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DEFICIENT COOLING

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MASTER

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ABSTRACT

The 8600-kW ATLAS heat transfer facility and the 16-rod test section and bundle are described. Operation with the first 16-rod bundle produced no useful data due to electrical shorting between heaters and grid spacers. The test bundle was rebuilt with more compatible materials, and testing will be completed in the following quarter.

The one-rod swell heater was redesigned, to avoid brazing difficulties, and testing will be completed in the following quarter.

A list of quarterly reports issued to date is included in the Appendix, with notes of their content, and also a list of topical reports.

SUMMARY

The salient features of the new ATLAS 8600-kW heat transfer test facility are described. The 16-rod steady-state and transient tests in this program will be conducted in ATLAS because of its power supply capabilities and computer-controlled, high-speed data acquisition system. Details of the test section, rod bundle, and instrumentation are described. The 16-rod test schedule is outlined and discussed.

Details of the new design for the one-rod swell heat transfer test are presented.

1. INTRODUCTION

1.1 GENERAL

The work on the Deficient Cooling Program is directed toward extension of out-of-pile heat transfer data in support of current nuclear-powered, water-cooled plants. A brief review of current technology reveals that the large bulk of data is of the steady- or quasi-steady-state class. Thus, a need exists for data taken under transient conditions of power, flow, and pressure in combination with different geometries. Analysis and study of such data are expected to contribute to the following overall objectives:

- a. A firmer understanding and establishment of heat transfer safety limitations, and
- b. Extension of safe operating limits to higher power densities.

Deficient cooling conditions are considered to be those which result in inadequate cooling of the fuel. They may result in potentially excessive cladding temperature which, in combination with the internal gas pressure of the fuel, may cause some type of fuel damage. Resultant temperatures depend on the mode or modes of heat transfer that arise during deficient cooling conditions. A brief description of these modes of heat transfer follows.

Three different modes of heat transfer can generally occur when heat is removed, during high quality conditions, from a surface such as a fuel rod: nucleate boiling, with a liquid film on the heated surface; transition boiling; and the liquid deficient region. Nucleate boiling occurs at very high surface heat transfer coefficients, usually in excess of $10,000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$, with steady surface temperatures. Increase of heat flux eventually results in "film dryout" and a change of mode to transition boiling, which is usually characterized by temperature oscillations of varying amplitude on the heated surface. The transition mode of heat transfer exhibits relatively low heat transfer coefficients, on the order of $500 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$. Continued increase of heat flux eventually results in the liquid deficient regions, with some liquid droplets impinging on the heated surface which is mostly steam cooled.

The heat flux at which transition is made from the nucleate to transition mode of boiling, as evidenced by onset of surface temperature oscillations, is usually termed the critical heat flux (CHF). For low qualities, it is also known as the departure from nucleate boiling (DNB), or burnout point. Considerable effort has been made on parametric definition of the CHF with pressure, coolant flow, vapor quality, and the geometry of the system. These parametric studies serve as design data and indicate the operating conditions beyond which fuel surface temperatures increase substantially, with greater risk to fuel cladding integrity.

Most of the work defining limits of CHF has been done with small incremental heat flux increases which would allow it to be classified as quasi-steady-state data. However, the actual modes of heat transfer become more complex in some cases involving transients, when the dynamic characteristics of the fuel-coolant system come into play. Therefore, practical cases of transient conditions which are considered worthwhile for study and test under this Deficient Cooling Program are those involving:

- a. Transient reduction of flow,
- b. Transient increase of power, and
- c. Transient decrease of pressure which causes flashing and potentially deficient cooling conditions due to increased void content.

Furthermore, it is proposed to study typical geometry changes such as those due to fuel spacer components and those caused by rod bowing and swelling. Such studies are expected to define more clearly heat transfer safety limitations, and will eventually result in extension of safe operation to higher limits.

1.2 PROGRAM OBJECTIVES

The objectives expected to be accomplished under this program are as follows:

- a. Determine experimentally the CHF and temperature regimes which may occur in water reactor fuel assemblies due to nonstandard cooling conditions that result from power, flow, or pressure transients, and determine these values relative to steady-state data.
- b. Evaluate temperature regimes likely to be experienced by water reactor fuel assemblies in loss-of-coolant accidents, and determine areas of similarity or relationship between these regimes and those found in transient critical conditions.
- c. Evaluate consequences of fuel rod geometry changes on heat transfer performance from simulated tests on single and nine-rod electrically-heated bundles.
- d. Provide a plan for further in- and out-of-pile work necessary for more complete performance and definition of the performance of water reactor fuel under deficient cooling conditions.

