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U PROP INFO		1	50-269 50-270 50-287

DESCRIPTION:
Ltr advising of the installation of flow restrictors in each core flooding nozzles entering the reactor vessel....trans the following:

ACKNOWLEDGED DO NOT REMOVE

PLANT NAMES: Oconee Units 1-2-3

ENCLOSURES:
Ltr 1-15-73 fm B&W to Giambusso w/attached....
Babcock & Wilcox Report..including...
Atchmt 1 - Description & mechanical design of the device.
Atchmt 2 - Analysis of the core flooding line break with the flow restrictor in place.
Atchmt 3 - Shows effect of the device on the operation of ECCS for both large & small breaks.

1-16-73 fod

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A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

January 12, 1973

RECEIVED
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U.S. ATOMIC ENERGY COMMISSION
Regulatory Mail Section

Mr. Angelo Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

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

Dear Mr. Giambusso:

Duke Power Company is installing flow restrictors in each of the core flooding nozzles entering the reactor vessel at Oconee Nuclear Station. The purpose of these restrictors is to limit the magnitude of the blowdown and retain more water in the reactor vessel during a core flood line break loss-of-coolant accident. As discussed with members of your staff in Bethesda, Maryland on January 4, 1973, we will file a revision to the Final Safety Analysis Report by January 29, 1973 to document necessary information for your review on the core flood line restrictors. The purpose of this letter is to provide you with the substance of the information that will be filed on January 29 at an earlier date for your review.

The information, which has been transmitted to you under separate cover, provides a description of the mechanical design of the flow restrictor, provides an analysis of the core flood line break with the flow restrictor in place, and shows the effect of the restrictor on the operation of the emergency core cooling system for both large and small breaks. These results demonstrate that all of the AEC interim acceptance criteria for emergency core cooling systems are satisfied.

Please advise if you have additional questions.

Sincerely,

A. C. Thies

A. C. Thies

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Babcock & Wilcox

Power Generation Division

P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (703) 384-5111

January 15, 1973

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Received w/Ltr Dated 1-18-73

Mr. A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing
U.S. Atomic Energy Commission
Washington, D. C. 20545

Re: Oconee Nuclear Station
Docket No. 50-269

Dear Mr. Giambusso:

Attached are responses to several questions received over the last few months in regard to the Core Flooding Tank Line Break and, in particular, the questions received by telecon and discussed at our meeting of January 4, 1973 in Bethesda, Maryland.

Duke Power Company is installing a flow restrictor in the core flooding nozzle to restrict the magnitude of the blowdown and retain more water in the reactor vessel during the accident.

Attachment 1 provides a description and mechanical design of the device.

Attachment 2 provides an analysis of the Core Flooding Line Break with the flow restrictor in place.

Attachment 3 shows the effect of the device on the operation of the Emergency Core Cooling System for both large and small breaks.

The results for the Core Flooding Line Break for the Oconee reactor demonstrate that all of the AEC Interim Acceptance Criteria for Emergency Core Cooling Systems are satisfied.

Very truly yours,

BABCOCK & WILCOX COMPANY
Nuclear Power Generation

James F. Mallay

James F. Mallay
Manager, Licensing

JFM/db
Attach.

ATTACHMENT I

CORE FLOODING NOZZLE

MECHANICAL DESIGN

LIST OF FIGURES

The core flooding nozzle modification for the Oconee I nuclear plant is basically a variable diameter thermal sleeve which reduces the nozzle opening from 0.72 ft^2 to 0.44 ft^2. This section reviews the design, fabrication, and field installation of the sleeve.

The existing core flooding nozzle and thermal sleeve are shown in Figure 1. The existing sleeve will be removed to enable installation of the modified sleeve.

The restrictor is fabricated in two stages as shown on Figure 2. Because of schedule restraints related to installing this modification at Oconee, it was necessary to use available material. Thus, Type 304 stainless steel pipe was used and weld overlay was deposited to meet the required 9" I.D. The pipe and weld overlay were U.T. examined before machining with P.T. examination after final machining. These examinations were performed in accordance with ASME Section III.

Removal of the existing sleeve is accomplished by grinding the weld buttons which hold it in place, and performing a P.T. examination on the ground areas (Figure 3). The nozzle is then ready for installing the restrictor. This will be accomplished by welding and machining the weld buttons and ring as shown in Figure 4. The restirctor is inserted and a full penetration weld with permanent backing ring is made in accordance with ASME Section III (Figure 5). A progressive P.T. is performed to insure a quality weld. The weld buttons center the restrictor. A drain hole is drilled at the bottom of the weld to allow a small flow of water behind the restrictor to prevent crud buildup.

The restrictor and attachment weld (Figure 5) are evaluated in accordance with ASME Section III. The significant transients which affect the restrictor and weld are reactor coolant system heatup and cooldown including the core flooding system periodic test transient and decay heat removal initiation. All transients are considered as normal operating conditions and are considered in determining thermal stresses and the fatigue usage factor. The fatigue analysis includes a strength reduction factor of two on the weld per ASME Section III. The weld has also been designed to withstand the faulted condition where a differential pressure of up to 2250 psi may occur because of a core flooding line LOCA. A dynamic magnification factor of two was applied to the pressure to account for instantaneous application. Based on these criteria, the average shear stress in the weld yields a safety margin of 1.4. These assumptions and safety margin are sufficient to insure the structural integrity of the nozzle, restrictor, and weld for all operating and faulted conditions.

During the core flooding transient, the maximum Δp across the nozzle is expected to be approximately 200 psi. This is a factor of greater than 20 less than the design loading assumptions. Therefore, it is not considered credible that the restrictor retaining weld would fail during core flooding tank discharge.

During operation of the decay heat system, the Δp loads on the restrictor are insignificant.

FIGURE 1 EXISTING CORE FLOODING NOZZLE SLEEVE

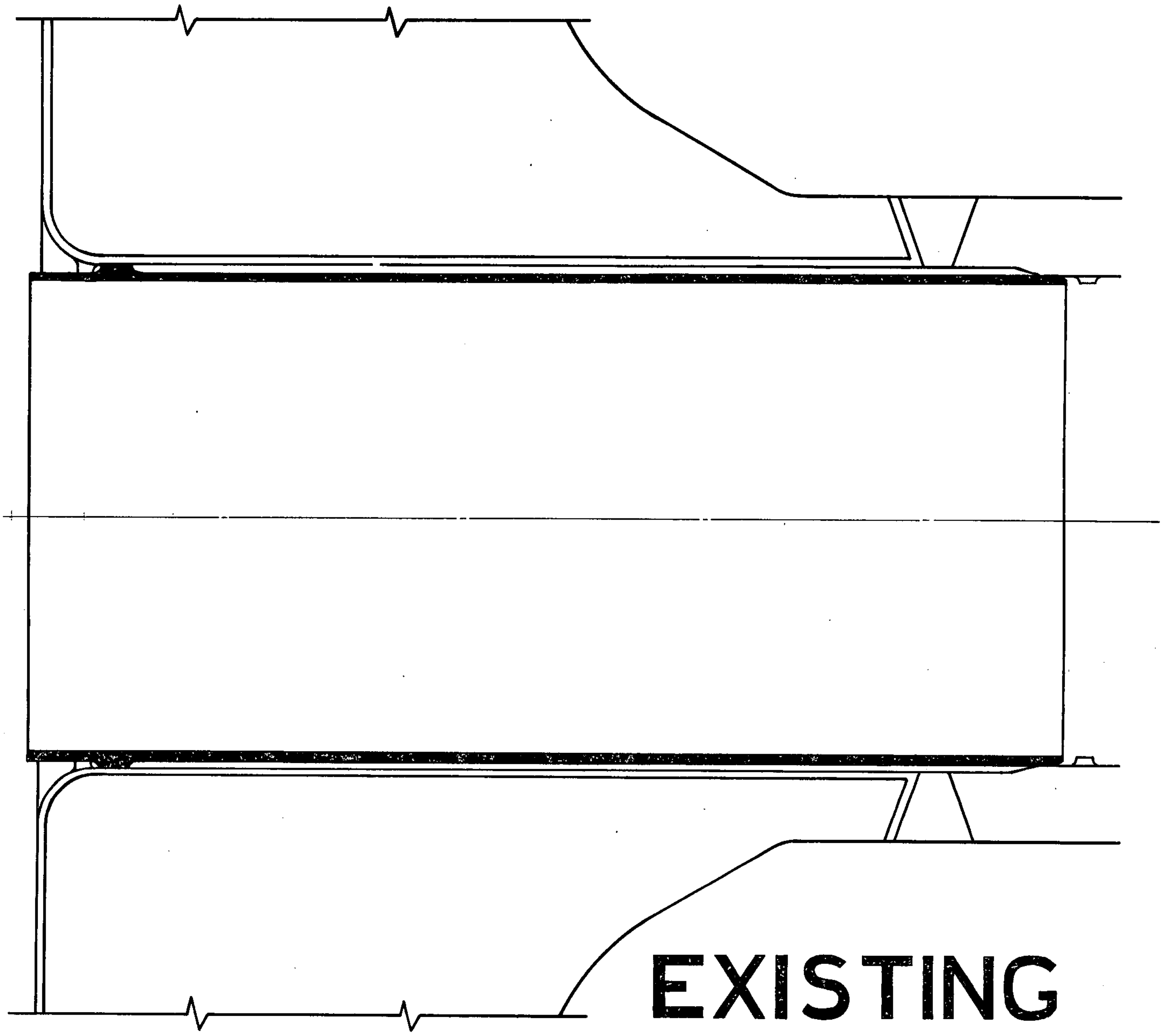
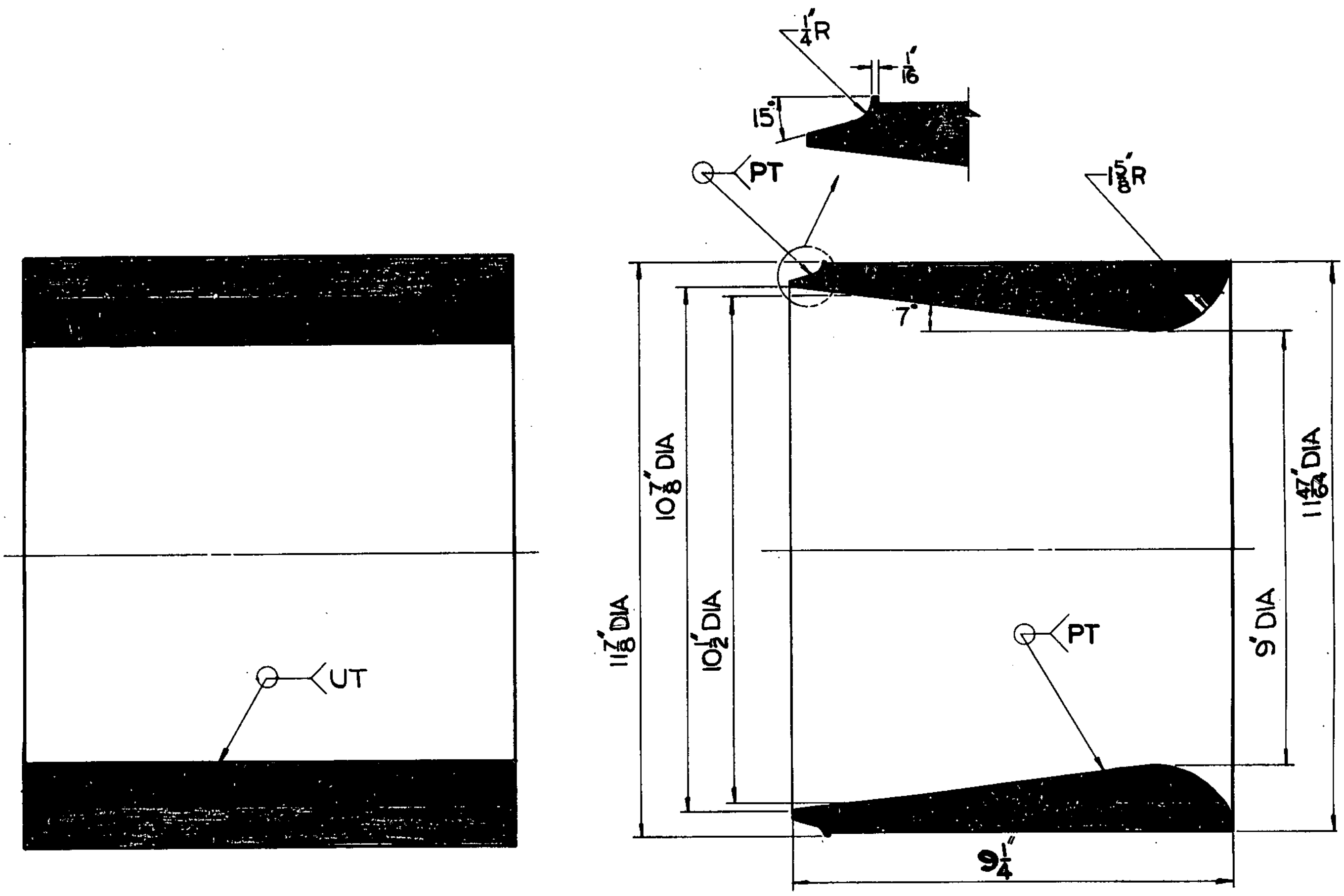


FIGURE 2 OCONEE I CORE FLOODING NOZZLE INSERT



STAGE 1 STAGE 2

FIGURE 3 CORE FLOODING NOZZLE AFTER REMOVING EXISTING SLEEVE

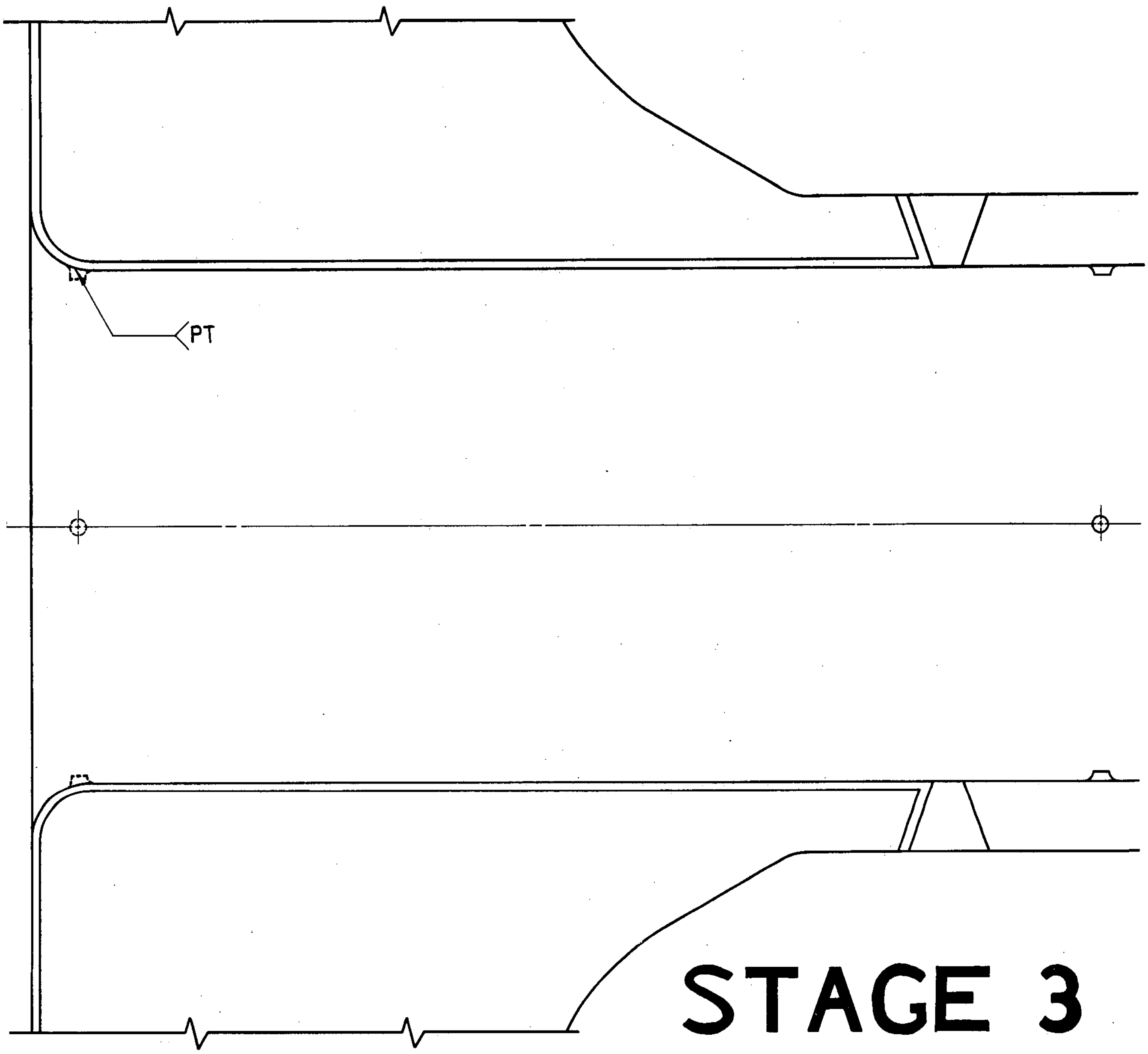


FIGURE 4 CORE FLOODING NOZZLE WELD PREPARATION PRIOR TO INSERTING MODIFIED INSERT

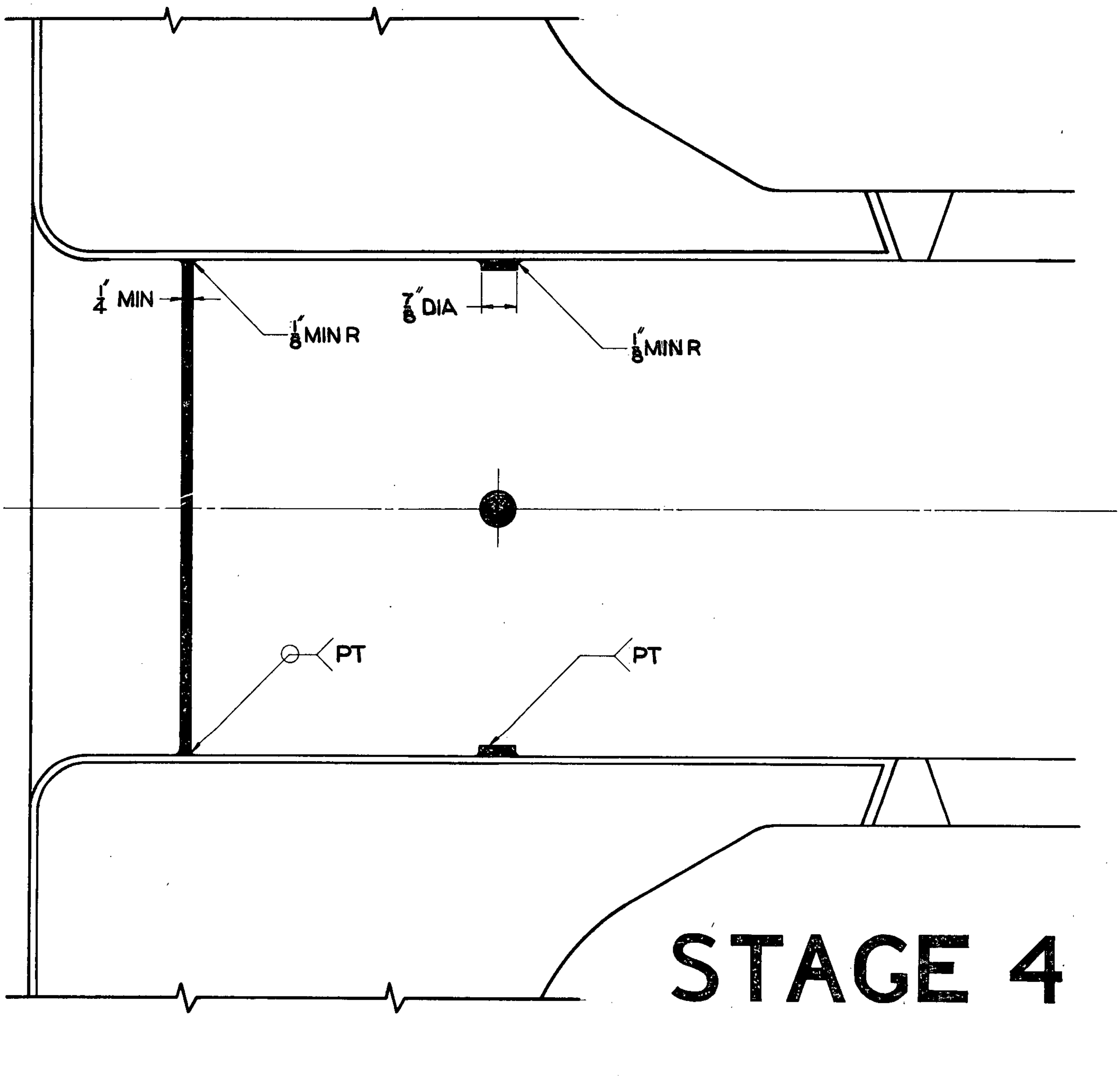
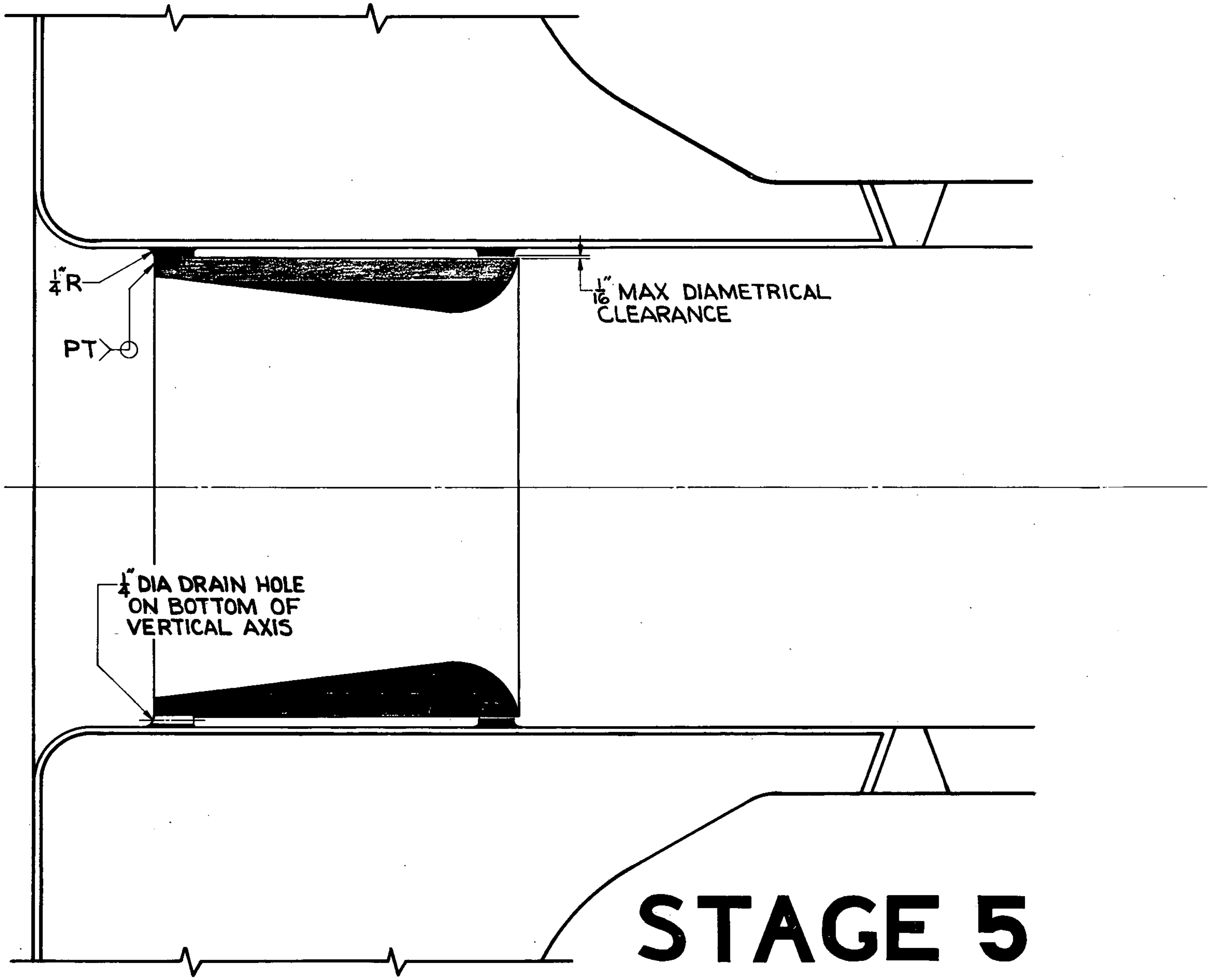


FIGURE 5 CORE FLOODING NOZZLE INSERT INSTALLED IN CORE FLOODING NOZZLE



ATTACHMENT II

ANALYSIS OF A CORE FLOODING LINE BREAK
LOSS OF COOLANT ACCIDENT FOR THE OCONEE 1
REACTOR WITH INSERT IN CFT NOZZLE

LIST OF FIGURES

LIST OF FIGURES

(Continued)

1. INTRODUCTION

The scope of this analysis is a guillotine break of the core flood tank line between the reactor vessel nozzle and the first check valve. Flow out of the reactor vessel is limited to an effective area of 0.44 ft^2 due to a flow limiting insert.

