Proposed change #117
trans w/ltr dtd 7-31-74

411 LOSS-OF-COOLANT ACCIDENT

GENERAL

A loss-of-coolant accident is defined as a rupture of the main coolant system piping which results in interruption of the normal mechanism for removing heat from the reactor core. The Yankee Nuclear Power Station is equipped with a safety injection system (see Section 212) which assures adequate protection for the core in the unlikely event of a loss-of-coolant accident by providing injection flow of borated water to cool the core and maintain a shutdown condition.

Should a major break occur, depressurization of the main coolant system results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer or main coolant system low pressure trip setpoint is reached. A safety injection system signal is actuated when the appropriate setpoint is reached. Safety injection system actuation is also provided by a high containment pressure signal. These countermeasures will limit the consequences of the accident in two ways:

(1) Reactor trip and borated water injection complement void formation in causing the rapid reduction of power to a residual level corresponding to delayed fission and fission product decay heat.

(2) Injection of borated water provides heat transfer from core and prevents excessive clad temperature.

The safety injection system consists of [illegible] safety injection storage tank, three low pressure injection pumps, one accumulator with its nitrogen supply, three high pressure injection pumps and associated pipes, valves and instrumentation.

Safety injection initiation is automa[illegible] low main coolant or low pressurizer pressure generates a safety [illegible]ction and alarm signal (SIAS). The low pressure safety injection (LPSI) pumps take suction from the safety injection storage tank and discharge to the LPSI header and/or to the high pressure safety injection (HPSI) suction header. The HPSI pumps take suction from the discharge of the LPSI pumps and discharge to the HPSI header.

Safety injection flow enters the main coolant system loops through four lines branching from the safety injection header in the vapor container. Flowmeters are provided in each of these lines. Excessive flow in any one of the four lines will indicate to the operator the possibility of a break in that line. The operator could then isolate that line with either one of the redundant remotely operated branch block valves and continue safety injection through the other three lines.

The accumulator is a passive component which is pressurized to an assumed minimum of 420 psia with nitrogen from the high pressure N_2 storage bottles. During normal operation, the accumulator is separated from main coolant system

8011060658

pressure by several check valves. Flow from the accumulator starts automatically when the main coolant system pressure falls below the pressure of the accumulator. A constant pressure is maintained in the accumulator by the gas pressure regulating valves. After sufficient volume has flowed from the accumulator, a level control signal vents the nitrogen through the pilot operated relief valves.

All active components of the safety injection system are redundant. The system will perform as required after the failure of any single active component and a loss of normal AC power.

Safety injection system power is furnished from three 480 volt emergency busses. Each bus is powered by a 500 KVA diesel generator if normal AC power is lost. Each bus powers one LPSI pump and one HPSI pump.

Where a valve may be required to close during the course of an accident, there are two valves in series so that flow will be stopped if only one valve operates. Similarly, where a valve is required to open, there are two valves in parallel and each is sized for 100 percent flow (for example, the accumulator vent valves or the nitrogen regulators).

In order to evaluate the loss-of-coolant accident, a range of break sizes from small leaks up to a complete double-ended severance of a main coolant pipe has been considered.

METHOD OF ANALYSIS AND RESULTS

The analysis specified by 10CFR50.46[5], "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors", is presented in this section. The analytical techniques used are in compliance with Appendix K of 10CFR50 and are described in the topical report, "Westinghouse ECCS Evaluation Model - Summary"[6]. The exceptions to the assumptions described in Reference 6 are:

(1) The rated power level is increased by 3 percent to allow for instrumentation errors.

(2) The fraction of the locally generated gamma energy that is deposited in the fuel and cladding is 0.87, based on calculations[1] for the Yankee core (which utilizes cruciform control rods).

(3) The worst single failure is determined to be the failure of a diesel to start following the postulated loss of offsite power.

The method of analysis used for the larger size breaks is somewhat different than the method used for the smaller size breaks.

Small Break Analysis

Ruptures of small cross section will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the main coolant system through the postulated break against the charging system makeup flow at maximum operating main coolant system pressure, i.e., 2090 psia. A makeup flow rate from the charging system is typically adequate to sustain pressurizer level at 2090 psia for a break through a 3/8 inch diameter hole. This break results in a loss of approximately 7.2 lb/sec. Should a larger break occur, the inability of the charging system to maintain pressurizer level results in a depressurization of the main coolant system.

Prior to the break, the plant is at equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from decay, hot internals and the vessel continues to be transferred to the main coolant system. The heat transfer between the main coolant system and the secondary system may be in either direction depending on the relative temperatures. In the case of continued heat addition to the secondary, steam pressure increases and steam relief through the dump or safety valves may occur. Make-up to the secondary side is provided by the auxiliary feedwater system. The secondary flow aids in the reduction of main coolant system pressure. When the main coolant system depressurizes to 420 psia, the accumulator begins to inject water into the main coolant loops. The main coolant pumps are assumed to be tripped at the initiation of the accident and the effects of pump coastdown are included in the blowdown analyses.

For small breaks (less than 1.0 ft^2) the WFLASH[(2)] digital computer code is employed to calculate the transient depressurization of the main coolant system as well as to describe the mass and enthalpy of flow through the break. The WFLASH program used in the analysis of the small break loss of coolant accident is an extension of the FLASH-4[(3)] code developed at the Westinghouse Bettis Atomic Power Laboratory. The WFLASH program permits a detailed spatial representation of the main coolant system.

The main coolant system is nodalized into volumes interconnected by flowpaths. The broken loop is modeled explicitly with the intact loops lumped together. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system.

The use of WFLASH in the analysis involves, among other things, the representation of the reactor core as a heated control volume with the associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant transient.

Safety injection flow rate to the main coolant system as a function of the system pressure is used as part of the input. The safety injection (SI) system was assumed to be delivering full flow to the main coolant system 40 seconds after the generation of a safety injection signal.

For these analyses, the safety injection delivery considers the injection flow which is depicted in Figure 411-1 as a function of main coolant system pressure. The 40 second delay includes time required for diesel startup and loading of the safety injection pumps onto the emergency busses. Minimum Emergency Core Cooling System capability and operability has been assumed in these analyses.

Peak clad temperature analyses are performed with the LOCTA IV program which is described in Reference 8.

The small breaks analyzed were 2, 3, and 4 inch diameter ruptures. The worst break size is a 4 inch diameter break. The depressurization transient for this break is shown in Figure 411-2. The extent to which the core is uncovered is shown in Figure 411-3.

During the early part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the main coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The resultant heat transfer cools the fuel rod and clad to very near the coolant temperatures as long as the core remains covered by a two phase mixture.

The maximum hot spot clad temperature calculated during the transient is 1265°F including the effects of fuel densification as described in Reference 4. The peak clad temperature transient is shown in Figure 411-4 for the worst break size; i.e., the break with the highest peak clad temperature. The steam flow rate for the worst break is shown in Figure 411-5. When the mixture level drops below the top of the core, the steam flow computed in WFLASH is used to determine the cooling to the upper portion of the core. The rod film coefficients for this phase of the transient are given in Figure 411-6. The hot spot fluid temperature for the worst break is shown in Figure 411-7 and the normalized core power transient following the accident (relative to reactor scram time) is shown in Figure 411-8.

The reactor shutdown time is equal to the reactor trip signal time pluse 3.2 sec for rod insertion and delay. During this rod insertion period the reactor is conservatively assumed to operate at rated power.

Additional break sizes were analyzed. Figures 411-9A and 411-9B present the main coolant system pressure transient for the 3 and 2 inch breaks respectively and Figures 411-10A and 411-10B present the volume history (mixture height) plots for both breaks. The peak clad temperatures for both cases are less than the peak clad temperature of the

4 inch break. The peak clad temperatures for the 3 inch break is given in Figure 411-11. The peak clad temperature for the 2 inch break is not shown since the core is not uncovered during the transient.

The analyses presented demonstrate that the Emergency Core Cooling System provides sufficient core flooding to keep the calculated peak clad temperatures below the required limits of 10CFR50.46. Hence, adequate protection is afforded by the Emergency Core Cooling System in the event of a small break loss of coolant accident.

Large Break Analysis

At the beginning of the blowdown phase of a major break, the entire main coolant system contains subcooled liquid which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break develops, the time to departure from nucleate boiling is calculated, consistent with Appendix K of 10CFR50[5]. Thereafter, the core heat transfer is unstable, with both nucleate boiling and film boiling occurring. As the core becomes uncovered both turbulent and laminar forced convection and radiation are considered as core heat transfer mechanisms.

When the main coolant system pressure falls below 420 psia the accumulator begins to inject borated water. The conservative assumption is made that accumulator water injected bypasses the core and goes out through the break until the termination of bypass. This conservatism is again consistent with Appenxix K of 10CFR50.

The Emergency Core Cooling System, even when operating during the injection mode with the most severe single active failure meets the Acceptance Criteria[5].

The description of the various aspects of the LOCA analysis is given in WCAP-8339[6]. This document describes the major phenomena modeled, the interfaces among the computer codes and features of the codes which maintain compliance with the Acceptance Criteria. The individual codes are described in detail in separate reports[7-10]. The containment parameters used in the containment pressure code[10] for determination of ECCS backpressure are given in Table 411-2.

Table 411-1 presents the peak clad temperatures and hot spot metal reaction for a range of break sizes. The analysis of the loss of coolant accident is performed at 103 percent of rated power. The peak linear power, and core power used in the analyses are also given in Table 411-1. along with the equivalent core peaking factor at the license application power level. Since there is a margin between the value of the peak linear power density used in this analysis and the value expected in operation, a lower peak clad temperature would be obtained by using the peak linear power density expected during operation.

Figures 411-12 through 411-24 present the transient values of the principle parameters for the break sizes analyzed. These figures present the following information:

411-12	Value of fluid quality at the clad burst location and at the hot spot.
411-13	Mass velocity at the clad burst location and at the hot spot.
411-14	Film heat transfer coefficient calculated by LOCTA IV at the clad burst location and at the hot spot.
411-15	Pressure calculated in the core.
411-16	Break flowrate. For guillotine breaks this is the sum of the flows out both ends of the break.
411-17	Core pressure drop from the lower plenum, near the core, to the upper plenum at the core outlet.
411-18	Clad temperature transient at the hot spot and at the burst location.
411-19	Fluid temperature at the hot spot and burst location.
411-20	Core flowrate (top and bottom).
411-21	Core reflood transient.
411-22	Accumulator flow. Established in the refill-reflood calculations and is the sum of that injected in the intact cold legs.
411-23	Pumped ECCS flow.
411-24	Containment pressure transient.

The clad temperature analysis is based on a total peaking factor of 2.96. The hot spot metal water reaction reached is 8.6 percent, which is well below the embrittlement limit of 17 percent, as required by 10CFR50.46. In addition, the total core metal-water reaction is less than 0.3 percent for all breaks as compared with the 1 percent criterion of 10CFR50.46.

The results of several sensitivity studies are reported in Reference 11. These results are for conditions which are not limiting in nature and hence are reported on a generic basis.

CONCLUSIONS

For breaks up to and including the double ended severance of a reactor coolant pipe:

(1) The calculated peak fuel element clad temperature provides margin to the requirement of 2200°F.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1 percent of the total amount of Zircaloy in the reactor.

(3) The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The cladding oxidation limits of 17 percent are not exceeded during or after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Therefore, the Emergency Core Cooling System will meet the Acceptance Criteria as presented in 10CFR50.46.

REFERENCES

(1) J. N. Hamawi, "Yankee Rowe Core XI Decay Heat Redistribution Factor During Shutdown Conditions", YAEC-1071, June 1974.

(2) V. J. Esposito, K. Kesavan, B. A. Maul, WFLASH - A FORTRAN IV Computer Program for Simulation of Transients in a Multi-Loop PWR, WCAP-8261 Rev. 1, July 1974.

(3) Porsching, T. A., Murphy, J. H., Redfield, J. A., and Davis, V. C., "FLASH-4: A Fully Implicit FORTRAN-IV Program for the Digital Simulation of Transients in a Reactor Plant", WAPD-TM-84, Bettis Atomic Power Laboratory (March, 1969).

(4) J. M. Hellman, "Fuel Densification Experimental Results and Model for Reactor Application", WCAP-8219, October, 1973.

(5) "Acceptance Criteria for Emergency Core Cooling System for Light Water Cooled Nuclear Power Reactors" 10CFR50.46 and Appendix K of 10CFR50. Federal Register, Volume 39, Number 3 January 4, 1974.

(6) Bordelon, F. M., Massie, H. W., and Zordan, T. A., "Westinghouse ECCS Evaluation Model - Summary" WCAP-8399, July 1974.

(7) Bordelon, F. M., et al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant", WCAP-8306, June 1974.

(8) Bordelon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis", WCAP-8305, June 1974.

(9) Kelly, R. D., et al., "Calculational Model for Core Reflooding after a Loss-of-Coolant Accident (WREFLOOD Code)", WCAP-8171, June 1974.

(10) Bordelon, F. M., and Murphy, E. T., "Containment Pressure Analysis Code (COCO)", WCAP-8326, June 1974.

(11) Buterbaugh, T. L., Johnson, W. J. and Kopelic, S. D., "Westinghouse ECCS - Plant Sensitivity Studies", WCAP-8356, July 1974.

TABLE 411-1
LARGE BREAK ANALYSIS
SUMMARY OF RESULTS*

Break Type	DECLG** (CD = 1.0)	DECLG (CD = 0.6)	0.6 x DECLS***	1.0 ft^2 S
Peak Clad Temperature, ^{o}F	2150	2175	2125	1850
Location of Peak, ft	3.8	3.8	3.8	4.1
Local Maximum Zr-H_2O Reaction, %	7.1	8.6	6.5	2.6
Location of Max. Zr-H_2O Reaction, ft	3.8	3.8	3.8	4.1
Total Zr-H_2O Reaction, %	<0.3	<0.3	<0.3	<0.3
Hot Rod Burst Time, sec	81.2	83.6	85.4	101.2
Hot Rod Burst Location, ft	3.8	3.8	3.8	3.8

*Calculations are based on

Reactor Power, Mwt	618
Peak Linear Power, kw/ft	13.06
Peaking Factor at Rated Power	2.96
Accumulator Water Volume, ft^3	700
Fuel Cycle	11
Fuel Region	B

**DECLG - Double-ended cold leg guillotine break.

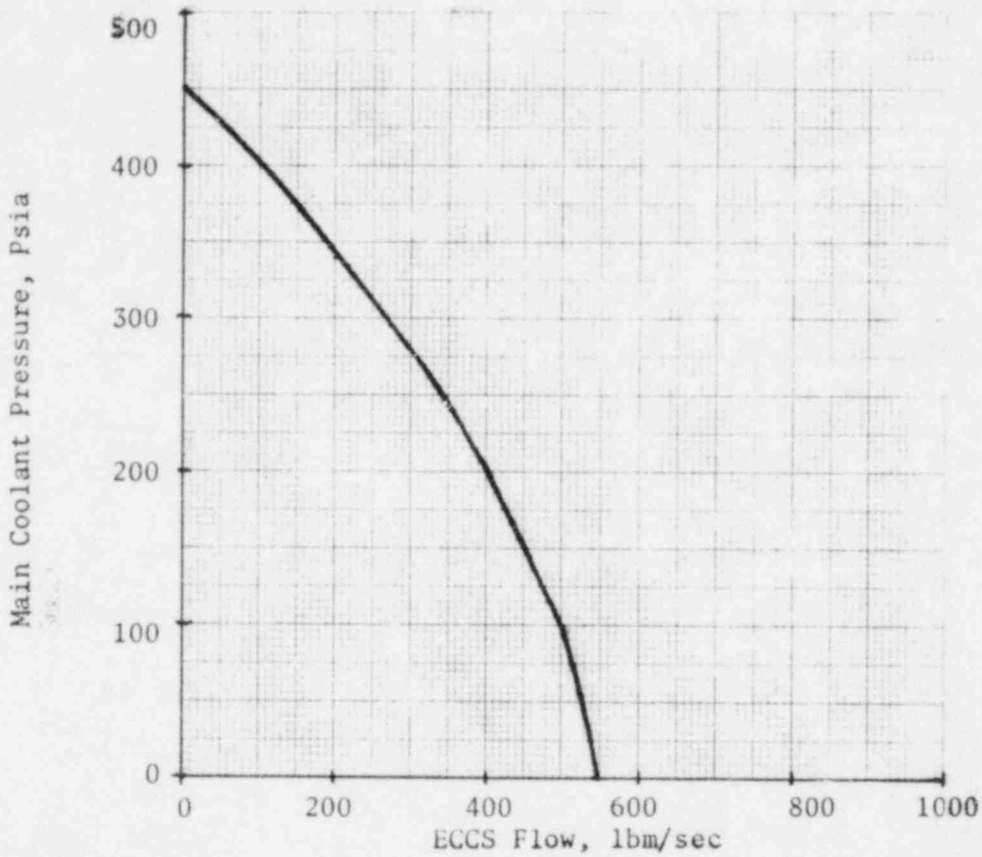
***DECLS - Double-ended cold leg slot break.

411.08
8/2/74
DRAFT

TABLE 411-2
LARGE BREAK ANALYSIS
CONTAINMENT DATA

Net Free Volume	903,000 ft^3
Initial Conditions	
Pressure	14.7 psia
Temperature	90°F
RWST Temperature	40°F
Service Water Temperature	33°F
Outside Temperature	-30°F
Structural Heat Sinks	
Material (thickness)	Area, Ft^2
Concrete (60 in)	50,000
Carbon Steel (0.875 in)	49,000
Carbon Steel (0.055 in)	8,000
Stainless Steel (0.225 in)	10,800

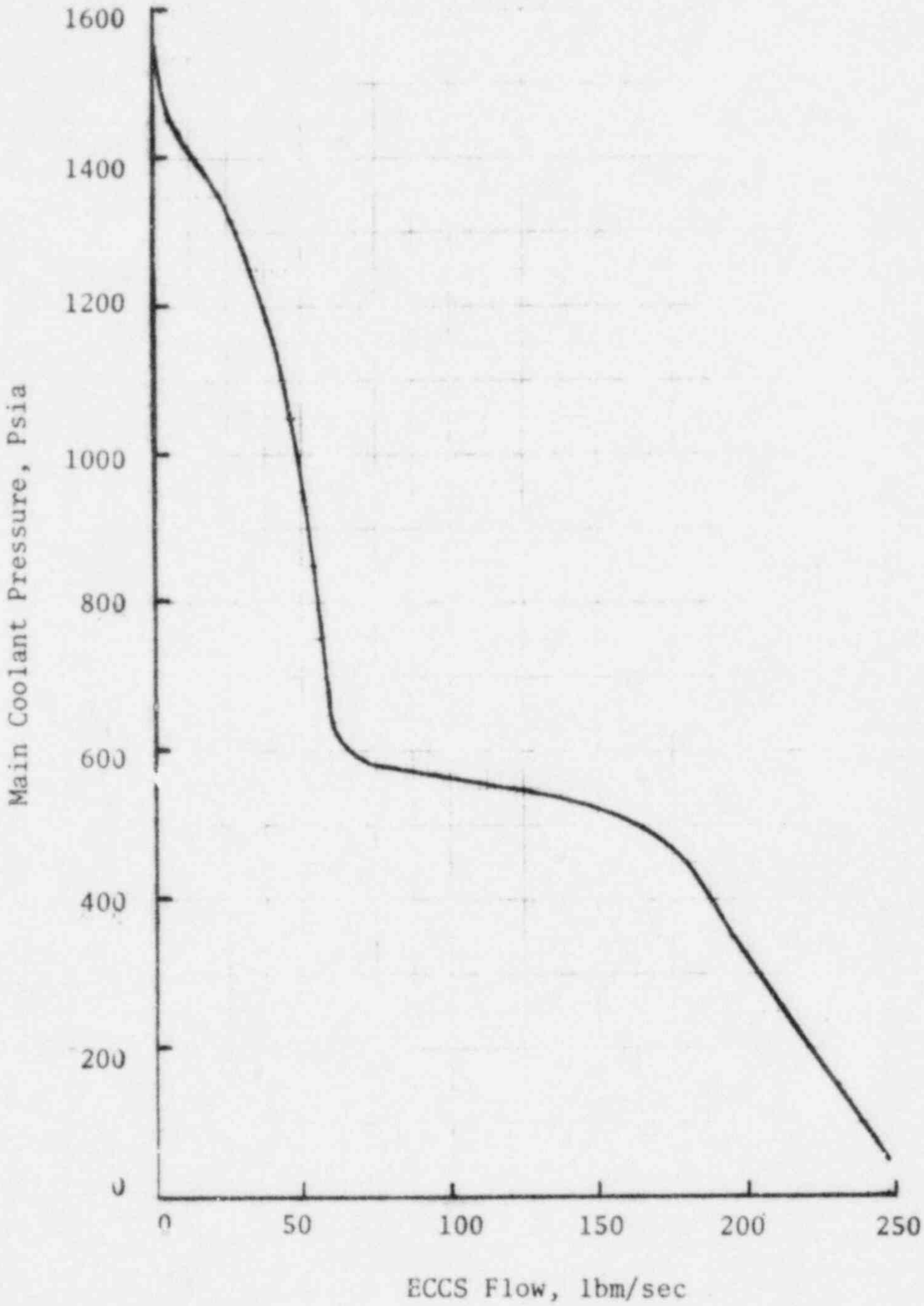
411:10
8/2/74
DRAFT



POOR ORIGINAL

YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW VS. PRESSURE FOUR INCH DIAMETER BREAK	FIGURE 411-1A

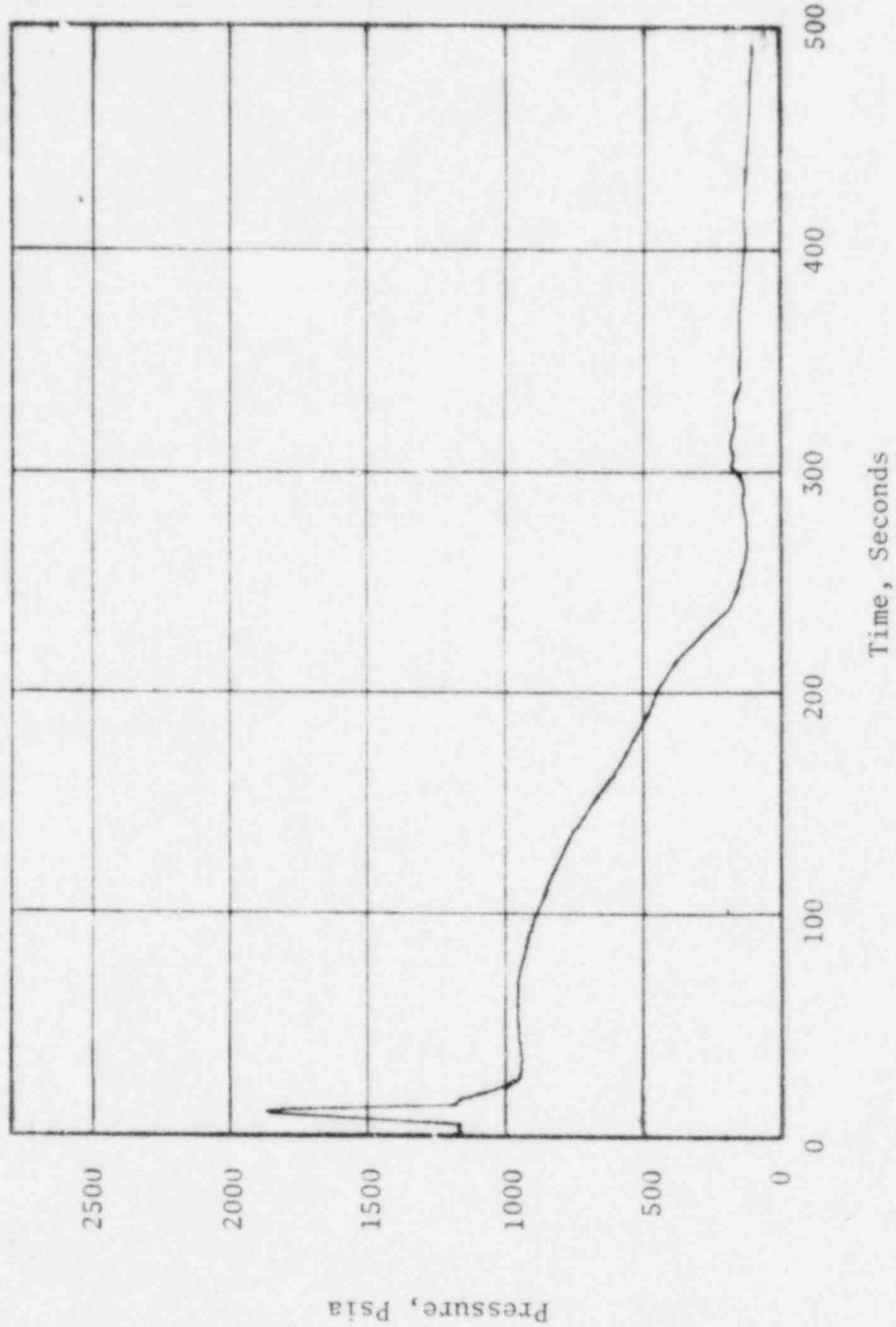
411.11
8/2/74
DRAFT



POOR ORIGINAL

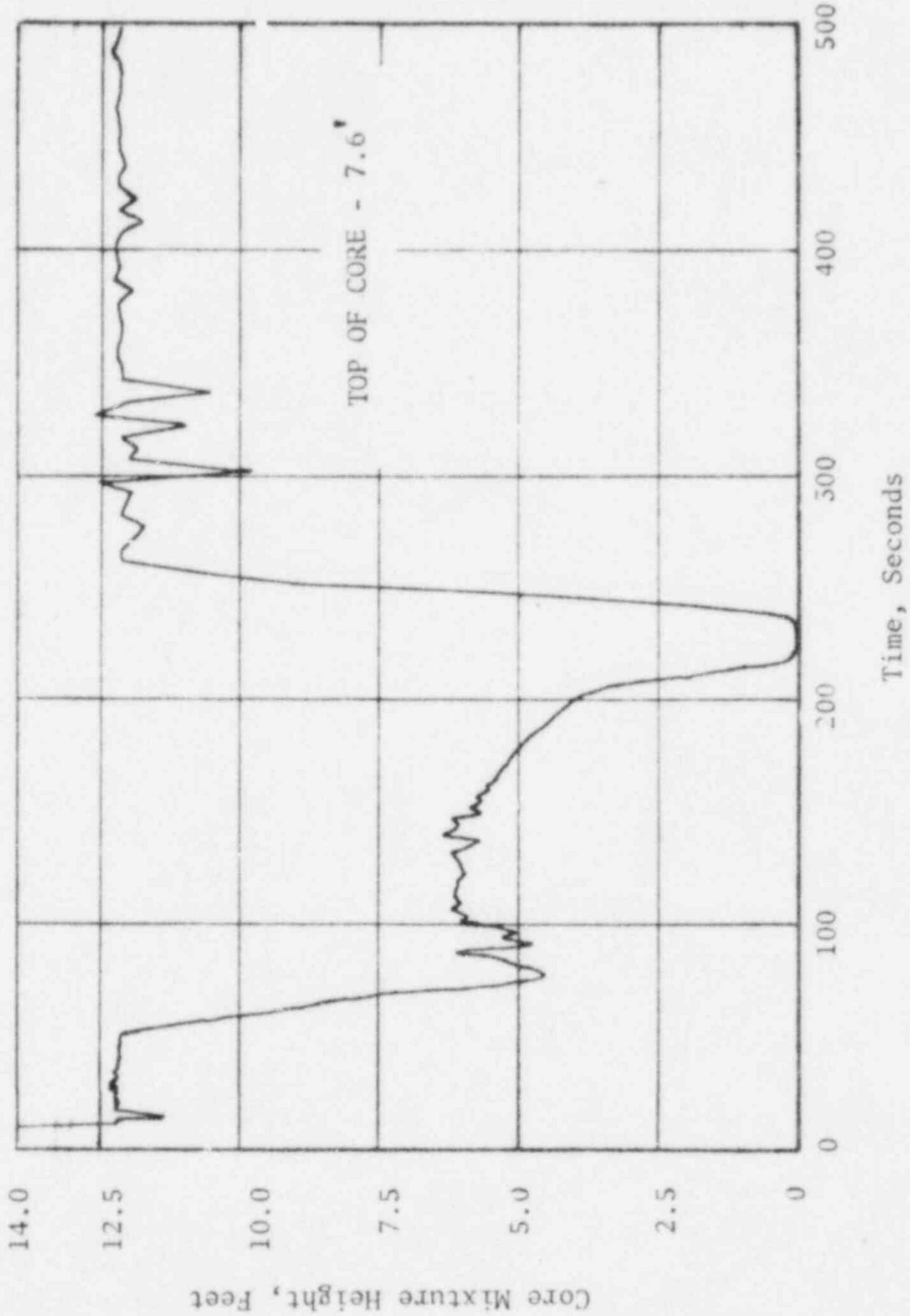
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW VS. PRESSURE TWO AND THREE INCH DIAMETER BREAKS	FIGURE 411-1B

