is evident from Table 1 that the liquid content of the flow combined with the droplet size gives the largest variation in the calculated room pressures. Physical arguments can be used to show that the variation of this parameter is probably much lower and that the true uncertainty in the room pressure due to the liquid content of the vent flow is much smaller. There is some evidence from this study that supports the argument that this variation may be limited to that of drop size since the drop deposition rate may come into equilibrium with the drop entrainment rate in a very short time. Even the range of drop size is probably limited to between 2 μm and 200 μm rather than 2 to 2000 μm, and the flow is probably a mixture of drop sizes. Both of these factors would reduce the probable effect of the vent flow liquid content. This uncertainty can probably be reduced by a more sophisticated analysis of the flow within the break compartment and more complete measurement of drop flows in all vents leading from the break compartment.

One important determination of this study was that, even if all the input parameters are set to give maximum vent pressure drops, the pressure drops are still underpredicted using Gesellschaft für Reaktorsicherheit (GRS) values for vent discharge coefficients. A larger loss coefficient must be specified for the vent containing the blowdown nozzle before reasonable comparisons with the measured vent pressure drops can be obtained. The minimum uncertainty in the loss coefficient of this vent is more significant than the probable uncertainty of the liquid content of the vent flow. This uncertainty should be removed either by closing this vent in future tests or by measuring the true flow through the vent or true loss coefficient for the vent.

It has been concluded that:

- The Uchida condensation heat transfer coefficient is an adequate model for large-scale containments provided that the distribution of steam throughout the containment is calculated correctly.
- Many nodes must be used to calculate the correct steam distribution.
- The liquid content of the vent flow is an important parameter, but it can probably be shown that the true uncertainty due to this parameter is small.
- The magnitude of the uncertainty in the effective vent loss coefficients needs to be quantified.
- The convection currents within the containment must be correctly calculated to predict the long-term pressure response of the containment.

TABLE 1. Summary of Parametric Study

Parameter	Range of Parameter Variation	Variation of Computed Results
Drop size	2 to 2000 μm	Pressure: 0.21 bar or 9.2% of total ΔP: 0.14 bar or 24% of total
Drop carryover (combined with drop size)	0 to 100%	Pressure: 0.35 bar or 15% of total ΔP: 0.35 bar or 59% of total
Condensation	0 to Uchida	ΔP: 0.04 bar or 6% of total
Vent loss[a] coefficient	2.0 to 8.0 vent U0140 and U0143	Pressure: 0.18 bar or 8% of total ΔP: 0.13 bar or 22% of total
Blowdown flow	Figure 3	Small effect in 0.3 to 2 s. Potential effect is large; uncertainty must be defined.

(a) Variation is the probable minimum; actual uncertainty could be larger depending on the values of other parameters.

