UNITED STATES ATOMIC ENERGY COMMISSION

Regulatory File Cy.

IN THE MATTER OF:

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

(Indian Point Station, Unit No. 2)

Docket No. 50-247

Place - Springvale Inn, Croton-on-Hudson, N.Y.

Date - October 5, 1971 Pages 1475-1620

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UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

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Springvale Inn
Croton-on-Hudson, N. Y.

Tuesday, October 5, 1971

The above-entitled matter came on for hearing, pursuant to notice, at 9:00 a.m.

BEFORE:

SAMUEL W. JENSCH, Esq., Chairman,
Atomic Safety and Licensing Board.

DR. JOHN C. GEYER, Member.

MR. R. B. BRIGGS, Member.

APPEARANCES: On behalf of the Applicant:

LEONARD M. TROSTEN, Esq., LEX K. LARSON, Esq.,
1821 Jefferson Place, N.W., Washington, D.C.
20036

On behalf of the Regulatory Staff:
MYRON KARMAN, Esq., and KARL KNIEL, Esq.,
Office of General Counsel, U.S. Atomic Energy
Commission, Bethesda, Maryland.

On behalf of Intervenor, Citizens' Committee for the Protection of the Environment, and on behalf of the Environmental Defense Fund:

ANTHONY B. ROISMAN, Esq., 1910 N Street, N.W., Washington, D. C.

On behalf of Intervenor, Hudson River Fishermen's Association:

ANGUS MACBETH, Esq., 36 West 44th Street, New York, N. Y. 10036

I N D E X

CHAIRMAN JENSCH: Please come to order.

This proceeding is a further evidenciary session of the proceedings involving Consolidated Edison Company of New York, Inc., in reference to Indian Point Station Unit No. 2, Docket No. 50-247, of the Atomic Energy Commission proceedings. This hearing is convened in accordance with an order convening hearings which was issued on September 17, 1971, that order providing for two sessions of evidentiary hearings, one commencing today on October 5 and the second convening on November 1, 1971, both at the Springvale Inn, 500 Albany Post Roade, Croton-On-Hudson, New York.

This order convening these two sessions provided for two types of presentations. The first, that is the October 5th session, today's session, will be generally concerned with receiving further direct evidence from Consolidated Edison Company of New York, the Regulatory Staff, and if the Atomic Energy Commission and the New York State Atomic Energy Council desires to present its evidence at this time the Board will give consideration to that presentation.

There has been some correspondence since the issuance of the order convening the hearings from the New York State Atomic Energy Council indicating that they did not desire to bring their witnesses in today's session. It was suggested in reply that perhaps the New York State Atomic

Council could confer with counsel participating in the proceedings as to a manner of presentation that would obviate the presentation of their witnesses but would provide the record would be complete with their offering and have the record available for cross-examination on the November 1, 1971 session.

The November 1, 1971 session was indicated to be of a limited duration in order to complete all matters of cross-examination, and all matters as to which complete submittals have been made by direct evidence at that time.

The general public notice was given of this order convening these two sessions of evidentiary hearings, which included distribution to several of the news media in this area and also included publication in the Federal Register, which is the official federal publication media for orders from regulatory agencies, among other notices, and publication in the Federal Register occurred on September 24, 1971, as reflected by Volume 36 of the Federal Register at page 18973.

As a preliminary, perhaps, we could have a statement of appearances formally again for the record.

On behalf of the Applicant?

MR. TROSTEN: Yes, Chairman. My name is Leonard M. Trosten. I am appearing on behalf of the Applicant. My office address is 1821 Jefferson Place, N. W., Washington, D. C. Appearing with me here today at the counsel table is my associate Lex K. Larsen, whose office address is the same as mine.

CHAIRMAN JENSCH: Thank you, sir. Appearance on behalf of the Regulatory Staff, the Atomic Energy Commission.

MR. KARMAN: Mr. Chairman, my name is Myron Karman. I am counsel for the Regulatory Staff of the Atomic Energy Commission. My office is 7920 Norfolk Avenue, Bethesda, Maryland.

CHAIRMAN JENSCH: Thank you, sir.

Appearances for the Intervenors, Hudson River Fishermen's Association.

MR. MAC BETH: Mr. Chairman, my name is Angus MacBeth, and I am appearing for the Hudson River Fishermen's Association. My office address is 36 West Fourth Street, New York, New York.

CHAIRMAN Jensch; Thank you, sir.

New York State Atomic Energy Council.

MR. MARTIN: Mr. Chairman, my name is Bruce L. Martin. I am appearing on behalf of the New York State Atomic Energy Council. My address is 112 State Street, Albany, New York.

CHAIRMAN JENSCH: Thank you, sir.

Environmental Defense Fund of the Citizens for the

Protection of the Environment.

MR. ROISMAN: Mr. Chairman, my name is Anthony Z. Roisman. I am appearing on behalf of the Environmental Defense Fund of the Citizens for the Protection of the Environment. My office address is 1910 N Street, N. W., Washington, D. C.

CHAIRMAN JENSCH: Thank you, sir.

I believe that completes the statement of all appearances.

B1Wt1

CHAIRMAN JENSCH: As indicated in the order convening these hearings, this session of the hearing, the Board, in making itself available to receive further submittals of evidence, has the Applicant further evidence to present? If so, we will then proceed.

