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Quarterly Progress Report on Blowdown Heat Transfer Separate-Effects Program for October-December 1980

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Prepared for the U.S. Nuclear Regulatory Commission
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QUARTERLY PROGRESS REPORT ON BLOWDOWN HEAT TRANSFER SEPARATE-EFFECTS PROGRAM FOR OCTOBER—DECEMBER 1980

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ABSTRACT

Analysis of the first small break loss-of-coolant accident (LOCA) heat transfer test series was completed. The second small break LOCA heat transfer test series, consisting of 16 tests, was conducted. Analysis of the second transient film boiling test produced results consistent with those of the first such test — the Dougall-Rohsenow film boiling correlation substantially overpredicts the heat transfer. A series of steady-state film boiling tests and a third transient film boiling test were conducted. Initial evaluation of the in-bundle gamma densitometer system reveals anomalous results when fluid temperature exceeds 589 K (600°F).

Keywords: pressurized-water reactor, rod bundle heat transfer, separate effects, loss-of-coolant accidents, Thermal-Hydraulic Test Facility, electrical heater rods, bundle hydraulics, two-phase flow, dynamic response, fuel rod simulator, THTF bundle 3, ORINC, ORTCAL, bundle uncovery tests, small break LOCAs, high-pressure reflood.

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1. ANALYSIS

1.1 Electric Pin Analysis

L. J. Ott

Three papers on fuel rod simulator (FRS) analysis were presented at the International Symposium on Fuel Rod Simulators — Development and Application held in Gatlinburg, Tennessee.

Final runs of ORTCAL Parts 1, 2, and 3 have been made. ORTCAL is a computer program that determines FRS thermophysical properties from data taken with the rod bundle installed in the Thermal-Hydraulic Test Facility (THTF).

The FRS-058, which was removed from the THTF bundle 3 during the refurbishment of the bundle in July and August, was cross-sectioned and microphotographed, and the internal dimensions were measured. These measurements were analyzed; some obvious stress relief and swelling in the

FRS during its lifetime were apparent. These diametrical changes were incorporated in ORTCAL and in ORINC, a computer program that determines FRS surface temperatures and surface fluxes from internal thermocouple and electrical current data using one-dimensional (radial) equations. The computer program ORMDIN, a report of which was published as ORNL/NUREG/CSD/TM-17, performs the same function as ORINC except that it uses two-dimensional (radial and azimuthal) equations.

Documentation of several electric pin analysis computer programs and analysis techniques has been started and is ~20% complete.

1.2 Small Break LOCA Heat Transfer Analysis

T. M. Anklam M. D. White

Analysis of Small Break Loss-of-Coolant Accident (LOCA) Heat Transfer Test Series I was completed during this period. A final report is being prepared and should be completed early in 1981. A presentation of these results was made at a Blowdown Heat Transfer (BDHT) Review Group meeting. The primary conclusions concerning Test Series I follow.

1. Heat transfer data have been obtained in rod bundle geometry for conditions typical of small break LOCAs.

- Bulk Reynolds number range is $3500 \leq Re_b \leq 10,200$.
- FRS surface temperature range is 850 to 1040 K (1070 to 1412°F).
- FRS to steam temperature ratio range is 1.2 to 1.65.
- Pressure range is 2.5 to 7.0 MPa (363 to 1015 psia).
- Linear powers range from 0.8 to 1.4 kW/m (0.24 to 0.43 kW/ft).

2. Total heat transfer coefficients were between 0.01 and 0.019 W/cm^2-K (17.6 to 33.5 Btu/h-ft^2-°F).

- Thermal radiation accounts for 22 to 37% of the total heat flux, depending on the particular test.
- Radiation to unheated rods accounts for 3.5 to 6% of the total heat flux.

3. A reference temperature correlation based on the fuel pin surface temperature can be used to evaluate the convection heat transfer coefficient under conditions typical of these tests.

4. Various radiation models, when combined with a reference temperature correlation, give a good fit with the data. Best overall fit to data resulted from the use of an Oak Ridge—developed radiation model showing:

- standard error of 6.2%,
- maximum overprediction of 16.1%, and
- maximum underprediction of 9.6%.

Small Break LOCA Heat Transfer Test Series II was performed using the THTF. The test matrix (Table 1) included uncovered bundle tests, reflood tests, and boiloff tests.