Since the leak area is less than 0.5 ft^2, the B&W small leak evaluation model, BAW-10052[1], "Multinode Analysis of Small Breaks for B&W's 2568-MWt Nuclear Plants", is used. Consideration is given to three different axial power distributions. The following assumptions are made for conditions and system responses during the accident:

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

2. A single failure is assumed in addition to the CFL break. The worst single failure results in an injection flow from only one high pressure injection pump and the second core flooding tank.

3. The leak occurs instantaneously, and a discharge coefficient of 1.0 is used for the entire analysis.

4. The reactor trips on low pressure at 2050 psig.

2. SUMMARY AND CONCLUSIONS

The maximum cladding temperature for this analysis is a function of the axial power shape. For the three axial power shapes analyzed, the maximum cladding temperature is 1172 F. A peak temperature of this magnitude or less presents no potential for either metal-water reaction or cladding swelling; therefore, the core geometry remains unchanged and amenable to cooling. This analysis indicates that adequate core cooling is maintained. The operator may initiate additional injection to refill the vessel in about one-half hour. Even without this additional long term cooling established, however, all conditions of the AEC Interim Acceptance Criteria are met.

The peak temperatures for the three different axial power shapes analyzed in this report are as follows:

Elevation of Power Peak from Bottom of Core ft	Elevation of Peak Cladding Temperature from Bottom of Core ft	Peak Cladding Temperature F
5.5	5.5	663
7.8	11.4	666
10.6	11.4	1172

3. METHOD OF ANALYSIS

The method of analysis used to determine the cladding temperature response is the same as described in BAW-10052. This is consistent with the interim policy statement because, although the CFT line cross-sectional area is 0.72 ft^2, the flow from the reactor vessel is limited in the vessel nozzle by an insert which has a cross-sectional area of 0.44 ft^2.

The basic assumptions of the CRAFT[2] model, used for core hydrodynamics, are the same as those applied to the small breaks except that the noding scheme is somewhat different due to the nature of the break. The noding scheme and legend are shown in Figure 3-1.

A THETA-1B[3] model is used to determine the heat transfer in the flow controlled region. The 10 axial region model is necessary to describe in sufficient detail the three axial power shapes that are analyzed. The Quench model, as described in BAW-10052, is used when the heat transfer regime is not flow controlled.

When the flow in the core has decayed sufficiently, the core is in a relatively quiescent condition in which the lower portion is covered by a two-phase mixture, which is cooled by pool boiling, and the upper portion is steam cooled at a flow rate consistent with the boil-off rate in the lower portion This steam cooling results in reduced heat transfer coefficients and significant temperature transients. Therefore, it is very important to properly and conservatively determine the height of the two-phase mixture. Steam production as calculated by CRAFT and the Redfield-Murphy bubble rise model[4] is used to determine the two-phase mixture height. Conservatism is applied by limiting the void fraction in the inner vessel mixture to a maximum value of 0.5 for thermal analysis. The void fraction as calculated by CRAFT results in higher mixture levels in the inner vessel. By limiting the void fraction to 0.5, more of the core is uncovered, which results in less steam flow and more superheating in the upper region.

The CRAFT model was examined to determine its sensitivity to various noding schemes. Figure 3-2 shows the results of three noding approaches in terms of inner vessel liquid volume and confirms our conservatism in using a two node inner vessel model because it eliminates the "pancaking"effect.

4. RESULTS OF ANALYSIS

The CRAFT run was made with a single core flow path and is applicable to any of the three axial power shapes examined. Figures 4-1 and 4-2 show the core pressure and power history for this accident. Figure 4-3 depicts the inner vessel fluid volumes as used for the heat-up calculations. The vent valves are located above the top of the active region; therefore, the vent valve flow shown in Figure 4-4 will appear to be inconsistent with the mixture height shown in Figure 4-3. In section 3, it was explained that a conservative limit was placed on the core void fraction. This effect is visible here because CRAFT shows a mixture height in the area of the vent valves while the mixture height which is used in the thermal analysis is much lower. Mixture flowing out of the vent valves is conservative because it removes liquid water from the core region.

The leak flow as a function of time is shown in Figure 4-5. The core and downcomer water heights are shown in Figure 4-6, while Figure 4-7 shows that calculated fluid velocities between the core and downcomer are not of sufficient magnitude to consider entrainment of the CFT water.

The 10 axial region THETA model was used until 300 seconds for the three axial profiles shown in Figures 4-8, 4-18, and 4-28. The Quench code was then used to analyze the cladding thermal response for the remainder of the transient.

The center-peaking power shape used in analyzing Case 1 is shown in Figure 4-8. The upper portion of the core is uncovered during the transient; but the low power obtained in this region does not result in significant cladding temperature increases. Therefore, the center region of the core, axial level 5, remains the hottest portion of the core with a peak cladding temperature of 663 F. Figures 4-9 through 4-11 show data related to axial level 5 and Figures 4-12 through 4-14 and 4-15 through 4-17 are relevant to axial levels 9 and 10 respectively.

Outlet power peaks are important to this analysis because higher power regions will be uncovered and the amount of superheat in the cooling steam will be higher. Case 2, Figure 4-18, is typical of an outlet peak experienced in the core. The maximum power peak occurs in axial level 7. It is never uncovered by the two-phase mixture, and the cladding temperature remains low. Figures 4-19 through 4-21 provide information on this level. Axial level 9 is uncovered for only a short period of time between 740 and 770 seconds and does not undergo a large cladding temperature rise. Figures 4-22 through 4-24 apply to this level. The highest peak cladding temperature is achieved at level 10 which is uncovered from approximately 1300 seconds to 2000 seconds and reaches a peak cladding temperature of 666 F. Information for this level is shown in Figures 4-25 through 4-27. The cladding temperature is declining slightly at the end of the analysis.

To see the effect of an outlet peak, Case 3 which is shown on Figure 4-28 was chosen for examination. This power shape represents one of the most adverse tilts toward the exit of the core. This shape is used for design purposes and while it is not an expected or normal operating condition, this shape is allowable for operation for the last few days of core life. In the heatup calculation, the power peak occurs at axial level 9. The temperature at the location of peak power, level 9, does rise as it is uncovered, but the recovering of the level keeps this from being the worst location. Figures 4-29 through 4-31 provide information at this location. The peak cladding temperature, 1172 F, occurs at level 10 because of the extended steam cooling from 1300 seconds on. The cladding temperature, Figure 4-34, is high because of the low heat transfer coefficient, Figure 4-33, and the degree of superheat, Figure 4-32. Both effects are caused by the shift of power to the upper regions of the core.

5. HIGH PRESSURE INJECTION FLOW

The analysis of the core flooding line break shows that the core can be cooled using only one core flooding tank and one high pressure injection (HPI) pump. The HPI flow used in the analysis was based on test results from Oconee 1. A least-squares regression analysis of the data was performed which showed a flow of 352 gpm and 353 gpm in each of two strings at a RCS pressure of 1500 psig. Considering an instrumentation error of ± 1% and the relative error in the regression fit, the flow at 1500 psig was reduced to 340 gpm for use in the analysis. Similarly at 600 psia, the least-squares fit resulted in a flow of 457 gpm which was reduced to 440 gpm for use in the analysis. Using these points, together with the tested pump head capacity curve, the high pressure injection over the full pressure range was established.

6. LONG TERM ECCS OPERATION

The preceding analyses were based on ECCS capability from one HPI pump and one core flooding tank. This condition assumed that the active LPI pump was lined up to pump to the core flood line that had the break and the other LPI pump was inoperative by the criteria of a single active failure.

Increased long term safety margin can be obtained by operator action to initiate low pressure injection through the unbroken core flood line. This action can easily be taken during the first 30 to 60 minutes after the CFT line break. The operator will open control room operated cross connect valves at the LPI pump discharge and check flow indicators in each of LPI lines to determine that some flow is going through each line. Equalization of flow in the lines can be accomplished from the control room by positioning of control valves in each LPI line. When flow is equalized through each line the LPI flow into the reactor vessel will be at least 1500 gpm with one pump operating and 3000 gpm if both pumps are operating.

The Oconee station operating procedures will be changed as follows:

1. Prior to switching suction on the ECCS and RB Spray pumps from the BWST to the RB sump or before shutting off all HPI pumps, check LPI flow indication and LPI pump operation to assure flow into the reactor vessel. This requires flow indication in each line since it is not known which line has the break.

2. If only one pump is running the operator should take the following actions.

   a. Attempt to start idle LPI pump. Failure to start may be ES actuation failure. Operator can operate valves and start pumps by remote manual control from control room.

   b. If pump (LP-P1C) is available, place in operation on the LPI string where pump is not running by opening valves in suction and discharge crossover lines and starting pump LP-P1C from the control room. Observe flow indication in the LPI line. This action produces 3000 gpm through each LPI line.

   c. If operator cannot start either of the two LPI pumps (steps a & b), perform the following steps to achieve flow into the reactor vessel from the one active LPI pump:

      Open discharge crossover valves to get LPI flow into each of the LPI lines.

      Monitor LPI flow indication to assure flow through each line.

Adjust the throttle valve in each LPI line until a flow balance is achieved. This will give approximately 1500 gpm through each line.

All valves also have local handwheels that can be manually actuated.

7. COMPARISON OF EVALUATION MODEL WITH APPENDIX A, PART 4 OF THE AEC INTERIM ACCEPTANCE CRITERIA FOR EMERGENCY CORE COOLING SYSTEMS

Although Appendix A, Part 4 is strictly appropriate only for breaks larger than 0.5 ft^2, a check list comparison of that evaluation model to the one used in this report may be of convenience to the reader. Evaluation and explanations are provided on a point by point basis with a subdivision consistent with Appendix A, Part 4.

The first paragraph of Appendix A, Part 4 lists several reports written by B&W which document techniques to be applied in the large break evaluation model. These reports are appropriate as follows:

1. CRAFT - This report and the code described are used for the CFT line break.

2. REFLOOD - This code is used for the purpose of calculating the refilling of a vessel once that vessel has reached end of blowdown. As that situation does not occur for the CFT line break, the code and its report do not apply and are not used.

3. THETA 1-B - This report and the code described are used for the CFT line break.

4. BAW-10034 - This report is written for large breaks.

Appendix A, Part 4 goes on to list specific instructions for the large break evaluation model.

1.1 Core and System Noding

1.1.1 Only one core node has been used in the CFT line break analysis.

1.1.2 The Theta model used during the flow controlled heat transfer regime had 6 fuel nodes, 2 clad nodes, and 10 axial levels. After the flow controlled heat transfer regime, after 300 seconds, the Quench code is used. This code has 1 fuel and 1 clad node and must be applied individually at separate axial levels.

1.2 Pump Model

This model is the same used in BAW-10052 and is discussed in that report. It is different from the model used in large break analysis though both models are consistent with the Appendix A, Part 4 guideline.

1.3 Break Characteristics

This statement does not apply to a specific break like the CFT line break.

1.4 Discharge Coefficient

As suggested by Appendix A, Part 4, a discharge coefficient of 1.0 has been used.

1.5 Decay Heat

The decay heat curve suggested in Appendix A, Part 4 was used in the CFT line break analysis.

1.6 Time to Departure from Nucleate Boiling (DNB)

This was done as suggested in the large break evaluation model.

1.7 Film Boiling Heat Transfer

This was done as suggested by the large break evaluation model except that for pool film boiling used in the QUENCH code, the Morgan correlation was used.

1.8 Metal-Water Reaction Rate

This was done as suggested by the large break evaluation model. Temperatures for this accident, however, prohibit any significant metal-water reaction.

1.9 Core Flow Rate

This was done as suggested by the large break evaluation model while flow was controlling the heat transfer.

1.10 Enthalpy and Pressure

This was done as suggested by the large break evaluation model.

1.11 Core Flooding Tank Bypass

As downcomer steam flows were insufficient to cause entrainment of core flooding tank water, this bypass assumption was not imposed on the CFT line break analysis.

Appendix A, Part 4 then proceeds to describe the evaluation model for the reflood portion of the large break. As there is no classic reflood portion for the CFT line break, this section does not apply to the analysis.

LIST OF REFERENCES

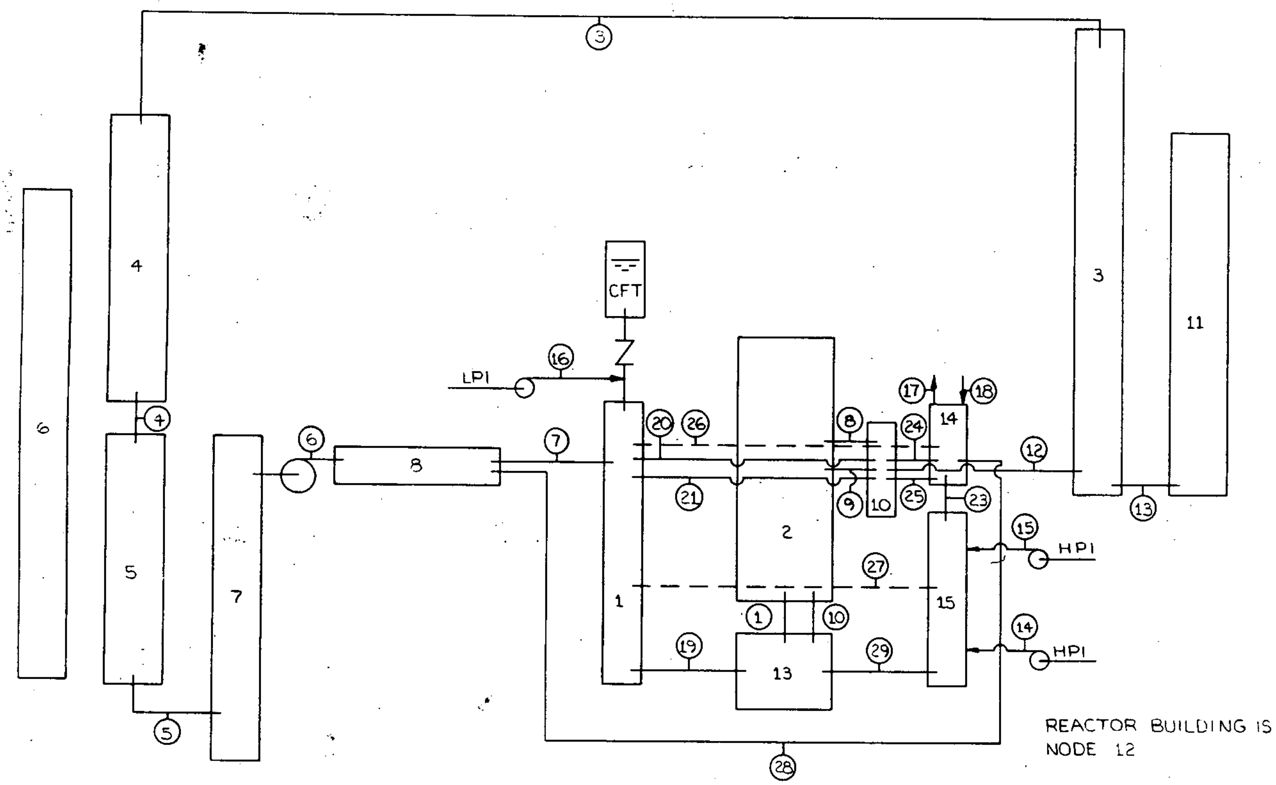
1. C.E. Parks, B.M. Dunn, and R.C. Jones, Multinode Analysis of Small Breaks for B&W's 2568-MWt Nuclear Plants, BAW-10052, Babcock and Wilcox, Lynchburg, Virginia, September, 1972.

2. CRAFT - Description of Model for Equilibrium LOCA Analysis Program, BAW-10030, Babcock and Wilcox, Lynchburg, Virginia, October, 1971.

3. C.J. Hocever and T.W. Wineinger, THETA 1-B - A Computer Code for Nuclear Reactor Core Thermal Analysis, IN - 1445, Idaho Nuclear Corp., February, 1971.

4. J.A. Redfield and J.H. Murphy, Void Fraction and Residual Water Predictions During Loss of Coolant, WAPD-T-2155, Westinghouse, BAPL, September 1968.

FIGURE 3-1 CRAFT EVALUATION MODEL



LEGEND

NODE NO.	IDENTIFICATION	PATH NO.	IDENTIFICATION
1	DOWNCOMER	1	CORE
2	CORE & UPPER HEAD	3	HOT LEG PIPING
3	HOT LEG PIPING	4	SG TUBES
4	SG	5	COLDLEG PIPING
5	SG	6	PUMP
6	SG(SECONDARY SIDE)	7	COLD LEG PIPING
7	COLD LEG PIPING	8	PLENUM TO OUTLET NOZZLE
8	COLD LEG PIPING	9	PLENUM TO OUTLET NOZZLE
10	OUTLET NOZZLE	10	CORE BYPASS
11	PRESSURIZER	12	HOT LEG PIPING
12	CONTAINMENT	13	PRESSURIZER SURGE LINE
13	LOWER HEAD	14	HIGH PRESSURE INJECTION
14	DOWNCOMER	15	HIGH PRESSURE INJECTION
15	DOWNCOMER	16	LOW PRESSURE INJECTION
		17	LEAK PATH
		18	RETURN PATH
		19	DOWNCOMER
		20	VENT VALVES
		21	LEAKAGE
		23	DOWNCOMER
		24	VENT VALVES
		25	LEAKAGE
		26	CROSS FLOW-DOWNCOMER
		27	CROSS FLOW-DOWNCOMER
		28	COLD LEG PIPING
		29	DOWNCOMER

