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FORWARDING ANALYSIS OF SMALL BREAKS IN THE REACTOR COOLANT PUMP DISCHARGE PIPING FOR THE B&W LOWERED LOOP 177 FA PLANTS, PURSUANT TO PROVISIONS OF THE NRC APRIL 1978 ORDER FOR MODIFICATION OF LIC FOR SUBJECT FACILITY"S, UNITS 1, 2, & 3.

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

May 15, 1978

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. Reid, Chief
Operating Reactors Branch #4

Reference: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

Pursuant to the provisions of the NRC's April 26, 1978 Order for Modification of License for Oconee Nuclear Station, Units 1, 2, and 3, the attached analysis of small breaks in the reactor coolant pump discharge piping is submitted. This analysis is the same as the generic 2568 MWt analysis, submitted by B&W on May 1, 1978 (letter of James H. Taylor to Robert L. Baer), and is applicable to Oconee Units 1, 2, and 3 with the exception of the required operator actions. For Oconee units the required operator actions are as follows:

1. Upon ESFAS signal check for flow through both HPI trains.

2. If there is no flow in one train,

   (a) open pump header cross-connect valve,

   (b) check HPI valve position and open, if closed.

The results of the generic analysis will be conservative for Oconee units since HPI flow from two pumps is assured by the operator action. The analysis indicates that the ECCS cooling performance calculated in accordance with the B&W Evaluation Model for operation of Oconee units at the rated core thermal power of 2568 MWt with operating procedures described in our letter of April 21, 1978, is wholly in conformance with the provisions of 10 CFR 50.46.

Very truly yours,

William O. Parker

William O. Parker, Jr.

PMA:scs
Attachment

A001 S/1

ANALYSIS OF SMALL BREAKS

IN THE

REACTOR COOLANT PUMP DISCHARGE PIPING

FOR THE

B&W LOWERED LOOP 177 FA PLANTS

May 1, 1978

1. Introduction

On April 14, 1978, B&W reported that previous small break analyses had not been based on the worst break location. This report indicated that the worst case break had now been determined to be at the reactor coolant pump discharge. A spectrum of small breaks has been examined for the B&W 177-FA lowered loop plants using the small break evaluation model described in BAW-10104, Rev 3, "B&W's ECCS Evaluation Model." These results show that it is necessary to use operator action during the early stages of the postulated accident, to effectively mitigate the accident consequences and meet the criteria of 10 CFR 50.46. Operator action is used to achieve sufficient and balanced flow through all four HPI injection lines. This report shows that operation up to at least 2568 MWt is possible within the criteria of 10 CFR 50.46 and Appendix K.

2. Evaluation

2.1. Method of Analysis

The analysis method used for this evaluation is that described in Chapter 5 of BAW-10104, Rev 3, "B&W's ECCS Evaluation Model." Specifically, the model, except for break size, break location, and core power, is the same as utilized in Appendix C of BAW-10103A, Rev 3, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS." The analysis uses the CRAFT2 code to develop the history of the reactor coolant system hydrodynamics. The CRAFT model uses 19 nodes to simulate the reactor coolant system, two nodes for the secondary system, and one node for the reactor building. A schematic diagram of the model is shown in Figure 1 along with the node descriptions. Control volumes (nodes) in and around the vessel are all connected by a pair of flow paths to permit counter-current flow. The break is assumed to be located at the bottom of the cold leg piping between the reactor coolant pump discharge and the reactor vessel. The Wilson, Grenda and Patterson average bubble rise model is used for all nodes. Within the core region, however, a multiplier of 2.38 is applied to the calculated bubble rise velocity.

Appendix F of BAW-10104 demonstrates that a multiplier of 2.38 in CRAFT2 gives a mixture height within ±2% of that predicted by FOAM. Thus, no FOAM analysis will be needed if the CRAFT2 mixture level remains above the core by 2% of the active length.

The following assumptions are made for conditions and system responses during the accident:

a. The reactor is operating at 102% of the steady-state power level of 2568 MWt.

b. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis. Bernoulli's equation was used for the subcooled portion of the transient, while Moody's correlation was used in the two-phase portion.

c. No offsite power is available.

d. The reactor trips on low pressure at 1900 psia.

e. The safety rods begin entering the core after a 0.5 second delay from the time the reactor trip signal is reached.

f. The RC pumps trip and coast down coincident with reactor trip.

g. One complete train of the emergency safeguards system fails to operate, leaving two CFTs and only one HPI and one LPI system available for pumped injection to mitigate the consequences of a cold leg break.

h. The auxiliary feedwater (FW) system is assumed to be available during the transient. Its main function is to remove heat from the upper half of the steam generator during the initial stages of the transient. When the secondary side of the steam generator becomes a source of heat to the primary system, the assumption of auxiliary FW maximizes the energy that must be relieved.

i. ESFAS signal error band is considered in the analysis to signal the actuation of the HPI system.

j. The peak linear heat generation rate in the hot pin is the maximum allowed by the Technical Specifications at the 10.5 ft level.

k. Operator action is taken to increase the HPI flows to the intact cold legs at 10 minutes following the ECCS initiation signal. This assumption is explained more fully below and in section 3.