1.3 ORGANIZATION OF PROGRAM

The work in this program was organized into tasks, an outline of which, together with the tasks' work content, is shown in Table 1-1. These task designations have been preserved throughout the program to provide systematic documentation and execution of the program.

Tasks 3, 4, 5, and 8 involve the experimental work. Tasks 3 and 8 have been completed; the results of Task 8, on nine-rod geometry changes, were included in the previous quarterly report (No. 10). The nine-rod transient flow and depressurization tests of Task 4 are complete.

The 16-rod transient tests in Task 4 will be accomplished in the new ATLAS (8600-kW) test facility during the next quarter. The one-rod swell test, Task 5, will also be completed during the next quarter.

Table 1-1
TASK DESIGNATION AND CONTENT

Number	Title	Content
1	Planning and Analysis	a. Review current technology and plan appropriate tests to integrate with required safety programs b. Analysis as necessary to determine test parameters
2a	Design	a. Design new equipment or modifications of existing equipment to carry out test plans
2b	Manufacture	a. Manufacture or purchase new equipment required
3	One-Rod Exploratory Transient CHF Tests	Exploratory tests involving a. Steady-state and transient flow, power, and pressure b. Modifications of heaters for nonuniform axial heat flux
4	Multirod Transient Tests 9- and 16-Rod	Tests involving: a. Steady-state and transient CHF conditions. Power transients with 16-rod tests only. b. Combinations of intra-bundle clearance, power distribution, and spacer components
5	One-Rod Tests with Geometry Changes	a. Steady-state CHF with simulated swelling
6	Data Reduction and Analysis	a. As required for test Tasks 1, 3, 4, 5, and 8
7	Project Administration, Reporting, and Recommendations	a. Administration of entire program b. Regular and topical reporting c. Future recommendations for continued efforts
8	Multirod Geometry Changes 9-Rod	a. Steady-state CHF data with spacer components and simulated fuel rod bowing and clearances b. Subchannel flow and enthalpy measurements

2. EXPERIMENTAL WORK ON 16-ROD BUNDLE

2.1 ATLAS TEST FACILITY (B. Matzner, D. A. Wilhelmson, D. W. Radcliffe)

2.1.1 Introduction

ATLAS is the name of the new water heat transfer facility, recently completed at General Electric's Nuclear Energy Division at San Jose, California, in which the 16-rod tests of this project will be performed. The facility is capable of providing design information from tests on full-scale, electrically heated rod bundles which simulate reactor fuel, under either steady-state or transient conditions. In terms of available test section power, this facility at 8.6 MW is now the largest of its kind in the western hemisphere. Plans have been made to double the power supply in 1973 which will make it the largest in the world.

The ATLAS facility is unique in many respects. It is the only facility of its kind owned and operated by a U.S. nuclear reactor supplier. It is the only U.S. facility which can simulate the thermal-hydraulic performance (including geometry, power and fluid conditions) of full-scale, water-cooled fuel bundles. It will be possible to determine thermal safety margins directly on simulated BWR fuel under both steady-state and transient conditions. Accurate experimental pressure drop data can be obtained at design conditions with Product Line spacers. Hypothetical reactor accidents typified by flow, power, and pressure transients can also be performed.

ATLAS has high-speed loop instrumentation and data acquisition equipment, including a minicomputer for measuring and recording many variables during transient tests. In the past, application of the foregoing types of test data to the real reactor case required modelling of either geometry or working fluid. Most of the results from ATLAS full-scale tests can be analyzed without resorting to modelling techniques.

2.1.2 Overall Capabilities

The major components of the ATLAS facility are: the high-pressure water heat transfer loop, including instrumentation and control system; SCR d-c power supply; high-speed data acquisition system with a computer; and a heater development area.

The ATLAS all stainless-steel piping system is coded for operation at a pressure of 2250 psig and a temperature of 655°F. The ATLAS test vessel has a 10-3/4-in. i.d. and is almost 18 ft long. It can easily accommodate the dimensions of a standard 7x7 12-ft heated length BWR fuel bundle. Test section flows up to 1000 gpm may be obtained with the existing piping. The present power supply is rated for continuous operation at 7.2 MW (40,000 A at 180 V). Under special conditions, however, it may be operated at 20% overpower to give a maximum available power rating of 8.6 MW. Plans have been made to double the power supply in 1973.

