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Exhibit B-6

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

SIXTH SUPPLEMENT TO:
PRELIMINARY SAFETY ANALYSIS REPORT

1235

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FINAL ISSUE

PRELIMINARY ISSUE

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1.0 PREFACE

In its letter of August 16, 1966 on the Indian Point No. 2 facility, the Advisory Committee on Reactor Safeguards recommended to the Commission that certain aspects of the design of applicant's facility be scrutinized both by it and by the Commission during the evolution of the final design of the facility. The material contained in this Supplement constitutes the information requested on the three matters mentioned in the aforesaid letter: increase of emergency core cooling capacity; effect of maximum loss of coolant accident on integrity of reactor core and pressure vessel internal components; and primary system design, fabrication techniques and inservice inspections.

2.0 EMERGENCY CORE COOLING

2.1 INTRODUCTION

The Indian Point Unit No. 2 Safety Injection System has been modified to provide improved core protection for postulated loss-of-coolant accidents for break sizes from the surge line up to the double ended severance of the reactor coolant pipe. The principal change to the system is the addition of a high-flow fast-acting accumulator system to perform the Safety Injection function formerly provided by the two recirculation pumps. Figure 2-1 presents the Process Flow Diagram for the Safety Injection System and shows the addition of the accumulators.

The accumulator system consists of a nitrogen pressurized borated water tank attached to the cold leg of each reactor coolant loop. The accumulators will quickly re-flood the core upon the depressurization of the reactor coolant system. This rapid re-flooding ensures that the fuel elements remain in place and substantially intact.

In addition better analytical techniques now available and described on pages 2-20 to 2-25 have been used to evaluate the loss of coolant blowdown and core thermal transients. This report defines the improved core protection design objectives, provides a complete description of the revised Safety Injection System and the status of the equipment, evaluates the actual performance of the revised system, and describes the new analytical techniques. Special attention has been given to the sensitivity of the analysis to system parameters, moderator coefficient, film heat transfer coefficient, and thermal-mechanical stability of the core.

The Safety Injection System parameters are now set. As will be demonstrated in the report, the performance of the system exceeds the minimum core protection objectives by a substantial margin when the moderator coefficient is negative or zero. This margin extends the capability of the Safety Injection System to meet the core protection objectives with a positive moderator coefficient.

The positive moderator coefficient can be controlled by the addition of fixed shims, if necessary, and during the detail design of the core the necessity for adding fixed shims will be determined on the basis of the core's ability to dissipate the heat generated in the reactivity transient without violating the objectives of the core cooling. Space is available for these shims in the RCC thimbles not used for control clusters.

2.2 SAFETY OBJECTIVES OF DESIGN

The following objectives have been adopted as a basis for the design of the Safety Injection System and for the evaluation of the adequacy of system performance under postulated accident conditions.

2.2.1 ACCUMULATOR FUNCTION

The accumulators shall be designed to provide sufficient injection of borated water following a large-area rupture of the Reactor Coolant System to keep the fuel elements in place and substantially intact. Large area ruptures are defined for the purpose of this statement of objectives as those larger than the largest connecting pipe to the reactor coolant system up to and including the double ended severance of the reactor coolant pipe. For smaller breaks protection is afforded by the operation of one or more residual heat pumps and two or more safety injection pumps, and the associated valves, control systems and power supplies, when triggered by the safety injection signal. The important safety criterion is that UO_2 must be retained essentially in the original configuration although clad rupture and localized Zirconium-water reaction are not necessarily precluded. The objective will be considered met if the calculated fuel clad temperatures for all of the core is less than the melting point when the core is in its original geometry.

The accumulator operation shall be so designed as to be self-energized; that is, no external source of power or signal transmission shall be required in order to accomplish its design function.

The accumulator status relating to its readiness to perform as designed shall be continuously monitored in the central control room. All functions requiring mechanical action, except the operation of the check valves at the loop connection shall be capable of testing under conditions of reactor power operation. The operation of the entire system including these check valves shall be possible with the reactor at cold shutdown.

2.2.2 PUMP FUNCTION

Pumps which are called upon to operate to fulfill core cooling objectives are designed to provide equal or greater protection against fuel damage than that defined above, for all break sizes up to and including the severance of the pressurizer surge line. In the evaluation of this performance it shall be assumed that the accumulators function according to design; in the event that system pressure falls below accumulator pressure in a postulated accident, account will be taken of the flow of borated water into the system from the accumulators according to the pressure differential and pipe line resistance. However, it will be assumed that only those components are operable which can be supplied by two of the three diesel generators in conjunction with the remaining engineered safeguard loads, as set forth in the Preliminary Safety Analysis. Further, the system must be tolerant of a failure at the time of the accident of any single active component to respond to the safety injection signal.

Pumps actuated by the safety injection signal supplement the accumulator function by making up for evaporation and spillage of coolant after accumulator discharge. This function generally will not govern the head or flow parameters for any pump, but may influence the mode of operation under specific circumstances, as for example during the change-over from injection to recirculation cooling mode.

The minimum performance of accumulators and pumps required to meet the safety objectives is described in Section 2.4.

2.3 SYSTEM DESCRIPTION

2.3.1 GENERAL DESCRIPTION

The safety injection system arrangement is shown on Figure 2-1.

The principal components of the safety injection system which provide emergency core cooling immediately following a loss of coolant are the four accumulator tanks, the three high head safety injection pumps, and the two low head residual heat removal pumps of the auxiliary coolant system. The high-head safety injection pumps and residual heat removal pumps are located in the auxiliary building and take suction directly from the refueling water storage tank located adjacent to the auxiliary building.

The accumulator tanks and the residual heat removal pumps discharge into the cold legs of the reactor coolant piping, thus assuring core cooling by rapidly restoring the water level to a point above the top of the core for large breaks.

The safety injection pumps deliver borated water to the hot and cold legs of the reactor coolant loops. These pumps augment the flow-pressure characteristics of the accumulator tanks and residual heat removal pumps, providing specifically for the makeup of coolant following a small beak which does not immediately depressurize the reactor coolant system to the accumulator cut-in pressure.

The design capacity of the accumulator tanks is based on one of the four tanks spilling and the remaining three containing sufficient water to fill the volume outside the core barrel below the nozzles, the bottom plenum and one half the core. The accumulator tanks are located inside the containment. The location of each tanks is outside the crane wall and therefore each is protected against missiles generated from reactor coolant loops.

The level of borated water in each accumulator tank can be adjusted remotely during normal plant operations. Refueling water is added using a high-head safety injection pump. Water level can be reduced by draining to the reactor coolant drain tank. Samples of the solution in the tanks can be taken in the sampling station for periodic checks of boron concentration.

The capacity of the refueling water storage tank is based on the requirements for filling the refueling canal and is approximately 350,000 gallons. This capacity provides borated water to assure:

a) A volume sufficient to refill the reactor vessel above the nozzles, plus

b) The volume of borated refueling water needed to increase the concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive RCC assembly, inserted into the core.

The Safety Injection System also contains the components necessary to assure long-term cooling of the core following delivery of the borated water in the refueling water storage tank. Either of the two recirculation pumps is capable of supplying the necessary long term flow of water for continued core cooling, in addition to containment spray flow. The recirculation pumps take suction from a sump in the containment floor and deliver spilled reactor coolant and borated refueling water back to the core through one of two residual heat exchangers which are also located within the containment.

In the event of a large rupture of the reactor coolant system, the recirculation flow path is entirely within the containment. For the smaller breaks in the reactor coolant system where recirculated water must be injected against higher pressures, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by that route, to the reactor coolant loops. The system is also arranged to allow either of the residual heat removal pumps to take over the function of a recirculation pump should both of those pumps fail. Water is delivered from the containment to the residual heat removal pumps from a separate sump inside the containment.

The recirculation pumps, the residual heat exchangers and piping and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

2.3.2 SAFETY INJECTION SYSTEMS RELIABILITY CRITERIA

To meet the safety objectives of design, the following reliability criteria have been established:

a) Borated cooling water is to be supplied to the core through separated and redundant flow paths; Eight points of injection are provided. Four of these are the combined accumulator and residual heat removal pump injection points, and the other four are the smaller connections from the high-head safety injection pumps.

b) Loss of injection water through a severed reactor coolant loop or safety injection branch line is considered. For the double-ended severance of a reactor coolant loop, loss of all safety injection water delivered to that loop is assumed. For rupture of an injection branch line between the loop and check valve, spilling flow is determined according to Reactor Coolant System pressure.

c) Natural phenomena characteristic of the site are considered. All associated components, piping, structures, power supplies, etc. are designed to Class I seismic criteria.

d) Layout and structural design specifically protects the injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Individual injection lines penetrate the missile barrier, with injection headers in the missile-protected area. Individual injection lines are connected to the injection header, pass through the barrier and then connect to the loops. Maximum practical separation of the individual injection lines is provided. Movement of the injection line associated with a rupture of a reactor coolant loop is accommodated by line flexibility and by the design of the pipe supports such that no damage beyond the missile barrier is possible.

e) Response of the injection systems is automatic, with appropriate allowances for delays in actuation of circuitry and active components. The injection systems are automatically actuated by coincidence of low pressurizer water level and low pressurizer pressure. In addition, manual actuation of the entire injection system and individual components can be accomplished from the control room. Delays in reaching the programmed trip points and in actuation of components are conservatively established on the basis that only emergency on-site power will be available.

f) Redundancy of instrumentation, components and systems is incorporated, to assure that postulated malfunctions will not impair the ability of the systems to meet the design objectives. System effectiveness will exist in the event of loss of normal station auxiliary power coincident with the loss of coolant. The system will also be tolerant to failures of a single component or instrument channel to respond actively in each system.

g) In addition to the manufacturer's performance tests and preoperational test results, provisions for periodic tests are capable of demonstrating the state of readiness and functioning capability of the injection systems. The systems, including their power supplies, are designed to permit complete demonstration of readiness and functioning capability of the injection systems. The systems, including their power supplies, are designed to permit complete demonstration of readiness and functioning capability when the reactor is operating at power or at a hot shutdown. In addition, extensive shop performance testing of characteristics and preoperational functional testing will be carried out.

2.3.3 SYSTEM OPERATION

Injection Phase

Safety injection is actuated automatically following a rupture in the Reactor Coolant System.

The safety injection signal, a coincidence of low pressure and low water level signals from two of the three pressure channels and two of the three level channels in the pressurizer, trips the reactor and starts the safety injection pumps and the residual heat removal pumps and aligns system valving for delivery of the pump output to the reactor coolant system. The items on Figure 2-1 marked with an "S" receive the safety injection signal. Under conditions of low pressurizer pressure and water level, both charging pumps would already be running. A volume control tank low level signal shifts the charging pump suction from the volume control tank to the refueling water storage tank. Suction for the high head safety injection pumps and residual heat removal pumps is always aligned to the refueling water storage tank when the reactor is in operation.

The accumulator tanks are always available for immediate injection of their charge of borated water in the event that Reactor Coolant System pressure falls below 600 psig and are not dependent upon the injection signal. The tanks are charged with borated water at refueling water boron concentration and nitrogen gas, at a pressure of about 660 psig. During normal plant operation each tank is at 660 psig, and is isolated from the Reactor Coolant System by two check valves in series. Should the Reactor Coolant System pressure fall below the accumulator pressure, the check valves will open and the pressure difference between the tanks and the reactor will drive borated water into the reactor.

A remotely operated isolation valve is provided at the accumulator discharge. This valve is normally open, but would be closed when:

1. The reactor is purposely depressurized below accumulator pressure.

2. It is desired to test the seating effectiveness of the injection line valve by depressurizing the pipe between the check valve and the accumulator and measuring water flow into the test line.

3. Leakage through the check valves interferes with accumulator readiness, until an orderly shutdown can be effected for repair of the check valve. This is expected to be an unusual condition, not an intended mode of operation.

In the event of a large area break in the Reactor Coolant System, flow of borated water begins immediately once the pressure falls below 600 psig and continues as pressure continues to fall until the volume in the reactor vessel outside the core barrel below the nozzles, the bottom plenum, and at least one half of the core are filled. The automatic actuation of the residual heat removal pump and the high-head safety injection pumps assure the delivery of borated water from the refueling water storage tank necessary to complete the cooling of the core, to refill the reactor vessel to the nozzles and to augment the containment spray in delivery of water to the containment floor in order to start the long-term cooling using the recirculation loop.

Recirculation Phase

The injection mode of operation is terminated following the delivery of a sufficient amount of water from the refueling water storage tank to assure adequate boration of the spilled reactor coolant water in the containment sump to maintain shutdown. Containment pressure will have been significantly reduced by this time by the containment ventilation air cooling units and the containment spray.

The switchover to the recirculation mode of operation is accomplished manually from the central control room after the appropriate level alarms have sounded indicating delivery of approximately 75% of the water from the refueling water storage tank, and after the operator has determined the status of the electric power supply for the engineered safeguards.

The switchover under conditions of minimum emergency power is accomplished in the following steps. At no time during this procedure is emergency cooling flow to the core terminated.

1) Terminate the automatic injection signal in preparation for shutdown of individual items of pumping equipment.

2) Close switch one:

This drops the following unnecessary loads from the diesels preparatory to adding the recirculation equipment loads:

a) Trips the containment spray pumps.
b) Closes the containment isolation valves at spray pump discharge.
c) Trips one containment ventilation fan, and isolates the thiosulfate tank.

3) Close switch two:

This performs the following functions to provide component cooling water flow to the residual heat exchangers:

a) Starts a third service water pump.
b) Starts one component cooling water pump.

4) Close switch three:

a) Trips the residual heat removal pump
b) Closes the isolation valve at residual heat removal pump suction.

5) Close switch four:

a) Starts one recirculation pump.

At this point, full recirculation of water from the containment to the reactor is established for large area breaks in the Reactor Coolant System. A small break will have been apparent to the operator due to high reactor coolant pressure or no indicated flow from the residual heat removal pumps. With the high-head safety injection pumps still running, the last in the procedure is to either align the suction of these pumps to the recirculation pump discharge or to shut them down.

6) Check Reactor Coolant System Pressure

a) If pressure is greater than 150 psig,
<u>Close switch five</u>

(1) Aligns flow from residual heat exchanger to high-head safety injection pump discharge.

(2) Closes safety injection pump minimum flow recirculation valves.

b) If pressure is less than 150 psig,
<u>Close switch six</u>

(1) Trips the high-head safety injection pumps.

(2) Closes the safety injection pump minimum flow recirculation valves.

7) <u>Close switch seven</u>:

This completes the isolation of the flow path to the refueling water storage tank by closing the isolation valve at the tank discharge.

With the normal sources of electric power available or with all three diesel generators in operation, additional containment cooling equipment will be energized. This includes an additional service water pump and all containment fan cooling units.

Operation of the recirculation loop of the Safety Injection System is continued until cleanup and repair of the plant have made normal residual heat removal loop operation feasible or until the core is removed.

Testing

Initial Tests

The initial tests of individual components in the Safety Injection System and the tests of the systems as a whole complement each other to assure performance of the system and to prove proper operation of the actuation circuitry. The details of the initial checkout and testing of the system and components is essentially as described in the Fifth Supplement to the Preliminary Safety Analysis Report.

The recirculation pump, not mentioned in the Fifth Supplement, will be subjected to a shop test program including the following:

a) Establishment of flow-head characteristics and NPSH requirements over the range of flows possible during the recirculation operation.

b) A test demonstrating the function of the pump under the suction pressures, water temperatures, and maximum flow expected during the recirculation phase of operation.

This pump will not take part in an integrated flow test of the system as a whole.

The accumulator vessels will be subjected to a normal hydrostatic test in the manufacturers shop. In addition, flow will be introduced into the Reactor Coolant loops through the accumulator discharge line to demonstrate operability of the check valves and remotely actuated stop valve. Tests of the nitrogen supply and venting equipment and filling and draining will be performed as part of the routine systems tests prior to plant startup.

Periodic Testing

A program of periodic testing of the Safety Injection System has been described in the Fifth Supplement to the Preliminary Safety Analysis Report.

With the exception of the following the testing of the present system with the accumulators will be the same.

Testing of the Accumulators

Permanent test lines are installed to determine leakage through the check valves in the injection lines and to ascertain that these valves seat when the Reactor Coolant System pressure is raised.

Although they are open during plant operation, the ability of the remote stop valve in each accumulator discharge line to open upon a safety injection signal may be tested by opening the remote test valve just downstream of the stop valve (test circuits, piping and valves are not shown on Figure 2-1). Flow through the test line can be observed on existing instruments and the opening and closing of the discharge line stop valve may be sensed on this instrumentation. Remote position indicators are provided in the control room and will alarm when the valve is off the full-open position.

Testing of the Recirculation Pumps

The recirculation pumps will normally be in a dry sump. These pumps will be started periodically and allowed to reach full speed. No flow testing of these pumps can be performed druing refueling operations.

2.3.4 EQUIPMENT DESIGN PARAMETERS

Accumulator Tanks

Number of tanks	4
Design Pressure - psig	700
Normal Pressure - psig	660
Design temperature - °F	300
Operation temperature - °F	70-120
Total volume - cu. ft.	1100
Normal water volume - cu. ft.	700
Material of Construction	Carbon steel with stainless steel cladding
Code	ASME III - Class C

Residual Heat Removal Pumps

Number of pumps	2
Type	Horizontal centrifugal
Design pressure - casing - psig	600
Design pressure - suction - psig	600
Design temperature - °F	400
Design flow - gpm	3000
Design head - ft.	350
Material	Austenitic Stainless Steel

High-Head Safety Injection Pumps

Number of pumps	3
Type	Horizontal centrifugal
Design pressure - casing - psig	1750
Design pressure - suction - psig	250
Design temperature - °F	300
Design flow - gpm	400
Design head - ft.	2500
Minimum shut-off head - ft.	3500
Material	Austenitic Stainless Steel

Recirculation Pumps

Number of pumps	2
Type	Vertical Type
Design pressure - casing - psig	175
Design temperature - suction - psig	---
Design temperature - °F	300
Design flow - gpm	3000
Design head - ft.	350
Material	Austenitic Stainless Steel

INDIAN POINT ENGINEERED SAFEGUARDS EQUIPMENT STATUS

As of March 1967

1. Containment and Primary Building

The architect-engineer's layout work complete. Lower elevations and ground work are under construction.

2. Accumulator Tanks

The final design has been set, and the purchase order has been placed. This design reflects conformance to the design bases described in Section 2.3.1, and can be accommodated within the availabe space in the containment.

3. High Head SIS Pumps

These pumps are on order. The engineering flow diagram has been issued for final layout by the architect.

4. Recirculation Pumps

Quotes have been received. The purchase order will be following shortly. The sump that the pumps will be located in is in the final design stages.

5. Residual Heat Removal Pumps

The pumps parameters are in the final stages of review. Process piping has been released to the architect.

6. Containment Spray Pumps

The pump parameters are in the final stages of review. Process piping has been released to the architects.

7. Thiosulfate Tank

The tank parameters are under review. Final outline will be released shortly. Process piping has been released to the architects.

2.4 PERFORMANCE CRITERIA

2.4.1 CORE DAMAGE CRITERIA

The basic criterion for assuring that fuel elements will remain in place and substantially intact during a loss of coolant accident is that the calculated Zircaloy clad temperature for the entire core will not exceed the Zircaloy melting temperature, when the core is in its original heat transfer geometry. This includes the theoretical maximum hot spot temperature. Although localized melting would not affect the integrity of the core, zero melting will be calculated for the entire core. Also Zircaloy-water reactions will be limited to an insignificant amount. A Zircaloy-water reaction starts when the cladding temperature reaches 1800°F. A tenacious oxide layer is formed by the Zircaloy-water reaction on the outside of the fuel pellet clad. Zircaloy metal melts as 3375°F; however, the Zirconium oxide will not melt until 4800°F. Although it is expected that the clad will not fail to hold the fuel column intact until the oxide melting temperature is reached, the design criterion is based on a Zircaloy melt temperature of 3375°F.

2.4.2 MINIMUM CORE COOLING REQUIREMENTS

Analysis of loss of coolant accidents involving large breaks indicate that if the core hot spot is re-flooded before a temperature of 3100°F is reached fuel clad melting will be prevented. During the period the core is uncovered following such an accident the heat transfer coefficients are low and the steam temperatures are high. When the bottom of the core is recovered with water, steam flow through the core is increased and the rate of rise of clad temperature is decreased due to the improved heat transfer coefficient. The heat transfer is further improved when the core is flooded, film boiling occurs on the rod, and the environmental temperature is greatly reduced. The calculational models used to simulate this accident are discussed in Section 2.5.1.

The minimum amount of core cooling required to prevent clad melting has been analyzed for a range of break sizes. Figure 2-2 plots the clad temperature versus time with no accumulator or safety injection. Break sizes presented on the curve are:

a) Double ended severance of the Reactor Coolant Pipe,
b) 6 ft^2, and 3 ft^2.

As will be discussed in Section 5, the protection for break sizes for the surge line and smaller is afforded by one residual heat removal and two safety injection pumps, even if the accumulator function is neglected.

In the case of the largest break considered (Fig. 2-2) the clad reaches melting temperature in 52.5 seconds. Thus if the hot spot were covered in 51.5 seconds (T@ 51.5 seconds = 3100°F) for the worst case clad melting would be prevented.

2.5 EVALUATION OF EXPECTED PERFORMANCE

Performance of the safety injection system described in Section 2.3 was evaluated for a full range of break sizes. A negative moderator coefficient was assumed in this part of the analysis. In Section 2.6.5 the sensitivity to A positive moderator will be discussed.

2.5.1 DISCUSSION OF CALCULATIONAL MODEL AND ASSUMPTION

Blowdown Analysis - Flash Code

Analysis of the transient flows, coolant inventory, temperature, and pressure of the Reactor Coolant System following a large-area rupture was performed using the digital computer code FLASH.[1] This code calculates rate of coolant blowdown and rate of influx from the Safety Injection System, pressure drop and flow through the core and intact loops, and accounts for energy entering and leaving the system, by way of the core and steam generators. Reactor power is controlled by moderator reactivity entered as a function of time, and a reactor trip model which represents insertion of RCC's. The moderator reactivity density contribution is pre-calculated using a more detailed core model than is now available in FLASH, with predicted pressure and flow transients which are checked with FLASH results for consistency and conservatism.

