304
GEAP-10221-9
AEC RESEARCH AND
DEVELOPMENT REPORT
OCTOBER 1971

MASTER

DEFICIENT COOLING

NINTH QUARTERLY PROGRESS REPORT
JULY 1-SEPTEMBER 30, 1971

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 55

GENERAL ELECTRIC

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DEFICIENT COOLING

NINTH QUARTERLY PROGRESS REPORT
July 1 – September 30, 1971

Contributors

Engineers	R. T. Lahey
	Y. H. Kong
	D. W. Radcliffe
	B. S. Shiralkar
Engineering Assistant	Sheila A. Kiernan

Approved:

E. E. Polomik
Project Engineer

Approved:

F. A. Schraub, Manager
Core Development

Prepared for the
U.S. Atomic Energy Commission
Contract AT(04-3)-189
Project Agreement 55

ATOMIC POWER EQUIPMENT DEPARTMENT • GENERAL ELECTRIC COMPANY
SAN JOSE, CALIFORNIA 95125

38/125-Dev. Eng.
AG-1/72

GENERAL ELECTRIC

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

TABLE OF CONTENTS

ABSTRACT	vii
SUMMARY	ix
1. INTRODUCTION	1-1
1.1 General	1-1
1.2 Program Objectives	1-2
1.3 Organization of Program	1-3
2. EXPERIMENTAL WORK	2-1
2.1 Steam-Water Steady-State Critical Heat Flux in a Nine-Rod Bundle	2-1
2.2 Steam-Water Transient Flow in a Nine-Rod Bundle	2-1
2.3 Transient Analysis	2-19
REFERENCES	1
DISTRIBUTION	3

LIST OF ILLUSTRATIONS

Figure	Title	Page
2-1	Nine-Rod Test Section Showing Grid Spacer Locations	2-2
2-2	Nine-Rod Steady-State CHF Data (Symmetrical Assembly), 800 psi	2-4
2-3	Nine-Rod Steady-State CHF Data (Symmetrical Assembly), 1000 psi	2-5
2-4	Nine-Rod Flow Decay Transient Loop Schematic	2-6
2-5	Nine-Rod Flow ON/OFF/ON Transient Loop Schematic	2-8
2-6	Nine-Rod Depressurization Transient Loop Schematic	2-9
2-7	Flow Transient Run No. 9-150	2-11
2-8	Flow Transient Run No. 9-152	2-12
2-9	Flow Transient Run No. 9-162	2-13
2-10	Flow Transient Run No. 9-163	2-14
2-11	Flow Transient Run No. 9-181	2-15
2-12	Flow Transient Run No. 9-182	2-16
2-13	Flow Transient Run No. 9-175	2-17
2-14	Flow Transient Run No. 9-170	2-18
2-15	MAYU-2 Prediction of Steam-Water Transient CHF (Constant Inlet Flow)	2-20
2-16	MAYU-2 Prediction of Steam-Water Transient CHF (Actual Inlet Flow)	2-21
2-17	Void Fraction Response to Flow Decay (Freon-114)	2-23
2-18	Void Fraction Response to Flow Decay (Freon-114)	2-24

LIST OF TABLES

Table	Title	Page
1-1	Task Designation and Content	1-3
2-1	Nine-Rod Steady-State CHF Data in a Symmetrical Assembly	2-3
2-2	Nine-Rod Transient Flow Runs (1000 psi)	2-10
2-3	Single-Rod Steam-Water Depressurization, CHF Data and Analysis	2-19

ABSTRACT

Nine-rod steady-state CHF data were taken at 1000 and 800 psi for comparison with CHF data obtained during transient flow and depressurization. Seventy-seven transient flow and depressurization runs were made, and several plots of transient flow are presented, showing conditions during the transients. The MAYU code has been improved by providing time-history input of flow, pressure, and heat flux, and accommodations for nonuniform axial heat flux. This new code, MAYU-2, shows an improvement over MAYU in the prediction of time to CHF during steam-water depressurization. MAYU-2, and the TRFLOW program, which incorporates an extension of Zuber's void propagation analysis, show excellent agreement with each other and the data for prediction of transient voids in Freon-114.

SUMMARY

Steady-state critical heat flux data were taken, in a nine-rod symmetrical 3 X 3 bundle with a 0.564-inch-o.d. calrods of a 6-foot heated length, at pressures of 1000 and 800 psi and mass fluxes of 0.25, 0.50, and 1.00×10^6 lb/h-ft². Typical reactor grid spacers were used, with the last spacer located 19-1/2 inches upstream of the end of the heated length. These data will be used for comparison with critical heat flux (CHF) data taken with transient flow and depressurization. The data compare very well with those taken several years ago in an almost identical geometry.