The ATLAS facility has the full range of steady-state testing capabilities for critical heat flux and pressure drop at both BWR and PWR operating conditions. Some of the transient capabilities specifically designed into the ATLAS facility are as follows: to simulate a flow coastdown, test section flow may automatically be varied from any initial value to 50% of that flow on a prescribed transient as short as 0.5 sec; to simulate hypothetical reactor power transients, the power may automatically be (a) varied sinusoidally or on a square-wave shape with frequencies between 1/5 and 5 cps; (b) increased on a ramp at rates up to 100%/sec to a maximum power limited only by the rating of the power supply, and then decreased to the initial value; or (c) decreased from 100 to 6% to simulate a reactor scram, and then decreased on a prescribed transient to simulate decay heat; and to simulate a hypothetical BWR loss-of-coolant accident, provisions will be made to decrease the pressure from 1000 to 750 psi within 2 sec.

The loop is designed to hold all other parameters nearly constant when one of the parameters undergoes a transient. If desired, combinations of the above transients can also be tested.

2.1.3 High-Pressure Water Heat Transfer Loop

Schematically, the loop, as shown in Figure 2-1, has been designed around a so-called "flywheel" circuit which includes the main 350-hp centrifugal pump, mixer, direct-contact condenser, and steam drum. The purpose of the flywheel circuit is to stabilize loop operating conditions (temperature and pressure) when other test section operating parameters are being varied, especially during fast transients.

Figure 2-1. Schematic of ATLAS Heat Transfer Test Facility

The steam drum is the key component for loop pressure stabilization. To provide close communication with the loop, it has been designed as an in-line type pressurizer, as opposed to the side-arm type used by many other loops. Steam is generated for the pressurizing section of the steam drum by a 900-kW, continuously variable, electric heater. Fast response to pressure changes in the loop is obtained by regulating the flow of subcooled water to the spray nozzles in the top of the steam drum.

The mixer is the key component for providing a stable inlet temperature to the test section. It blends flow coming from the flywheel circuit with saturated temperature water from the pressurizer section of the steam drum and highly subcooled water from the air cooler. Fast response is achieved with a fast-acting 2-in. valve in the saturated water line coming from the auxiliary pump and pressurizer.

The main function of the direct-contact condenser is to condense the vapor generated in the test vessel as close to the test vessel as possible. This type of condenser consists of a perforated pipe through which subcooled water is sprayed to condense the steam coming from the test vessel.

The last major component in the flywheel circuit is the 350-hp main circulating pump. It is a centrifugal pump with horizontal shaft and tandem mechanical seals. At BWR operating conditions, it is rated for 1600-gpm flow at a head of 780 ft.

The air cooler is connected in parallel with the flywheel circuit. It takes flow from the pump discharge and sends highly subcooled water, as dictated by the loop control system, to the direct-contact condenser, mixer, and spray nozzles in the steam drum. This air cooler has a rated heat rejection capacity of 9.1 MW.

The test vessel and its associated flow control valves and flow instrumentation are also connected in parallel with part of the flywheel circuit. Three fast-response valves (1-, 2-, and 4-in. sizes) connected in parallel serve to provide automatic fine control or manual control of the flow to the test vessel.

There are several loop auxiliary systems not shown in Figure 2-1. They are concerned mainly with cooling other loop components which are not rated for high-temperature operations such as pump seals, pump bearings, test section O-ring seals, etc. One prominent exception is the high-pressure demineralizer which is required to maintain water quality while the loop is in operation.

2.1.4 D-c Power Supply

The d-c power supply represents the single most expensive piece of equipment in the ATLAS system. The power supply consists of initially two and later four modules, each rated at 20,000 A d-c and 180 V. This gives an initial rating of 7.2 MW. However, the two installed modules can be operated in series at 20% overpower (24,000 A at 360 V) under special conditions, to give a maximum available power rating of 8.6 MW. Plans have been formulated to install two identical modules by the end of 1973, thus bringing the loop and power supply to its final 14.4-MW normal rating (17.2 MW maximum available).

Incoming three-phase a-c power is at 12 kV via underground feeder from the 40 MVA site substation. Local feeder switching is performed with an air circuit breaker, complete with metering and protective equipment. Two separate rectifier transformers in a single tank have their 12-kV primaries fed from this single breaker. The 192 V secondaries form a modified six-phase star which includes an interphase transformer to aid rectifier commutation.

Each rectifier unit (180 Vdc, 20,000 A) has six coaxial SCR "drum" assemblies—one for each phase. Each drum, in turn, has 16 SCR cells connected in parallel and mounted in a circular pattern on a heavy collector plate. High current connections to this plate are made by welded coaxial tubular bus work. This arrangement results in balanced resistance and reactance paths for each of the 16 cells; when the cells are reasonably well matched in characteristics, the load current is properly shared among the cells.