MR. TROSTEN: The Applicant wishes to have received in evidence today two documents prepared by the Applicant concerning the emergency core cooling system for Indian Point 2. The first of these documents has already been offered in evidence at the session on July 13 under the sponsorship of Mr. James Moore as reflected on transcript Page 898. Thos document is entitled, "Additional testimony of applicant concerning emergency core cooling system performance," dated July 13, 1971. Mr. Moore, who is sponsoring this document, is here today, and I would like to ask that the Board received this document in evidence today.

CHAIRMAN JENSCH: Were those documents identified on July 13th? If so, in what manner?

MR. TROSTEN: They were identified in the record on Page 898. The title of the document, as I said, Mr. Chairman, "Additional testimony of applicant concerning emergency core cooling system performance," dated July 13, 1971. This document was prepared by Mr. James S. Moore of the Westinghouse Electric Corporation. Mr. Moore swore to the accuracy of the contents at that time.

B1wt2

CHAIRMAN JENSCH: Is there any objection to that offer on behalf of the Regulatory Staff?

MR. KARMAN: No objection, Mr. Chairman.

CHAIRMAN JENSCH: Hudson River Fishermen's Association.

MR. MAC BETH: No objection.

CHAIRMAN JENSCH: New York State Atomic Energy Council?

MR. MARTIN: No objection.

CHAIRMAN JENSCH: Environmental Defense Fund.

MR. ROISMAN: No objection with the usual reservation, Mr. Chairman.

CHAIRMAN JENSCH: Very well. The additional testimony of the Applicant with reference to the emergency core cooling system, which applicant's counsel just referred to, is received in evidence.

My recollection is that that offer of evidence was in a previously prepared statement; is that correct?

MR. TROSTEN: That is correct.

CHAIRMAN JENSCH: Do you have copies sufficient for the report to physically incorporate within the transcript? If not now, can you supply it?

MR. TROSTEN: There were copies already supplied to the reporter.

CHAIRMAN JENSCH: They do have sufficient copies?

MR. TROSTEN: Yes.

Blwt3

CHAIRMAN JENSCH: The additional evidence of the Applicant as prepared by witness James S. Moore is received in evidence.

MR. TROSTEN: Yes.

The next item of evidence I would like to offer today, Mr. Chairman, is a document entitled, "Additional information on Emergency core cooling analysis." It is dated August 16, 1971. This document, I will mention, Mr. Chairman, was prepared in response to questions which were asked orally by the Division of Reactor Licensing. A copy of this document was sent to the Division of Reactor Licensing by letter dated August 16, 1971, and copies of the same document were furnished to the Board and the parties to this proceeding by letter from Applicant's counsel dated August 16, 1971. I would like to have Mr. Moore stand now, please.

J A M E S S. M O O R E, previously sworn, testified as follows:

BY MR. TROSTEN:

Q Mr. Moore, I show you now a document entitled, "Additional Information on Emergency Core Cooling Analysis." Was this document prepared by you, Mr. Moore, or under your supervision and direction?

A Yes.

Q Are the contents of this document true and correct to the best of your knowledge?

Blwt4

A Yes.

Q Do you desire to have this document introduced in evidence in this proceeding as if read?

A I do.

MR. TROSTEN: Mr. Chairman, I now offer in evidence the document entitled, "Additional Information on Emergency Core Cooling Analysis," and ask that this be received in evidence and incorporated in the transcript as if read.

CHAIRMAN JENSCH: Is there any objection on behalf of the Regulatory Staff?

MR. KARMAN: No objection.

CHAIRMAN JENSCH: Hudson River Fishermen's Association.

MR. MACBETH: No objection.

CHAIRMAN JENSCH: New York State Atomic Energy Council.

MR. MARTIN: No objection.

CHAIRMAN JENSCH: Citizen's Committee for the Protection of the Environment.

MR. ROISMAN: No objection.

CHAIRMAN JENSCH: Very well. The offer of the evidence to which Applicant's counsel has just referred, entitled, 'Additional Information on Emergency Core Cooling Analysis" is received into evidence.

Do you have copies sufficient for the reporter?

BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

In the Matter of)

Consolidated Edison Company of) Docket No. 50-247
New York, Inc.)
(Indian Point Station, Unit No. 2))

ADDITIONAL TESTIMONY OF

APPLICANT CONCERNING EMERGENCY

CORE COOLING SYSTEM PERFORMANCE

July 13, 1971

This testimony consists of a report on the emergency core cooling system performance of Indian Point Unit No. 2, with emphasis on the manner in which the AEC interim policy statement concerning emergency core cooling systems, published on June 29, 1971, are complied with. This report also contains the information requested by the Division of Reactor Licensing (DRL) in a letter of July 7, 1971. Appendix C provides references to the body of the report where responses to specific items are found and provides additional information requested by DRL not covered in the body of the report.