Seventy-seven transient flow and depressurization runs have been made in the nine-rod test section. Transient flow runs were performed at a pressure of 1000 psi, with mass flux varied from 1.00/0.50, 1.00/0.30, and $0.60/0.30 \times 10^6$ lb/h-ft². Power was held constant during the runs. Time for the flow to decay from the high to the low value was in the range of approximately 0.3 to 1.00 second. CHF did not result during the flow transient but appeared about 1 second after establishment of the low flow. Steady-state CHF criteria were found to be an accurate guide for prediction of the CHF occurrence at the low flow conditions. Data reduction and analysis of the data are not complete; however, several computer plots are presented which show typical results during the flow transients.

The MAYU program, for prediction of CHF conditions during transient flow and depressurization with constant uniform heat flux, has been improved. The improved program, called MAYU-2, provides for input of an arbitrary time history of system flow, pressure, and heat flux, and can also be used for nonuniform axial heat flux. Single-rod steam-water depressurization CHF data showed that MAYU predictions for time to CHF were very conservative. The same data processed with MAYU-2 show predicted times to CHF to be within 10 to 20% of the data.

Freon-114 transient void fraction predictions, using MAYU and presented in the previous quarterly report, were rerun with more realistic void-quality parameters ($\bar{V}_{gj}$ and C_o) which resulted in excellent agreement with actual data. Another direct prediction of transient voids was made with the TRFLOW program, which incorporates an extension of the void propagation analyses of Zuber. The predictions of TRFLOW were in excellent agreement with MAYU, and the data. These results indicate that MAYU and MAYU-2, although they include assumptions of homogeneous flow and steady-state void-quality relations, are highly useful programs which yield accurate predictions.

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 $1000 \text{ Btu/h-ft}^2\text{-}^\circ\text{F}$ or less. 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

Acquisition of the ATLAS 8500 kW (eventually 17,000 kW) 2400 psi heat transfer facility at the General Electric, San Jose, California, site has stimulated interest in obtaining full-length multirod heat transfer data in 4 X 4 and 7 X 7 heater arrays. Such data will be more valuable than most existing data due to its closer simulation of actual flow and heat distributions in large reactor bundles.

The new facility has a power supply with capability of very rapid transients. Also, the facility includes a high speed data acquisition system which is ideally suited for the transit power and flow tests required on this program.

The work scope of this program has been reoriented to include transient power and flow tests in the large ATLAS heat transfer facility with 4 X 4 heater arrays with typical reactor lengths of 12 feet. These will use cosine type axial heat flux variation with direct current powered resistance heaters. In addition, Task 8 has been expanded to include subchannel enthalpy measurements in a nine-rod bundle to better evaluate the local effects of geometry changes. The large amount of work added by inclusion of the 16-rod arrays and subchannel measurements necessitated some reorientation within the program subtasks of specific tests considered of lesser priority, or that had been accomplished elsewhere since the inception of this program.

Tasks 3, 4, 5 and 8 in Table 1-1 designate the revised experimental program. Tasks 4 and 8 show the inclusion of 16-rod transient work and 9-rod subchannel measurements, respectively.

Table 1-1
TASK DESIGNATION AND CONTENT

No.	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 at constant coolant pressure drop
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 swelling b. Subchannel flow and enthalpy measurements.

2. EXPERIMENTAL WORK

2.1 STEAM-WATER STEADY-STATE CRITICAL HEAT FLUX IN A NINE-ROD BUNDLE

(E. E. Polomik, Y. H. Kong)

Steady-state CHF data were taken in a symmetrical assembly (i.e., with no reduced clearances or purposely bowed rods) whose clearances were typical of those in an actual BWR fuel bundle. Details of the test section are shown schematically in Figure 2-1. Grid spacers were made from typical fuel spacer components, so the corner and center rod-spacer geometry simulated that in an actual fuel bundle. Spacers were located on 19-1/2-inch centers.

CHF data were obtained in the symmetrical test section at pressures of 800 and 1000 psi, with mass fluxes at each pressure of 0.25, 0.50, and 1.00×10^6 lb/h-ft². These "symmetrical" data will be used later, for comparisons with data taken in the same bundle with reduced rod-wall corner clearances and bowed rods, to determine the effect of such geometry changes on CHF. The symmetrical data will also be used for comparison with CHF conditions obtained during flow and pressure transients in this nine-rod test section.

The steady-state CHF data are tabulated in Table 2-1 and are shown in Figures 2-2 and 2-3.

