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DECEMBER 1970

MASTER

**DEFICIENT COOLING
SIXTH QUARTERLY PROGRESS REPORT
OCTOBER 1-DECEMBER 31, 1970**

U.S. ATOMIC ENERGY COMMISSION
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SIXTH QUARTERLY PROGRESS REPORT October 1–December 31, 1970

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SUMMARY

Nine-rod subchannel sampling with typical grid spacers, 4 inches upstream of the sampling section, was accomplished for the first time with the simultaneous sampling of three unique subchannels in the 3 X 3 array. Data were taken with 80 to 100°F water at 100 and 1000 psig, several exit qualities, and mass velocities of 0.5, 1.0, 1.5, and 2 X 10⁶ lb/h-ft².

The single phase runs showed a good mass balance between the total measured flow and that calculated from subchannel measurements. It was concluded that the center subchannel measurements were reliable. The corner and side subchannel measurements are under some question due to a possibility of leakage past the sampling tube gaskets. The general effect of the grid spacer, when data is compared to that in a clean channel with button spacers instead of grids, is to decrease flow in the center channels and increase it in the side and (apparently) the corner channels.

The two-phase data showed a mass defect of over 10% which was attributed to loss of sampling seals discovered at disassembly. This data will be rerun with redesigned sampling tubes which should avoid leakage.

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 Btu/h-ft², 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 Btu/h-ft² 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,
- c. Oscillation of power into and out of the transition boiling region to determine effects of prolonged operation at CHF. (Note: This case will be omitted in future reports. See revisions in the work scope discussed in paragraph 1.3 below.)
- d. Transient decrease of pressure which causes flashing and potentially deficient cooling conditions due to increased void content.

Furthermore, it is proposed to study, under transient conditions, 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, electrically-heated rods.
- 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, which will be ready for operation in the spring of 1971, will have 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 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 below 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	<ul style="list-style-type: none"> a. Review current technology and plan appropriate tests to integrate with required safety programs b. Analysis as necessary to determine test parameters
2a	Design	<ul style="list-style-type: none"> a. Design new equipment or modifications of existing equipment to carry out test plans
2b	Manufacture	<ul style="list-style-type: none"> a. Manufacture or purchase new equipment required
3	One-Rod Exploratory Transient CHF Tests	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> 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	<ul style="list-style-type: none"> a. Steady-state CHF with simulated swelling at constant coolant pressure drop
6	Data Reduction and Analysis	<ul style="list-style-type: none"> a. As required for test Tasks 1, 3, 4, 5, and 8
7	Project Administration, Reporting, and Recommendations	<ul style="list-style-type: none"> a. Administration of entire program b. Regular and topical reporting c. Future recommendations for continued efforts
8	Multirod Geometry Changes 9-Rod	<ul style="list-style-type: none"> a. Steady-state CHF data with spacer components and simulated fuel rod bowing and swelling b. Subchannel flow and enthalpy measurements.

2. EXPERIMENTAL WORK (TASK 8)

2.1 NINE-ROD SUBCHANNEL SAMPLING MEASUREMENTS

Subchannel data have been taken in the subchannel sampling system previously discussed in the third (GEAP-10196) and fifth (GEAP-10221-5) quarterly reports.

Prior to actual data acquisition, the nine-rod bundle shown schematically in Figure 2-1 was installed without splitters which enclose the flow areas to be sampled. Typical grid-type spacers shown in Figure 2-2 were used to maintain rod clearances. These spacers were located on 19.5 inch centers, with the last spacer being 4 inches ahead of the sampling point (i.e., the end of the heated length). Calibration runs were then made to determine what the isokinetic settings should be during the subsequent flow sampling with splitters installed. The measured pressure differentials were found to correlate with dynamic head, indicating slight impacting. The resultant calibration curves for a system pressure of 100 psig and 1000 psig are shown in Figures 2-3 and 2-4, respectively. It can be noted that there is a small but noticeable increase in isokinetic δP as the system pressure increases.

The nine-rod array was reassembled with 10 mil stainless steel splitters in all three subchannels and single-phase; isokinetic subchannel data were taken using the sampling system shown schematically in Figure 2-5. System pressure was 100 psig with 100°F water temperature. These cold single-phase data are tabulated in Table 2-1. It can be seen that, with the exception of the lowest flow case, the mass defect is quite small. This indicates that the isokinetic settings shown in Figure 2-3 are consistent. These data are unique, since they are the first subchannel data ever taken successfully in which all the representative subchannels were sampled simultaneously.

High credibility is placed on the center subchannel results presented in Table 2-1 due to the low mass defect and good repeatability. However, the corner and side subchannel data, which are somewhat higher than expected are subject to question due to a possibility of leakage into the sampling system, discovered after subsequent check runs and disassembly (see discussion in the following paragraph). A corner subchannel contains approximately 2.5% of the total flow. Reliable data in the latter will be obtained by repeatability in successive tests.