Voltage to the test section is controlled to within 1/4% by operator adjustment of a 0- to 10-V demand signal to the feedback control system of each rectifier unit. At the rectifier, the output voltage is compared with this demand signal. Any difference causes a shift in the phase of the firing pulses at the gates of all the SCRs, thus correcting the error. While the usual mode of control is manual adjustment of the demand signal, built-in features include a preset takedown and recovery program, a curve follower for special power transients, and provision for superposition of oscillator signals on the demand signal. The time constant of the power supply is estimated to be 10 msec.

Special attention was given to arranging the bus work for each changeover from series to parallel operation. The change involves moving bolted links, and can be made in about 2 man-hours. As a matter of interest, the d-c bus work contains more than 10 tons of copper.

D-c power is measured with a wattmeter specially designed to account for any a-c power components generated by the SCRs. The bus current at each rectifier unit is measured with a form of d-c current transformer. It has a transformation ratio of 25,000/5 so that at 20,000 A, its output is 4 A. Any small a-c component superimposed on the d-c is transformed at the same ratio and appears in the output. The transducer output is then multiplied by the test section voltage drop in a carefully compensated Hall-effect multiplier whose output is a voltage proportional to the instantaneous test section power dissipation. This result is displayed on a digital panel meter reading directly in kilowatts. Other outputs are available for input to the data acquisition system as well as for test loop control functions.

During initial power supply operation, the a-c power components present in the d-c power supply output have been found to be very small, except at low power outputs. For this reason, plans are being made to install two standards d-c current measuring shunts which can be used with only small correction factors, to provide an alternate method of power measurement.

2.1.5 Data Acquisition System

The ATLAS Data Acquisition System (DAS), shown schematically on Figure 2-2, samples the analog signals from the loop instrumentation, digitizes the signals, and stores them on mass storage devices, all under computer control. The DAS also monitors critical loop parameters and can automatically initiate data acquisition and loop control. In addition, the DAS can process either off-line or real-time data and print out the loop conditions in engineering units.

2.1.5.1 Hardware

The DAS consists of two separate input systems, which select and sample analog channels under the control of an HP 2116B computer. The low speed input system is used for steady-state operation, while the high speed system is used to monitor critical thermocouples and for transient operation. The low speed system, which is a HP 2911 crossbar switch and a HP 2402 integrating digital voltmeter, can scan 200 input channels at a rate of about 25 channels per second. The HP 2402 voltmeter offers 0.01% accuracy, with a very high noise rejection ratio (100 V of 60-cycle noise will cause less than 0.005 mV signal error), allowing steady-state parameters to be read to the full capabilities of the transducers. The high-speed input system consists of a HP 2930 low-level multiplexer and a Raytheon analog-to-digital converter, capable of sampling 64 input channels at rates up to 15,000 total samples per second. The multiplexer accepts input signals from 10 mV full scale to 10 V full scale, amplifies the signal by one of 11 programmable gains, and feeds it to the converter. The overall accuracy of this system is 0.1%, or 0.03 mV.

The computer serves as a storage buffer for the digitized signals, which are either processed to engineering units and printed out on a teletype, or passed on to one of the storage devices. There are four storage devices available to the ATLAS system: a high-speed paper-tape punch, a strip printer, and two digital magnetic tape units. The paper tape punch is capable of 120 characters per second, or 60 data points per second at 2 characters (16 bits) per data point. The strip printer can handle 20 data points per second for a permanent visual record. The magnetic tape units are the HP 7970B, with 45 in. per second operation. Each unit is capable of storing up to 18,000 data points per second, and more than 10 million data points can be stored on one tape.

The paper tape punch and the magnetic tape units' data can be stored and processed later by either the DAS computer or the GE 635 data processing center. This allows use of the processing center facilities for complex data reduction programs with extensive printed and plotted output. With this system, all lengthy and tedious handling of the raw data is eliminated.

Figure 2-2. Computer-Controlled Data Acquisition System, ATLAS Test Loop

The computer can also exercise some control over the loop operation through a relay output board, which is connected to the power trip, power takedown, and alarm circuits.

2.1.5.2 Software

Software is at least as important as hardware in computer controlled data acquisition systems. For the ATLAS loop, several specialized computer programs have been developed with NED funds to complement the existing data handling and reduction routines.