The CHF data at 800 psi are about 15% higher on the average than data at 1000 psi and corresponding mass flux and quality. This is the usual pressure trend that would be expected. Acquisition of steady-state data at 800 psi was interrupted to take transient data, and was not resumed due to failure of three or four heaters. Additional data at 800 psi will be taken, when steady-state operations are resumed, to establish a firmer trend with quality at each mass flux.

The 1000 psi data in Figure 2-3 show the usual general trend of decreasing CHF with increasing quality. The mean data of Janssen¹ are shown for comparison. The latter data were taken in an almost identical geometry, with the exception that the last spacer was 18 inches upstream of the end of the heated length, with the balance of spacers on 18-1/2-inch axial (compare Figure 2-1 where the respective numbers are 19-1/2 inches). The present data compare favorably with the largest deviations of $\pm 20\%$ occurring at the mass flux of 0.50×10^6 lb/h-ft².

More scatter exists in the present 1000-psi data than is usually obtained. Part of this may be due to the fact that the data were taken in three assemblies. There is a possibility that one assembly may distort enough during operation, compared with another assembly, to account for some of the scatter.

Some of the scatter is attributable to different boiling lengths at the same quality, at different inlet subcoolings. This has been noted before and indicates CHF is not strictly a local condition (that is, a function of local quality and heat flux) but that the upstream history (boiling length) has some influence on the CHF. Thus, in Figure 2-3, at 22% quality, it may be noted that the higher subcooling (shorter boiling length) results in about 15% higher heat flux at the same quality. Similarly, at 37% quality and 0.50×10^6 lb/h-ft², a 20% higher heat flux is obtained at the same quality.

2.2 STEAM-WATER TRANSIENT FLOW IN A NINE-ROD BUNDLE

(Y. H. Kong, R. T. Lahey, E. E. Polomik, D. W. Radcliffe)

2.2.1 Equipment

The general arrangement of the test section, test loop, and basic equipment is shown schematically in Figure 2-4. As noted in the Fourth Quarterly Report (GEAP-10221) Subsection 4.2 and related figures, a simple valve positioning operation could not be used to get flow decays, due to the large change that resulted in pressure loss distribution and the fact that CHF may occur along the transient and preclude flow adjustments at the low values. Thus, a feedback flow control system was devised, as shown in Figure 4-6 of the forementioned report and in Figure 2-4 of this report.

The GEMAC flow controller, in the AUTO position, compares the actual flow from the Potter vane meter with a preset desired flow value, and an error signal is produced if the two are different. The error signal is amplified to produce a controller output which, through the Conoflow transducer, actuates the valve in the proper direction to reduce the error to zero at the desired flow.

Figure 2-1. Nine-Rod Test Section Showing Grid Spacer Locations

Table 2-1
NINE-ROD STEADY-STATE CHF DATA
IN A SYMMETRICAL ASSEMBLY

Run No.	Assembly No.	Mass Flux $G/10^6$ (lb/h-ft ²)	Heat Flux $q''/10^3$ (Btu/h-ft ²)	Subcooling Δh (Btu/lb)	Steam Quality x (wt %)	Boiling Length ($l_{in.}$)
800 psi						
379	21	0.2517	437.6	328.2	49.6	36.7
383	21	0.2421	296.0	30.5	64.0	67.3
143	14	0.2538	345.2	108.1	60.6	57.2
148	14	0.2636	447.5	290.3	52.8	40.1
245	15	0.5218	662.2	267.4	32.1	32.6
142	14	0.5108	518.1	65.7	47.4	60.0
146	14	0.5122	657.1	37.5	37.4	37.45
382	21	0.4872	482.4	124.9	37.4	48.4
381	21	0.5046	517.3	159.8	34.2	42.9
243	15	1.0076	698.0	91.0	25.7	47.5
244	15	0.9933	685.6	66.3	29.2	54.1
384	21	0.9833	708.8	117.9	23.2	41.4
385	21	0.9929	599.0	29.1	29.7	63.0
1000 psi						
375	21	0.2490	295.8	27.3	66.4	67.7
378	21	0.2696	443.5	345.2	44.7	32.8
233	15	0.2525	296.6	64.0	60.0	61.8
236	15	0.2471	328.9	99.59	64.0	58.0
238	15	0.2515	368.5	170.5	60.9	50.3
147	14	0.2535	434.0	322.8	51.8	36.8
155	14	0.2570	355.8	137.5	61.0	53.5
227	15	0.4990	495.2	123.1	40.0	48.8
216	15	0.5653	539.9	144.6	34.5	43.7
218	15	0.5086	501.4	147.7	36.0	44.0
229	15	0.5063	460.0	46.3	47.1	62.5
231	15	0.5076	558.4	213.3	32.6	35.8
144	14	0.5040	523.7	99.3	46.5	54.2
145	14	0.5215	620.0	226.1	35.6	36.5
374	21	0.4977	395.2	14.1	44.9	68.7
376	21	0.5176	452.0	98.2	36.6	51.0
377	21	0.5186	546.3	212.5	30.0	34.4
220	15	0.9956	620.4	81.23	24.6	47.6
140	13	0.9787	534.8	7.99	31.2	69.3
141	13	0.9921	764.8	155.8	21.7	34.3
240	15	1.0039	670.7	39.0	21.6	39.0
223	15	1.0087	589.8	65.6	24.7	51.0
225	15	0.9964	674.3	129.0	20.4	36.4
373	21	0.9937	515.2	6.91	29.7	69.5
380	21	1.0019	623.7	94.5	22.4	43.6