411'.12
8/2/74
DRAFT



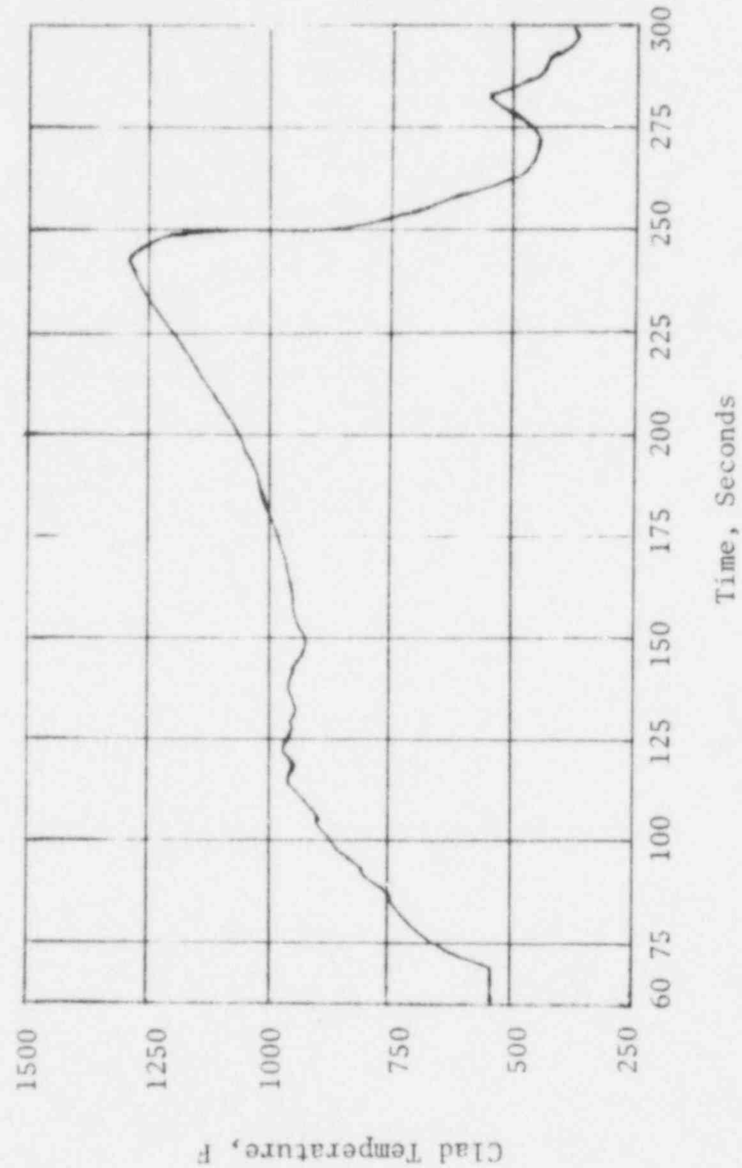
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT MAIN COOLANT PRESSURE VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-2

411:13
8/2/74
DRAFT



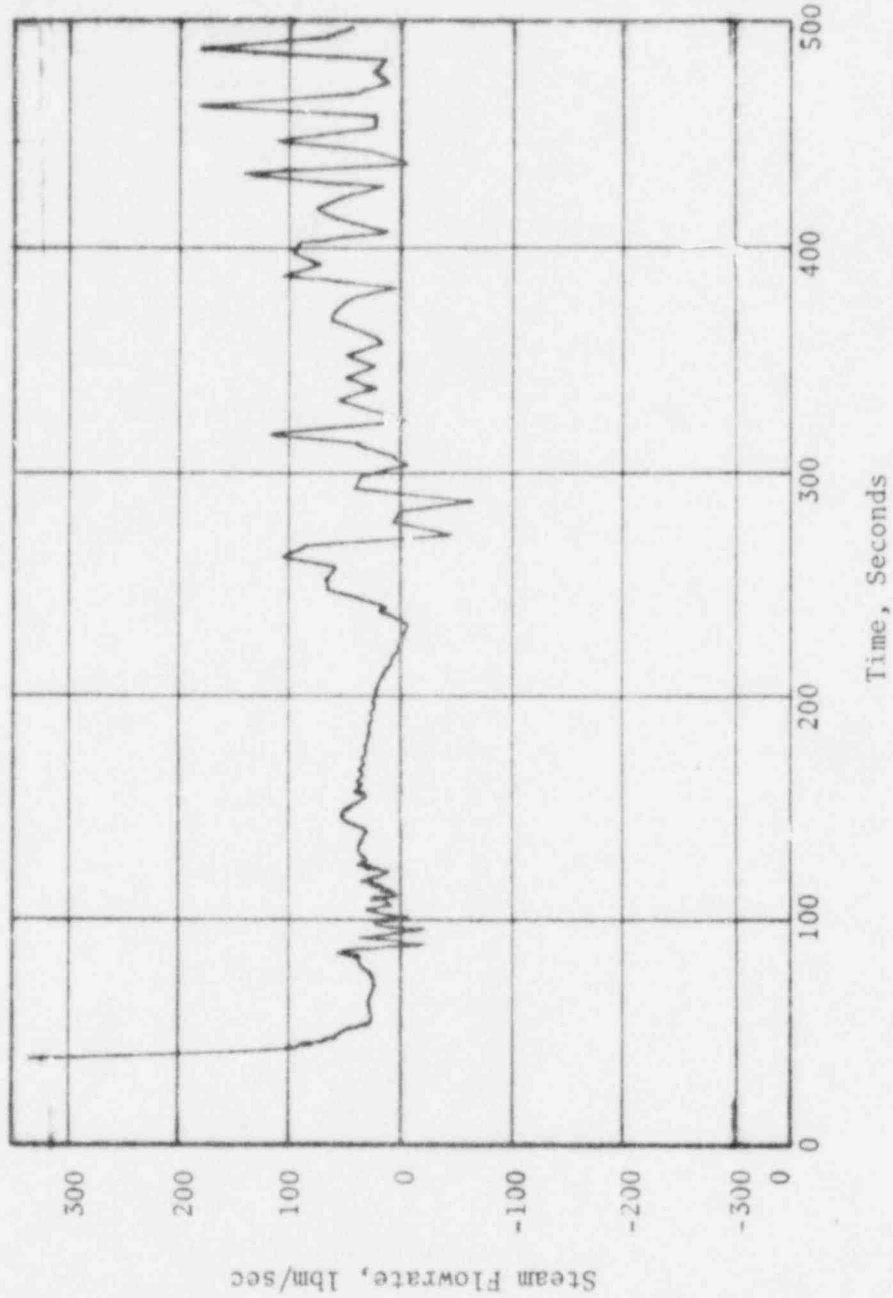
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE COOLANT LEVEL VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-3

411.14
8/2/74
DRAFT



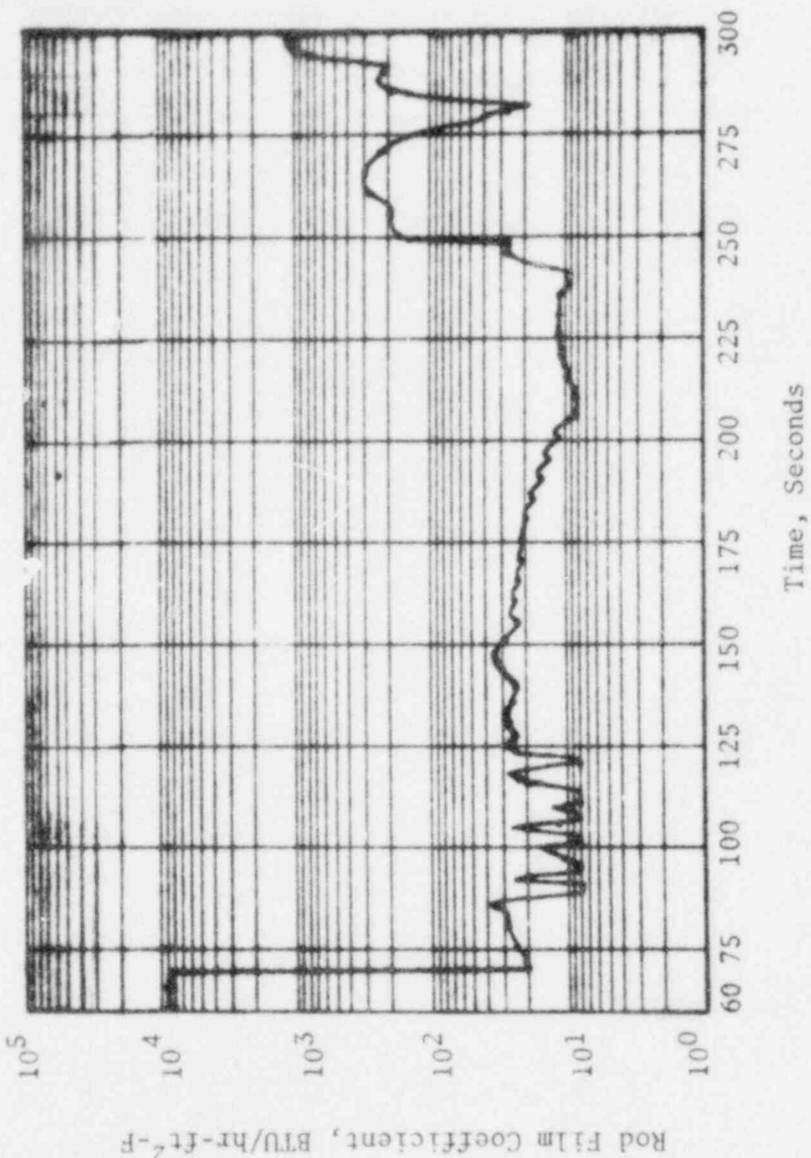
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PEAK CLAD TEMPERATURE VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-4

411'.15
8/2/74
DRAFT

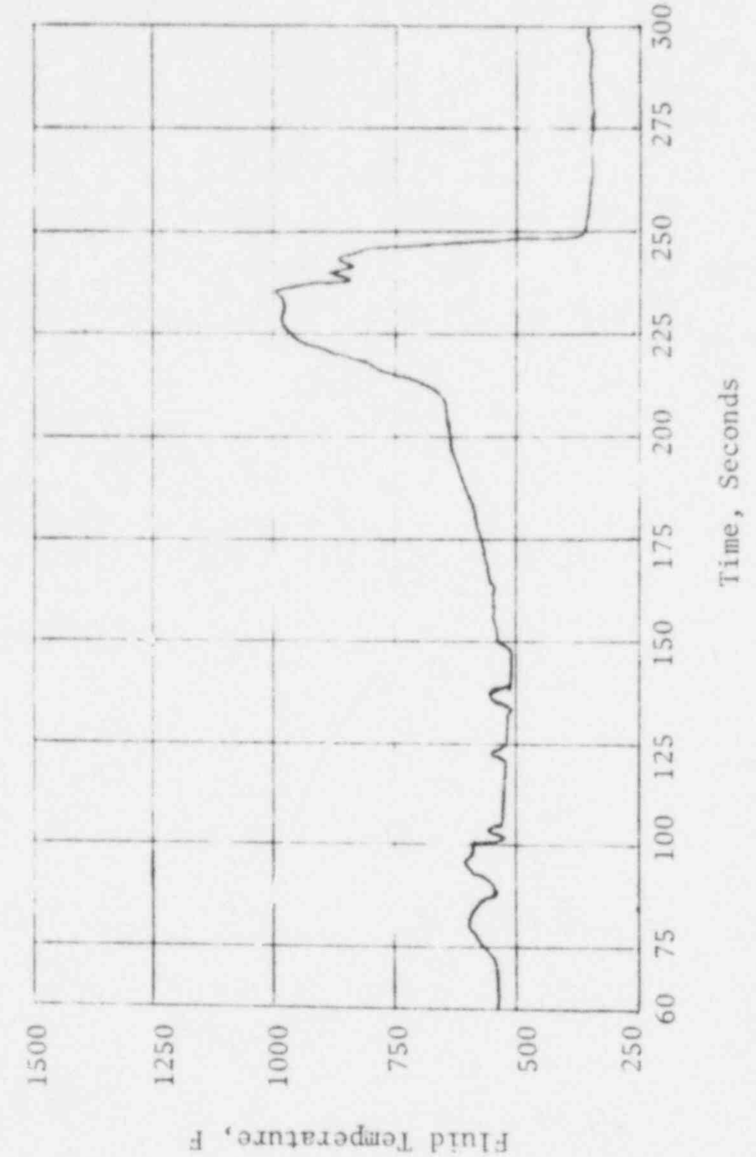


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE STEAM FLOWRATE VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-5

411.16
8/2/74
DRAFT

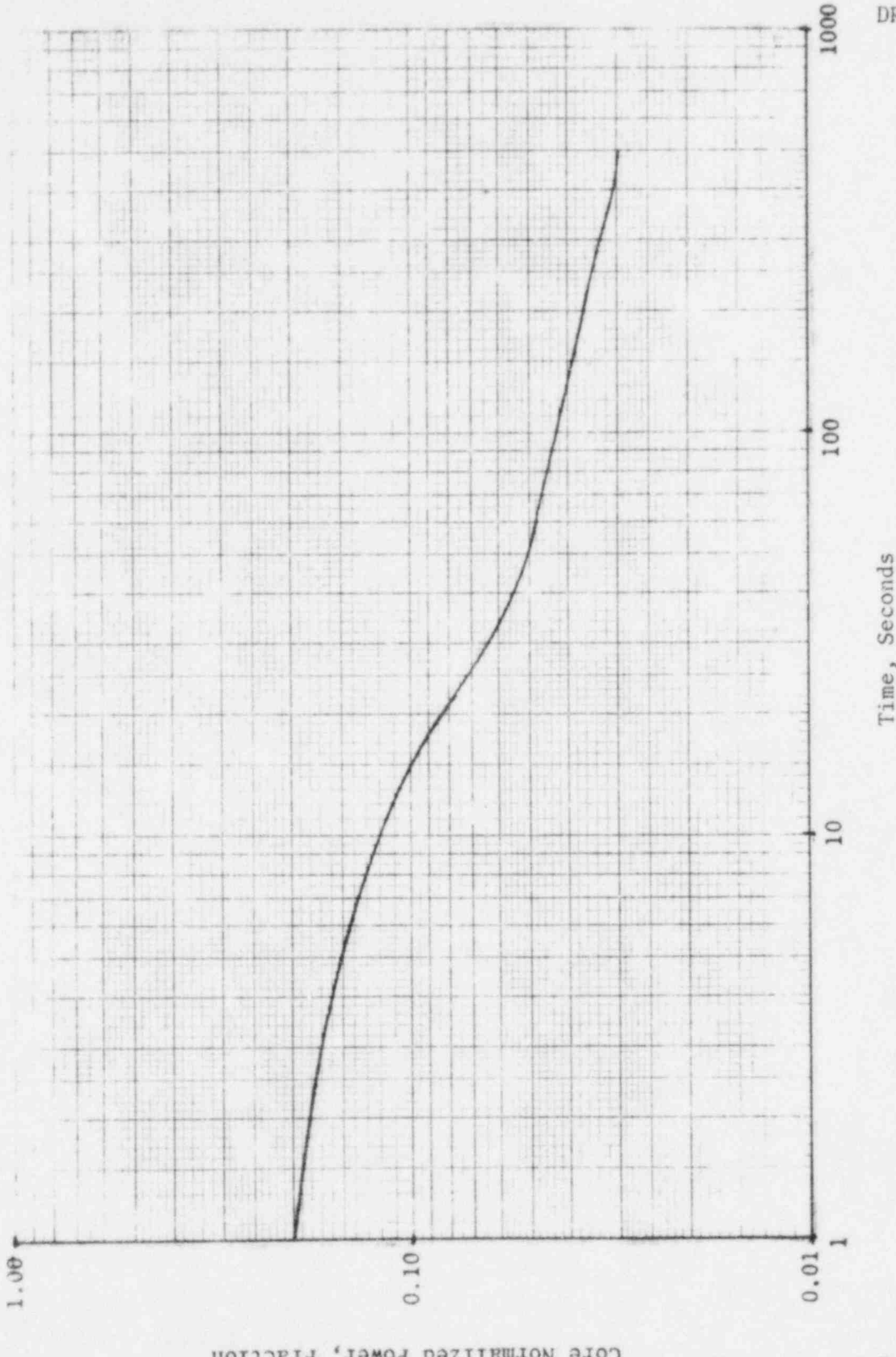


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FILM COEFFICIENT VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-6



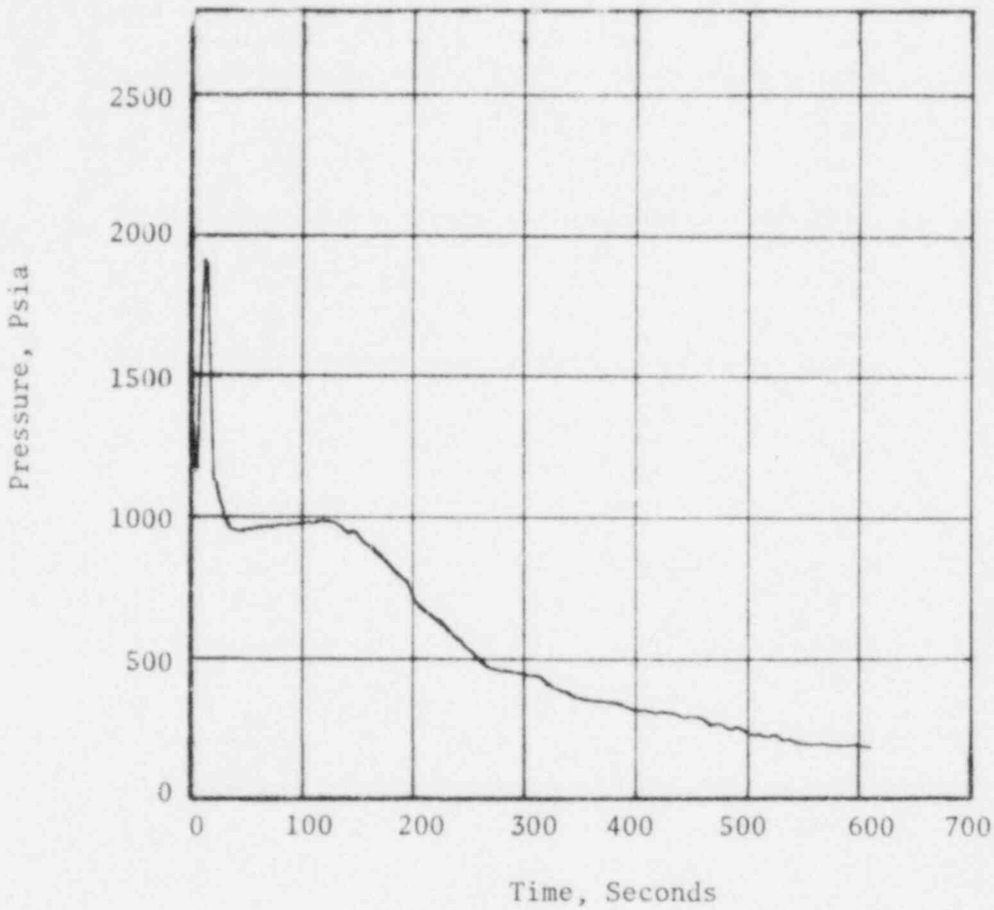
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FLUID TEMPERATURE VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-7

411:18
8/2/74
DRAFT



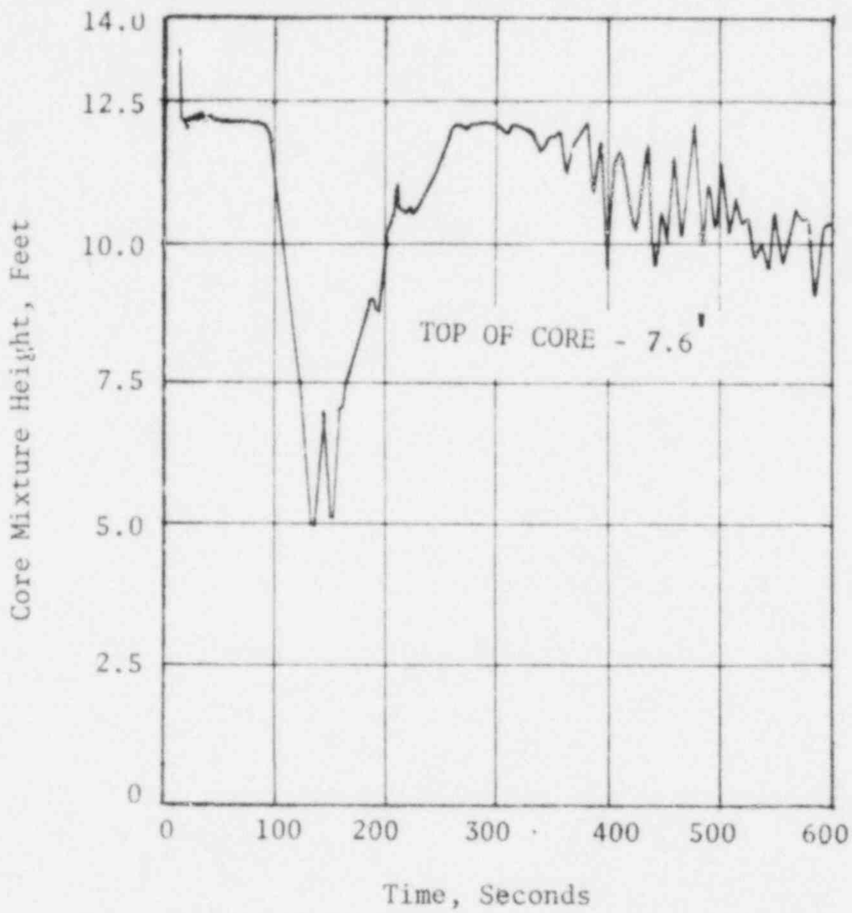
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE POWER FRACTION VS. TIME FOUR INCH DIAMETER BREAK	FIGURE 411-8

411.19
8/2/74
DRAFT

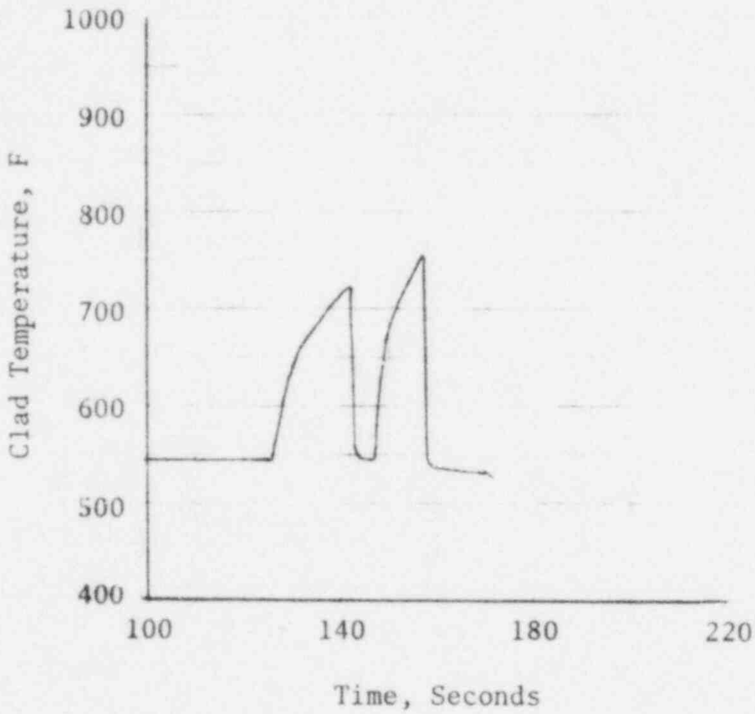


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT MAIN COOLANT PRESSURE VS. TIME THREE INCH DIAMETER BREAK	FIGURE 411-9A

411.20
8/2/74
DRAFT

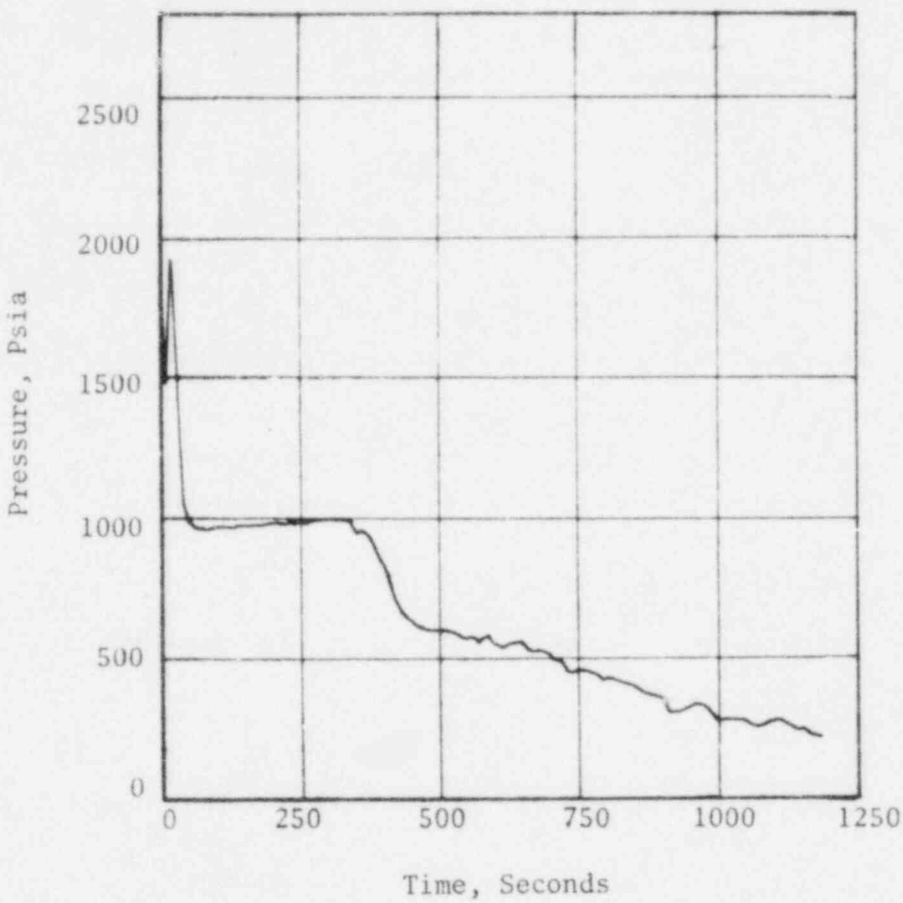


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE COOLANT LEVEL VS. TIME THREE INCH DIAMETER BREAK	FIGURE 411-10A

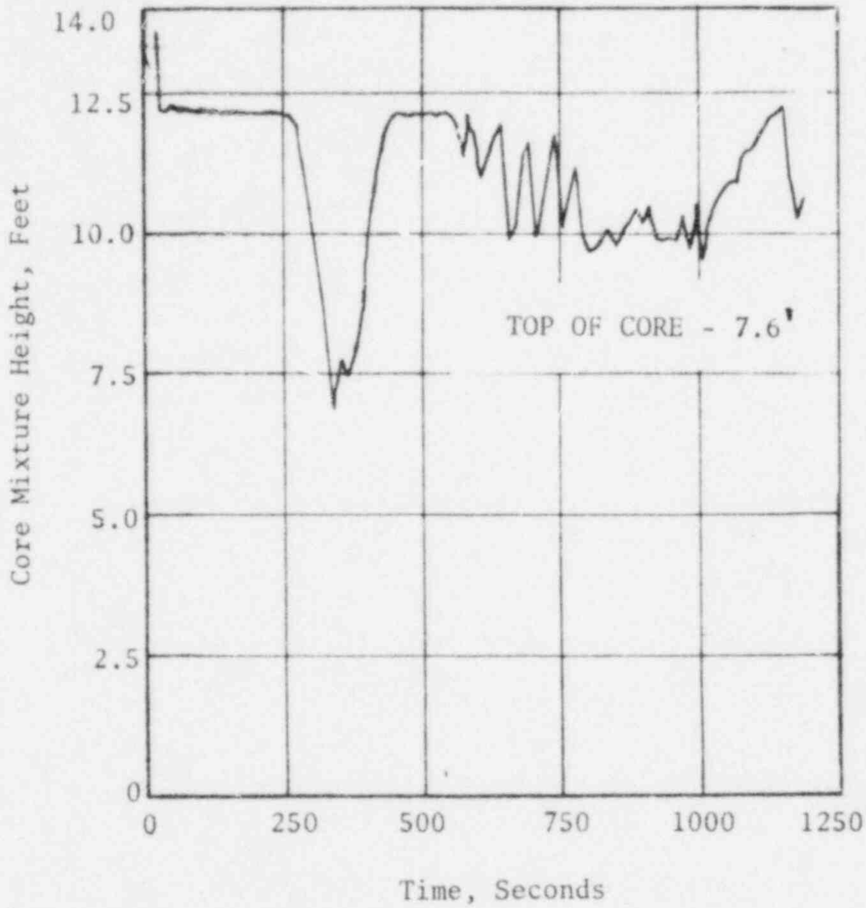


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PEAK CLAD TEMPERATURE VS. TIME THREE INCH DIAMETER BREAK	FIGURE 411-11

411.22
8/2 /74
DRAFT

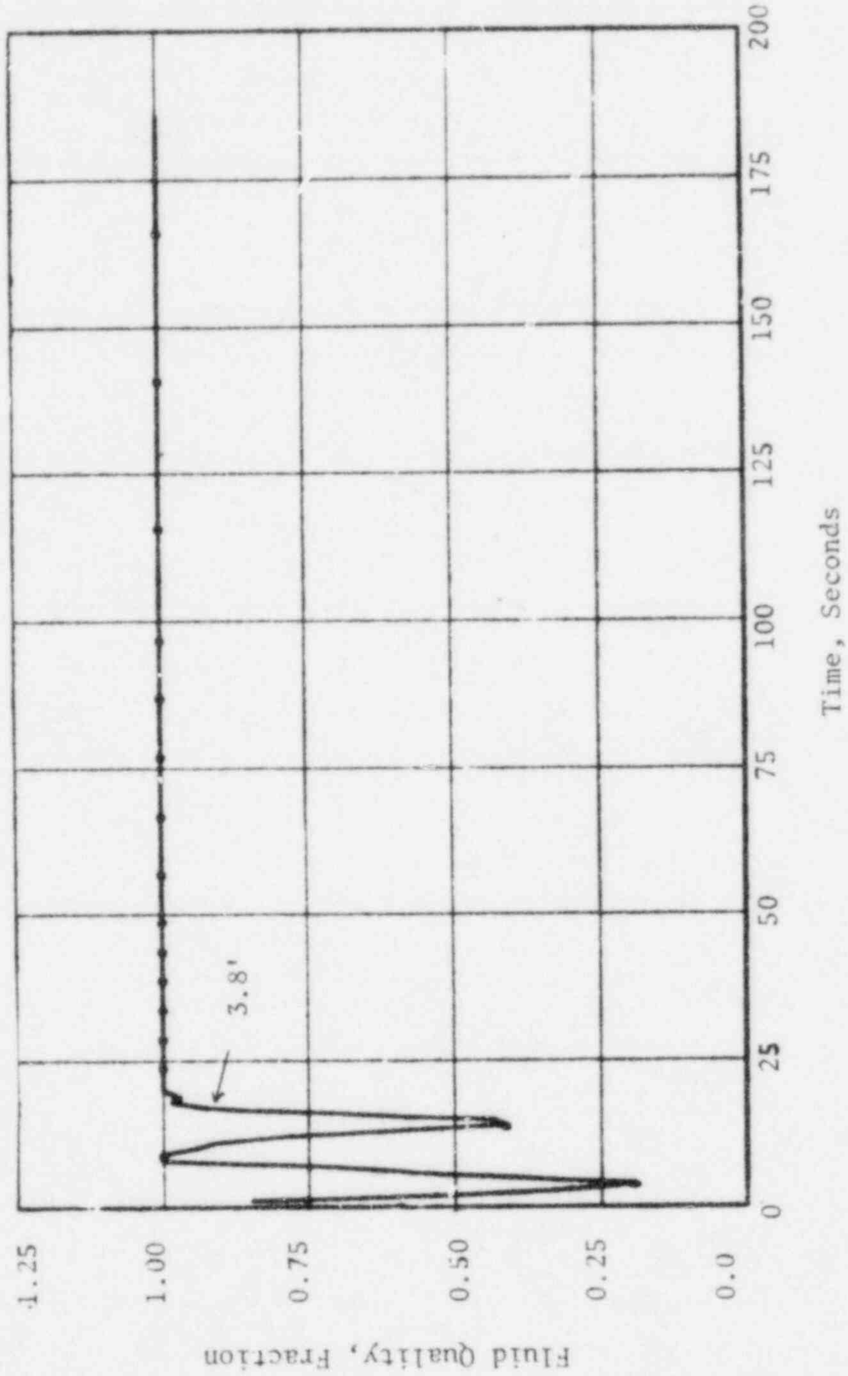


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT MAIN COOLANT PRESSURE VS. TIME TWO INCH DIAMETER BREAK	FIGURE 411-9B