The FLASH code treats the Reactor Coolant System as if it were comprised of three control volumes, and calculates the pressure and inventory of each separately. The selection of these volumes in a PWR system is made in

[1] "FLASH: A Program for Digital Simulation of the Loss of Coolant Accident" by S. G. Margolis and J. A. Redfield WAPD-TM-534, May 1966.

such a way as to group those portions of the system whose temperatures and pressure are relatively uniform thoughout the transient:

Volume 1 includes the reactor outlet plenum, the hot leg piping and the steam generator tubes.

Volume 2 includes the loop cold leg piping, the reactor coolant pump, and the reactor downcomer and inlet plenum.

Volume 3 includes the pressurizer and surge line.

Modifications were made to the original FLASH program to account for the specific system configurations of the IPP Unit 2 system: these included the single-pass rod-type core, and the location of the reactor coolant pump, the accumulator and injection pumps characteristics. A sub-routine was added to the FLASH program to determine the flow rate into the Reactor Coolant System for the accumulator. The flow rate calculation is based on the pressure difference between the accumulator lines. The accumulator tank gas pressure is assumed to expand isentropically to replace the injected accumulator water. The accumulator pressure, and liquid and gas inventories are continually calculated. Accumulator injection continues until the tanks are emptied.

The results of FLASH, core cooling inventory, pressure, quality, flow rates through the core, etc. are used for a detailed analysis of the core thermal transient.

CORE POWER TRANSIENT DURING BLOWDOWN

The basic tool used for the reactor kinetics calculation is the CHIC-KIN[4] code, which has a point kinetics model and a single channel fuel and coolant description. In this study the channel was divided axially into five sections, with density in each section a function of pressure and enthalpy, plus

[4] Redfield, V. A., "CHIC-KIN--A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor" WAPD-TM-479, January 1, 1965

nucleate boiling void. A nucleate boiling model for highly subcooled conditions was used, even though a large part of the coolant is saturated throughout the transient. This was done to minimize apparent void formation in order to retard reactor shutdown and yield a conservatively high energy input. Average core pressure was input as a function of time from the FLASH output. For small breaks with forward flow, the core inlet flow as shown by the FLASH code calculations was used as input to CHIC-KIN. For large cold leg breaks, with violent flow reversal and then near-stagnation, the core pressure drop as indicated by FLASH was assumed to be a reasonable representation of the forcing action between the two large liquid regions of the system. This pressure drop was used as input to CHIC-KIN, which calculated flow response taking into account inertia and losses at inlet and outlet and from grids and friction in the fuel. The resulting flow transients were very close to those obtained by FLASH.

Each axial fuel rod section was divided into nine radial regions for the heat transfer calculation. Fuel properties are constant in the code, and best estimate fuel thermal properties were used to determine the core average fuel heat transfer response, and the resulting void formation. Since hot channels have greater than average void fraction, even if DNB occurs in them, neglecting the effect of the hot channels reduces the total apparent void and thus yields a conservatively high energy input.

Trip would be actuated by overpower for the cases with a significantly positive moderator coefficient or by low pressurizer pressure with a zero or negative coefficients. For the small break cases, depressurization after the initial subcooled blowdown is slow, and thus shutdown or voids is also slow. Trip is required in these cases, and in CHIC-KIN this was simulated by a ramp insertion of negative reactivity starting. For the large breaks trip would be similarly actuated, but because void formation is adequate for shutdown, trip was not simulated in these studies.

Doppler reactivity feedback was simulated as a function of the average fuel temperature, with a weighting factor of 1.4 used as a lower limit for the initially unrodded core.

Six groups of delayed neutrons were used. The total effective fraction was 0.0061, a conservative minimum, for the positive moderator coefficient cases. A conservative maximum of 0.0072 was used for the zero or negative coefficient cases to slow down power decay.

Core Cooling Analysis

The LOCTA-R2 transient digital computer program was developed for evaluating fuel pellet and cladding temperatures during a loss of coolant accident. It determines the extent of the Zircaloy-steam reaction and the magnitude of the resulting energy release in Zircaloy clad cores.

The transient heat condition equation is solved by means of finite differences, considering only heat flow in the radial direction. A lumped parameter method is used- the fuel containing three nodes and the cladding one node.

Internal heat generation can be specified as a function of time, or the decay heat from any initial power level can be calculated by the code. The decay heat is based on the heat generated from:

a) Fission Products,
b) Capture Products, and
c) Delayed Neutrons

It is assumed that the core has been irradiated for an infinite period of time.

In addition to decay heat, the code calculates the heat generated due to the Zircaloy-steam reaction. The Zr-H_2O reaction is governed by the parabolic rate law unless there is an insufficient supply of steam available, then a "steam limited" evaluation is made. The buildup of the Zircaloy-oxide film is calculated as a function of time, and its effect on heat transfer is considered. An isothermal clad melt is considered based on the heat of fusion of Zircaloy. Once the Zircaloy metal melts, it is retained by the Zirc-oxide, and slups against the fuel. The Zircaloy-steam reaction may continue until the oxide melts. If the oxide melts the remaining Zircaloy is assumed to fall, and 10% of this metal is assumed to react with additional water which is available in the vessel.

The code has been developed to stack axial sections and thereby describe the behavior of a full length region as a function of time. A mass and energy balance is used in evaluating the temperature rise in the steam as it flows through the core.

The initial conditions of the fuel rod are specified as a function of power. The following core conditions are also introduced as a function of time; as determined by the Flash Code:

a) Mass flow rate through the core
b) Coolant quality
c) Pressure
d) Liquid level

Heat transfer coefficients during the various phases of the accident are evaluated in the following manner:

a) Nucleate boiling film coefficients on the order of 20,000 Btu/hr ft^2 °F are used until DNB.

b) When DNB occurs, it is assumed that the fuel rods can immediately develop a condition of stable film boiling. No credit is taken for higher transition boiling coefficients that exist prior to the establishing of a stable film on the fuel rods. The correlation used during this period is

$$h = .023 \frac{k_v}{D_e} \left(\frac{\rho_v D_e}{\mu_v} \frac{Q_1 + Qv}{A_c} \right)^{0.8} \left(\frac{c_p \mu}{k} \right)^{0.4}$$

c) During the time the core is uncovered (period of steam flow through the core), laminar or turbulent forced convective coefficients and radiative coefficients and radiative coefficients are evaluated.

For laminar forced convection to steam:

$$\left(\frac{hD}{k}\right)_{iso} = 3.66$$

$$h/h_{iso} = \left(\frac{Tb}{Tw}\right)^{.25}$$

For turbulent forced convection to steam:

$$\frac{hD}{k} = 0.020 \, (Re_b)^{0.8} \, (Pr_b)^{0.4} \left(\frac{Tw}{T_b}\right)^{-0.5}$$

d) Conservative heat transfer coefficients of the order of 25 Btu/hr ft^2 °F is all that is needed to turn back the rising clad temperature when the core is recovered.

Information generated by LOCTA R 2 as a function of time includes:

a) Fuel temperature,
b) Clad temperature,
c) Steam temperature,
d) Amount of metal water reaction
e) Volume of core melt, and
f) Total heat released to coolant

Symbols for Equations

h - Heat transfer coefficient on outer surface of fuel rod (Btu/hr-ft^2-°F)

De - Equivalent diameter of flow channel - (ft)

ρ - Density (lbs/ft^3)

μ - Viscosity (lbs/ft-hr)

Q - Volumetric flow rate (ft^3/hr)

Ac - Area of flow channel (ft^2)

Cp - Specific heat (Btu/lb-°F)

k - Thermal conductivity (Btu/hr-ft-°F)

T - Temperature °F

Subscripts

v - Evaluation of the property at the saturated vapor condition

l - Evaluation of the property at the saturated liquid condition

b - Evaluation of the property at the saturated bulk fluid condition

w - Evaluation of the preperty at the saturated bulk clad condition

2.5.2 BLOWDOWN AND RECOVERY TRANSIENT - LARGE BREAKS

The operation of the Safety Injection System with accumulators was analyzed for the following range of break sizes:

a) Double ended severance of the Reactor Coolant Pipe,
b) 6 ft^2,
c) 3 ft^2, and
d) .5 ft^2

All break sizes studied were cold leg breaks. During the blowdown portion of the transient no credit was taken for the safety injection pumping system. It was also assumed that all the contents of one accumulator tank spilled through the break for all cases.

The Reactor Coolant Blowdown transient was evaluated using the FLASH computer program (described in Section 2.5.1). Figures 2-3 through 2-6 present the results of the pressure and liquid volume transients for the above break sizes. The significant points of the transient, core uncovering, accumulator flow starting, core recovery, accumulator completion can be noted from the curves. The volumes presented in the curves are "quiet" water levels, i.e., no credit is taken for an increased froth height due to voids created by boiling in the core. The quiet levels were used in the calculation of heat transfer coefficients.

For all break sizes considered the quiet level was restored above the level of the hot spot before the accumulator tanks are emptied of water. The total flow delivered by the accumulators exceeds the requirement of restoring the level to the mid-plane of the core. In all cases due account is taken for the water in the reactor vessel downcomer and reactor vessel inlet piping. This is of particular importance in cold leg breaks, where a static head of water in the downcomer is required to drive the steam generated in the core around the loop to the assumed cold leg break location. This will be discussed further in Section 2.6.2.

2.5.3 CORE POWER TRANSIENT DURING BLOWDOWN

Nuclear transient response to loss of coolant can be separated into two classes according to the break assumed. The first is a "small" break, in which pressure is maintained or increased for a few seconds after subcooled blowdown, and forward flow is maintained through the core until trip is effective. This type is characterized by the 1/2 ft^2 cold leg rupture. The second class is that of "large" breaks, in which pressure continues to drop rapidly after subcooled blowdown, until the cold leg is saturated. This type is represented by the 3 ft^2 and double-ended cold leg ruptures, both of which result in reversed core flow.

For the 1/2 ft^2 cold leg break, the initial subcooled decompression, about 1/4 second in duration, does not form enough void to shutdown the core unless the moderator coefficient is strongly negative. For a zero coefficient, power drops 5% to 15% before the trip becomes effective. For a positive moderator coefficient, the maximum moderator reactivity is achieved within the first 1/2 second of the transient, and subsequent void formation tends to decrease reactivity.

Thus the transient is very similar to that for a near-step addition of reactivity with a negative moderator coefficient, with the reduction of coolant flow and heat capacity acting to increase the negative moderator effect. Calculations for various total moderator reactivity insertions have shown no sharp discontinuity in behavior as the magnitude of this insertion exceeds the delayed neutron fraction. This is partly because of the time taken to insert the moderator reactivity, and the response of fuel temperature which helps to prevent actually reaching a prompt critical condition. Also, as the initial power burst becomes greater, the subsequent shutdown is also quicker, so that energy generated between the power burst and trip becomes less significant. This smooth variation of energy with inserted reactivity through the prompt critical insertion was also observed in studies of the control rod ejection accident from full power. This small break accident represents essentially the maximum energy which could be generated in the loss of coolant accident. A smaller break would take longer to reach the maximum moderator reactivity after actuating trip, thus it would have less time with the moderator reactivity inserted before trip becomes effective. A larger break reaches the maximum reactivity sooner, but tends to shut down on voids before the rods come in. However, core cooling studies indicate that the energy input for a small break is more than offset by the longer cooling time before the core is uncovered.

For the large breaks the faster subcooled blowdown and subsequent rapid continued depressurization introduce voids much more rapidly and extensively than in the case of a small break. Backflow through the core also forces a saturated steam-water mixture from the reactor outlet plenum down into the core, adding to the voiding. The result is that for the 3 ft^2 and double-ended cold leg breaks studied, the reactor either shuts down immediately, or in the case of a positive moderator coefficient undergoes a power burst followed by shutdown within 1/4 second. If the available reactivity is less than prompt critical the energy is strongly dependent on the time taken to pass through the adverse density condition.

For higher available reactivity, the power transient becomes faster, relative to the void transient, and thus the energy becomes less dependent upon the pressure and flow transient. With an insertion of over two times the delayed neutron factor, the total energy generated approaches the total generated in the prompt burst following a step reactivity insertion. That is, it becomes insensitive to the nature of the blowdown.

2.5.4 CORE THERMAL TRANSIENT

The core thermal transient was determined using the blowdown and recovery data, and the core power for negative moderator which were described in the previous sections. Figure 2.8 presents the results of this study. The maximum clad temperature reached for the range of break sizes was 1800°F and this occurred for the double loaded severance of the reactor coolant pipe. It was conservatively assumed that DNB occured for all cases. The peak temperature for the double ended break occurred while the core was uncovered. A peak temperature of 1745°F was calculated for the 0.5 ft^2 at 11 secs. This clad temperature is higher than that occuring in the 3 ft^2 break due to a generation of 3.3 full power seconds after the break prior to shutdown. This results from the pressure hang-up for this small break. Other small breaks which require rod insertion for shutdown will behave in a manner similar to the 0.5 ft^2 break.

2.5.5 DISCUSSION OF SMALL BREAK CASES

The preceeding paragraphs have demonstrated adequacy of accumulator injection concept to terminate core exposure and limit the temperature rise of fuel cladding in conformance to safety objectives for large area breaks. The FLASH computer code was employed in this evaluation in order to provide detail as to the coolant flow phenomena affecting reactor shutdown and core heat transfer during the early post-rupture, blowdown and fill processes.

In smaller area breaks, previous calculations using the single coolant region code LOCO have provided the basis for design of the high and low head pumping systems whose capacity assures that the safety objectives are met in the case of these ruptures as well. LOCO calculates the blowdown history by small incremental steps in system pressure, determining the make-up and blowdown mass flows from mass and energy balance and nozzle flow equations.

Results for a postulated 4" break using the LOCO analysis are presented in Figure 2-7. It was assumed that the engineered safeguards were operated by diesel power, and one safety injection pump and one residual heat removal pump were operating. For this case no credit is taken for the accumulators. For this case the core hot spot is briefly uncovered. The safety injection system then quickly recovers the core. A core thermal analysis (LOCTA-R) on this case resulted in no metal water reaction, and no fuel clad melting. In the actual case accumulator flow would start when system pressure reached 600 psig.

2.6.1 EXPECTED VS. REQUIRED PERFORMANCE COMPARED

In the preceeding sections the minimum core protection requirements and the actual performance of the accumulators were discussed. Figures 2-9 and 2-10 present a comparison of the required and actual core protection.

Figure 2-9 presents the maximum clad temperatures with accumulators operating versus break size. The maximum clad temperature reached for any break size would be 1800°F. Since this is 1575°F below the clad melting point there will be no clad melting.

Figure 2-10 compares the minimum core re-flooding time to prevent clad melting with the actual re-flooded times. As stated in Section 2.4.2, the fuel hot spot must be reflooding within 52.5 seconds to prevent clad melting. The accumulators re-flood the core in 35 seconds. This represents a margin of 17.5 seconds. For smaller breaks an equal or greater margin exists.

2.6.2 ACCUMULATOR SYSTEM PARAMETERS

A study was performed to determine the effect on system performance by varying accumulator parameters. Figure 2-11 presents the results of this study. The fixed parameters in this study were the accumulator line size, line length, and accumulator tank water volume. The line size and line length were determined from the actual plant layout. The 700 ft^3 of water in each tank together with 40 ft^3 in each accumulator line are more than sufficient to meet the requirement of filling the downcomer, loop inlet piping, and reactor vessel to level of the hot spot, with a conservative allowance for boil off. (Refer to Figure 2-3 through Figure 2-7). Once the hot spot is recovered, the flow from the minimum available pumps is sufficient to continue the refill and decrease the hot spot clad temperature below the maximum. Thus a larger volume of accumulator water would have little effect on the maximum clad temperature.

The parameter study was performed for the case of the double ended reactor coolant pipe break. Four different total accumulator volumes were studied (with the fixed water volume of 700 ft^3 this in effect varies the gas volume). For each volume, the relation between gas pressure and the time required to recover the core hot spot was determined and plotted in Figure 2-11. For a reference point, the performance of the accumulator system being procured (600 psi, 1100 ft^3, etc) is circled on the figure. The 35 second hot spot recovery time for this case was derived from the blowdown and recovery transient presented in Figure 2-3.

The effect of varying pressure can be seen by referring to the 1100 ft^3 accumulator curve. For a 600 psi accumulator the hot spot is recovered at 35 secs., which is 17 secs. faster than required. If the accumulator pressure were increased to 1000 psi the recovery time would be decreased to 25 seconds. However, as will be discussed in Section 2.6.3 there are some disadvantages to a high accumulator pressure. Decreasing accumulator pressure has the effect of increasing the recovery time. The study was conducted only as low as 550 psig, which is below the lower limit for which the margin is considered to be adequate. If performance at a lower pressure is of interest, the curves can be extrapolated by extending the curves to an infinite recovery time as the accumulator pressure asymptotically approaches the minimum reactor vessel back pressure, i.e. with zero pressure differential between the reactor coolant system and the accumulator tanks there can be no accumulator injection.

Figure 2-11 also illustrates the effect of increased gas volume. The accumulator injection driving force is an isentropic expansion of the gas volume. As the total volume is increased from 900 to 1000 ft^3, and the corresponding initial gas volume is increased by 50% (from 200 to 300 ft^3), a large increase in accumulator performance results. The next 100 ft^3 increase causes a 33% increase in gas volume and a smaller improvement in accumulator performance. Each incremental increase in gas volume then results a smaller incremental increase in performance. Thus for a 600 psi initial gas pressure, an increase from 1100 to 1200 ft^3 total accumulator volume, a 2.5 second improvement in hot-spot recovery time results.

It can be seen that although increases in the accumulator parameters would result in improvements in the margins demonstrated, these improvements are minimal. The accumulators have been sized according to the parameters stated in section 2.3.4. This design more than adequately meets the core cooling criteria for a core operating with zero or negative moderator coefficient.

2.6.3 INJECTION FLOW TO THE CORE

Several factors have been considered that could adversely affect the accumulator and safety injection flow to the core. These will be discussed in the following paragraphs.

One factor considered was the possibility that the injection flow would be diverted to some other part of the reactor coolant system rather than the core. Figure 2-12 through 2-15 presents the delivery characteristics of 3 out of 4 accumulators for the range of break sizes analyzed in Section 2.5.2. The combined accumulator flow rate which enters the loop piping and flows through the downcomer to the core is shown for each case. The maximum accumulator flow rate for any break is 6600 lbs/sec, which occurs for the double ended break. This flow rate is approximately 17.5% of the steady state flow rate of 37,800 lbs/sec for normal plant operation, and thereby there is no possibility of choking the downcomer and backing the flow to other parts of the system. Flow into the inlet of the vessel is also enhanced by the reactor coolant pump, which would be coasting down during the transient and would tend to force coolant in the direction of the reactor. Further, a characteristic of the reactor coolant pumps prevent back-flow through the pumps under the injection condition. Each pump has a diffuser which effectively serves as a weir to impede back-flow through the pump. The discharge pipe has to be full of water before the weir can be over-topped. The water required to fill the discharge piping between the reactor coolant pipe and the vessel is accounted for in the calculations.

Another concern is the possibility that the injected accumulator water could be carried out with the blowdown. This can be illustrated by the following examples. If the accumulator injection were somehow held until the blowdown were complete, all of the accumulator water would enter the system since there would be no driving pressure differential to carry water out the break. On the other hand, if the accumulator system was pressurized to the initial Reactor Coolant Pressure and the contents injected instantaneously, conceivably a large fraction of the contents could be carried out with the blowdown, because these accumulators would act essentially as an extension to the reactor coolant system.

This situation was assessed by comparing FLASH blowdown runs for 1000 and 600 psi accumulators. The other parameters were in accordance with specifications for Indian Point Unit No. 2 (Section 2.3.4). As shown in Figure 2-3 the 600 psi accumulators re-flood the core well past the hot spot. The run for the 1000 psi accumulator resulted in a re-flooded volume 275 ft^3 less than that for the 600 psi accumulator, or just barely reflooding the hot spot. This difference in volume delivered can be seen by comparing the ratio of time the accumulators are injecting during blowdown to the total blowdown time. For the 1000 psi accumulators this ratio is 73% of the time, while for the 600 psi accumulators the ratio is 43%. Although it was shown in Section 2.6.2 that the 1000 psi accumulator increases the margin in recovery time by 10 secs over the 600 psi accumulator, there was a significant sacrifice in terms of water delivered for the 1000 psi accumulator.

Another factor considered was the possibility of a short circuit flow path of the injected accumulator water to the break. The worst break location for this is a cold leg break. It is assumed in the analysis that all of the flow from one accumulator is lost through the break. The remaining accumulators inject into the other loop cold legs. At the time the accumulators begin to inject for large area ruptures, the water level has dropped below the bottom of core. (Refer to Figure 2-3 through Figure 2-6). The injected water would enter the vessel and fall into the water into the lower vessel plenum. To reach the break the flow from two of the accumulators would have to travel 180° around the downcomer annulus and circumvent two outlet nozzles which bridge across the annulus. The third accumulator injection point is in a pipe adjacent to the postulated broken loop, but the flow would have to overcome gravity to reach the nozzle of the broken loop. It is concluded that since the accumulator pressure is low enough to delay injection until after the nozzles are well uncovered, the short circuit problem does not obtain.

Finally the effect of steam generation in the core was considered. When the core is reflooded and steam is generated a steam bubble is formed. The steam pressure is relieved by the flow of steam through the break. The worst location is a cold leg break where the steam must travel through the steam generator, pump, and all of the loop piping to reach the break. A static head of water in the downcomer above the core water level is sufficient to drive the generated steam to the cold leg break location. In the recovery calculations this required downcomer head is taken into account. It should be noted that any steam generation in excess of that which can be relieved by the downcomer head provides of itself sufficient flow to cool the core.

2.6.4 SENSITIVITY TO POSITIVE MODERATOR COEFFICIENT AND FILM HEAT TRANSFER COEFFICIENT.

Sensitivity studies were performed on a typical PWR core with Zircaloy clad fuel rods to illustrate the effects of varying reactivity insertion and film heat transfer coefficients on clad temperature.

For the purpose of showing the effects of reactivity insertion and film heat transfer coefficients during the steam cooling phase of the accident, the double-ended break was analyzed. For the variation of film heat transfer coefficients during blowdown the 3 ft^2 break was used because for a 1.10% reactivity insertion this particular break reaches its peak clad to temperature during this time.