INDIAN POINT UNIT NO. 2

EMERGENCY CORE COOLING PERFORMANCE

This report covers the analytical techniques described in the topical report "Westinghouse PWR Core Behavior Following a Loss-of-Coolant Accident," WCAP-7422-L, January 1970 (Proprietary); the supplementary proprietary Westinghouse report "Emergency Core Cooling Performance," dated June 1, 1971; and includes the additional assumptions identified in Appendix A of the AEC Interim Policy Statement, "Criteria for Emergency Core Cooling Systems for Light Water Power Reactors," published in the Federal Register June 29, 1971, Vol. 36, p. 12247.

The acceptance criteria found in the Interim Policy Statement are addressed on page 2 of the attached report and results showing compliance with these limits are given on page 18. The additional assumptions identified in Appendix A are incorporated in the text describing the calculational procedure. These assumptions are repeated below and the pages noted where each is discussed.

1. The break discharge coefficient, (C_D), used with the Moody discharge flow model should be equal to 1.0 for all break sizes. Page 8
2. The decay heat curve described in the proposed ANS Standard, with a 20% allowance for uncertainty, should be used. The fraction of decay heat generated in the hot rod may be considered to be 95% of this value. Page 34

3. For large breaks in the range 0.6 to 1.0 times the total area of the double-ended break of the largest cold-leg pipe, two break models should be used. The first model should be the double-ended severance ("Guillotine"), which assumes that there is break flow from both ends of the broken pipe, but no communication between the broken ends. The second model should assume discharge from a single node ("split"). Page 18
4. The time after the break for the onset of departure nucleate boiling at the hot spot should be equal to 0.1 second. Page 24
5. For cold leg breaks, all of the water injected by the accumulators prior to end-of-blowdown shall be assumed to be lost. In this context the end-of-blowdown shall be specified as the time at which zero break flow is first computed. The containment back pressure assumed for the blowdown analysis should not be higher than the initial pre-break pressure plus 90% of the increase in pressure calculated for the accident under consideration. Page 9, 10
6. The pump resistance, K, used for analysis should be fully justified. The effect of pump speed upon K should be considered. The more conservative of two assumptions (locked or running) should be used for the pump during the blowdown calculation. Page 18
7. A calculation for the reflooding heat transfer should be performed. The containment back pressure assumed for the analysis should not be higher than the initial pre-break pressure plus 80% of the increase in pressure calculated for the accident under consideration. Page 13

The following items should be constraints on the calculation:

a. No steam flow should be permitted in intact loops during the time period that accumulators are injecting. Page 11

b. Core exit quality should be calculated from entering mass flow rate and nominal FLECHT heat transfer. Page 13

c. Pump resistance should be calculated on the basis of a locked rotor. Page 12

d. The effects of the nitrogen gas in the accumulator, which is discharged following accumulator water discharge, should be taken into account in calculating steam flow as a function of time. Page 14

e. The pressure drop in the steam generator should be calculated with the existing fluid conditions and associated loss coefficients. Page 1

f. All effects of cold injection water, in either a hot or cold leg, on steam flow (and ΔP) should be included in the calculation. Page 13

g. The heat transfer coefficient during reflood should be derived from FLECHT data. Page 30

General

A LOCA would result from a rupture of the Reactor Coolant System (RCS) or of any line connected to that system up to the first closed valve. The charging pumps have the capability to make up for leakages resulting from ruptures of a small cross section, thus permitting an orderly shutdown. A small quantity of the coolant containing fission products present in the coolant would be released to the containment.

For a postulated large break, reactor trip is initiated when the pressurizer low pressure set point is reached while the Safety Injection System (SIS) signal is actuated by coincident pressurizer low pressure and low level. The reactor trip and SIS actuation are also initiated by a high containment pressure signal. The consequences of the accident are limited in two ways:

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

2. Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

Before the reactor trip occurs, the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. After reactor trip and turbine trip, core heat, heat from hot internals and the vessel is transferred to the RCS fluid, and then to the secondary system. The secondary system pressure increases until steam dump to the condenser occurs. Without offsite power,condenser cooling is lost and the power operated atmospheric relief valves and the safety valves open to the dump steam to the atmosphere. Make-up to the secondary side is automatically provided by the auxiliary feedwater pumps. The SIS signal stops normal feedwater flow by closing the main feedwater line isolation valves and initiates auxiliary feedwater flow by starting the motor driven auxiliary feedwater pumps. The secondary flow aids in the reduction of RCS pressure. When the RCS pressure falls below 600 psia, the accumulators begin to inject borated water.

Acceptance Criteria

The reactor is designed to withstand thermal effects caused by a LOCA including the double ended severance of the largest RCS pipe. The reactor core and internals together with the ECCS are designed so that the reactor can be safely shut down and the essential heat transfer geometry of the core preserved following the accident.

The ECCS, even when operating at partial effectiveness due to loss of one vital power bus (limiting case without off-site power), is designed to limit the cladding temperature to well below the melting temperature of Zircaloy-4 and below the temperature at which gross core geometry distortion, including clad fragmentation, can be expected. In addition, the core metal-water reaction is limited to less than 1% of the available Zircaloy.