Figure 2-2. Nine-Rod Steady-State CHF Data (Symmetrical Assembly), 800 psi

GEAP-10221-9

Figure 2-3. Nine-Rod Steady-State CHF Data (Symmetrical Assembly), 1000 psi

Figure 2-4. Nine-Rod Flow Decay Transient Loop Schematic

The GEMAC flow controller was adjusted so the combination of settings on sampling rate, reset, proportional band, and bleed valve settings on the boosters gave the fastest smooth flow decay. The best settings resulted in flow decays from $G = 1.0$ to $0.50 \times 10^6 \text{ lb/h-ft}^2$ in 0.2 to 0.3 second. Slower flow decays could be obtained by adjustment of the bleed valves at the boosters.

The desired transient data for flow, pressure, power, inlet temperature, and CHF temperatures were recorded on a precalibrated Ampex magnetic tape recorder at 15 in./sec. Fourteen magnetic tape channels were available. In addition, all data were recorded at 4- to 5-second intervals on a DYMEC digital data acquisition system. All key transient data were also displayed on a Sanborn recorder, so a visual check was available on the course of the transient and the occurrence of CHF. The Sanborn CHF temperature markers were connected, through switches and a relay, to the power supply. Adjustments were provided on each switch so that power could be tripped off when any of the CHF thermocouples reached a preset value. These power trip temperatures were set in the range of 800 to 1000°F to avoid excessive overheating of the Calrod heating elements.

The sequence of operations for a transient flow decay run is indicated in the diagram shown at the upper right-hand corner of Figure 2-4. The run is identified by voice: the tape data marker was turned on, and the flow decay was initiated by turning the GEMAC control to AUTO. The purpose of the tape marker is to identify the position on the tape at which digital data reduction is to be made when the tape is processed later. When the desired data have been recorded—as judged from observation of the Sanborn recordings—the flow controller is turned back to MANUAL which initiates return to the initial flow; the tape marker is turned off, signalling the end of desired data to be reduced; and the tape recorder is stopped.

Another type of flow transient is one in which a complete loss of flow results but is re-established after a given off time. It is shown schematically on Figure 2-5 as a flow ON/OFF/ON transient. This transient partially simulates a recirculation line break in reactor coolant, where flow is stopped and then re-established due to flashing in the lower plenum. The ON/OFF/ON flow transient is accomplished by means of the automatic flow valve control system shown in the lower right-hand corner of Figure 2-5.

The pneumatic valve controller has three positions: No. 1 is for manual control, and Nos. 2 and 3 are independent systems, on each of which a given valve position can be set. For this transient, position 3 is used to set the desired high initial flow, and position 2 is off. The 6-second timer is set for the desired flow OFF time. Actuation of the push-button causes the controller and valve to switch from the high flow to the OFF positions for the prescribed time. A tape marker is actuated to identify desired data on the tape, and a 0.4-second delay provides sufficient time to record the initial conditions at the start of the transient.

At the end of the prescribed time for flow OFF, the 6-second timer switches back to the original flow and the tape marker de-energizes to indicate the end of data on the tape. The 15-second timer is connected to the 6-second timer; and the former is used to produce an adjustable delay period, at the end of the OFF period, during which time re-establishment of flow is taking place. The two timers can be connected in combination to produce any combination of 21-second total time for the flow OFF and delay period, when flow and test conditions are being re-established.

The equipment and instrumentation for a blowdown or depressurization transient is shown in Figure 2-6. Depressurization is accomplished by a quick-opening 3-inch valve which vents the steam-water mixture to atmosphere before it reaches the condenser. The rate of depressurization is controlled by handwheel adjustment of the valve stem's maximum opening. A microswitch on the valve stem provides a signal which indicates the time the valve is open. The valve, which bypasses the subcooler, is closed, and sufficient subcooling is maintained to the test section inlet to avoid flashing, at the lowest pressure, in the main pumps and flow metering elements. The control system shown in Figure 2-6 is essentially the same as that for the flow ON/OFF/ON transient previously described. Here, the control system keeps the valve closed or open to the position desired for a particular depressurization rate. A 0.4-second delay timer, to record initial conditions, and a tape marker are also used.