For steady-state CHF testing, the MASS program scans the loop conditions with the low-speed input system as often as every 3 sec. Concurrently, the high-speed input system monitors the loop power trip circuits, up to 42 critical thermocouples, and an operator event marker button, every 20 milliseconds. When the monitor routine indicates an event has taken place, the computer issues the appropriate loop control instruction, prints a message describing the event, stores the latest loop conditions on magnetic tape, and processes and prints these conditions, while continuing to monitor for other events. Between events, the operator can update constants, update limits, request loop conditions, and scan loop conditions with the high-speed input system, again continuing to monitor for events. The processing program essentially completes the data reduction, but the raw data are available on tape for reprocessing at any time.

A transient program, TRANS, has been written which samples the loop conditions up to 50 times per second in addition to monitoring critical events. The raw data are written on magnetic tape for later processing, and the time and description of any critical events are printed out. Future programming is planned in this area to allow faster sampling rates on selected channels.

The transient data can be sampled immediately after the test by SAMP, which processes and prints out selected points. For detailed reduction, the data tapes are transferred to the data processing center where existing transient reduction programs are used. The 16-rod transient data on this project will be processed at the GE 635 computer data processing center by the program TRDATA, which has been written specifically for that purpose. The program contains the option of linking to a computer plotting program which will plot up to six variables for any desired time interval of a particular run.

2.1.6 General Test Loop Instrumentation

Wherever possible, redundant measurements of test section operating variables are made. Test section inlet temperature is measured to $\pm 1^\circ\text{F}$ by a calibrated RTD (resistance temperature detector) and checked by three chromel-constantan thermocouples. Pressure is measured to ± 5 psi by a 0- to 2400-psi Heise gage and a differential pressure transducer which transmits to the data acquisition system. Test section flow is measured to $\pm 1\%$ in either of the two flow lines by both a Daniel turbine flowmeter connected to visual and analog outputs, and a calibrated orifice connected to a servomanometer. Power is measured to $\pm 1\%$ as previously described. There are also 20 differential pressure transducers available to measure test section pressure drops.

2.1.7 Typical Experimental Operation

Steady-state CHF data are generally obtained by holding subcooling, flow, and pressure constant and slowly increasing power until one or more rod thermocouples indicate the onset of cladding surface temperature instabilities. Typically, a 20 to 30°F temperature rise is sufficient to indicate incipient CHF. This method of CHF evaluation not only gives the most conservative and reproducible results but also exposes expensive test bundles to minimum risk of premature failure.

Transient runs are completely recorded on magnetic tape as described previously. The data acquisition system is activated, and the desired transient is started about 2 or 3 sec later, to allow determination of the initial conditions. Transients involving loss of flow at high power can exceed CHF heater temperatures at very fast temperature-time gradients, which could cause damage to the heaters if not properly controlled. The high-speed DAS system has two alarm circuits for power reduction in such events: the HI alarm, for a moderately high preset temperature level above CHF, which reduces power about 60% when the heater reaches that temperature; and the HI-HI alarm, which trips power off on reaching a preset maximum desired heater temperature.

2.2 SIXTEEN-ROD TEST SECTION

2.2.1 General Description

A schematic of the ATLAS 16-rod test section is shown in Figure 2-3. The 10-3/4-in. i.d. vessel houses a stainless steel flow channel, which is lined with accurately ground ceramic segments. The clearances of the 4x4 rod array are controlled by nine spacer grids which fit snugly inside the ceramic liners. The direct-current-powered heaters are connected in parallel with a liquid metal common junction at the bottom, and a common nickel bus at the top. The ceramic liners provide electrical insulation between the rod bundle and flow channel and vessel, and form the interior flow boundary.

The rods, spacers, and ceramic liner were manufactured to closely toleranced dimensions so the clearances in the resultant flow path are typical of those in a BWR fuel channel.

Inconel direct heaters,* 0.563-in. o.d. with a 12-ft effective heated length, were chosen for these tests as they offered longer expected service life for the difficult transient tests and could be instrumented more readily. These characteristics should result in the most economical test operations. Axial and radial heat flux profiles, details of which are given later in this report, were obtained by varying the wall thicknesses of the heaters.

Chromel-constantan thermocouples in 0.020-in. Inconel sheaths were attached to the inner walls of several heaters at various axial locations for detection of CHF. These emerged at the lower end of the heaters and passed through shielded connectors to the cold junction. Thermocouple signals at the cold junction and other test signals were led, through permanent lines, to the control room panels from which they were distributed to the various recording and computer channels.

The grid spacers were manufactured to reactor spacer dimensions in order to maintain the same rod-rod and rod-wall clearances, with the exceptions of (a) smaller overall dimensions due to the number of rods, and (b) thicker springs to provide additional stiffness to withstand the magnetic forces generated by the currents flowing through the rods. Typical details of spacer construction may be seen in Figure 3-14 of GEAP-10196, the third quarterly report.