FIGURE 3-2 SENSITIVITY OF INNER VESSEL LIQUID VOLUME TO NODING

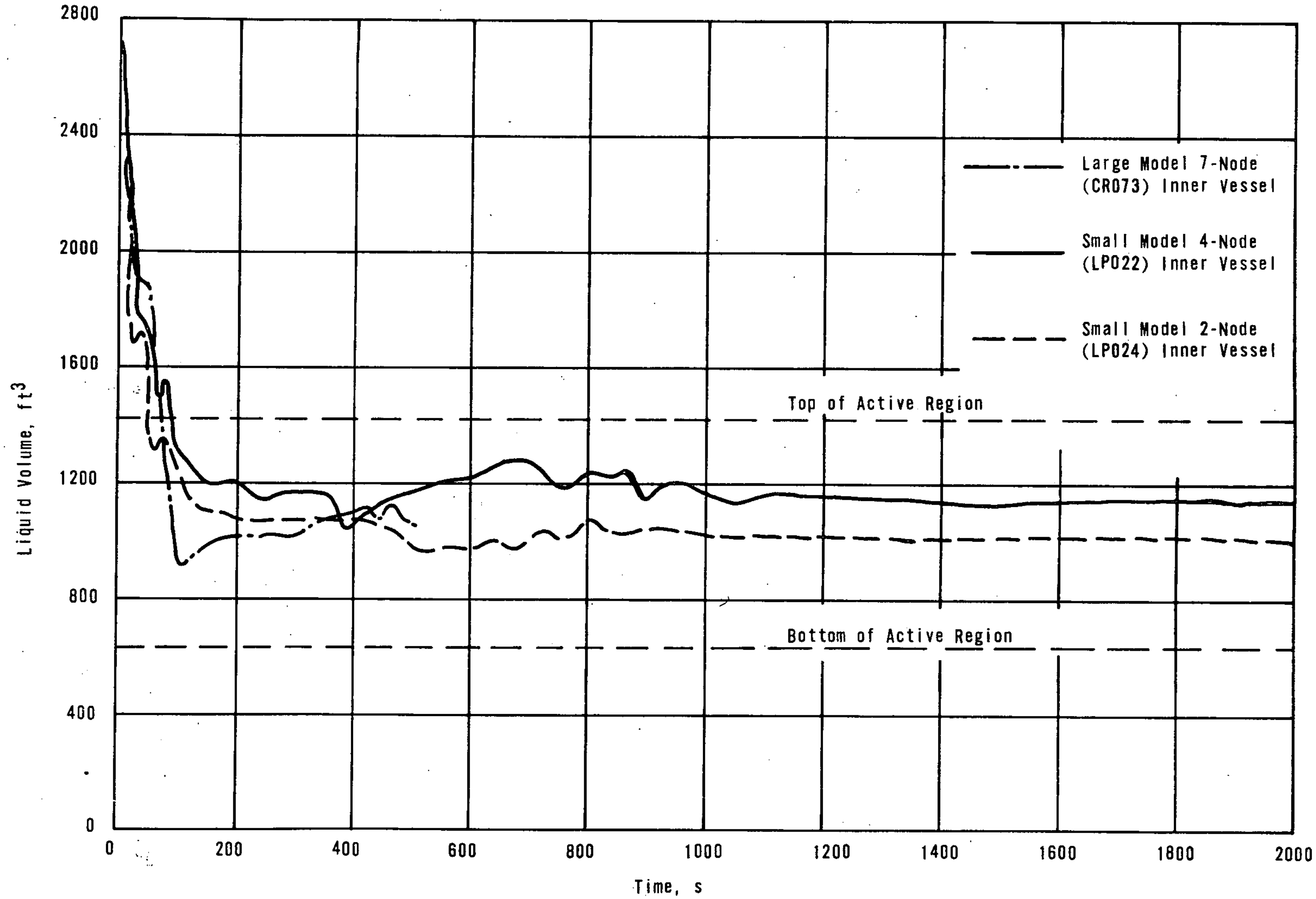


FIGURE 4-1 CORE PRESSURE VERSUS TIME

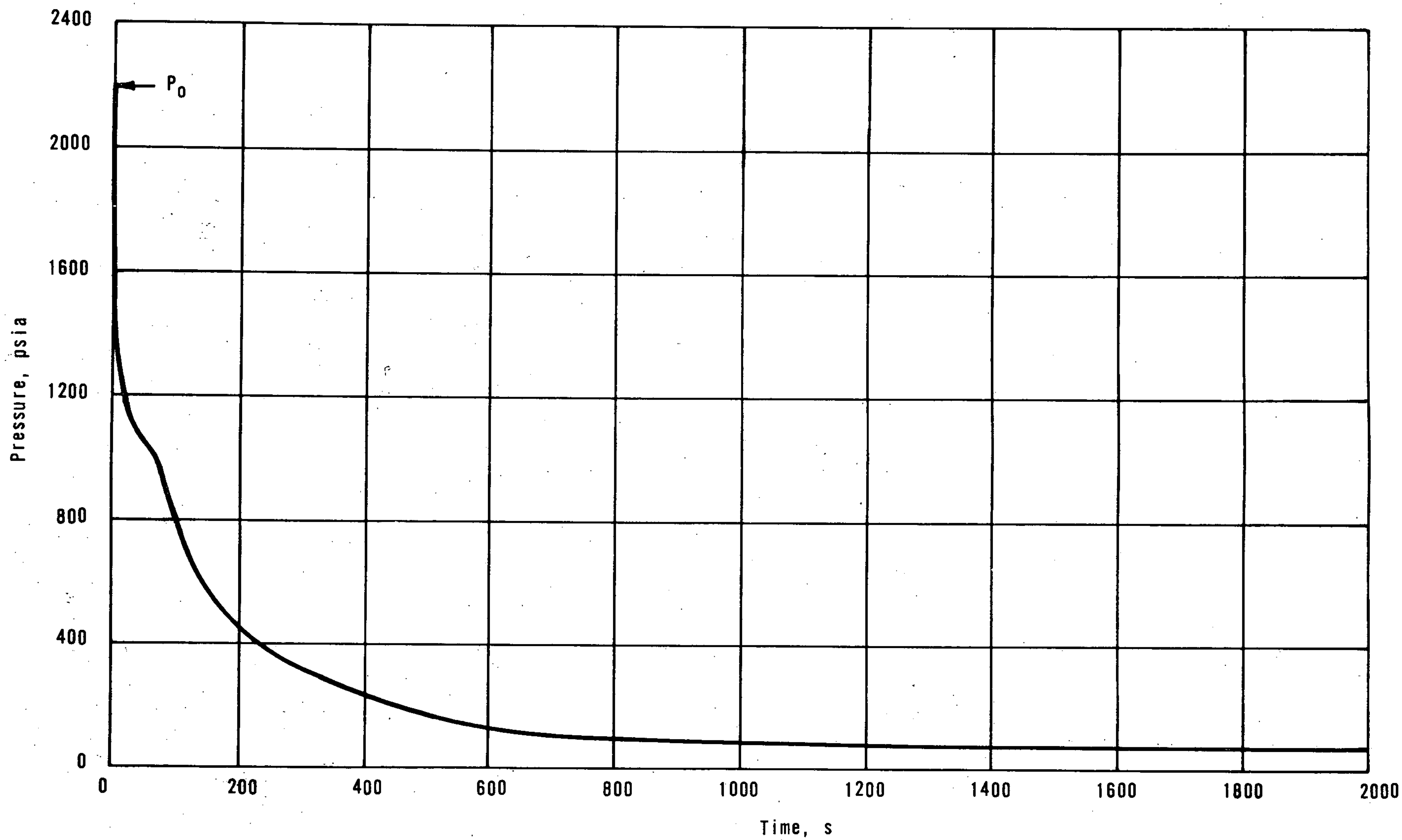


FIGURE 4-2 CORE POWER VERSUS TIME

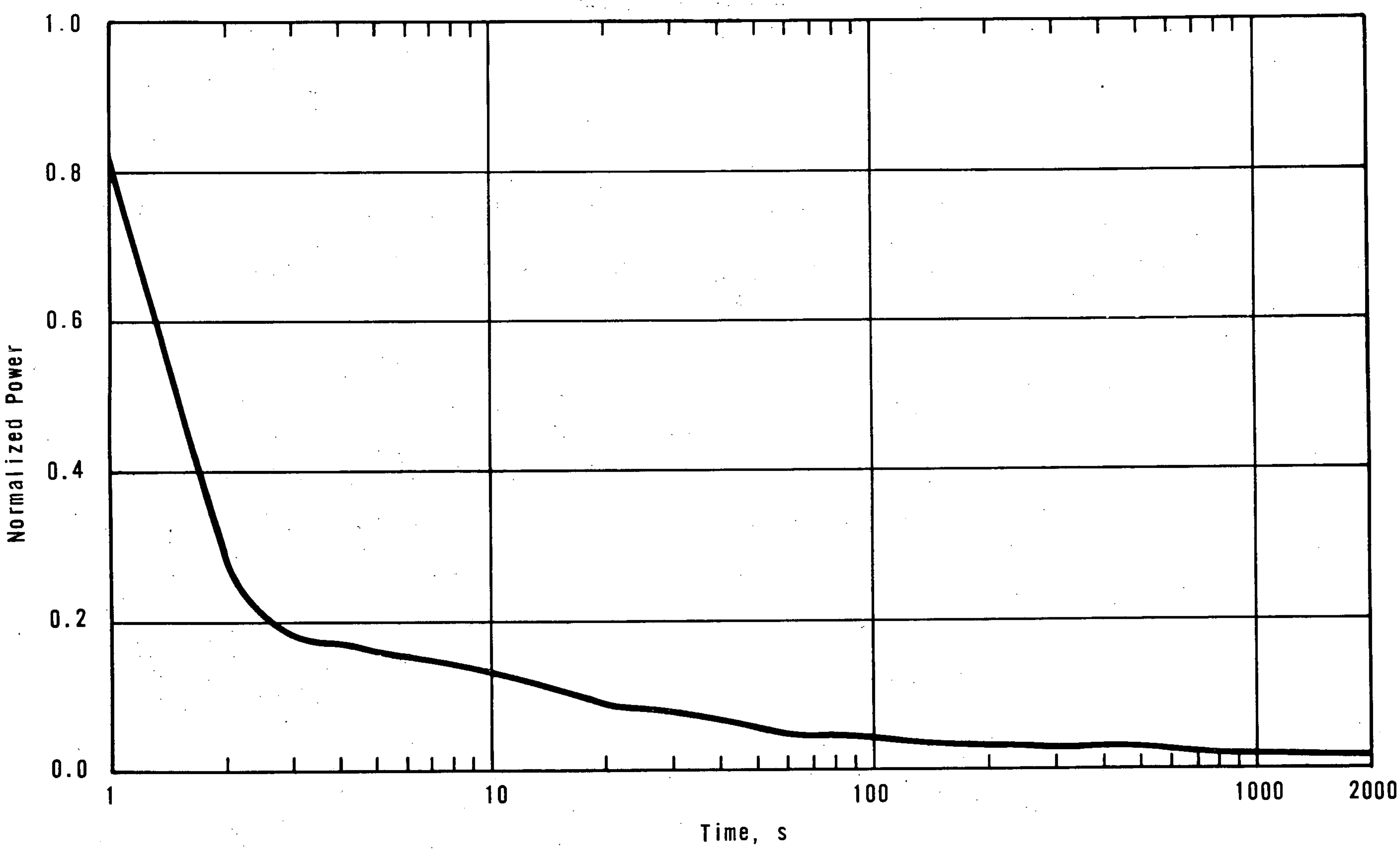


FIGURE 4-3 INNER VESSEL FLUID VOLUMES VERSUS TIME

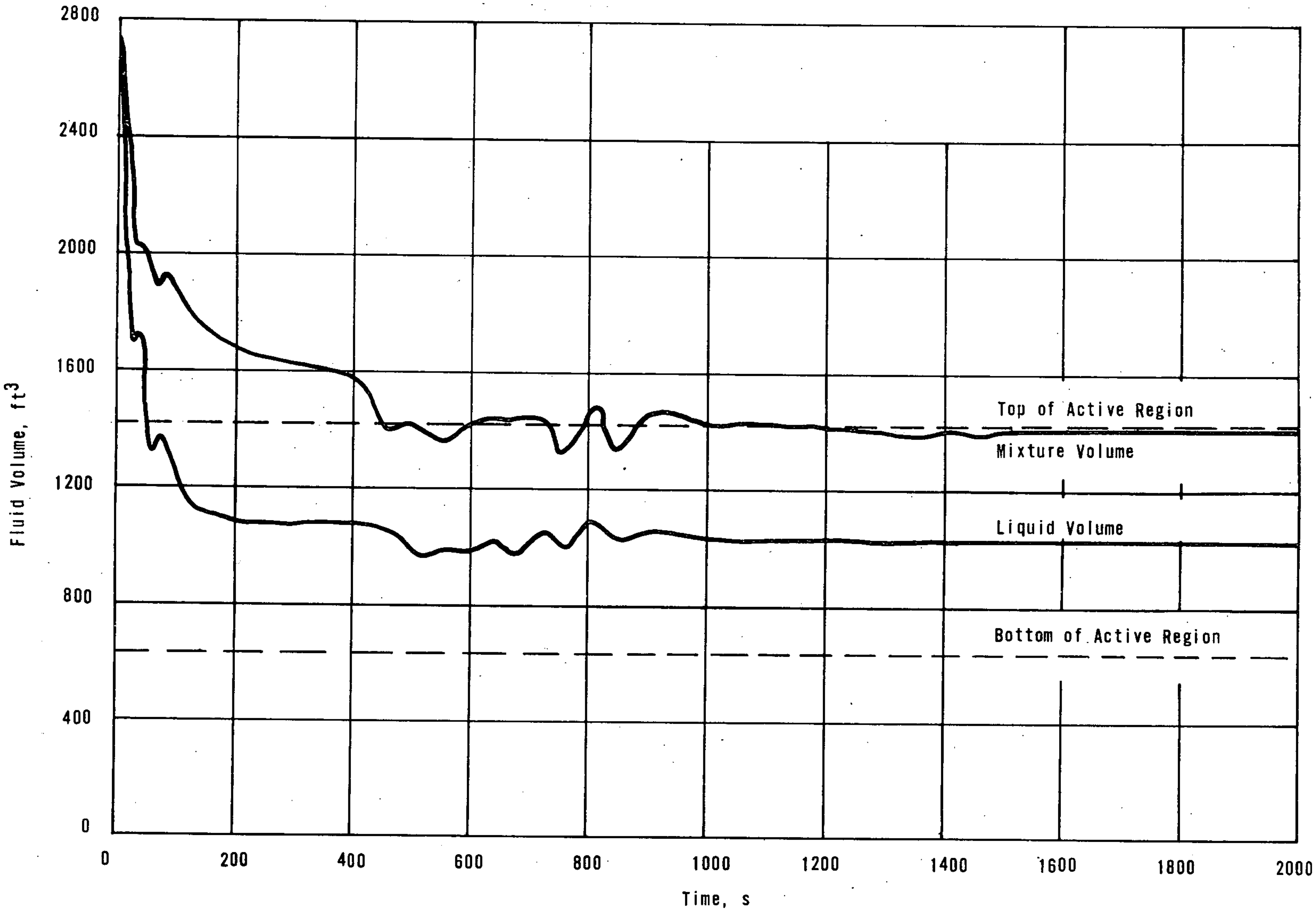


FIGURE 4-4   VENT VALVE FLOW VERSUS TIME

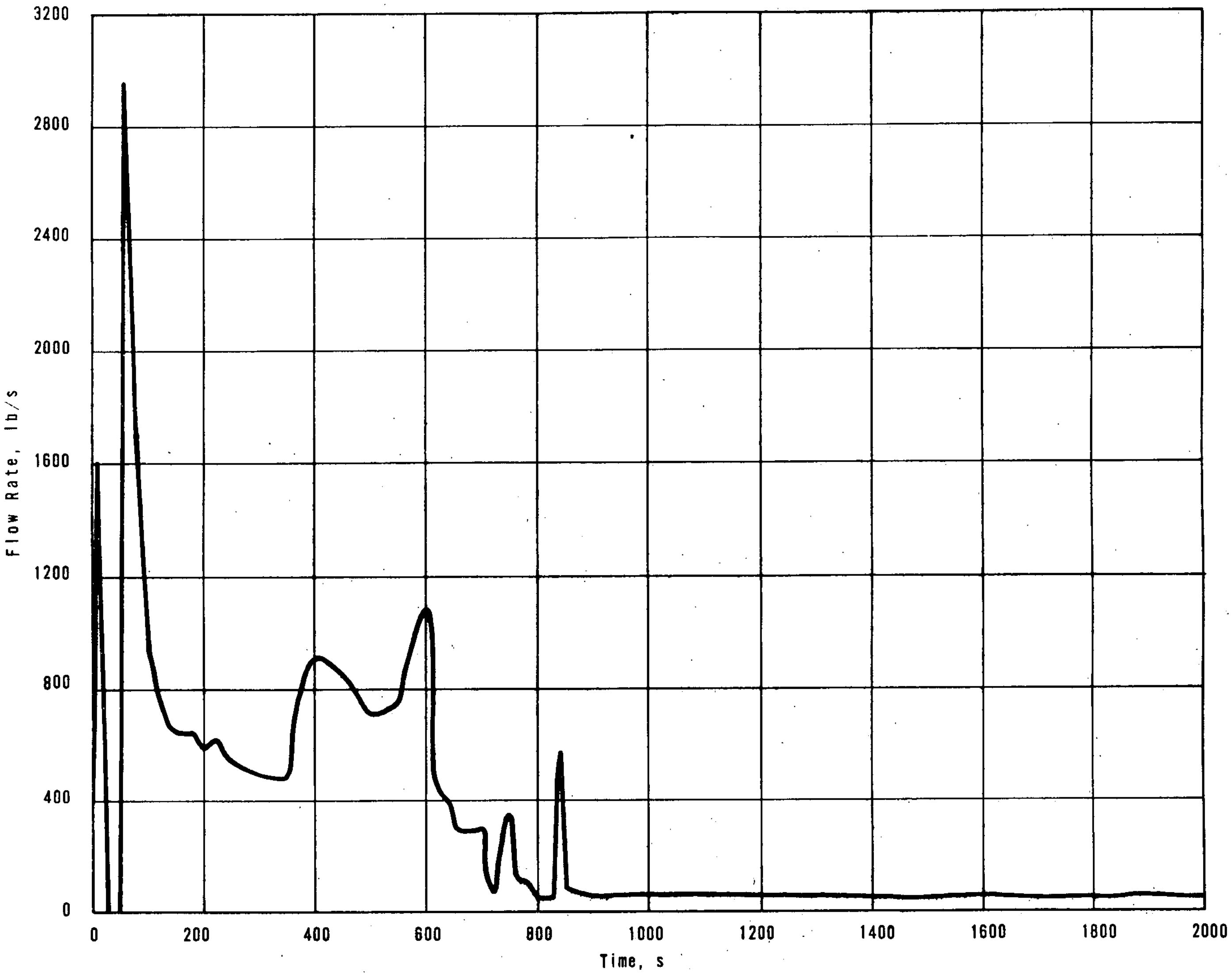


FIGURE 4-5 LEAK FLOW VERSUS TIME

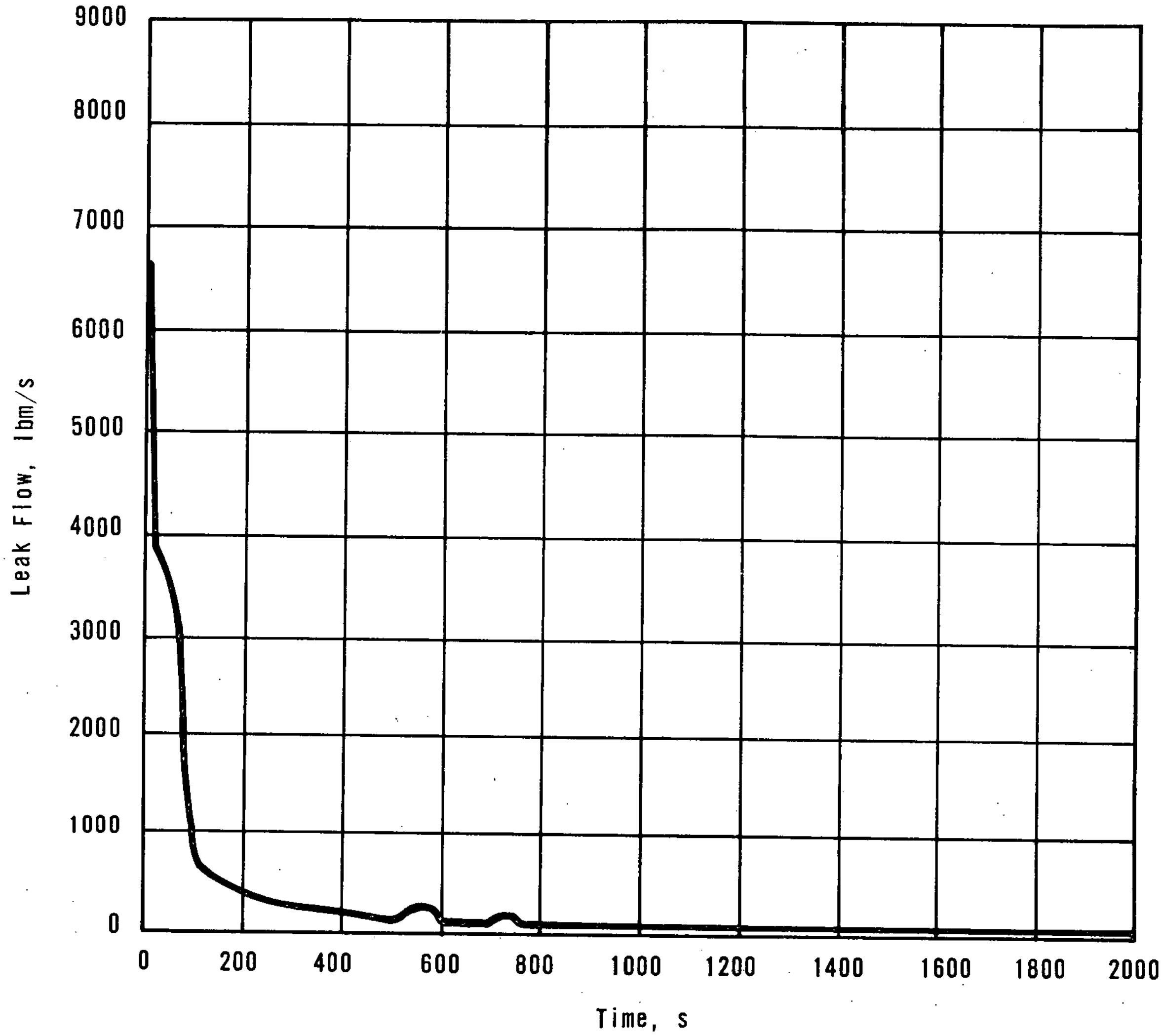


FIGURE 4-6 WATER HEIGHT IN CORE AND DOWNCOMER VERSUS TIME

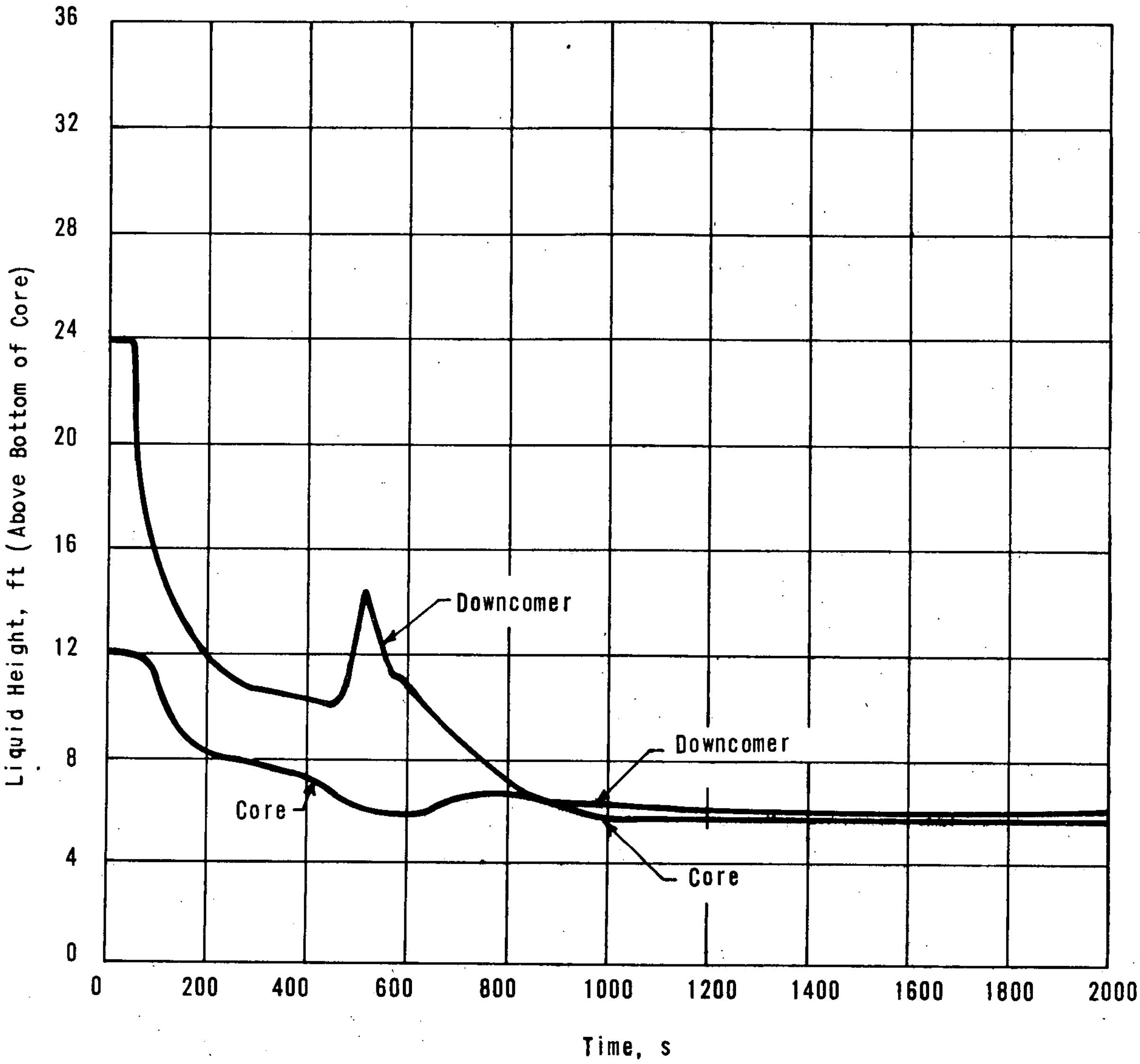


FIGURE 4-7 FLUID VELOCITIES FROM THE DOWNCOMER TO THE LOWER HEAD DURING CFT INJECTION

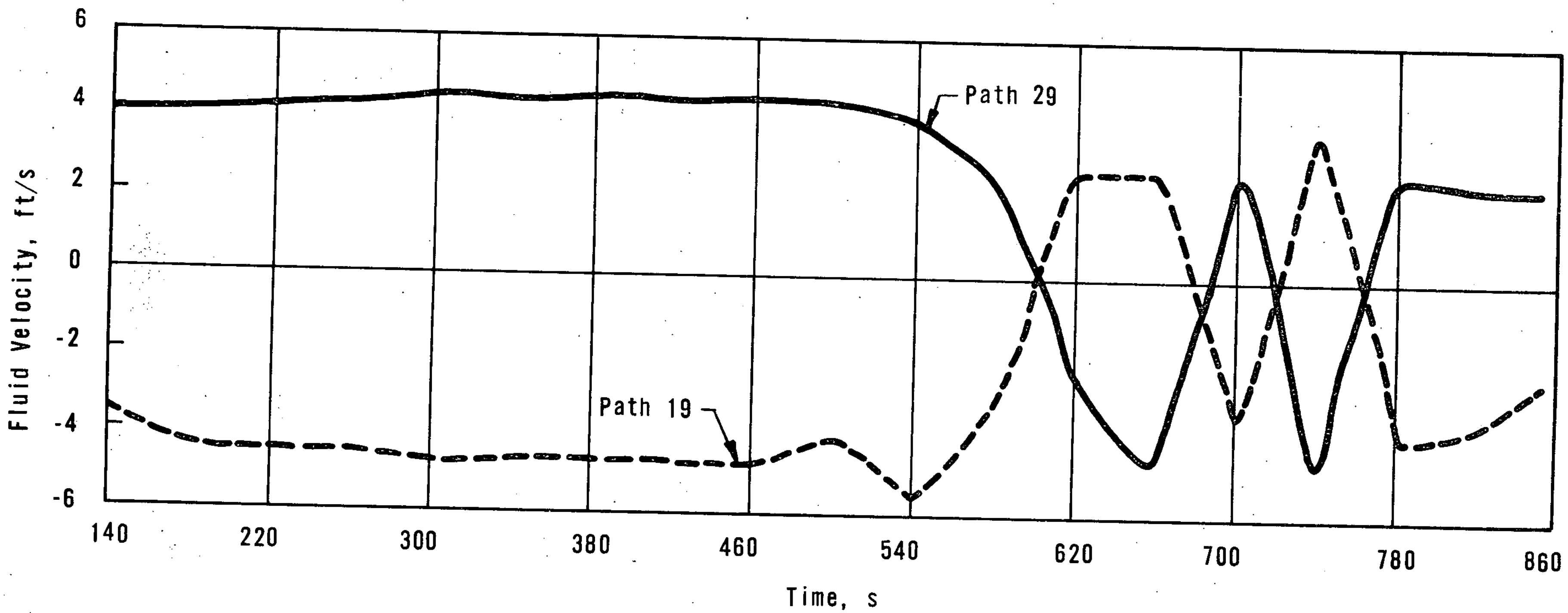


FIGURE 4-8 AXIAL POWER SHAPE FOR CASE 1

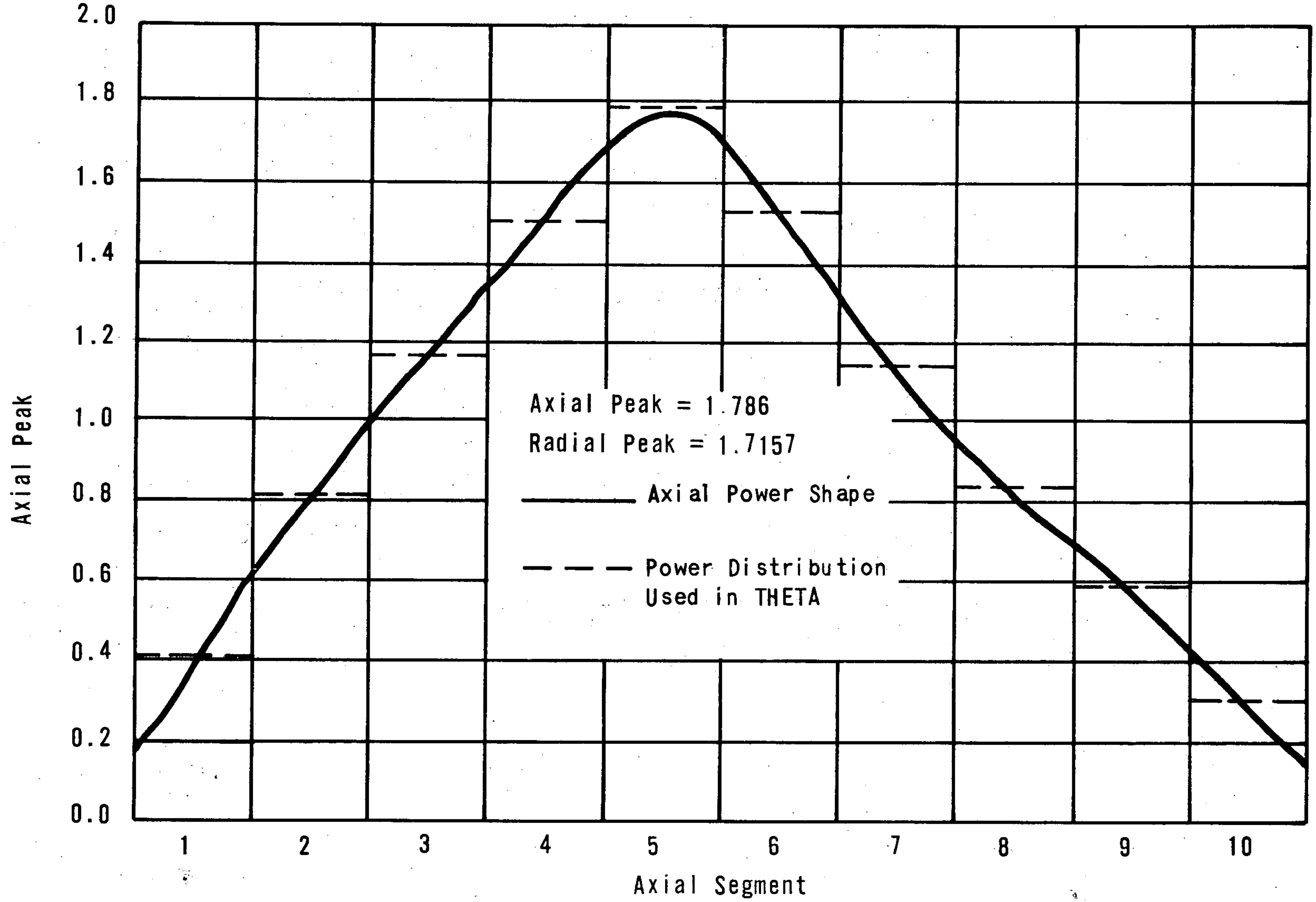


FIGURE 4-9   CASE 1, LEVEL 5 - SINK TEMPERATURE VERSUS TIME

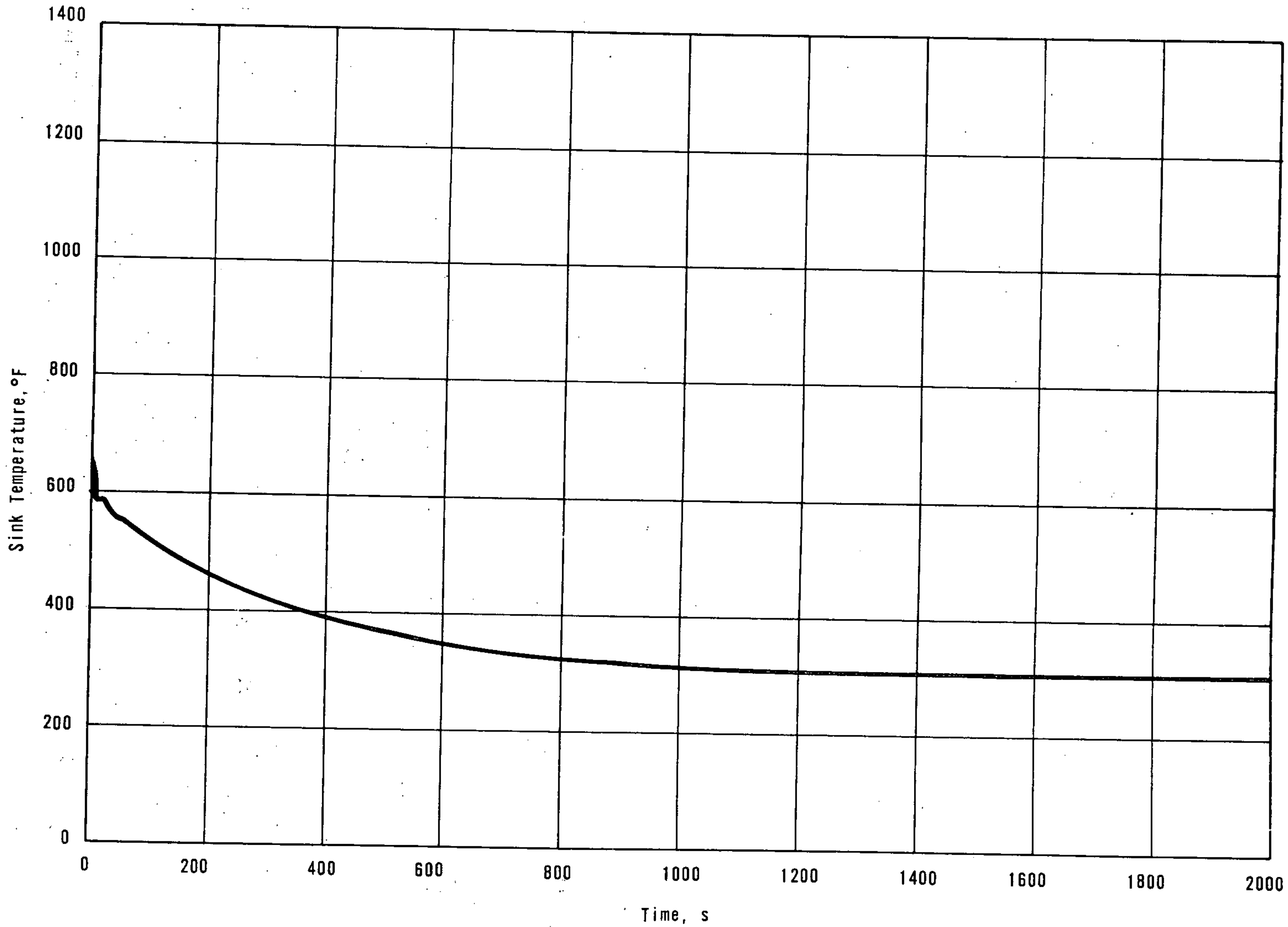


FIGURE 4-10 CASE 1, LEVEL 5 - HEAT TRANSFER COEFFICIENT VERSUS TIME

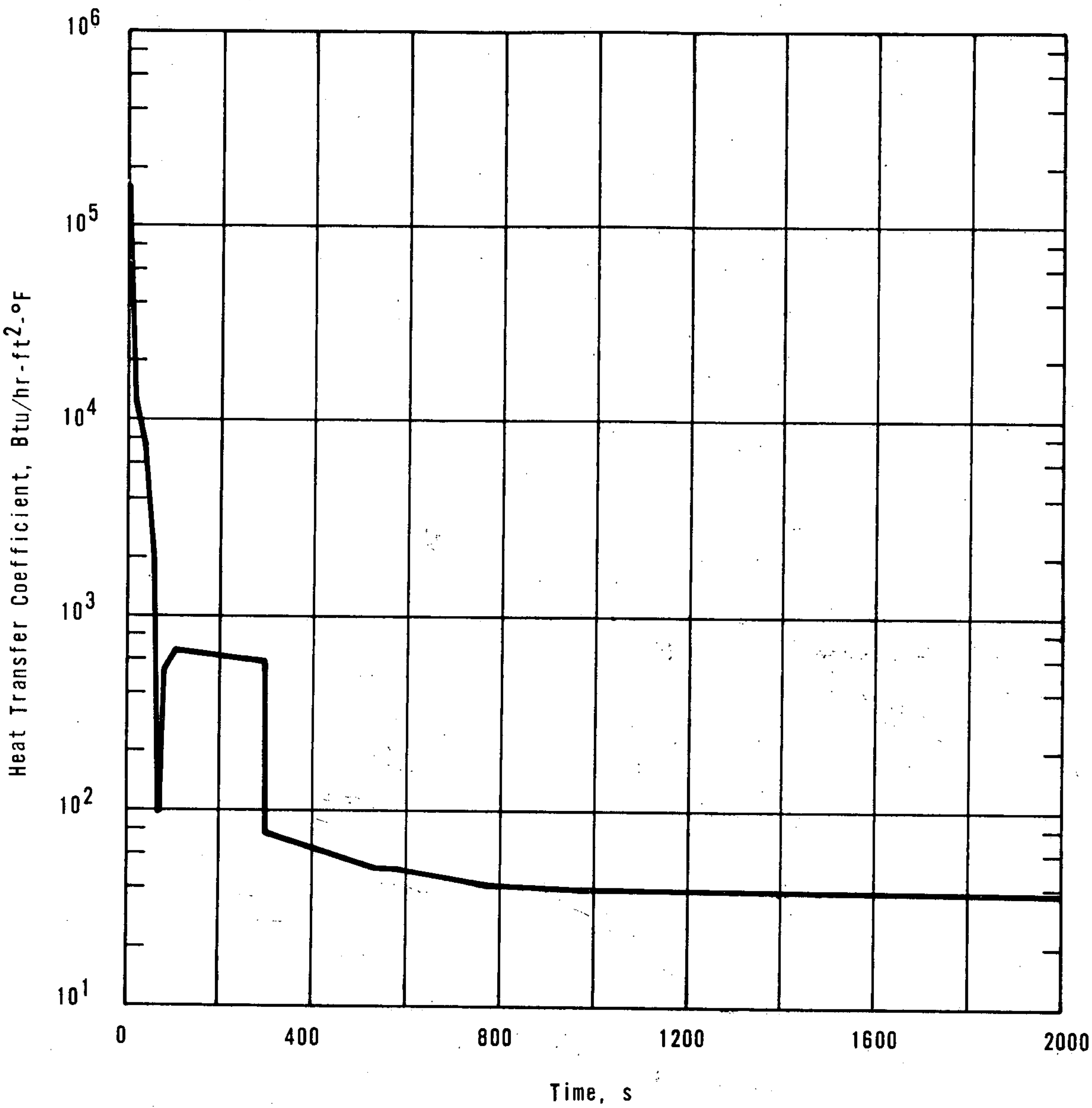


FIGURE 4-11 CASE 1, LEVEL 5 - CLADDING TEMPERATURE VERSUS TIME

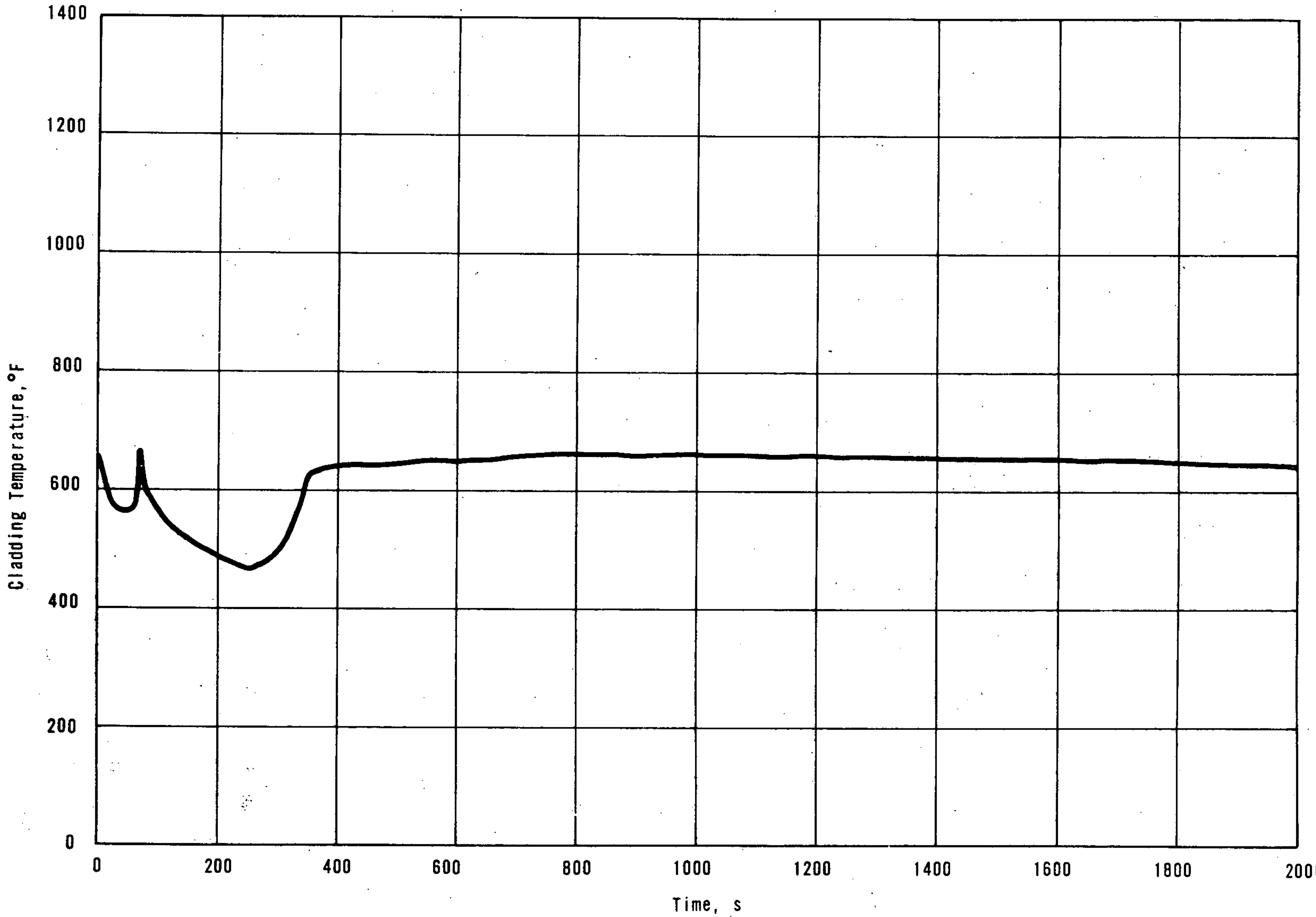