Table 1. Test conditions for Small Break LOCA Heat Transfer Test Series II

Test	Type	Pressure [MPa (psia)]	Linear power [kW/m (kW/ft)]	Flooding rate [cm/s (in./s)]	Depressurization rate [kPa/s (psi/s)]
3.09.10UI	Uncovered	4.1 (600)	2.0 (0.6)		
3.09.10UJ	Uncovered	4.1 (600)	1.0 (0.3)		
3.09.10UK	Uncovered	4.1 (600)	0.3 (0.1)		
3.09.10UL	Uncovered	7.6 (1100)	2.0 (0.6)		
3.09.10UM	Uncovered	7.6 (1100)	1.0 (0.3)		
3.09.10UN	Uncovered	7.6 (1100)	0.3 (0.1)		
3.09.10RO	Reflood	4.1 (600)	2.0 (0.6)	15 (6)	
3.09.10RP	Reflood	4.1 (600)	1.0 (0.3)	7.6 (3)	
3.09.10RQ	Reflood	4.1 (600)	1.0 (0.3)	2.5 (1)	
3.09.10RR	Reflood	7.6 (1100)	2.0 (0.6)	7.6 (3)	
3.09.10RS	Reflood	7.6 (1100)	1.0 (0.3)	7.6 (3)	
3.09.10BT	Boiloff	6.2→4.1 (900→600)	1.0 (0.3)		~14 (~2)
3.09.10BU	Boiloff	7.9→6.2 (1150→900)	2.0 (0.6)		~21 (~3)
3.09.10BV	Boiloff	7.9→[illegible].2 (1150→900)	0.7 (0.2)		~21 (~3)
3.09.10BW	Boiloff	7.9→6.2 (1150→900)	0.3 (0.1)		~14 (~2)
3.09.10BX	Boiloff	8.6 (1250)	0.7 (0.2)		~0 (~0)

1.3 Film Boiling Heat Transfer Analysis

C. B. Mullins

An interim report describing results of analysis of THTF test 3.06.6B (film boiling in transient upflow) was completed and distributed to the Nuclear Regulatory Commission (NRC) and BDHT Review Group members. The results of analysis performed on the test indicate that (1) the Dougall-Rohsenow correlation generally overpredicts the heat transfer, (2) the Groeneveld 5.7, Groeneveld 5.9, and Condie-Bengston IV correlations perform better than Dougall-Rohsenow, and (3) the Groeneveld-Delorme correlation is overly conservative in its prediction of film boiling heat transfer. Results of analysis for test 3.06.6B concur with those found in analysis of THTF test 3.03.6AR (film boiling in transient upflow) and the preliminary analysis of steady-state film boiling tests (test series 3.07.9). An abstract covering analysis of tests 3.06.6B and 3.03.6AR has been accepted by the 20th National Heat Transfer Conference for presentation in August 1980. Preparation of the manuscript is 70% complete.

An interim report to NRC covering preliminary results of test series 3.07.9 (steady-state film boiling) is in preparation and over 50% complete.

A 45-min presentation was made at the Review Group Meeting in Gaithersburg, Maryland, on October 30. The presentation covered the preliminary film boiling heat transfer results contained in the two interim reports submitted to NRC this year.

2. THTF OPERATIONS

D. K. Felde G. S. Mailen

The THTF was used to conduct the steady-state film boiling test series (3.07.9), the transient upflow film boiling test (3.08.6C), and the second small break LOCA heat transfer test series (3.09.10I-N). Each of these tests was successfully completed; tests and operations currently scheduled for bundle 3 at the THTF are concluded. The facility is being restored to its original configuration with 4-in. piping, and a facility description report is in preparation.

For the steady-state film boiling test series, system pressure was maintained at 12.4 MN/m^2 (1800 psig) while the bundle power was varied from 30 to 120 kW/rod and the test section flow rate was varied from 2.14 $\times 10^{-3}$ to 6.3×10^{-3} m^3/s (34 to 100 gpm). During these tests, more than half of the bundle was uncovered. At the end of the test series, a transient upflow film boiling test (3.08.6C) was performed. The outlet break area was 25%, with 100% corresponding to a break area of 0.00125 m^2 (0.0135 ft^2).

For the small break LOCA heat transfer test series, the test section inlet and outlet piping size was reduced for handling the low test section flow rates used in these tests [0 to 3.15×10^{-4} m^3/s (0 to 5 gpm)]. An orifice manifold was fabricated and installed on the test section outlet for measuring the bundle outlet steam flow rates. Data were taken at three different system pressures, ~4.1, 7.6, and 10.3 MN/m^2 (~600, 1100, and 1500 psig). Bundle power was varied from 0 to 10 kW/rod while varying the test section flow rate. The upper part of the bundle (5 ft) was uncovered. Section 1.2 contains a summary of the tests that were conducted.