Effect of Reactivity Insertion

To illustrate the effects of reactivity insertion on clad temperature, a double-end cold leg break was analyzed for 0.63, 0.86 and 1.10% reactivity insertion. Core parameters such as mass flow rate, pressure core inlet temperature and heat transfer coefficients as functions of time were the same for all cases. DNB was assumed to have occurred in less than 0.5 seconds after the break.

Results of the study indicated that the peak clad temperature for the entire transient occurred during the blowdown phase of the accident and at the same time for all variations of reactivity insertion. A curve of peak clad temperature versus reactivity insertion is shown in Figure 2-16 with 1.10% reactivity insertion producing a maximum peak clad temperature of 2534°F.

Effect Of Film Heat Transfer Coefficient During Blowdown

A 3 ft^2 cold leg break with 1.10% reactivity insertion and film heat transfer coefficients of 500, 300 and 100 Btu/hr ft/°F during blowdown was analyzed. All peak clad temperatures occurred during the blowdown phase of the accident and at approximately the same time.

A curve of peak clad temperature versus film heat transfer coefficient during blowdown is presented in Figure 17 with a film heat transfer coefficient of of 100 Btu/hr ft^2 °F producing a maximum peak clad temperature of 2650°F.

Effect Of Film Heat Transfer Coefficient During Steam Cooling Period

A double-end cold leg break with 1.10% reactivity insertion and film heat transfer coefficients of 75, 100, and 125 percent of actual design film heat transfer coefficients during the steam cooling phase of the accident was analyzed.

Results of the analyzis are presented in Figure 18 with peak clad temperature versus percent of actual design film heat transfer coefficient during the steam cooling period. A film heat transfer coefficient of 75 percent of the calculated value gave a maximum peak clad temperature of 2620°F.

2.6.5 THERMAL - MECHANICAL STABILITY OF FUEL

Chapter 3 describes the evidence obtained by analysis of the sub-cooled expansion phase of the loss of coolant accident which indicates that the most severe hydromechanical forces will not cause deformations or failures capable of interferring with shutdown and post-accident core cooling.

Thermal effects on the core geometry are of interest during the period when high quality and/or core uncovering leads to clad temperature rise to a range where significant changes in the mechanical properties of Zircaloy occur. This phase may be reached in 3 to 30 seconds, depending on the break size, and is terminated when reflooding is accomplished. As in the case of hydromechanical forces, permanent deformation should be limited to the extent that the core remains permeable to the coolant when injection begins, and that the UO_2 remains essentially in place.

Studies of the thermomechanical effects of this transient are limited by the knowledge of the plastic behavior of cladding under the prevailing conditions. The loss of coolant analysis provides a description of the peak cladding temperature, and the spatial dependence of clad temperature, up to a point where the core geometry is no longer definable. When the coolant flashes to a high quality conservatively represented in the model as a step transition to film boiling, clad temperature rises and local yielding occurs due to the unbalanced internal pressure of fission gases in high burnup rods. This results in a diametral growth of the rod and partial restriction of the flow channel cross section. Two factors limit the progress of this growth:

a) Reduction in internal gas pressure due to the expanded void space between pellet and clad, and
b) Perforation of the clad at a point of excessive stress or high oxidation rate.

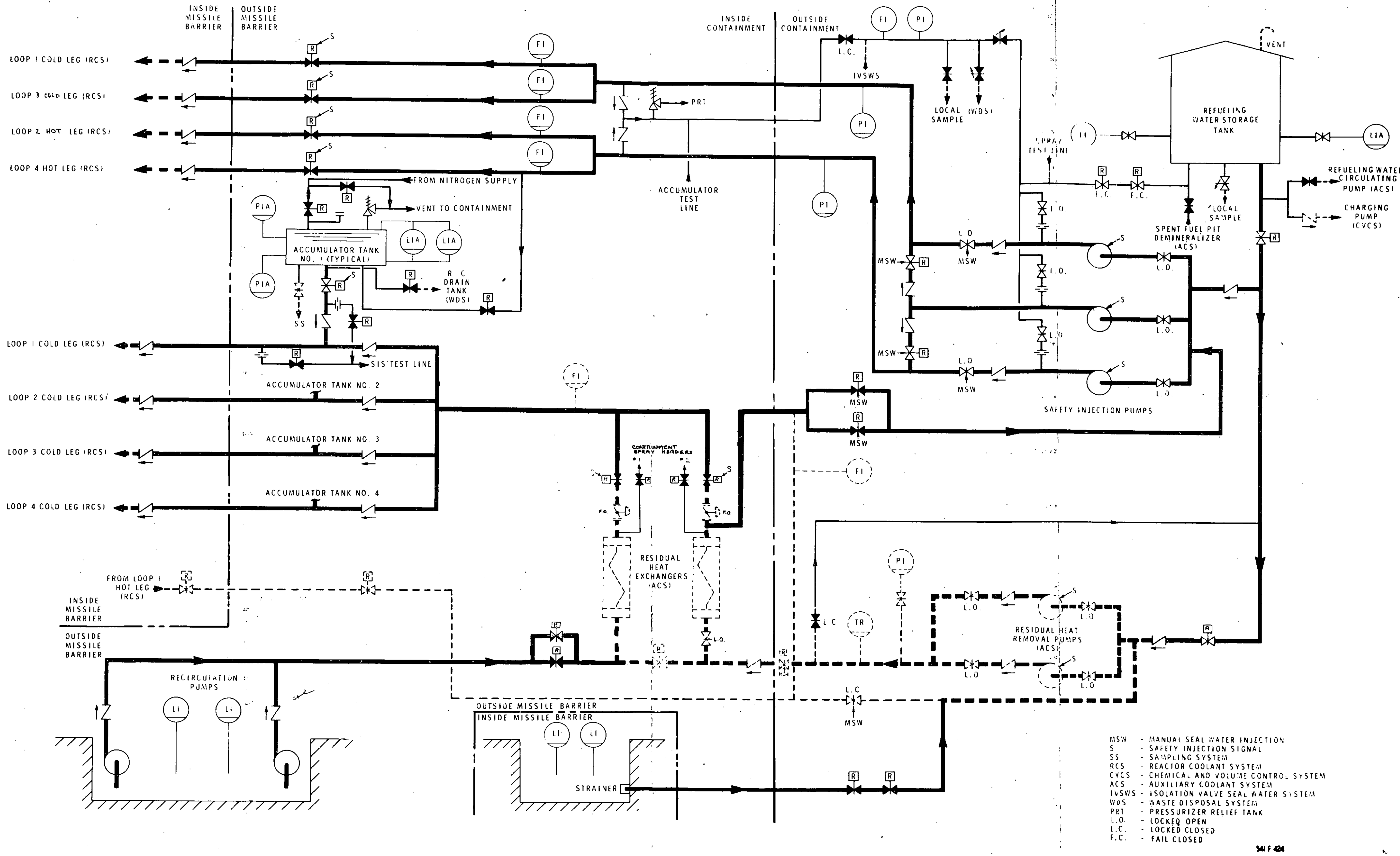
Qualitative understanding of these limiting phenomena suggests that rod growth will not take place to such an extent that gross blockage of coolant flow could occur. In the first place, diametral growth of the rod could take place resulting in a five-fold reduction in gas pressure without seriously impeding coolant flow. Unless plastic strain occurs uniformly (a behavior which is improbable considering the characteristics of the material and possible uneven heating effects due to pellet cracking) the clad would exceed ultimate strain locally, relieving the pressure stress before this amount of growth is realized.

Because of the extrapolation of material behavior which is necessary to refine this model of fuel rod failure, it is planned to conduct experiments with internally heated and pressurized rods. These tests will seek to define the thermal failure mode of the cladding and lead to a basis for prediction of the effects of heatup rate, external pressure, coolant flow, burnup, etc. It is expected that these effects will fall within the tolerances allowed in the present accident model.

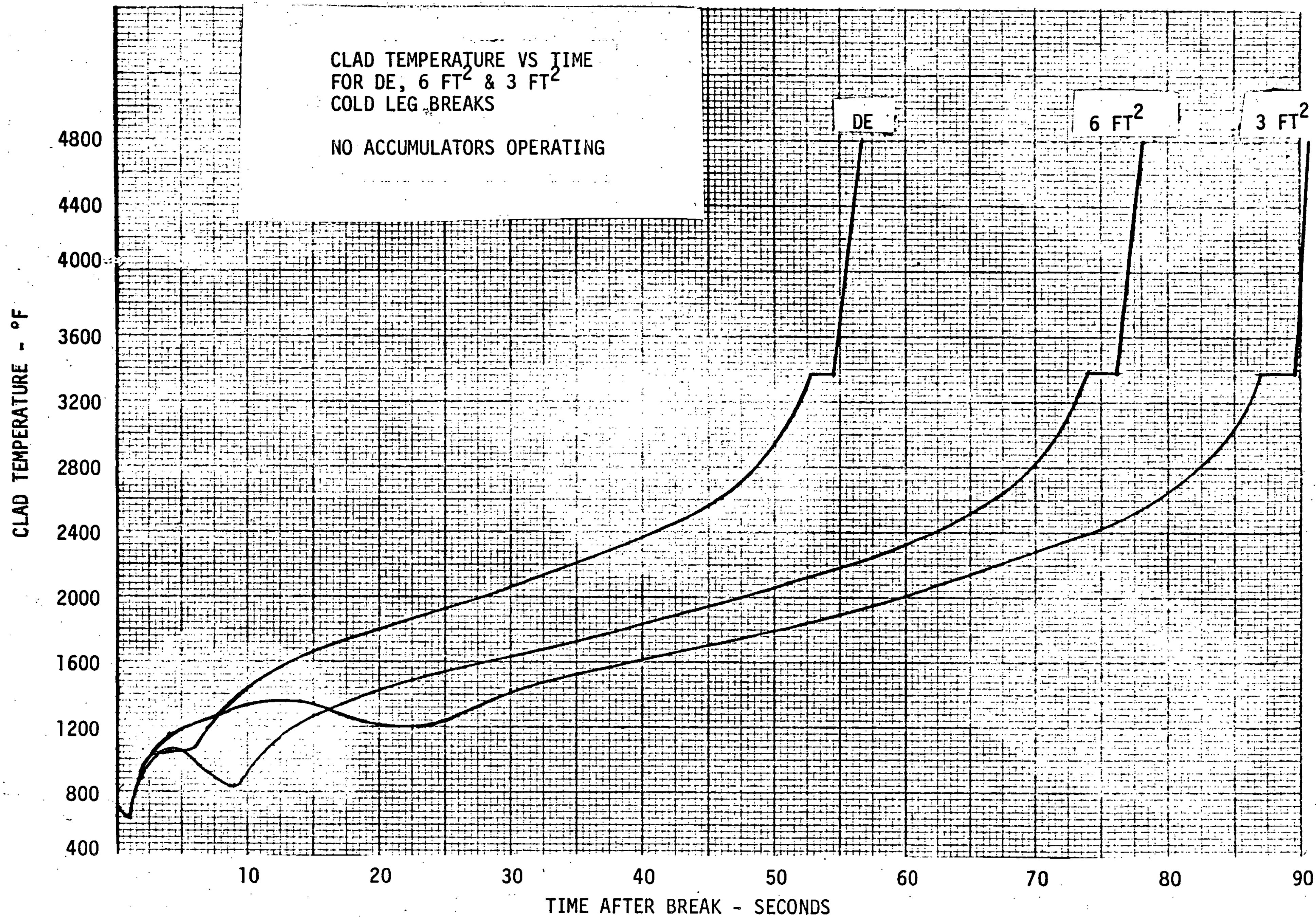
2.7 CONCLUSIONS

The analysis carried out and presented in this report demonstrates that the safety injection system meets the core cooling design objective with suitable margin for all break sizes up to and including the double ended severance of the reactor coolant pipe.

The safety injection and accumulator parameters have been set. When the detailed core design is complete and the magnitude of the positive moderator coefficient determined, additional core cooling analysis will be made. If the core cooling objectives cannot be met with a positive moderator coefficient, fixed shims will be added to limit or eliminate the positive coefficient.

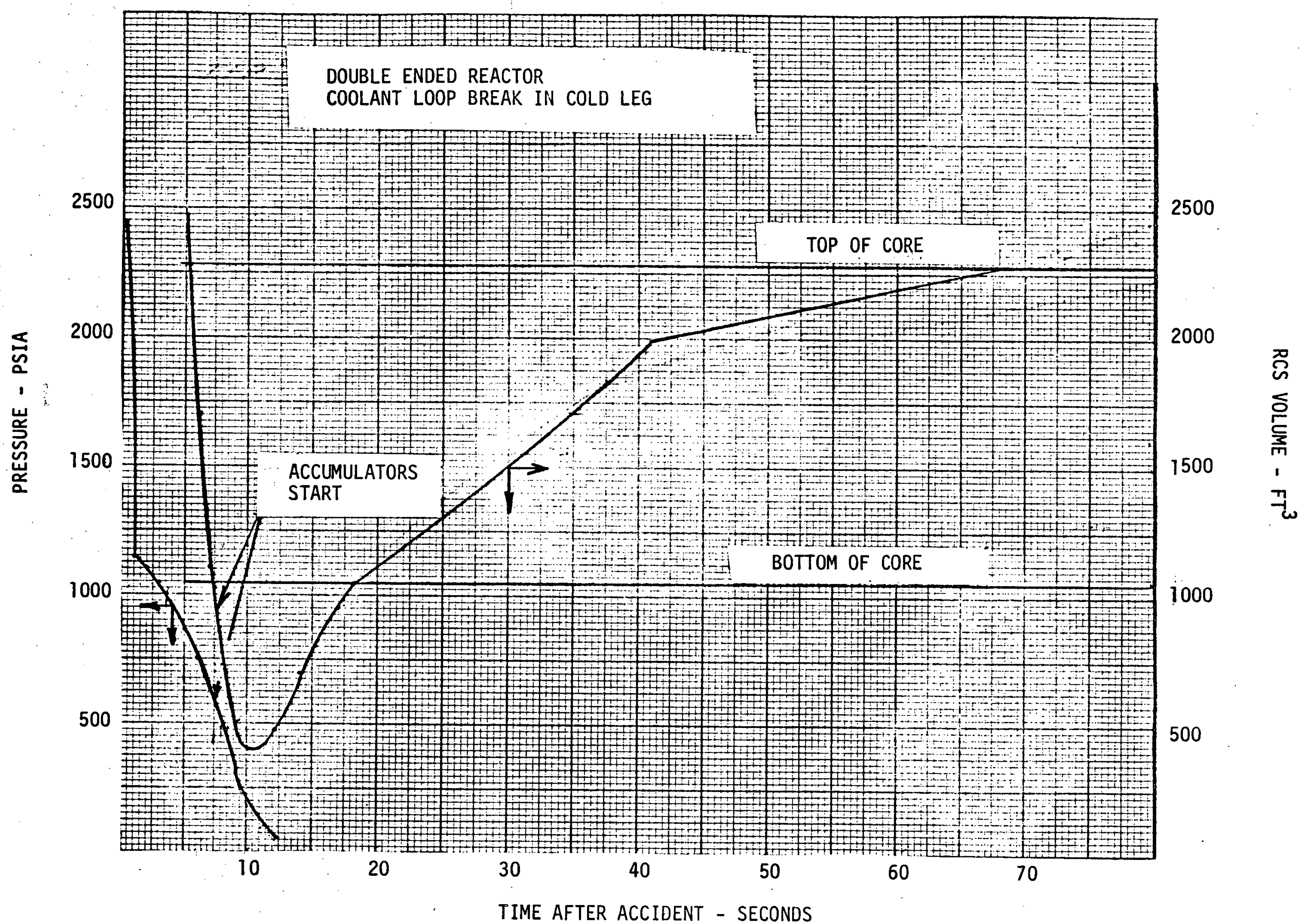


SAFETY INJECTION SYSTEM
FIG. 2-1


CLAD TEMPERATURE VS TIME
FOR DE, 6 FT^2 & 3 FT^2
COLD LEG BREAKS

NO ACCUMULATORS OPERATING
DE
6 FT^2
3 FT^2
CLAD TEMPERATURE - °F
4800
4400
4000
3600
3200
2800
2400
2000
1600
1200
800
400
10
20
30
40
50
60
70
80
90
TIME AFTER BREAK - SECONDS

FIG. 2-2



DOUBLE ENDED REACTOR
COOLANT LOOP BREAK IN COLD LEG
TOP OF CORE
BOTTOM OF CORE
ACCUMULATORS
START
PRESSURE - PSIA
RCS VOLUME - FT^3
2500
2000
1500
1000
500
10
20
30
40
50
60
70
TIME AFTER ACCIDENT - SECONDS

FIG. 2-3

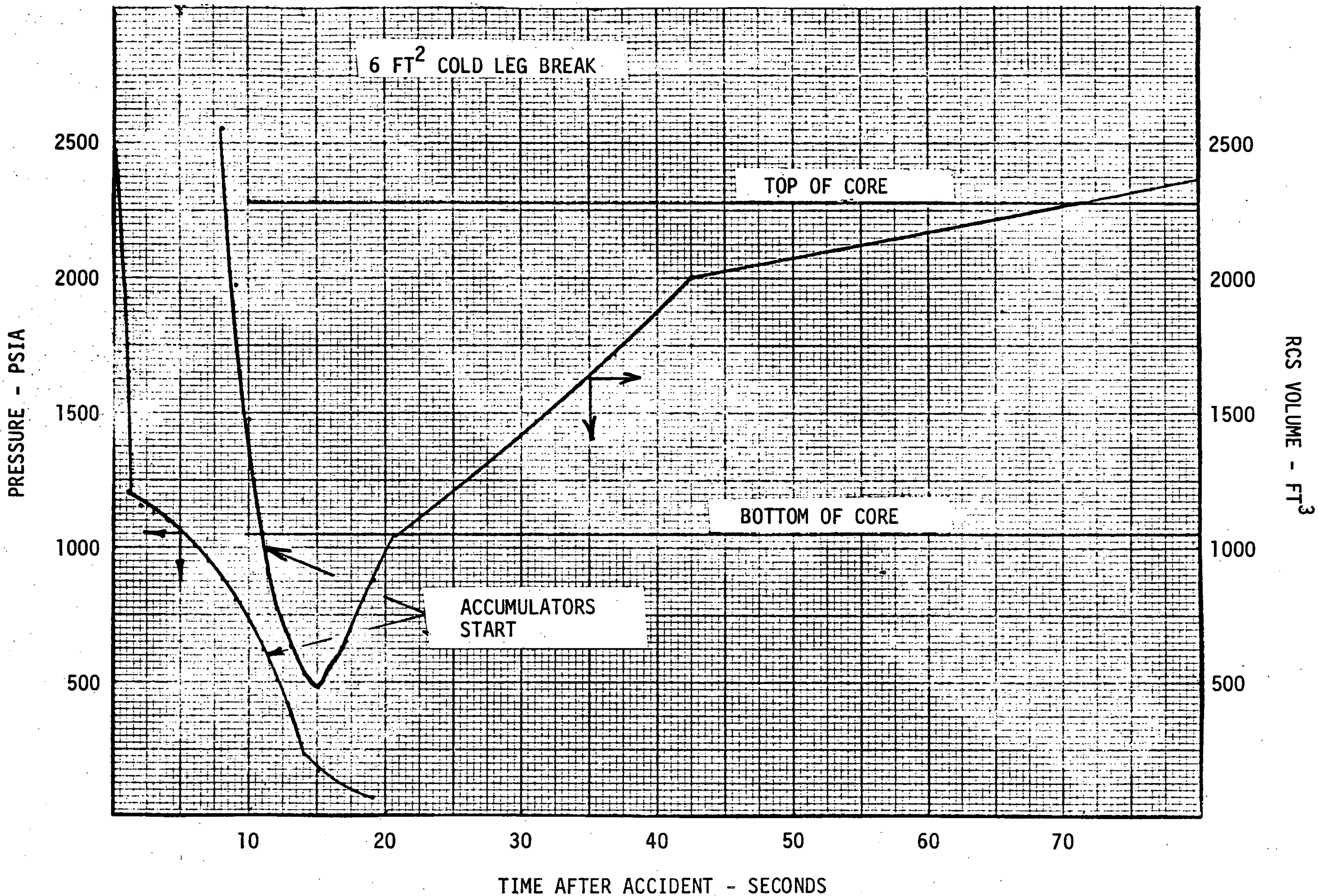
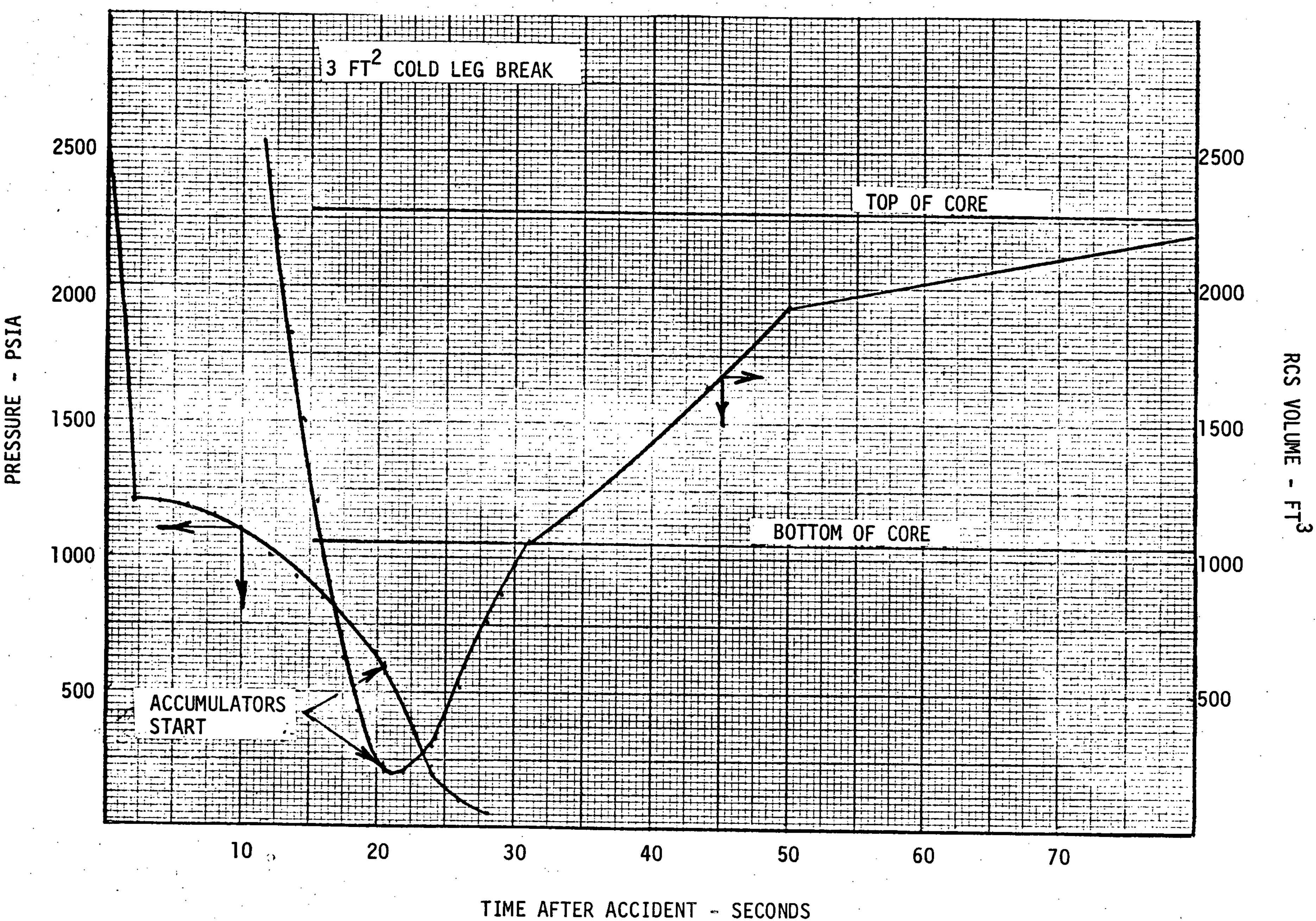