FIGURE 4-12 CASE 1, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

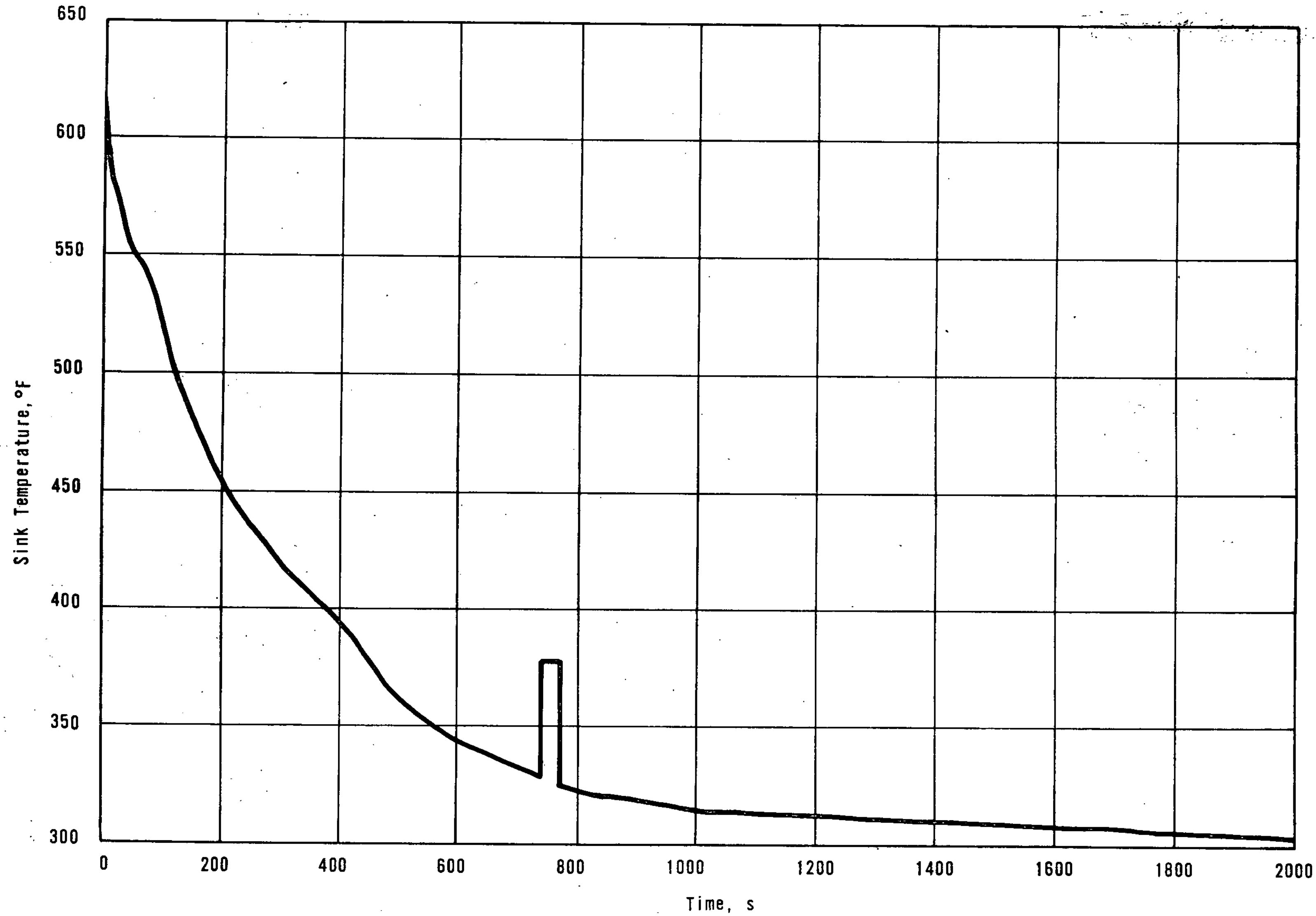


FIGURE 4-13 CASE 1, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

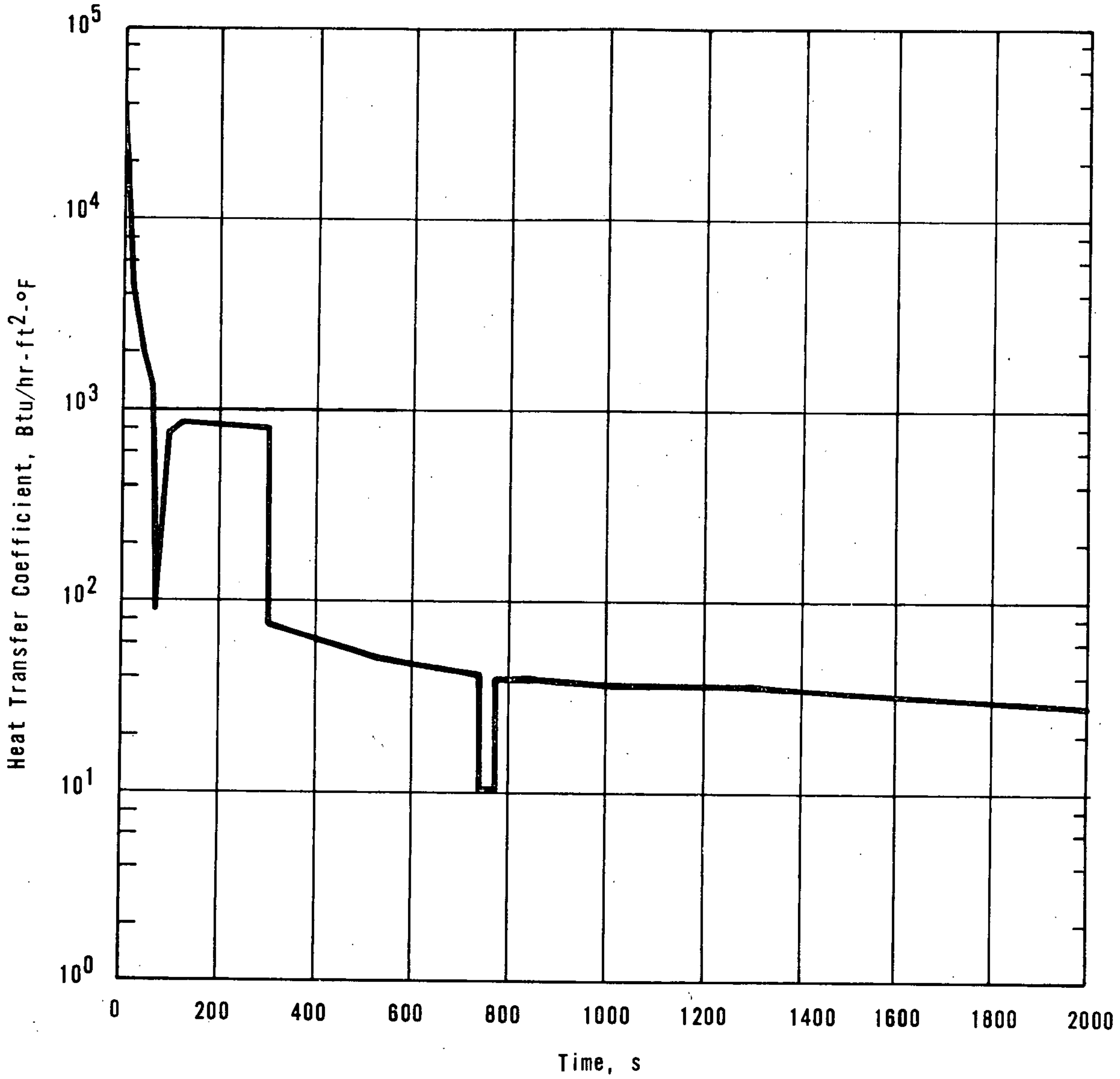


FIGURE 4-14   CASE 1, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME

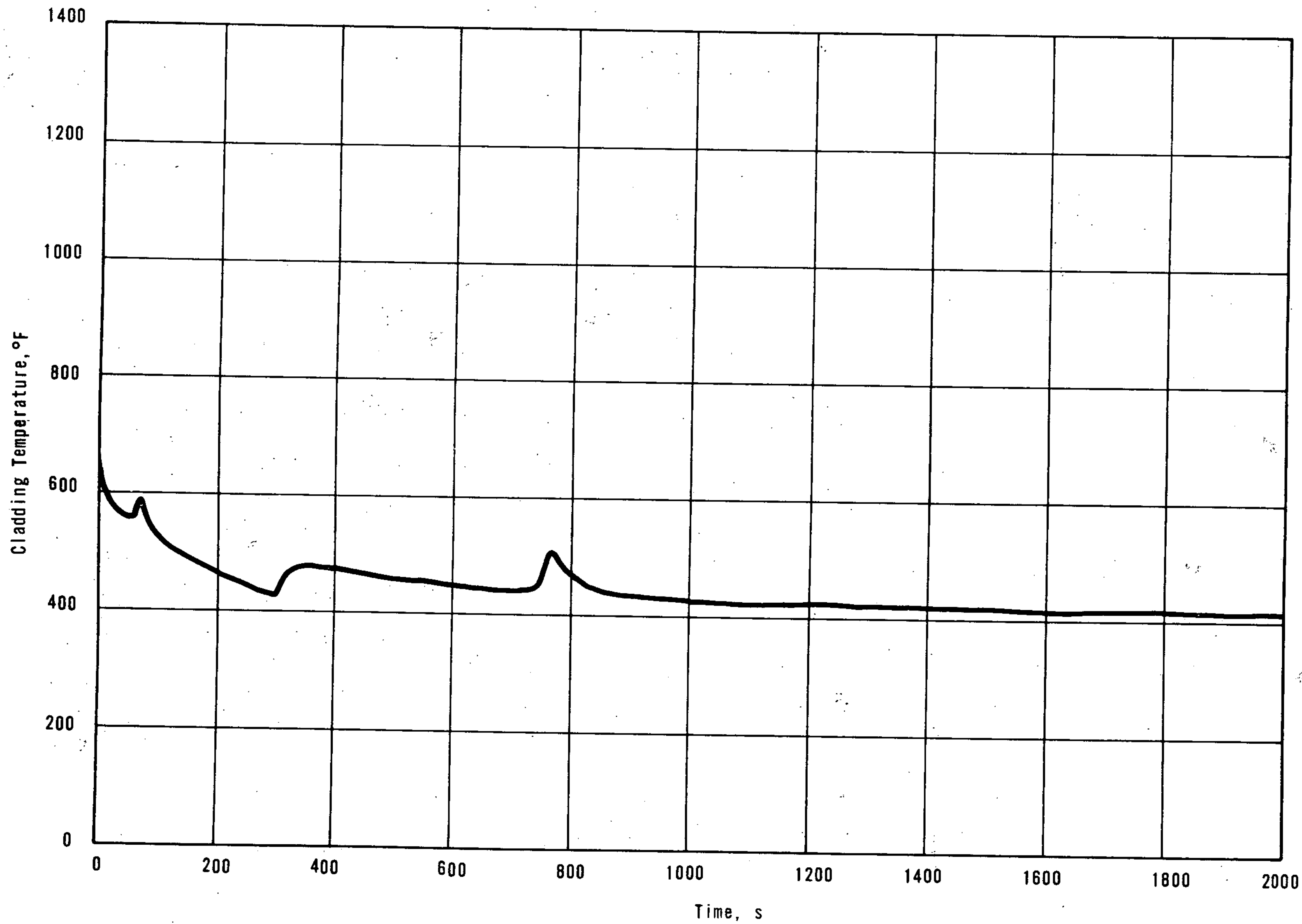


FIGURE 4-15 CASE 1, LEVEL 10 - SINK TEMPERATURE VERSUS TIME

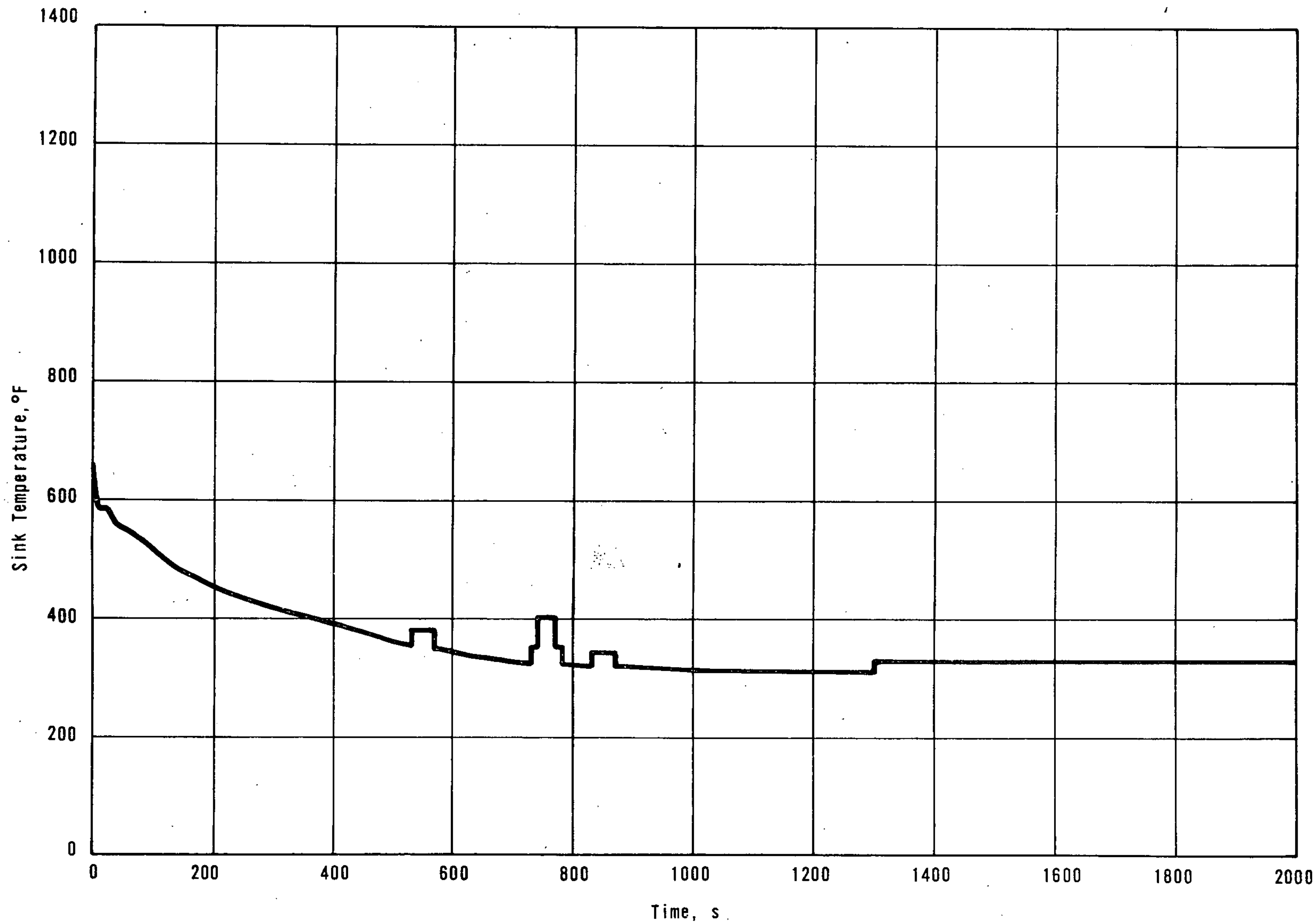


FIGURE 4-16 CASE 1, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

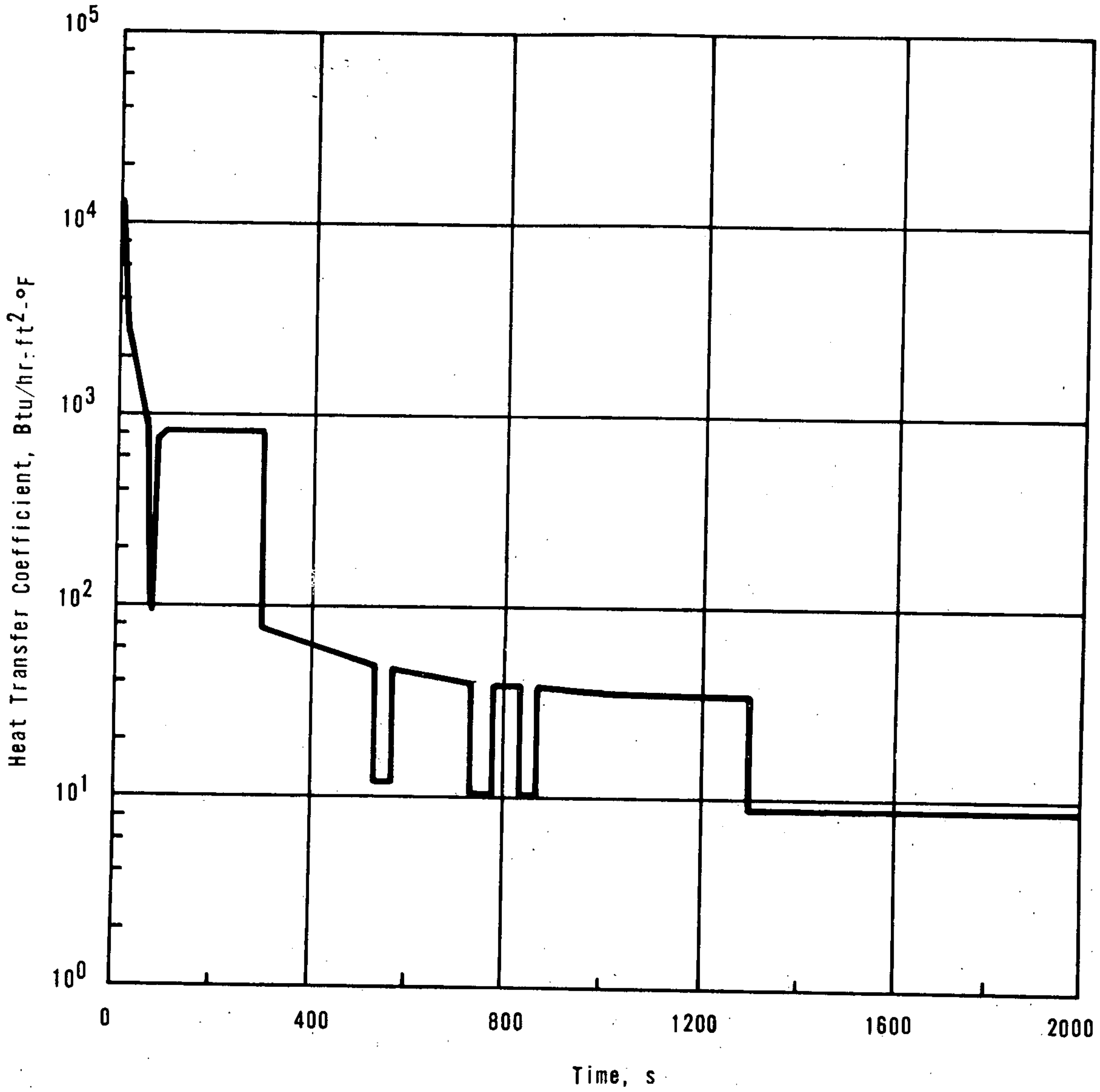


FIGURE 4-17 CASE 1, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME

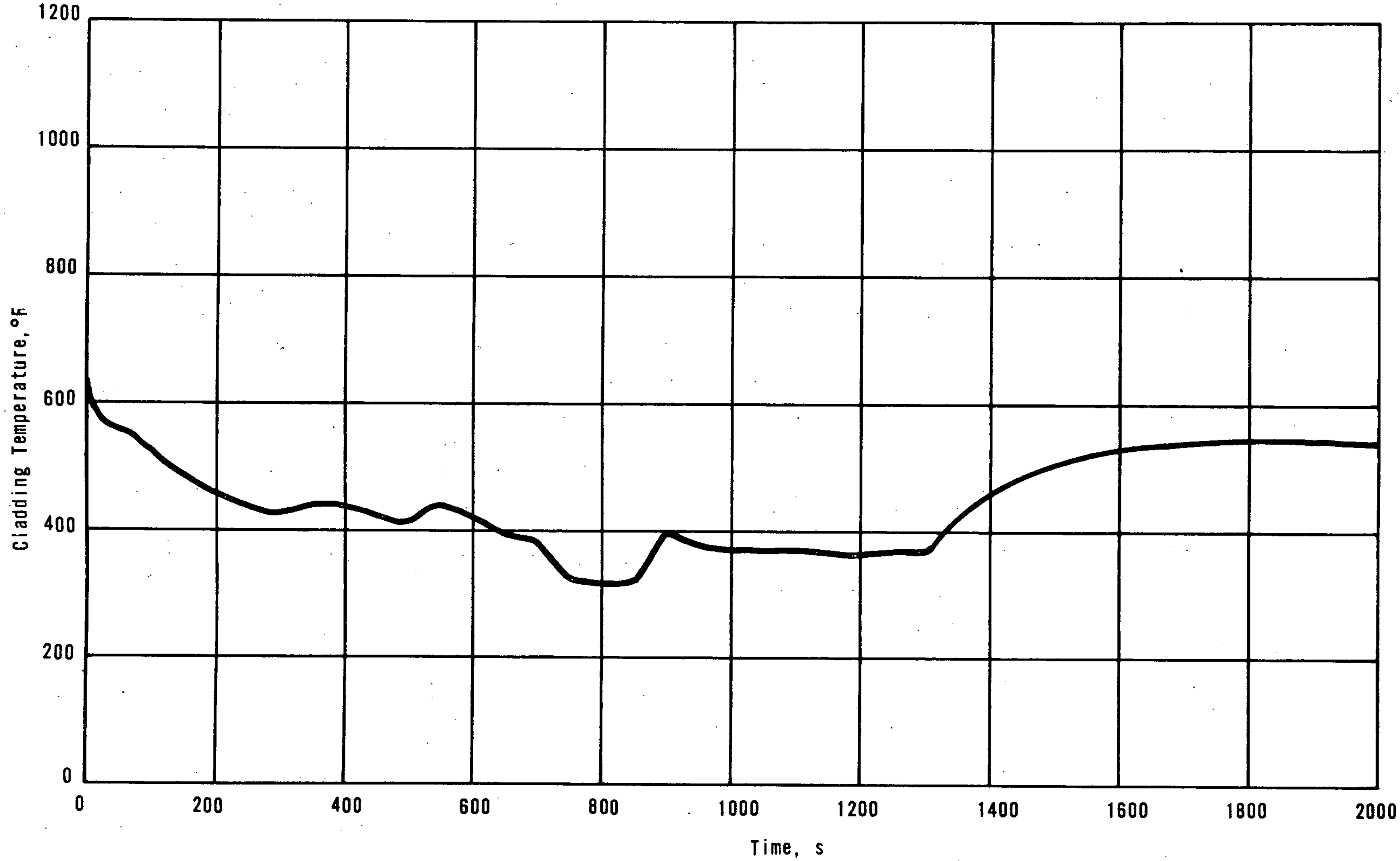


FIGURE 4-18 AXIAL POWER SHAPE FOR CASE 2

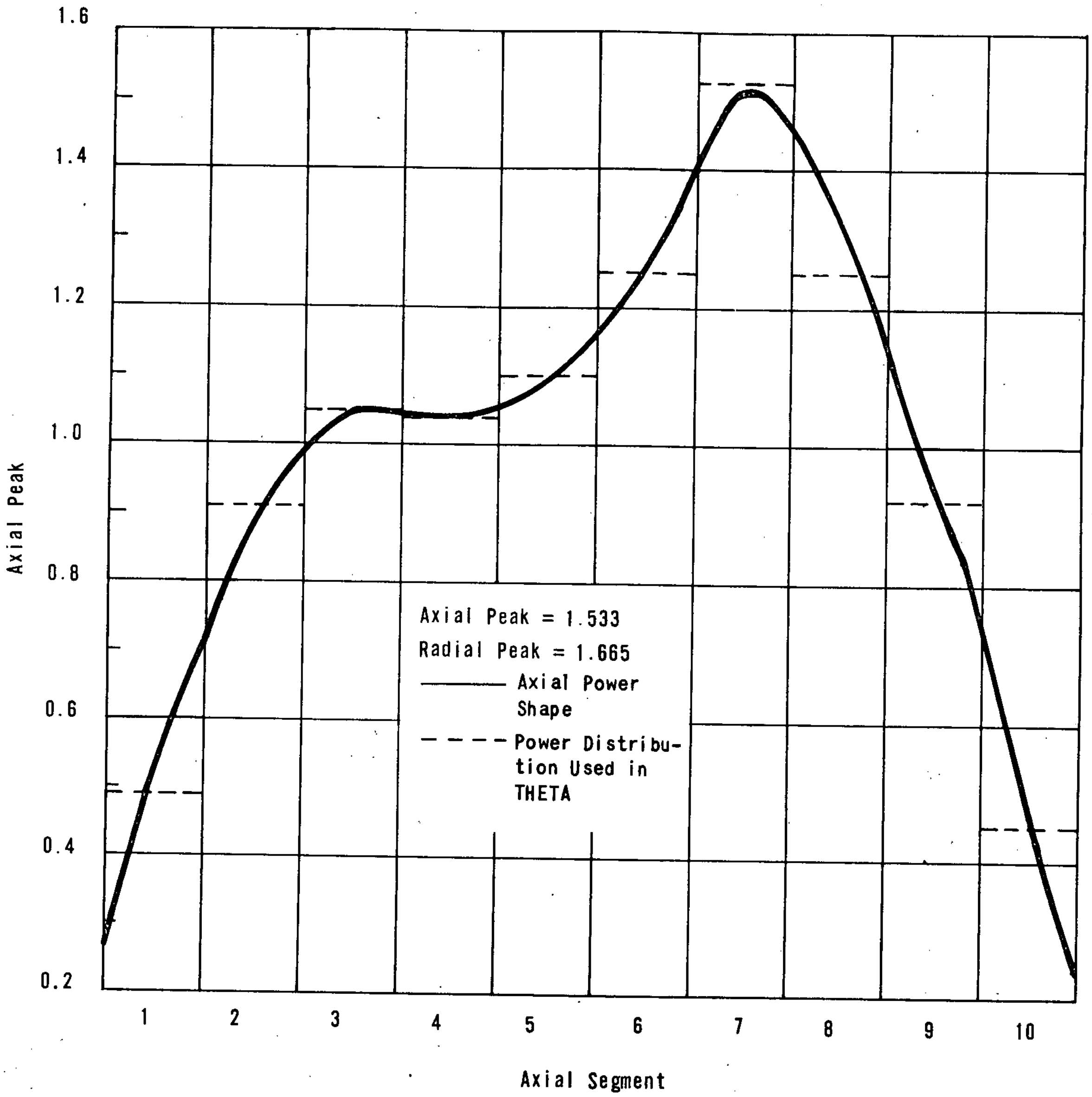


FIGURE 4-19   CASE 2, LEVEL 7 - SINK TEMPERATURE VERSUS TIME

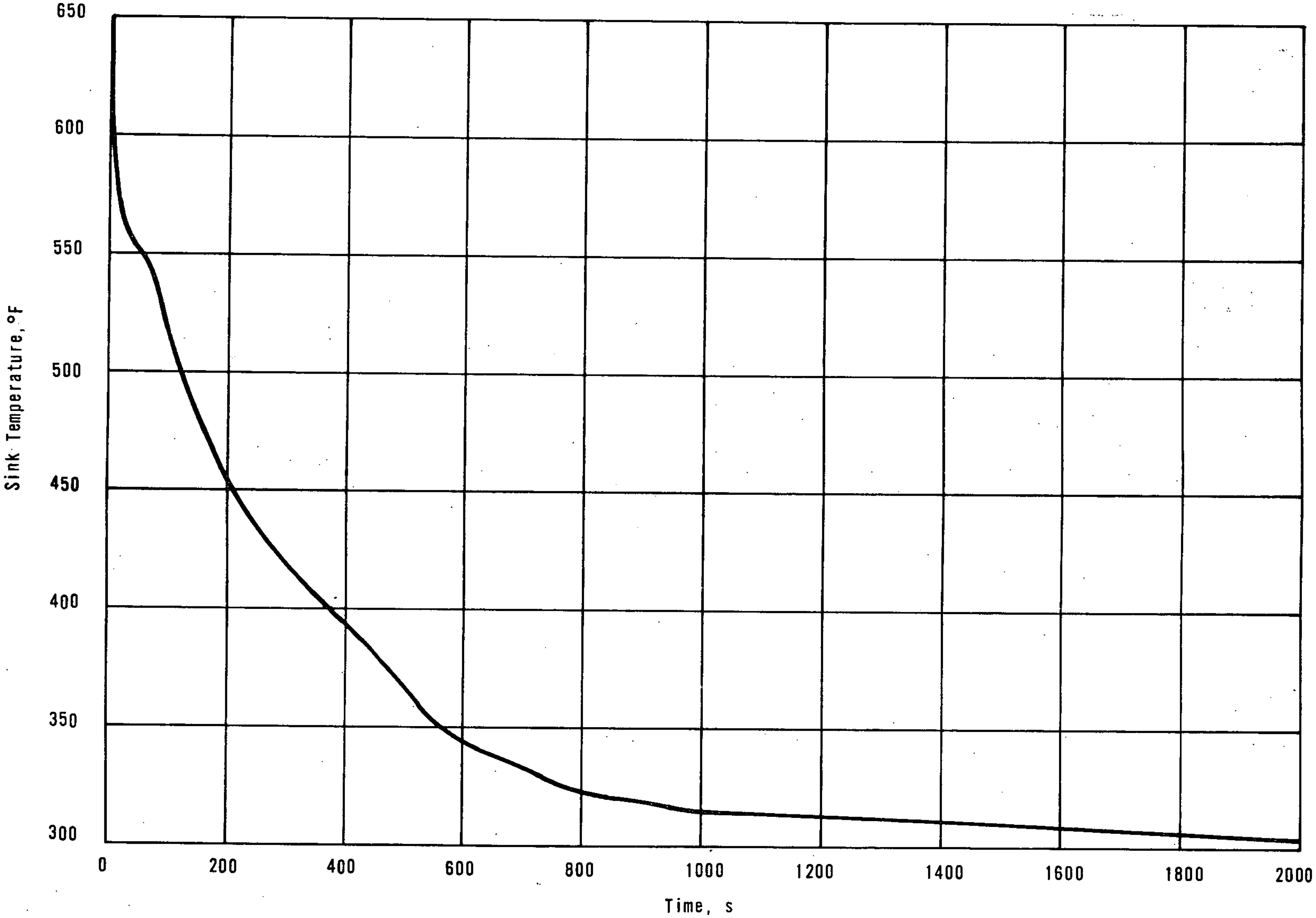


FIGURE 4-20 CASE 2, LEVEL 7 - HEAT TRANSFER COEFFICIENT VERSUS TIME

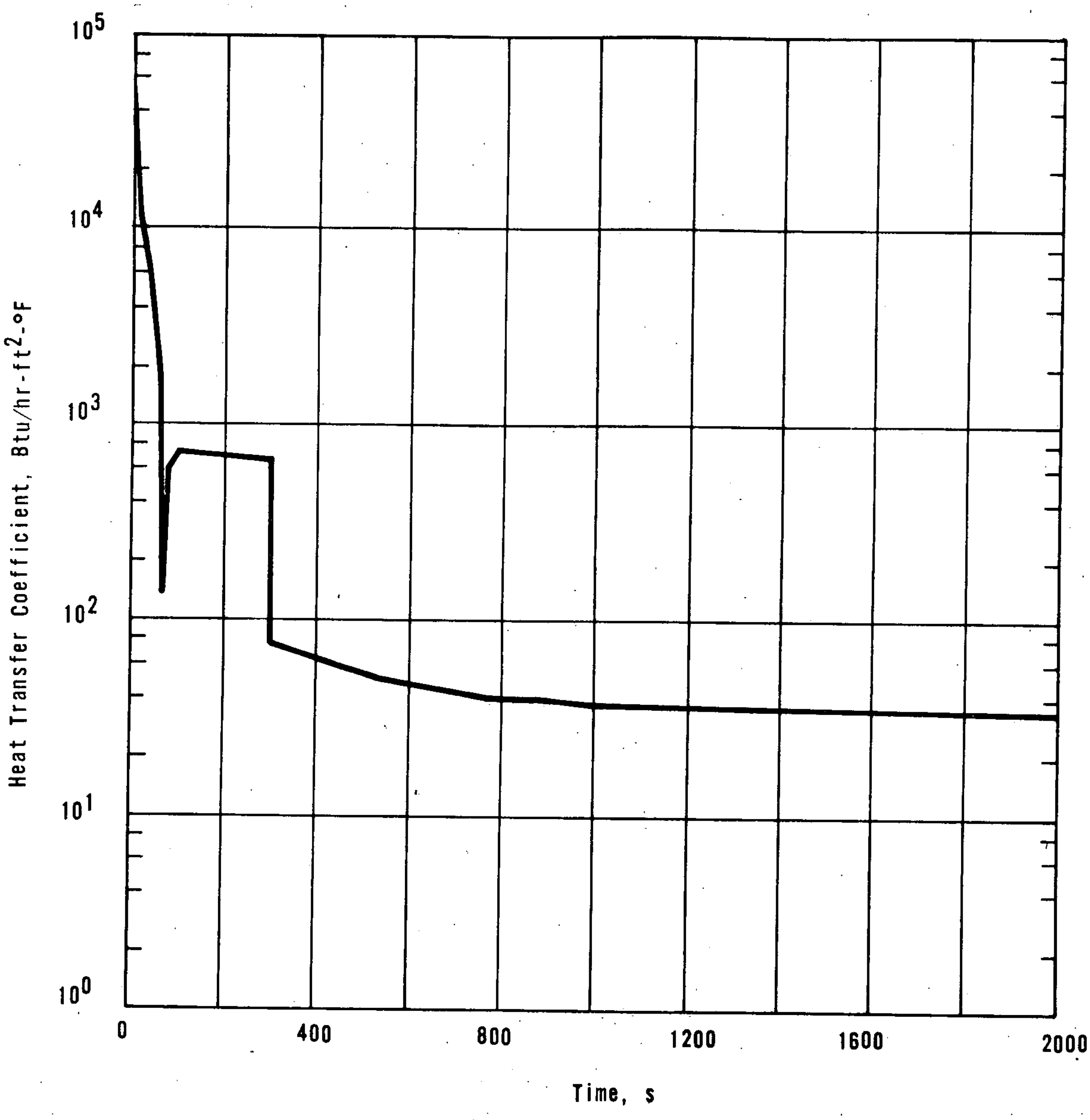


FIGURE 4-21 CASE 2, LEVEL 7 - CLADDING TEMPERATURE VERSUS TIME

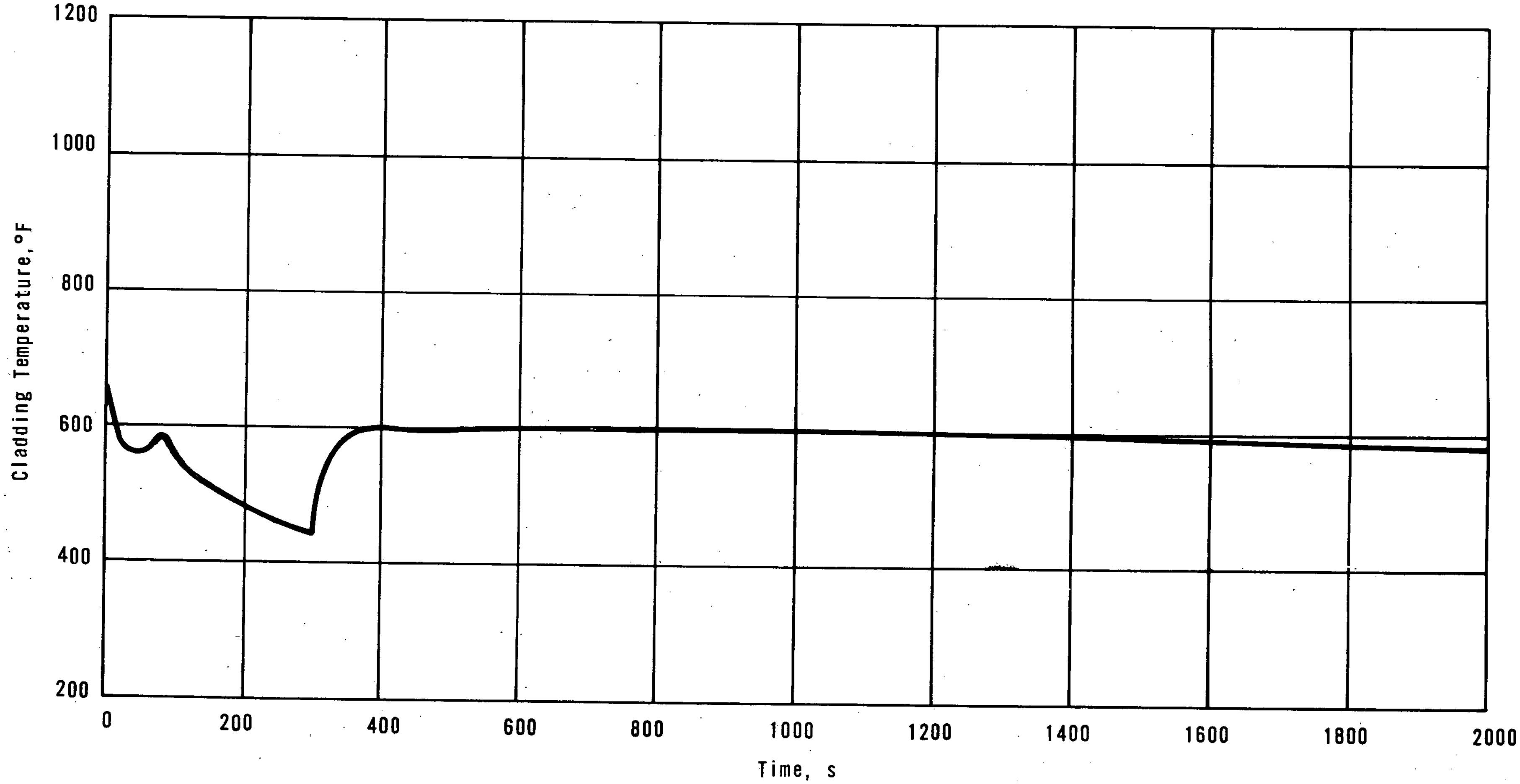


FIGURE 4-22 CASE 2, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