3. TWO-PHASE INSTRUMENT DEVELOPMENT

D. H. Cook D. K. Felde

Investigation of the performance of the in-bundle gamma densitometer system is continuing. Results from the upflow film boiling test 3.06.6B indicate temperature-effect problems at fluid temperatures above ~589 K (600°F). This agrees with results from oven heating tests conducted in air prior to installation in the THTF bundle. Figure 1 shows the density measurement from one of the in-bundle densitometers positioned at thermocouple level F in the bundle [63.5 cm (25 in.) below the top of the heated length]. Calibration of the in-bundle densitometer is based on preblowdown subcooled data and postblowdown empty pipe data. For comparison, the density measurement at the test section outlet is shown in Fig. 2. The in-bundle densitometer shows the same trend as the spool piece densitometer until ~3 s into the transient. From ~3 until 11 s into the transient, the in-bundle densitometer shows an apparently nonphysical behavior. The reason for this behavior appears to be a temperature effect experienced by either the ion chamber or triaxial signal cable. Figure 3 shows a plot of the fluid temperature measured at level F and the temperature inside the

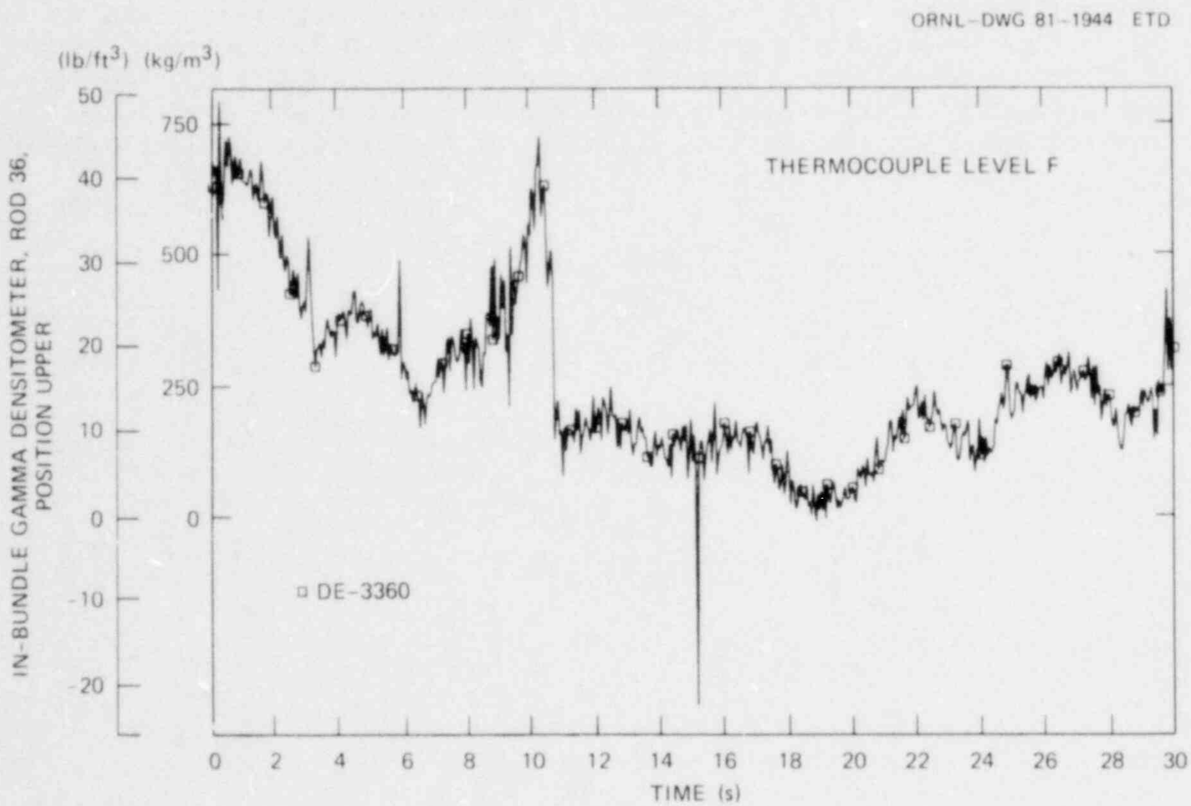

Fig. 1. In-bundle gamma densitometer measurement at thermocouple level F for test 3.06.6B (film boiling in upflow).

