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ABSTRACT

During various flow transients in a sodium-cooled reactor, localized boiling can occur. If this boiling does not result in dryout, significant reactor core damage is not likely.

A full-length electrically heated 19-pin bundle was used to determine the extent to which dynamic boiling can be sustained before dryout occurs. Over 30 boiling runs were made with runs at three flow-power conditions culminating in dryout. Continuous boiling for time periods exceeding 20 sec was observed.

Preliminary data analysis suggested that thermal inertia of the duct walls, which were backed with thermal insulation, was higher than designed and was contributing to boiling incoherence. Posttest examination confirmed that the insulation annulus had become permeated with sodium, resulting in a significantly increased thermal inertia.

Detailed comparisons of experimental results with the results of several different analytical techniques indicate that incoherent boiling caused by bundle thermal inertia was responsible for the long time periods between boiling inception and dryout. This suggests that thermal inertia designed into the reactor core could prevent or delay core damage during various flow-power mismatch transients.

INTRODUCTION

Various flow reduction accidents can lead to sodium boiling in LMFBRs; these include the Loss of Flow (LOF), Loss-of-Pipe Integrity (LOPI), and Loss of Shutdown Heat Removal Systems (LSHRS). During a LOPI transient in a loop-type sodium-cooled reactor, due to stored thermal energy in the fuel pins, even though reactor scram immediately occurs, sodium saturation temperatures are exceeded for short time periods in portions of the reactor core. If the ensuing localized boiling does not result in clad dryout, significant reactor core damage should not occur.

For the LOF, noncoherent boiling could delay the time to sodium expulsion and the attendant reactivity increase. Also, for the LSHRS, long-term stable boiling could significantly delay the onset of fuel failure. Thus boiling stability in reactor cores is important in establishing the extent to which core damage will result from flow reduction transients. A test program was completed in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at ORNL to determine the extent of continuous boiling at low flow—low power conditions.[1,2]

THORS FACILITY

The THORS facility is an engineering-scale sodium loop for testing sodium-cooled simulated reactor fuel subassemblies at normal and off-normal operating conditions. Flow is provided by a 40 liters/sec centrifugal pump, and for the test program described heat was removed by a 0.5-MW sodium-to-air heat exchanger. The facility had an argon-covered expansion tank simulating the reactor sodium plenum and a bypass line simulating parallel subassemblies to allow dynamic testing of an electrically heated simulated reactor subassembly. The cross-sectional area of the bypass line was approximately 13 times the net cross-sectional flow area of the bundle. A test-section inlet valve was adjusted to simulate the pressure drop of the reactor inlet flow paths and a bypass line valve was adjusted to control the bypass-to-test section flow ratio.

TEST BUNDLE

The bundle under test (THORS bundle 6A) consisted of 19 electrically heated pins 5.84 mm in diameter spaced by 1.42-mm-diam helical wire-wrap spacers on a 305-mm helical pitch. Each pin had a 0.9-m heated length (with a 1.3 peak-to-mean chopped-cosine heat flux) and a 1.2-m simulated downstream fission-gas plenum which included a 150-mm nickel axial reflector simulating the configuration and thermal characteristics of the Fast Test Reactor.

The 19 pins were contained in a hexagonal stainless steel duct 0.51 mm thick sized so that the gap between the peripheral pins and the inner duct wall was 0.71 mm — half that of the pin-to-pin gap. Use of the half-size edge gaps had been previously shown[3] to give flatter transverse temperature profiles. An insulation annulus surrounded the 0.51-mm-thick hexagonal can to provide low thermal inertia so that the bundle would respond similarly to the central region of a full 217-pin subassembly during thermal transients. As shown in Fig. 1, the insulation annulus was approximately 20 mm thick.

Heated-section thermometry consisted of heater-internal thermocouples attached to the inner surface of the heater sheath, wire-wrap thermocouples embedded in the helical wire wraps, and duct-wall thermocouples located in counterbored holes in the hexagonal bundle duct wall. The simulated fission gas plenum was instrumented by wire-wrap thermocouples and by duct-wall thermocouples.