6 FT^2 COLD LEG BREAK
TOP OF CORE
BOTTOM OF CORE
ACCUMULATORS START
PRESSURE - PSIA
RCS VOLUME - FT^3
2500
2000
1500
1000
500
10
20
30
40
50
60
70
TIME AFTER ACCIDENT - SECONDS

FIG. 2-4



3 FT2 COLD LEG BREAK
TOP OF CORE
BOTTOM OF CORE
ACCUMULATORS START
PRESSURE - PSIA
2500
2000
1500
1000
500
RCS VOLUME - FT3
10
20
30
40
50
60
70
TIME AFTER ACCIDENT - SECONDS

FIG. 2-5

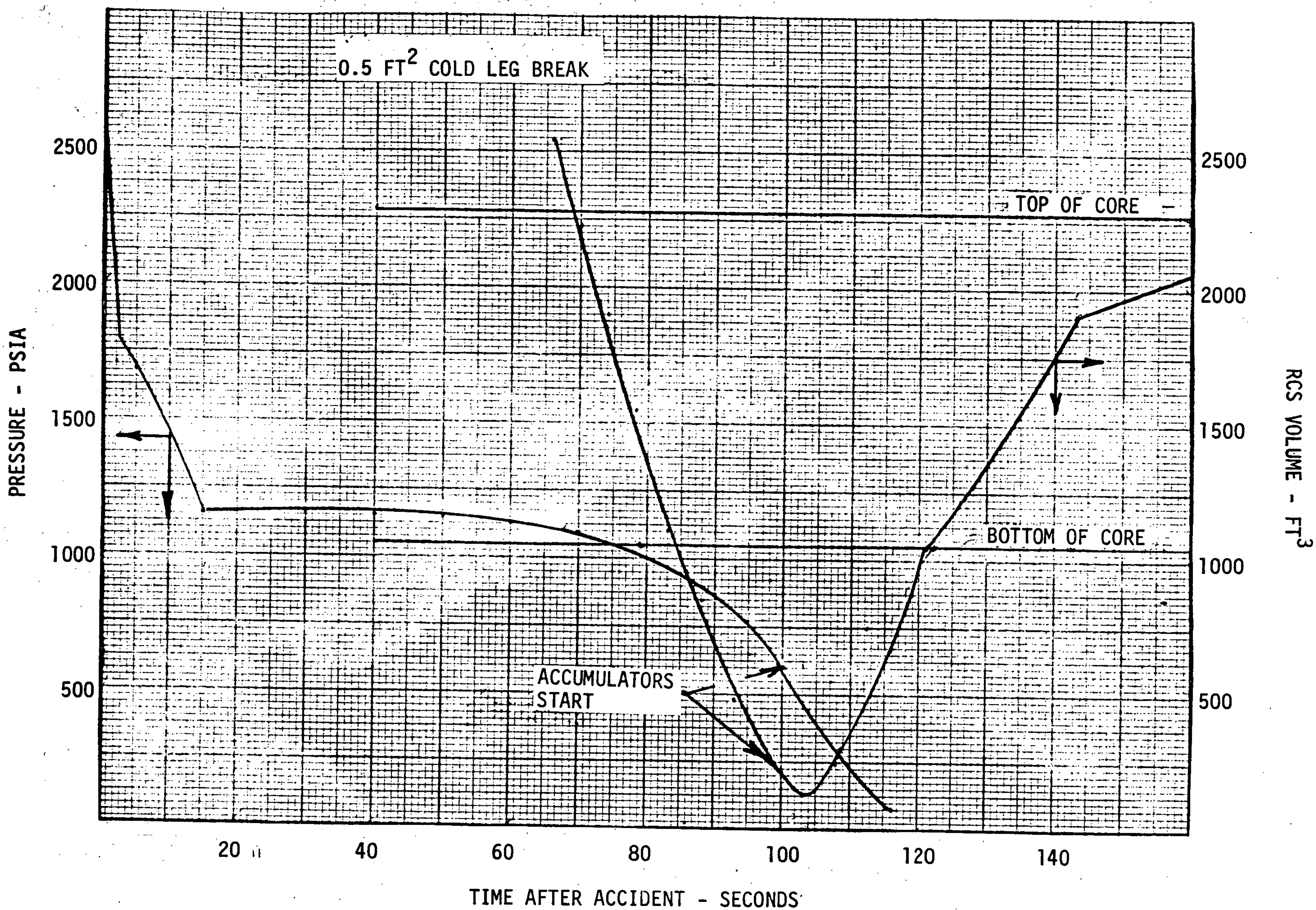

0.5 FT2 COLD LEG BREAK
TOP OF CORE
BOTTOM OF CORE
ACCUMULATORS START
PRESSURE - PSIA
RCS VOLUME - FT3
TIME AFTER ACCIDENT - SECONDS
2500
2000
1500
1000
500
20
40
60
80
100
120
140

FIG. 2-6

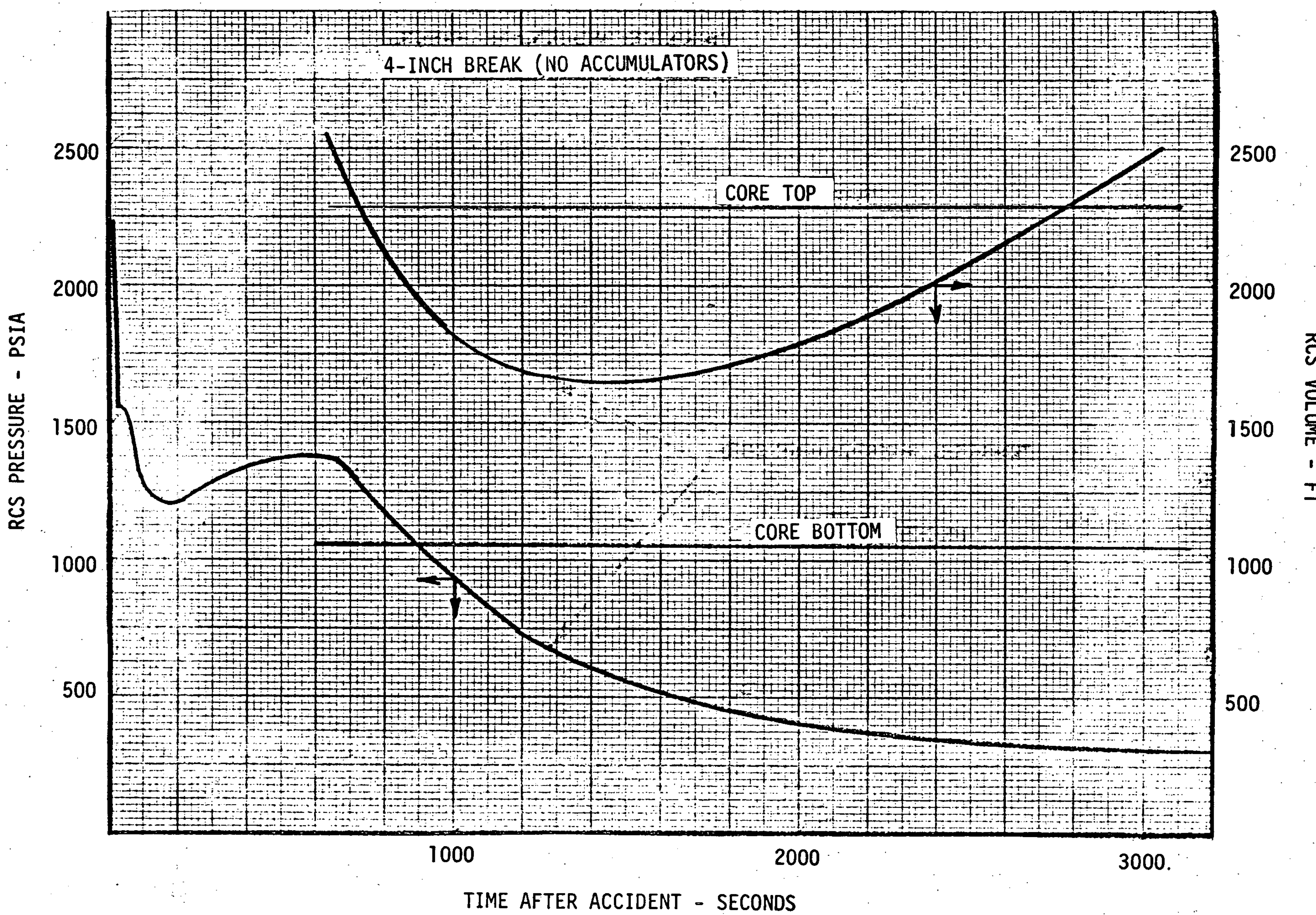

4-INCH BREAK (NO ACCUMULATORS)
CORE TOP
CORE BOTTOM
RCS PRESSURE - PSIA
2500
2000
1500
1000
500
RCS VOLUME - FT^3
2500
2000
1500
1000
500
1000
2000
3000.
TIME AFTER ACCIDENT - SECONDS

FIG. 2-7

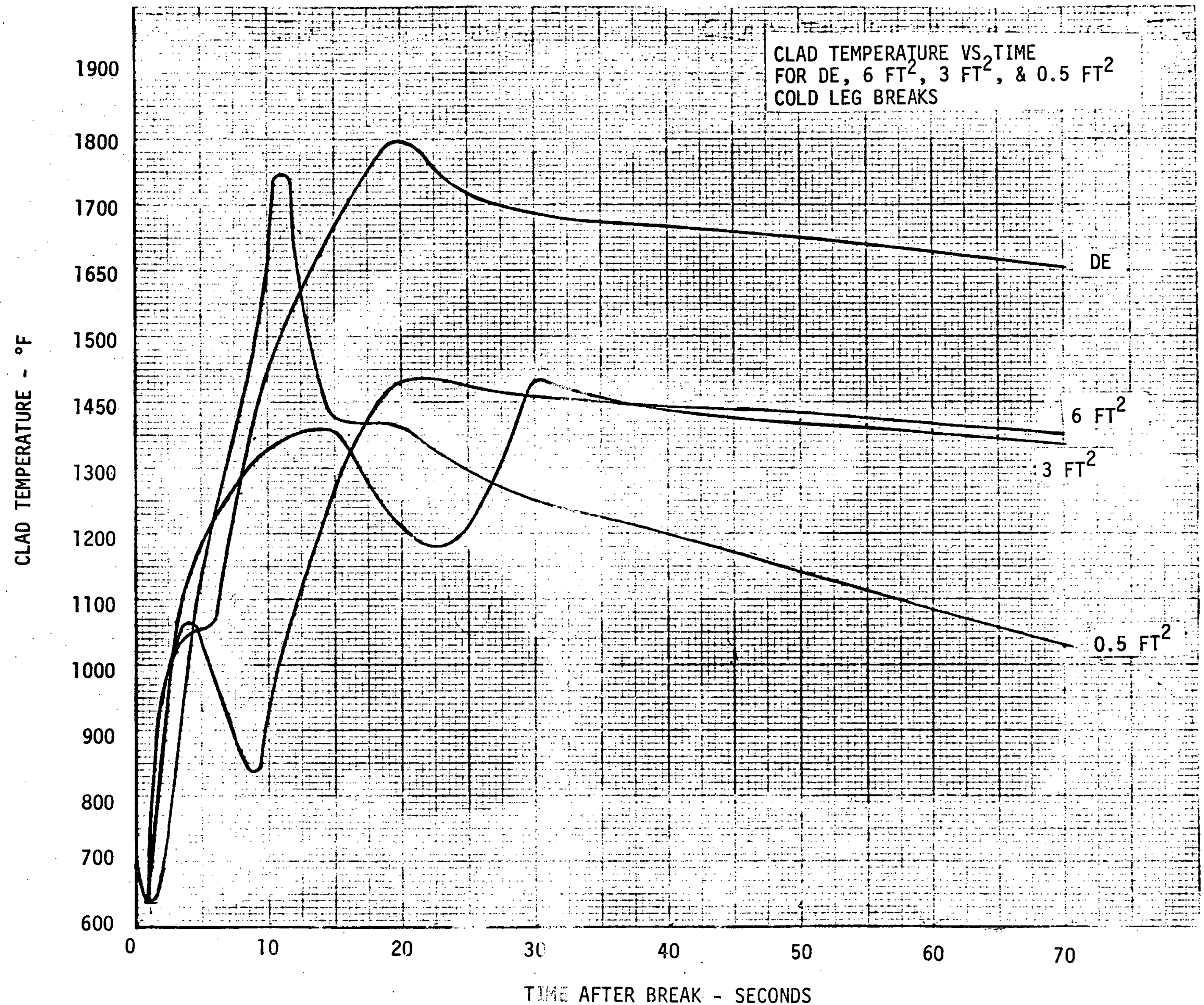
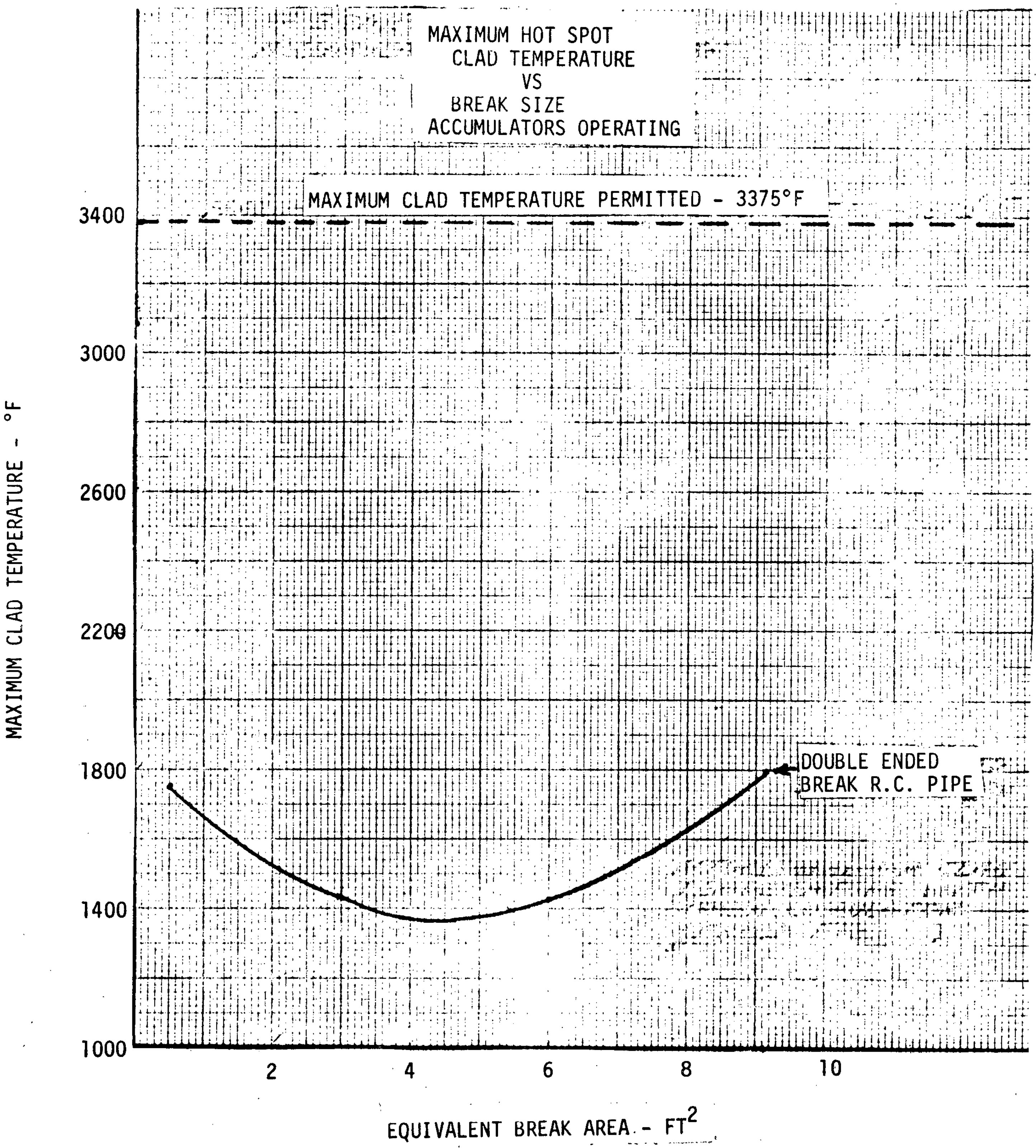
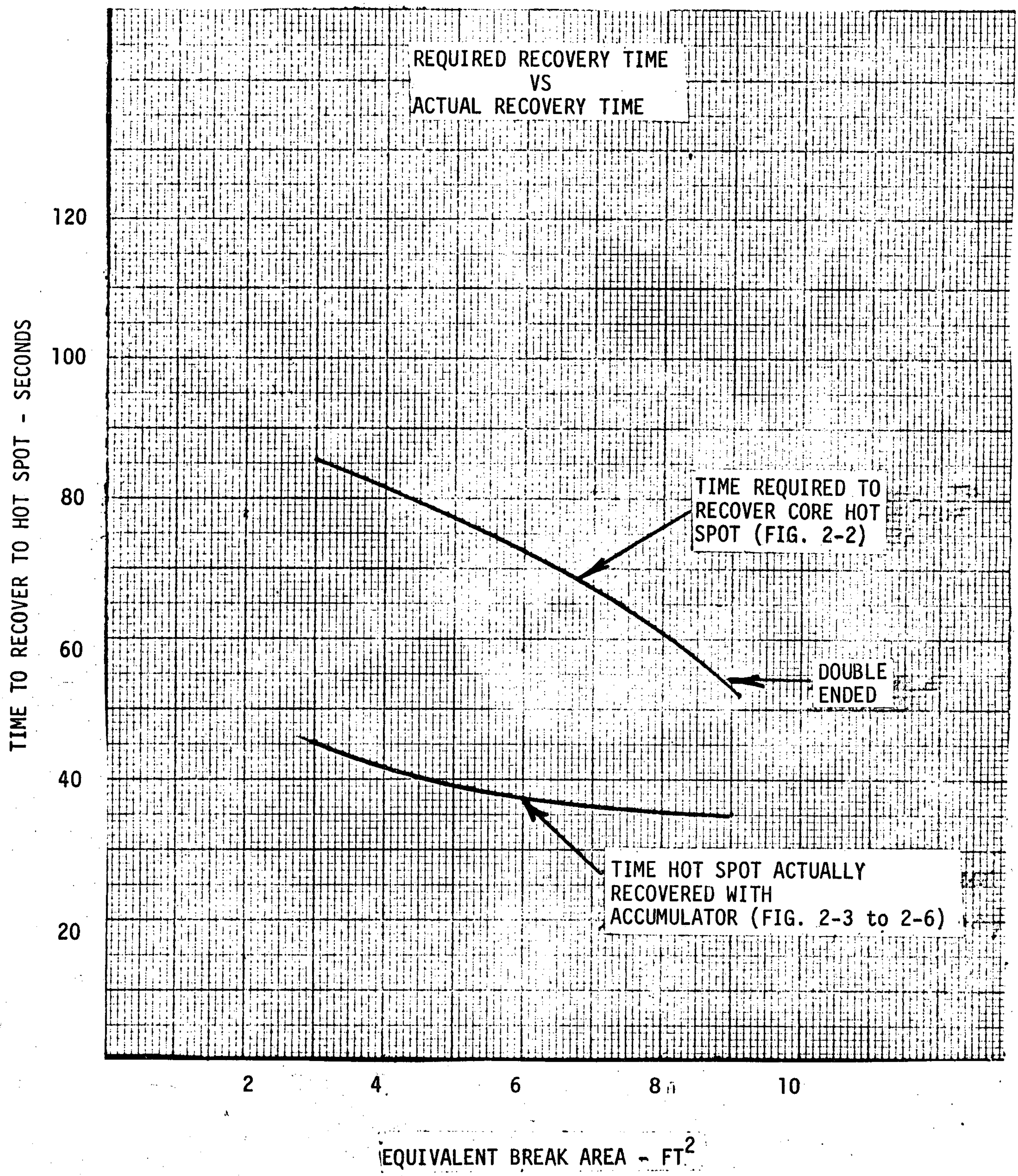

CLAD TEMPERATURE VS TIME
FOR DE, 6 FT^2, 3 FT^2, & 0.5 FT^2
COLD LEG BREAKS
CLAD TEMPERATURE - °F
1900
1800
1700
1650
1500
1450
1300
1200
1100
1000
900
800
700
600
0
10
20
30
40
50
60
70
TIME AFTER BREAK - SECONDS
DE
6 FT^2
3 FT^2
0.5 FT^2