In order to assure effective cooling of the core, limits on peak clad temperature and local metal-water reaction have been established. It has been demonstrated in the single rod burst test phase of the Westinghouse Rod Burst Program,(1) that for conditions within the area of safe operation, as shown in Figure 1, fuel rod integrity is maintained. Additional experimental data could further increase this area of safe operation. Results of the multi-rod burst test phase of the Westinghouse Rod Burst Program(2) show that peak clad temperature calculated during the accident increases less than 100°F due to geometry distortion; thus peak clad temperatures determined on the basis of no geometry distortion should be limited to 100°F below the limits presented in Figure 1. However, in order to increase the safety margin, the peak clad temperature calculated without geometry distortion will be limited to 2300°F.

Environmental consequences of the ruptures are well within the limits of 10CFR100 guidelines.

THERMAL ANALYSIS

Method of Analysis

This analysis is performed by considering the following aspects of the accident: blowdown hydraulics, reactor kinetics, and the core cooling.

a) Blowdown Hydraulics. This calculation provides a description of the hydraulic response to a rupture of the RCS through completion of depressurization and the operation of the safety injection systems. The basic information concerning the dynamic behavior of the reactor core environment is thus provided for use in reactor kinetics and core cooling analyses. The code used in this part of the analysis is SATAN-V.

b) Reactor Kinetics. The core power transient is forced by the blowdown dynamics which in turn affects the blowdown. The kinetic calculation determines the energy added to the core which forms an essential input to the core cooling analysis. The SATAN-V code includes reactor kinetics calculation.

c) Core Cooling. Based on the above information, a detailed analysis of reactor core cooling is performed. The code used in this part of the analysis is LOCTA-R2 for fuel pellet temperature, cladding temperature, and metal-water reaction evaluations.

THERMAL ANALYSIS

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b) Reactor Kinetics. The core power transient is forced by the blowdown dynamics which in turn affects the blowdown. The kinetic calculation determines the energy added to the core which forms an essential input to the core cooling analysis. The SATAN-V code includes reactor kinetics calculation.

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Blowdown Hydraulics (SATAN-V)

The blowdown hydraulic analysis for large size ruptures is required to provide the basic information concerning the dynamic behavior of the reactor core environment for the reactor kinetics and core cooling analysis. This requires the ability to predict the flow, quality, and pressure of the fluid passing through the core region. The SATAN-V code was developed with a capability to provide this information.

In a SATAN-V analysis the RCS is divided into a finite number of control volumes. The control volumes in the loop are divided into broken and unbroken loops and two control volumes are used to describe the coolant in the core region. This analysis replaces the FLASH-R analysis which had a limitation in that only three control volumes were available. The current multi-control volume analysis, particularly with two control volumes in the core, permits execution of a detailed parametric survey of the important phenomena affecting the blowdown process.

The adequacy of the SATAN code to predict the hydraulic behavior of the reactor coolant system during a loss-of-coolant accident has been verified by comparing the SATAN results of loss-of-coolant accidents of various sizes and locations performed at the LOFT and CSE facilities.[12]

In selecting the control volumes for the SATAN code, stability requirements dictate control volume selection of equal volume and length. In order to provide detail of the core region two core control volumes were selected.

This enables the flow at the top, middle, and bottom of the core to be calculated simultaneously. A total of 71 control volumes overall were selected for this analysis. Studies were performed to determine the effect of using more and less control volumes. The studies for less control volumes (15-20) showed that in addition to not having available the detail necessary to assess the flow regimes in the reactor vessel and core regions, significant changes in the answers resulted from small modeling changes. On the other hand a study was performed using 10 rather than 2 control volumes in the core resulted in only an insignificant change in the results. The current control volume selection is therefore felt to be adequate.

Later in this section a discussion is presented which shows the methods used to establish a conservative use of all the main assumptions used in the analysis. This study was performed to establish the conservative upper limit of the "confidence band" and forms the basis for this analysis.

Description of SATAN-V Code

The SATAN-V[4] (System Accident and Transient Analysis of Nuclear plants) computer code performs a comprehensive space-time dependent analysis of a LOCA and is designed to treat all phases of the accident. The three major phases are:

a) A subcooled phase where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the RCS internals and support structures.

b) A two-phase depressurization phase.

c) A refill-phase where the system pressure approaches the containment pressure.

The code employs a one-dimensional lumped parameter approach in which the entire reactor coolant loop system is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coast-down and cavitation, core and steam generator heat transfer including the W-3 DNB correlation in addition to the reactor kinetics are incorporated in the code.

The reactor kinetics considered in the code is a point kinetics model identical to the CHIC-KIN[9] model. The reactor kinetics equations are the point kinetics equations with appropriately weighted feedback effects. The major reactivity feedback mechanisms incorporated in the kinetics equations are the moderator density change, moderator temperature change, Doppler broadening as well as the control rod motion. These reactivity forcing functions are inputs in tabular form as functions of time. A block diagram of this continuous feedback model is shown in Figure 2.

Fundamental Equations

The fundamental equations of conservation of mass, energy, and momentum are applied for single and two phase system in order to determine the main dependent variables: density and internal energy. A set of auxiliary dependent variables is defined by the system governing algebraic equations. A critical flow calculation for subcooled, two phase, or superheated break flow is incorporated in the analyses.