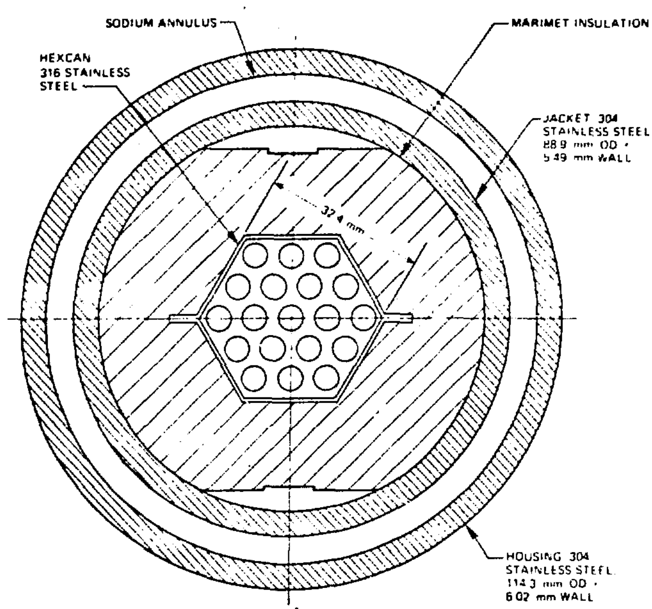

Fig. 1. Cross-section of THORS Bundle 6A showing the extent of the insulation annulus.

Data acquisition was by a PDP-8E computer-controlled data acquisition system. Selected data were logged onto magnetic tape at the rate of 10,000 points per second. High-low limits were set on each data input channel so that when any reading exceeded a preset range for three successive samplings, the test was automatically terminated by disconnecting heater power and pump motor power.

TESTING PROCEDURE

The portion of the THORS loop between the test bundle and the expansion tank could withstand temperatures of 900°C (compared with sodium saturation temperatures of ~950°C) for only short time periods; and although mixing of hot sodium from the test bundle and cold sodium from the bypass line tended to protect the remainder of the facility from excessive temperatures (700°C), it was necessary to limit the time during which boiling could occur during each run.

Each run was conducted as follows. The test section inlet valve was set to simulate the nominal-flow pressure drop of the reactor inlet flow configuration. The bypass valve was set to give a bypass-to-test section flow ratio of 10:1 (this flow ratio remained relatively constant during the run). The test-section inlet sodium temperature was set at 390°C. At a specified constant bundle power, a sodium flow was established to give a test section bulk outlet temperature of 700°C. Using a programmable automatic pump motor control system, the flow was decreased to a low value, held there for a prescribed time period (dwell), and then increased to a value which resulted in a test-section outlet temperature of 540°C. The test-section flow during the dwell time was preset to result in test-section temperatures near saturation or to produce varying intensities of boiling. The duration of the dwell times (between 20 and 90 sec) was a compromise between being long enough to approach steady-state conditions but not so long that excessive loop temperatures between the test section and the expansion tank occurred.

RESULTS

Using the above procedure, approximately 50 runs were conducted using a 10:1 bypass-to-test section flow ratio at bundle powers ranging from 3.0 to 15.4 kW/pin. Boiling was observed in approximately 30 runs, and 5 runs at 3 test conditions culminated in dryout. As shown in Table I, the times between boiling inception and dryout were significant. Dryout was observed for each run only after boiling occurred throughout the bundle cross section.

Table I
Summary of THORS Bundle 6A dryout runs

Test, run (condition)	Bundle power (kW/pin)	Dwell flow (liter/sec)	Time to dryout (sec)
Test 73E, Run 102A (high)	15.4	0.50	~14
Test 72B, Run 101 (intermediate)	9.9	0.24	~17
Test 71H, Run 101 (low)	6.6	0.11	~23

Early in the test program, a comparison of (nonboiling) experimental transient temperatures with analytical predictions indicated that the thermal inertia of the bundle containment was much higher than had been planned, and it was suggested that sodium had leaked into the insulation annulus.[4] Posttest examination verified that the porous block insulation had become permeated with sodium.