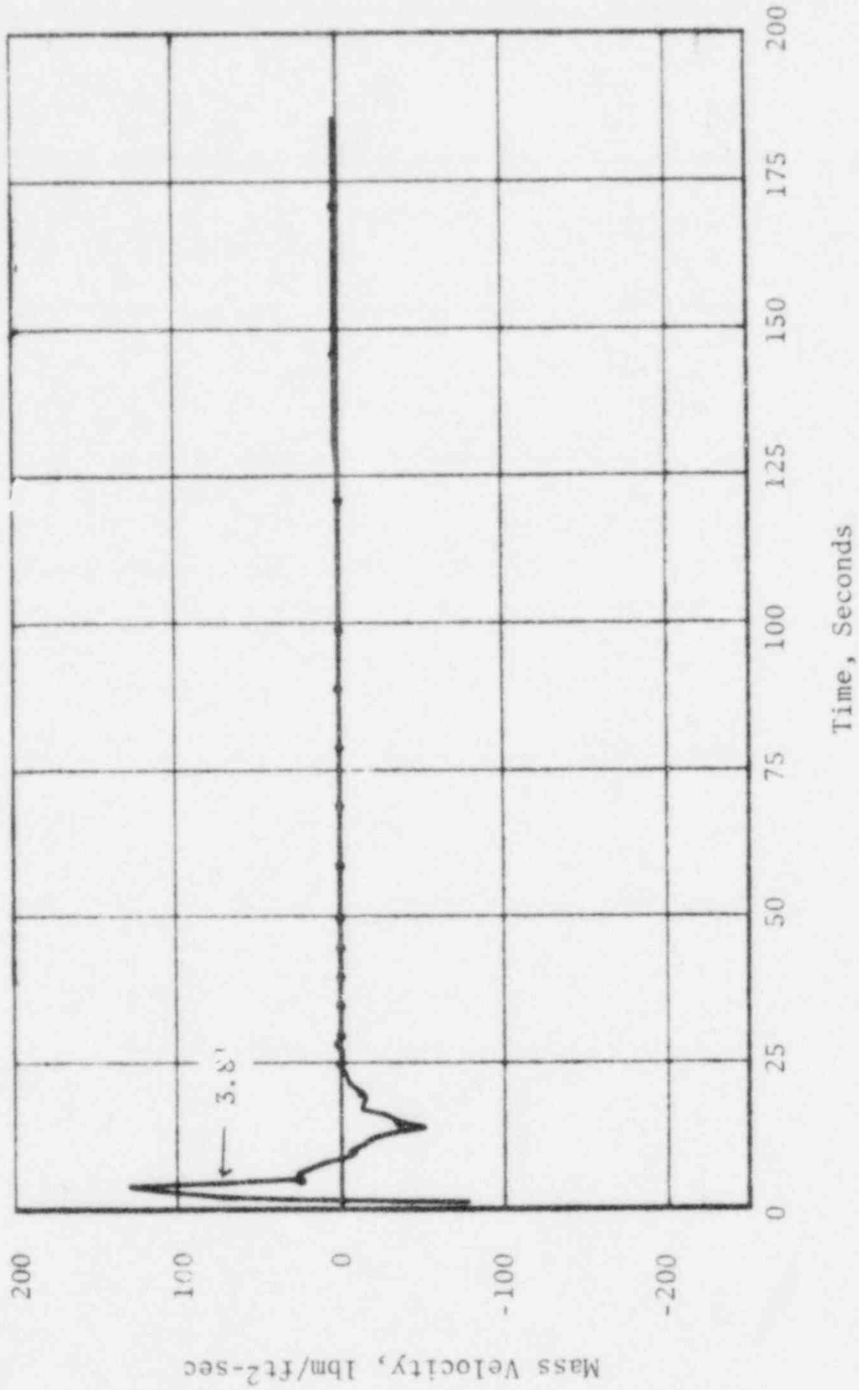
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE COOLANT LEVEL VS. TIME TWO INCH DIAMETER BREAK	FIGURE 411-10B

411:24
8/2/74
DRAFT



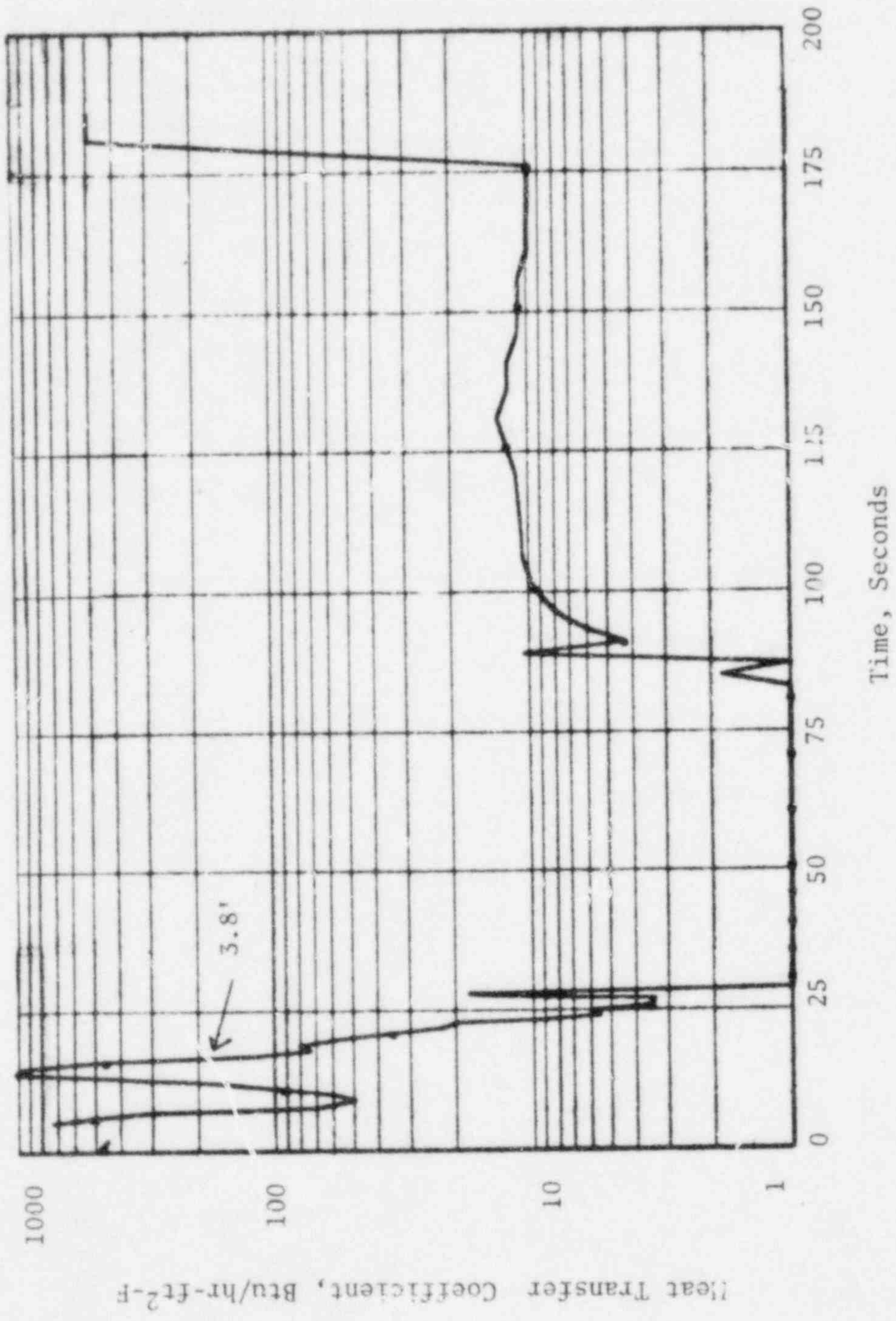
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT QUALITY VS. TIME DECLG (CD=1.0)	FIGURE 411-12A

411:25
8/2/74
DRAFT



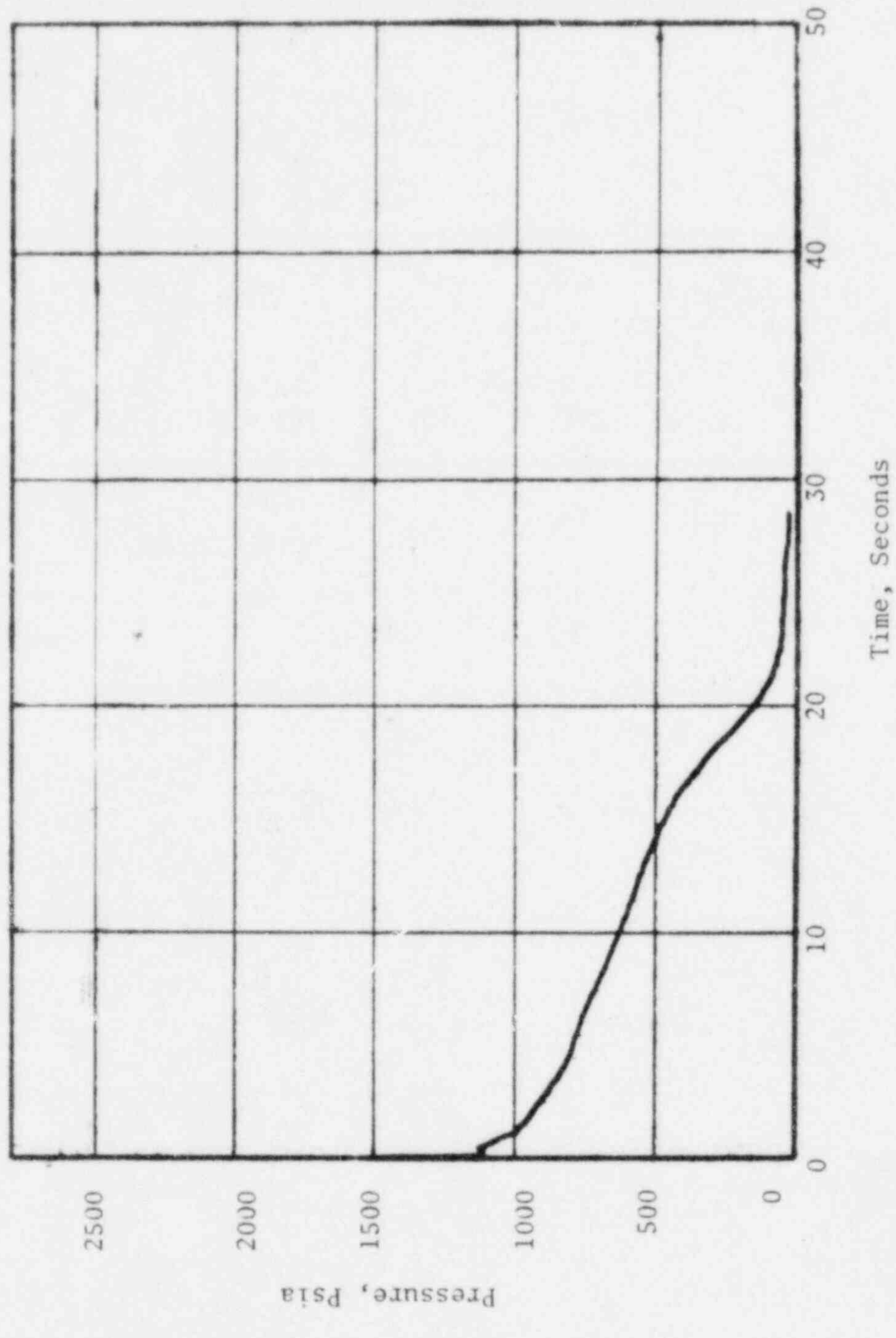
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT MASS VELOCITY VS. TIME DECLG (CD=1.0)	FIGURE 411-13A

411:26
8/2/74
DRAFT



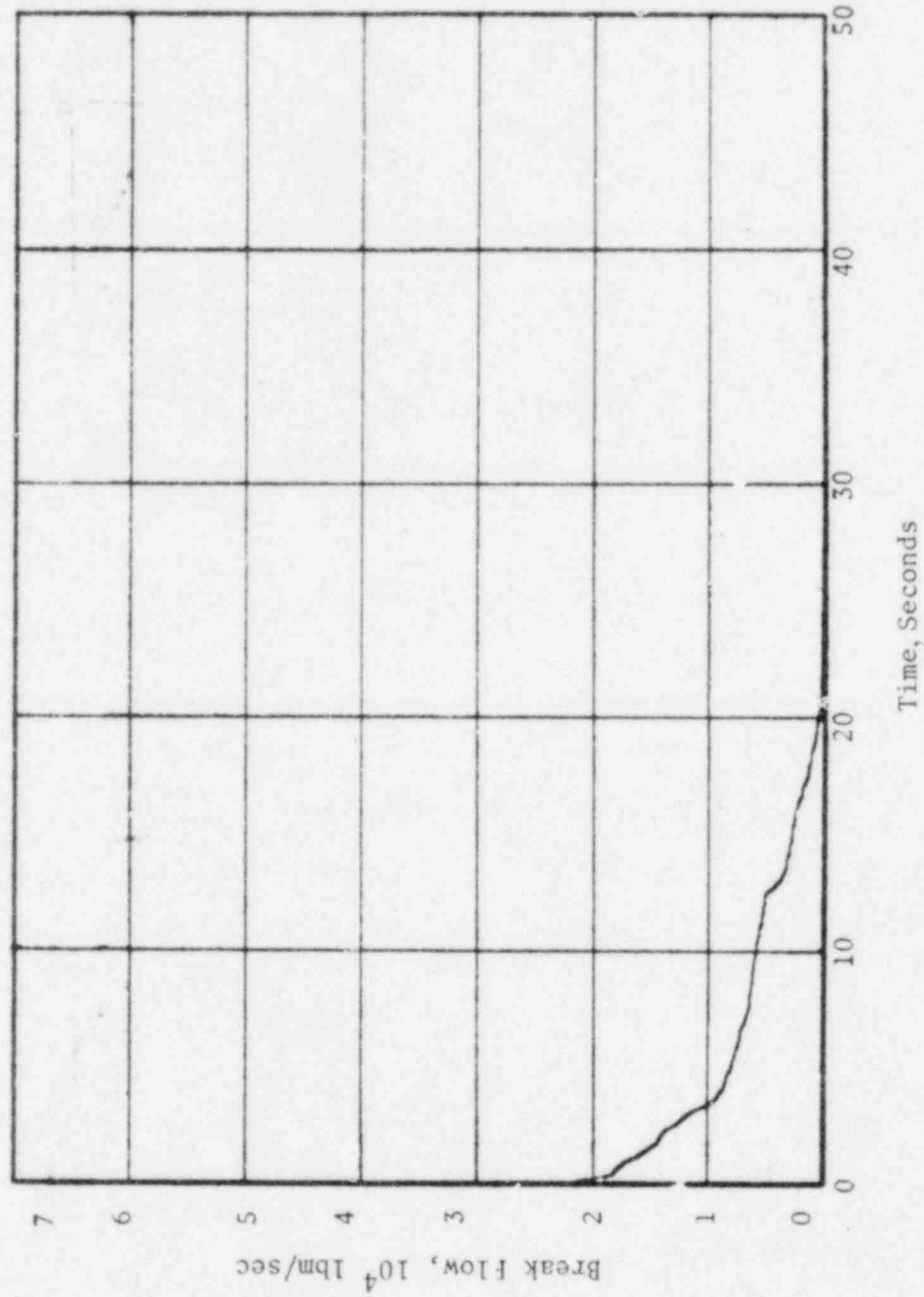
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FILM COEFFICIENT VS. TIME DECLG (CD=1.0)	FIGURE 411-14A

411.27
8/2/74
DRAFT



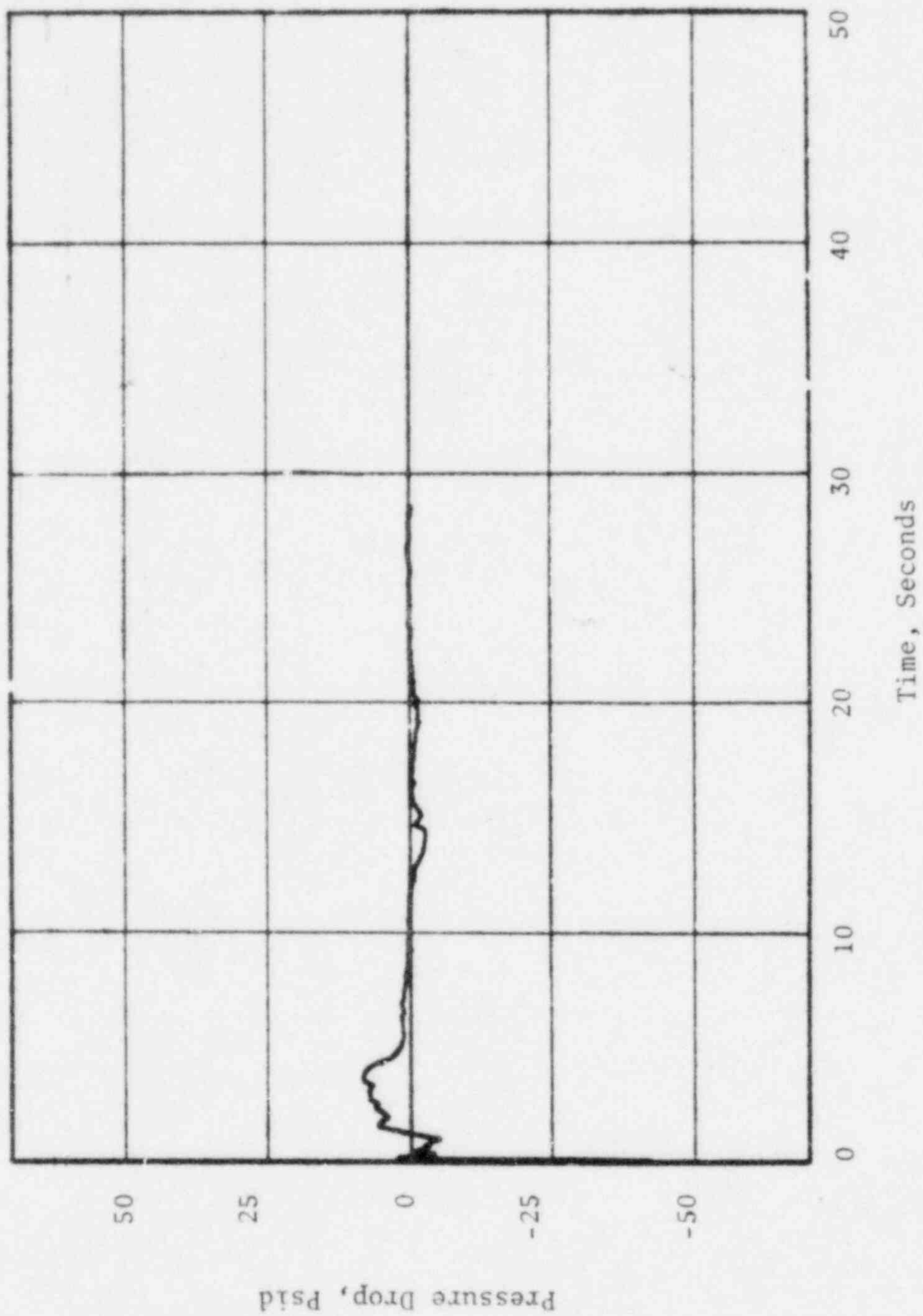
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE VS. TIME DECLG (CD=1.0)	FIGURE 411-15A

411.28
8/2/74
DRAFT



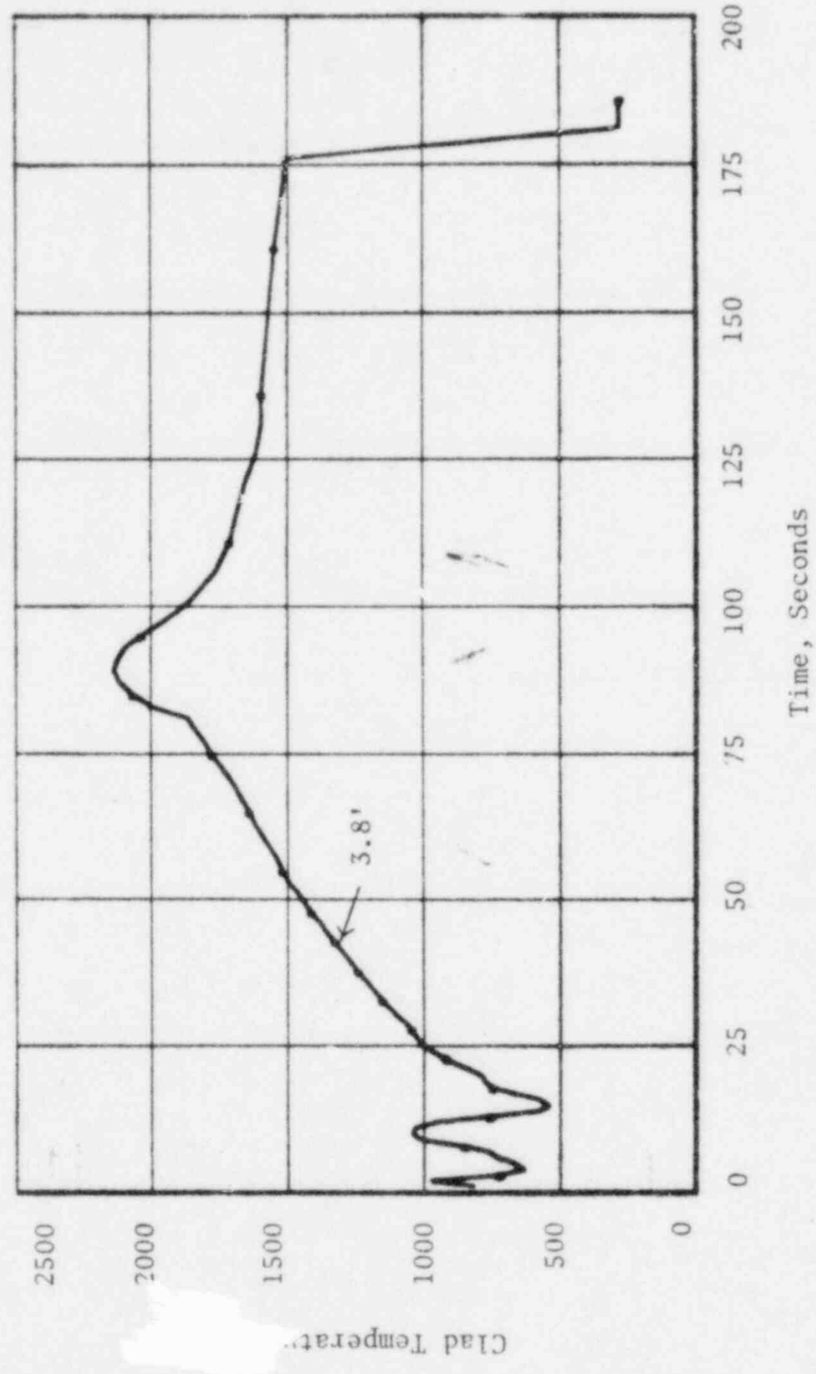
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT BREAK FLOWRATE VS. TIME DECLG (CD=1.0)	FIGURE 411-16A

411:29
8/2/74
DRAFT

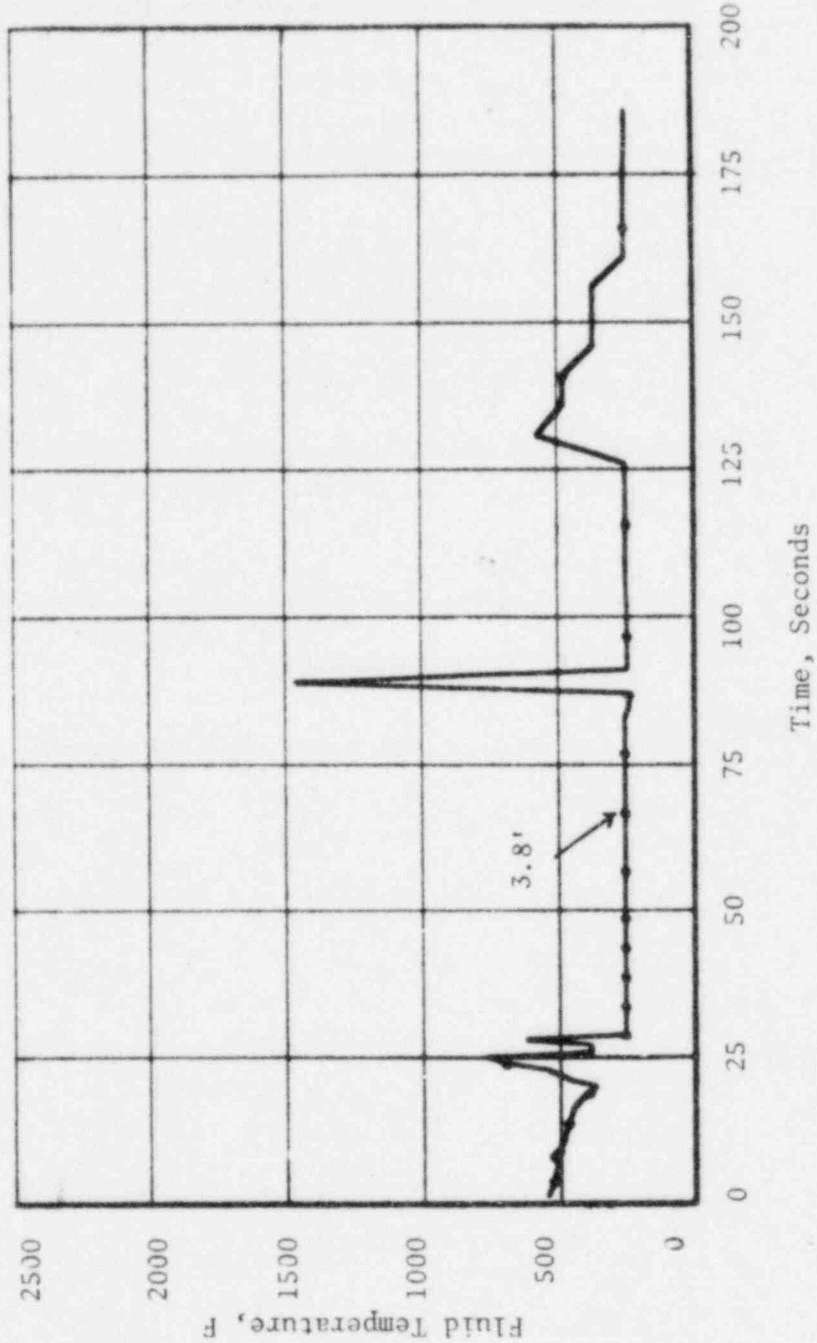


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE DROP VS. TIME DECLG (CD=1.0)	FIGURE 411-17A