Figure 2-5. Nine-Rod Depressurization Transient Loop Schematic

2.2.2 Experimental Data

A considerable number (77) of transient flow and depressurization runs were made during this quarter in the nine-rod test section described in subsection 2.1.

Fifty-three transient flow runs were made at 1000 psi with mass flux varied in the ranges of 1.0/0.5, 1.0/0.3, and 0.6/0.3 $\times 10^6$ lb/h-ft². Various times for the transients and levels of heat flux were used until definite CHF responses were obtained. Where it was safely possible, temperatures were allowed to continue beyond CHF, and rewet for the ON/OFF/ON transients so the complete history of the transient could be obtained.

Twenty-four depressurization runs were made with pressure varying from 1000 to about 800 psi. Nominal blowdown rates of 50 and 100 psi/sec were used at mass fluxes of 0.25, 0.50, and 1.0 $\times 10^6$ lb/h-ft².

Data reduction has extended beyond the quarter, and reduction and analysis are not yet complete. However, a few typical cases of the results are shown in Figures 2-7 through 2-14.

The initial conditions at the beginning of the transients and the changes in flow are shown in Table 2-2.

Table 2-2
NINE-ROD TRANSIENT FLOW RUNS (1000 psi)

Figure No.	Run No.	Initial Conditions			$G_{\min}$ Minimum Mass Flux during Transient (approx.)	τ Time at Min. Mass Flux (sec) (approx.)	CHF
		$q''/10^3$ (Btu/h-ft ²)	X Exit Quality (wt %)	G Mass Flux (lb/h-ft ²)			
2-7	9-150	424.9	19.32	1.007	0.5	7.4	No
2-8	9-152	457.8	22.78	0.9827	0.5	25	Yes
2-9	9-162	426.1	19.73	0.9961	0.5	15.5	Yes
2-10	9-163	441.5	20.60	0.9953	0.5	12.5	Yes
2-11	9-181	366.3	14.81	1.014	0.3	8	Yes
2-12	9-182	367.4	15.54	1.021	0.3	27	Yes
2-13	9-175	347.6	28.08	0.6221	0.3	18	Yes
2-14	9-170	357.1	27.2	0.6299	0.3	19	Yes

Figure 2-7 shows that flow can be reduced from 1.0 to 0.5 $\times 10^6$ lb/h-ft² for over 7 seconds, at a constant heat flux of 424,900 Btu/h-ft² with no CHF resulting. The test section was operating at the lower flow and higher quality, with no indication of CHF due to the flow decay transient itself, or due to the higher quality after establishment of equilibrium at the lower flow.

Runs in which CHF was obtained at the lower flow are shown in Figures 2-8 through 2-10. Here also the flow was decayed from $G/10^6$ of 1.0/0.5 lb/h-ft². The CHF occurs at about 1/2 to 1 second after reaching the lower flow.

Transient flow decays with $G/10^6$ varied from 1.0/0.3 lb/h-ft² are shown in Figures 2-11 and 2-12. Figure 2-11 indicates that the transition boiling condition is removed by re-establishment of the flow. When the low flow was maintained for a considerably longer time of 27 seconds (Figure 2-12), higher temperatures of about 950 to 1050°F were reached. Rewetting again occurred by re-establishment of the flow and the resultant reduction of quality.

Flow decays of $G/10^6$ from 0.6/0.3 lb/h-ft² are illustrated in Figures 2-13 and 2-14. The time at low flow is about the same for these two runs: 18 and 19 seconds. The effect of an additional 10,000 Btu/h-ft², in run 9-170 (Figure 2-14) is to increase the heater temperature about 100°F.

Figure 2-7. Flow Transient Run No. 9-150

FLOW TRANSIENT RUN NO 9-152

NOVEMBER 09, 1971

Figure 2-8. Flow Transient Run No. 9-152

FLOW TRANSIENT RUN NO 9-162

NOVEMBER 09, 1971

Figure 2-9. Flow Transient Run No. 9-162

FLOW TRANSIENT RUN NO 9-163

NOVEMBER 09, 1971

Figure 2-10. Flow Transient Run No. 9-163

Figure 2-11. Flow Transient Run No. 9-181

Figure 2-12. Flow Transient Run No. 9-182

FLOW TRANSIENT RUN NO 9-175

NOVEMBER 15, 1971

Figure 2-13. Flow Transient Run No. 9-175

Figure 2-14. Flow Transient Run No. 9-170

In general, the thermocouple on Rod "I" indicated a stronger CHF, got hotter and rewet more slowly than the thermocouple on the diametrically opposite corner rod (Rod "A"). This is probably due to a number of effects, including a closer clearance (8 to 10 mils) and variations in the thermocouple mounting.