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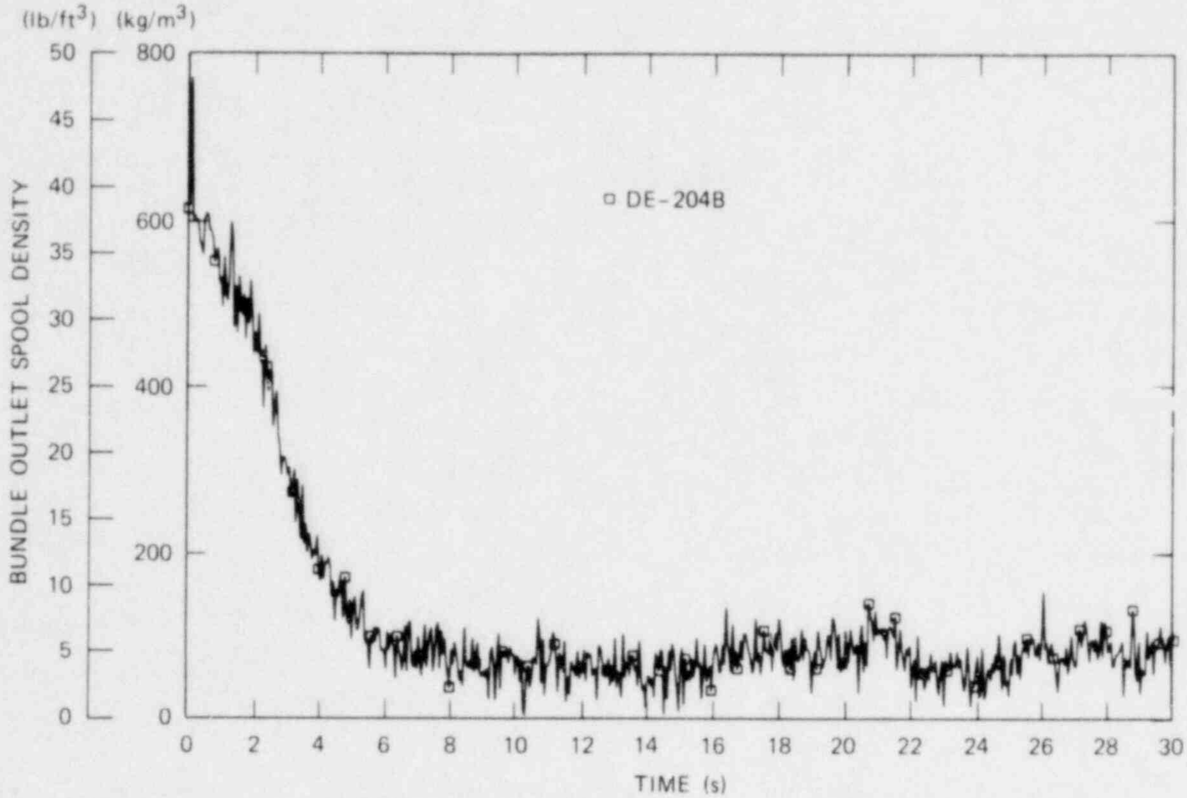

Fig. 2. Bundle outlet spool piece gamma densitometer for test 3.06.6B (film boiling in upflow).

hollow instrument rod in which the ion chamber is located. This temperature is measured by a thermocouple tack-welded onto the ion chamber support tube ~7.6 cm (3 in.) above the ion chamber. The apparently nonphysical behavior shown by the ion chamber occurs at the same times as increases in the fluid temperature. The first gradual fluid temperature peak at ~4.5 s and the second sharper peak at ~10 s are apparently mirrored by the in-bundle densitometer response. As the fluid temperature decreases after ~11 s, the in-bundle densitometer shows more reasonable trends. The temperature trace shown by the thermocouple tack-welded near the ion chamber shows somewhat the same trends as the fluid thermocouple temperature trace, although the former is considerably damped by the instrument rod wall. The in-bundle densitometer actually follows the fluid thermocouple trend more closely than the thermocouple tack-welded near the ion chamber. This may indicate that the ion chamber has greater thermal contact with the hollow instrument rod wall than with the thermocouple tacked near the ion chamber.