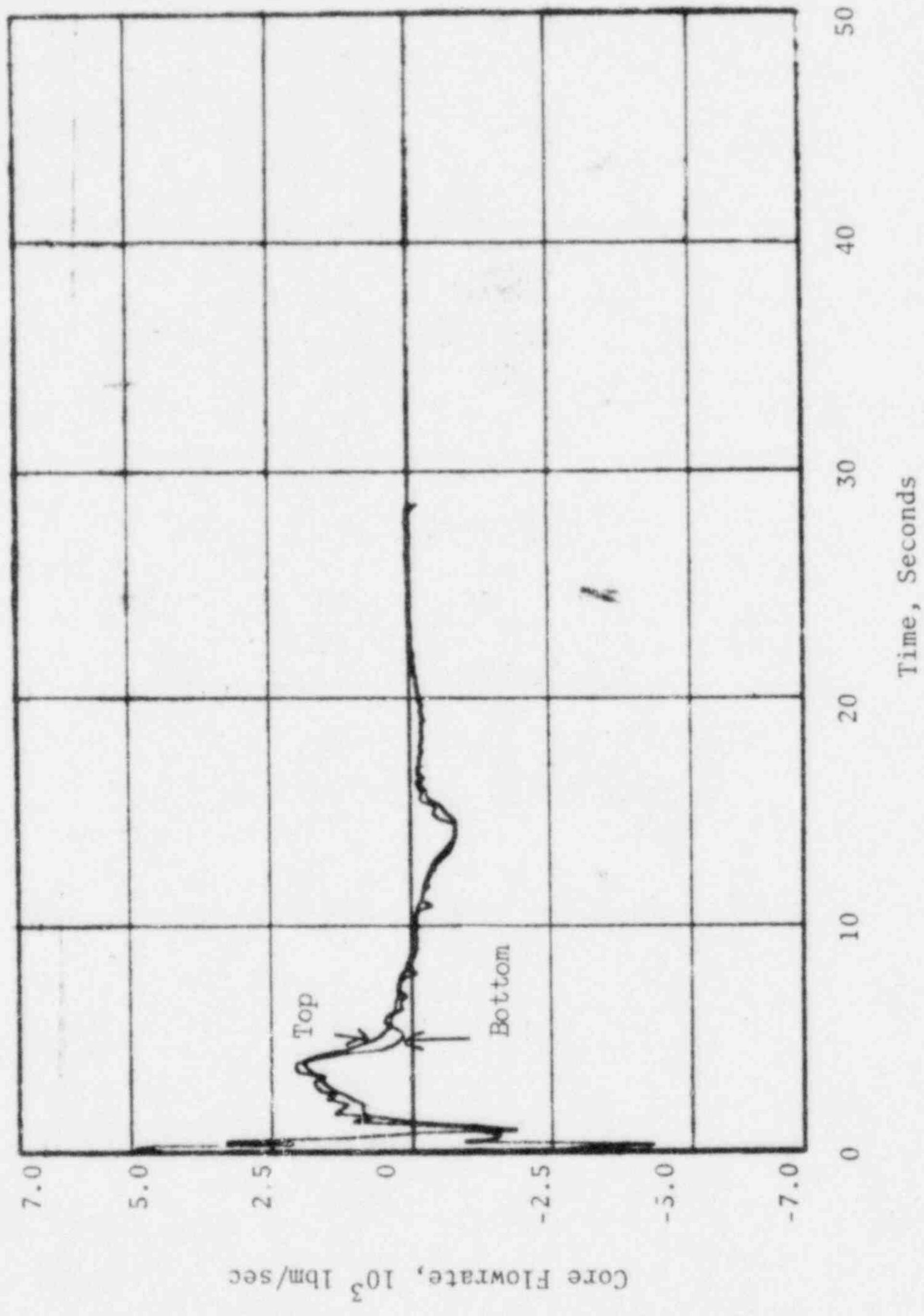
411:30
8/2/74
DRAFT



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT CLAD TEMPERATURE VS. TIME DECLG (CD=1.0)	FIGURE 411-18A

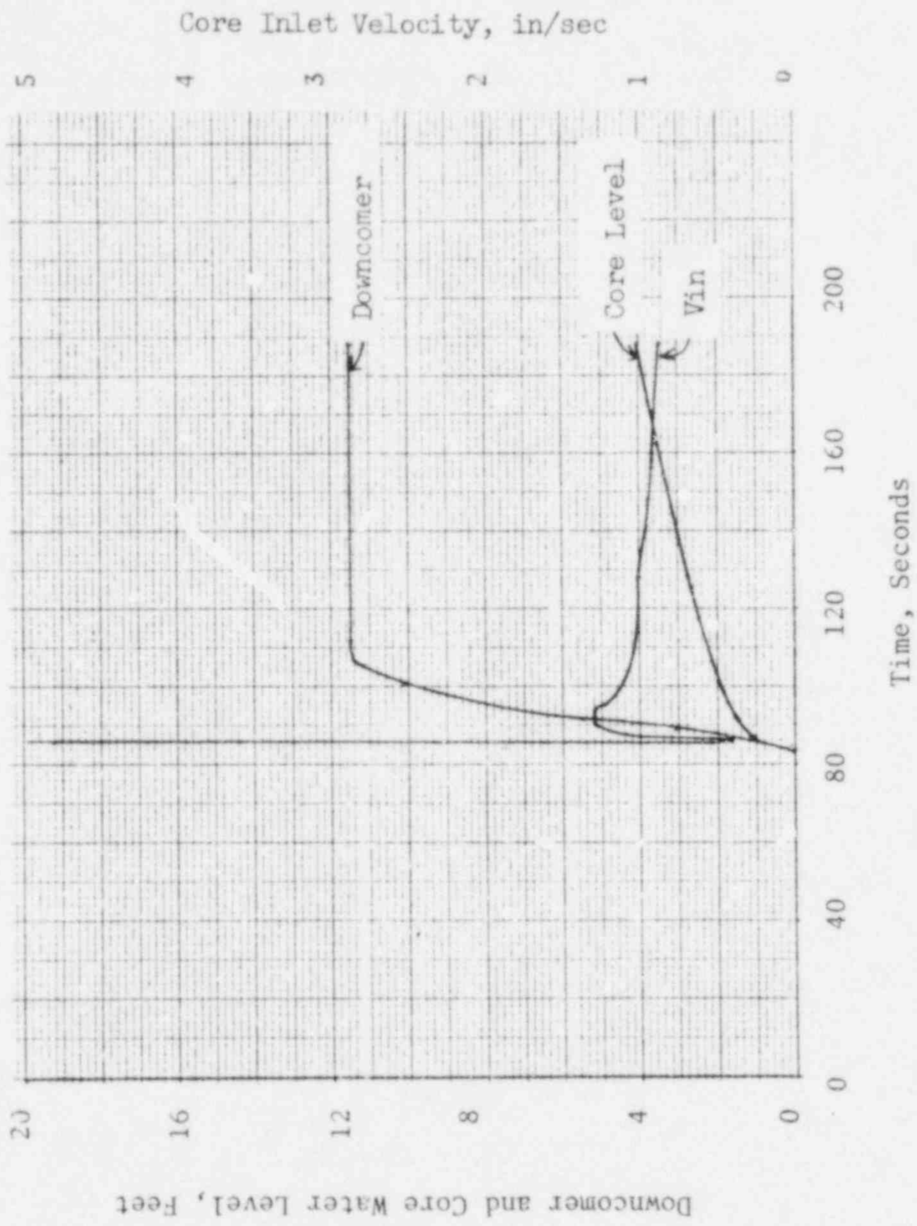


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FLUID TEMPERATURE VS. TIME DECLG (CD=1.0)	FIGURE 411-19A

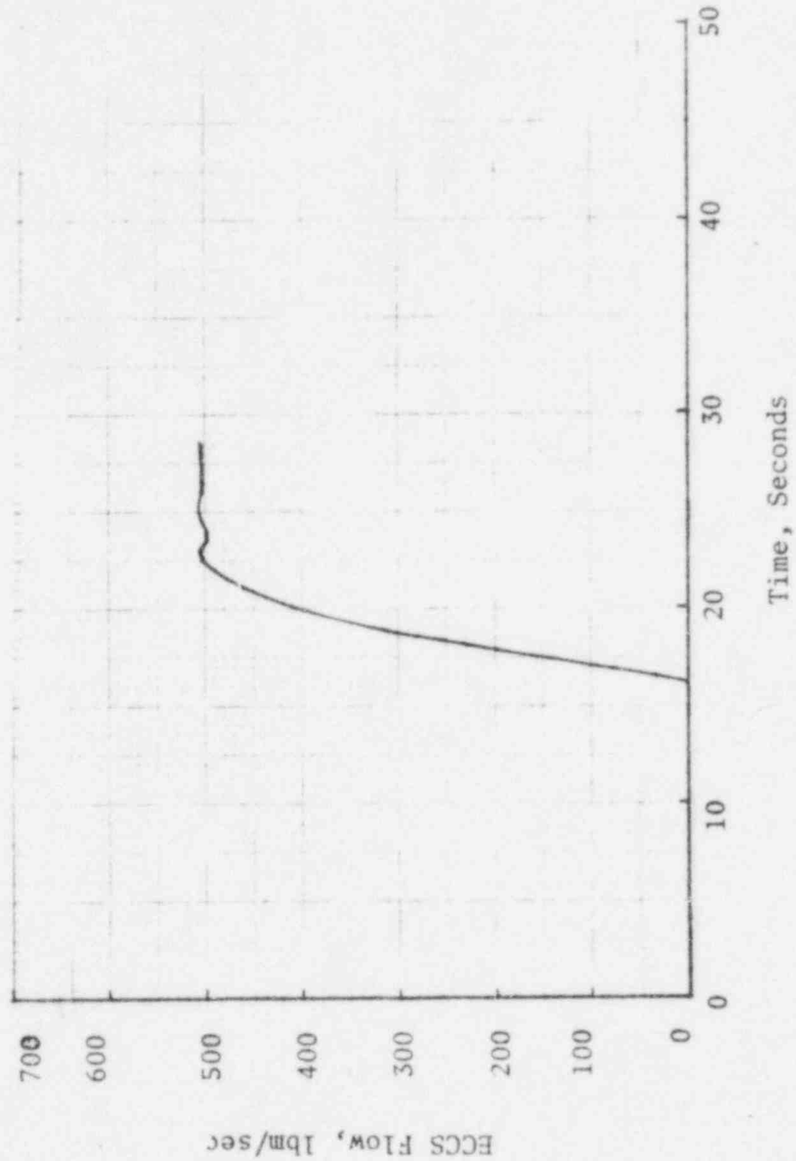


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE FLOW (TOP AND BOTTOM) VS. TIME DECLG (CD=1.0)	FIGURE 411-20A

411:33
8/2/74
DRAFT

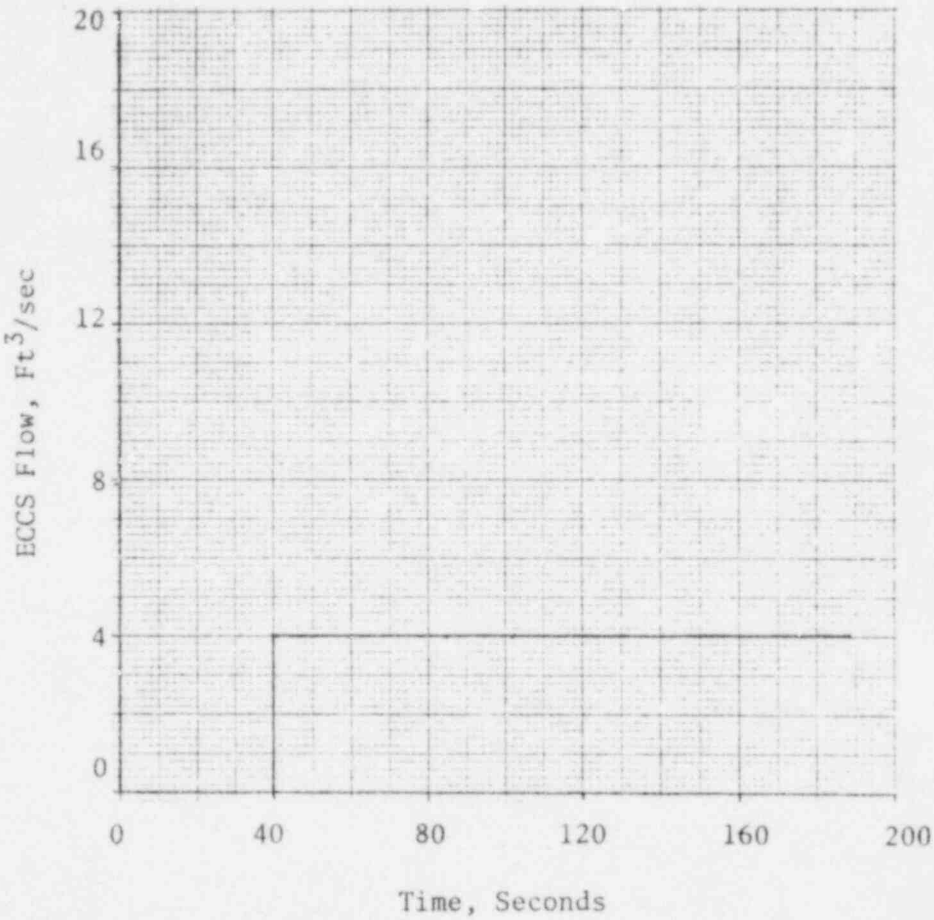


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE REFLOODING VS. TIME DECLG (CD=1.0)	FIGURE 411-21A



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW (BLOWDOWN) DECLG (CD=1.0)	FIGURE 411-22A

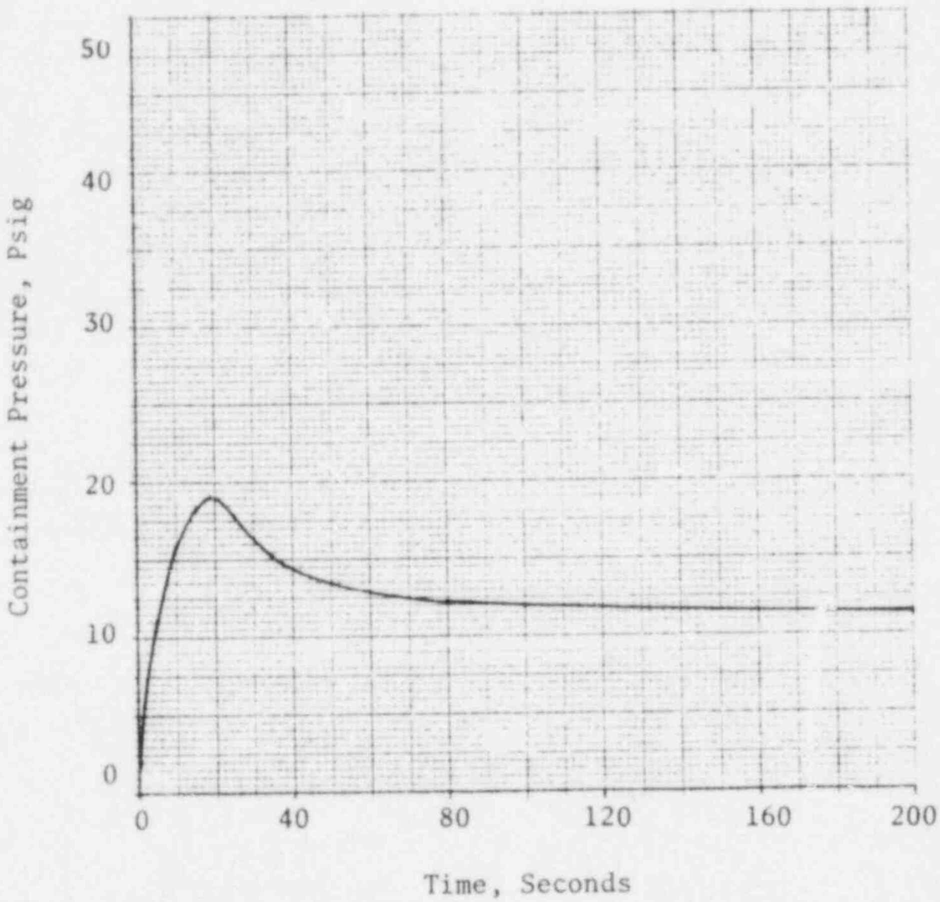
411.35
8/2/74
DRAFT



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PUMPED ECCS FLOW (REFLOOD) DECLG (CD=1.0)	FIGURE 411-23A

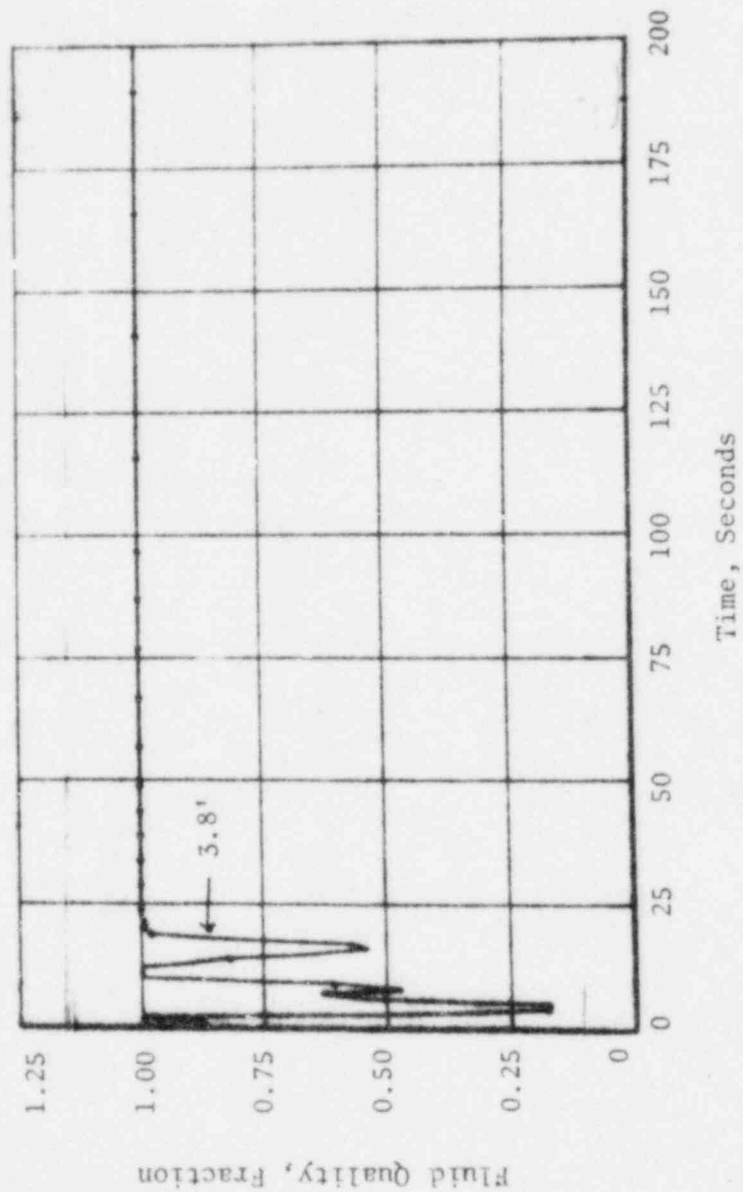
POOR ORIGINAL

411:36
8/2/74
DRAFT



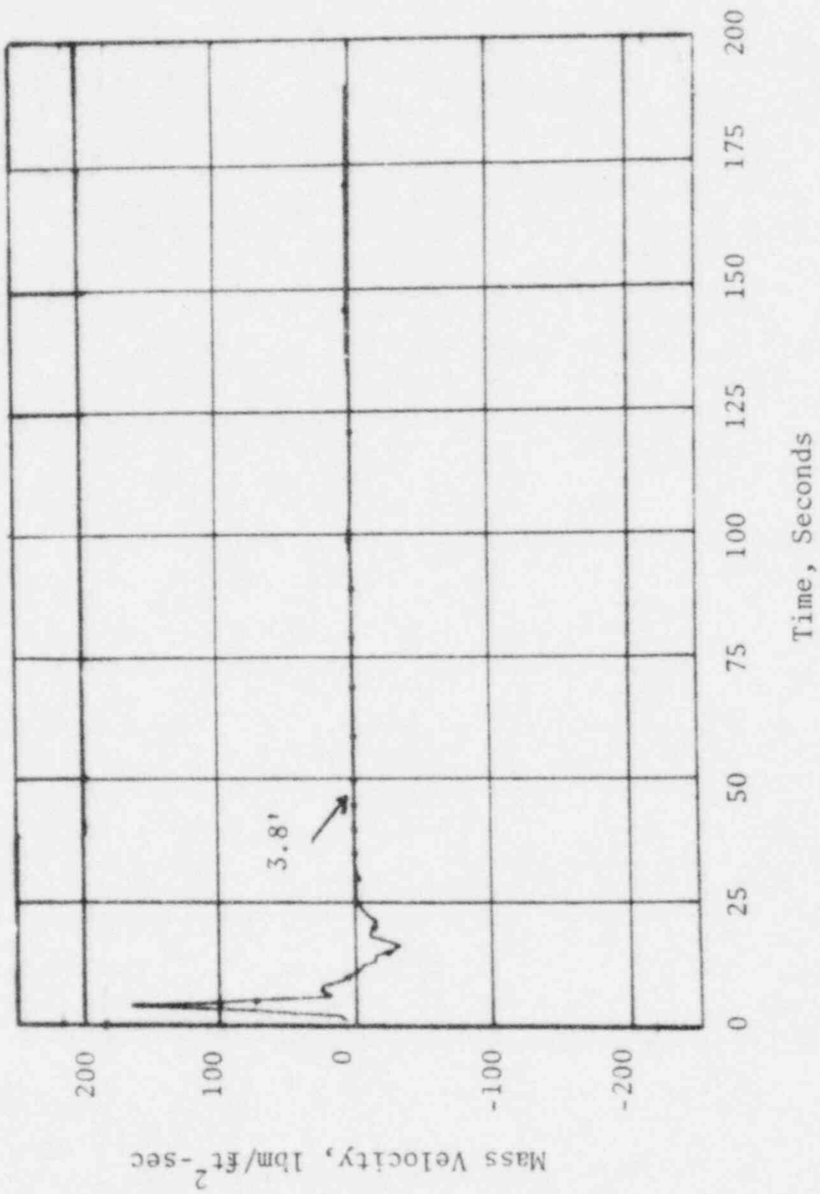
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CONTAINMENT PRESSURE VS. TIME DECLG (CD=1.0)	FIGURE 411-24A

411'.37
8/2/74
DRAFT



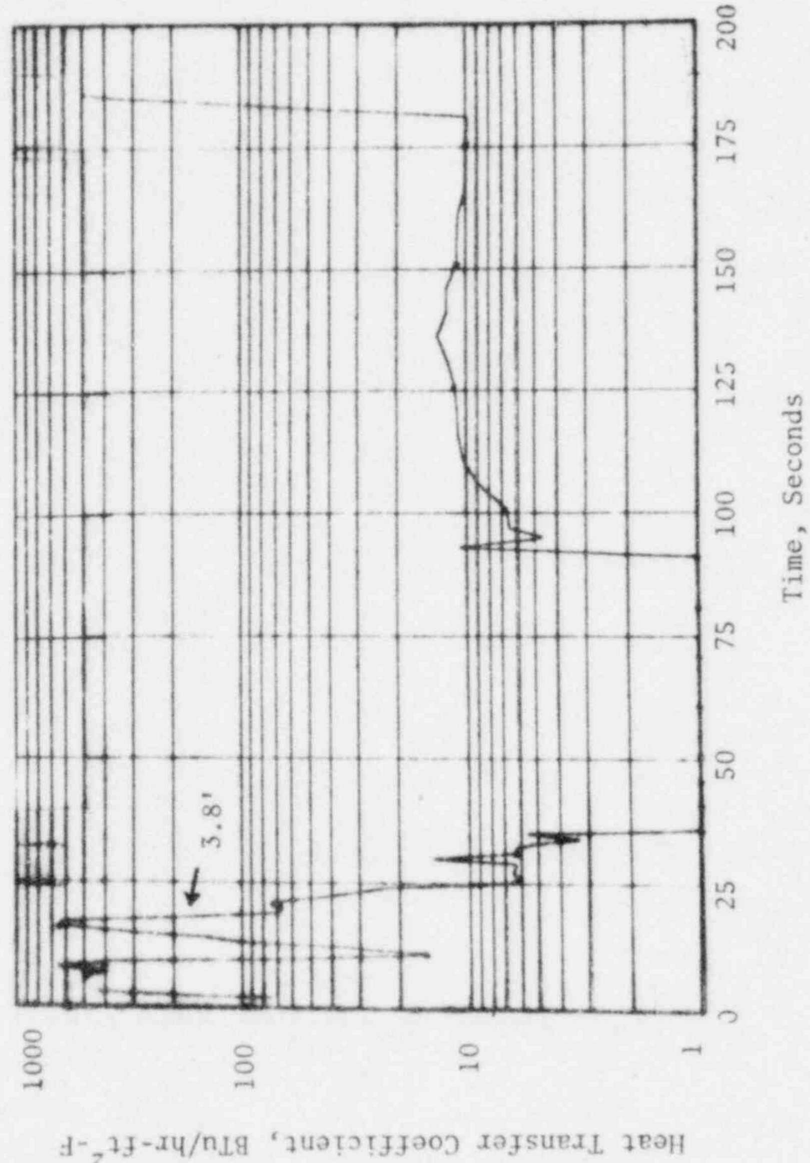
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT QUALITY VS. TIME DECLG (CD=0.6)	FIGURE 411-12B

411:38
8/2/74
DRAFT



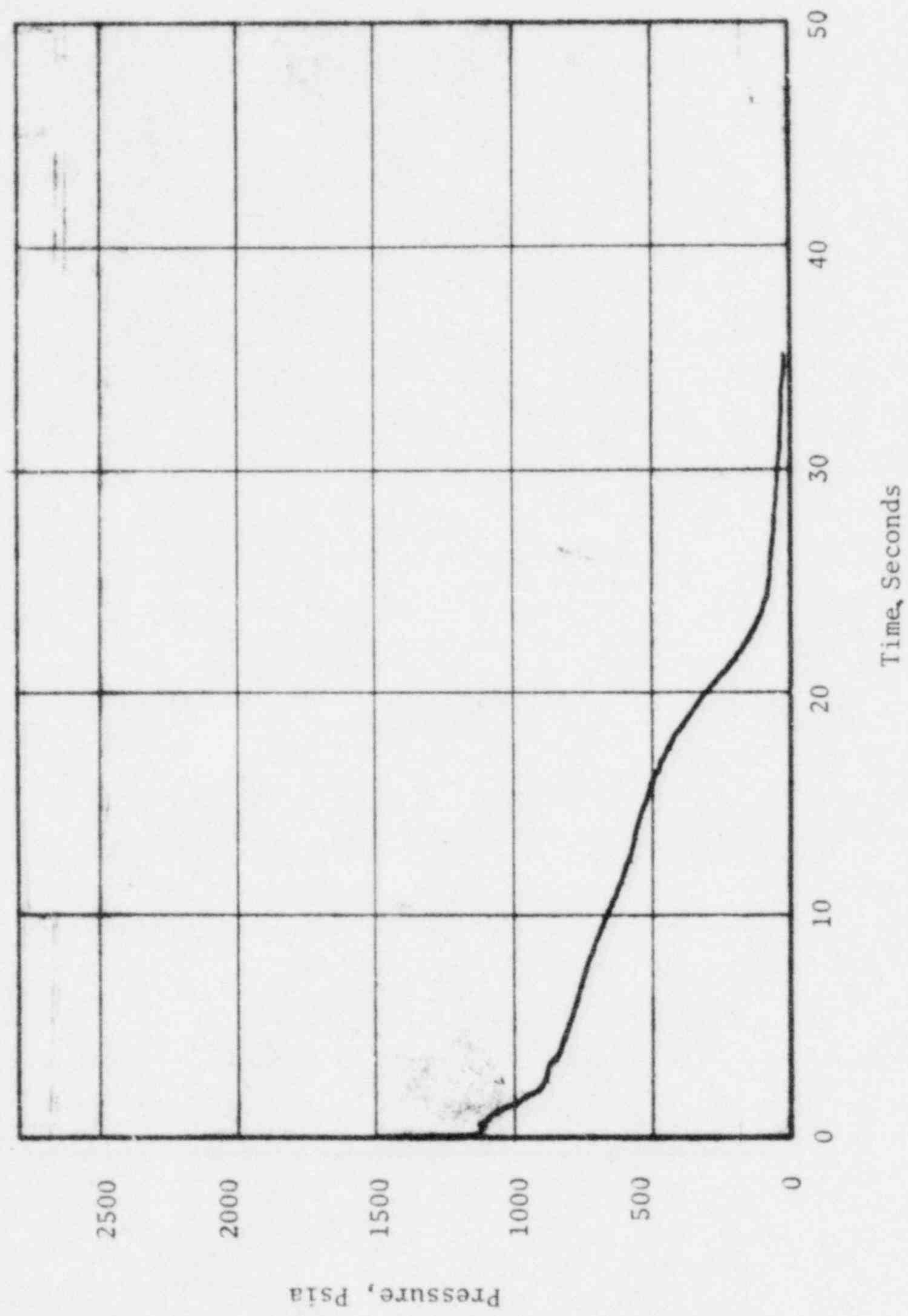
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT MASS VELOCITY VS. TIME DECLG (CD=0.6)	FIGURE 411-13B

411:39
8/2/74
DRAFT



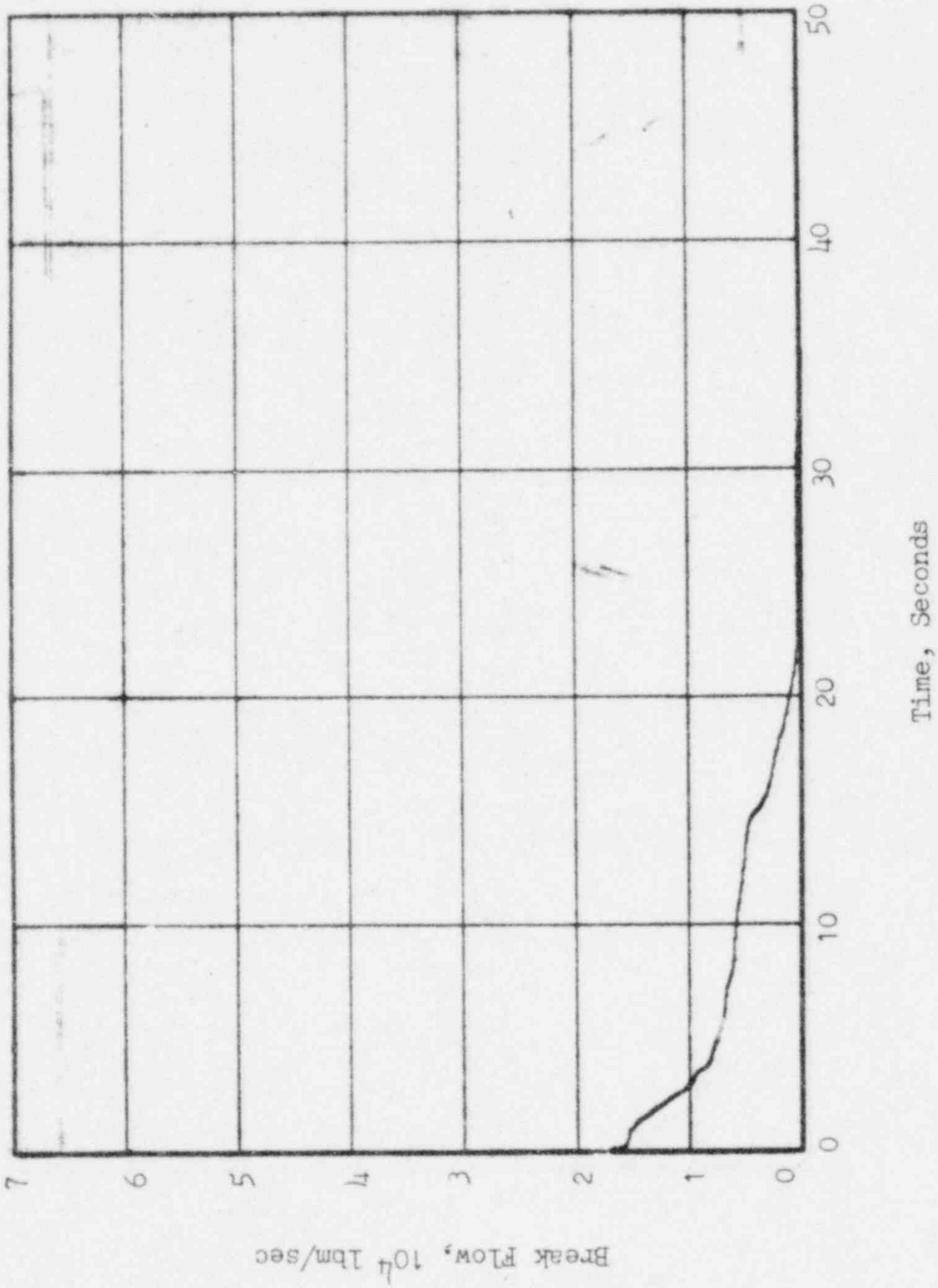
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FILM COEFFICIENT VS. TIME DECLG (CD=0.6)	FIGURE 411-14B