The foregoing flow transient runs are typical of data reduced thus far. Blowdown runs and the balance of flow transient data will be reported as soon as data reduction and analysis are completed.

2.3 TRANSIENT ANALYSIS (R. T. Lahey)

2.3.1 Transient CHF Predictions

In the previous quarterly report (GEAP-10221-8), it was observed that the transient computer code, MAYU, apparently predicted the onset of CHF quite conservatively for many of the depressurization transient data. This conclusion was based on analysis using the appropriate steady-state CHF correlation and assumed constant system flow inputs. Closer examination of the test data indicated that the inlet flow rate was not truly constant in time; in fact, in many cases it dipped appreciably near the start of the depressurization transient.

The MAYU code was not programmed to input a time-flow history such as was experimentally observed in the single-rod data. Moreover, MAYU is able to handle only cases with constant, uniform, axial heat flux. Thus, it was decided to write a new version of the code which can handle an arbitrary time history of system flow, pressure, and heat flux as well as accommodate nonuniform axial heat flux profiles. This code, called MAYU-2, uses the same conservation equations as MAYU; however, it has been written in FORTRAN-IV using the classical second-order method of characteristics numerical scheme.²

MAYU-2 agrees quite well with MAYU predictions for equivalent inputs, but is far more flexible. Specifically, MAYU-2 can easily handle the observed inlet flow fluctuations during the single-rod depressurization runs. The analysis presented previously in Table 2-2 of GEAP-10221-8 was rerun with MAYU-2 using the observed inlet flow rate. A comparison between Figures 2-15 and 2-16 indicates that by using the actual inlet flow, the CHF prediction is much more realistic (now only slightly conservative). In both cases the steady-state CHF correlation was used to estimate the critical quality (X_{crit}) at which CHF would be expected. The other runs previously analyzed were also rerun with MAYU-2 and showed improved prediction capabilities as listed in Table 2-3.

Table 2-3
SINGLE-ROD STEAM-WATER DEPRESSURIZATION,
CHF DATA AND ANALYSIS

Run No.	Initial Conditions			Data	Time to CHF (sec)	
	$q''/10^6$ (Btu/h-ft ²)	$G/10^6$ (lb _m /h-ft ²)	Steam Quality $X_{(exit)}$		MAYU Prediction	MAYU-2 Prediction
167	0.669	0.96	0.058	No CHF	No CHF	No CHF
171	0.829	0.99	0.085	4.2	2.53	3.85
172	0.638	0.52	0.136	6.0	3.15	4.70
173	0.714	0.49	0.151	4.13	1.64	3.70

2.3.2 Transient Void Fraction Predictions

As discussed previously (GEAP-10221-8), an experiment was performed in Freon-114 in which the exit void fraction was measured during various flow transients. These data afford a unique opportunity to check the prediction capability of analytical techniques such as MAYU and MAYU-2. Analytical predictions using MAYU had previously (GEAP-10221-8) indicated the correct trend but incorrect end points. Further analysis of the situation has indicated that the values assumed for parameters C_0 and V_{gj} in the Zuber-Findlay void-quality correlation were not appropriate for the test conditions that existed.

Figure 2-15. MAYU-2 Prediction of Steam-Water Transient CHF (Constant Inlet Flow)

Figure 2-16. MAYU-2 Prediction of Steam-Water Transient CHF (Actual Inlet Flow)

The predictions were rerun using the values recommended by Zuber:³

$$C_0 = 1.13,$$

$$\bar{V}_{gj} = 1.41 \left[\frac{\sigma g (\rho_f - \rho_g) g_c}{\rho_f^2} \right]^{1/4}$$

It can be seen in Figures 2-17 and 2-18 that the predictions are now quite good, compared with those previously presented. Except for a small deviation on Figure 2-17, the predicted transient void fraction is within the error band of measurements. These results support the validity of MAYU type predictions.

The other predictions (TRFLOW), shown in Figures 2-17 and 2-18, are due to an extension of the void propagation analysis of Zuber.⁴ This analysis was performed in order to obtain a direct prediction of the transient void fraction, in contrast to the MAYU predictions, which assumes the validity of a steady-state void-quality correlation during transients. As can be seen in Figures 2-17 and 2-18, the agreement is quite good.