The subcooled break flow is calculated using a modified version of the Zaloudek correlation.[5] The results are consistent with test results and do not produce a discontinuity between the subcooled and two-phase break flows. The two-phase break flow is calculated by using the Moody[6] two-phase correlation. Comparison of these correlations with experiments indicate that the break area must be reduced by a discharge coefficient to match the experimental break flow and depressurization rates. However, in all the design calculations, a break discharge coefficient of 1.0 is used.

Geometry

The SATAN-V Code is equipped to analyze a maximum of two loops plus "special" elements such as the reactor core, pressurizer and accumulators. This means that there is allowance for no more than two reactor coolant pumps and two steam generators. The unbroken loops are grouped together with proper scaling.

Method of Solution

The RCS is divided into a finite number of elements depending on the resolution desired and the computer is provided with the folliwng information:

a) Description of the way in which the elements are interconnected. Each element can have as many as two flows out and up to five flows in, plus a break flow out.

b) Initial thermodynamic data for each element: enthalpy, pressure, mass flow rates and heat transfer characteristics.

c) Physical data for each element: dimensions and flow area, heat transfer area, length, volumes and friction characteristics.

Starting from a set of initial conditions the program evaluates the updated values of the dependent variables for all system elements using an integration routine based on the Taylor series. The computed updated values are then substituted as the initial conditions for the next integration cycle and the procedure repeated until the final problem time is reached.

Description of the Core Reflooding Model

The SATAN calculations are performed until the completion of blowdown. In this context the end of blowdown is defined as the time at which zero break flow is first computed. The containment break pressure assumed for the blowdown analyses is equal to the initial pre-break pressure plus 90% of the increase in pressure calculated for the accident analyzed. At this time, the normal blowdown transient calculations are terminated and the reflooding calculations are performed. The reflooding model consists of three reference volumes which represent the downcomer region, the lower plenum region, and the active core region. The core and the downcomer volumes both communicate with the lower plenum volume via non-resistive flow paths. An input containment back-pressure is assumed to act directly on the top of the downcomer volume, and any steam generated in the core region is vented to the containment via a flow path whose resistance simulates the flow path to the break. The model is shown in Figure 3.

At the end of blowdown, the lower plenum volume is assumed to contain all the water remaining in the downcomer region below the nozzle and in the lower plenum. The water injected by the accumulator prior to the end of blowdown is conservatively assumed to be lost. No credit is taken for water remaining in any other control volume. The lost accumulator water delays the reflooding, extends the period of core recovery and results in a longer adiabatic core heat-up. Accumulator flow and cold-leg safety injection flow are continuously calculated as they were during the blowdown transient and are added to the downcomer region. Until the water reaches the bottom of the core, the water levels in the downcomer and in the region below the core are equal. When the water level in the downcomer region reaches the bottom of the inlet nozzle, water from the injection flow is assumed to spill through the break into the containment sump. Provisions for heat transfer from vessel walls and reactor internals to injection water are also included in this model.

When the bottom of the core is reflooded by the accumulator water, steam is generated by the hot fuel rods, causing a pressure build-up in the core region. This retards the core reflooding process. The steam generated must be vented from the system through the break, and the flooding rate is limited by the resistance of the loop to the steam and water flow. There are two paths available for the steam and entrained water to flow to the break. The first path is directly to the break through the broken loop. The other path is through the intact loops, back into the inlet annulus, and finally to the break through the inlet nozzle in the broken leg. These flow paths are illustrated in Figure 4 and 5. The sketches in Figures 4 and 5 show the path the steam must follow for the cold leg break. (Note that figure 5 shows one intact loop and one broken loop for the purpose of illustration.) The pressure drops along these two paths are calculated with the existing fluid conditions and associated loss coefficients. The pressure drop across the pump is maximized by assuming that the rotor is locked. In addition, it is postulated that the accumulator water injected in the cold leg pipe completely fills the pipe, thus forming a plug that prevents venting of the steam generated in the core during reflooding. These assumptions tend to reduce the core flooding rate and the fuel rod heat transfer, thus resulting in increased peak clad temperature.

The amount of mass evaporized and entrained as a function of core flooding rate and time after reflooding is obtained from an analysis of the FLECHT(7)(8) results. These results indicate that several flow regimes are present in the rod bundle during reflooding. For the first few seconds of the reflooding transient, until the core floods to approximately 20 inches, most of the heat transferred from the rod to the coolant goes to increase the liquid enthalpy. During this period almost no steam generation takes place and the core flooding rate equals the vessel cold flooding rate. Following this initial period the steam velocity increases above the value required for entrainment and a dispersed flow regime begins. This flow pattern is characterized by a continuous vapor phase with dispersed droplets and by a fast increase in rod heat transfer coefficient.

It is during this phase of the reflooding transient that the flooding rate into the core is reduced by the resistance of the flow paths from the core to the break. The core flooding rate transient during this period is a function of the core and loop resistance, the fraction of coolant ~~evaporized~~ evaporized and entrained, and the difference in water level between the downcomer and the core. The fraction of coolant evaporized, entrained and leaving the core is not constant during the transient, but increases from zero at the beginning to 0.7 - 0.8 several seconds after initiation of reflooding depending on the core flooding rate. At the same time, due to the reduction in core flow, the water level in the downcomer region increases at a faster rate thus providing the water head required to discharge the increased exit core flow to the break.