As most of the breaks evaluated in this spectrum showed core uncovery, temperature calculations were necessary. Once core uncovery occurs a spatial swell distribution analysis is necessary to assure that only the core covered by mixture is included in the swell level. B&W uses the FOAM code. The code was utilized under the same assumptions as described above with the following additions:

1. The power shape shown in Figure 6 was used but implemented with a radial peaking factor of 1.0. This represents the average channel condition which is appropriate for use in swell level calculations.

2. Steam production due to heat from the primary metal, core and lower plenum flashing, was conservatively underpredicted. Although the CRAFT model accurately predicted these effects, full credit was not included in the FOAM simulation as a conservative computational convenience. This simulation, therefore, underpredicts both the swell level and the steaming rate. Consequently, more core uncovery and lower coolant flow are used in the heat-up evaluation.

The heat-up calculation was performed using the THETA code in the manner described in section 5 of BAW-10104. The following additional assumptions are utilized in the THETA evaluation:

1. The power shape of Figure 5 was used with a radial power factor of 1.8. This maximizes steam superheating and sets the peak local power at 10.5 ft at the technical specification LOCA limit.

2. Coolant flow and mixture level were taken directly from the FOAM calculations.

3. End of life pin pressures were used to conservatively predict the incidence of fuel pin rupture.

2.2. High Pressure Injection System Performance

The previous arrangement of the HPI system allowed for one pump to inject into the reactor coolant system (RCS) at two locations. As one injection point could be in the region of the break, 50% of the one HPI flow could fail to penetrate the reactor vessel. This flow would, therefore, not be available to provide core cooling. The proposed operator action, section 3, will provide four points of penetration of the RCS. Therefore, only 25% of the HPI flow would be lost.

Since the flow from one HPI pump will now be distributed to four injection points and to assure conservatism in allowing for injection line loss differences, this analysis assumes 30% of the HPI is injected into the broken cold leg. The implemented action starts at 5 minutes after an ECCS signal and is concluded 15 minutes after the signal. The resultant HPI flow can be conservatively represented as a linear ramp from 5 to 15 minutes. This ramp

was simulated in our present CRAFT code as a step function at 650 seconds (600 seconds for action, 50 seconds for ECCS signal). This is illustrated in Figure 7.

2.3. Break Spectrum and Results

All evaluations reported in this analysis assume the high pressure injection performance as described in section 2.2. Breaks of 0.04, 0.07, 0.1, 0.13, 0.15, and 0.17 ft^2 were evaluated. The evaluation of a 0.5 ft^2 break was reported in BAW-10103A, Rev 3, and shows complete core covery at all times and thus no temperature excursion. The 0.5 ft^2 break results are independent of HPI flow and remain valid.

Figure 2 shows the RCS pressure transient for each break. As shown, each accident initiates CFT flow within 2000 seconds except for the 0.04 ft^2 break.

Figure 3 shows (CRAFT) mixture height as a function of time for each break of the spectrum. Various uncovery levels and times are observed but all trends are consistent throughout the spectrum.

The 0.04 ft^2 break achieves a match up of effective ECCS (the HPI injected into the intact cold legs) with the core decay heat and the RCS metal heat at 2500 seconds. After 2500 seconds the mixture level will rise in the core due to excess HPI injection. As the 0.04 ft^2 break has a level of 14 feet at this time the core never uncovers and no temperature excursion occurs. For breaks smaller than 0.04, the match up will occur at approximately the same time and the core mixture levels will drop slower; thus, for all smaller breaks the core will remain covered.

Figure 4 shows the time duration of uncovery for three core elevations as a function of break size. These results are from CRAFT. As can be seen, the maximum degree of uncovery and the maximum time of uncovery occur for the 0.13 ft^2 break and is the worst case break in this analysis. This break can thus be identified as the worst case for operation up to 2568 MWt. The 0.07, 0.10, 0.13, 0.15, and 0.17 ft^2 breaks were analyzed for temperature response. The results are shown in Figure 5 and are well within the criteria of 10 CFR 50.46. They provide positive assurance that all breaks of the spectrum are within acceptance criteria.

The evaluation of break sizes of 0.2 and 0.3 ft^2 was reported in the report of April 25, 1978 (J.H. Taylor to R.L. Baer) for a power level of 2772 MWt and is thus conservative for the analysis herein.

Local metal-water reaction is shown in Table 1. The highest value is 1.72% for the 0.13 ft^2 break. This value is well below the local oxidation limit for the large breaks utilized in BAW-10103 for the whole-core metal-water reaction calculation. Thus, the whole-core metal-water reaction results given in section 8 of BAW-10103 is conservative for small breaks. The degree of clad damage is bounded by the large break results which produce higher clad temperatures. Thus, all criteria of 10 CFR 50.46 are met. This analysis is conservative for many reasons as detailed in the writeup and meets all evaluation criteria. This analysis shows that all 177 lowered loop plants meet the criteria of 10 CFR 50.46 if operated at or below 2568 MWt power and in conjunction with the specified operator action.