ANALYSIS

In an effort to determine the extent to which the long time periods between boiling inception and dryout were caused by the high thermal inertia region surrounding the bundle, several analyses were made using both the thermal inertia of the insulation annulus in the as-planned or dry condition and using its thermal properties calculated to exist if it had been degraded by having been completely permeated with liquid sodium.

For the low flow—low power dryout run (0.11 liter/sec, 6.6 kW/pin), a simple one-dimensional transient conduction model was first considered.[5] A right-circular cylinder having homogenized isotropic thermal properties and cross-sectional area of the bundle (sodium and electric heaters) was surrounded by an annular shell with the thermal properties and cross-sectional area of the insulation annulus. With an initial uniform temperature of 700°C, at time zero, a uniform volume heat generation rate corresponding to 6.6 kW/pin was imposed on the bundle cylinder. The resulting analytical temperature profiles are compared with experimental data in Fig. 2. The analytical profiles using the thermal properties of degraded insulation (Fig. 2*b*) are in reasonable agreement with experimental data (Fig. 2*a*). For comparison, analogous analytical profiles using the thermal properties of dry insulation are shown in Fig. 2*c*. The radial temperature profiles are much flatter and the entire bundle cross-section reaches saturation temperature within 10 sec.

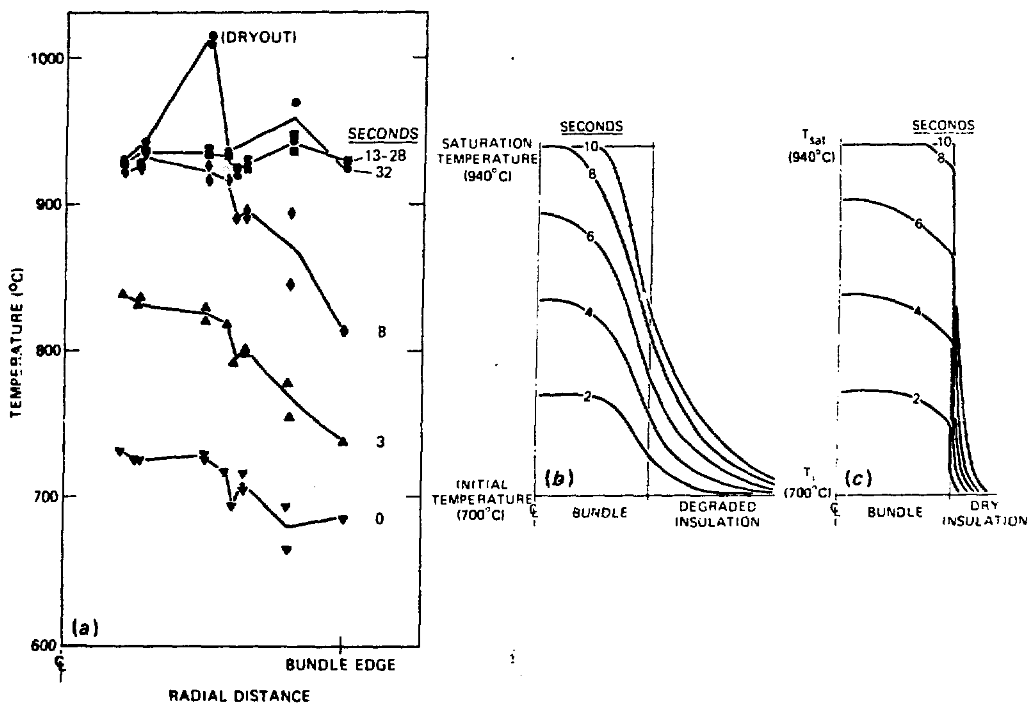

Fig. 2. A comparison of experimental and analytical radial temperatures vs time for the conditions of THORS Bundle 6A, Test 71H, Run 101: dwell flow = 0.11 liter/sec, bundle power = 6.6 kW/pin. (*a*) Experimental results. (*b*) One-dimensional conduction model with thermal properties of degraded insulation. (*c*) One-dimensional conduction model with thermal properties of dry insulation.