411.40
8/2/74
DRAFT



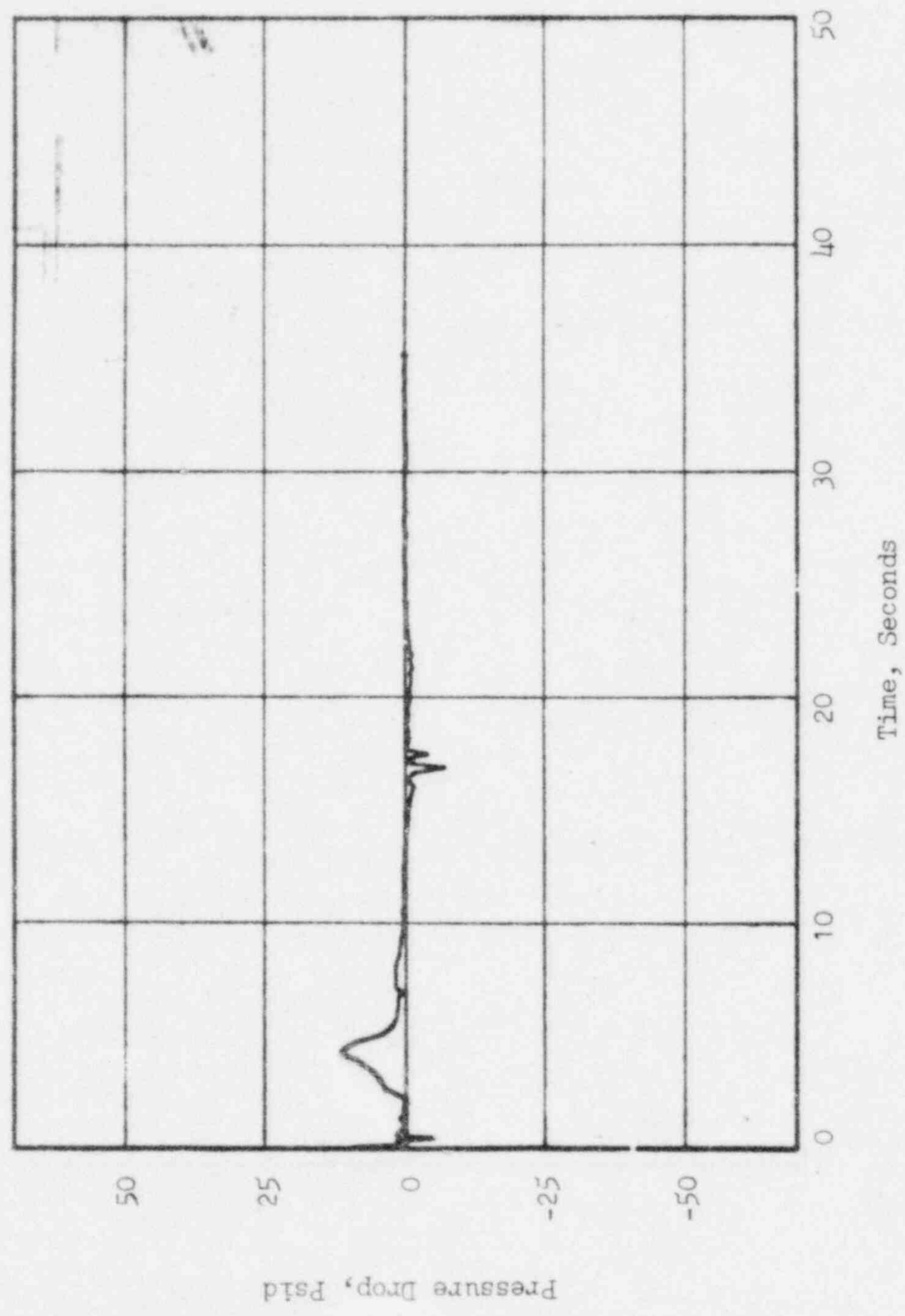
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE VS. TIME DECLG (CD=0.6)	FIGURE 411-15B

411:41
8/2/74
DRAFT



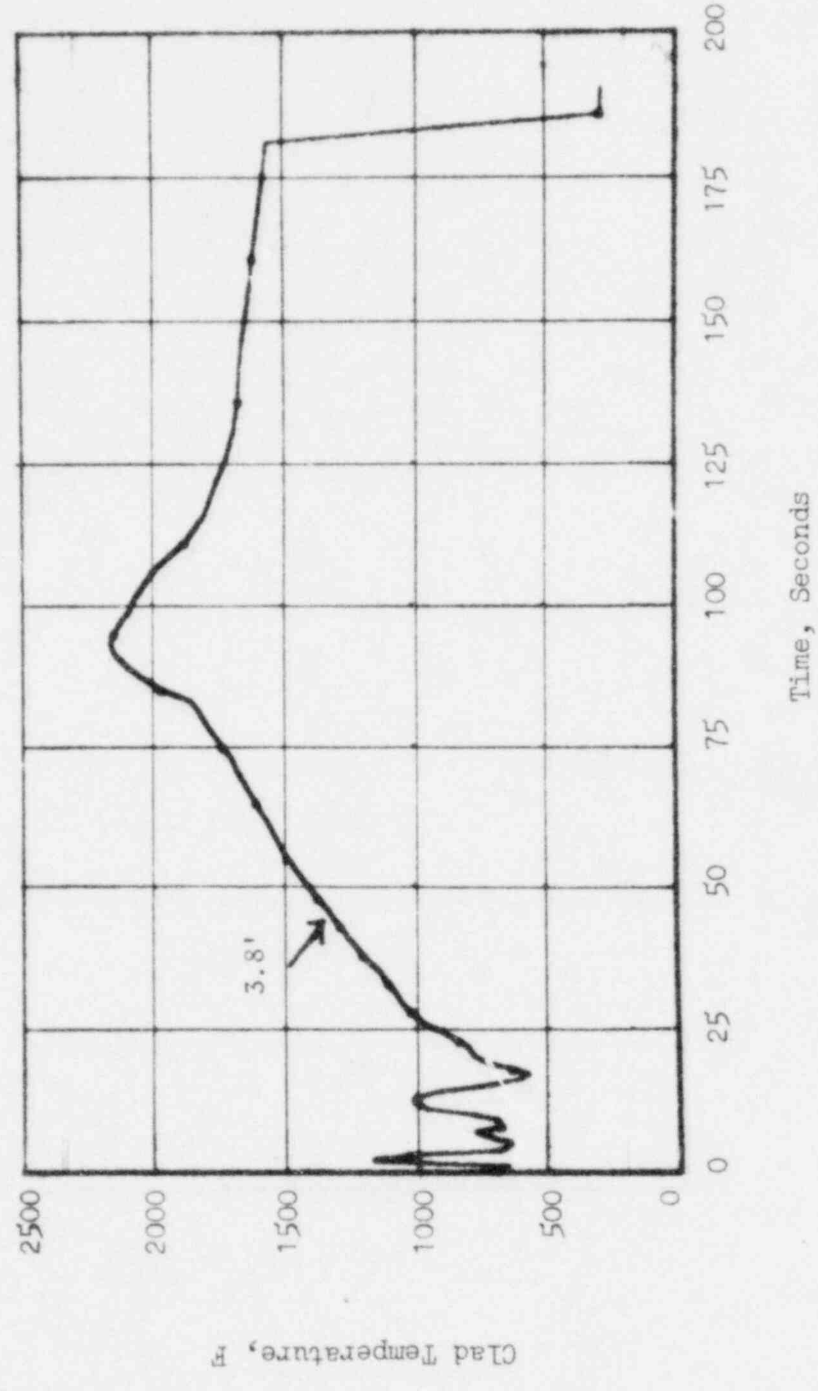
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT BREAK FLOWRATE VS. TIME DECLG (CD=0.6)	FIGURE 411-16B

411.42
8/2/74
DRAFT



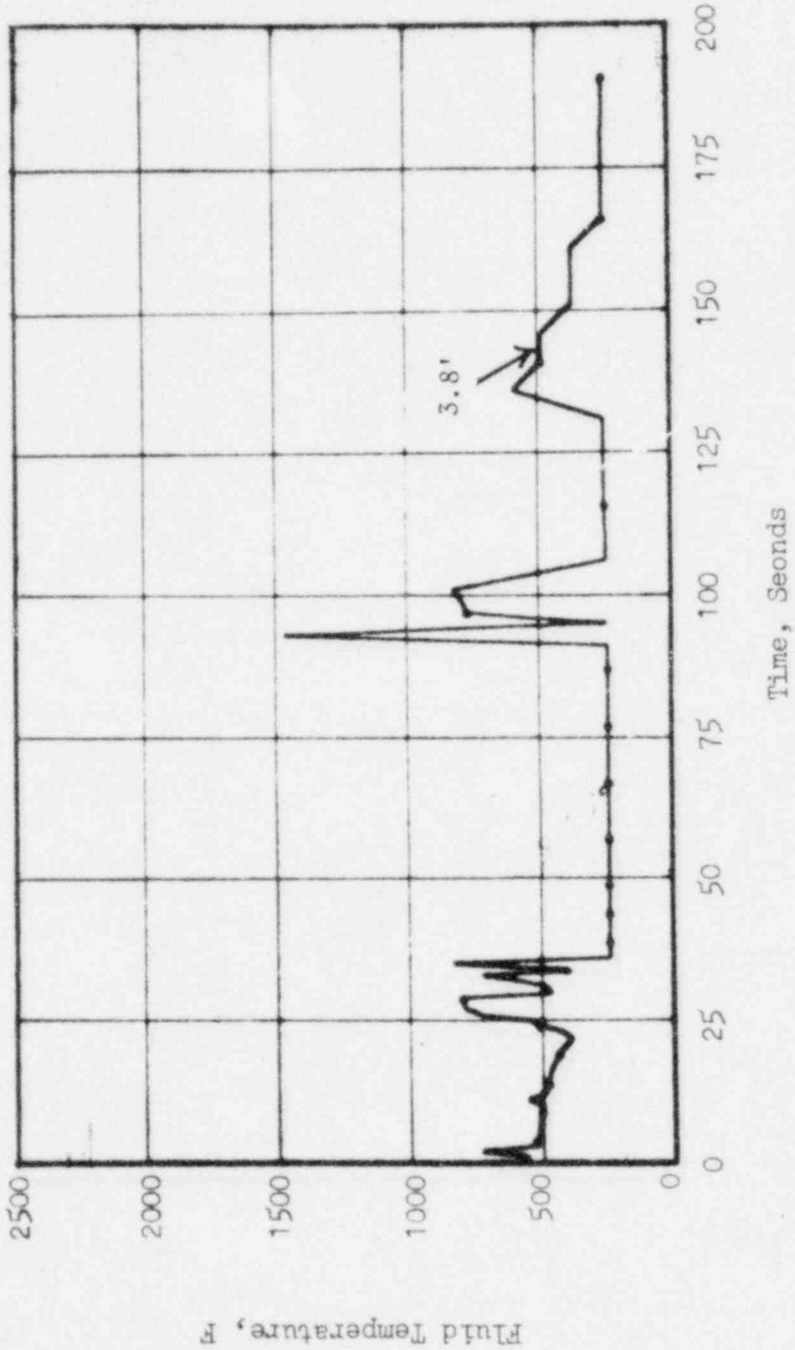
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE DROP VS. TIME DECLG (CD=0.6)	FIGURE 411-17B

411.43
8/2/74
DRAFT



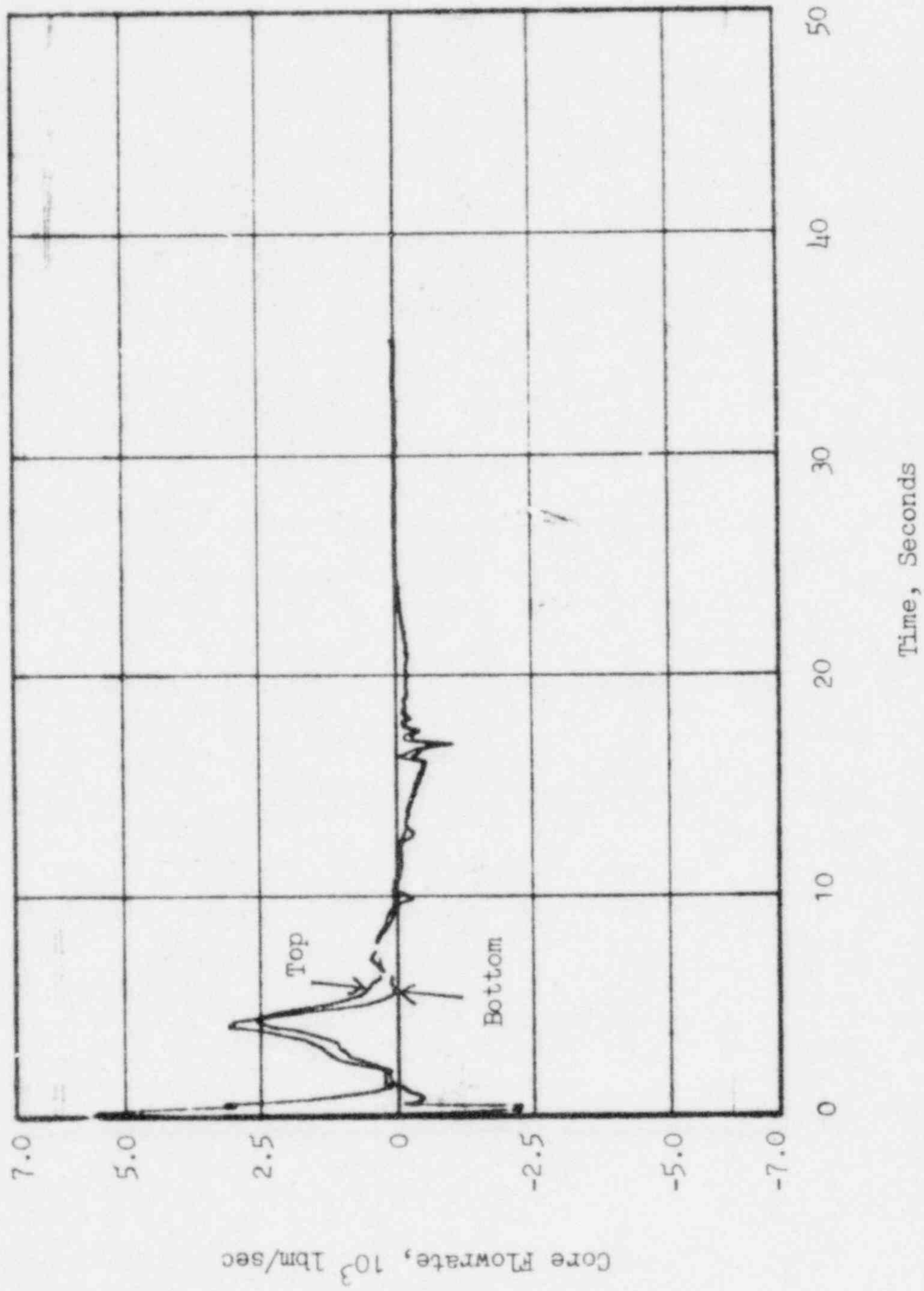
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT CLAD TEMPERATURE VS. TIME DECLG (CD =0.6)	FIGURE 411-18B

411.44
8/2/74
DRAFT



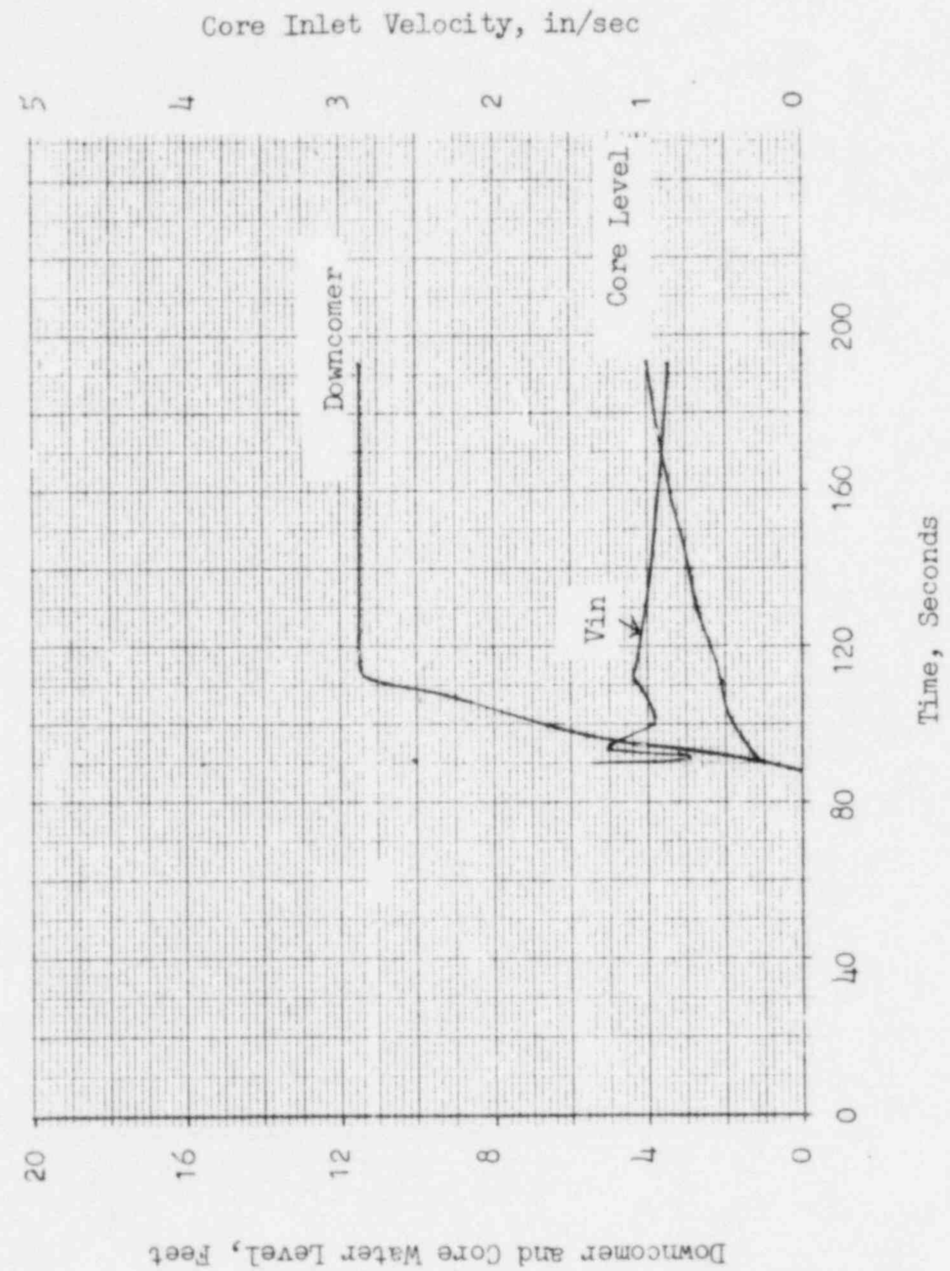
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FLUID TEMPERATURE VS. TIME DECLG (CD=0.6)	FIGURE 411-19B

411:45
8/2/74
DRAFT



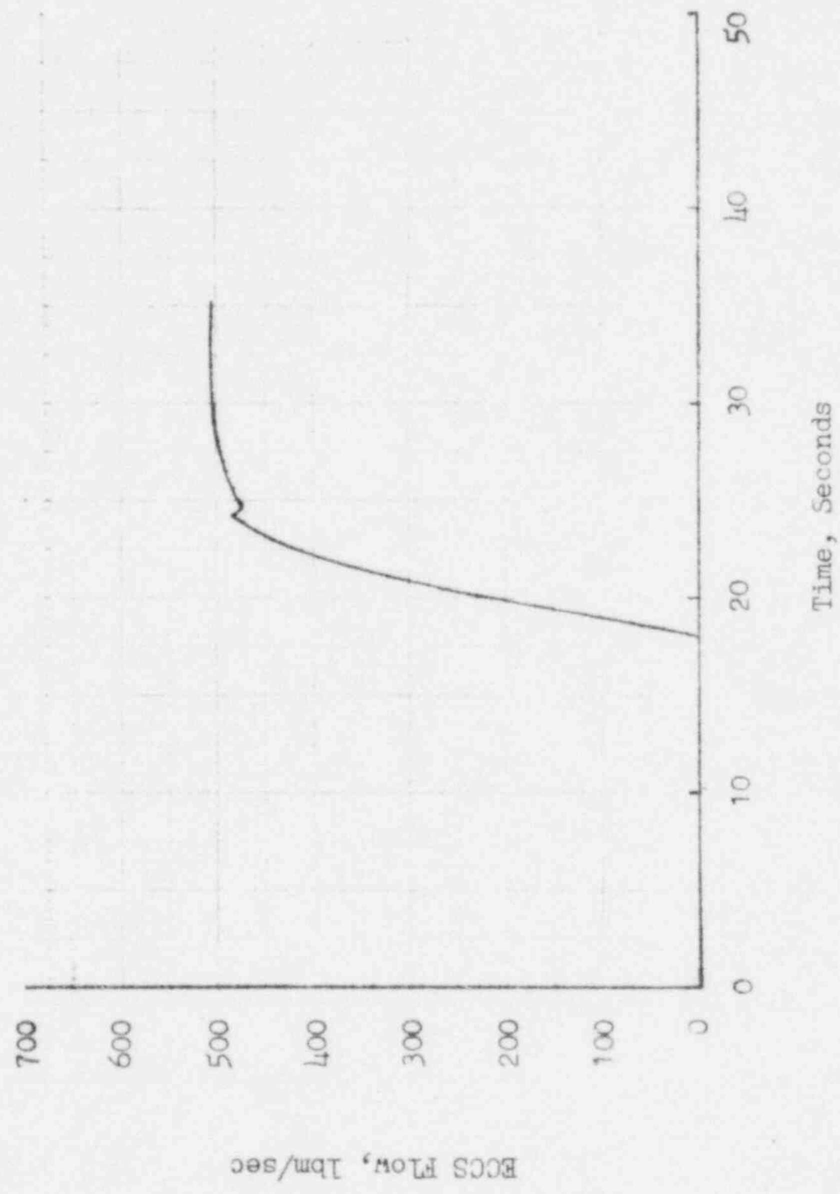
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE FLOW (TOP AND BOTTOM) VS. TIME DECLG (CD=0.6)	FIGURE 411-20B

411.46
8/2/74
DRAFT



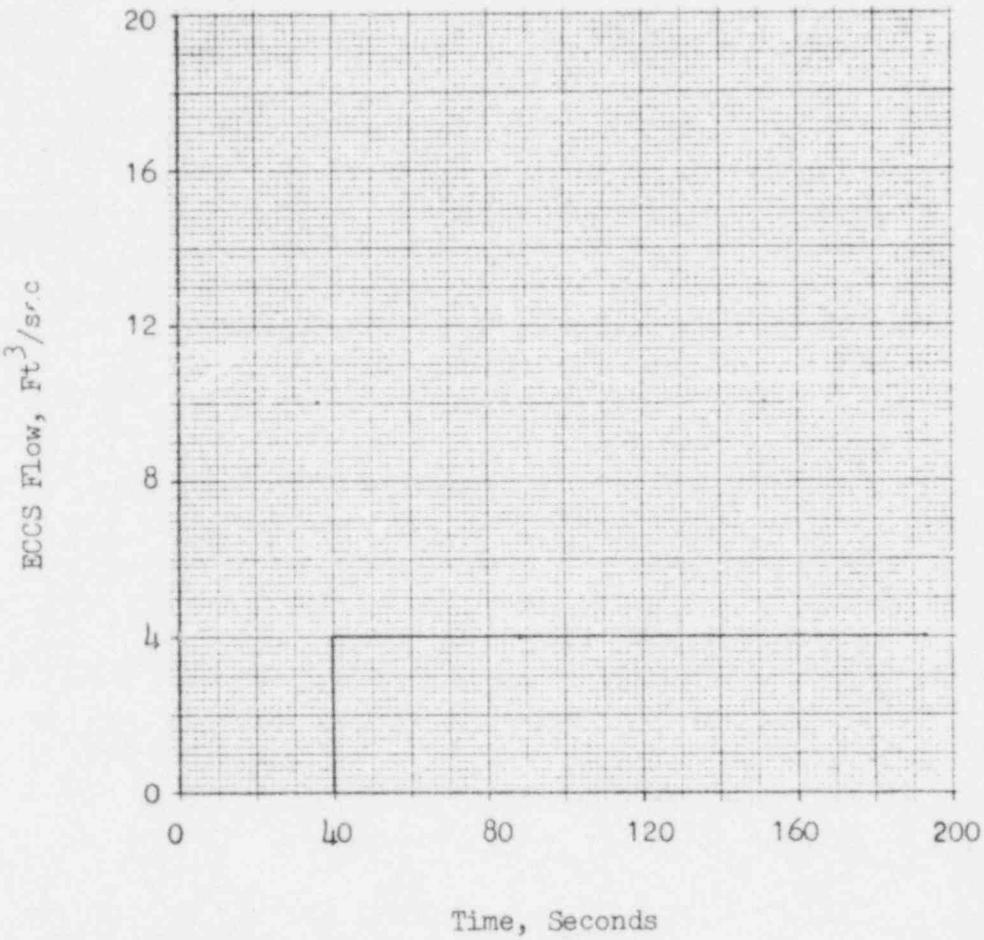
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE REFLOODING VS. TIME DECLG (CD=0.6)	FIGURE 411-21B

411:47
8/2/74
DRAFT



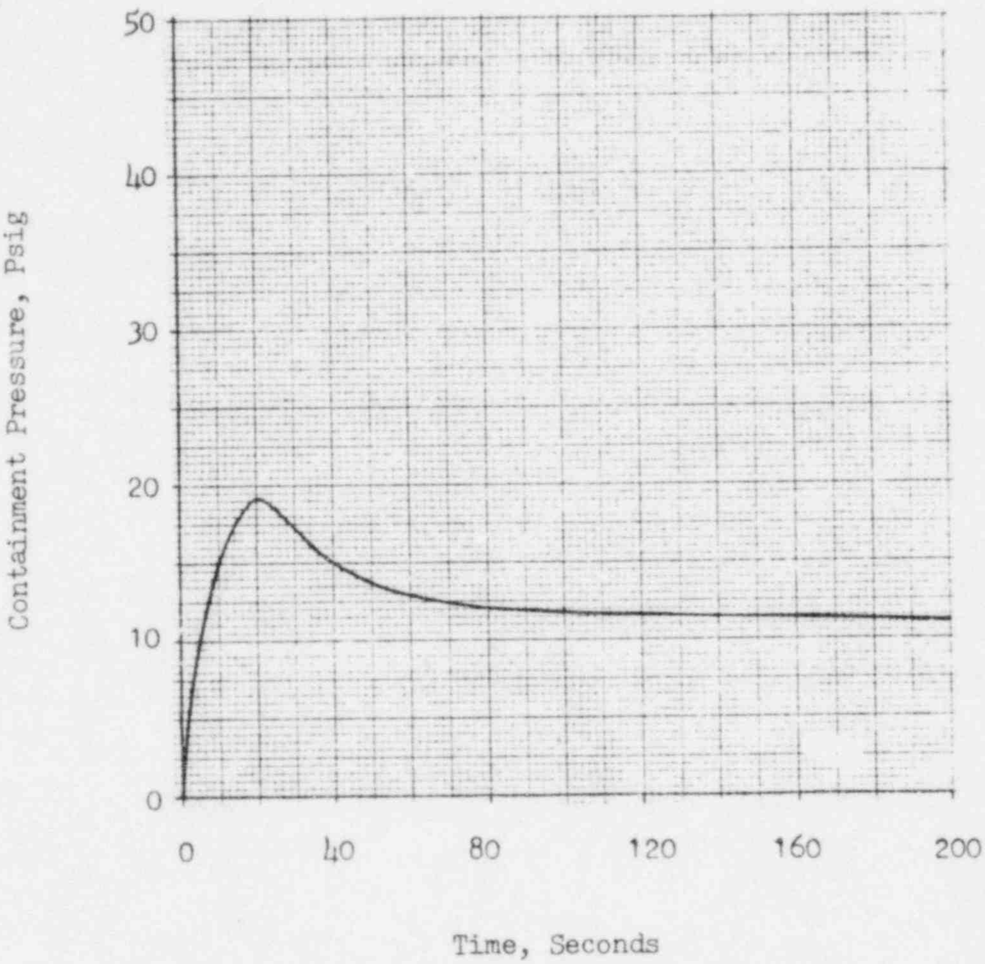
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW (BLOWDOWN) DECLG (CD=0.6)	FIGURE 411-22B