3. Operator Action

The ECCS analysis used as a basis for this report assumes that the operative HPI train (one train is lost due to a single active failure) provides emergency core cooling water to the RC loop containing the break. It is conservatively assumed that the break is on the lower portion of the reactor coolant pump discharge piping resulting in the total loss to the system of 50% of the available HPI flow. Acceptable mitigation of the accident requires more than the 50 % of this flow from one HPI pump. If, following the LOCA, it is assumed that one train of HPI does not start it is necessary to take operator action to achieve a flow split wherein no more than 30% of the remaining pump's flow goes into the cold leg containing the break. The following is a description of the action required for a typical plant.

1. Upon ESFAS signal check for flow through both HPI trains.
2. If no flow in one train:
   - open pump header cross-connect valves
   - check HPI valve position and open if closed
3. Secure flow through normal makeup line if flow is indicated
4. Throttle HPI valves as required to balance flow and meet run out limits

TABLE 1

PEAK CLADDING TEMPERATURE VERSUS BREAK SIZE
(All at 2568 MWt)

Break Size (Ft^2)	Peak Cladding Temperature (^oF)	Maximum Local MW Reaction (%)	Time of Peak Temperature (sec.)
0.04	Core Stays Covered With No Temperature Excursion		
0.07	1320	.73%	1600
0.1	1440	1.68%	1080
0.13	1551	1.72%	820
0.15	1455	1.67%	740
0.17	1248	0.72%	650

The above actions initiated at five minutes and completed within 15 minutes subsequent to the ESFAS actuation ensures adequate HPI flow for accident mitigation. In the analysis, credit is taken for the HPI flow as the HPI injection valves are opened. Figure 7 shows the calculated HPI flow for a typical plant as a function of time for a 10 minute valve opening. As shown in Figure 7, the majority of the HPI pump capacity would be delivered with a partial valve opening. For the small break analysis, a linear flow versus valve position response was simulated by a step function increase, 10 minutes after ESFAS actuation.

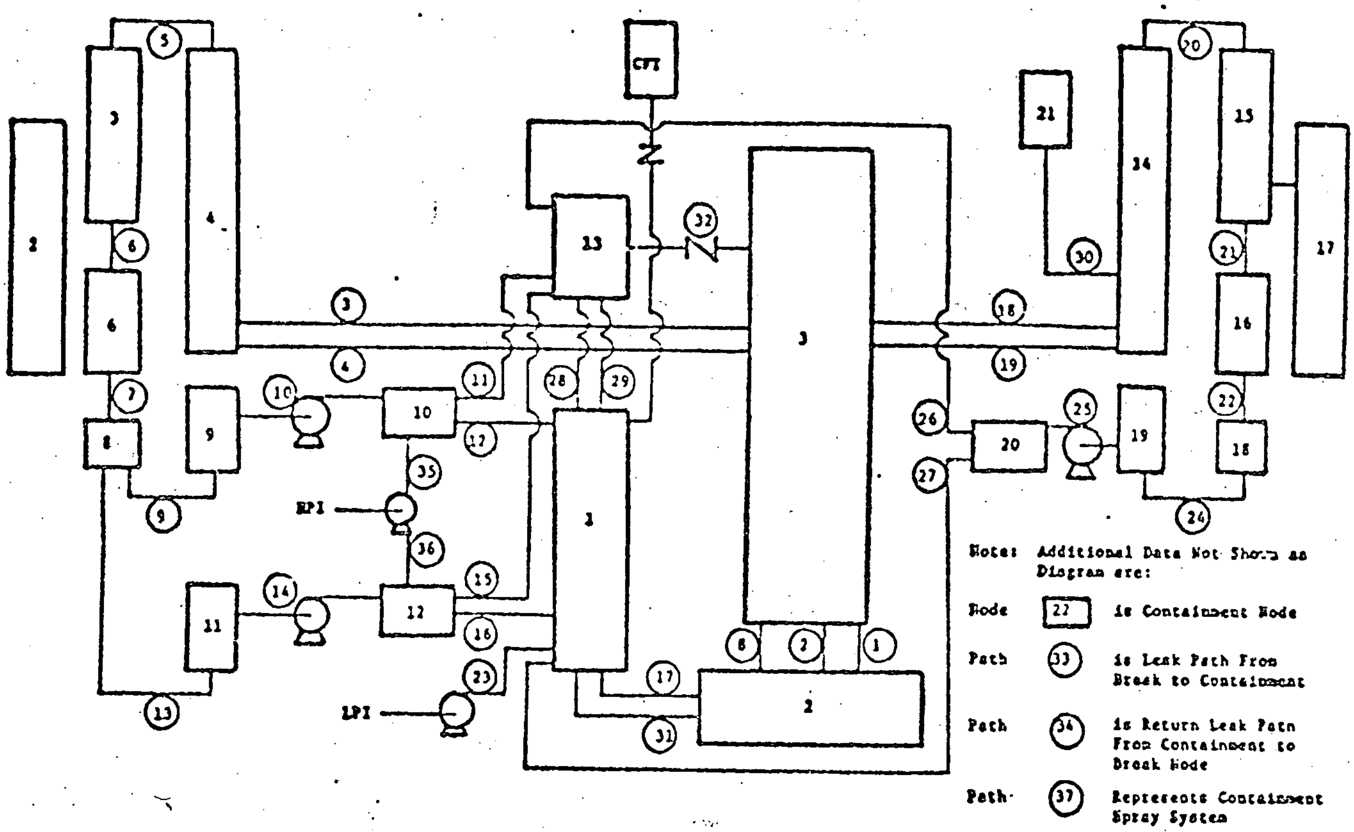
4. Evaluation of Other B&W Supplied Plants