FIG. 2-8

FIG. 2-9


MAXIMUM HOT SPOT CLAD TEMPERATURE VS BREAK SIZE ACCUMULATORS OPERATING
MAXIMUM CLAD TEMPERATURE PERMITTED - 3375°F
DOUBLE ENDED BREAK R.C. PIPE
3400
3000
2600
2200
1800
1400
1000
2
4
6
8
10
MAXIMUM CLAD TEMPERATURE - °F
EQUIVALENT BREAK AREA - FT2


REQUIRED RECOVERY TIME
VS
ACTUAL RECOVERY TIME
TIME REQUIRED TO RECOVER CORE HOT SPOT (FIG. 2-2)
DOUBLE ENDED
TIME HOT SPOT ACTUALLY RECOVERED WITH ACCUMULATOR (FIG. 2-3 to 2-6)
TIME TO RECOVER TO HOT SPOT - SECONDS
120
100
80
60
40
20
2
4
6
8
10
EQUIVALENT BREAK AREA - FT^2

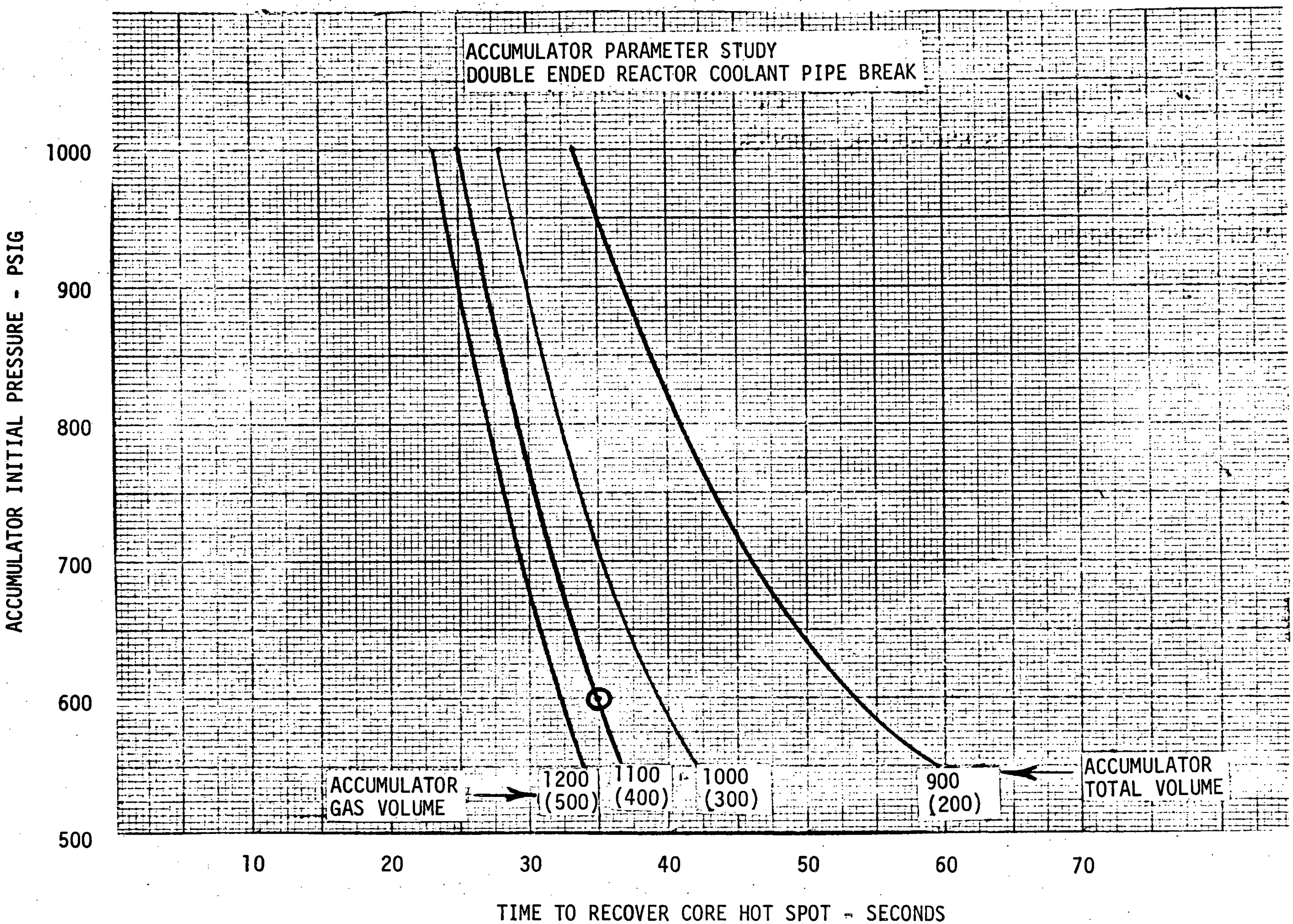
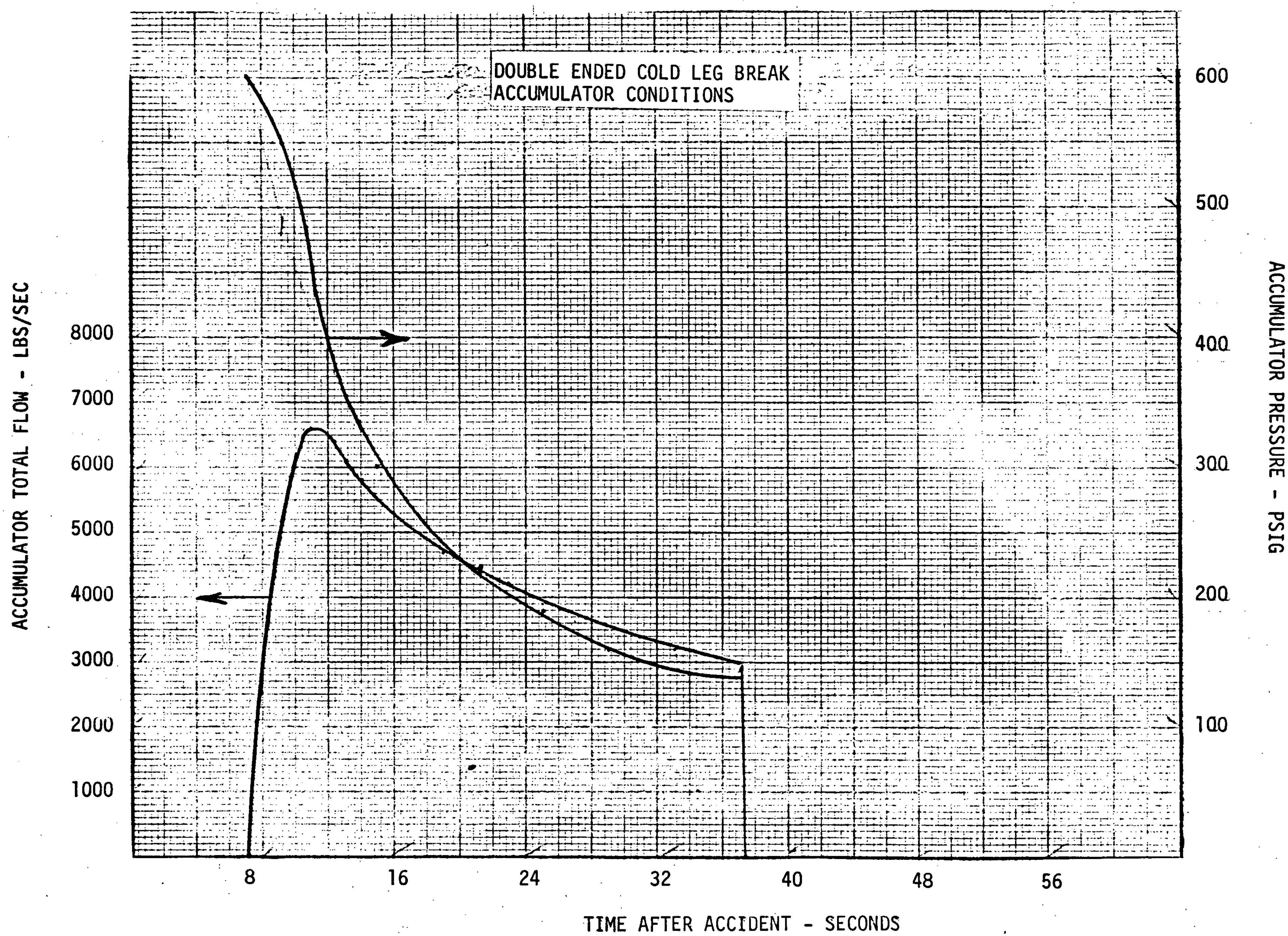

ACCUMULATOR PARAMETER STUDY
DOUBLE ENDED REACTOR COOLANT PIPE BREAK
ACCUMULATOR INITIAL PRESSURE - PSIG
1000
900
800
700
600
500
10
20
30
40
50
60
70
TIME TO RECOVER CORE HOT SPOT - SECONDS
ACCUMULATOR GAS VOLUME
1200 (500)
1100 (400)
1000 (300)
900 (200)
ACCUMULATOR TOTAL VOLUME

FIG. 2-11



DOUBLE ENDED COLD LEG BREAK
ACCUMULATOR CONDITIONS
ACCUMULATOR TOTAL FLOW - LBS/SEC
8000
7000
6000
5000
4000
3000
2000
1000
ACCUMULATOR PRESSURE - PSIG
600
500
400
300
200
100
8
16
24
32
40
48
56
TIME AFTER ACCIDENT - SECONDS

FIG. 2-12

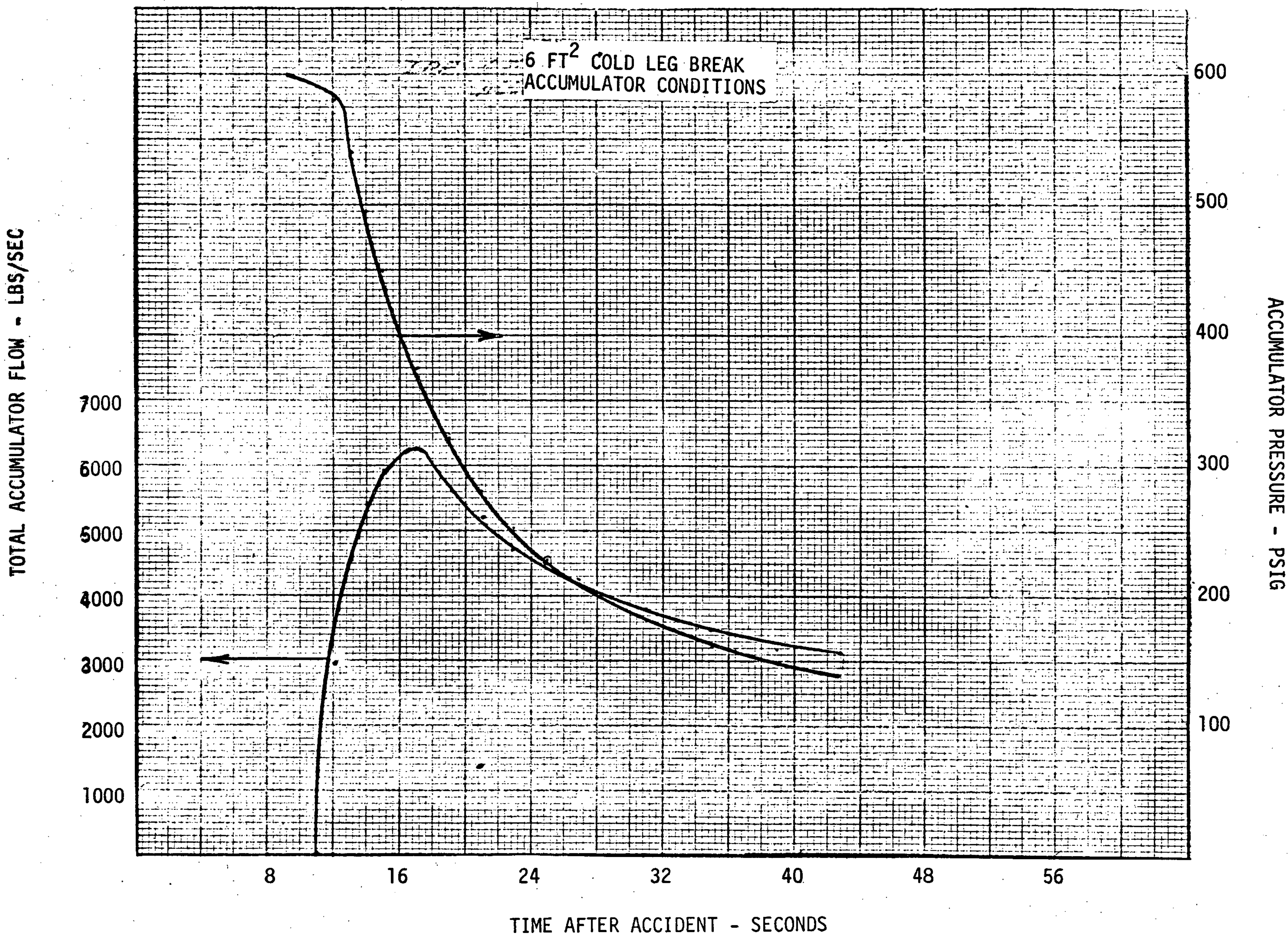
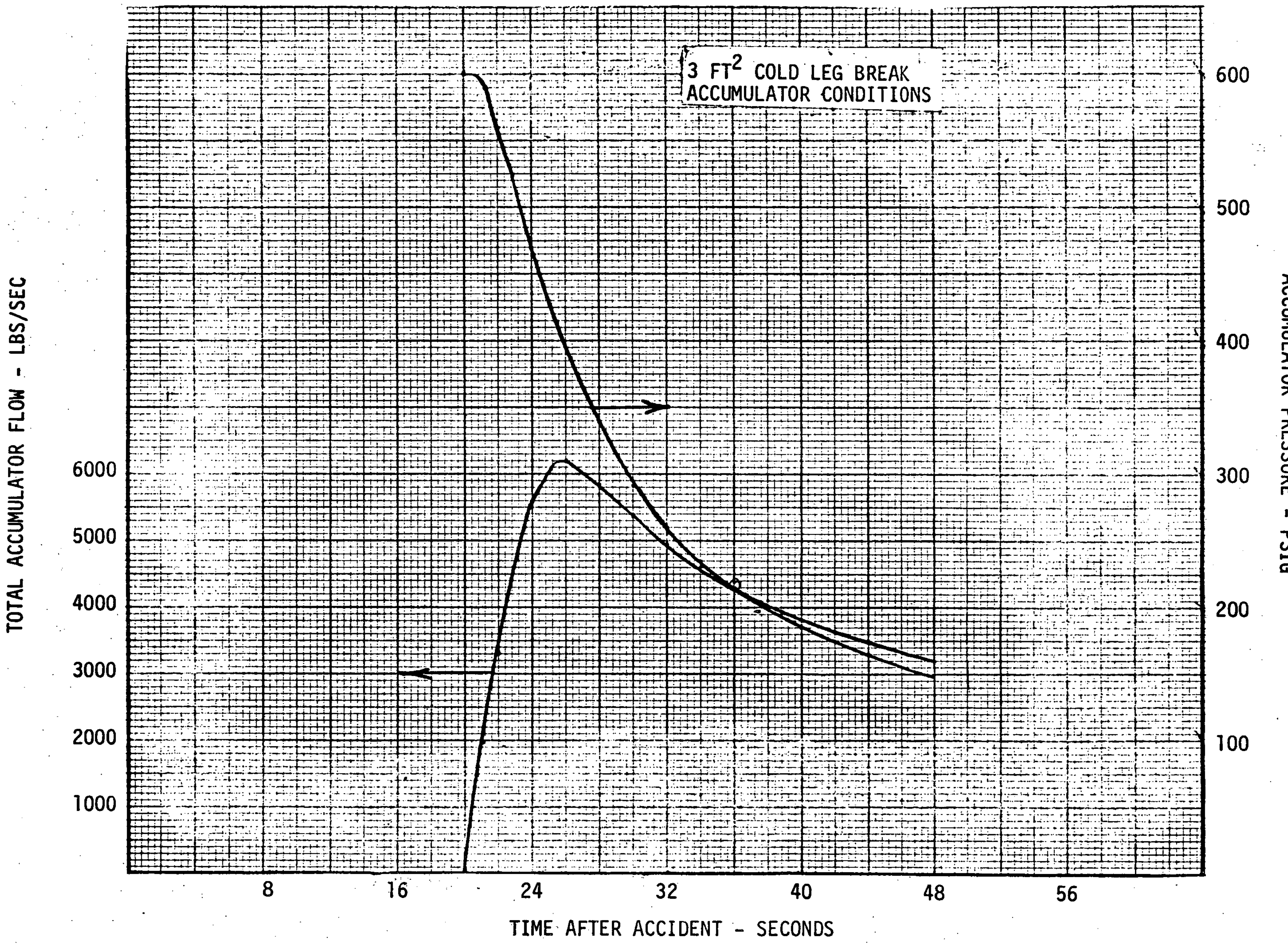
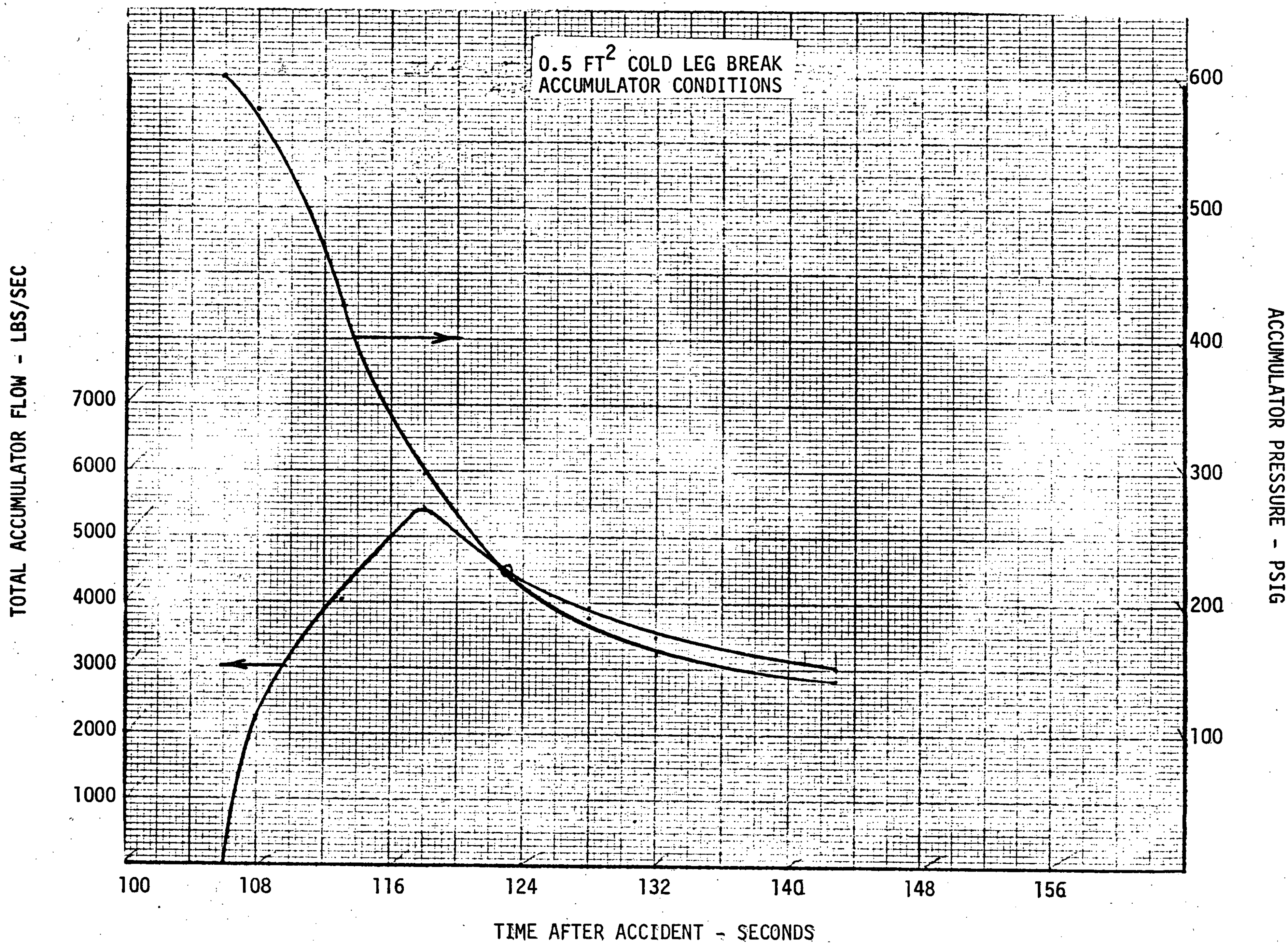

6 FT2 COLD LEG BREAK
ACCUMULATOR CONDITIONS
TOTAL ACCUMULATOR FLOW - LBS/SEC
7000
6000
5000
4000
3000
2000
1000
ACCUMULATOR PRESSURE - PSIG
600
500
400
300
200
100
8
16
24
32
40
48
56
TIME AFTER ACCIDENT - SECONDS

FIG. 2-13



3 FT² COLD LEG BREAK
ACCUMULATOR CONDITIONS
TOTAL ACCUMULATOR FLOW - LBS/SEC
6000
5000
4000
3000
2000
1000
ACCUMULATOR PRESSURE - PSIG
600
500
400
300
200
100
8
16
24
32
40
48
56
TIME AFTER ACCIDENT - SECONDS

FIG. 2-14



0.5 FT² COLD LEG BREAK
ACCUMULATOR CONDITIONS
TOTAL ACCUMULATOR FLOW - LBS/SEC
1000
2000
3000
4000
5000
6000
7000
ACCUMULATOR PRESSURE - PSIG
100
200
300
400
500
600
100
108
116
124
132
140
148
156
TIME AFTER ACCIDENT - SECONDS