The analysis that gave the TRFLOW predictions extended Zuber's original work⁴ (which discussed sinusoidal inlet flow variations with saturated inlet conditions) to subcooled inlet conditions and exponential flow decays. The kinematic wave equation which was integrated was

$$\frac{D\langle\alpha\rangle}{Dt} = \frac{\partial\langle\alpha\rangle}{\partial t} + C_k \frac{\partial\langle\alpha\rangle}{\partial z} = \left[1 - \frac{C_0 (\rho_f - \rho_g)}{\rho_f} \langle\alpha\rangle \right] \frac{\langle\Gamma_g\rangle}{\rho_g}$$

where

$$C_k = \frac{\Delta}{dt} = C_0 V_{if} + \bar{V}_{gj} + \frac{C_0 (\rho_f - \rho_g)}{\rho_f} \int_{\lambda}^z \frac{\langle\Gamma_g\rangle}{\rho_g} dz$$

and

$$\lambda = \text{Boiling boundary, and}$$

$$\langle\Gamma_g\rangle = \frac{q'' P_H}{A_{xs} h_{fg}}$$

P_H is the heated perimeter and A_{xs} is the flow cross section. It was assumed that $V_{if} = V_{\infty} (1 + \beta e^{-\omega t})$ and integrated to yield an exact solution of the form $\langle\alpha(z,t)\rangle$.

Since the solution function $\langle\alpha(z,t)\rangle$ is rather complicated, it was evaluated numerically, using a time-share code (TRFLOW), and the results were plotted in Figures 2-17 and 2-18.

In summary, it appears that the assumptions made in MAYU-2 and MAYU yield analytical results which agree quite well with transient CHF and void fraction data. Moreover, the void propagation equation (which assumes slip flow) agrees quite well with the predictions of MAYU, even though MAYU assumes homogeneous flow and uses steady-state void-quality relations. This is indeed encouraging since MAYU and MAYU-2 are far more versatile and useful than the void propagation analysis, and are apparently as accurate.

Figure 2-17. Void Fraction Response to Flow Decay (Freon-114)

Figure 2-18. Void Fraction Response to Flow Decay (Freon-114)

REFERENCES

1. Janssen, E., *Two-Phase Flow and Heat Transfer in Multirod Geometries, 19th Quarterly Progress Report, April 1-June 30, 1970*, USAEC Contract A7(04-3)-189, P.A. 44.
2. Lister, M., "Numerical Solution of Hyperbolic Partial Differential Equations by the Method of Characteristics," *Mathematical Methods for Digital Computers*, John Wiley, 1967.
3. Zuber, N., Staub, F. W., Bijwaard, G., and Kroeger, P. G., *Steady-State and Transient Void Fraction in Two-Phase Flow Systems, Final Report for the Program of Two-Phase Flow Investigation*, 1967, (GEAP-5417).
4. Zuber, N. and Staub, F. W., "The Propagation and the Wave Form of the Vapor Volumetric Concentration in Boiling, Forced Convection System under Oscillatory Conditions," *Int. J. Heat and Mass Transfer*, 1966.

DISTRIBUTION

Argonne National Laboratory Reactor Engineering Division 9700 South Cass Avenue Argonne, Illinois 60439 Attn: M. Petrick	1	General Electric Company Atomic Power Equipment Department 175 Curtner Avenue San Jose, California 95125 Attn: Dr. F. Schraub	2
Argonne National Laboratory Reactor Engineering Division 9700 South Cass Avenue Argonne, Illinois 60439 Attn: R. Stein	1	General Electric Company Research and Development Center P. O. Box 8 Schenectady, New York 12301 Attn: Fred Staub	1
Atomics International P. O. Box 309 Canoga Park, California 91305 Attn: L. Bernath	1	Knolls Atomic Power Laboratory General Electric Company P. O. Box 1072 Schenectady, New York 12301 Attn: Document Librarian	1
Babcock & Wilcox Company Atomic Energy Division P. O. Box 1260 Lynchburg, Virginia 24505 Attn: D. F. Judd	1	Massachusetts Institute of Technology Department of Mechanical Engineering Cambridge, Massachusetts 02139 Attn: Dr. W. Rohsenow	1
Brookhaven National Laboratory Chemical Engineering Division Upton, New York 11973 Attn: Dr. O. E. Dwyer	1	Massachusetts Institute of Technology Department of Mechanical Engineering Cambridge, Massachusetts 02139 Attn: Dr. P. Griffith	1
Columbia University Department of Chemical Engineering New York, New York 10027 Attn: J. Casterline	1	North Carolina State University Department of Chemical Engineering Raleigh, North Carolina 27607 Attn: Dr. J. K. Ferrell	1
Combustion Engineering, Inc. Nuclear Division P. O. Box 500 Windsor, Connecticut 06095 Attn: Dr. P. Zmola	1	Oak Ridge National Laboratory P. O. Box X Oak Ridge, Tennessee 37830 Attn: H. W. Hoffman	2
E. I. duPont de Nemours and Company Wilmington, Delaware 19898 Attn: J. S. Neill	1	Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352 Attn: D. Rowe	1
General Electric Company Atomic Power Equipment Department 175 Curtner Avenue San Jose, California 95125 Attn: Dr. S. Levy	1	Pacific Northwest Laboratory P. O. Box 999 Richland, Washington 99352 Attn: F. W. Zaloudek	1
Advisory Committee on Reactor Safeguards U. S. Atomic Energy Commission Washington, D. C. 20545	1	Stanford University Department of Mechanical Engineering Stanford, California 94305	1