Figure 2-4 is a view, looking upstream, of the partially assembled flow channel. The end of the heated length is a few inches downstream of the spacer at the lower right. A close-up view of the flow channel, ceramic liners, rods, and a spacer is shown in Figure 2-5.

2.2.2 Flow Cross Section, Radial and Axial Flux, and Instrumentation Details

Clearance dimensions in the flow channel are shown in Figure 2-6 for the first assembly, No. 16R-1. The dimensions are within a few mils of those in typical reactor fuel bundles. The flow area between spacers is 0.036 ft^2 and has a hydraulic diameter of 0.521 in. Inconel heater sections were available to provide a relative heat flux of 1.103 between the highest flux rod and the bundle average at any axial location. The high flux rod was placed in the No. 1 corner position and had more CHF thermocouples axially as CHF was expected along the No. 1 position due to flux peaking. Also, the No. 1 rod had the lowest rod-wall clearances due to shims on the spacer which held it in the No. 1 corner of the channel. The balance of 15 rods had local/average bundle heat flux ratios as shown in Figure 2-6. The Nos. 4, 13, and 16 corner rods also had several thermocouples, located at axial positions likely to experience CHF, to provide additional protection in the event CHF also resulted there, or if malfunction or loss of key thermocouples occurred at the No. 1 position.

The relative axial locations of the spacers and thermocouples, and the axial heat flux variation, are shown in Figure 2-7. It is known that CHF frequently occurs near the leading edge of flow obstructions. Thus, the four corner rods had thermocouples located 1/4 in. upstream of the spacers' leading edges. In addition, the No. 1 rod had three additional thermocouples upstream of spacers 2 and 3, where the steam quality is appreciable and the heat flux much higher than that at the exit. All the thermocouples, except No. 7, were attached to the interior surface of the heater which faced the corner of the channel (Figure 2-6). Thermocouple No. 7, on the peaked No. 1 rod, was placed 180 degrees from the corner, to monitor conditions in the interior subchannel of the peaked rod.

*GE uses both direct and indirect electrical heaters. In the direct heater, electricity is passed directly through the outer cladding of the rod, while in the indirect heater, electricity is passed through an internal coil which is separated from the outer cladding by an insulator.

Figure 2-3. Schematic of ATLAS Test Section

Figure 2-4. View of 16-Rod Assembly

Figure 2-5. Close-up View of Spacer

Figure 2-6. Flow Cross Section Details, (Assembly 16R-1)

2.3 SIXTEEN-ROD TEST PROGRAM

Realignment of the test program on this project was made in 1970 in order to take advantage of the much larger power supply and high-speed data acquisition system of the ATLAS test facility. These factors allowed the use of a larger test bundle—16 rods instead of 9, which is a closer simulation of reactor bundles—and execution of more difficult transients requiring synchronization and high-speed monitoring and control of test variables.

One- and nine-rod transient flow and depressurization tests have been performed in the 2000-kW facility, and were reported in previous quarterly reports. The 16-rod test program, discussed in the following paragraphs, includes various combinations of flow and power transients with limited manual control of pressure during the transient. Modifications for performance of pressure transients should be completed next year.

Principal transients of interest for determination of reactor fuel temperature safety margins are those involving decay and/or loss of flow. These can occur with moderate reductions in surface heat flux or a power coastdown due to control rod insertion. Another transient of interest involves a potential malfunction of the flow recirculation valve, in which flow and power increase from reduced to rated values. An outline of the steady-state and transient runs that will be performed with the 16-rod test bundle follows with brief comments.

a. First Assembly

Nominal cross-sectional clearances as described previously, with corner rod peaked 1.10, 12-ft heated length and cosine axial flux (Figure 2-7).

Steady State

- (1) Data at two pressures, 800 and 1000 psia and at four or five mass fluxes in the range of 0.25 to 1.0×10^6 lb/h-ft². Various inlet temperatures will be used, up to subcooling values of about 200 Btu/lb. The 1000-psia data are for normal operating conditions of a BWR, and the 800-psia data will be for evaluation of pressure effect on CHF and interpolation where necessary of CHF in analysis of transient runs where pressure dropped appreciably.