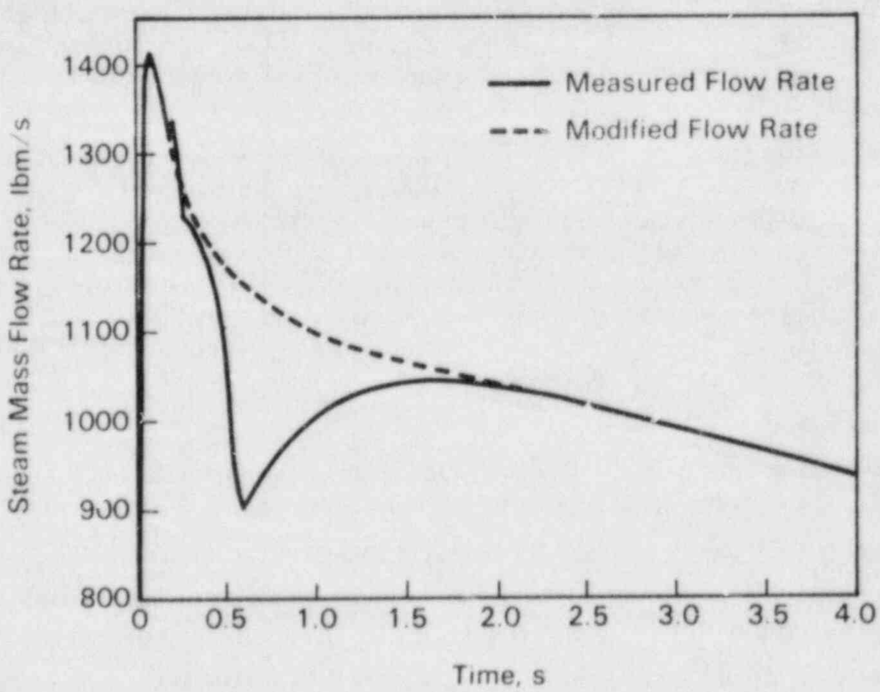

FIGURE 3. Measured and Modified Steam Flow Into Containment

The results of the parametric study of Test V44 using the lumped parameter model have been published as NUREG/CR-3749.[2]

Post-test calculations have been performed for HDR steam blowdown Test V45 and HDR water blowdown Test V21.3. Short time period (0 to 2.0 s) comparisons of measured and calculated pressures in the blowdown room are shown in Figures 4 and 5. The calculated blowdown room pressure is slightly higher than the measured pressure in both tests.

Major input variables from the parametric study were set to the optimum values for Test V44. One exception is the blowdown flow in Test V21.3. Figure 4 shows a large depression in calculated blowdown room pressure at 0.4 s. Similar dips in calculated pressure for Tests V44 and V45 were eliminated by assuming that a large decrease in blowdown flow enthalpy was due to an inaccuracy in measurement. The nature of the blowdown flow for Test V21.3 prohibits a similar smoothing of the blowdown flow enthalpy. The sharp decrease in measured and calculated pressure at 0.7 s in Test V45 is the result of liquid in the blowdown flow. Results of the post-test calculations for Tests V45 and V21.3 support the conclusions reached in the parametric study of Test V44.

A pretest calculation will be submitted for HDR steam blowdown Tests T31.1, T31.2, and T31.3. The difference for the three tests is in the angle of the jet impingement plate, which changes the flow pattern in the blowdown room. The COBRA-NC input is a 2-D model based on the 34-zone model of the HDR containment and does not consider flow patterns within a room. Therefore, all three tests can be predicted by a single COBRA-NC calculation with a 2-D input model.

A 3-D model of the blowdown room was set up and run to investigate the effect of internal flow patterns on the vent pressure drops. Initial results from this calculation indicate that the local pressure in front of the vent containing the blowdown nozzle is 2 to 3 psi lower than the average room pressure depending on how close the impingement plate is to the break nozzle. This pressure drop occurs within the room and is caused by the high-velocity jet issuing from the break nozzle. This internal

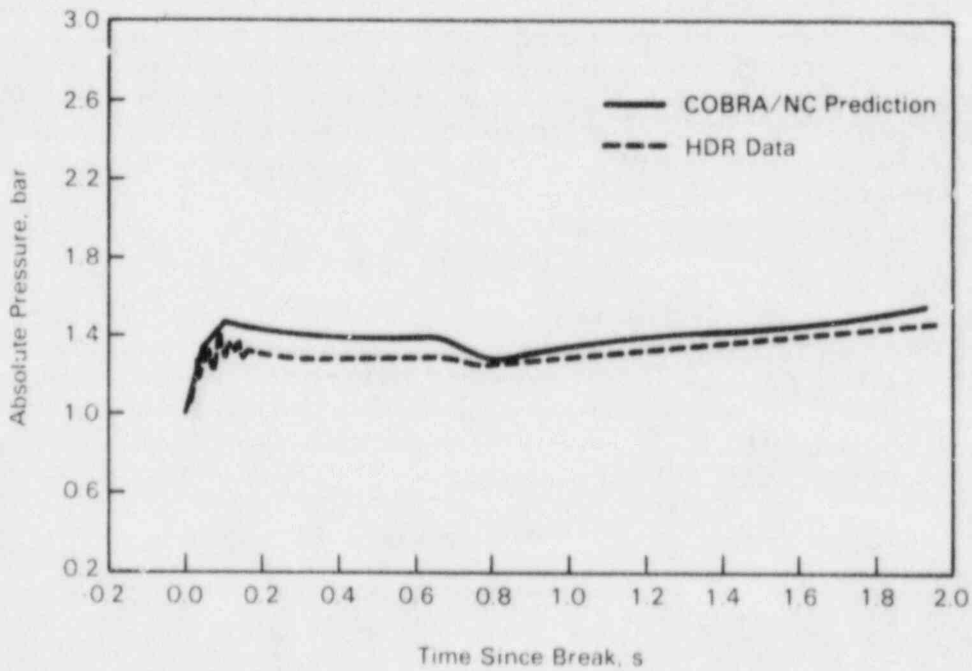

FIGURE 4. Measured and Calculated Absolute Pressures for HDR Steam Blowdown Test V45