A more realistic model was obtained by modifying a version of COBRA III-C,[6] a single-phase transient pin bundle thermal-hydraulic code, to include a radial-conduction model of the bundle containment.[7] A comparison of experimental and analytical results from the low flow—low power dryout run at the 32-in. level (810 mm downstream from the start of the heated section) 10 sec into the transient is shown in Fig. 3. The agreement between experimental results and analytical results for degraded insualtion is remarkable.

For the high flow—high power dryout run (0.50 liter/sec, 15.4 kW/pin), SIMBO,[8] a transient homogeneous boiling code, was used to compute sodium and insulation-annulus temperature development. Although SIMBO is one-dimensional in the bundle, it provides for a radial heat sink boundary in which different thermal properties can be specified. Comparisons of SIMBO results using the thermal properties of degraded insulation and experimental temperatures from thermocouples near the bundle periphery and SIMBO results using the thermal properties of dry insulation and experimental temperatures from thermocouples near the center of the bundle are shown in Fig. 4. With low thermal inertia bundle containment, SIMBO reasonably predicts experimental temperatures observed near the bundle center; with high thermal inertia, it reasonably predicts experimental temperatures observed near the bundle periphery.

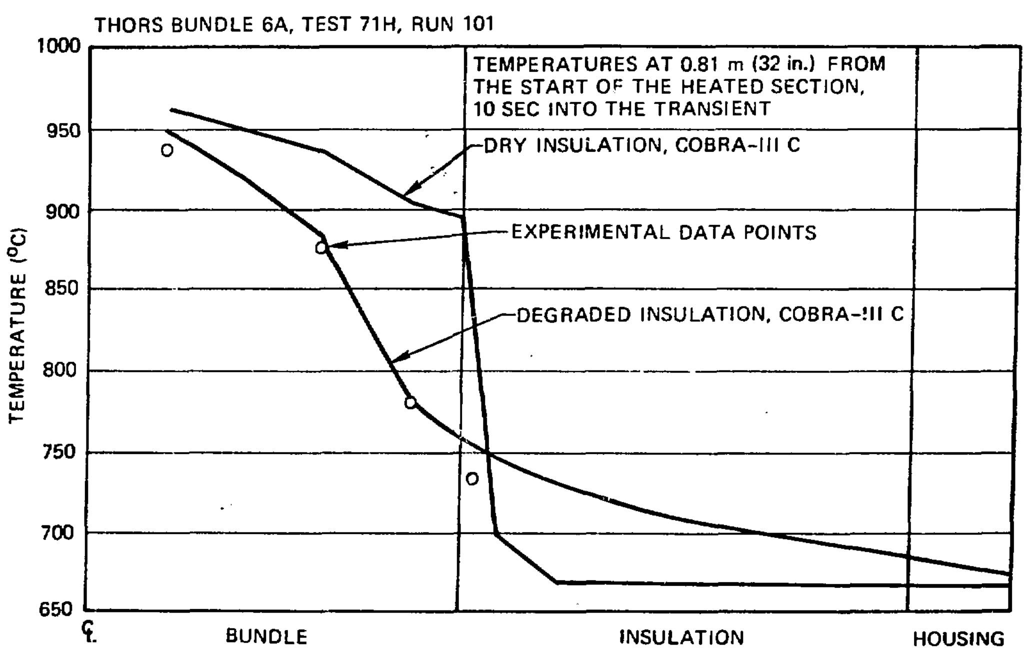

Fig. 3. A comparison of experimental and analytical radial temperature profiles predicted by COBRA-III C at 10 sec into the transient for the conditions of THORS Bundle 6A, Test 71H, Run 101: dwell flow = 0.11 liter/sec, bundle power = 6.6 kW/pin.