Transients

- (2) Flow decay—power constant. This transient will be somewhat more severe than a reactor one-pump trip which results in partial loss of flow with a reduction of 10 to 15% in the surface heat flux. The simple transient is represented by the dotted flow curve D in Figure 2-8. Several runs will be made with variations in initial and final flow; time to low flow, τ_D ; and the power level. Pressure will be approximately constant at 1000 psia.
- (3) Flow and power decay. This will be performed as (2), above, except power will also decay with the flow on a typical reactor power coastdown curve. This transient typifies a reactor two-pump trip.
- (4) Flow ON/OFF/ON—constant power. This transient, reported for a nine-rod bundle in the previous quarterly report, involves turning the flow off for a short period of time, and then re-establishing flow, to determine time to CHF occurrence and the time and characteristics of rod cooling (rewetting).
- (5) Flow ON/DECAY/OFF/ON with power decay. This is an important transient that occurs in the event of a recirculation line break. It is a design basis accident (DBA) that is also referred to as LOCA (loss of coolant accident). A typical transient of this sort, as calculated by the Atomic Energy Commission's Interim Acceptance Criteria Models, is shown by the solid lines in Figure 2-8. After loss of flow, a surge of flow occurs a few seconds later due to lower plenum flashing, and then decays.

The flow transient for this case will be simulated by runs as depicted by the dotted lines, 1 and 2. Parameters that will be varied are time, τ_1 , and magnitude of the lower flow on the initial flow decay; time flow is off, $\Delta\tau_1$, and the level of the initial power.

- (6) Flow and power ramp increases. Flow will be increased from 0.5 to 1.0×10^6 lb/h-ft², and power will be increased from various initial values to final power at various rates of increase over a 3-sec period.

ROD POSITION No.	CHF-TC No							
	1	2	3	4	5	6	7	8
1	•	•	•	•	•	•	•	•
4.13:16	•		•			•		•

Figure 2-7 Axial Location of Spacers and CHF Thermocouples (Assembly 16R-1)

Figure 2-8. Typical Conditions Following a Recirculation Line Break

- (7) Power step decrease. This transient will be run with heater temperatures below the subcooled boiling levels and also at typical conditions of net steam generation. It is anticipated that final data analysis may require corrections for the thermal time lag in the heater-thermocouple output system. Data from these runs may be used to estimate thermocouple output—heater time lags if it appears necessary.

b. Second Assembly

This assembly will be as the first, except that a corner rod will be bowed toward the corner radius of the flow channel to determine the effect on CHF. Data from other projects indicate a high frequency of CHF at the No. 2 spacer. Thus, the bow will be made between the Nos. 2 and 3 spacers shown in Figure 2-7, with a nominal minimum rod-wall clearance at midspan of 0.060 in. compared with normal values of 0.135.

Steady State

- (8) Data will be taken at conditions as noted in (1), above. Sufficient runs will be made to determine CHF variations that may result.

c. Third Assembly

This assembly is similar to the first and second, except that a corner rod will be translated laterally toward the corner, only between the Nos. 2 and 3 spacers, so the rod-wall clearance along the span is a nominal 0.060 in. There is no record of spacers breaking under reactor operation, so there is almost no probability of this type of clearance existing in practice. However, experience from the nine-rod data, presented in the previous quarterly report, indicates that drastic clearance reductions must be made to show reductions in CHF. Very little, if any, reduction in CHF was obtained in the normal operating region of 25% quality.

Steady State

- (9) Sufficient runs will be made at the conditions in (1), above, to determine the effects of this reduced rod-wall clearance.

Transients

- (10) A few transient runs will also be made with this assembly to note effects of a reduced clearance.

2.4 TEST ON FIRST 16-ROD ASSEMBLY

The first test bundle, Assembly No. 16R-1, was assembled and operated; however, no useful data were obtained due to electrical shorting between the heaters and spacer grids. Clearances were generally near nominal values, except in the No. 1 corner where they were lower, at an average value of 0.120 in. Operation of the bundle was characterized by thermocouple temperature excursions and spurious power trips at conditions known to be well below the CHF.

Inconel heaters were selected for the first assembly, rather than stainless steel, as the correct sizes were available to produce a bundle with a corner rod having peaked heat flux. Also, Inconel has a high yield stress at temperature and very little variation of electrical resistivity with the temperature excursions expected. The latter property would result in the least distortion of the axial cosine heat flux at the prolonged temperature excursions planned. A higher operating voltage is required with Inconel, compared with stainless steel, due to its higher resistivity. It was found on disassembly that the combination of higher voltage and insufficient resistance between rods and spacers, resulted in enough arcing to burn holes in the Nos. 2 and 7 rods, and to cause small pinholes in the Nos. 3 and 8 rods. It was decided not to risk potential further loss of expensive assemblies and schedule delays, by rebuilding the bundle with a safer combination of materials. Stainless steel heaters and autoclaved Zircaloy spacers were ordered.