411:48
8/2/74
DRAFT



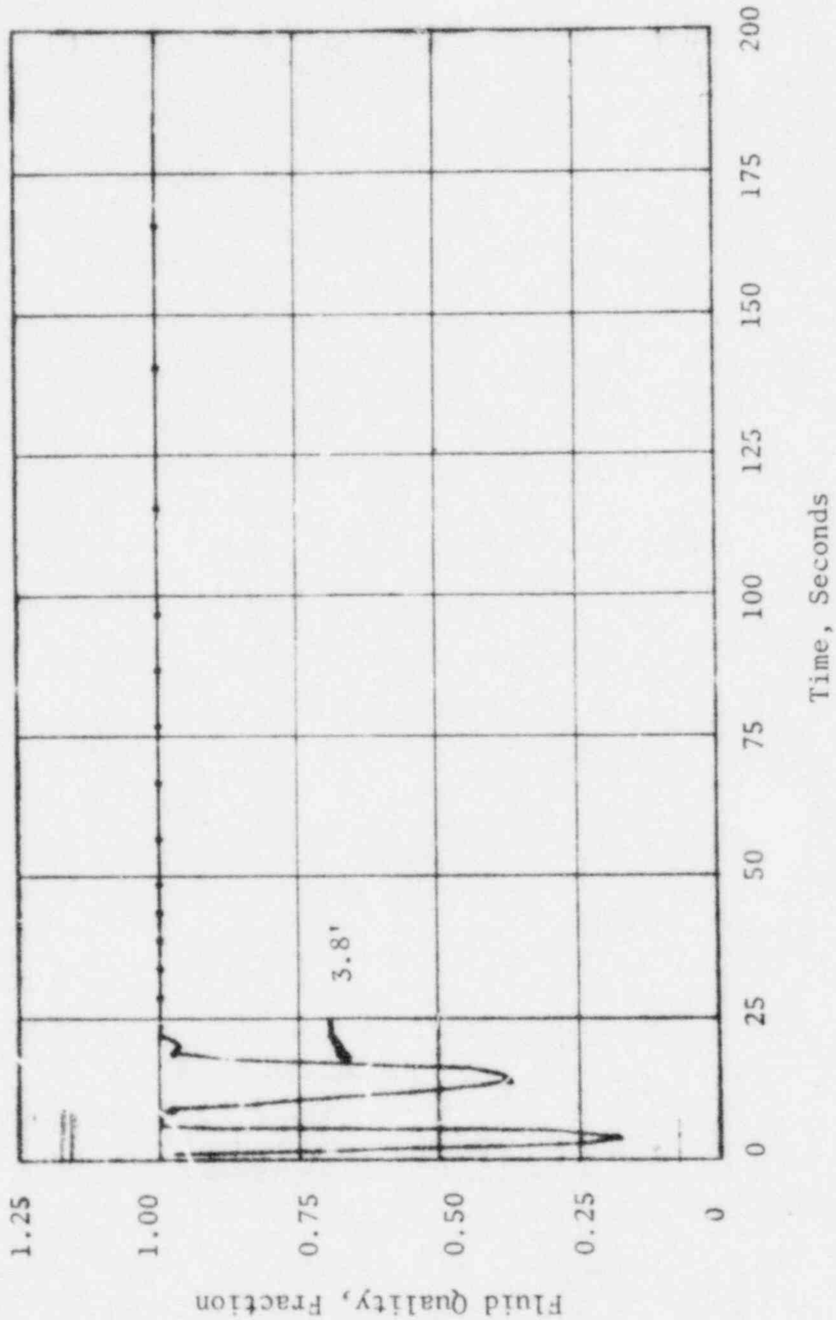
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PUMPED ECCS FLOW (REFLOOD) DECLG (CD=0.6)	FIGURE 411-23B

411:49
8/2/74
DRAFT

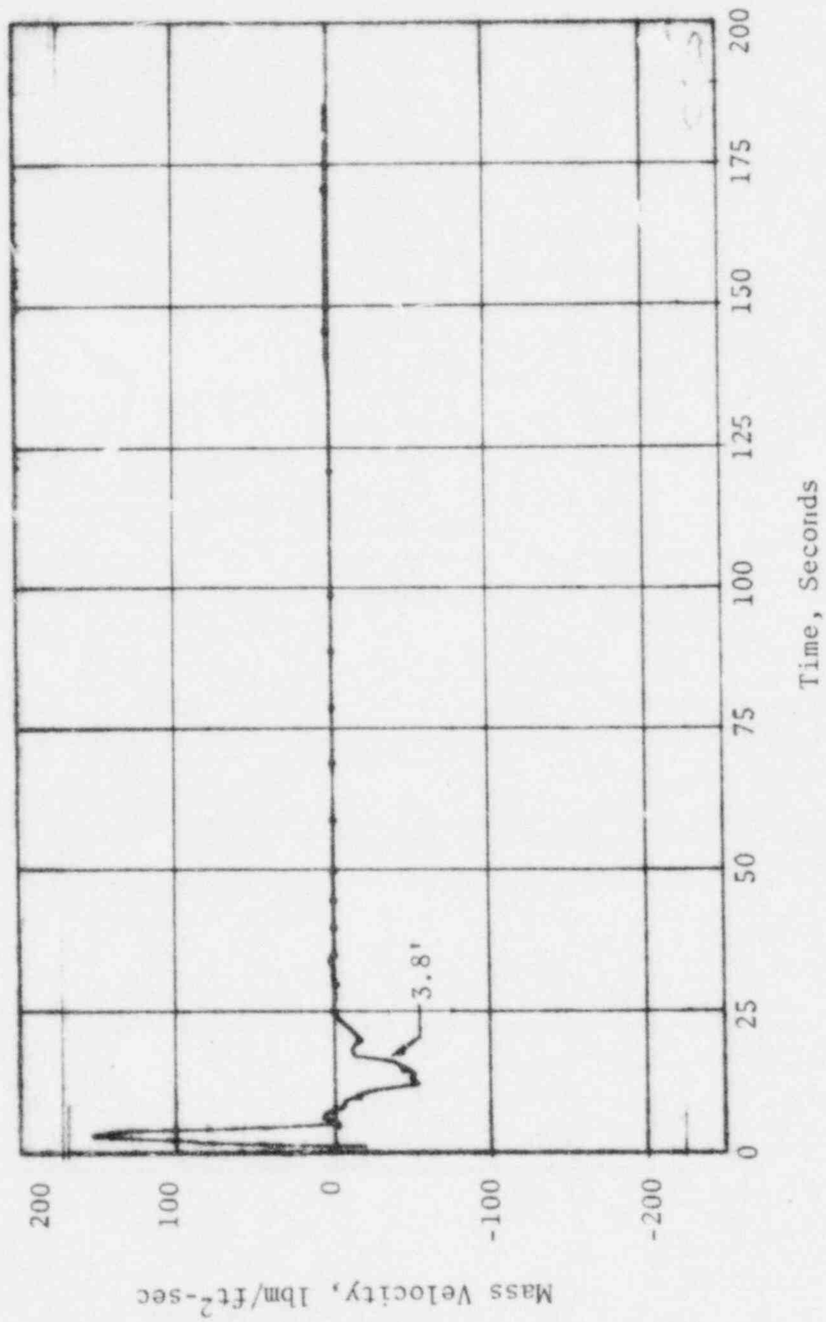


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CONTAINMENT PRESSURE VS. TIME DECLG (CD=0.6)	FIGURE 411-24B

411.50
8/2/74
DRAFT

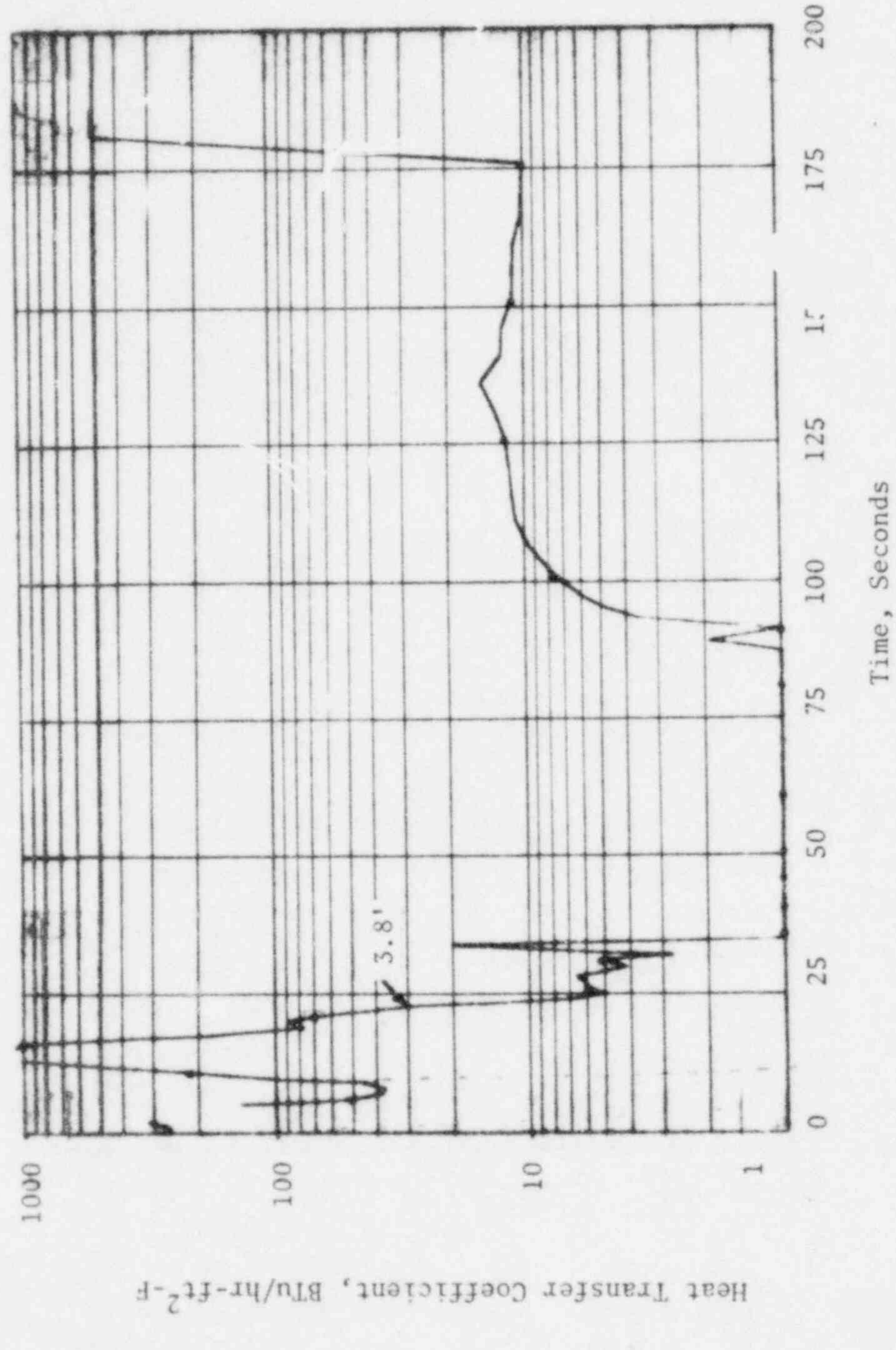


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT QUALITY VS. TIME 0.6 x DECLS	FIGURE 411-12C



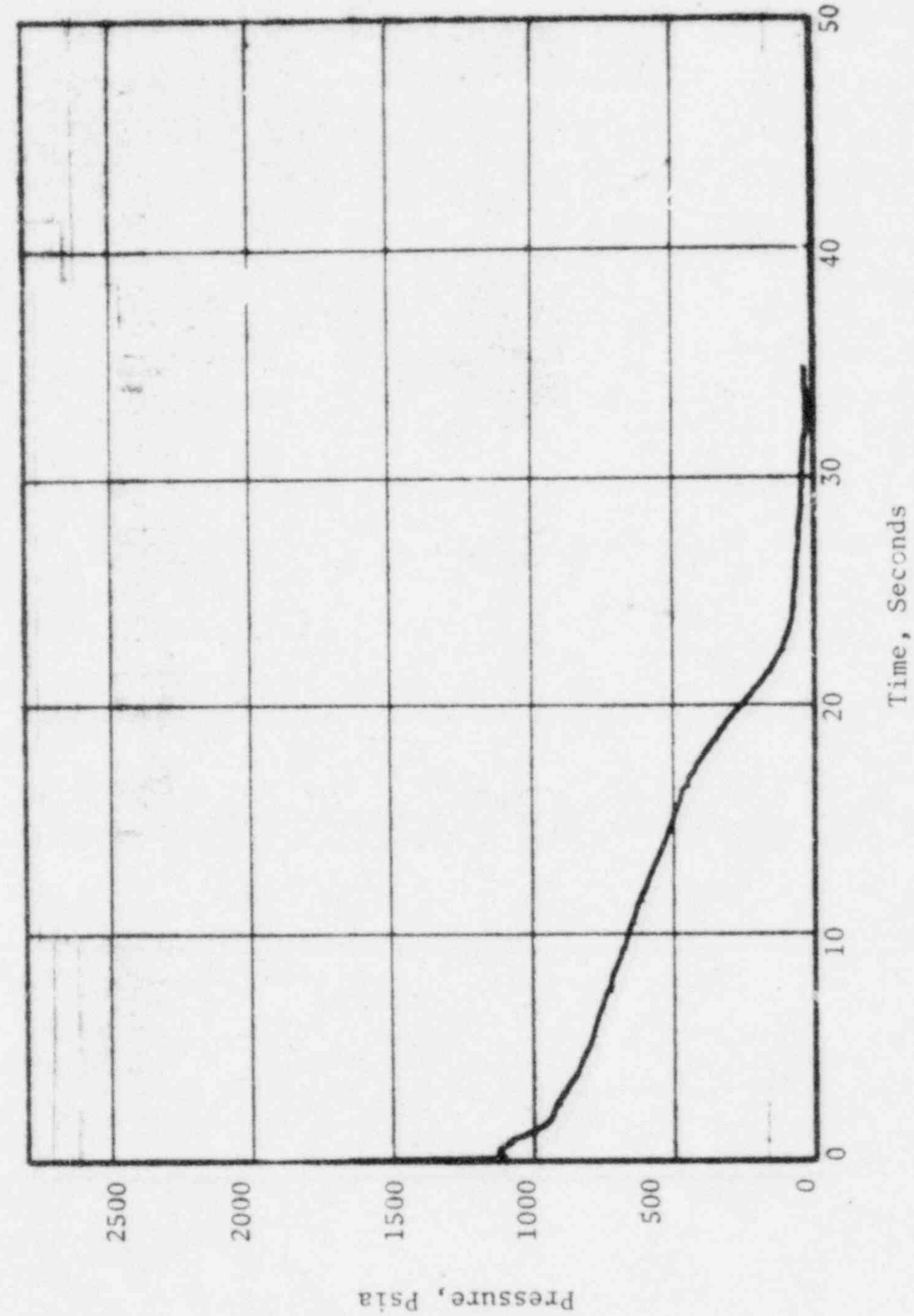
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT MASS VELOCITY VS. TIME 0.6 x DECLS	FIGURE 411-13C

411:52
8/2/74
DRAFT



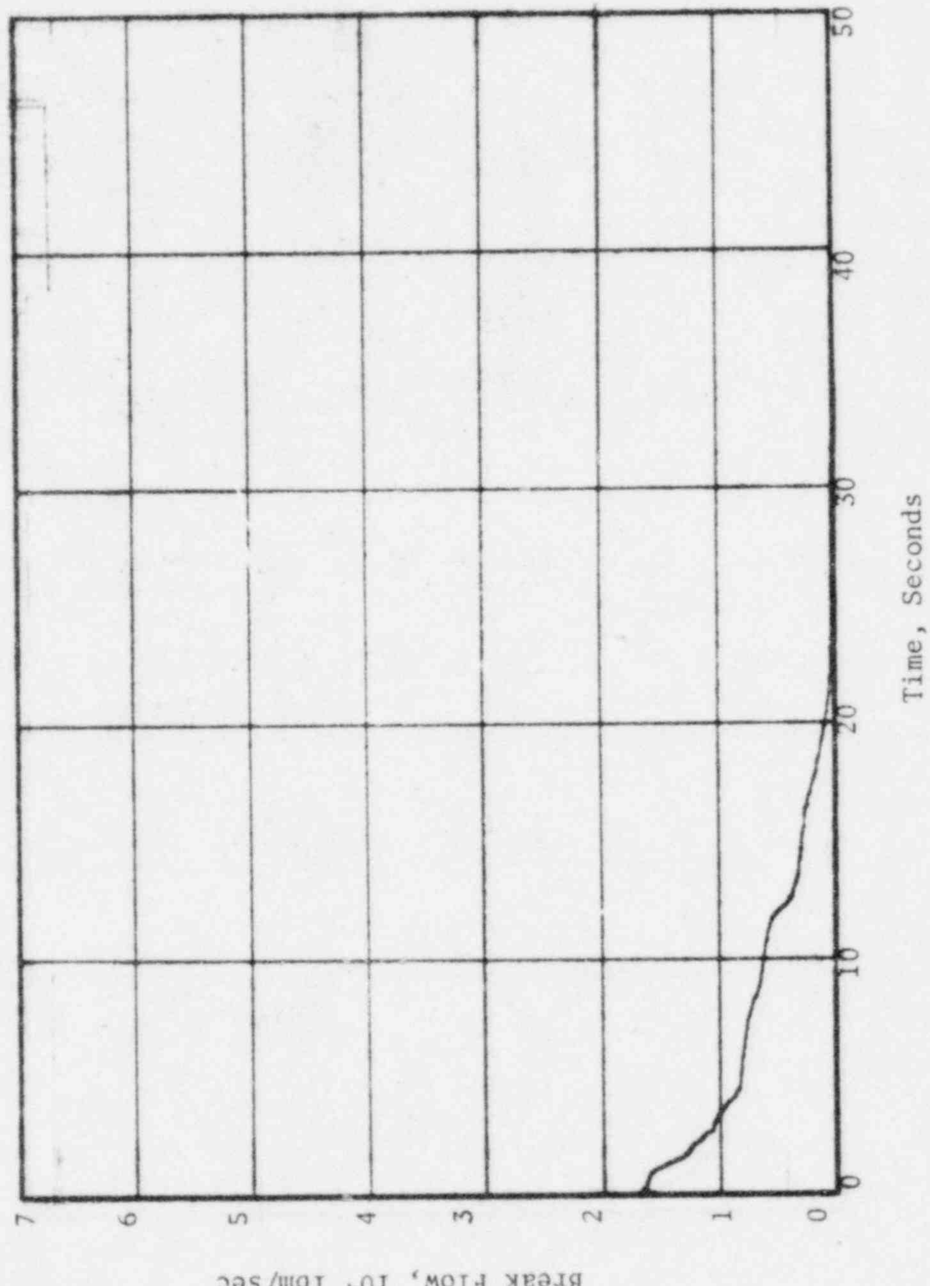
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FILM COEFFICIENT VS. TIME 0.6 x DECLS	FIGURE 411-14C

411.53
8/2/74
DRAFT



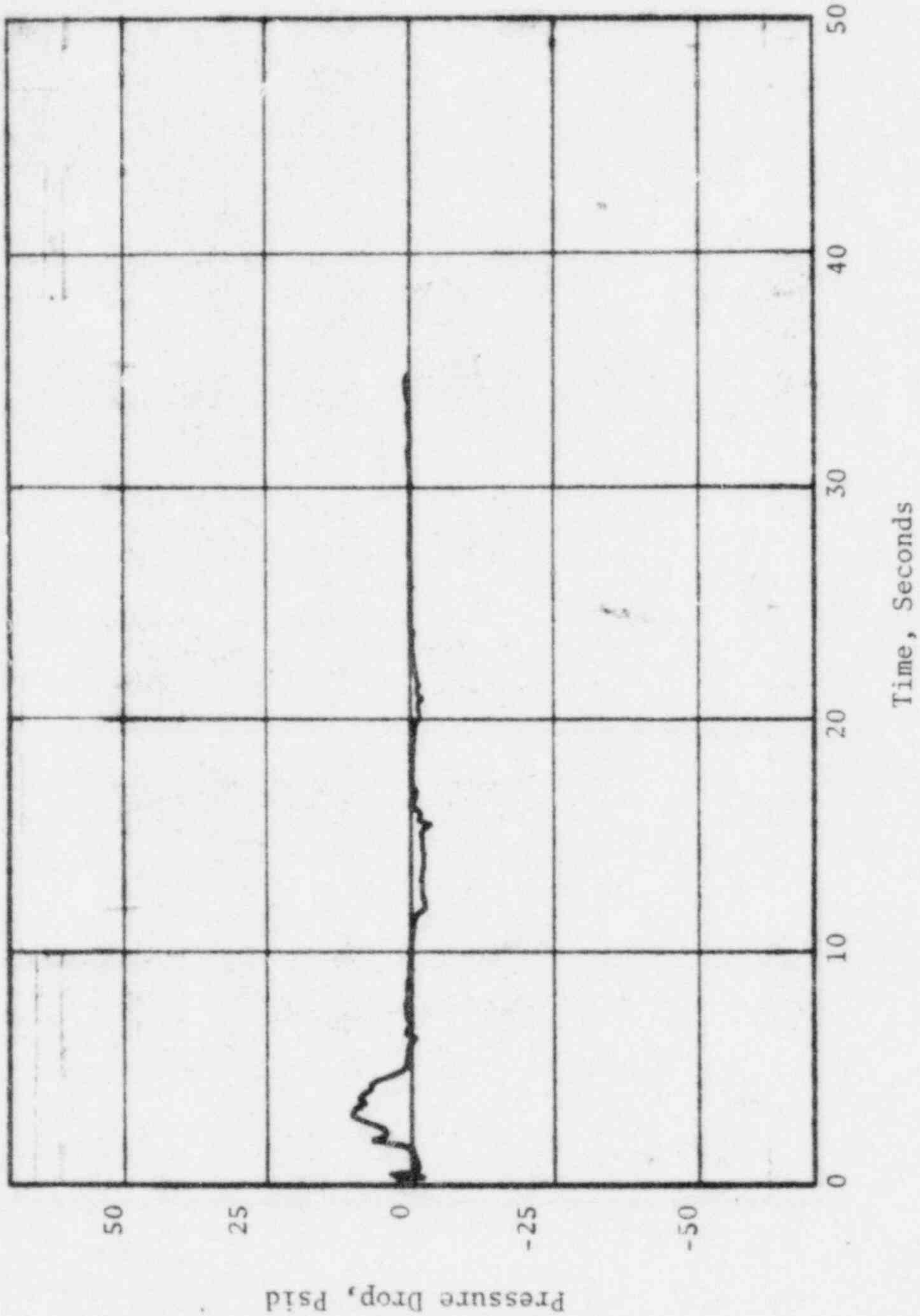
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE VS. TIME 0.6 x DECLS	FIGURE 411-15C

411.54
8/2/74
DRAFT



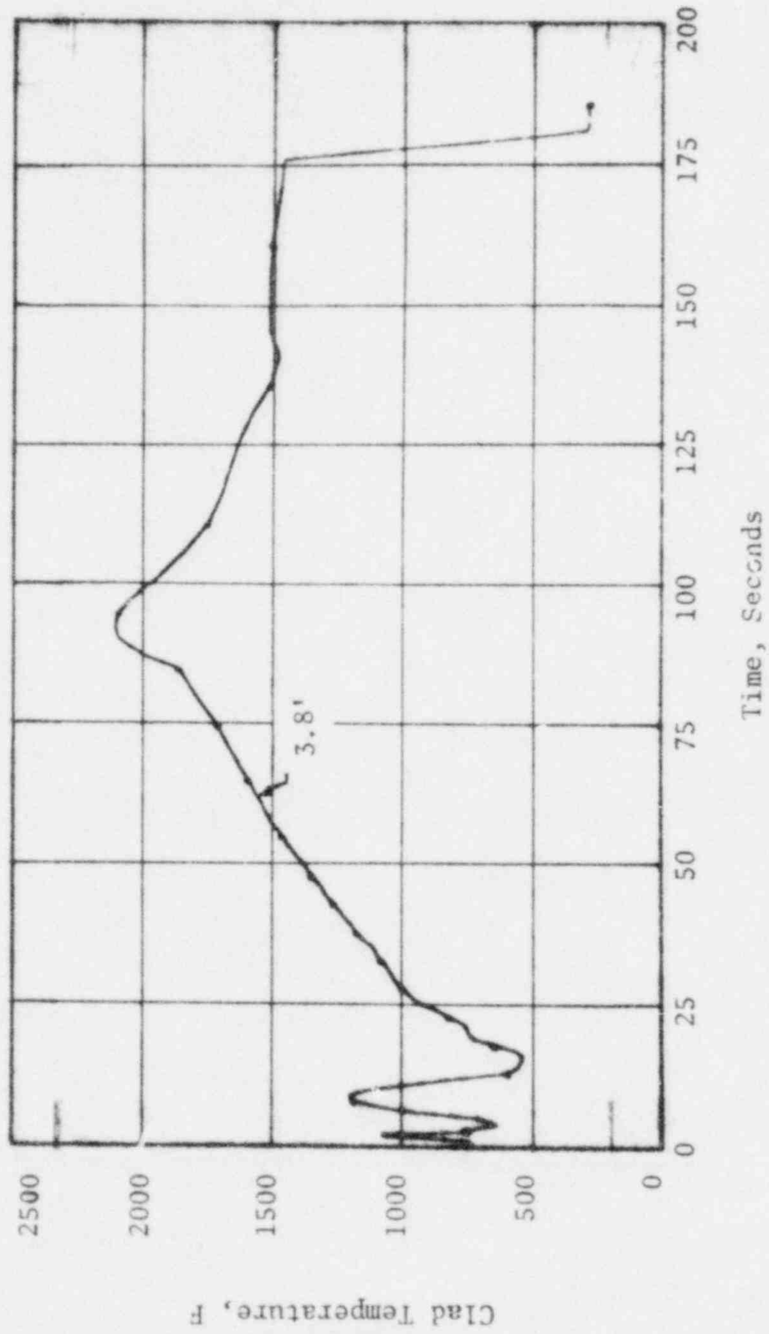
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT BREAK FLOWRATE VS. TIME 0.6 x DECLS	FIGURE 411-16C

411'.55
8/2/74
DRAFT



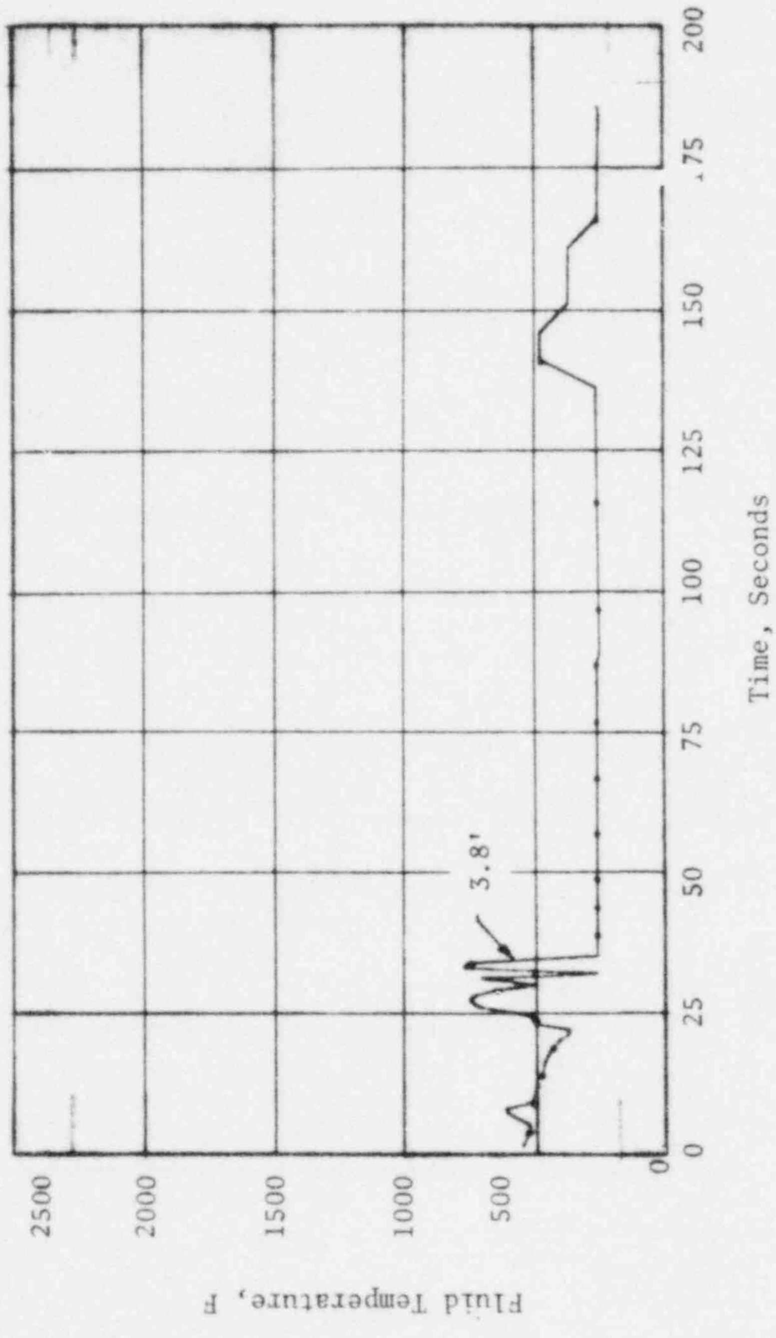
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE DROP VS. TIME 0.6 x DECLS	FIGURE 411-17C

411:56
8/2/74
DRAFT



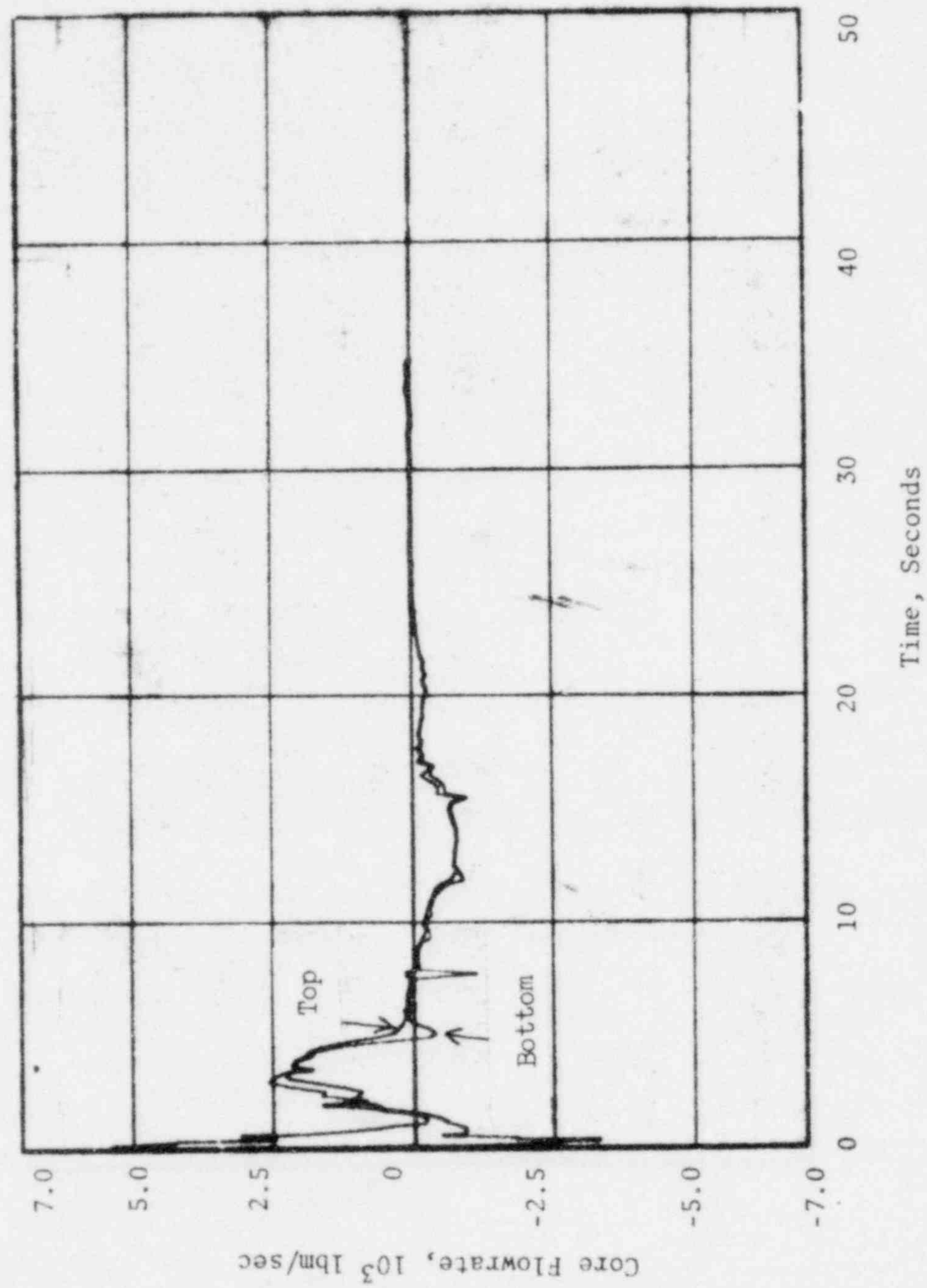
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT CLAD TEMPERATURE VS. TIME 0.6 x DECLS	FIGURE 411-18C