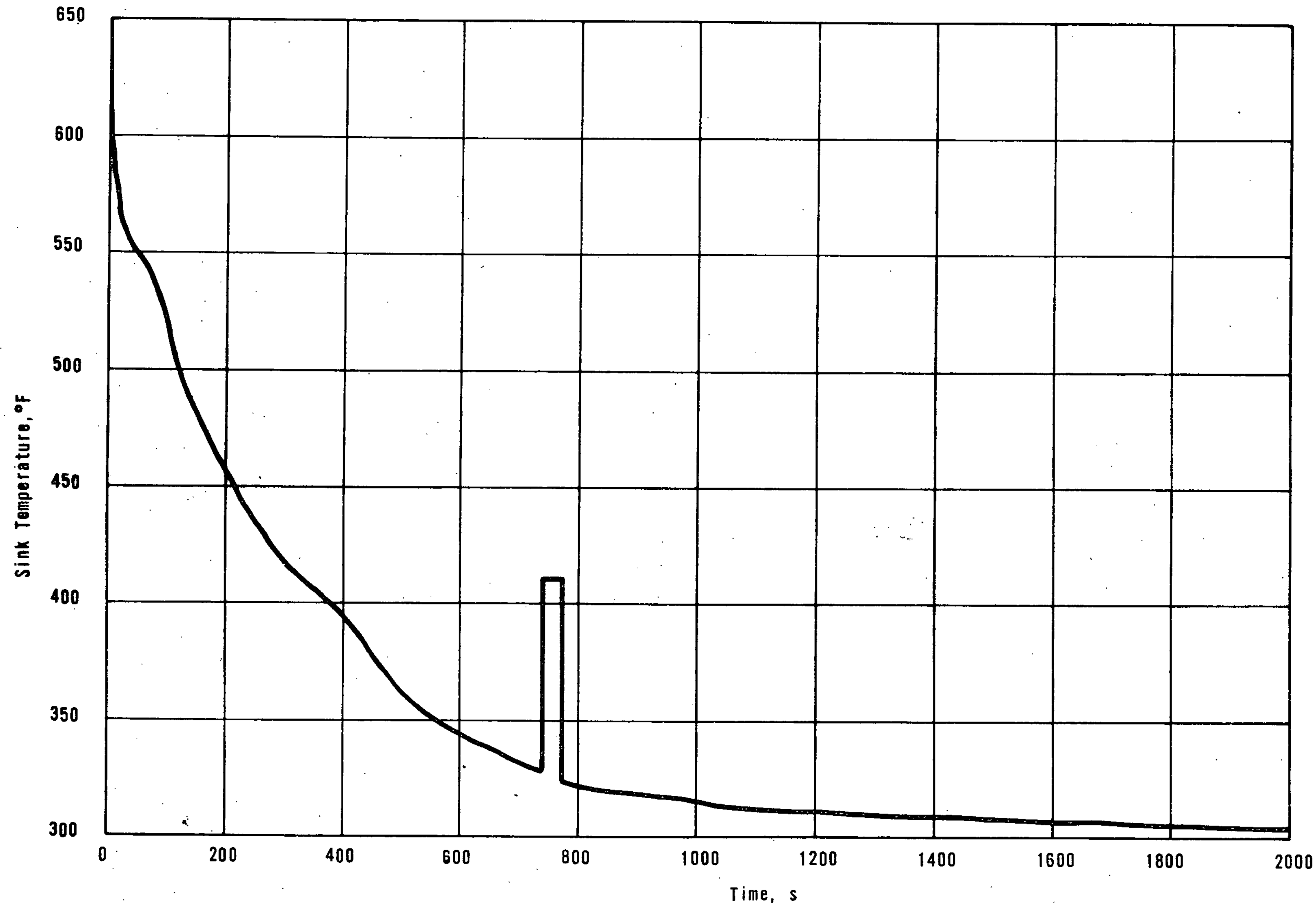


FIGURE 4-23 CASE 2, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

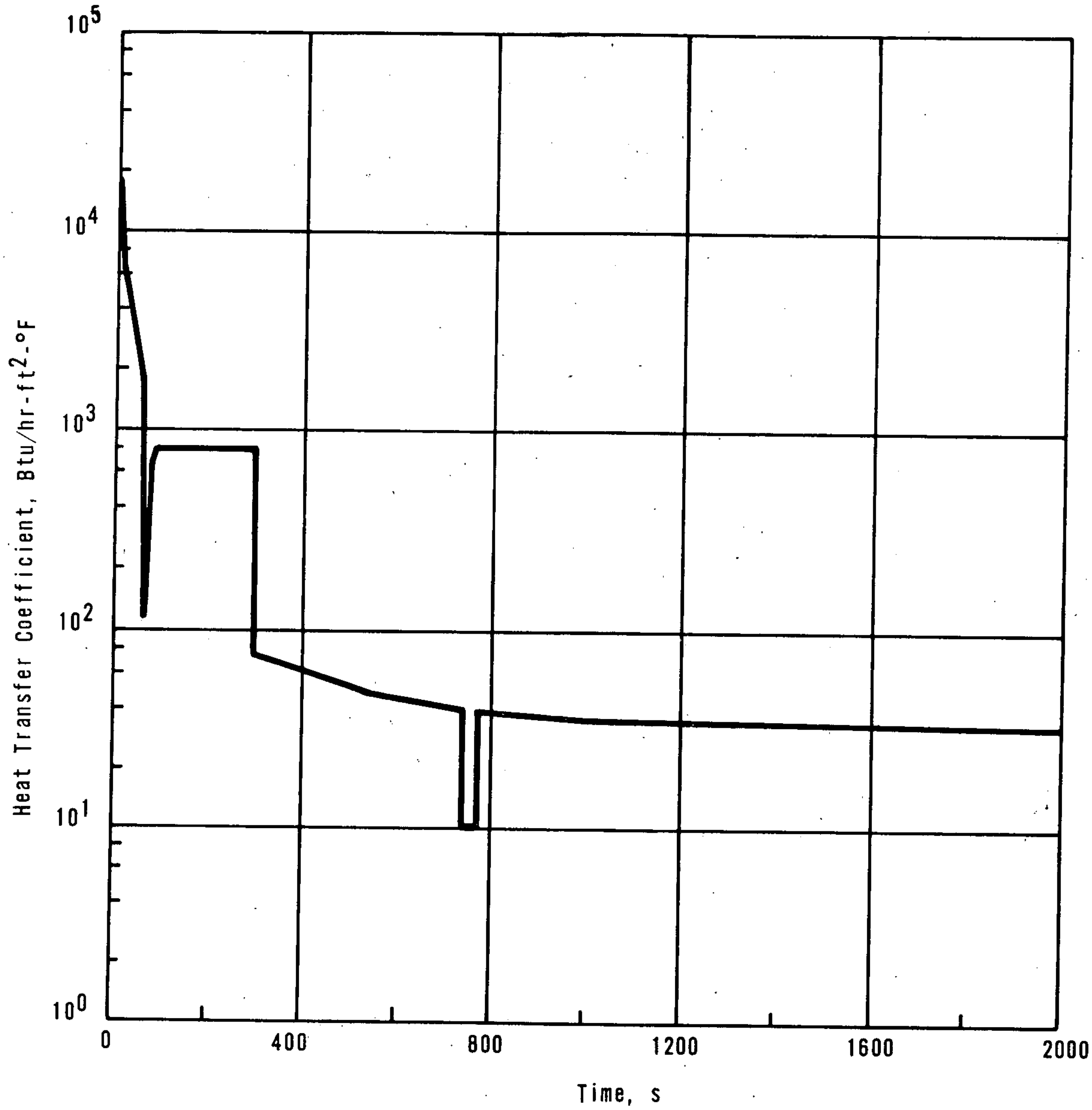


FIGURE 4-24 CASE 2, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME

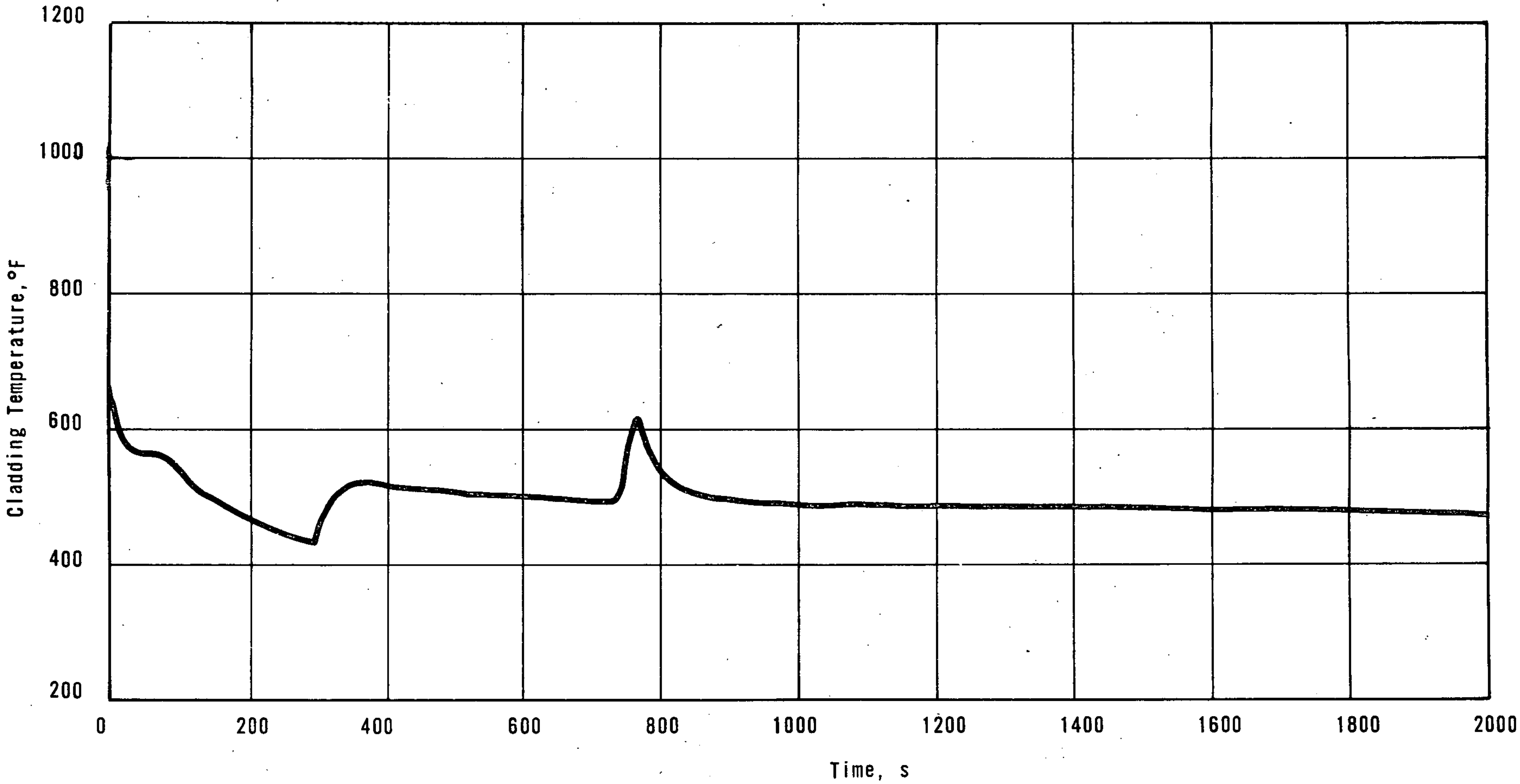


FIGURE 4-25 CASE 2, LEVEL 10 - SINK TEMPERATURE VERSUS TIME

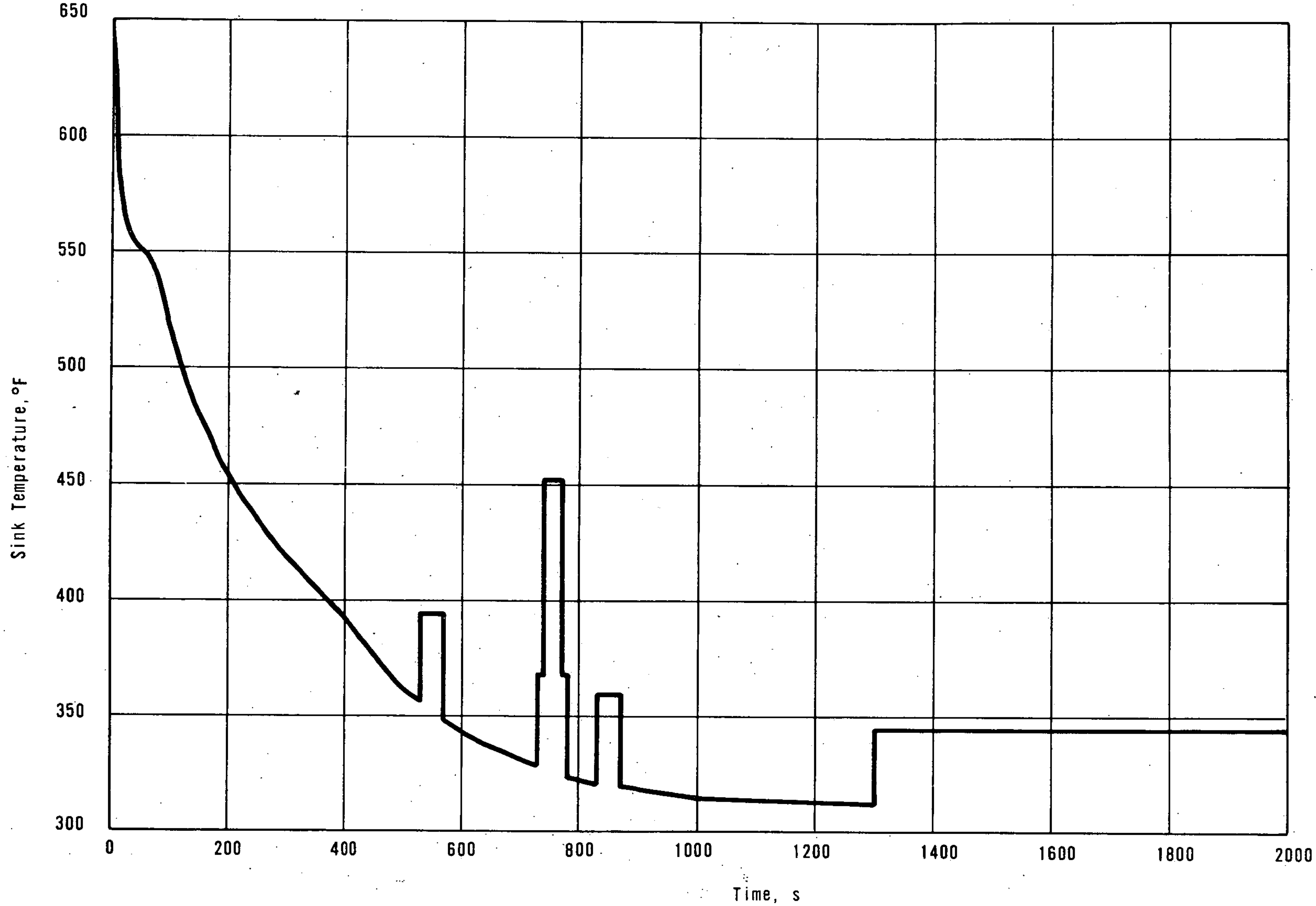


FIGURE 4-26 CASE 2, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

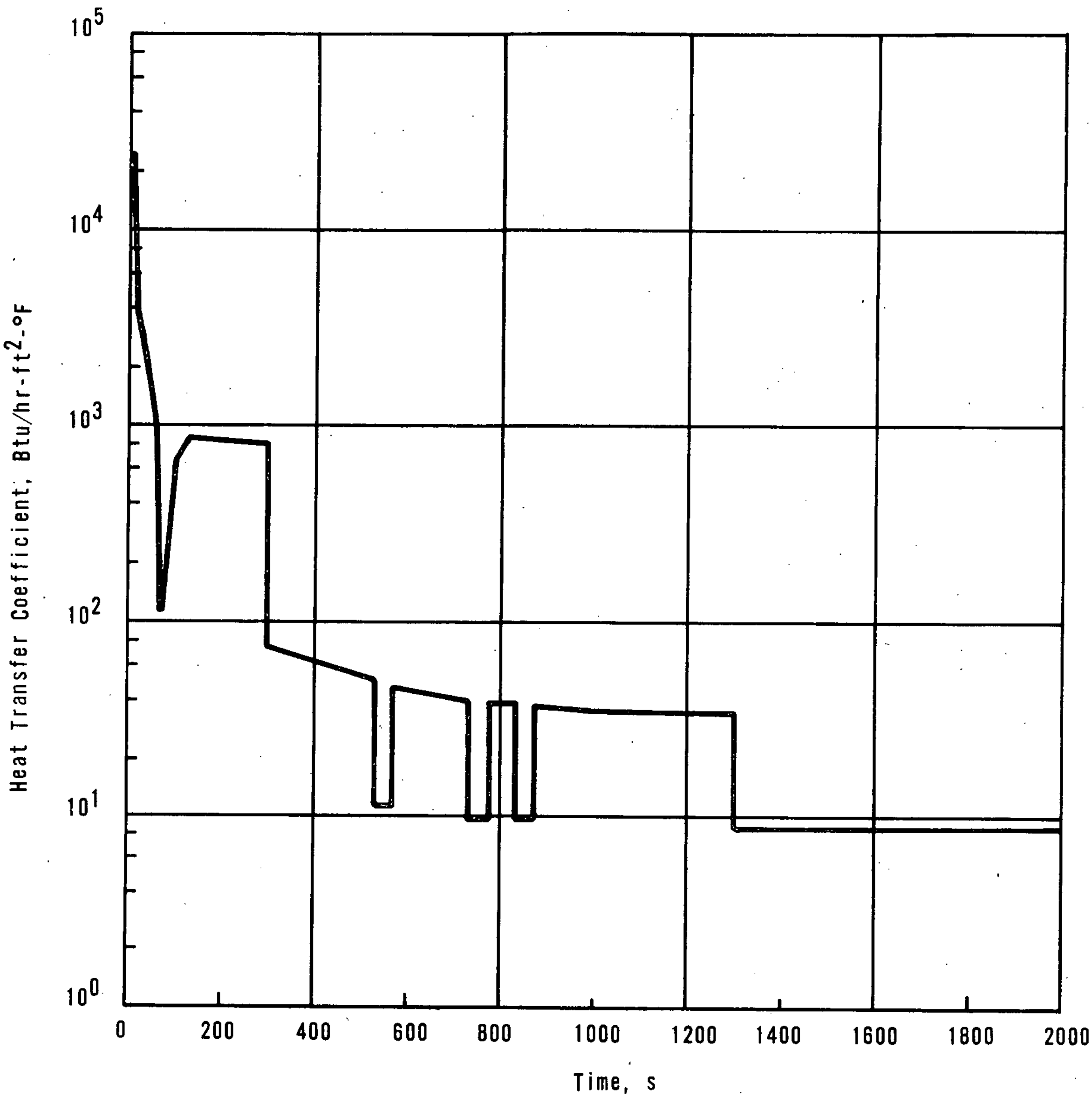


FIGURE 4-27 CASE 2, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME

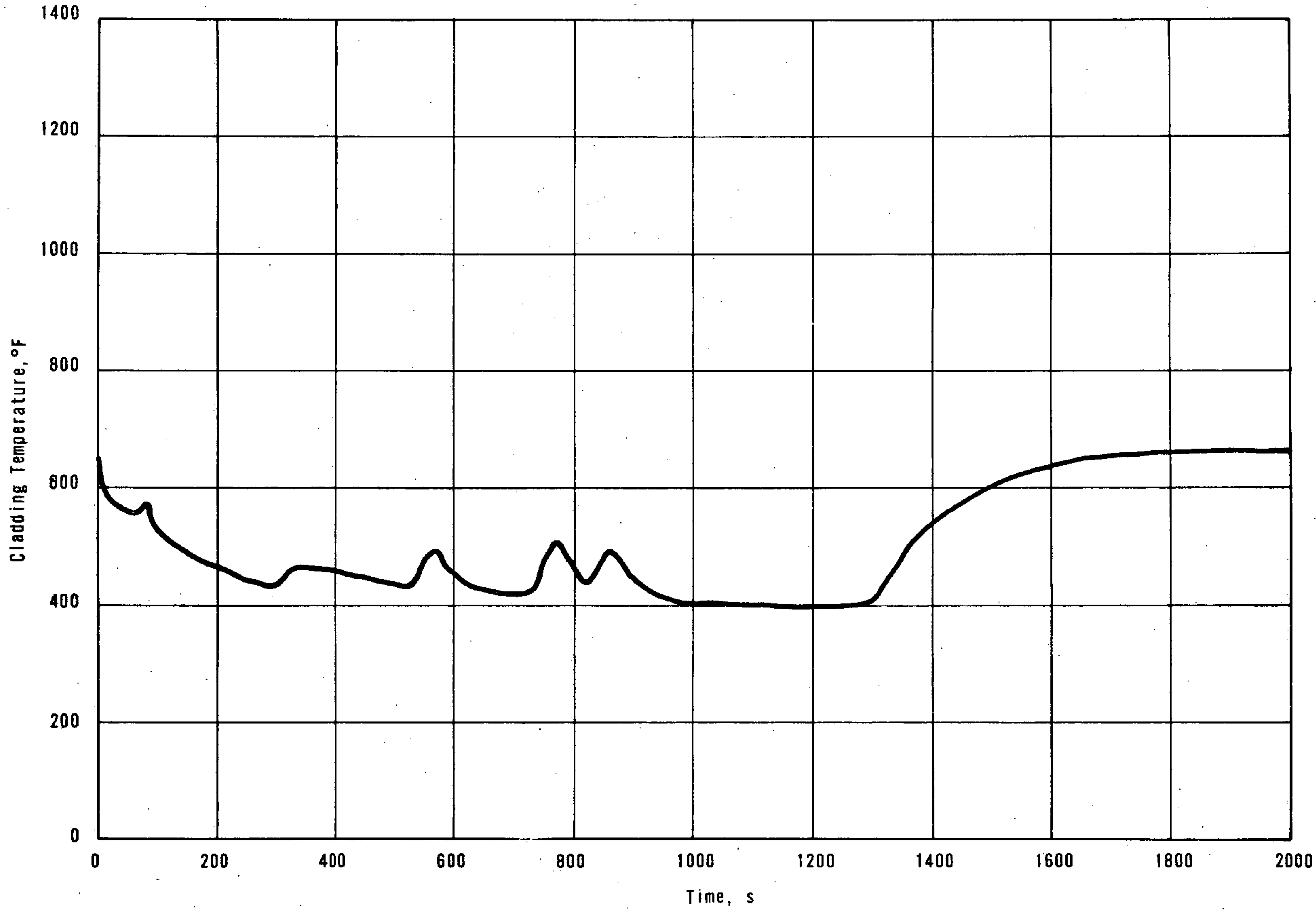


FIGURE 4-28 AXIAL POWER SHAPE FOR CASE 3

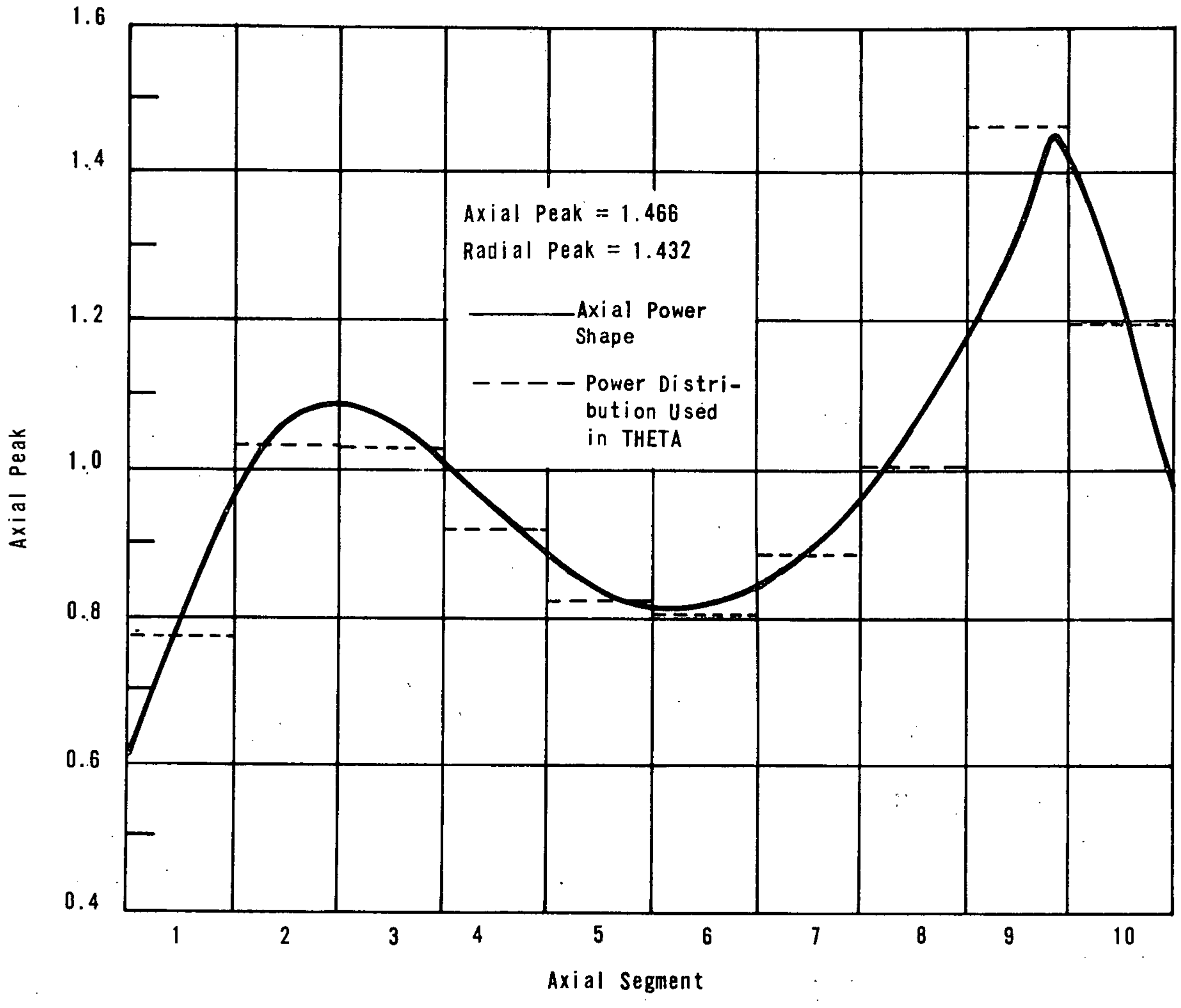


FIGURE 4-29   CASE 3, LEVEL 9 - SINK TEMPERATURE VERSUS TIME

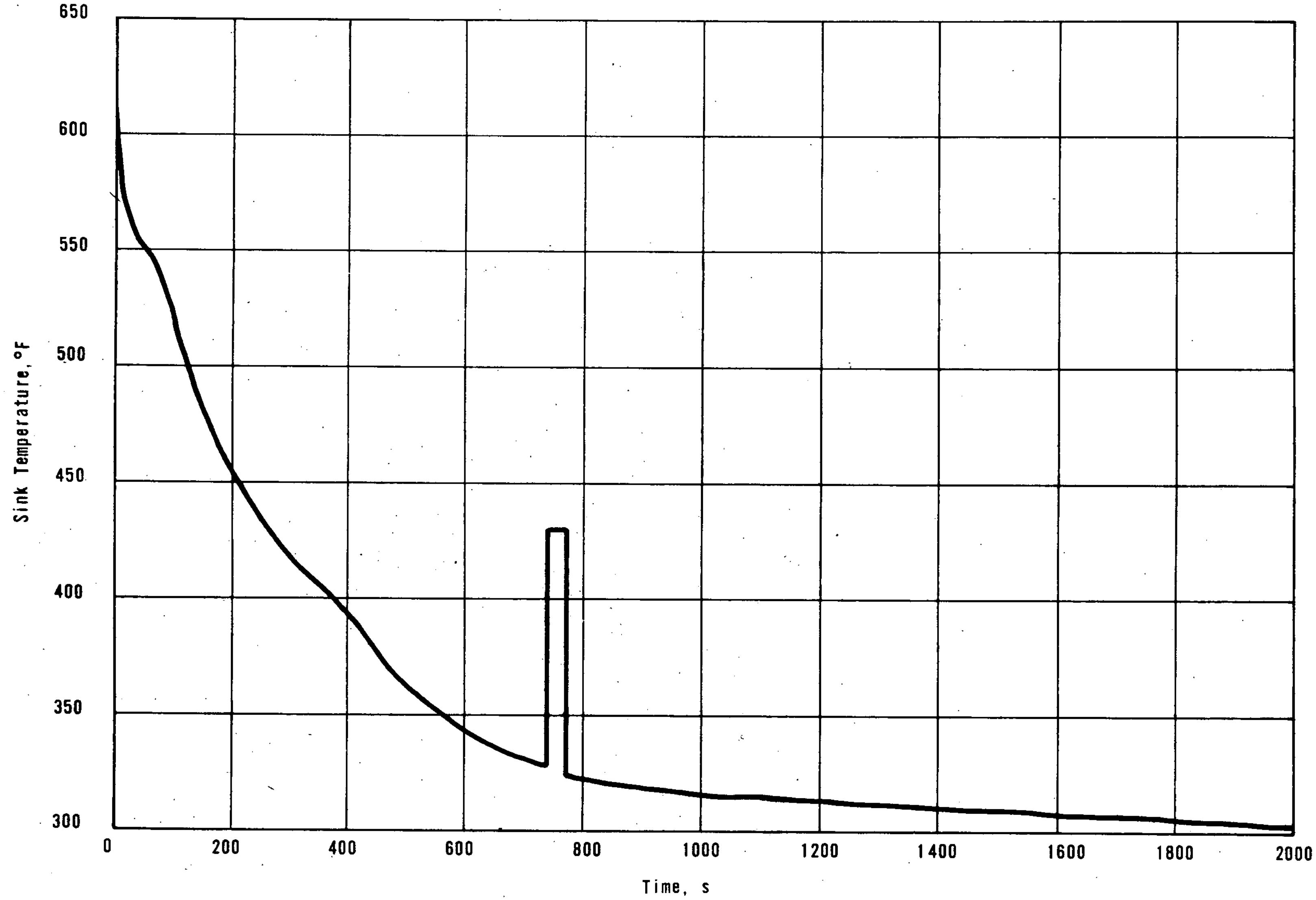


FIGURE 4-30 CASE 3, LEVEL 9 - HEAT TRANSFER COEFFICIENT VERSUS TIME

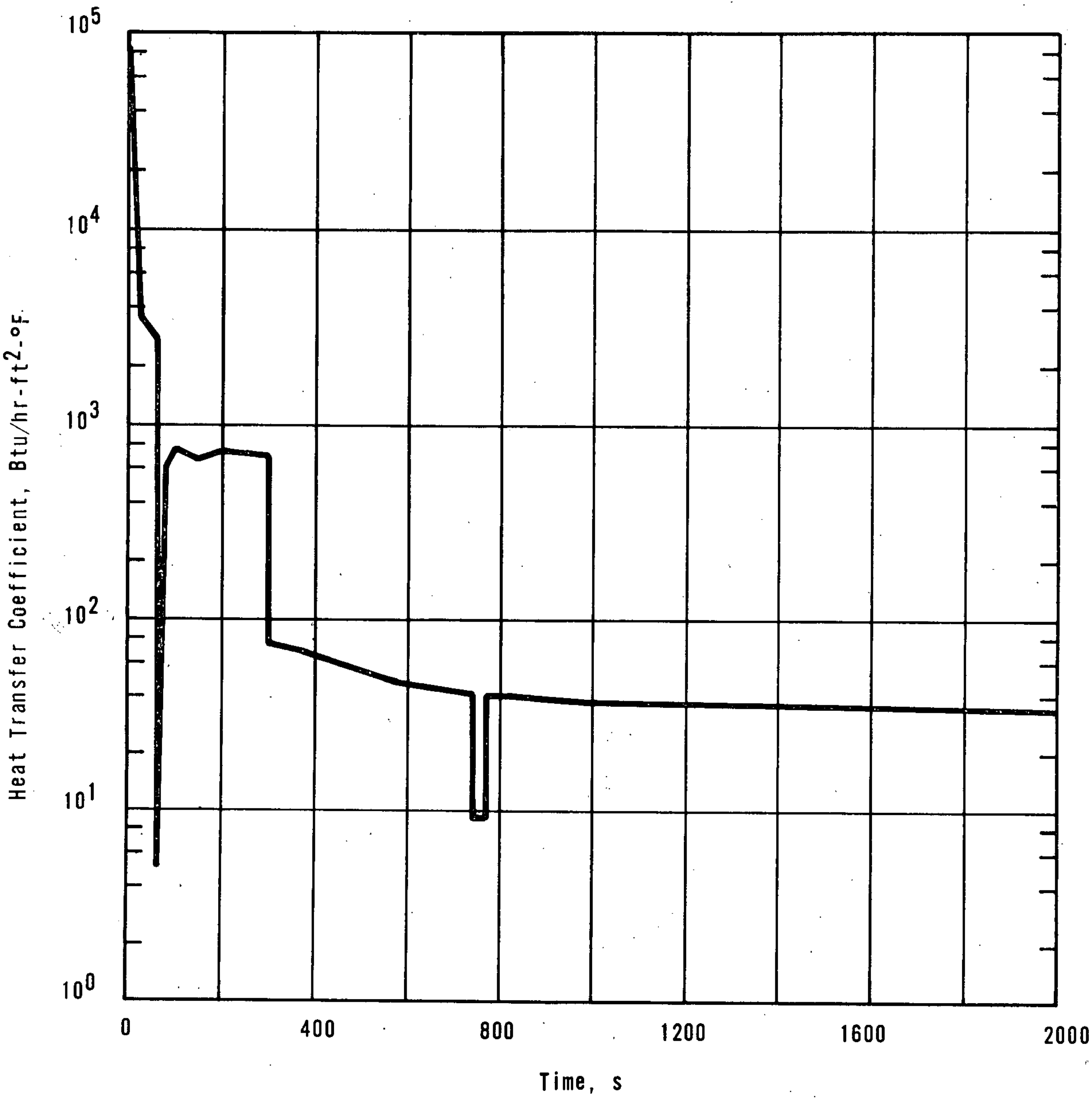


FIGURE 4-31 CASE 3, LEVEL 9 - CLADDING TEMPERATURE VERSUS TIME

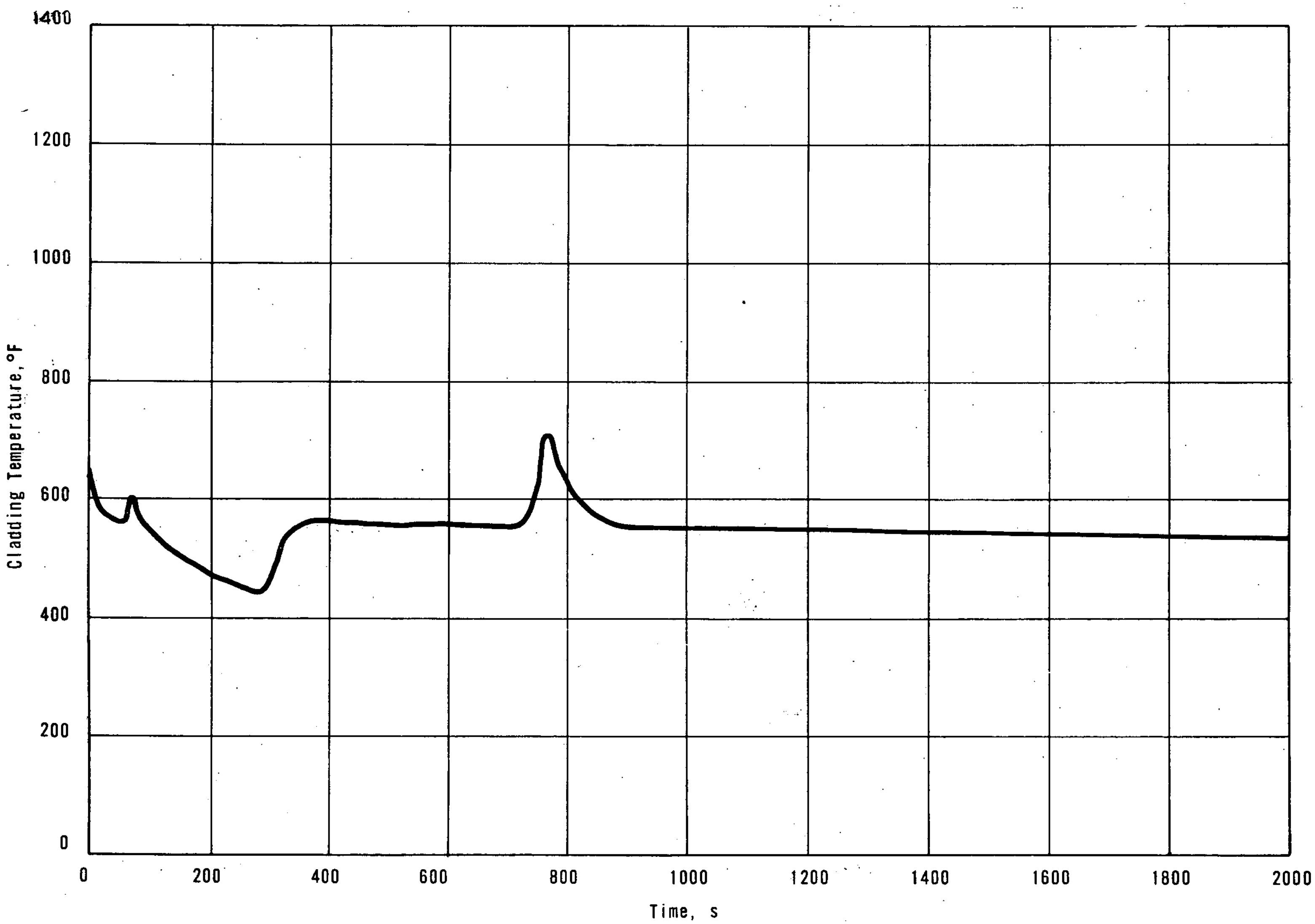


FIGURE 4-32 CASE 3, LEVEL 10 - SINK TEMPERATURE VERSUS TIME

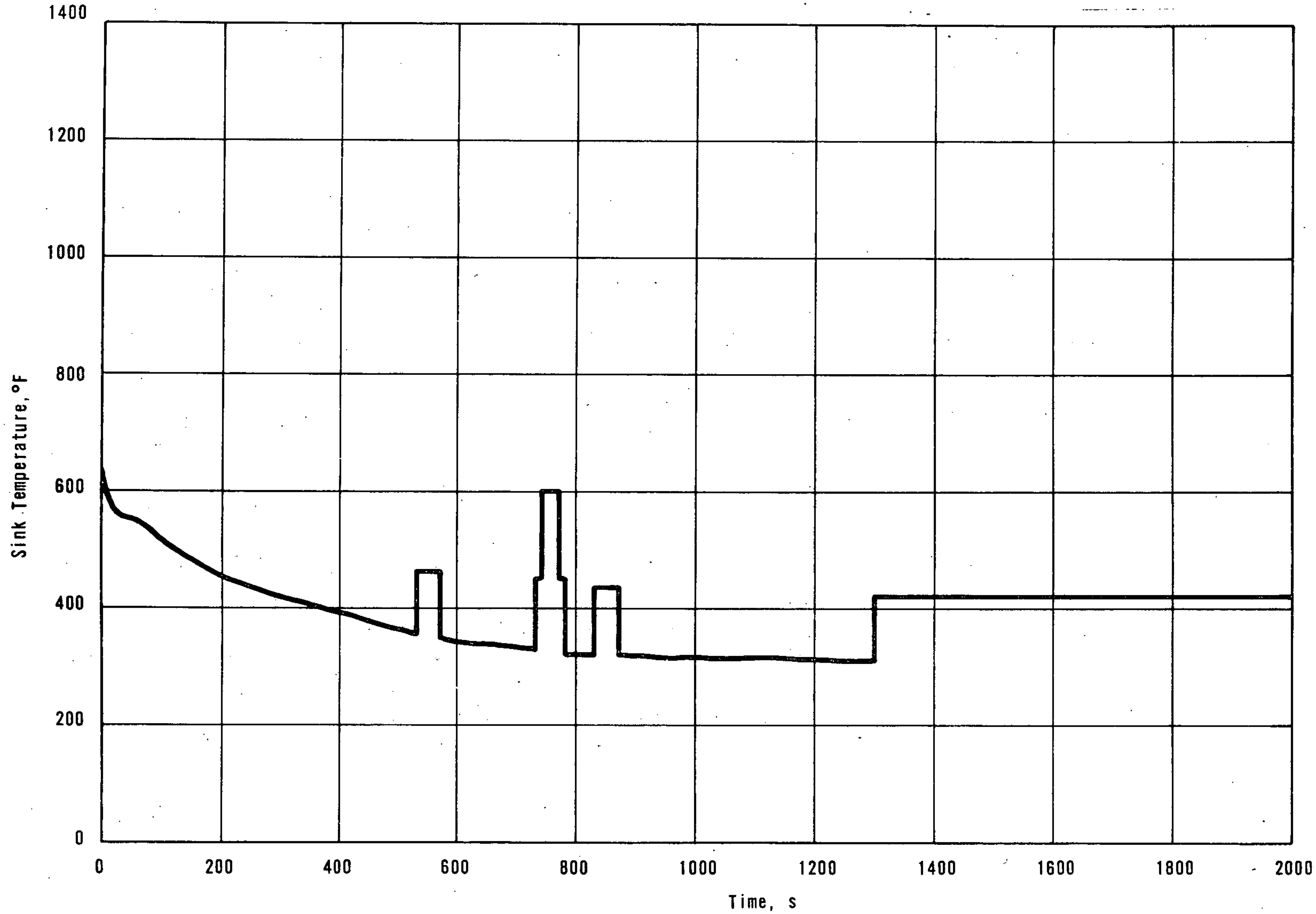


FIGURE 4-33 CASE 3, LEVEL 10 - HEAT TRANSFER COEFFICIENT VERSUS TIME