In the reflooding calculations in addition to assuming a locked rotor pump and the cold leg pipe plugged during accumulator injection, the following assumptions are made. Each of these assumptions is conservative because they result in increased venting path pressure drop and therefore lower flooding rate.

1. The fraction of coolant evaporized and leaving the core is assumed to be equal to 0.8 for flooding rates up to 2 in/sec and increases to 0.85 for flooding rates higher than 4 in/sec.

2. No transient effects are considered in the transition between high to low flooding rates, but, even during this period of time, the core flooding rate is calculated by assuming that the fraction of core inlet flow that has to be vented to the break is equal to the equilibrium values specified in No. 1 above.

3. No water-steam separation is assumed to occur in the upper plenum and the quality of the mixture entering the loops is calculated from the core mass flow rate and nominal FLECHT heat transfer.

4. All the coolant is assumed to be superheated to 500°F in the steam generators. The superheating of the steam is conservatively assumed to take place instantaneously at the steam generator <u>inlet</u>.

5. The containment back pressure is equal to the initial pre-break pressure plus 80% of the calculated pressure increase for the accident.

6. For the steam velocities occurring in the cold leg pipe and mass flow rate of the safety injection system, the acceleration pressure loss of accelerating all the water to the steam velocity is balanced by a small reduction in the steam temperature. Therefore, the injection of SIS water in the cold leg does not increase the loop pressure losses during reflooding.

7. The beneficial effects of nitrogen discharging after the end of blowdown are not taken into account during the reflooding time period. Since the nitrogen discharges into the cold leg, the downcomer will be pressurized a very short time period before the hot leg. This will result in an additional amount of downcomer water entering the core that is not accounted for in the reflood analysis.

Core Cooling - LOCTA-R2 Code

The LOCTA-R2 transient digital computer program was developed for evaluation of fuel pellet and cladding temperature during a LOCA. It also determines the extent of the Zircaloy-steam reaction and magnitude of the resulting energy release in Zircaloy clad cores.

The transient heat conduction 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 radial nodes and the cladding one node with a specified resistance simulating the pellet to clad gap.

Internal heat generation can be specified as a function of time. The decay heat is based on the heat generated from:

a) fission products;

b) capture products; and

c) residual fissions.

It is assumed that the core has been irradiated for an infinite period of time. The method used to calculate the total decay heat is presented in Appendix B.

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 equation(13)(14) unless there is an insufficient supply of steam available, then a "steam limited" evaluation is made. In the design calculations the parabolic rate equation is assumed for the full transient even when the fuel rod is assumed adiabatic. 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.

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 to evaluate a temperature rise in the steam as it flows through the core. Each radial region is considered independently and a number of axial sections which may be analyzed is essentially unlimited.

The initial conditions for the fuel rod are specified as a function of power. The following core conditions are also introduced as a function of time:

a) mass flow rate through the core;

b) coolant quality;

c) pressure.

To assure the conservatism of the hot channel temperature calculations the following procedure is used for the hot spot temperature calculation:

1. The higher of the two SATAN core control volume quantities is used in hot spot temperature calculations.

2. The flow used in the hot channel temperature calculations is the core midplane SATAN flow reduced by 20%.

Blowdown and reflooding heat transfer processes in the core are evaluated by means of using correlations summarized in Appendix A which have been validated with several Westinghouse Research and Development Programs. From the end of blowdown until core reflood the core is assumed adiabatic even though steam cooling is expected because of the local depressurization caused by the accumulators and by circulation promoted by the core and other heat sources. In summary, information generated by LOCTA-R2 as a function of time includes:

a) fuel temperature;

b) clad temperature;

c) steam temperature;

d) amount of metal-water reaction, if any;

e) volume of core melt, if any;

f) total heat released to coolant.

A more detailed explanation of the models and correlations used in LOCTA-R2 is presented in WCAP-7437-L[10].

Results

The detailed description of the Reactor Coolant System that can be obtained in the multi-control volumes SATAN-V code has been used to analyze important phenomena affecting the blowdown process, such as:

1. Heat transfer from core to the coolant during blowdown
2. Reactor Coolant Pump Characteristics
3. Steam Generator Heat Transfer Characteristics
4. Loop Resistances and Break Location
5. Accumulator Performance

A parametric survey study was performed for the Indian Point Unit No. 2 Final Safety Analysis Report,[11] Supplements 12 and 13 with the purpose of determining the most conservative combination of the above assumptions as input to the SATAN-V code. Additional parametric studies performed for Turkey Point Unit No. 2 and Indian Point Unit No. 2 are presented in the 6/1/71 AEC submittal "Emergency Core Cooling Performance".[3] The results presented in this section are based on the conservative assumptions determined in that study. These conservative assumptions were, for the limiting break condition (Cold Leg Double-Ended Guillotine Break), that the pump was assumed to trip at the time of the break and continued to coast down until cavitation conditions were reached (at this time the pump was assumed to not lock but continued to develop a conservative head). The pump speed is continually calculated as a function of prevailing conditions and the pump characteristics.

The analysis of the LOCA was performed at 102% of the maximum calculated power of 2758 MWt and at a peak linear power of 17.4 kw/ft. This value of peak linear power includes a 5% allowance for nuclear uncertainties.