411:57
8/2/74
DRAFT



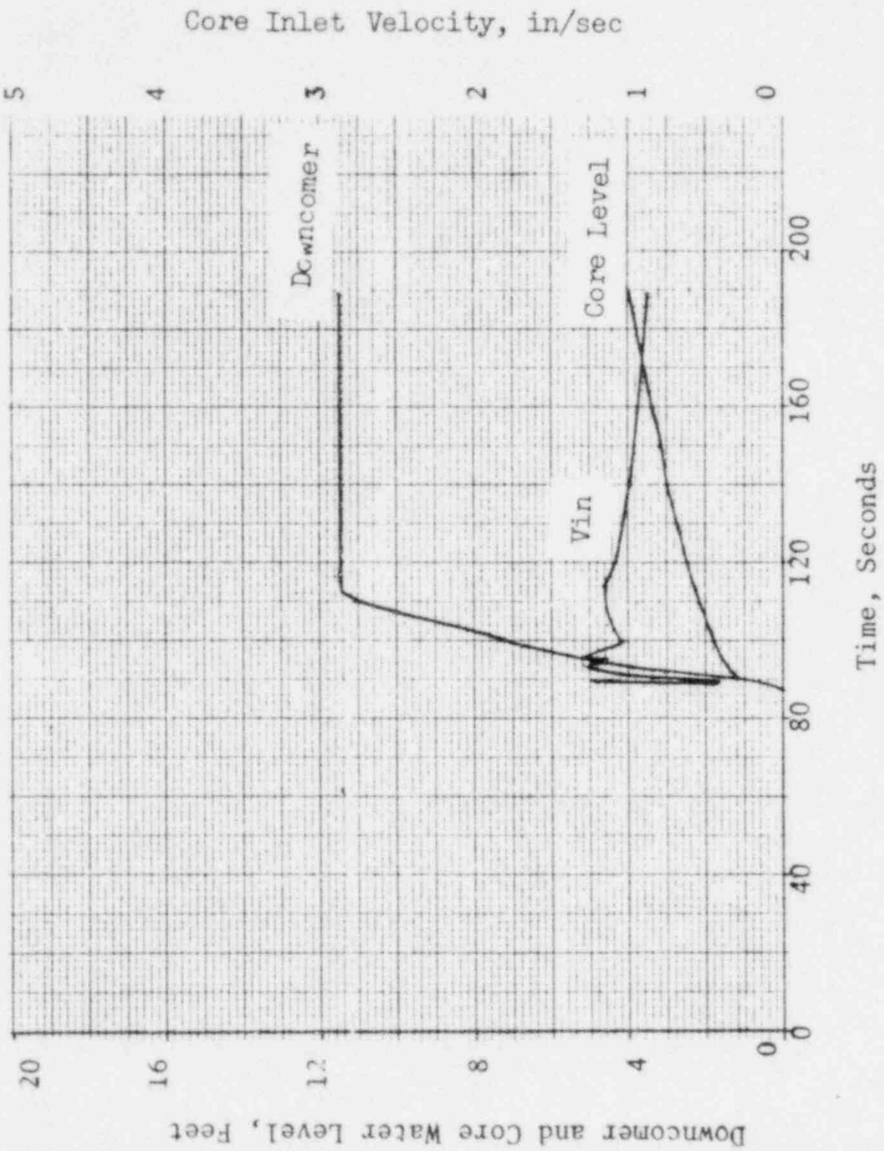
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FLUID TEMPERATURE VS. TIME 0.6 x DECLS	FIGURE 411-19C

411.58
8/2/74
DRAFT



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE FLOW (TOP AND BOTTOM) VS. TIME 0.6 x DECLS	FIGURE 411-20C

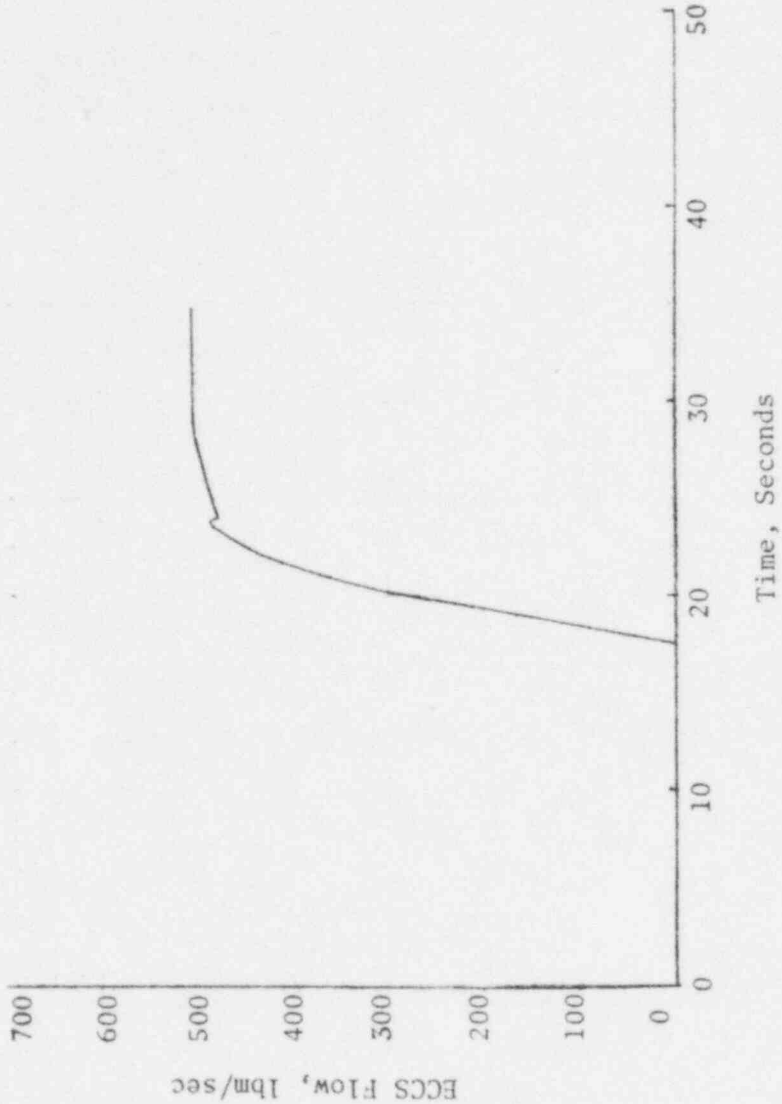
411:59
8/2/74
DRAFT



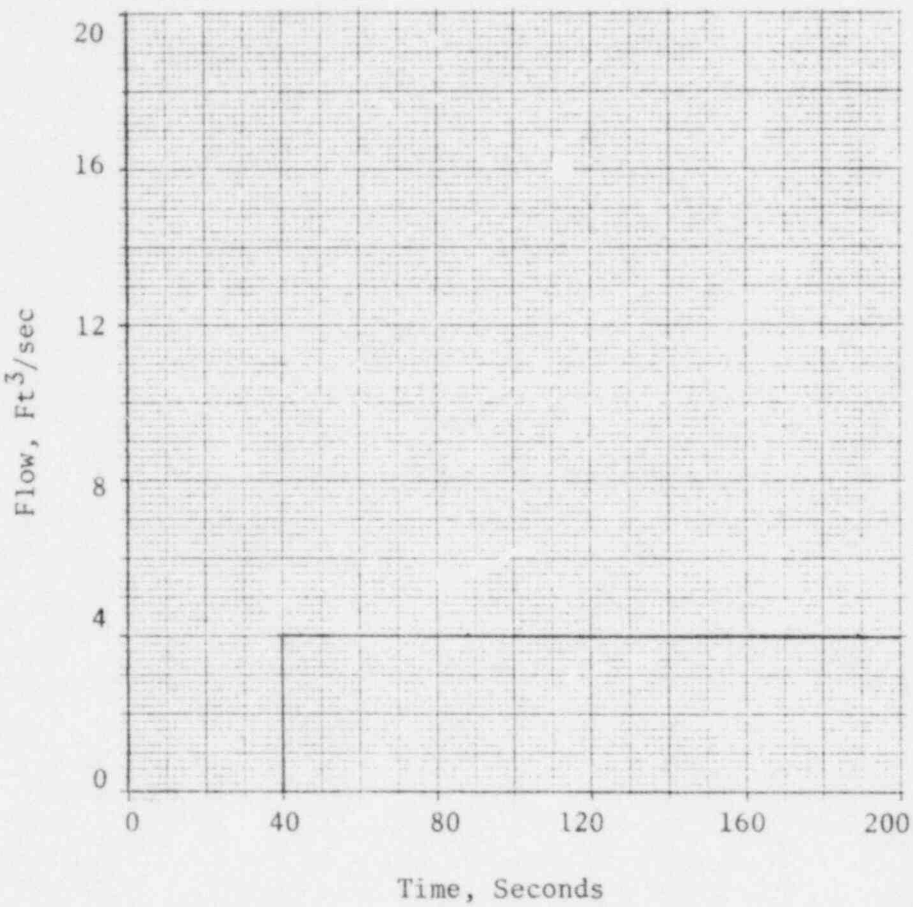
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE REFLOODING VS. TIME 0.6 x DECLS	FIGURE 411-21C

411:60

8/2/74
DRAFT

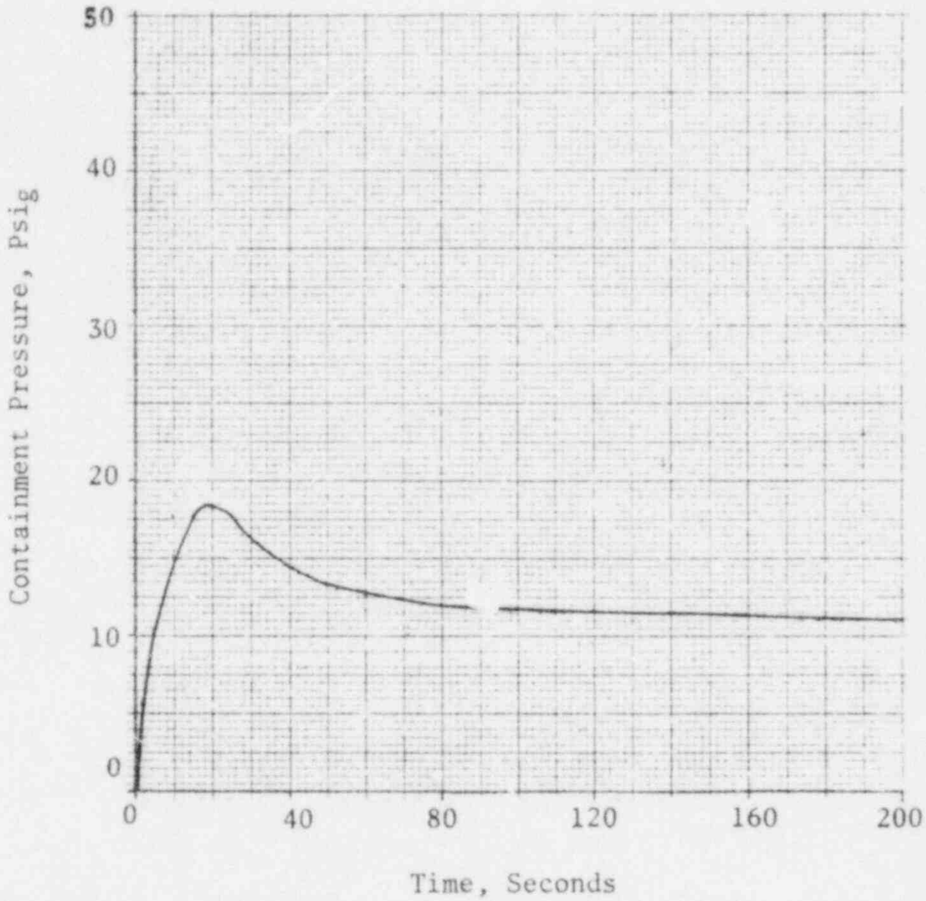


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW (BLOWDOWN) 0.6 x DECLS	FIGURE 411-22C



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PUMPED ECCS FLOW (REFLOOD) 0.6 x DECLS	FIGURE 411-23C

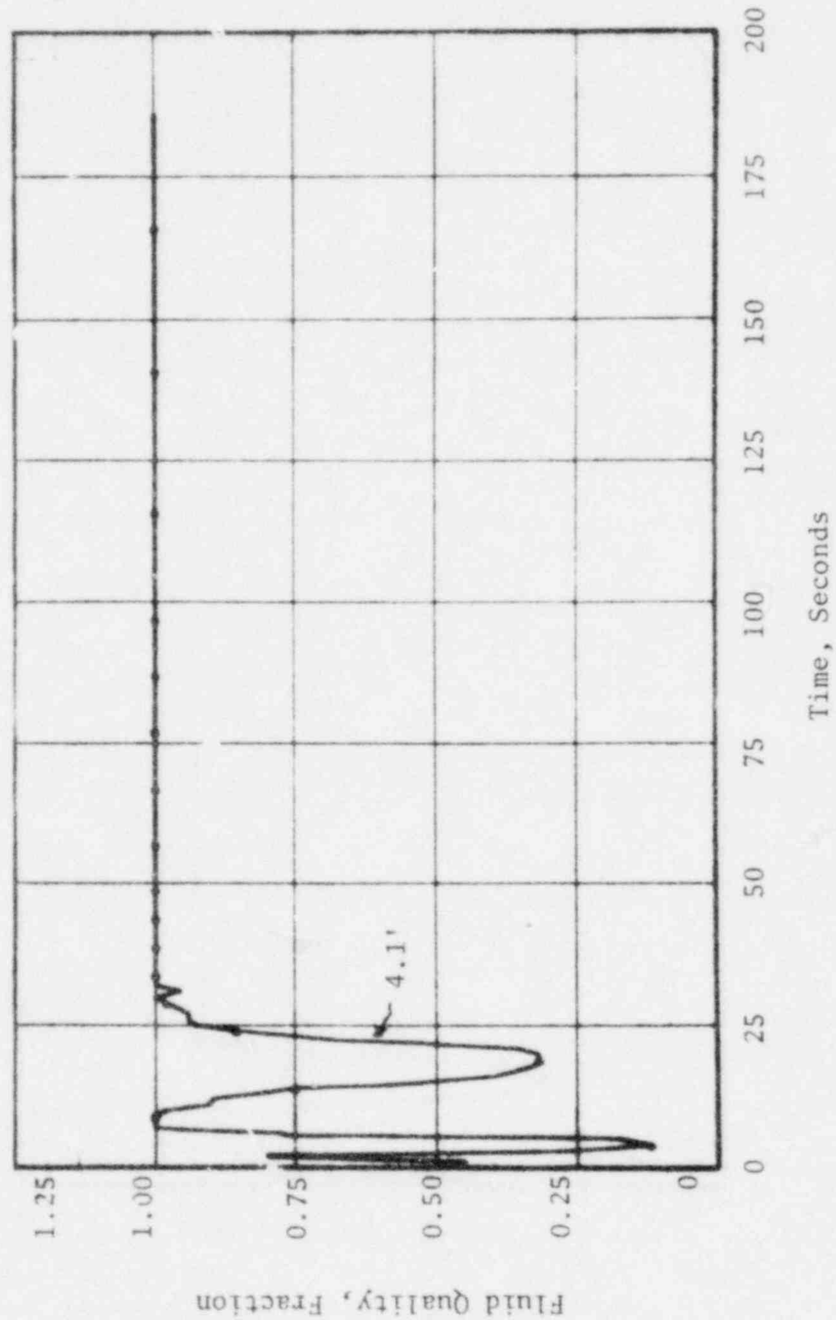
411.62
8/2/74
DRAFT



POOR ORIGINAL

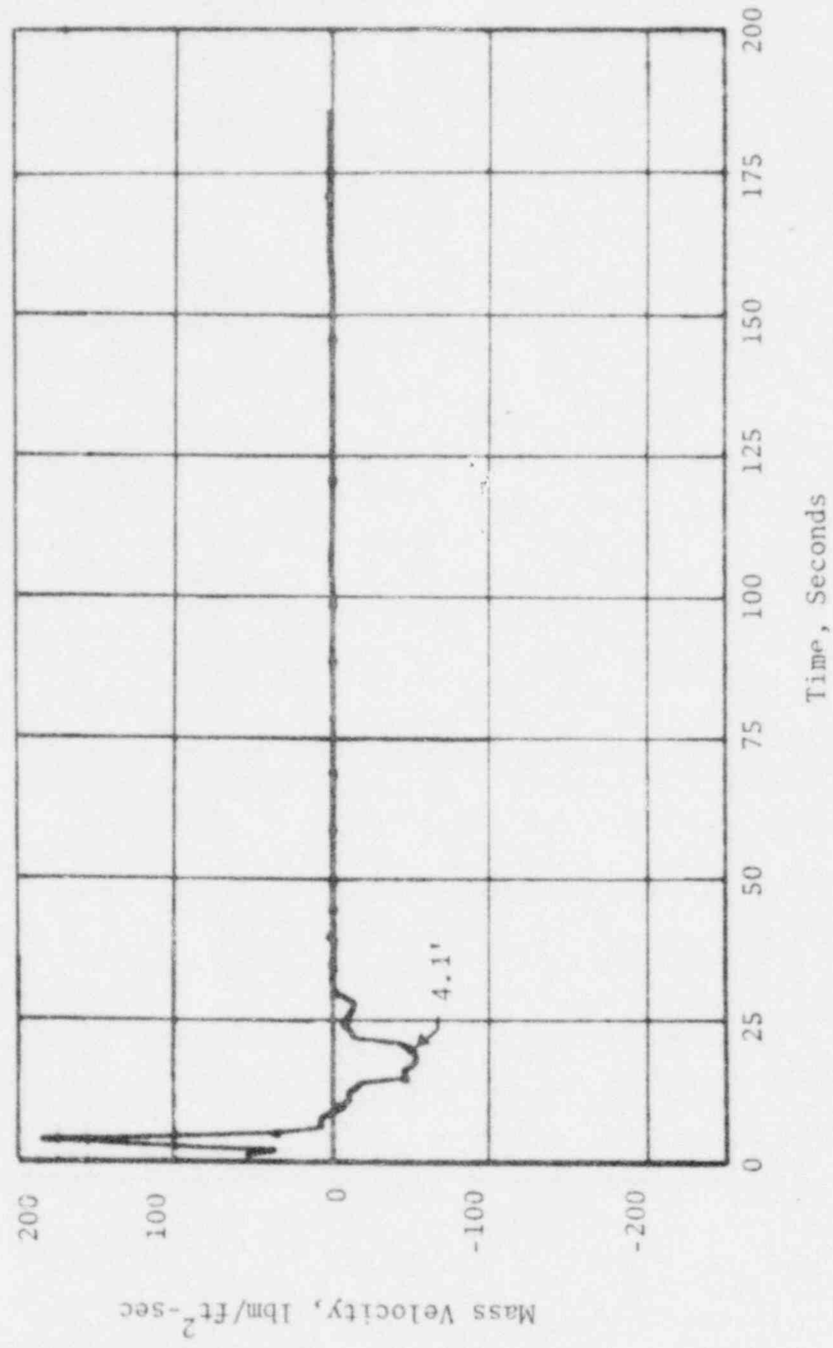
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CONTAINMENT PRESSURE VS. TIME 0.6 x DECLS	FIGURE 411-24C

411:63
8/2/74
DRAFT



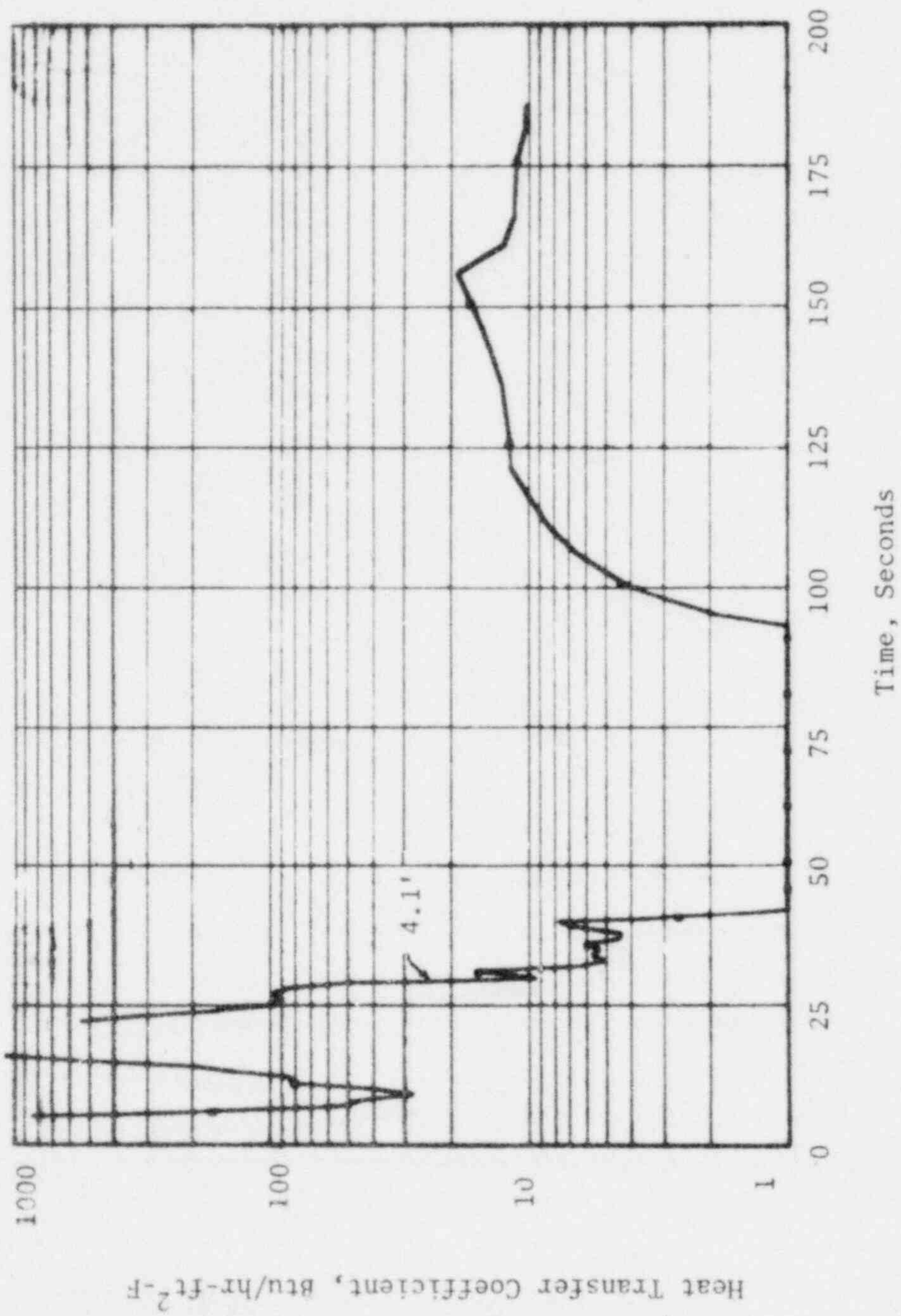
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT QUALITY VS. TIME 1.0 ft^2 SLOT	FIGURE 411-12D

411.64
8/2/74
DRAFT



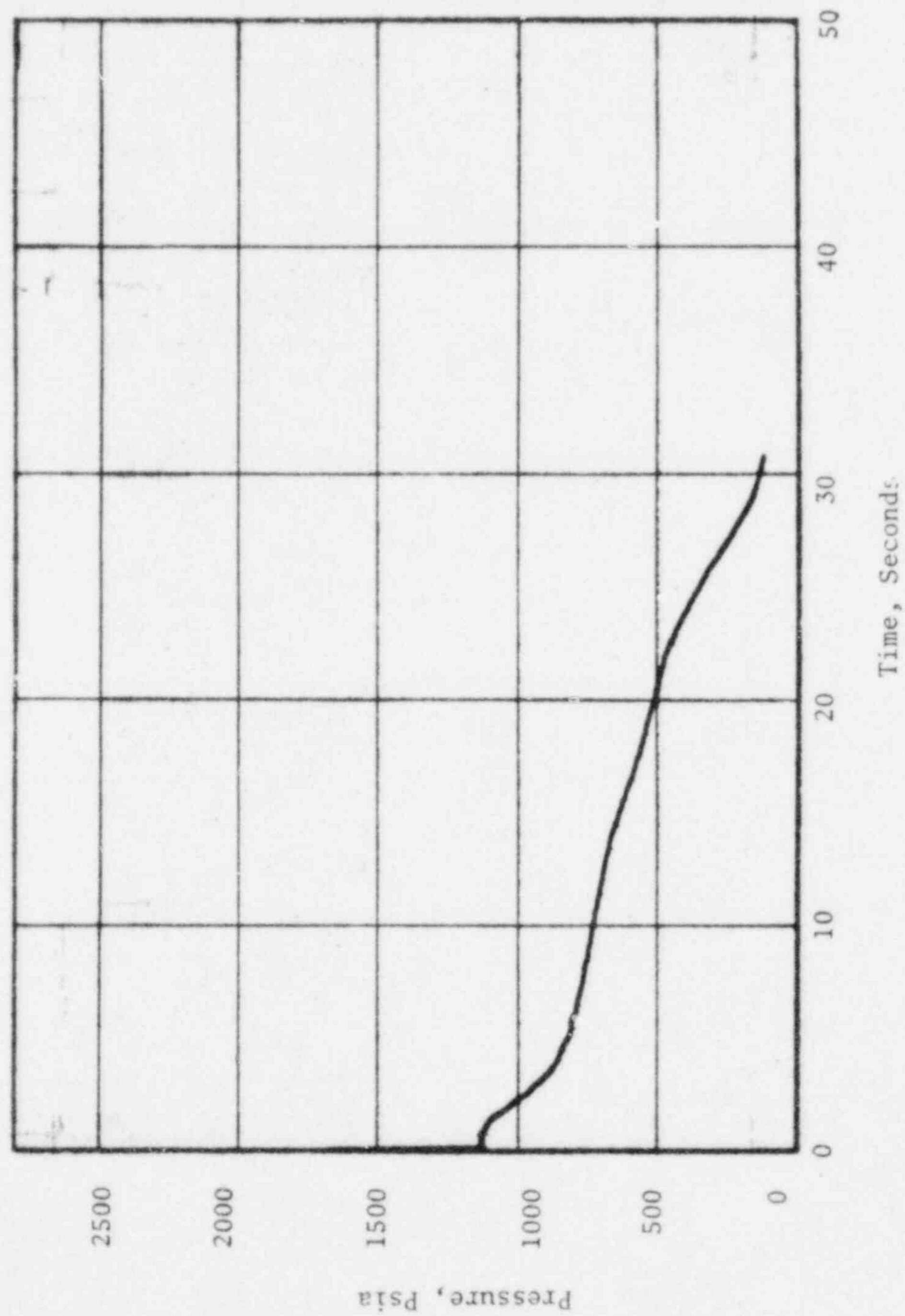
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT MASS VELOCITY VS. TIME 1.0 ft2 SLOT	FIGURE 411-13D