Eleven two-phase runs were made at nominal conditions typical of the two-phase sampling data previously* taken in a "clean" bundle (i.e., without grid type spacers). The mass defect** on these runs was found to average greater than 10% which indicated gross leakage. The cold runs were repeated and were found to have changed drastically, particularly in the side and corner subchannels which were lower and higher respectively than the previous values tabulated in Table 2-1. The head of the pressure vessel was removed and the splitter assembly was inspected. It was found that the corner subchannel sampling tube was loose, such that leakage into it could have occurred. In addition, a borescopic inspection of the side subchannel showed that two of the Rulon gaskets were missing. A simple orifice type calculation indicated that about 13% of the side subchannel flow probably leaked out during sampling, which was approximately in proportion to the total mass defect obtained by measurement.

Since leakage apparently occurred in both the corner and side subchannel, one cannot rigorously correct the two-phase data, and hence it will be repeated in future tests.

A review of the operating log indicated that during the heatup between the initial single-phase runs and the two-phase runs, movement of the sample tubes could have occurred. This movement was due to the reassembly required to seal the steam leaks in the external O-ring unions located in the same lines as valves V-1 through V-5, shown in Figure 2-5. This movement could have easily loosened the sample tubes and caused the observed leakage. Design modifications using bellows have been made to the sample tubes to make them more flexible, thus effectively eliminating this source of error in future runs.

* Two-Phase flow and heat transfer in multirod geometries AEC Contract AT(04-3)-189, PA44: "Subchannel and Pressure Drop Measurements in a Nine-Rod Bundle for Diabatic and Adiabatic Conditions," R. T. Lahey, et al., GEAP-13049, March 1970.

** $(\text{Total Measured Bundle Flow} - \text{Calculated Total of Subchannel Measurements}) / (\text{Total Measured Bundle Flow}) \times 100$.

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

Figure 2-2. Grid-Type Spacer Details

Figure 2-3. Isokinetic Calibration—100 psig, 80°F

Figure 2-4. Isokinetic Calibration—1000 psig, 80°F

Table 2-1
SINGLE PHASE MEASURED MEAN SUBCHANNEL MASS VELOCITIES
100 psig SYSTEM PRESSURE, 100° F

Run No.	$\bar{G}/10^6$ Based on Total Flow Area	Subchannel $\bar{G}/10^6$ lb/h-ft ²			Mass* Defect (%)
		No. 1 Corner	No. 2 Side	No. 3 Center	
9-44	0.489	0.550	0.462	0.421	6.8
9-44R	0.490	0.509	0.463	0.456	5.1
9-45	0.988	1.119	0.957	0.938	2.1
9-45R	0.984	1.121	0.953	0.953	1.4
9-46	1.484	1.688	1.442	1.447	1.0
9-46R	1.501	1.709	1.460	1.478	0.5
9-47	1.975	2.249	1.896	1.938	1.2

$$^+ \text{Mass Defect} \triangleq \frac{\left| \bar{G} \sum_i A_i - \sum_i (G_i A_i) \right|}{\bar{G} \sum_i A_i} \times 100$$

Figure 2-5 Flow Diagram for Three-Subchannel Sampling System (Nine-Rod Test Section)

The significance of this work is that it affords us an opportunity to appraise the effects of grid type spacers on subchannel flow distributions. Table 2-2 is a comparison of the normalized subchannel mass flux for "clean" conditions and with grid spacers. The "clean" data have been reported previously in GEAP-13049, referenced above, in which 0.570-inch diameter calrods were spaced using pin type spacers. The present data were taken in the same assembly in which 0.564-inch diameter calrods were held in place with grid type spacers. The effect of the different rod size is a negligible effect compared to the strong flow redistribution caused by the grid spacer. It is evident in Table 2-2 that at a location four inches downstream of the last grid, the flow split is quite different from the "clean" flow split. For example, the local flow is considerably lower in the center subchannel with the grid spacer, compared to that without the spacer.

This observation has important implications to reactor designers, since it shows that correct subchannel analysis of the thermal margins must include accurate spacer modeling. That is, if grid spacers are not adequately accounted for, flow distribution predictions can be grossly inaccurate.

Future effort in this area will be directed toward the accumulation of grid type spacer data for diabatic two-phase conditions, such that the effect of grids on the local enthalpy distribution can be appraised. In addition, the effect of rod bow and swell on subchannel conditions will be investigated using the sampling techniques just discussed.

Table 2-2
COMPARISON OF FLOW DISTRIBUTIONS WITH AND WITHOUT (CLEAN) GRID SPACERS

Clean*		With Grid**		G (Subchannel)/ $\bar{G}$ (Total Area)					
				No. 1 Corner		No. 2 Side		No. 3 Center	
				Clean	With Grid**	Clean	With Grid**	Clean	With Grid**
1B	0.480	9-44	0.489	0.648	1.125	0.963	0.945	1.096	0.861
		9-44R	0.490		1.039		0.945		0.931
1C	0.990	9-45	0.988	0.708	1.133	0.949	0.969	1.162	0.949
		9-45R	0.984		1.139		0.968		0.968
1D	1.510	9-46	1.484	0.725	1.137	0.954	0.972	1.119	0.975
		9-46R	1.501		1.139		0.973		0.985
1E	1.970	9-47	1.975	0.822	1.139	0.970	0.960	1.112	0.981

*Reported in GEAP-13049, PA44, Referenced above.

**This report, data of December, 1970, with last grid 4 inches upstream from the end of the heated length where sampling occurred.

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