



UNITED STATES
NUCLEAR REGULATORY COMMISSION
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DEC 7 1978

MEMORANDUM FOR: S. L. Israel, Section Leader, Reactor Systems Branch, DSS
FROM: J. A. Olshinski, Reactor Systems Branch, DSS
SUBJECT: EXPECTED FLOW RATES AND PRIMARY SYSTEM TEMPERATURES UNDER
NATURAL CIRCULATION CONDITIONS

As requested, I have briefly reviewed the frequency and extent of natural circulation events that have occurred to date. Printouts of natural circulation/loss of offsite power events were obtained from the Office of Management and Program Analysis (NRC) and from the Nuclear Safety Information Center (ORNL).

The more significant of these events include a Farley Unit 1 event in which forced reactor coolant system flow was lost for 2 hours and the St. Lucie Unit 1 event in which the primary coolant system was cooled down under natural circulation conditions.

Additionally, some data were available from natural circulation tests conducted at Maine Yankee, Calvert Cliffs Unit 2, Fort Calhoun, and Connecticut Yankee nuclear units. The data from these tests ranged from very detailed data to data that included only one or two temperature points. Although the test conditions varied significantly from test to test, the test data indicated a consistency in expected temperature differentials and natural circulation flow rates.

The Fort Calhoun test was initiated by tripping all reactor coolant pumps at a reactor power level of 35%. Thirty minutes after the reactor coolant pump trip, all feedwater addition was secured to the steam generators. Data was recorded for 105 minutes after the reactor coolant pump trip.

The Calvert Cliffs Unit 2 test was initiated from a reactor power level of 40%. Data were recorded for 45 minutes after the reactor coolant pump trip.

The Maine Yankee natural circulation test was initiated from 35% reactor power. The test was continued for 63 minutes.

The Connecticut Yankee test was initiated from hot standby conditions.

The results from the tests indicated that the initial core ΔT 's increased for 10 to 15 minutes, then decreased to a stabilized level for the remainder of the test. Figure 1 shows a plot of the core ΔT from time 0 to time 100 minutes for the Fort Calhoun and Calvert Cliffs natural circulation tests. Since the detail of the test data reported in the

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startup test reports varied from test to test, it was not possible to plot core ΔT versus time for all the tests. Descriptions of the test results for all the tests do indicate, however, that the measured core ΔT followed similar patterns for all the tests.

It is also significant to note that for each test, the maximum T_{HOT} recorded during natural circulation conditions was less than the T_{HOT} recorded immediately before the start of the test. Table 1 shows the initial T_{HOT} and the maximum T_{HOT} recorded during each of the natural circulation tests.

Table 2 lists the power/flow ratios recorded at various times during each of the natural circulation tests. The power levels were calculated either by using decay heat rates or by measuring boiloff from the steam generators. Flow rates were calculated utilizing loop transport time calculations or core ΔT calculations. Power is expressed as percentage of full power and flow rate is expressed as percentage of full forced circulation coolant flow rate.

Since the Connecticut Yankee core power history prior to the start of the natural circulation test is available, it is possible to calculate the expected natural circulation temperatures and flow rates assuming a loss of forced circulation flow at 100% reactor power. The heat flux and flow rates can be calculated utilizing the relationship that

$$Q^2 = K(\Delta T)^3$$

where Q = decay heat rate
 ΔT = core differential temperature

Assuming that T_{COLD} equals $T_{SECONDARY}$, T_{HOT} is calculated to be 588.8°F 1 minute after loss of all forced reactor coolant system flow from 100% power. Assuming a hot channel factor of 2.0, T_{HOT} (hot channel) at 1 minute after the loss of forced flow is calculated to be 629.6°F which is below the saturation temperature of the primary safety valve set pressure and is therefore acceptable. The peak heat flux at this point is 0.012 BTU/hr-ft²x10⁻⁶. The average natural circulation flow velocity is 0.1358 lbm/hr-ft²x10⁻⁶ based on these assumptions

Reference 1 addresses the critical heat flux in a heated bundle cooled by pressurized water. Figure C-19 (attached) from reference 1 presents a graph of critical heat flux versus average mass velocity for pressures about 2200 psia. As can be seen from examination of Figure C-19, the peak heat flux of 0.012 BTU/hr-ft²x10⁻⁶ remains well below the critical heat flux.

In summary, then, a number of plant events have occurred during which all forced reactor coolant system flow has been lost for significant lengths of time without any apparent adverse consequences. Additionally, a number of natural circulation tests have been conducted which indicated that the % power/% flow ratio during the tests was about 0.2. Based on these tests, calculations indicate that neither burnout nor loss of reactor coolant inventory due to lifting of primary safety valves would be expected to occur during natural circulation conditions following a loss of all forced reactor coolant system flow at 100% power.