   a. Davis-Besse — The DB-1, 2 and 3 Plants have been analyzed for a spectrum of small breaks at the RCP discharge in accordance with an approved small break evaluation model. This analysis is reported in BAW-10075A, Rev 1, March 1976. In addition, the Davis-Besse 1, 2, and 3 units have a split high pressure injection and makeup system design. The Davis-Besse HPI pumps, therefore, have considerably higher capacity at the system pressures experienced.

   b. 205 and 145 FA — These plants have been analyzed for a spectrum of small breaks at the RCP discharge in accordance with an approved small break evaluation model. These analyses are reported in BAW-10074A, Rev 1, and BAW-10062A, Rev 1, March 1976. In addition, the 205 and 145 FA HPI systems contain cross connects between the two HPI trains downstream of the HPI injection valves. These cross connects effectively achieve the same flow split as the operator action assumed in the current 177 FA lowered loop analysis and the flow split is achieved when the HPI pump is started.

   c. All B&W supplied plants except the 177 FA lowered loop plants have raised loops and thus do not trap a large volume of coolant in the cold leg. The raised loop design allows this coolant to drain into the core for core covering and cooling by boiloff.

Figur[illegible]-1. CRAFT2 Noding Diagram [illegible] Small Break



Node No.	Identification
1	Downcomer
2	Lower Plenum
3	Core, Core Bypass, Upper Plenum, Upper Head
4,14	Hot Leg Piping
5,15	Steam Generator Upper Head, SG Tubes (Upper Half)
6,16	SG Tubes (Lower Half)
8,18	SG Lower Head
9,11,19	Cold Leg Piping (Pump Suction)
10,12,20	Cold Leg Piping (Pump Discharge)
13	Upper Downcomer (Above the ℄ of Nozzle Belt)
21	Pressurizer
22	Containment

Path No.	Identification
1,2	Core
3,4,18,19	Hot Leg Piping
5,20	Hot Leg, Upper
6,21	SG Tubes
7,22	SG Lower Head
8	Core Bypass
9,13,24	Cold Leg Piping
10,14,25	Pumps
11,12,15,16,26,27	Cold Leg Piping
17,31	Downcomer
23	LPI
28,29	Upper Downcomer
30	Pressurizer
32	Vent Valve
33,34	Leak & Return Path
35,36	HPI
37	Containment Sprays