FIG. 2-15

FIG. 2-16

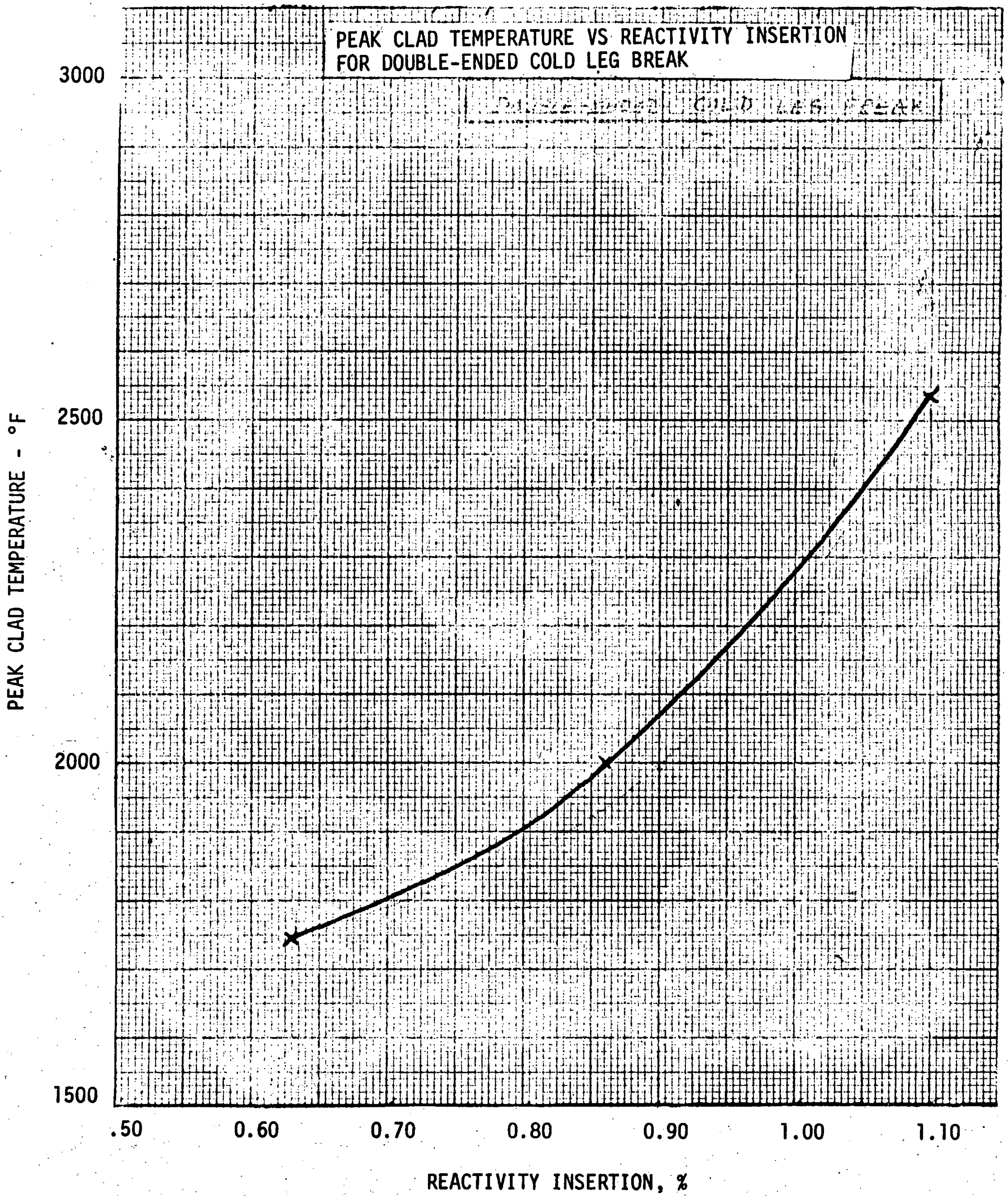

PEAK CLAD TEMPERATURE VS REACTIVITY INSERTION
FOR DOUBLE-ENDED COLD LEG BREAK
PEAK CLAD TEMPERATURE - °F
3000
2500
2000
1500
.50
0.60
0.70
0.80
0.90
1.00
1.10
REACTIVITY INSERTION, %

FIG. 2-17

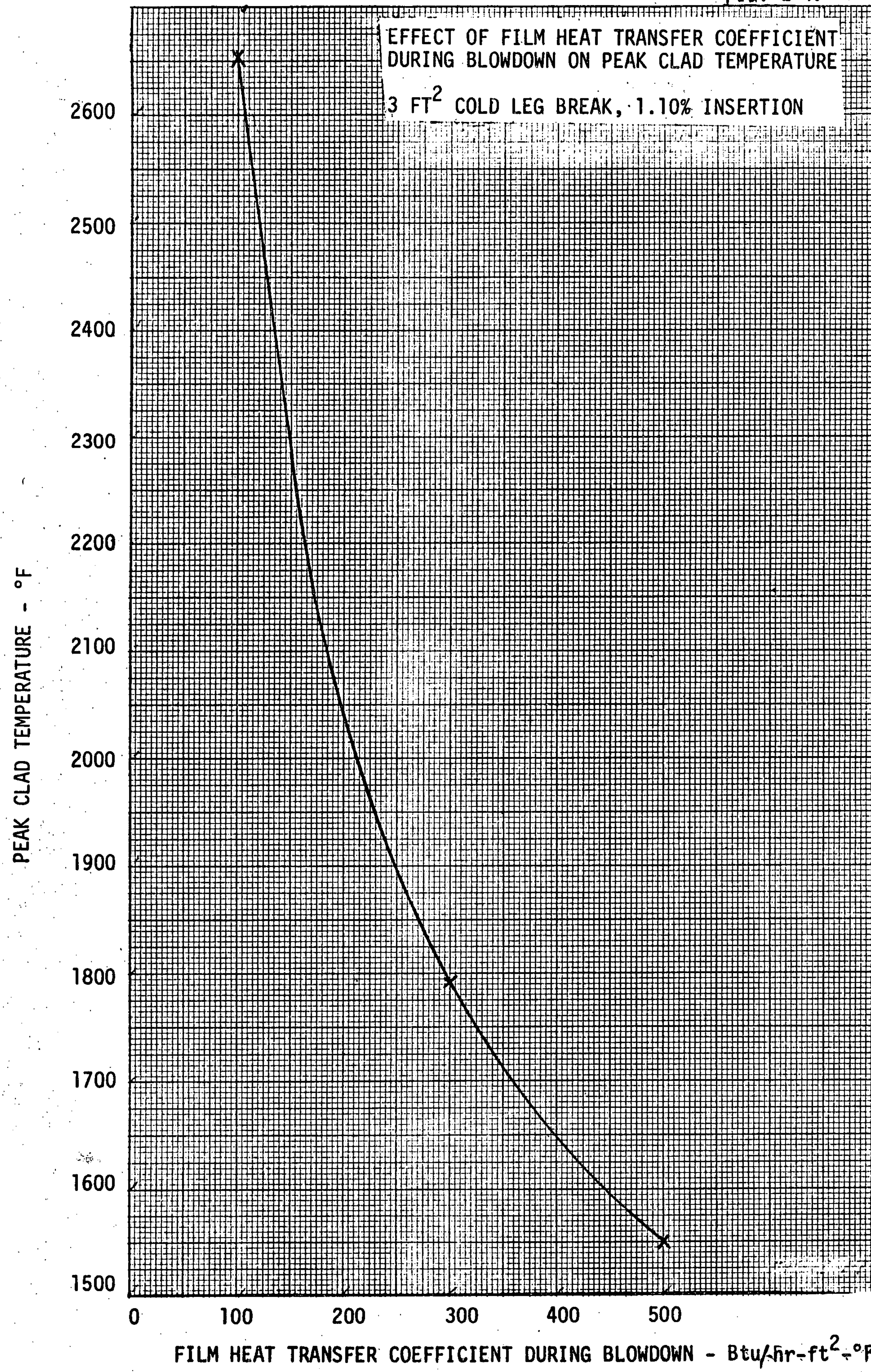
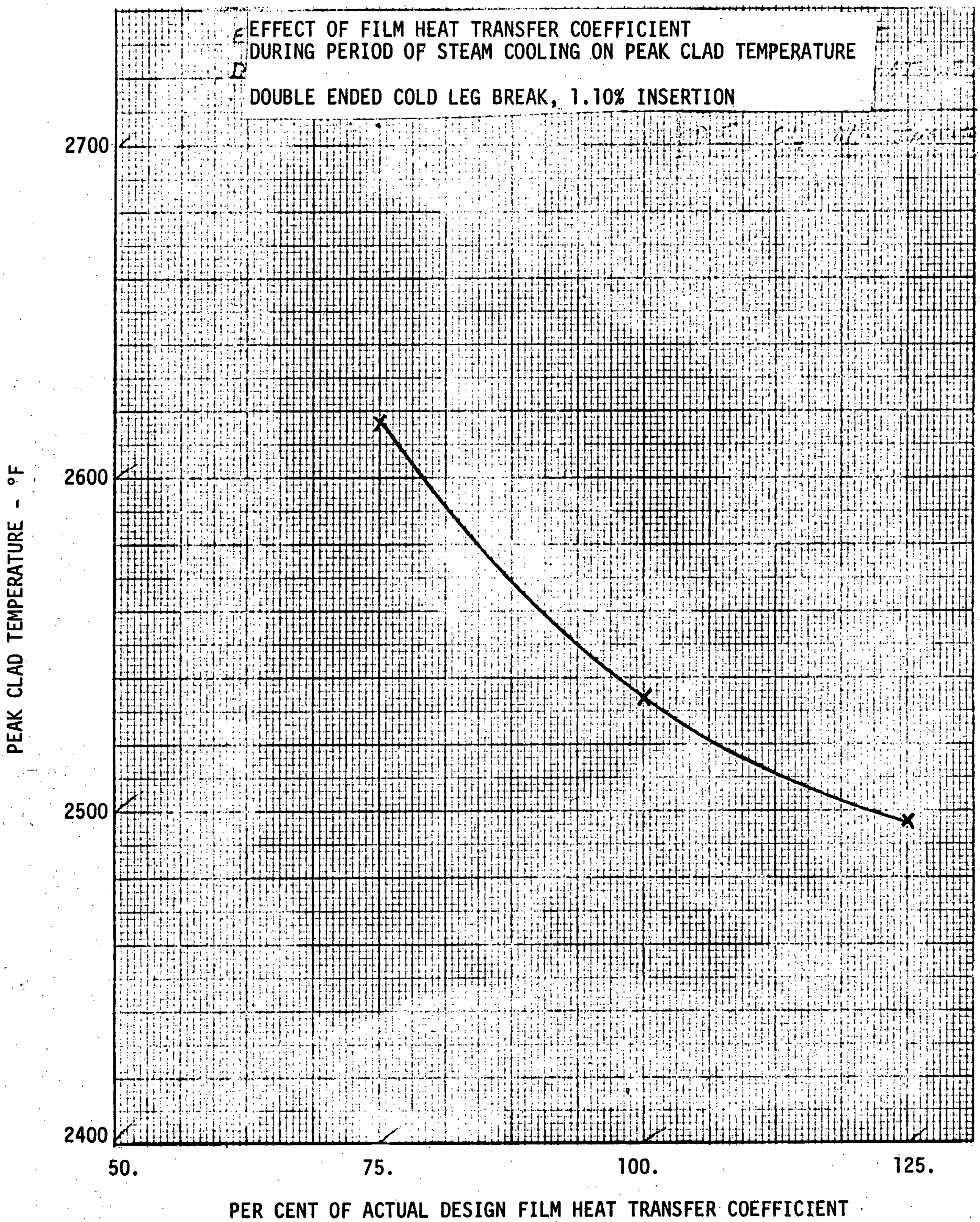
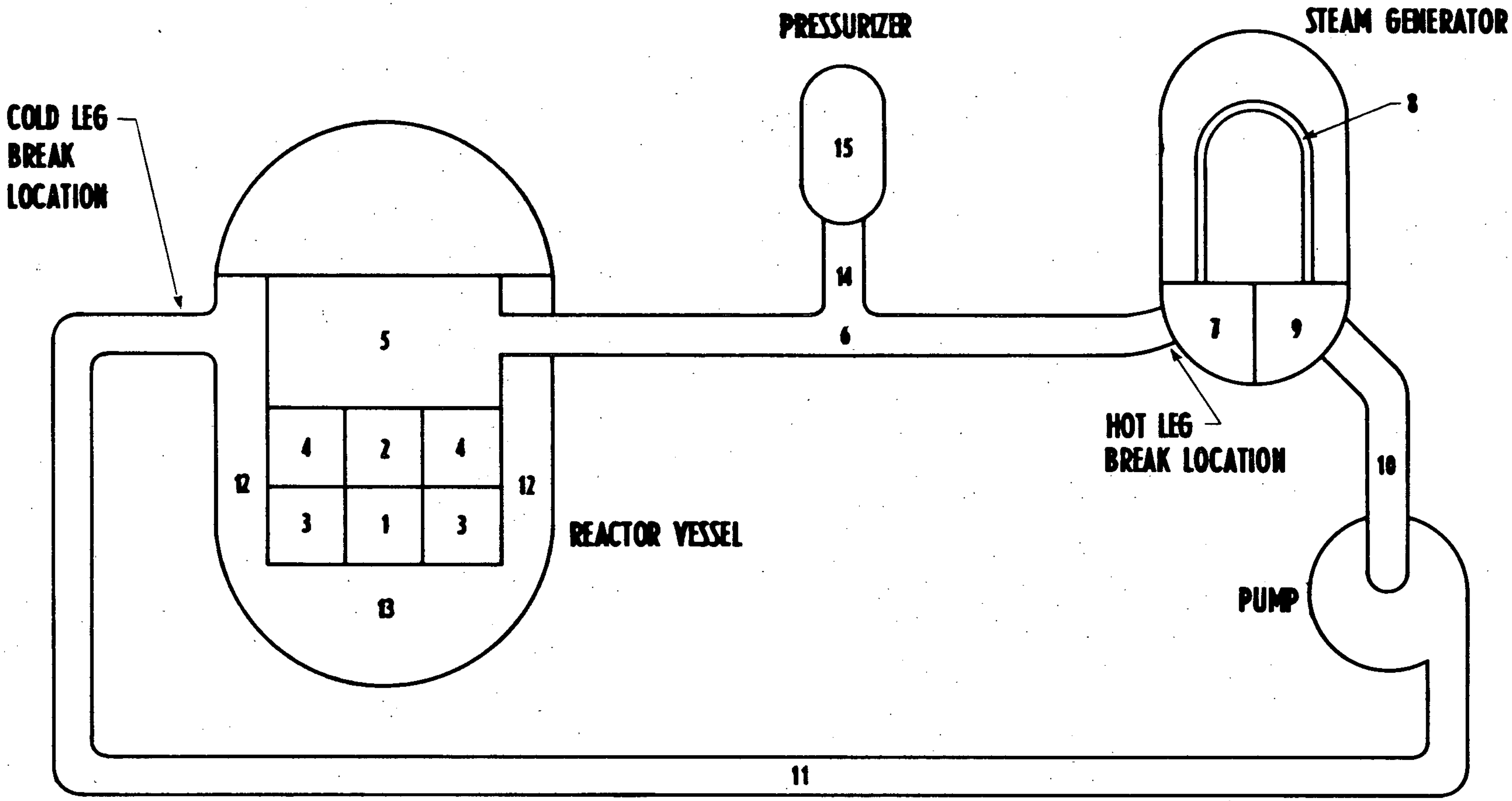

EFFECT OF FILM HEAT TRANSFER COEFFICIENT
DURING BLOWDOWN ON PEAK CLAD TEMPERATURE
3 FT2 COLD LEG BREAK, 1.10% INSERTION
PEAK CLAD TEMPERATURE - °F
2600
2500
2400
2300
2200
2100
2000
1900
1800
1700
1600
1500
0
100
200
300
400
500
FILM HEAT TRANSFER COEFFICIENT DURING BLOWDOWN - Btu/hr-ft2-°F

FIG. 2-18


EFFECT OF FILM HEAT TRANSFER COEFFICIENT
DURING PERIOD OF STEAM COOLING ON PEAK CLAD TEMPERATURE
DOUBLE ENDED COLD LEG BREAK, 1.10% INSERTION
PEAK CLAD TEMPERATURE - °F
2700
2600
2500
2400
50.
75.
100.
125.
PER CENT OF ACTUAL DESIGN FILM HEAT TRANSFER COEFFICIENT



REACTOR COOLANT SYSTEM ELEMENT DESIGNATION FOR SATAN CODE

FIG. 3-1

REACTOR COOLANT SYSTEM ELEMENT DESIGNATION FOR SATAN CODE 2 LOOP ANALYSIS

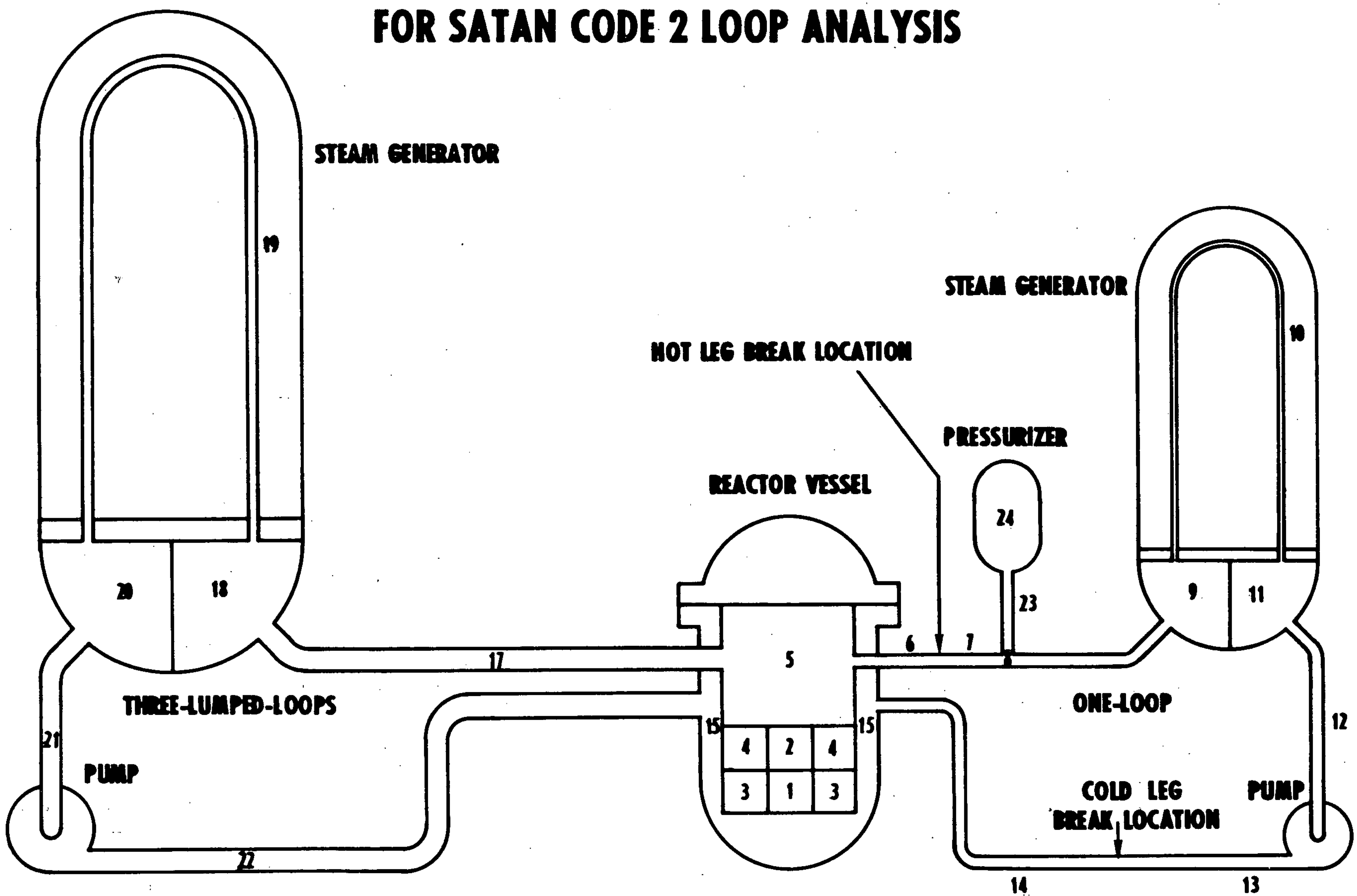


FIG. 3-2

COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT
(RESERVOIR BLOWDOWN)
VESSEL PRESSURE

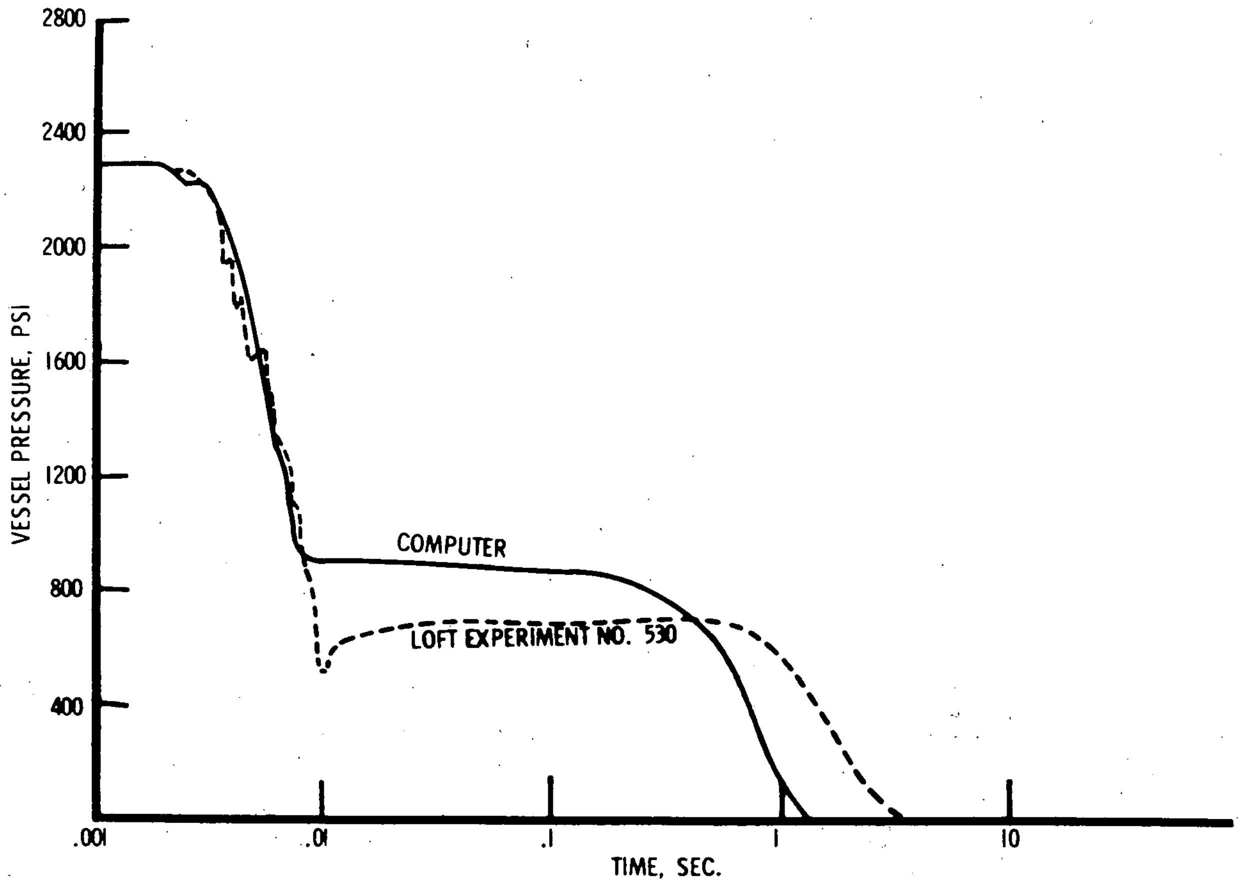


FIG. 3-3

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT
(RESERVOIR BLOWDOWN)
DISCHARGE PRESSURE