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Attachments:

1. Table 1
2. Table 2
3. Reference
4. Figure 1
5. Figure C-19

cc: T. Novak

TABLE 1

Temperatures Recorded During Natural Circulation Tests

	<u>Initial T_{HOT}</u> <u>Prior to test</u>	<u>Maximum T_{HOT}</u> <u>Recorded during test</u>
Fort Calhoun	545 ⁰ F	543 ⁰ F
Calvert Cliffs	556 ⁰ F	546 ⁰ F
Maine Yankee	541 ⁰ F	528 ⁰ F
Connecticut Yankee	555 ⁰ F	551 ⁰ F

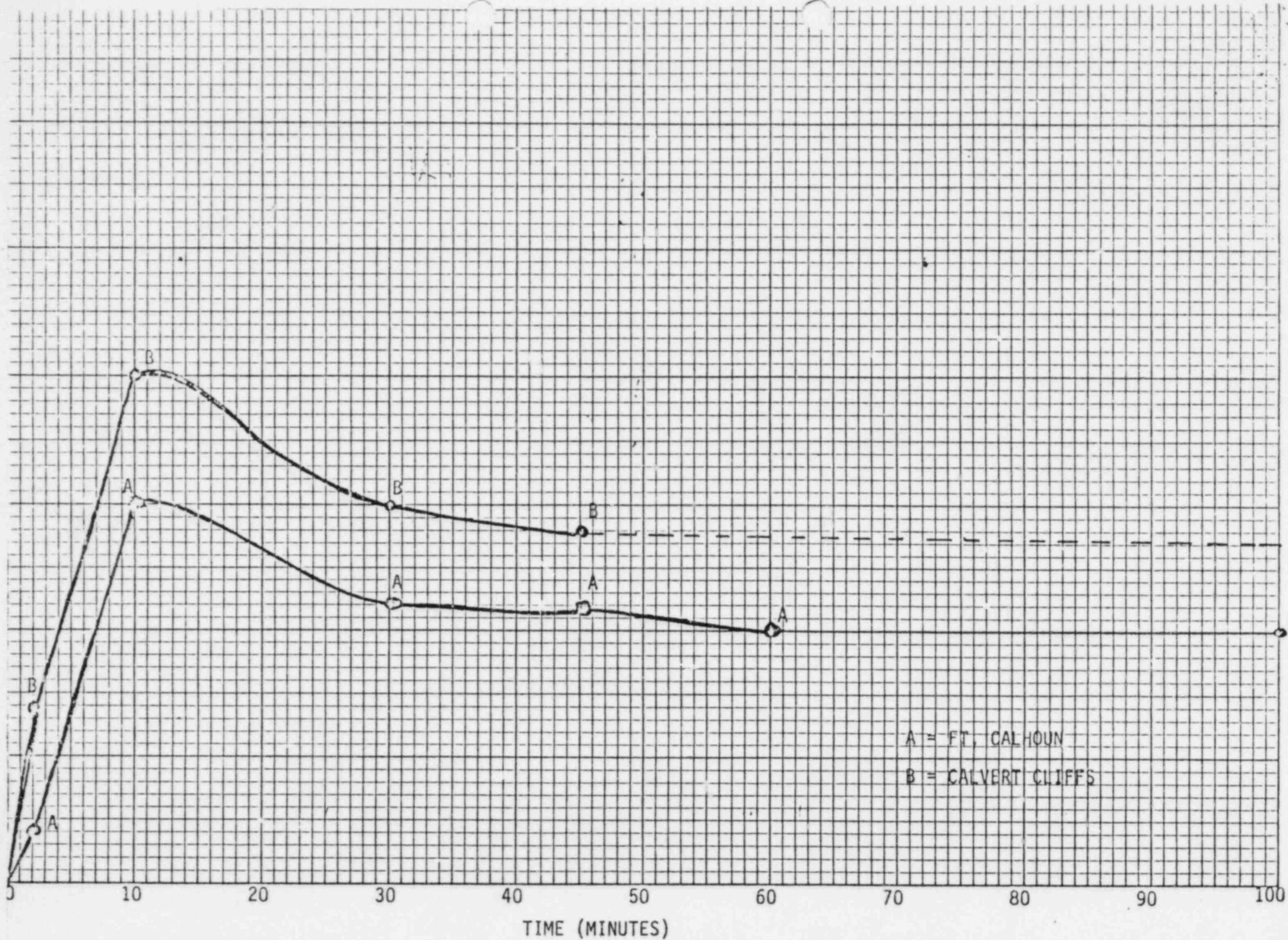
TABLE 2

% Power/% Flow During Natural Circulation Tests

	$\frac{\% \text{ Power}}{\% \text{ Flow}}$
Fort Calhoun	0.20
Calvert Cliffs	Unavailable
Maine Yankee	0.22
Connecticut Yankee	0.20