Figure 2 PRESSURE

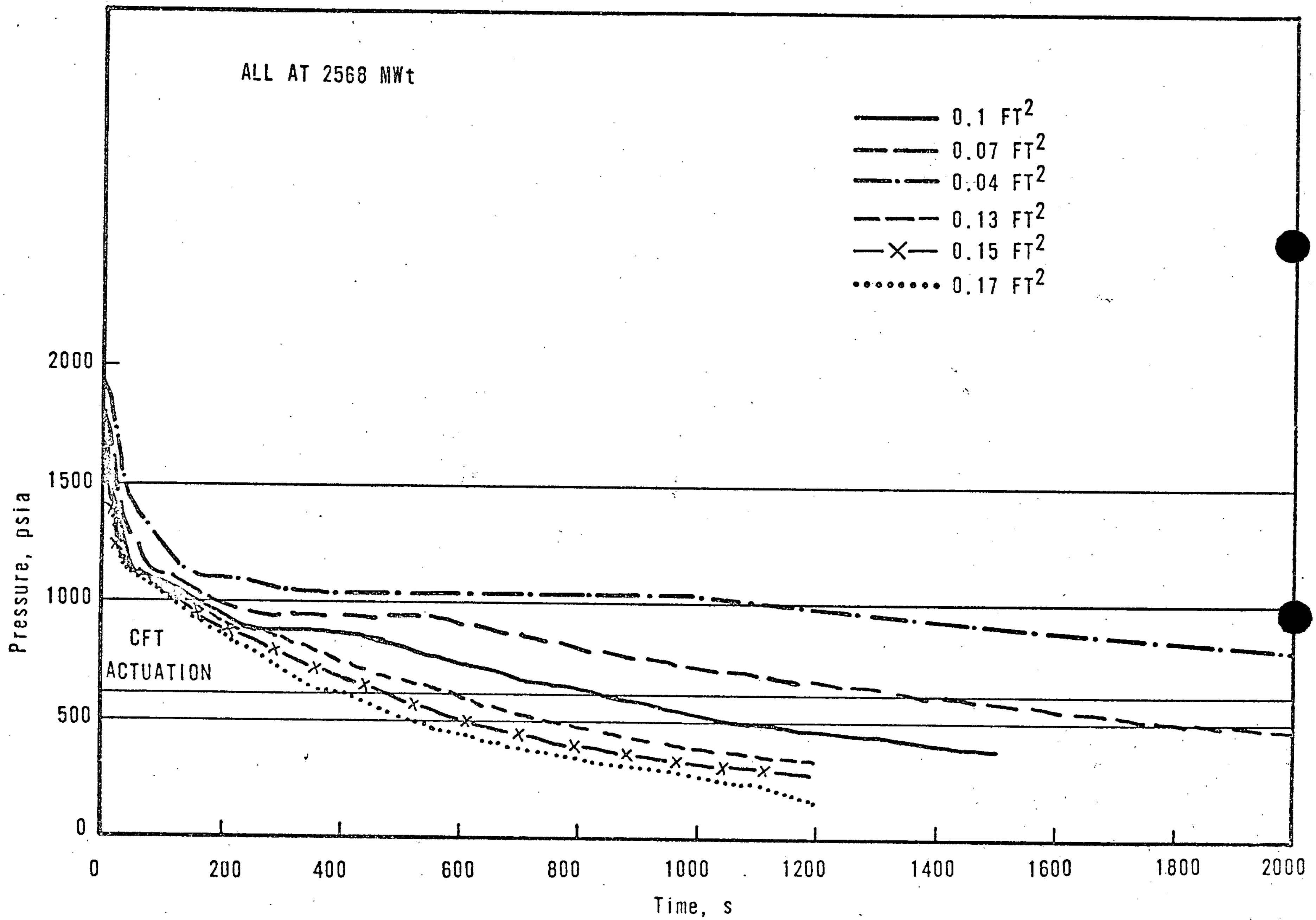


Figure 3 CORE MIXTURE HEIGHT
(CRAFT)

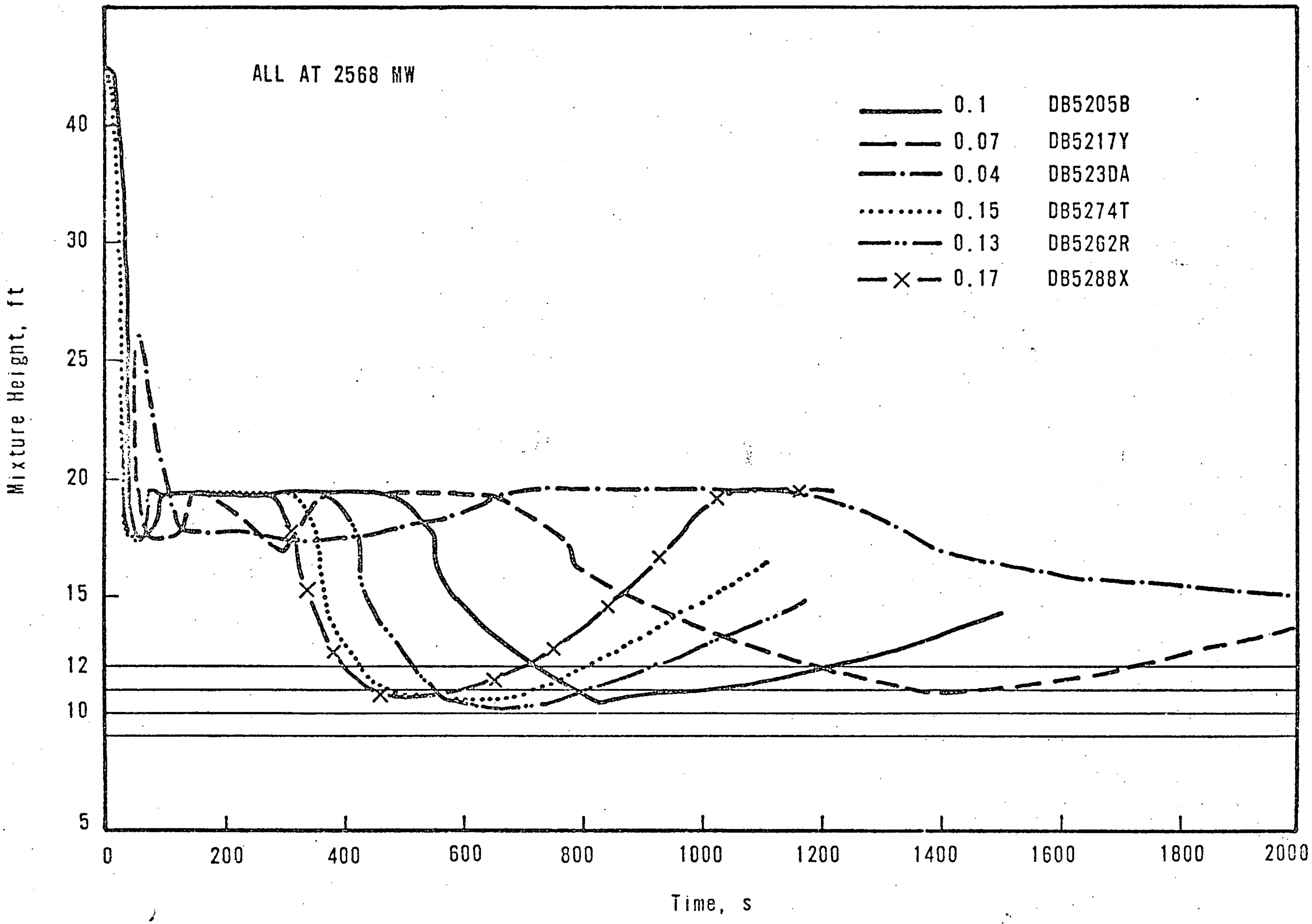


Figure 4 DURATION OF UNCOVERY FOR THREE CORE LEVELS
(11.5 FT, 11.0 FT, 10.5 FT)
(NOTE 10.0 FT DOES NOT UNCOVER)