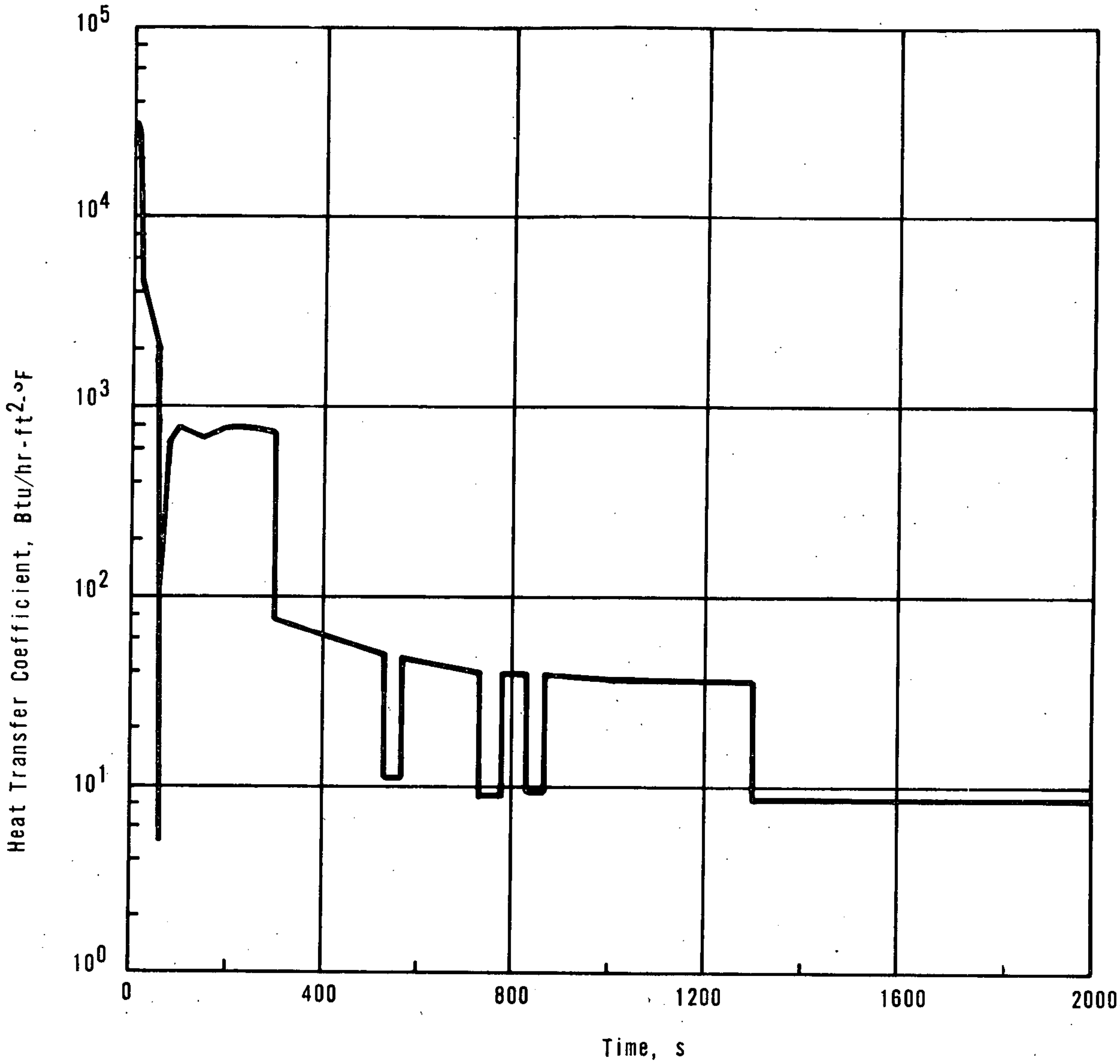
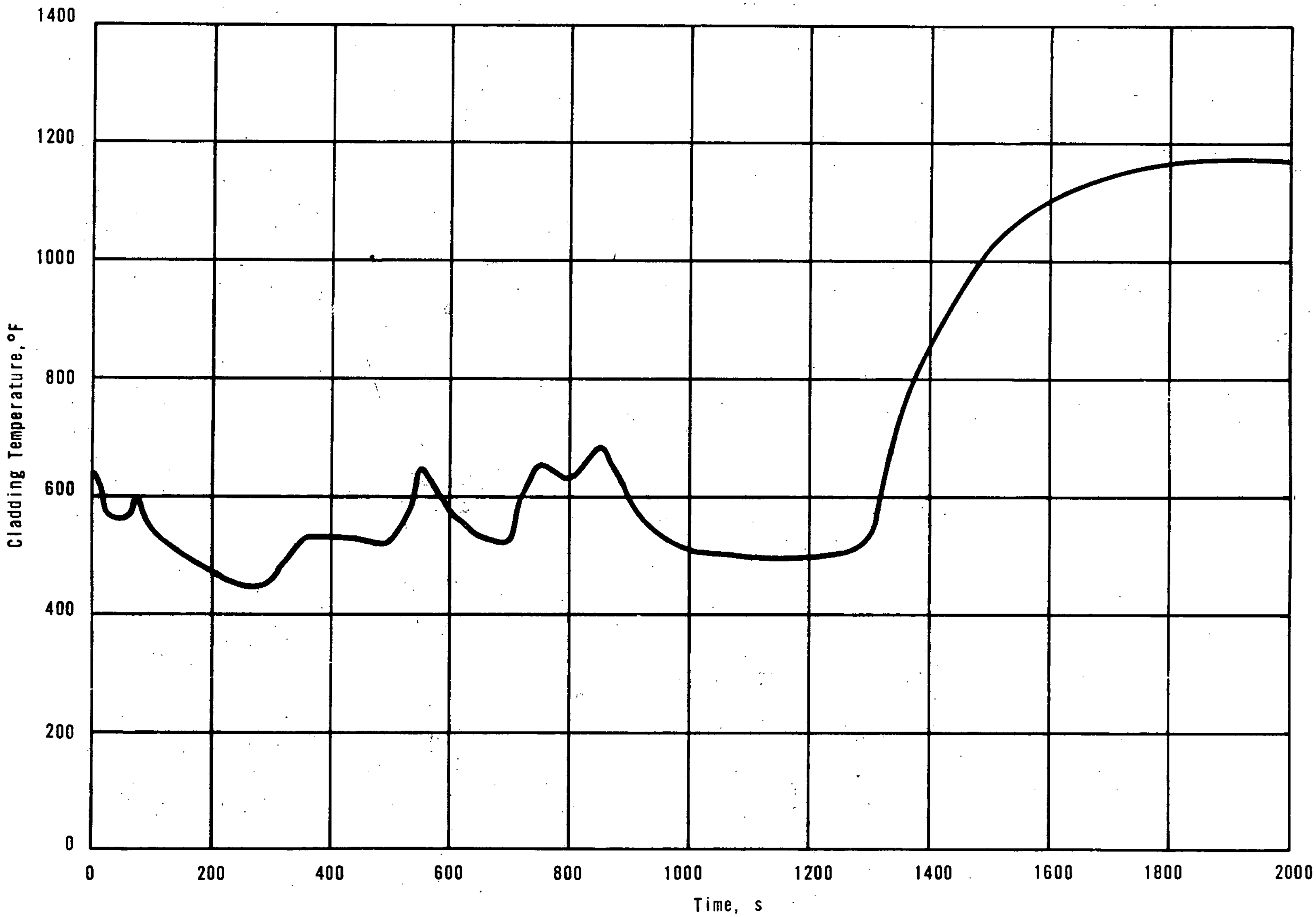


FIGURE 4-34 CASE 3, LEVEL 10 - CLADDING TEMPERATURE VERSUS TIME



ATTACHMENT III

Previous analyses have shown that a marginal amount of water is maintained in the core following a core flooding tank line break. This is due to a partial degrading of the ECC System as a natural consequence of the accident. In order to improve the safety of the design, a flow limiting insert is to be placed in the core flooding tank nozzle to control the accident. By limiting the violence of the blowdown, less reactor coolant system water will be ejected from the primary system. Thus, more water will be maintained in the core during and after blowdown.

The acceptability of the insert is dependent on three points. First, it must be shown that it can be built and installed to function as designed. This has been addressed in Attachment 1. Second, it is necessary to show that the insert as designed will produce the desired results. This has been addressed in Attachment II. Finally, it is necessary to show that the insert will not jeopardize the performance of the ECC System during other accidents. This is addressed in this attachment.