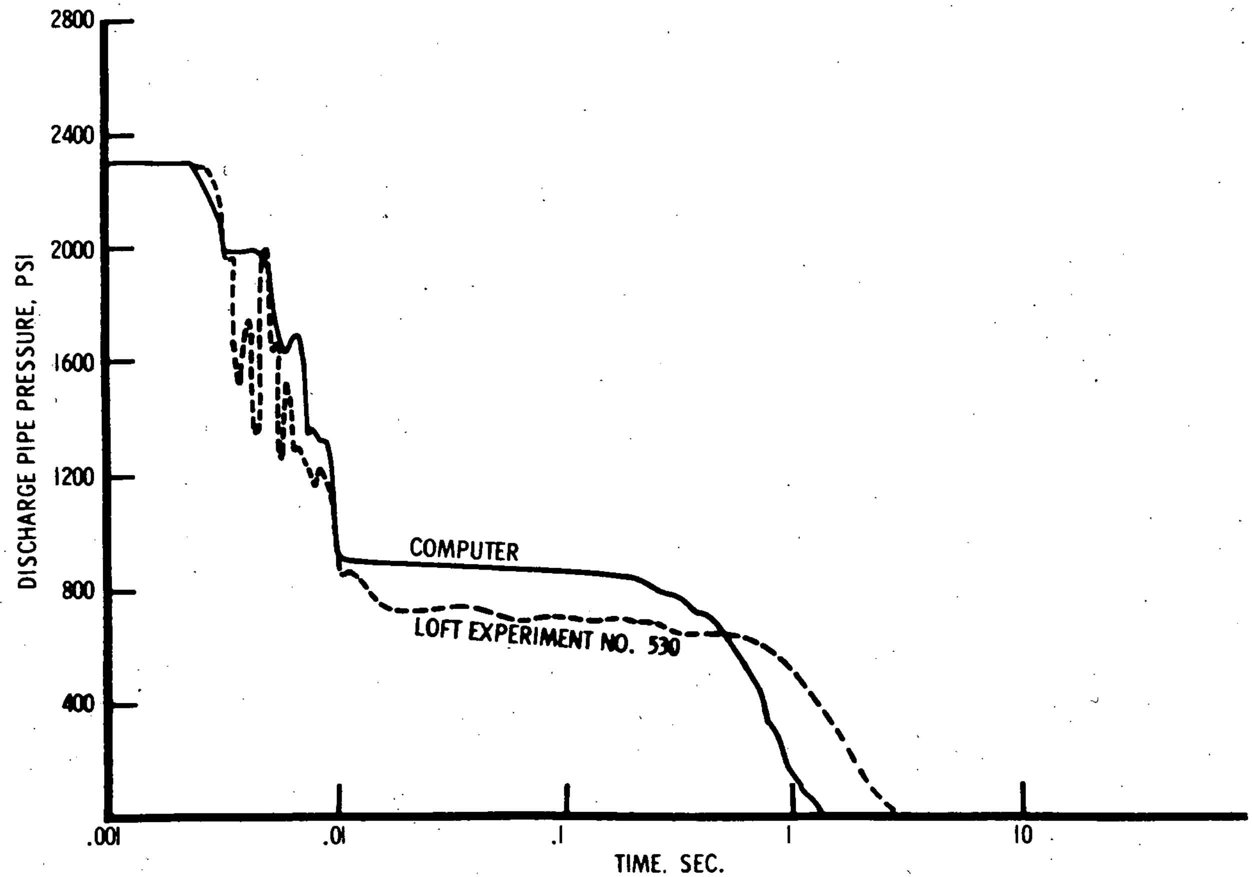


FIG. 3-4

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT
(RESERVOIR BLOWDOWN)
VESSEL TEMPERATURE

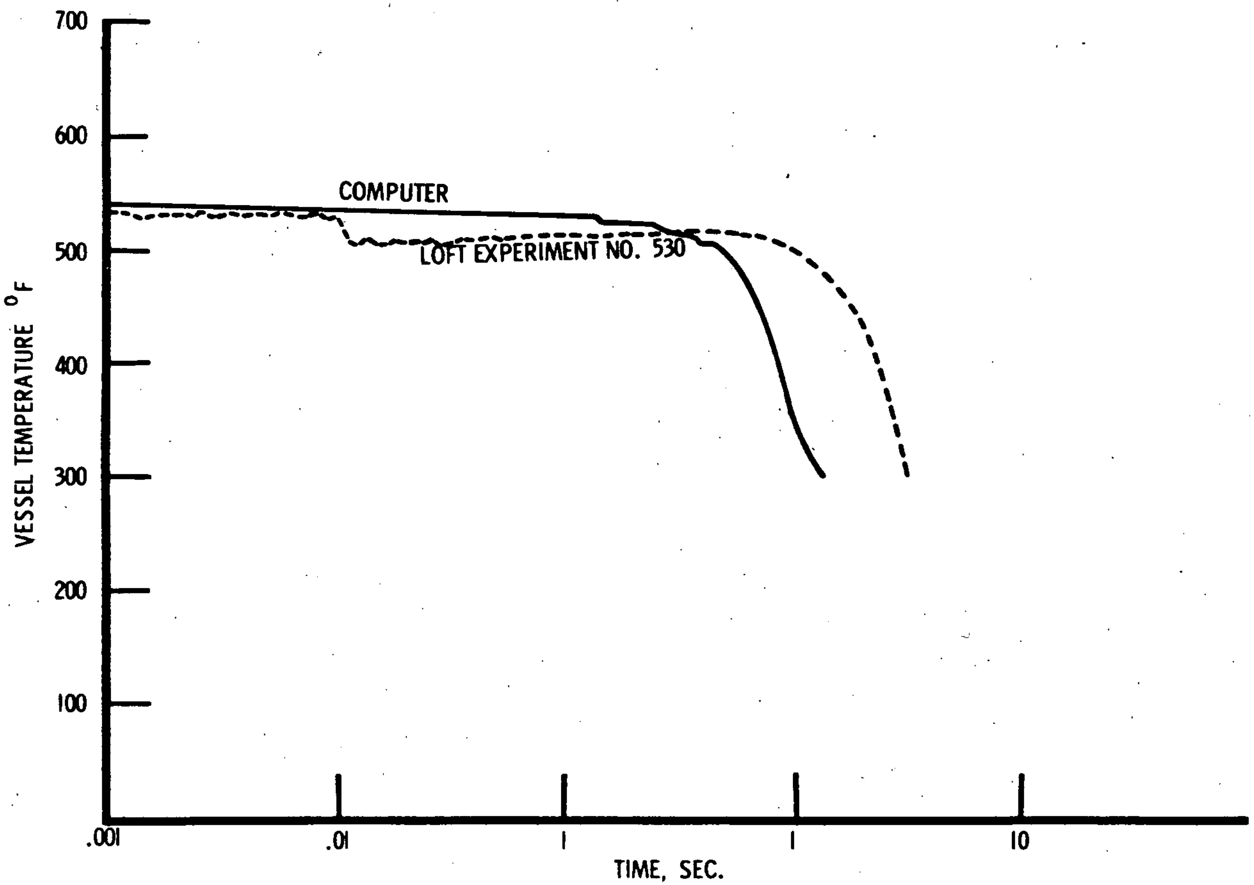


FIG. 3-5

A COMPARISON OF SATAN CODE RESULTS WITH LOFT EXPERIMENT
(RESERVOIR BLOWDOWN)
DISCHARGE PIPE TEMPERATURE

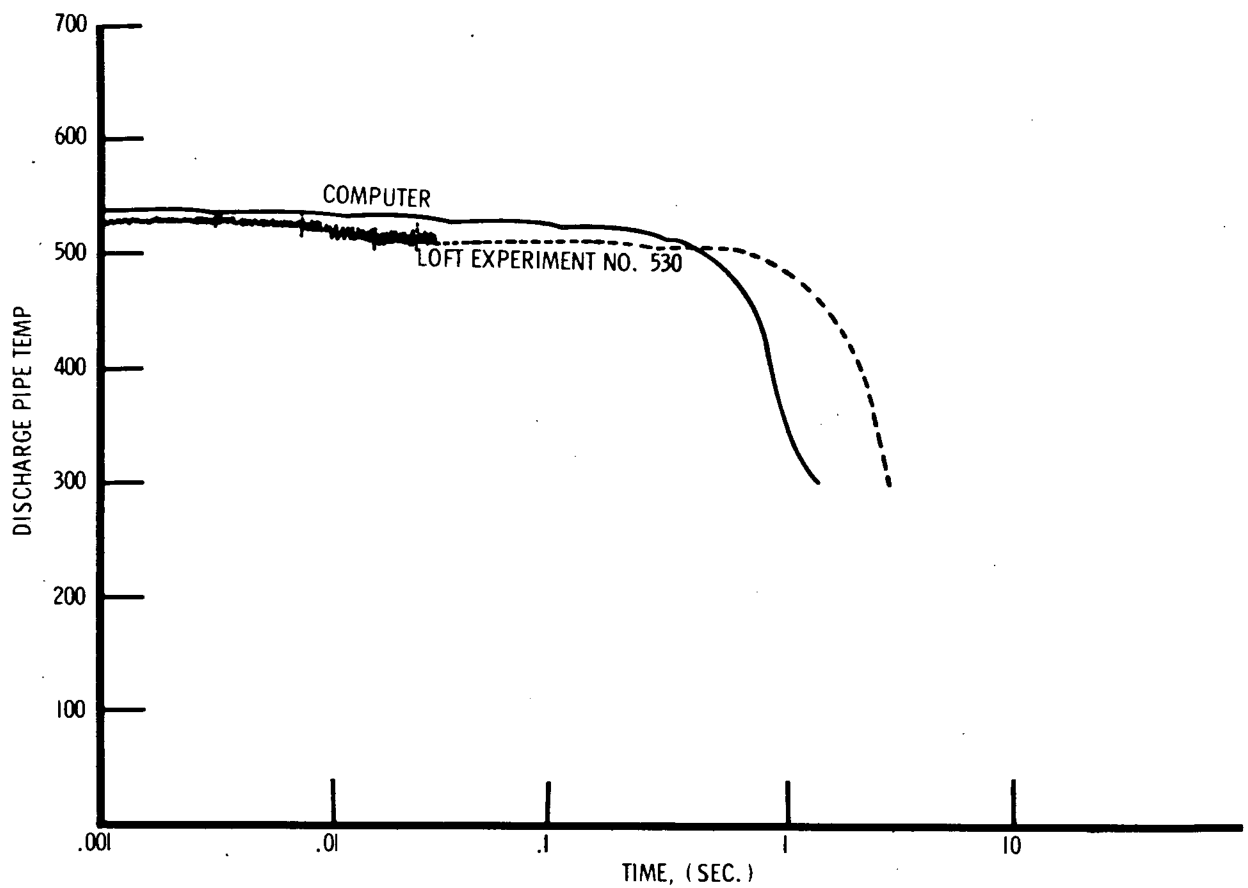


FIG. 3-6

HOT LEG - 0.0 SEC. RUPTURE TIME
CORE PRESSURE TRANSIENT, 0-4 SEC.

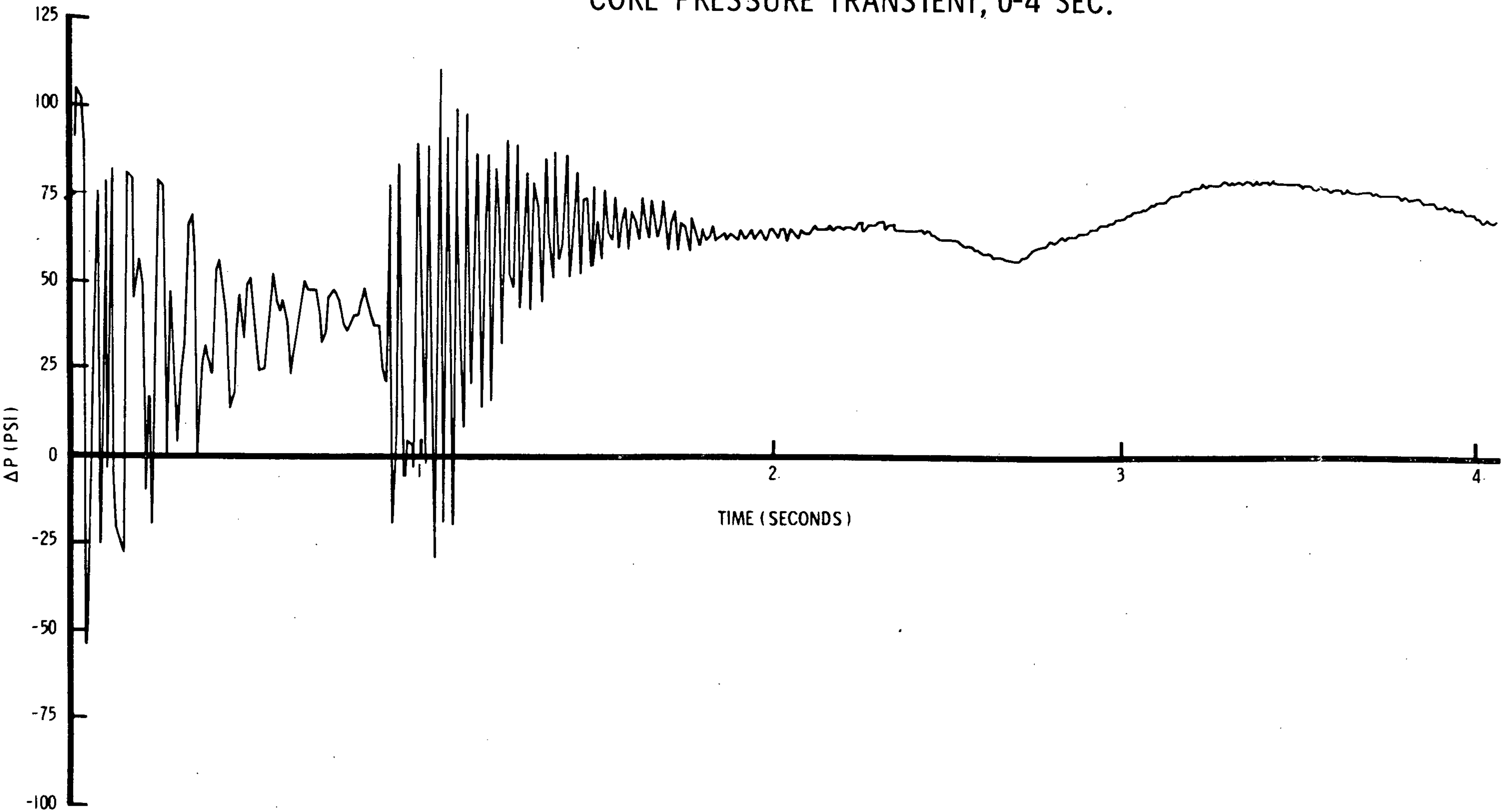


FIG. 3-7a

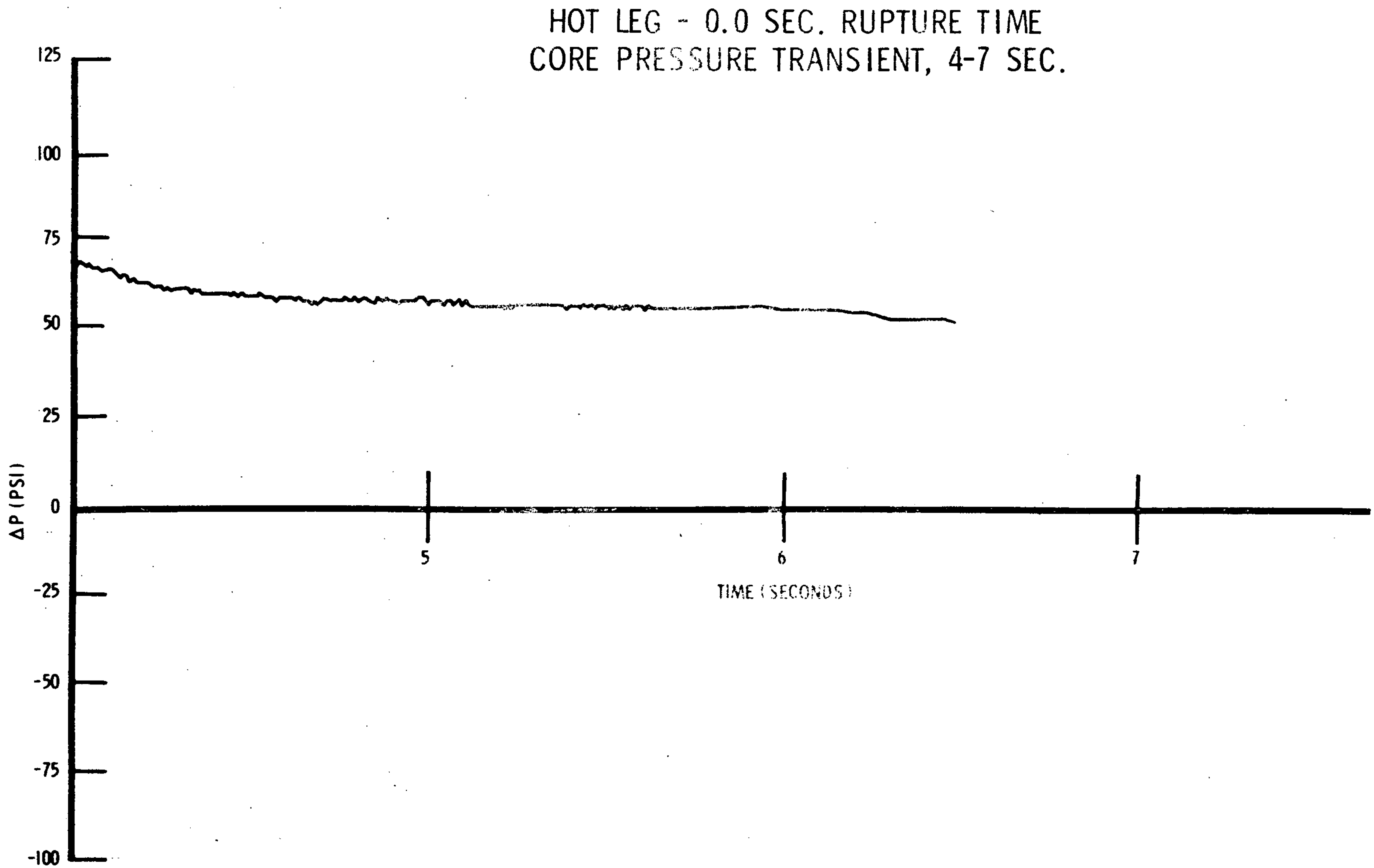

HOT LEG - 0.0 SEC. RUPTURE TIME
CORE PRESSURE TRANSIENT, 4-7 SEC.
125
100
75
50
25
0
-25
-50
-75
-100
ΔP (PSI)
5
6
7
TIME (SECONDS)

FIG. 3-7b

HOT LEG - 0.0 SEC. RUPTURE TIME
CORE BARREL PRESSURE TRANSIENT, 0-4 Sec.