Table 1 presents the results of the loss of coolant accident analyses for a range of break sizes.

Table 2 presents the hot spot metal water reaction for each break.

Figures 6 through 9 present the results of the SATAN-V analysis in terms of core flow, quality, and pressure as a function of time after the accident. Figures 10 through 13 present clad temperature transients. Figures 14 through 17 present core pressure drop. Figures 18 through 21 present hot spot fluid temperature. Figures 22 through 25 present flow in the broken and intact hot-leg and cold-leg piping and figures 26 through 29 present flow in the upper plenum, lower plenum and flow out of the break.

The peak clad temperature occurred following the double-ended cold leg break and did not exceed a value of 2300°F. The hot spot metal water reaction is 7.5% which meets the design criteria. In addition, the total core metal water reaction is less than 1%.

Conclusions - Thermal Analysis

For breaks up to and including the double-ended severance of a reactor coolant pipe, the ECCS with partial effectiveness will limit the clad temperature to 2300°F and assure that the core will remain in place and substantially intact with its essential heat transfer geometry preserved. The ECCS design meets the core cooling criteria as previously stated.

TABLE 1

PEAK CLAD TEMPERATURES FOR
INDIAN POINT UNIT #2

Break	Peak Temperature
Double Ended	2300 °F
0.8 Double Ended	2280 °F
4.5 Ft^2	2160 °F
3.0 Ft^2	1715 °F

TABLE 2

INDIAN POINT UNIT #2

HOT SPOT CLAD METAL - WATER REACTION

BREAK	PERCENT ZIRC REACTED
Double Ended	7.5
0.8 Double Ended	7.2
4.5 Ft^2	5.1
3.0 Ft^2	0.0

REFERENCES

(1) "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Single Rod Tests, "WCAP-7379-L, Volume I (Proprietary) and Volume II (Non-proprietary), September, 1969.

(2) "Topical Report - Performance of Zircaloy Clad Fuel Rods During a Simulated Loss-of-Coolant Accident - Multi-Rod Tests, "WCAP-7495-L, Proprietary, Volume I - Test setup and Results; Volume II - Analyses of Results, July 1969.

(3) "Emergency Core Cooling Performance" (W Proprietary) 6/1/71 AEC submitted.

(4) F. M. Bordelon, "A Comprehensive Space-Time Analysis of Loss-of-Coolant (SATAN Digital Code), WCAP-7263 (W Proprietary), March 1969.

(5) F. R. Zoloudek "Steam Water Critical Flow from High Pressure System," Hanford Laboratories, HW-80535, January 1964.

(6) F. H. Moody, "Maximum Flow Rate of Single Component, Two-Phase Mixture," Paper NO. 64-HT-35, and ASME Publication.

(7) J. O. Cermak, et al., "Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Tests," WCAP-7435, January 1970.

(8) F. F. Cadek, et. al., "Full Length Emergency Cooling Heat Transfer (FLECHT) Group II Test Report," WCAP-7544, September, 1970.

(9) V. A. Redfield, "CHIC-KIN. . . A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor, WAPD-TM-479 (January 1965).

(10) W. A. Bezella, et. al., "LOCTA-R2 Loss of Coolant Transient Analysis," WCAP-7437-L, January 1970.

(11) Supplement 12 to the IPP No. 2 FSAR, (Docket No.'s 50-247 and 50-286).

(12) "Topical Report - Westinghouse PWR Core behavior Following a Loss of Coolant Accident," WCAP-7422-L. (W Proprietary), January 1970.

(13) James C. Hesson, et. al., "Laboratory Simulations of Cladding - Steam Reactions Following Loss of Coolant Accidents in Water-Cooled Power Reactors" ANL-7609

(14) L. Baker, Jr. and Just, J. C., "Studies of Metal Water Reactions At High Temperature," ANL-6548, 1962

APPENDIX A

HEAT TRANSFER COEFFICIENTS USED IN THE LOCTA-R2 CORE THERMAL ANALYSES

The heat transfer correlations used in the LOCA analyses are presented below in order of application during the accident.

Time of break until occurrence of DNB

The time of the break until occurrence of DNB is taken conservatively to be 0.1 sec. The heat transfer regime during this period of the accident is forced convection turbulent heat transfer or fully developed nucleate boiling. The correlation for nucleate boiling is that by Jens and Lottes.(1)

$$T_W - T_{sat} = \Delta T_{sat} = 1.9\ e^{-p/900}\ (q'')^{0.25}$$

In the nucleate boiling regime the wall temperature is a function of the heat flux and pressure, not coolant velocity. The Jens and Lottes correlation is independent of geometry, i.e., valid for tubes, plates or rod bundles. It is also used for both local and bulk boiling. The correlation has been compared to subcooled water data obtained from single heated tubes having internal diameters from 0.143" to 0.288", length from 3" to 24.6" and pressure ranging from 500 to 2000 psia. The Dittus-Boelter correlation is used for forced convection turbulent heat transfer.

From DNB until time of uncovering (steam cooling period).

In the large break design calculations DNB is assumed to occur at 0.1 sec. After DNB occurs, the mode of heat transfer is unstable with both nucleate boiling and film boiling existing from times of short duration. The heat transfer correlation used in the transition to film boiling period is described in WCAP-9005(2) and the 6/1/71 AEC submittal, "Emergency Core

Cooling Performance." The verification of the use of this correlation for post DNB heat transfer during a transient blowdown is also presented.