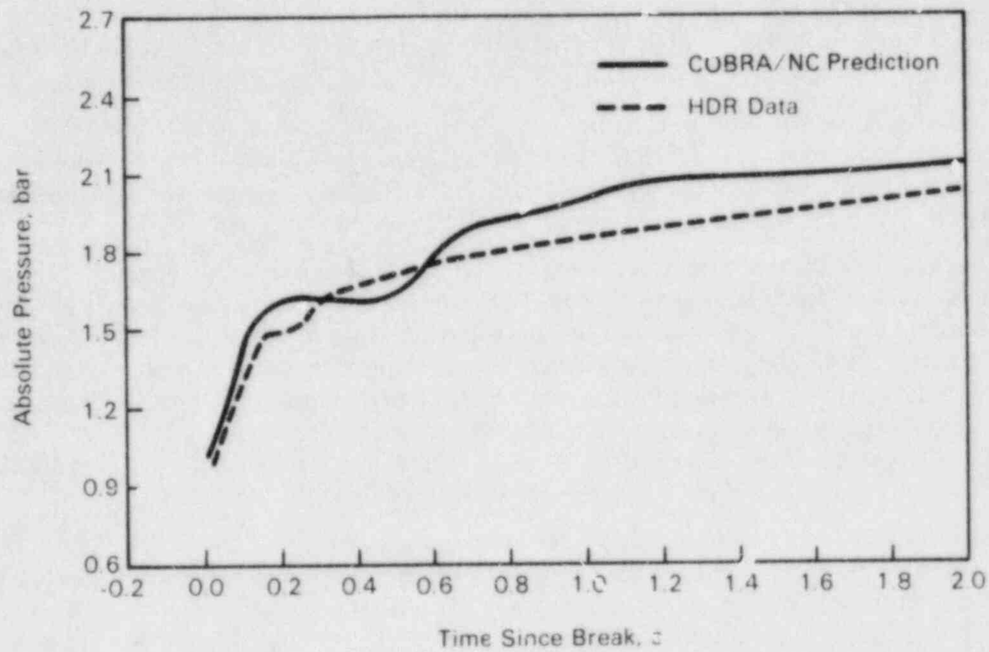

FIGURE 5. Measured and Calculated Absolute Pressures for HDR Water Blowdown Test V21.3

pressure drop accounts for most of the difference between the "apparent" loss coefficient and the geometrical loss coefficient for the vent. The results of the 3-D simulation will be published next quarter.

COBRA-NC is being used to simulate the removal of core debris from a PWR. The objective is to determine how core debris is dispersed when molten core material is ejected under pressure from the vessel into the reactor cavity. Several code modifications are required to perform the calculation. Liquid properties must be replaced with the properties of molten core material. Interfacial mass transfer between the vapor and the liquid must be eliminated, and interfacial drag must be evaluated for particles of molten core in a gaseous medium. The COBRA-NC input is nearly complete, and code modifications are under way.

FUTURE WORK

During the next quarter, analysis of natural circulation cooling during a severe core damage transient and calculation of core debris removal from the reactor cavity of a generic PWR will be completed. Results of the 3-D HDR simulation and a final report on FLECHT-SEASET analysis will be published. Work on simulation of BWR-FIST test GDBA1B will also be completed. Addition of an implicit solution scheme to COBRA-NC will begin next quarter. The implicit scheme is expected to improve the efficiency of the calculation significantly. Simulation of a JAERI CCTF test will also begin next quarter.