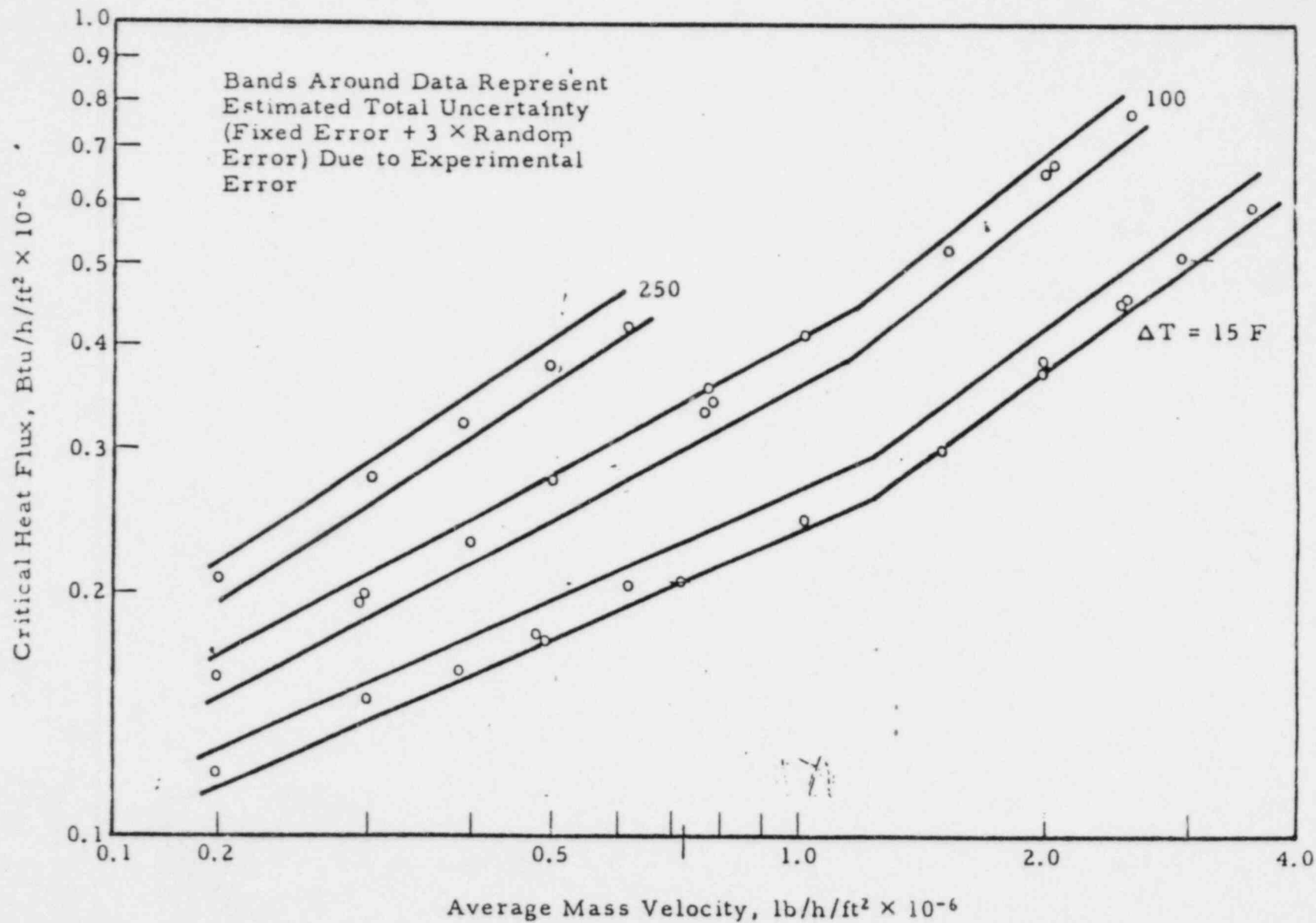
Reference

1. Topical Report BAW-10000A, Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, May 1976.



A = FT. CALHOUN
 B = CALVERT CLIFFS

Figure C-19. Experimental Errors for the Bundle



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Babcock & Wilcox

Source: Topical Report BAW-10000A (May 1976)