University of Michigan Dept. of Chem. & Metallurgical Engineering Ann Arbor, Michigan 48103 Attn: R. Balzhiser	1	U. S. Atomic Energy Commission Div. of Reactor Development & Technology Washington, D. C. 20545 Attn: Chief, Special Technology Br.	2
Westinghouse Electric Corporation Atomic Power Division P. O. Box 355 Pittsburgh, Pennsylvania 15230 Attn: L. S. Tong	2	U. S. Atomic Energy Commission Div. of Reactor Development & Technology Washington, D. C. 20545 Attn: Chief, Core Design Branch	1
Westinghouse Electric Corporation Bettis Atomic Power Laboratory P. O. Box 79 West Mifflin, Pennsylvania 15122 Attn: S. Green	1	U. S. Atomic Energy Commission Div. of Reactor Development & Technology Washington, D. C. 20545 Attn: Chief, Water Projects Branch	1
National Aeronautics & Space Admin. Lewis Research Center 21000 Brookpark Road Cleveland, Ohio 44135 Attn: Librarian	2	U. S. Atomic Energy Commission Div. of Reactor Development & Technology Washington, D. C. 20545 Attn: Chief, Thermal Reactor Safety	1
Atomic Energy of Canada Limited Chalk River, Ontario, Canada Attn: G. A. Wikhammer	1	U. S. Atomic Energy Commission Division of Reactor Licensing Washington, D. C. 20545 Attn: Dr. R. C. DeYoung	2
Atomic Energy of Canada Limited Chalk River, Ontario, Canada Attn: H. Smedley	1	U. S. Atomic Energy Commission San Francisco Operations Office 2111 Bancroft Way Berkeley, California 94704 Attn: Director, Contracts Division	2
Canadian General Electric Co. Ltd. Atomic Power Department 107 Park Street North Peterborough, Ontario, Canada Attn: Dr. J. T. Rogers	1	U. S. Atomic Energy Commission Division of Technical Information Extension P. O. Box 62 Oak Ridge, Tennessee 37831	3
Canadian Westinghouse Co. Ltd. Atomic Fuel Department Port Hope, Ontario, Canada Attn: J. Howieson	1	Idaho Nuclear Corporation P. O. Box 1845 Idaho Falls, Idaho 83401 Attn: Mr. J. C. Haire	1
University of Minnesota Dept. of Chemical Engineering Minneapolis, Minnesota 55455 Attn: Dr. Herbert S. Isbin	1	USAEC Site Representative General Electric Company Sunnyvale, California 94086 Attn: Russell H. Ball, RDT Senior Site Rep.	1
U. S. Atomic Energy Commission Director, Division of Reactor Licensing Washington, D. C. 20545	3	Georgia Institute of Technology School of Mechanical Engineering Atlanta, Georgia 30332 Attn: Dr. Novak Zuber	1
U.S. Atomic Energy Commission Director, Division of Reactor Standards Washington, D. C. 20545	2		

GEAP-10221-9

Georgia Institute of Technology
School of Mechanical Engineering
Atlanta, Georgia 30332
Attn: Dr. A. E. Bergles

1 U. S. Atomic Energy Commission
Div. of Reactor Standards
Washington, D. C. 20545
Attn: Dr. M. Rosen

1

University of Cincinnati
Dept. of Chemical & Nuclear Engineering
Cincinnati, Ohio 45221
Attn: Dr. Joel Weisman

William B. Cottrell, Director
Nuclear Safety Program
1 Oak Ridge National Laboratory
Union Carbide Corporation
Post Office Box Y
Oak Ridge, Tenn. 37830

4