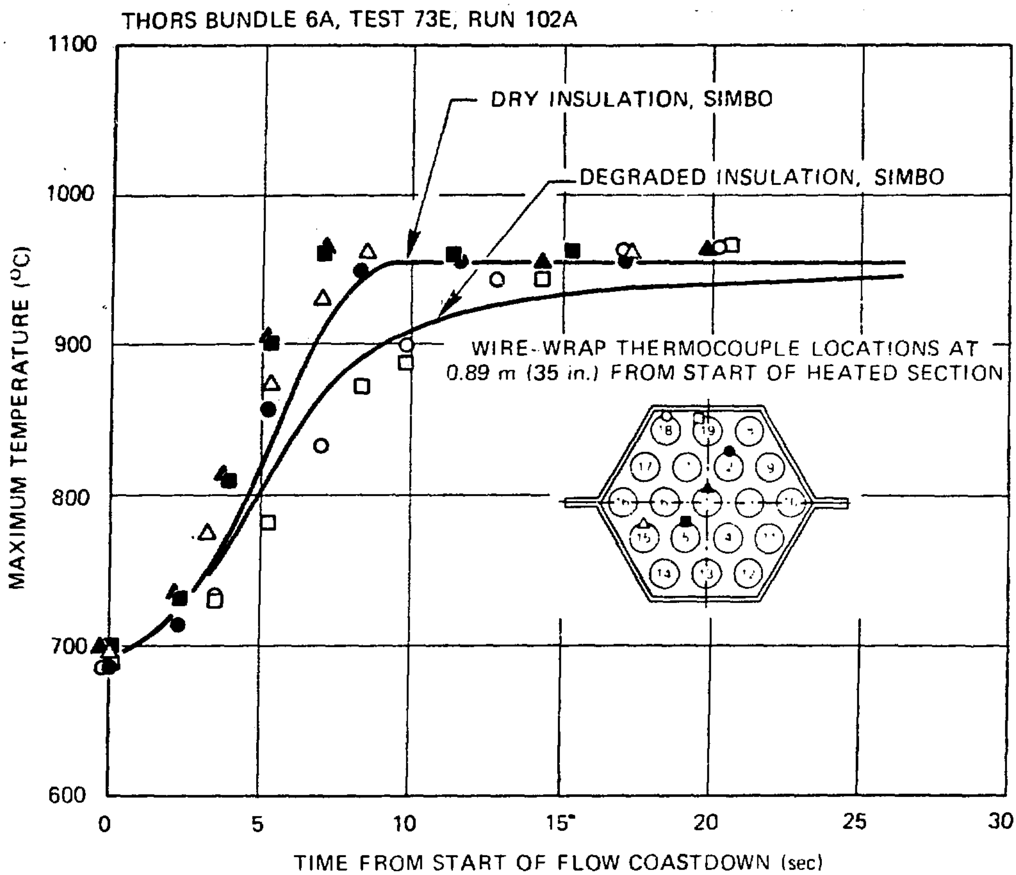

Fig. 4. A comparison of experimental and analytical maximum coolant temperatures predicted by SIMBO for the conditions of THORS Bundle 6A, Test 73E, Run 102A: dwell flow = 0.50 liter/sec, bundle power = 15.4 kW/pin.

CONCLUSIONS

As sustained dryout did not occur until the entire bundle cross-section reached saturation temperatues and thermal inertia of degraded insulation delayed the time necessary for the bundle cross-section to reach saturation temperature, it is concluded that the significant time periods between boiling inception and dryout were primarily due to the large thermal inertia of the bundle containment. It is suggested that by increasing reactor subassembly thermal inertia, radial boiling incoherence can be enhanced and clad dryout delayed. Steady-state thermal-hydraulic characteristics are insensitive to increased thermal inertia, and deleterious neutronic effects can be minimized by proper core design.

The radial growth of the boiling zone is influenced by the coolant flow which, for a fixed driving pressure, is strongly influenced by the axial extent of the boiling zone. Therefore, it is thought that experiments should be performed with additional thermal inertia in the downstream fission-gas plenum region of the bundle. This increased thermal inertia would result in enhanced downstream condensation and also would be expected to delay the onset of clad dryout conditions without adding deleterious neutronic effects.

ACKNOWLEDGEMENT

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