411:65
8/2/74
DRAFT



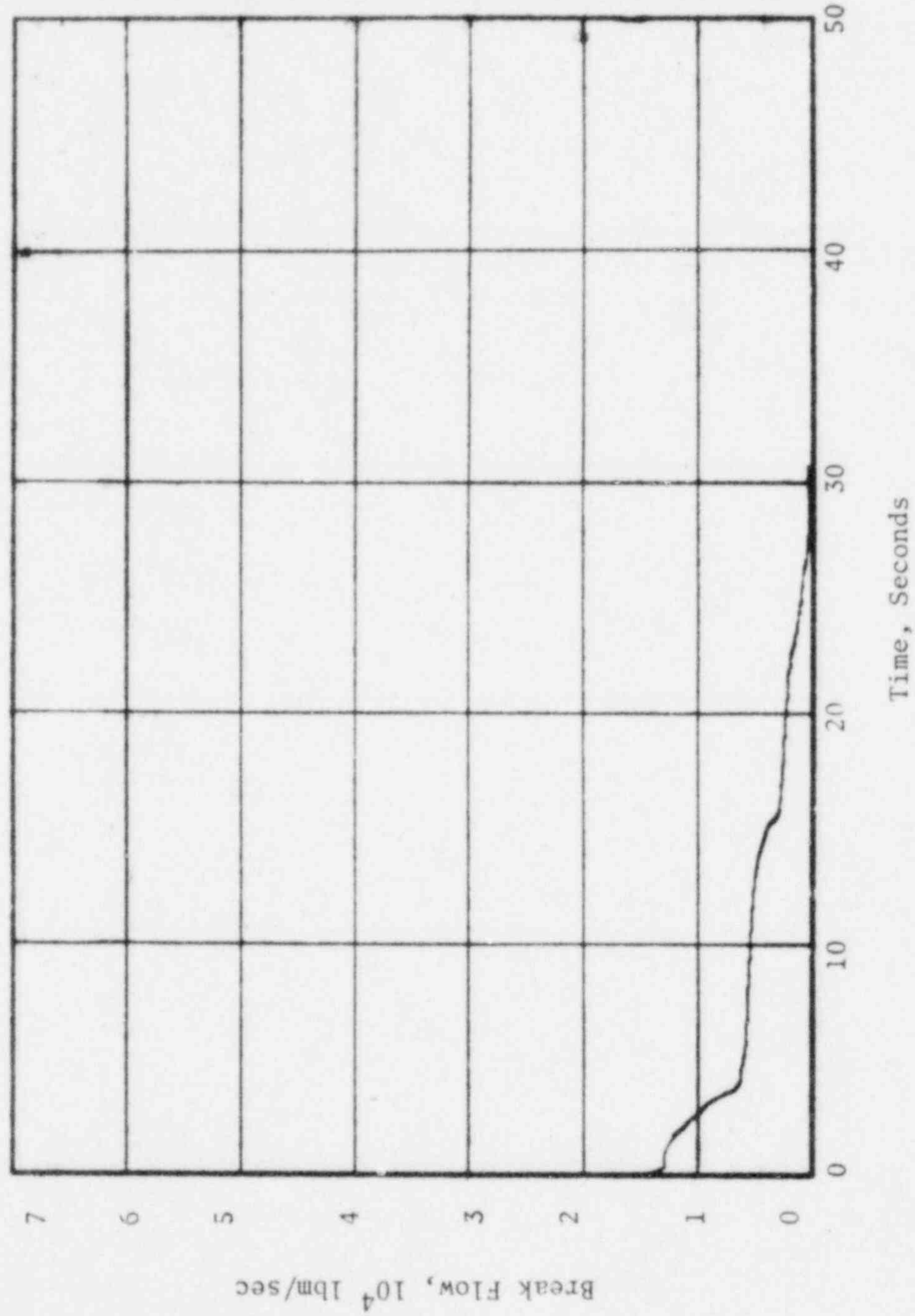
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FILM COEFFICIENT VS. TIME 1.0 ft^2 SLOT	FIGURE 411-14D

411:66
8/2/74
DRAFT



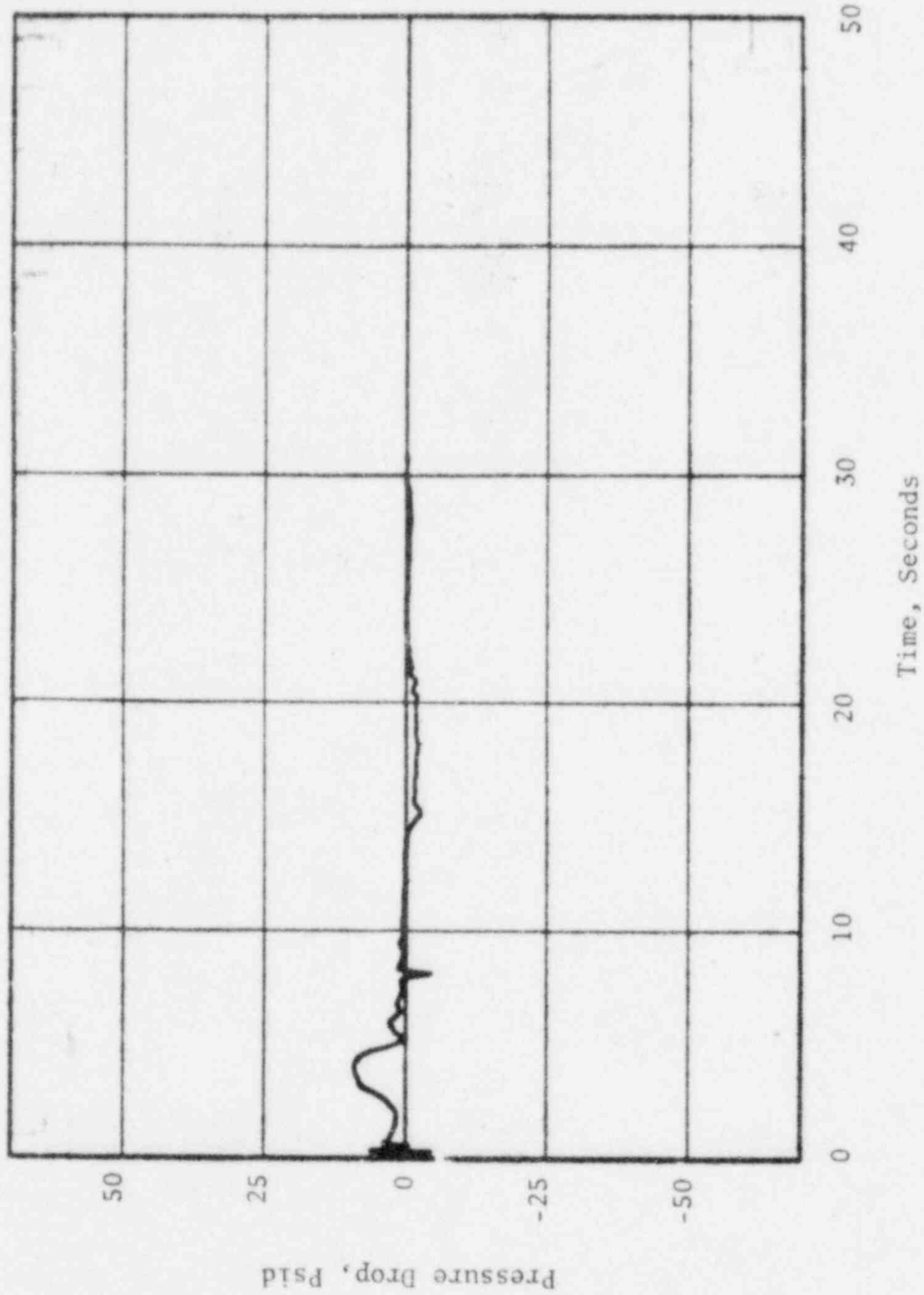
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE VS. TIME 1.0 ft^2 SLOT	FIGURE 411-15D

411:67
8/2/74
DRAFT



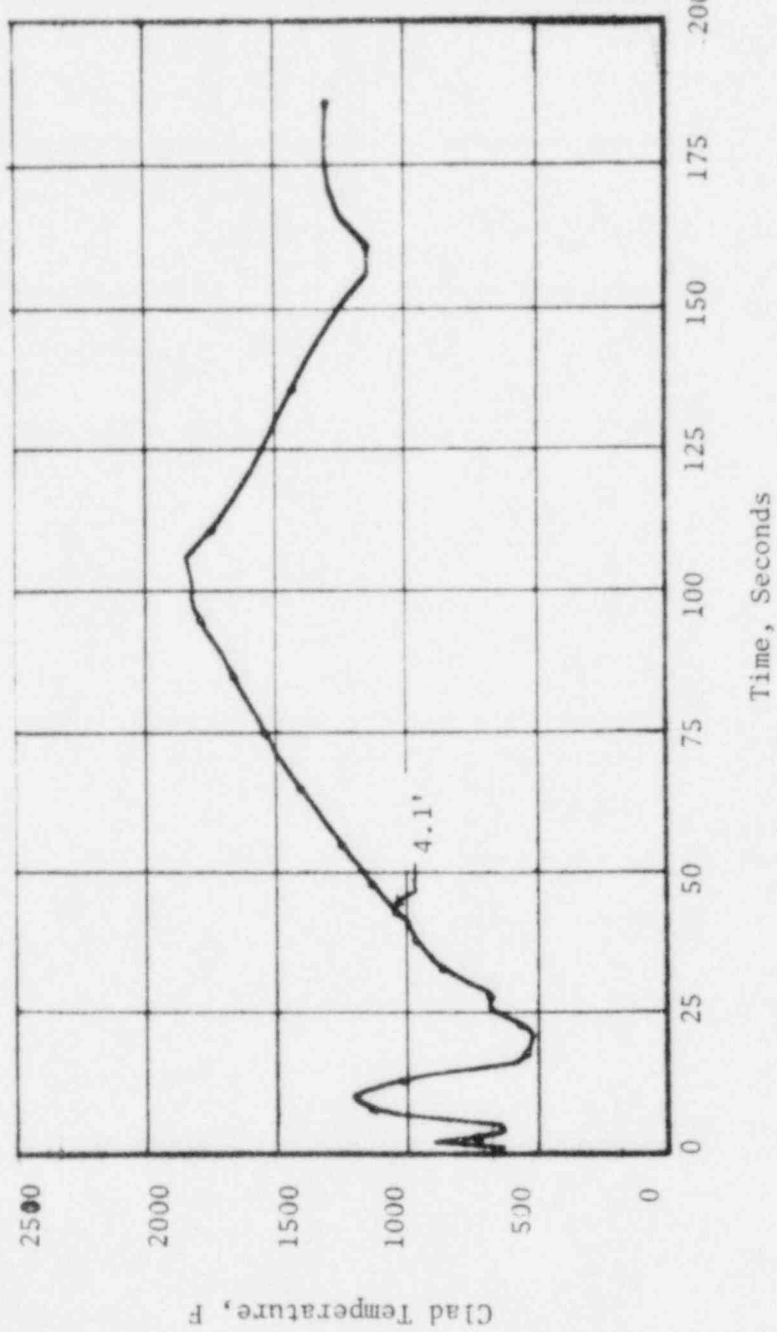
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT BREAK FLOWRATE VS. TIME 1.0 ft^2 SLOT	FIGURE 411-16D

411:68
8/2/74
DRAFT



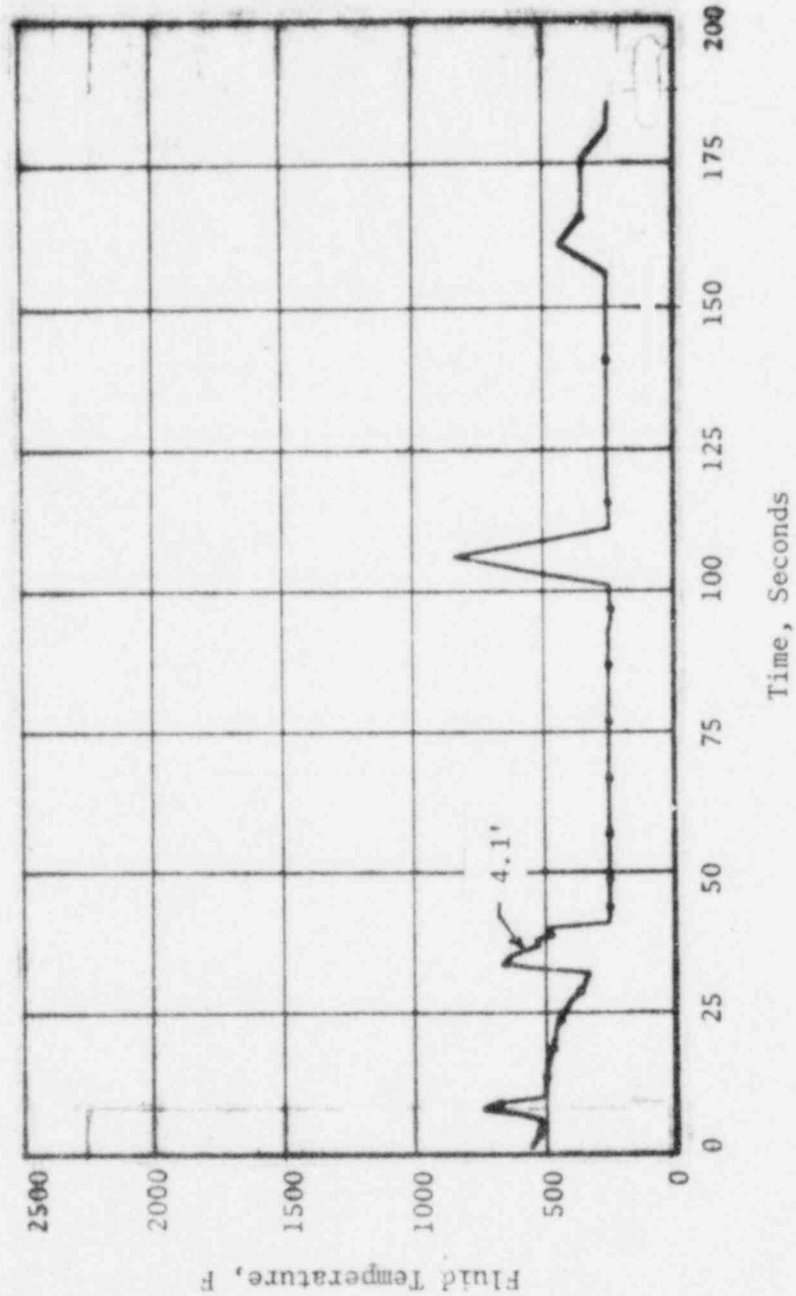
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE PRESSURE DROP VS. TIME 1.0 ft^2 SLOT	FIGURE 411-17D

411:69
8/2/74
DRAFT



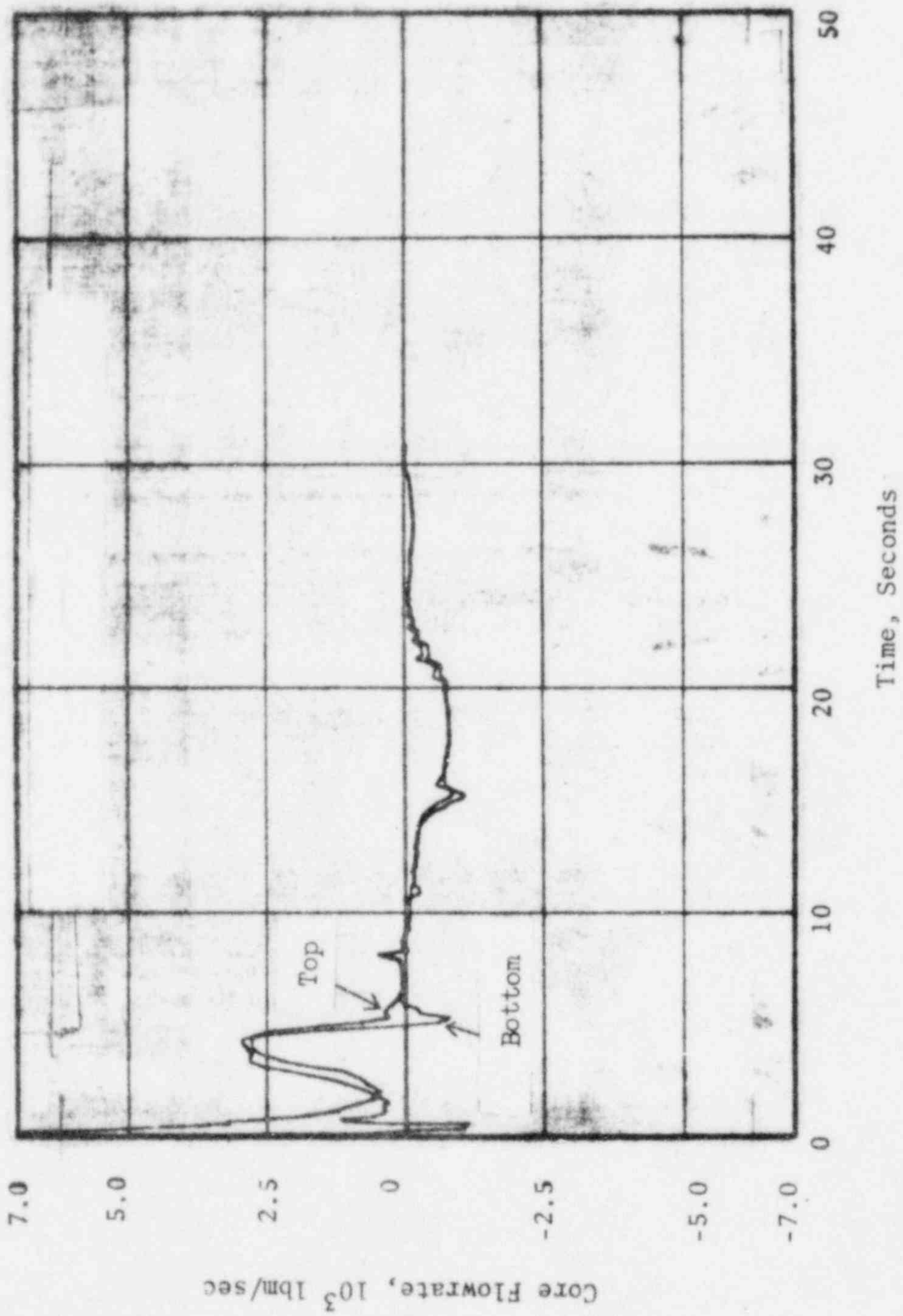
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT CLAD TEMPERATURE VS. TIME 1.0 ft^2 SLOT	FIGURE 411-18D

411:70
8/2/74
DRAFT



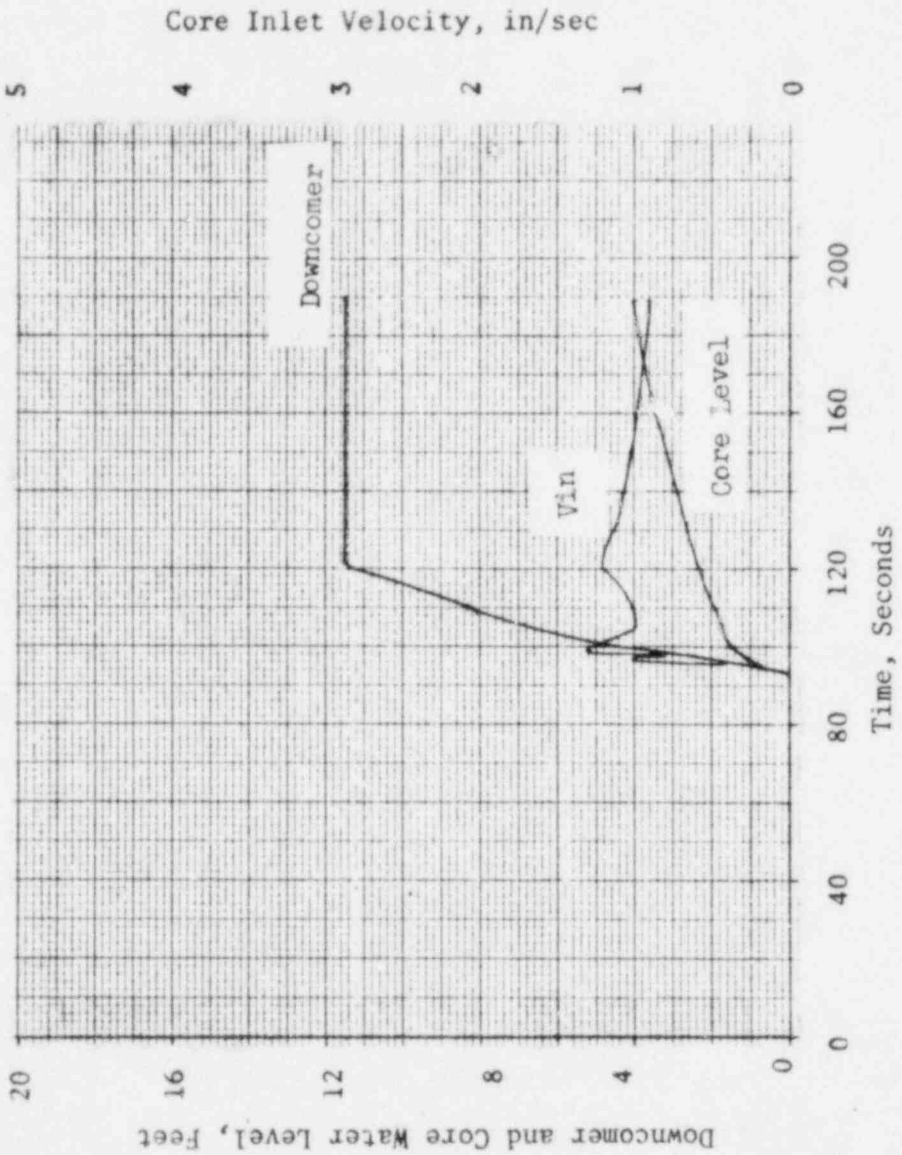
YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT HOT SPOT FLUID TEMPERATURE VS. TIME 1.0 ft^2 SLOT	FIGURE 411-19D

411:71
8/2/74
DRAFT



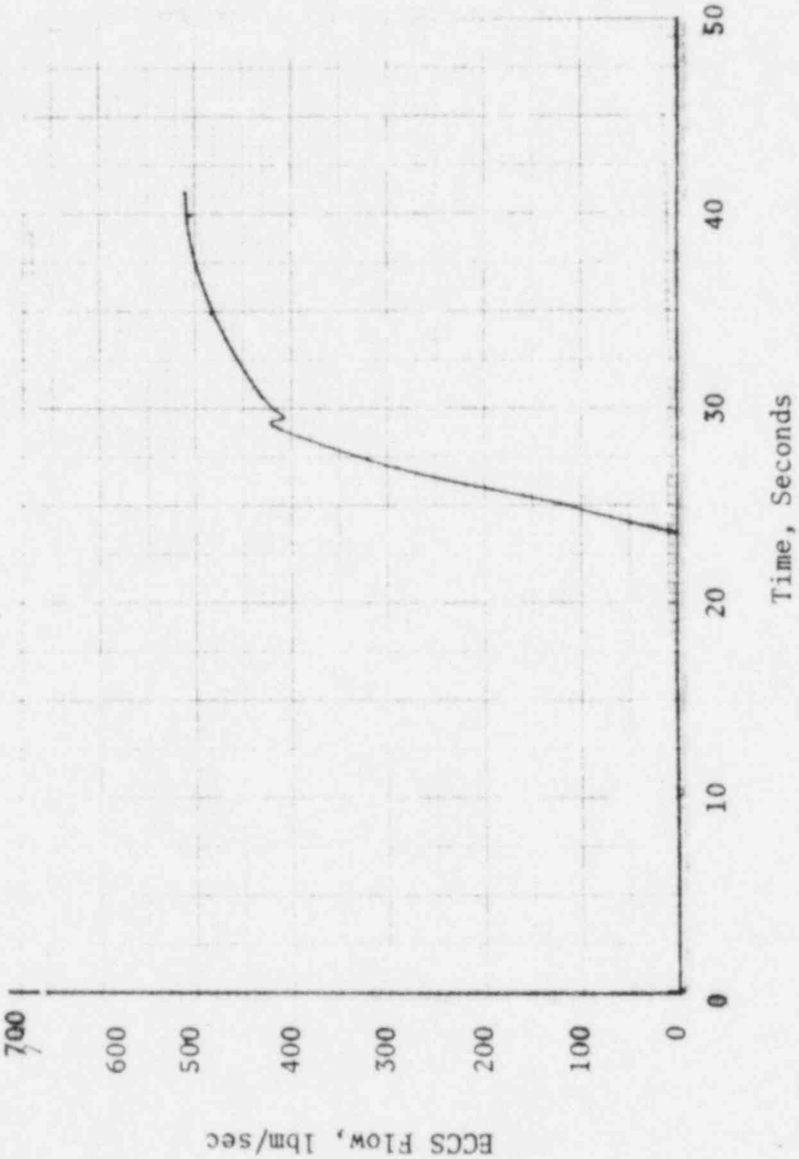
POOR ORIGINAL

YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE FLOW (TOP AND BOTTOM) VS. TIME 1.0 ft^2 SLOT	FIGURE 411-20D

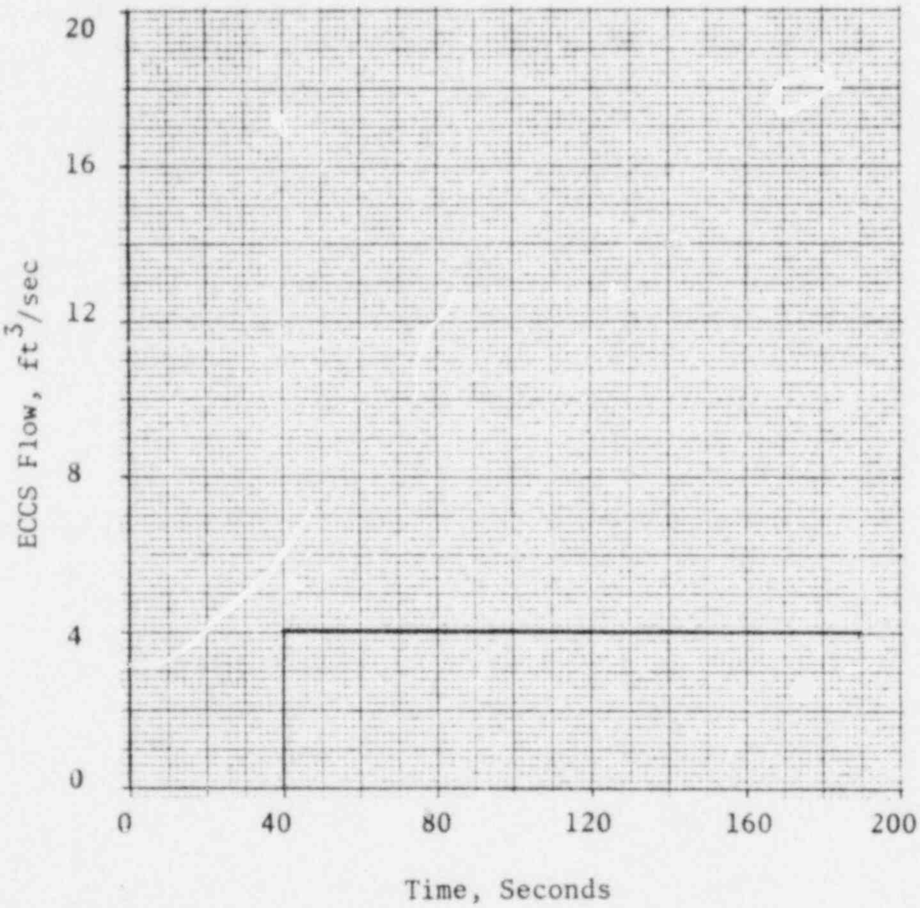


YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CORE REFLOODING VS. TIME 1.0 ft^2 SLOT	FIGURE 411-21D

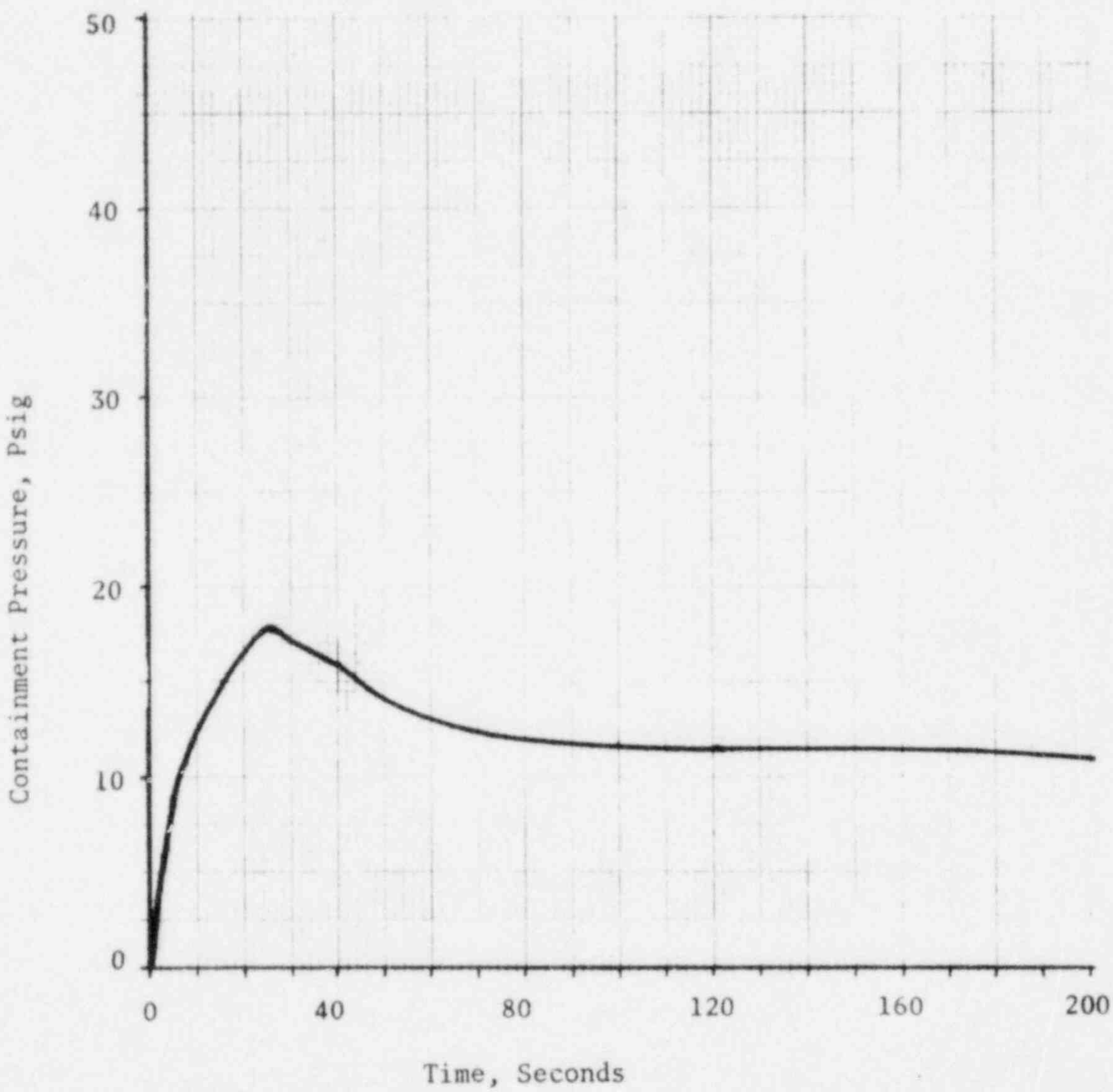
411:73
8/2/74
DRAFT



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT ECCS FLOW (BLOWDOWN) 1.0 ft^2 SLOT	FIGURE 411-22D



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT PUMPED ECCS FLOW (REFLOOD) 1.0 ft^2 SLOT	FIGURE 411-23D



YANKEE NUCLEAR POWER STATION	LOSS OF COOLANT ACCIDENT CONTAINMENT PRESSURE VS. TIME 1.0 ft^2 SLOT	FIGURE 411-24D