At this writing, materials have been received, the test bundle was rebuilt, and testing was satisfactorily completed, with the loss of only one rod due to leakage. When the replacement heaters were ordered, it was not possible to obtain desired sizes which would result in heat flux peaking. The new assembly uses chromel-alumel thermocouples instead of the chromel-constantan combination used previously.

The 16-rod steady-state and transient data will be reported in the next quarterly report, and the 1-, 9-, and 16-rod transient data will be covered in detail in a topical report in preparation.

3. ONE-ROD SWELL TEST (J. A. Kervinen, E. E. Polomik)

The one-rod swell test has been redesigned because of difficulties encountered with obtaining a reliable brazed joint between a simulated swell, in the form of a copper sleeve, and a stainless calrod. The new design uses a direct heater with the swell machined in the form of two cups which will be welded together and to the constant diameter portions of the heater. Heat flux is essentially constant over the swelled portion of the assembly and the balance of the heater. A schematic of the new design is shown in Figure 3-1. The constant heat flux from the swell was obtained by accurately machining the wall so the thickness varied axially with the diameter to generate the additional local power required. The maximum diameter of the swell results in a mass flux increase of 1.5/1.0. Calibrated thermocouples on the inner side of the heater are used for calculation of heat transfer coefficients and determination of CHF. During tests at 1000 psi, the heater's interior is pressurized to avoid any distortion of the thin walls at the maximum diameter of the swell.

At this writing, tests have been completed at several mass fluxes at 1000 psi. Tests were conducted for both single- and two-phase conditions, and results will be reported in the next quarterly report.

Figure 3-1. Schematic of One-Rod Swell

APPENDIX

ABSTRACTS OF QUARTERLY AND TOPICAL REPORTS

Quarterly Progress Reports

Abstract

First, July-September 1969:	Test planning and design of one-rod tests. Analysis of time lags in orifice pressure drop output data.
Second, October-December 1969:	Planning of one- and nine-rod tests. Calrod heater procurement. Analysis of pressure transducer transient, time lag by a low pass filter, and orifice flowmeter transient.
Third, January-March 1970:	Calrod design. Nine-rod subchannel design and preliminary results.
Fourth, April-June 1970:	Preliminary analysis for MAYU transient code. One-rod exploratory tests. Analog-digital conversion and TAPER code for reading digital tapes on GE 635 computer.
Fifth, July-September 1970:	Preliminary simplified transient results by MAYU code. Nine-rod subchannel equipment installation. TAPER code expansion. Transient flow analysis for exponential ΔP decay in orifice output.
Sixth, October-December 1970:	Nine-rod subchannel data with single-phase water.
Seventh, January-March 1971:	Nine-rod single- and two-phase subchannel data. One-rod transient flow and film boiling data.
Eighth, April-June 1971:	One-rod transient flow and depressurization data. Analysis of transient pressure by MAYU. Freon, void fraction transient data, and MAYU prediction. CHF data with bowed one-rod, and one-rod with 1/2 sin axial flux variation.
Ninth, July-September 1971:	Nine-rod steady-state CHF data. Nine-rod transient flow and depressurization. Improved MAYU-2 using time-history input of flow, pressure, and heat flux.
Tenth, October-December 1971:	Nine-rod steady-state CHF data with various bows and clearances at 800 and 1000 psi. Summary tabulation of nine-rod transient flow and depressurization runs.
Eleventh, January-March 1972:	Description of ATLAS 8600-kW facility test section, and 16-rod test bundle. Redesign of one-rod swell test shown.

Topical Reports

1. Lahey, R. T., Jr., et al., *Out-Of-Pile Subchannel Measurements in a 9-Rod Bundle for Water at 1000 psia*, Paper 3-1, International Symposium on Two-Phase Flow, Haifa, Israel, August 1971.
2. Lahey, R. T., Jr. and Shiralkar, B. S., *Transient Flow Measurements with Sharp-Edged Orifices*, November 1970, (ASME 71-FE-30).
3. Lahey, R. T., Jr., et al., *The Analysis of Transient Critical Heat Flux*, AEC AT(04-3)-189, P.A. 55, February 1972, (GEAP-13249).
4. *Transient CHF—Experimental Results*, GEAP-13295 in preparation.
5. Gonzalez, J. M. and Lahey, R. T., Jr., *An Exact Solution of Flow Decay Transients in Two-Phase Systems by the Method of Characteristics*, submitted for presentation at ASME Annual Winter Meeting 1972.