The direction of the temperature effect is not as would be expected because of current leakage at higher temperatures. As the resistance of the insulators in the triaxial cable and ion chamber decreases with increasing temperature, the resulting current leakage would be expected to

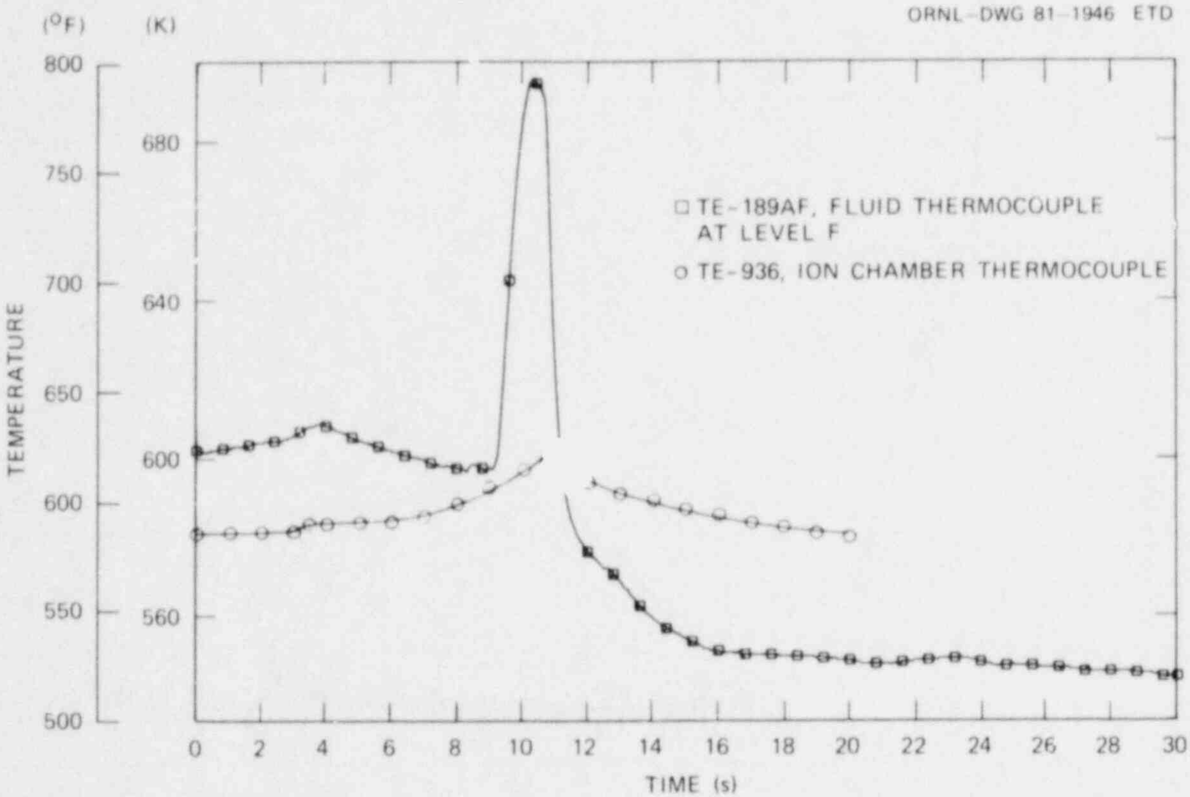

Fig. 3. Comparisons of fluid thermocouple and ion chamber thermocouple at thermocouple level F for test 3.06.6B (film boiling in upflow).

cause an increase in the measured current, thus decreasing the measured density. Because the output current is very low ($\sim 5 \times 10^{-10}$ A), piezoelectric effects possibly are producing the observed behavior. Temperature changes will cause the different materials of the triaxial cables to undergo different amounts of thermal expansion, which may cause piezoelectric effects. Observations during later tests showed that further increases in temperature caused a current swing in the direction expected for current leakage, presumably as a result of decreasing insulator resistance with temperature. Another possible explanation for the observed results may be rod geometry changes caused by high FRS temperatures. Small changes in rod position may produce significant changes in the subchannel geometry illuminated by the in-bundle densitometer system.

Further analysis of the in-bundle densitometer system will be made as results become available from the steady-state film boiling tests, some of which were run with lower fluid temperatures.

The spool piece characterization tests, run in conjunction with the steady-state film boiling test series (3.07.9), have been completed. The results should provide information on the model uncertainties used in obtaining mass flows from the spool piece instrumentation at high pressure and temperatures for high quality two-phase flows. Measurement of subcooled water mass flow at the test section inlet with 2-in. spool pieces will provide a standard for determining uncertainties in the measured two-phase and superheated steam outlet mass flows.

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