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COOLANT BOILAWAY AND DAMAGE PROGRESSION EXPERIMENTS IN THE NRU REACTOR[a]

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

SUMMARY

All preparations for the MT-6A and MT-6B tests were completed. The MT-6A test was conducted during the week of May 21; and the MT-6B test, during the week of June 11, 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 objective of the CBDP series of experiments is to evaluate fuel behavior during a simulated small-break LOCA that results in a partially uncovered reactor core. Fission heating is used to simulate decay heat generation to boil the coolant, which results in fuel damage progression and fission product release. Previous experiments were conducted to evaluate the thermal-hydraulic characteristics and fuel rod rupture characteristics of the heatup and quench phases of a simulated large-break LOCA.

A four-test program will cover fuel bundle damage behavior from 820 to 2230°C in a series of progressively more severe tests operating at prototypic power densities, thermal gradients, and steam mass fluxes. The four tests 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 rod 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 Text Matrix

Test	Peak Temperature, °C	Type of Shroud Insulation	Hydrogen Measurement	Planned Test Date
MT-6A	820	ZrO_2	No	5/84
MT-6B	1400	ZrO_2	No	6/84
FLHT-1	1880	ZrO_2	Yes	11/84
FLHT-2	2230	ZrO_2/ThO_2	Yes	4/84

(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 strategies, 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:

- axial temperature distribution for 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
- flow channel blockage behavior
- hydrogen evolution
- fission product release and transport
- inner and outer diameter cladding oxidation and embrittlement
- test train design verification for subsequent source term oriented 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 at other laboratories. 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 power densities and axial power 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 associated with length and power distribution scaling factors and the interpretation of the experimental results from small-scale separate effects tests.

The CBDP tests are 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.

TECHNICAL PROGRESS

The major accomplishment during the April-June 1984 quarter was the completion of the MT-6A and MT-6B experiments in the NRU reactor. The MT-6A test assembly was inserted in the reactor on May 21; preconditioning was conducted on May 23; and transient testing began on May 24. MT-6A was a LOCA test conducted to determine ballooning, blockage, and heat transfer characteristics of the ballooned and ruptured 21-rod bundle inside an insulated shroud. Fuel rod prepressurization levels and heating rates were selected to maximize Zircaloy cladding strain with rupture occurring at high temperatures in the alpha-phase region.

The MT-6B test assembly was inserted in the reactor on June 11, and transient testing was conducted on June 14. MT-6B simulated a small-break LOCA test with a coolant boildown. Post-test examination is not planned in the near future.

FUTURE WORK

During the next quarter, the following activities will be conducted in preparation for the FLHT-1 and FLHT-2 tests:

- Complete the test assembly design.
- Complete material and hardware procurements.
- Initiate test assembly fabrication.
- Continue software development for the upgraded data acquisition and control system.
- Complete the FLHT-1 final safety analysis report.
- Arrange for shipment of the MV/6000 computer, effluent control module, and test assembly from PNL to Chalk River.
- Complete cost and schedule estimates for subsequent source term oriented tests.

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Publishing Coordination (2)

NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG/CR-3810, Vol. 2
PNL-5106-2

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
Reactor Safety Research Programs
Quarterly Report, April-June 1984

5. DATE REPORT COMPLETED

MONTH	YEAR
September	1984

7. AUTHOR(S)
S. K. Edler, ed.

DATE REPORT ISSUED

MONTH	YEAR
November	1984

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Pacific Northwest Laboratory
P.O. Box 999
Richland, WA 99352

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Accident Evaluation
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Washington, DC 20555

10. PROJECT TASK/WORK UNIT NO.

11. FIN NO. B2043, B2452, B2383, B2084, B2456, B2864, B2455, B2391, B2466, B2041, B2277

13. TYPE OF REPORT
Quarterly

PERIOD COVERED (Inclusive dates)
April-June 1984

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

This document summarizes work performed by Pacific Northwest Laboratory from April 1 through June 30, 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.

17. KEY WORDS AND DOCUMENT ANALYSIS
quarterly progress
reactor safety research
FRAPCON computer code
severe fuel damage
COBRA applications
pellet-cladding interaction

17a. DESCRIPTORS
postirradiation examination
in-reactor testing
pipe-to-pipe impact tests
materials property tests
coolant boilaway and damage progression

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
unlimited

19. SECURITY CLASS (This report)
unclassified

20. SECURITY CLASS (This page)
unclassified

21. NO. OF PAGES
30

22. PRICE
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