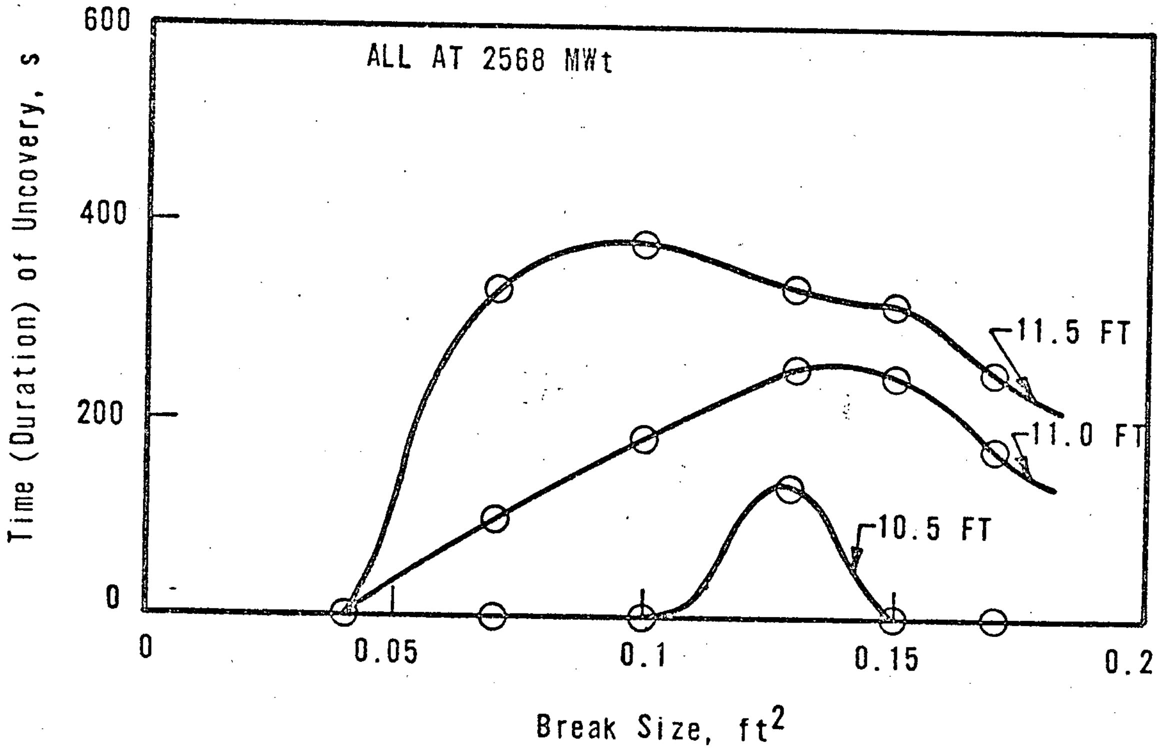


Figure 5 PEAK CLADDING TEMPERATURE PUMP DISCHARGE SMALL BREAKS

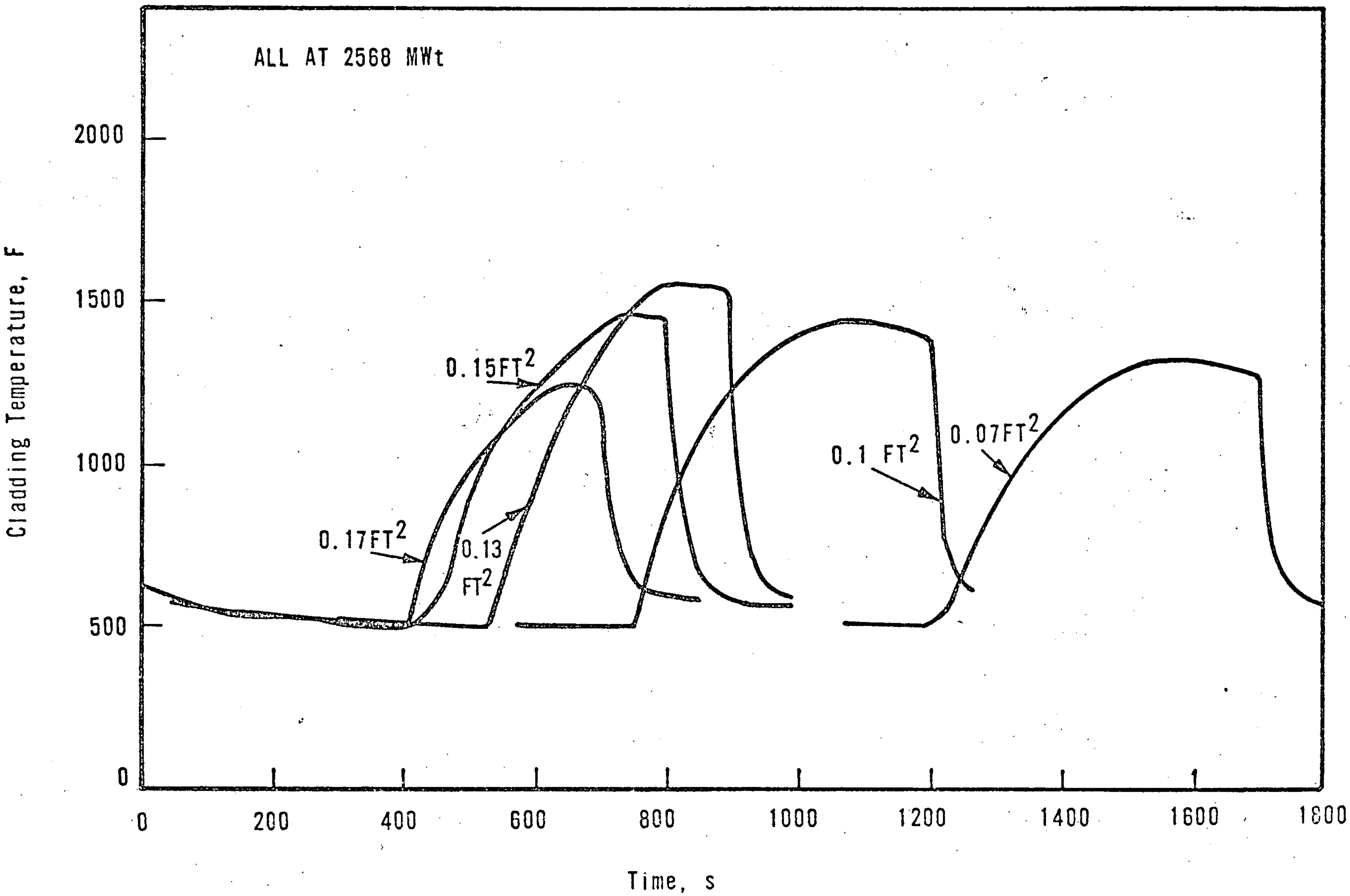