By standard methods of analysis, a k-factor has been determined for the insert. The value of this factor is 0.2 based upon an area of 0.7213 ft^2 and is used to solve for the pressure loss Δp in the following equation:

$$\Delta p = \frac{k|W|W}{288\ \rho g\ A^2}$$

where

W = flow rate, Lbm/sec

ρ = density, Lbm/ft^3

g = gravitional constant, $\frac{32\ \text{Lbm ft}}{\text{Lbf sec}^2}$

A = Area, ft^2

The k-factor for the core flooding line resistance used by B&W in the evaluation of LOCA's presented in BAW-10034 was 6.3. This value is typical of all plants of this type. The proposed insert would increase the resistance by only 3%. To show that such an increase is acceptable, an analysis of the worst case large break, an 8.5 ft^2 cold leg split, has been carried out for two different k-factors. The first value, k = 8.3, has an increase (6.3 + 2.0) an order of magnitude higher than the proposed insert. This was chosen before the insert had been designed in order to bound the result. The second value, k = 5.5, is based on an experimental measurement of the line k-factor without the insert, k = 4.8, plus a conservative evaluation of the insert effect, k insert = 0.7. With a more concrete design and a better evaluation of the effect of the insert, we expect the actual line resistance to be k = 4.8, plus k insert = 0.2 or k = 5.0.

The results of the two different k-factors are shown in Figures 1 and 2. Figure 1 shows the core flooding tank injection rate for two tanks. Figure 2 shows the resulting peak cladding temperatures. Both temperatures are acceptable and are within the AEC Interim Acceptance Criteria. These results show that there is no adverse effect of the insert for large breaks.

For small breaks, the core flooding tanks provide water at a very slow rate. Thus, the important parameter in the core flooding tank system is its pressure volume relationship and not line resistance. An increase of only 3% in CFT line resistance would have no effect on small breaks.

FIGURE 1 CORE FLOODING TANK INJECTION RATE DURING AN 8.5 FT^2 COLD LEG LOCA FOR VARIOUS FLOW RESISTANCES

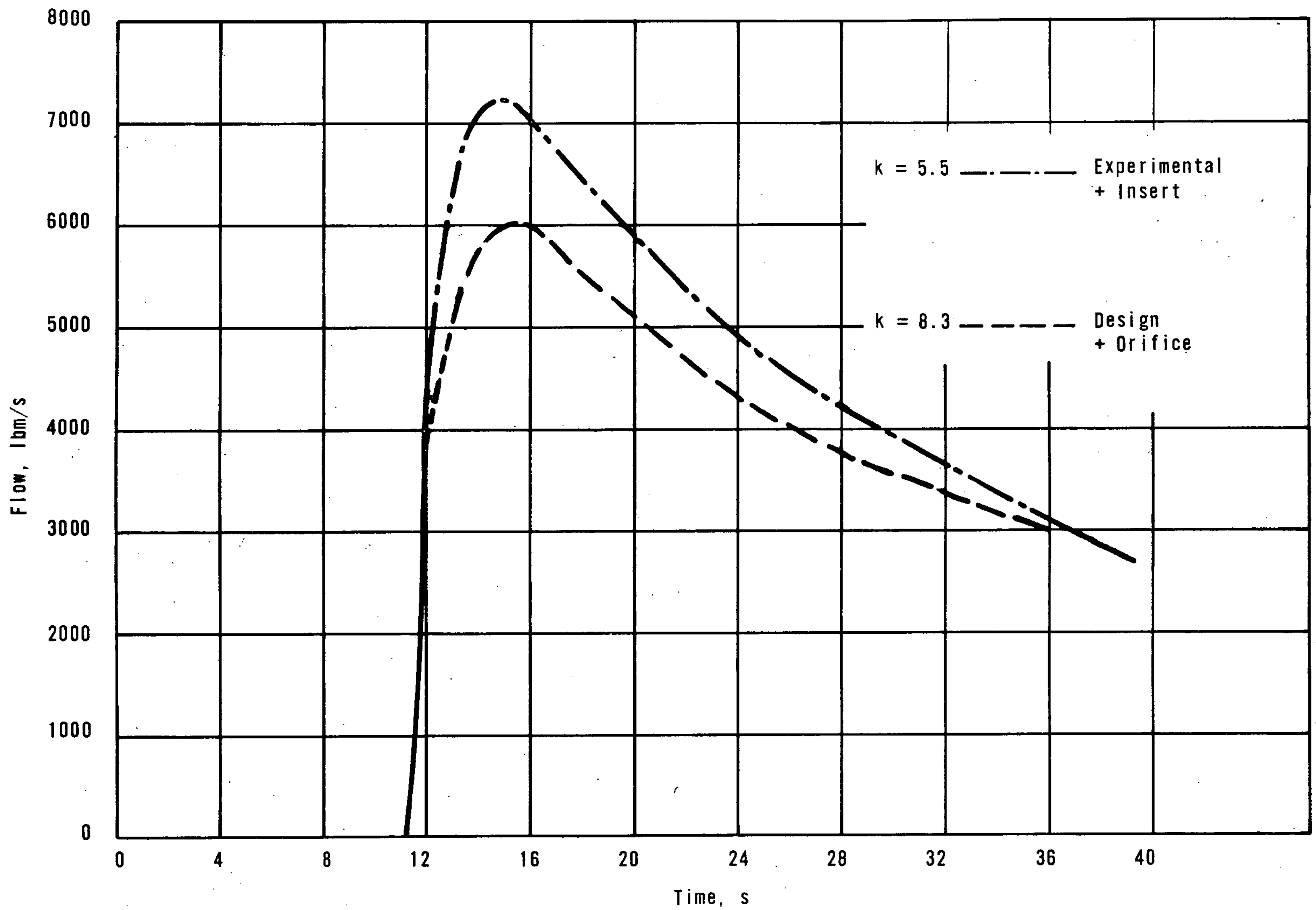


FIGURE 2 HOT SPOT CLADDING TEMPERATURE DURING AN 8.5 FT^2 COLD LEG LOCA FOR VARIOUS FLOW RESISTANCES IN THE CORE FLOOD TANK LINE