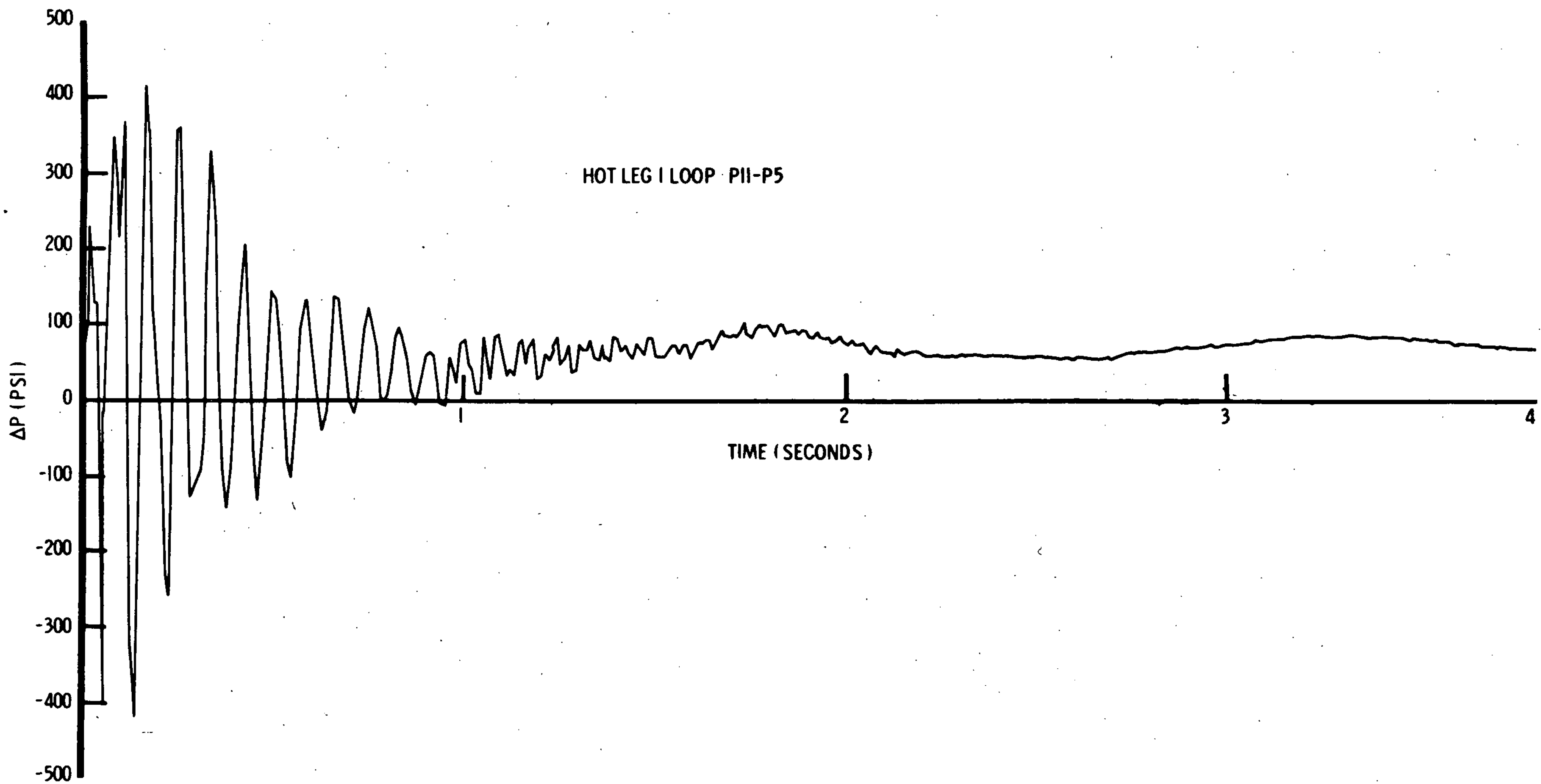


FIG. 3-8

3.0 EFFECT OF MAXIMUM LOSS OF COOLANT ACCIDENT ON REACTOR CORE AND INTERNALS INTEGRITY

3.1 CORE AND INTERNALS INTEGRITY REQUIREMENTS

The basic requirement of any loss of coolant accident, including the maximum hypothetical accident, is that sufficient integrity exist to permit the safe and orderly shutdown of the reactor. To insure that the basic requirement is met the following three sub-requirements must be met:

a) The basic configuration must be maintained. This implies that gross fracture and/or deformation of core and internals must not occur.

b) The ability to move control rods must be maintained so that they can be used to provide shut down even though insertion is not necessarily required following an accident.

c) Internals deformation must be sufficiently small so that primary loop flow, and particularly, adequate safety injection flow, is not impeded.

3.2 ANALYSIS OF FORCES AND PRESSURES ON INTERNALS AND CORE

The forces exerted on reactor internals and core, following a loss of coolant accident, are computed by employing the SATAN digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants.

This computer program generally provides a means for the study of the nuclear plant dynamic behavior for a variety of conceivable plant transient operation and accidents. In particular, it can be used to determine the reactor coolant system dynamic characteristics after the occurrence of a rupture in the main coolant piping. SATAN is generally similar to, yet more elaborate than, the FLASH code. It can be employed to determine the shock-wave effects in the reactor coolant system over short time periods in great detail. It can also handle a larger number of elements along the reactor coolant loops.

The SATAN mathematical formulation employs a three-dimensional approach (length along the flow path, radius and time) in the reactor core; a two-dimensional approach (length along the flow path and time) in the reactor coolant loops; and a lumped-parameter representation for the secondary system. The reactor coolant system is assumed to consist of one or two parallel loops. The reactor core and the primary loops are sectionalized into elements with variable spatial mesh size. The present memory capacity of the IBM-7094 allows a maximum of 70 spatial elements for the system, which may be distributed between the core and the reactor coolant loops in any desirable fashion. Two typical schemes for the simulation of the reactor coolant system are shown on Figures 3-1 and 3-2. Figure 3-1 portrays a one-loop representation which can accommodate leaks over an appropriate size range in the hot leg (element 7) or in the cold leg (element 11). Figure 3-2 is a two-loop representation. In this arrangement three parallel loops are lumped as one-loop and double-ended complete severance ruptures can be accommodated on the fourth loop either in the hot leg (between elements 6 and 7) or in the cold leg (between elements 13 and 14).

The thermodynamic state of the coolant in any element is defined by:

a) a set of main dependent variables; and
b) a set of auxiliary dependent variables.

The main dependent variables are those variables computed by the integration of a differential equation. The main dependent variables are: coolant pressure, enthalpy, element liquid mass, element vapor mass, axial mass flow rate and radial (or branch) mass flow rate. The auxiliary dependent variables are: coolant steam quality, density and temperature. To compute the main and auxiliary dependent variables, the following fundamental equations are first expressed for each element:

a) The continuity equation;
b) The energy equation;
c) The momentum equation;
d) The state equation.

These equations are then rearranged so that the time derivative of the main dependent variables and the values of the auxiliary dependent variables are explicitly defined in terms of the present system variables. These explicit equations form a system of simultaneous differential equations which is integrated numerically on the computer to obtain the time-variation of all dependent variables.

The SATAN program has several distinctive features summarized as follows. The code can accommodate:

a) A leak or a complete-severance double-ended rupture of a desired size with zero or a given rupture time for any element in the hot or the cold leg.
b) Flow reversal for any element.
c) Subcooled, two-phase, or superheat flow in any element.
d) Critical choking flow at the rupture point.
e) Heat transfer in the core and the steam generator.

To verify the correctness of the mathematical formulation and the various numerical techniques employed, the code was checked against LOFT experimental results presently available. The LOFT semi-scale blowdown test facility is a 12 inch dia. 10 ft. long cylindrical reservoir connected to a 4 inch dia. pipe plugged at the end by rupture discs. The reservoir is filled with high enthalpy, high pressure water. The rupture discs are then broken. The pressure and temperature time-variation in the reservoir and the discharge pipe are then recorded.

The SATAN program was used to simulate the LOFT reservoir blowdown test numbers 530 and 522. Figures 3-3 and 3-6 show that the pressure and temperature time history obtained from SATAN are in good agreement with LOFT experimental data. The discrepancies observed at the onset of two-phase blowdown is due to metastability phenomenon when the fluid undergoes a non-equilibrium flow and predicts higher pressures with subsequently larger and more conservative blowdown discharge and reduced blowdown time.

Having gained confidence in the validity of the SATAN program results, the code was applied to the Indian Point Unit No. 2 blowdown analysis. The plant was first brought to steady-state from initially uniform pressure and enthalpy distributions throughout the system. The computed steady-state variables were then checked against the available plant thermal and hydraulic characteristics and found to be in a very good agreement. This steady-state verification provided a further proof for the validity of the digital program. Starting from these steady-state values, the following blowdown analyses were performed:

1) One-loop system with large leak (area equal to double-ended rupture from one actual loop) on hot leg with zero rupture time;

2) One-loop system with large leak (area equal to double-ended rupture from one actual loop) on cold leg with zero rupture time;

3) Two-loop system with double-ended rupture (complete severance) on the hot leg of one actual loop with zero rupture time;

4) Two-loop system with double-ended rupture (complete severance) on the cold leg of one actual loop with zero rupture time.

From the pressure-time history of core inlet and outlet plenums and the vessel inlet and outlet plenums the pressure difference across the core and the upper plenum was respectively computed and used to determine forces on the reactor internals. Figures 3-7a, 3-7b, and 3-8 show typical pressure-time histories.

Sensitivity studies were performed to evaluate the effect of rupture time on pressure gradients. It was found that rupture times of the order of 0.3 sec. considerably reduces the values of pressure gradients on reactor internals by at least a factor of two as compared with the pressure gradients obtained with zero rupture time used in the present analysis.

3.3 EFFECTS OF FORCES AND PRESSURES ON CORE AND INTERNALS

3.3.1 CORE

During a hot leg break, the difference in pressure across the core is oscillatory for approximately 2 seconds (Figures 3-7a and 3-7b) and later on is approximately constant. The largest longitudinal force on the fuel assembly will occur during the initial transient and will reach a value of 6,700 pounds per fuel assembly in compression.

During a cold leg break the longitudinal compressive force on the fuel assembly has a peak value of approximately 9,900 pounds.

The force to buckle a fuel assembly is of the order of 85,000 pounds. The inconel grids connecting the fuel rods are able to maintain the rods in position during the transient.

3.3.2 CORE SUPPORT STRUCTURE

As a consequence of the dynamic effect during the initial transient following a hot leg break, the upper core support structure has a maximum deflection upward of 0.120 inches and the maximum total stresses (approximately 12,000 psi) occur a the grid and upper support plate ligaments.

After the first 2 seconds, the force on the upper core support structure becomes approximately constant; each fuel assembly exerts a force of approximately 2920 pounds. The upper core support structure deforms 0.057 inches under this load. Maximum total stresses are at the grid and upper support plate (approximately 12,700 psig).

Following a cold leg break, there will be no significant effects on the upper core support structure because the external forces on the core are predominantly downward.

3.3.3 LOWER CORE SUPPORT STRUCTURE

Following a hot leg break the maximum total stress occurs at the lower girth weld during the initial transient and is approximately 15,000 psi.

Following a cold leg break the maximum total girth weld stress is approximately 11,000 psi.

3.3.4 THERMAL SHIELD

The thermal shield is rigidly connected to the core barrel and will not be adversely affected by pressure and flow transients following either a hot or cold leg break.

3.3.5 UPPER CORE BARREL

The pressure across the upper core barrel becomes oscillatory during an accident and, to establish its behaviour, buckling and radial natural frequency as a short cylinder were computed. The results are:

Buckling Pressure	-	850 psi $< p_{cr} <$ 2400 psi
Extensional Natural Frequency	-	f > 1180 cps
Bending Natural Frequency	-	f = 60 cps - 100 cps

During the first 0.5 seconds following a hot leg break, the difference in pressure across the core oscillates between +400 and -400 psi at a frequency of 15 cps which is small compared with the natural frequencies listed above. The corresponding stress level is $\sigma = \pm$ 13,500 psi. After this initial transient, this difference in pressure remains approximatley constant and much smaller (∿75 psi inward).

The maximum pressure oscillation following a cold leg break is $\Delta p = \pm$ 600 psi at a frequency of f = 17 cps producing a stress level of $\pm$ 20,300 psi assuming completely elastic behaviour.

3.3.6 RCC GUIDE SYSTEM

The RCC guide system will not be adversely affected following either a hot or cold leg break. Compressive stresses (approximately 16,800 psi), which are below the yield strength of the material will be present in the fuel assembly thimbles without affecting the integrity and/or the stability of the system.

During a hot leg break, cross flow in region above the core will stress the guide column near the outlet nozzle leading to the break producing a maximum bending stress of 6,000 psi for the initial flow peak immediately after the break. Then the flow reduces to a value below the normal steady-state flow.

During a cold leg break the effect of transverse flow across the guide columns is much smaller than for the hot leg break.

In each of the cases described above, the stresses and deformations which would result following either a hot or cold leg break are less than those which would adversely affect the integrity of the structures. Also, the natural and applied frequencies are such that resonance problems would not occur. Therefore, it is concluded that:

a) The forces imposed, due to either a hot or cold leg break, can be sustained by the internals system,

b) The integrity of the internals system is maintained, and,

c) The basic requirements described in Section 3.1 are met.

STATUS ON INTERNALS DESIGN AND FABRICATION

Part	Stage of Completion		
	Design Committed	Material Ordered	Fabrication Started
Mechanism	X	X	
Drive Shafts	X	X	
Lower Internals	X	X	X
Upper Internals	X	X	X

4.0 PRIMARY SYSTEM DESIGN AND FABRICATION TECHNIQUES AND IN-SERVICE INSPECTION

4.1 PRIMARY SYSTEM DESIGN AND FABRICATION TECHNIQUES

4.1.1 INTRODUCTION

Extensive consideration has been given to the requirements of design, fabrication and inspection to produce highest quality components for use in the reactor coolant system, assuring adequate conservatism and full use of practical existing inspection techniques.

4.1.2 DESIGN

The rules of Section III Nuclear Vessels (ASME B&PV Code) provide an up-to-date industry-wide acceptable basis for design evaluation of nuclear vessels. The criteria established by Section III are used for evaluating the design of the reactor vessel, pressurizer, coolant pump casing and steam generator.

The ground rules established for fatigue analysis, unique to Section III of the boiler codes are based on low cycle fatigue considerations and cover discontinuities, stress raisers, and thermal as well as pressure and mechanically induced stresses. The conservatism of the fatigue design curves in Section III has been verified by the cyclic testing of pressure vessels carried out by Southwest Research Institute for the AEC and the Pressure Vessel Research Council.

The reactor coolant piping is analyzed in accordance with the requirements of ASA B31.1 Code for Pressure Piping. While this procedure does not categorize all stresses in the same manner as Section III, primarily the expansion stress which is peculiar to piping, it does provide a basis for fatigue analysis which has beencorrelated with strain cycling of piping subassemblies by Markl of Tube Turn, Inc. This analysis takes into account the interrelation of the primary system components, piping and supports.

The design specifications include all of the loads the reactor vessel, pressurizer, pump casing and steam generator will carry which include static and dynamic loads from internal or external sources, steady state, operational and abnormal transient conditions expected during the life of the system. The hydraulic forces imposed by the instantaneous failure of main coolant piping are considered in the design as previously reported in Supplement 5. The operational modes of the plant are based on a conservative evaluation of nuclear plant operation coupled with experience from existing nuclear plant such as Yankee-Rowe and include startup and shutdown cycles, load step changes, reactor trips and pre-startup hydrostatic tests.

The design pressure of the reactor coolant system components provides a 10% margin above the steady state operating pressure, providing a significant range to cover transients without exceeding the setting of the safety valves.

A complete stress analysis which reflects consideration of all design loadings detailed in the design specification is prepared by the manufacturer to assure compliance with the stress limits of Section III for the reactor vessel, steam generator, pressurizer and pump casing. Westinghouse independently will review these stress analyses. A similar analysis of the piping will be prepared by Westinghouse or for Westinghouse by a qualified piping analysis contractor.

As part of the design control on materials, and in addition to that reported previously for the reactor vessel, Charpy V-notch tests are run on all ferritic materials used in fabricating pressure parts of the steam generator and pressurizer to assure hydrotesting and operation in the ductile region at all times.

4.1.3 INSPECTION

The degree of conservatism considered in the quality assurance of materials and fabrication procedures is indicated in the attached table which delineates all the inspection requirements that are imposed by Westinghouse on its equipment suppliers. In addition to the inspections shown are those the equipment supplier performs to confirm the adequacy of material he receives, and those performed by the material manufacturer in producing the basic material. The inspections of reactor vessel, pressurizer, and steam generator are governed by ASME Code requirements. The inspection procedures and acceptance standards required on pipe materials and piping fabrication are governed by ASA B31.1 and Westinghouse requirements and are equivalent to those performed on ASME coded vessels. Table 4-1 at the end of this section summarizes the quality assurance program with regard to inspections performed on primary system components.

Procedures for performing the examinations are consistent with those established in Section III of the ASME B&PV Code and are reviewed by qualified Westinghouse engineers. These procedures have been developed to provide the highest assurance of quality material and fabrication. They consider not only the size of the flaws, but equally as important, how the material is fabricated, orientation and type of possible flaws, and the areas of most severe service conditions. In addition, the surfaces, most subject to damage as a result of the heat treating, rolling, forging, forming and fabricating processes, receive a 100% surface inspection by Magnetic Particle or Liquid Penetrant Testing after all these operations are completed. Although flaws in plates are inherently laminations in the center, all reactor coolant plate material is subject to shear as well as longitudinal ultrasonic testing to give maximum assurance of quality. (All forgings receive the same inspection.) In addition, 100% of the material volume is covered in these tests as an added assurance over the grid basis required in the code.

Westinghouse Quality Control engineers monitor the supplier's work, witnessing key inspections not only in the supplier's shop but at sub-vendors on the major forgings and plate material. Their normal surveillance includes verification of records of material, physical and chemical properties, review of radiographs, performance of required tests and qualification of supplier personnel.

As discussed in the Second Supplement to the Preliminary Safety Analysis Report in the answer to Question 1, an independent surveillance of the conformance to the fabrication and installation specifications and the quality control requirements of, among other things, the primary system components will be carried out by the United States Testing Company for Con Edison.

4.1.4 FABRICATION

The equipment specifications require that suppliers submit the manufacturing procedures (welding, heat treating, etc.) to Westinghouse where they are reviewed by qualified Westinghouse engineers. This also is done on the field fabrication procedures to assure that installation welds are of equal quality.

Section III of the ASME B&PV Code requires that nozzles carrying significant external loads shall be attached to the shell by full penetration weld. This requirement has been carried out in the reactor coolant piping, where all auxiliary pipe connections to the reactor coolant loop are made using full penetration welds.

Preheat requirements, non-mandatory under Code rules, are performed on all weldments, including P1 and P3 materials which are the materials of construction in the reactor vessel, pressurizer and steam generators. Preheat and post-heat of weldments both serve a common purpose -- the production of tough, ductile metallurgical structures in the completed weldment. Preheating produces tough ductile welds by minimizing the formation of hard zones whereas post-heating achieves this by tempering any hard zones which may have formed due to rapid cooling. Reactor coolant system components utilize both preheat and post-heat.

TABLE 4-1

Reactor Coolant System
Quality Assurance Program

Component	RT	UT	PT	MT	ET
1. Steam Generator					
1.1 Tube Sheet					
1.1.1 Forging		yes		yes	
1.1.2 Cladding		yes	yes		
1.2 Channel Head					
1.2.1 Casting	yes			yes	
1.2.2 Cladding			yes		
1.3 Secondary Shell & Head					
1.3.1 Plates		yes		yes	
1.4 Tubes		yes			yes
1.5 Nozzles (forgings)		yes		yes	
1.6 Weldments					
1.6.1 Shell, longitudinal	yes			yes	
1.6.2 Shell, circumferential	yes			yes	
1.6.3 Cladding		yes			
1.6.4 Nozzle to shell	yes			yes	
1.6.5 Support brackets				yes	
1.6.6 Tube-to-tube sheet			yes		
1.6.7 Instrument connections				yes	
1.6.8 Temporary attachments after removal				yes	
1.6.9 After hydrotest (all welds)				yes	
2. Pressurizer					
2.1 Heads					
2.1.1 Casting	yes			yes	
2.1.2 Clad			yes		
2.2 Shell					
2.2.1 Plates		yes		yes	
2.2.2 Clad			yes		
2.3 Heaters					
2.3.1 Tubing		yes	yes		
2.3.2 Centering of element	yes				
2.4 Nozzles		yes	yes		
2.5 Weldments					
2.5.1 Shell, longitudinal	yes			yes	
2.5.2 Shell, circumferential	yes			yes	
2.5.3 Cladding			yes		
2.5.4 Nozzles	yes			yes	
2.5.5 Nozzle Safe Ends	yes			yes	
2.5.6 Instrument connections			yes		
2.5.7 Support skirt				yes	
2.5.8 Temporary attachments, after removal				yes	
2.5.9 All welds after hydrotest				yes	

RT - Radiographic
UT - Ultrasonic
PT - Dye Penetrant
MT - Magnetic Particle
ET - Eddy Current

TABLE 4-1 (Cont'd)

Component	RT	UT	PT	MT	ET
3. Piping					
3.1 Fittings (Castings)	yes		yes		
3.2 Fittings (Forgings)		yes	yes		
3.3 Pipe		yes	yes		
3.4 Weldments					
3.4.1 Longitudinal	yes		yes		
3.4.2 Circumferential	yes		yes		
3.4.3 Nozzle to run pipe	yes		yes		
3.4.4 Instrument connections			yes		
4. Pumps					
4.1 Casting	yes		yes		
4.2 Forgings		yes	yes		
4.3 Weldments					
4.3.1 Circumferential	yes		yes		
4.3.2 Instrument connections			yes		
5. Reactor Vessel					
5.1 Forgings					
5.1.1 Flanges		yes		yes	
5.1.2 Studs		yes		yes	
5.1.3 Head Adapters		yes		yes	
5.2 Plates		yes		yes	
5.3 Weldments					
5.1.1 Main Seam	yes			yes	
5.1.2 CRD Head Adapter Connection			yes		
5.1.3 Instrumentation Tube			yes		
5.1.4 Main Nozzles	yes			yes	
5.1.5 Cladding			yes		
5.1.6 Nozzle Safe Ends	yes		yes	yes	

4.2 IN-SERVICE INSPECTION

The answer to Question 1 in the Second Supplement to the Preliminary Safety Analysis Report discussed Con Edison's plans for the in-service inspection of the reactor vessel. Subsequent to this submittal, Con Edison agreed to inspect all of the reactor vessel closure studs at each refueling. With regard to the other primary system components, the layout of the equipment and support structures will be designed to permit access to the following areas for examination during a plant shutdown. Access implies ability to contact surface and to visually examine surfaces.

1) The reactor coolant piping and fittings external to the primary shield surrounding the reactor vessel will be available for external surface and volumetric examination.

2) The pressurizer will be available for external surface and volumetric examination. Internal surface examination is possible.

3) The steam generator shell will be completely available for external surface and volumetric examination. Surface examination of the inside surface of the steam drum is possible.

4) The steam generator channel head will be completely available for surface and volumetric examination. Access to the inside surface is possible.

5) The external surfaces of the pump casing are available for surface examination. With removal of the pump, the internal surfaces and volumetric examination are possible.

These areas will be subjected to periodic in-service inspection at frequencies which will be established prior to initial operation of the plant. At this stage of the design, Con Edison has not developed a program dealing with the frequency of inspection and the methods for such inspections. At the operating license stage, such a definitive program will be submitted.

4.3 IN-SERVICE PRIMARY SYSTEM LEAKAGE DETECTION

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by one or more of the following:

a) Leakage through the head to vessel closure joint will result in a flow to the leak-off provided between the double gaskets of the closure joint which will show up as a high temperature in this line.

b) Any leakage will cause an increase in the amount of make up water required to maintain a normal level in the pressurizer.

c) The most sensitive indication of reactor coolant system leakage is the containment air particulate monitoring system. Experience has shown that the particulate activity in the atmosphere responds very rapidly to increased leakage. A system will be provided to monitor particulate activity from the areas enclosing the reactor coolant system components so that any leakage from them will be easily detected.

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