FIGURE 6. AXIAL POWER SHAPE

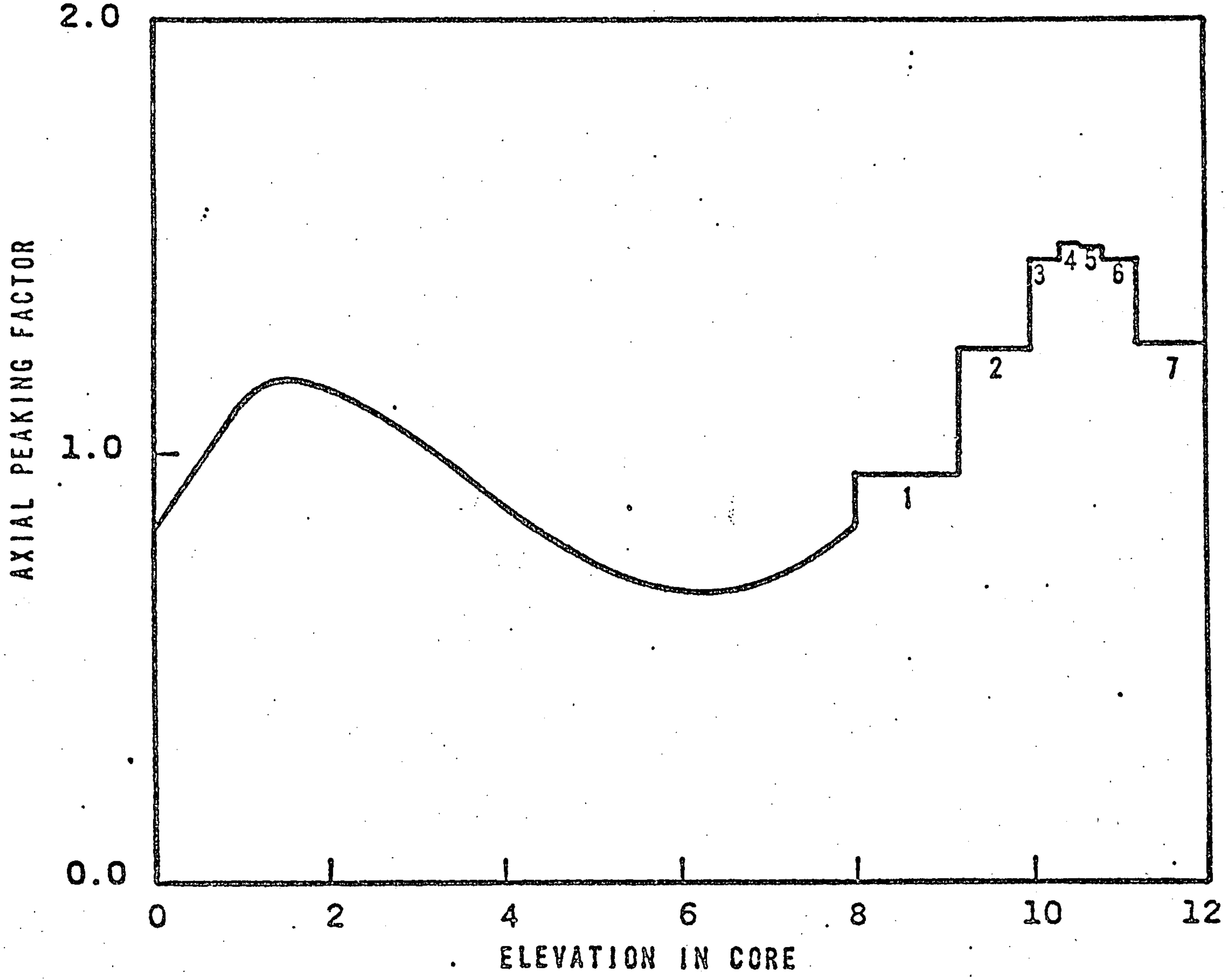


FIGURE 7. FLOW VS VALVE POSITION
(Based on Actual C_V VS Valve Position for Oconee HPI Valves)

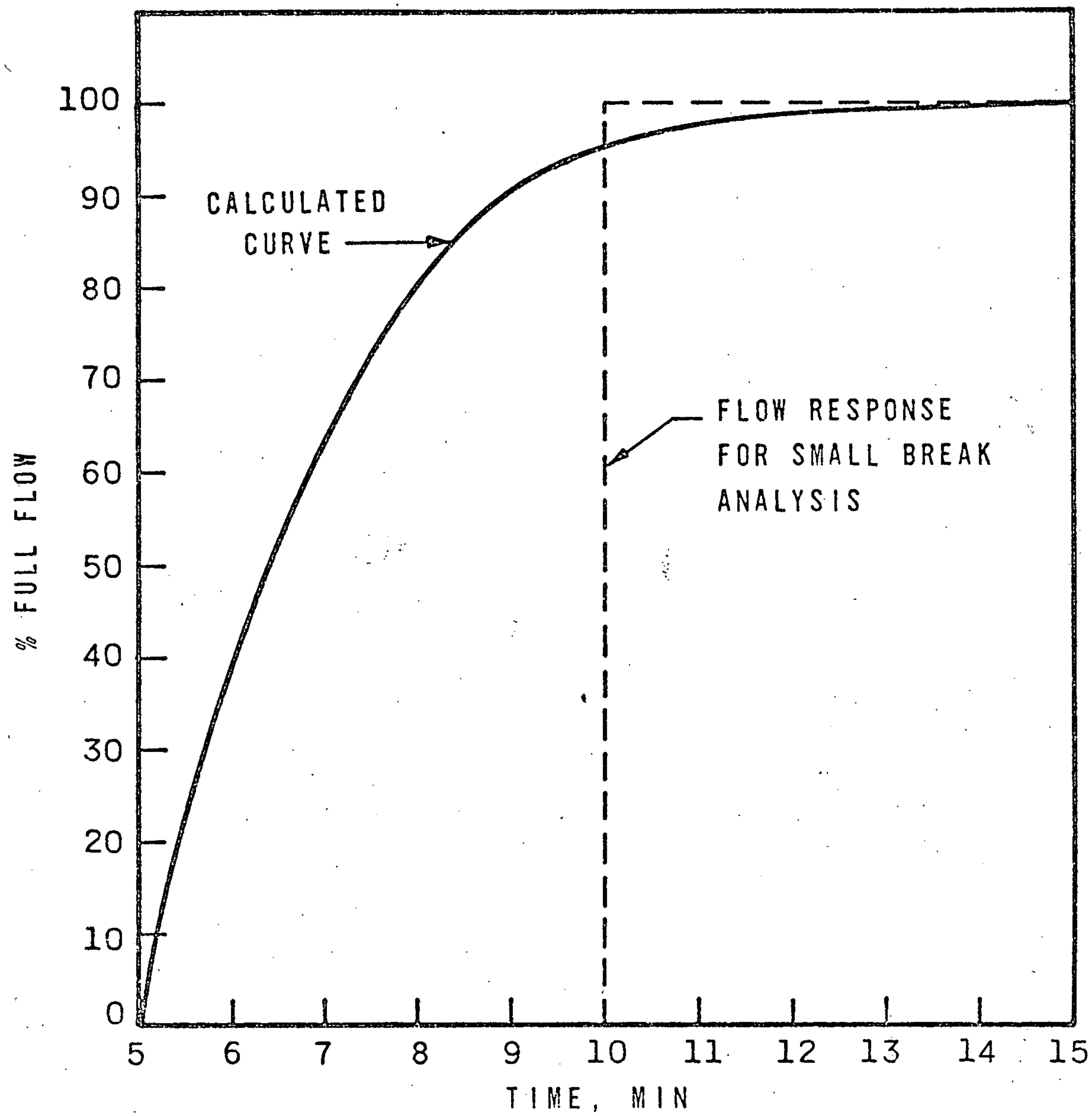


Figure 8 PEAK CLADDING TEMPERATURE
VS
BREAK SIZE