The initial operating conditions prior to blowdown are:

1.	Pressure	2250 psia
2.	Mass Velocity	1,000,000 to 2,500,000 lb/hr ft^2
3.	Inlet Temperature	480 to 560°F
4.	Heat Flux	635,000 to 1,100,000 Btu/hr ft^2

Blowdown conditions were:

1.	Initial blowdown rate	200 to 10,000 psi/sec
2.	Average flow decay rate	250,000 to 1,150,000 lb/hr ft^2 sec

The test section consisted of a 1/2 inch inside diameter circular tube 3 feet long. The 3 foot length is sufficient to establish fully developed flow at the exit of the test section (L/D = 72).

A total of 50 transient blowdown runs were performed. To determine the effect of flow decay 20 runs were performed during which the flow rate was maintained as close as possible to the initial value. Thirty runs were performed in which the flow to the test section was allowed to decay in addition to the depressurization. This latter condition more nearly resembles the predicted conditions in a PWR core during the large break LOCA. For this reason the majority of the runs were performed with flow decay.

The comparison of predicted heat transfer coefficient with the measured data is shown in Figure A-1 for all data points. It is readily apparent that the correlation is conservative with respect to the results of this test since the measured value is greater than predicted for 95% of the data. The degree of conservatism contained in the correlation increases with increasing values of the heat transfer coefficient.

During uncovering (steam cooling period)

This period of the loss-of-coolant accident considers either turbulent or laminar forced convection to steam combined with radiation from the fuel rod surface to the steam. Radiation between fuel rods is not considered.

a) For turbulent forced convection to steam a Dittus-Boelter type equation, modified by McEligot[3] [4] to account for the variation in fluid properties near the wall due to a large temperature gradient, is used.

$$N_u = 0.020\ (Re)_b^{0.8}\ (Pr)_b^{0.4}\ \left(\frac{Tw}{Tb}\right)^{-0.5}$$

The $\left(\frac{Tw}{Tb}\right)$ term is independent of geometry. The Dittus-Boelter type equation was developed from flow inside tubes with values of C=0.023. Weisman[5] has shown that C is higher for rod bundle data ($\sim$0.023). A lower value of C as shown in the above correlation is presently used in the loss-of-coolant analyses.

The McEligot correlation was compared to data with hydraulic equivalent diameter values of 0.125" and 0.25" and L/D greater than 150. Additional coolant conditions are described below.

Coolant	Reynolds Number	Maximum Wall Temperature, °F	Tw/Tb
Air	1450-15000	1520	2.17
Helium	7570-13400	1050	1.56
Nitorgen	18200-45000	1620	4.78

The Prandtl number of steam is similar to those obtained with the above coolants.

b) For laminar forced convection to steam, the heat transfer correlation used is based on theoretical calculations of laminar flow in tubes made by Hausen[6] and Kays[7].

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

$$h/h_{iso} = \left(\frac{Tw}{Tb}\right)^{-0.25}$$

These calculations indicate that the local Nusselt number is highest near the inlet and drops until it reaches a limit corresponding to fully-developed thermal conditions. For the case of constant wall temperature, the limiting Nusselt number is 3.66. For constant heat flux at the wall, the asymptote is 4.36.

Furthermore, these calculations indicate that the asymptotic values are reached for all practical purposes when $L/D/RePr > 0.05$. For the Reynolds numbers ranging from 100 to 1000 and a $Pr = 1.0$ for steam, the developing length is from 2.5" to 25".

The correlation was compared to data from laminar air flow in circular tubes where the L/D ranged from 42 to 80 and $Re < 3000$.

The $\left(\frac{T_w}{T_b}\right)^{-0.25}$ term is to account for variations in fluid properties near the wall due to large temperature gradients.

c) Radiation to steam is evaluated employing the emperical method of Hottel.[8]

$$h = 0.1713 \times \varepsilon \frac{\left[\left(\frac{T_w}{100}\right)^4 - \left(\frac{T_{H_2O}}{100}\right)^4\right]}{T_w - T_b}$$

where

$$\varepsilon = \frac{1}{\frac{1}{\varepsilon_W} + \frac{1}{\alpha_{H_2O}} - 1}$$

$$\alpha_{H_2O} = \varepsilon' H_2O \left(\frac{T_{H_2O}}{T_W}\right)^{0.45} C_{H_2O}$$

The present value of the correction factor, C, for ε_{H_2O} at higher pressure than 0 or 1 atmosphere is 2.0.

Verification of Correlations Used During Steam Cooling Period

The use of the above correlations during the steam cooling period was verified by the work performed at the University of Michigan under Westinghouse funding and direction. This was part of the Flashing Heat Transfer Program. The results of this phase of the program have been documented in WCAP-7396-L[9]. The primary objective of this test was to determine the behavior of radiation heat transfer to steam at elevated pressures (up to 5 atm.).

The heat transfer test facility consisted of an open heat transfer loop. Steam was delivered to the test section and discharged to the atmosphere through the necessary piping and control apparatus.

The test sections consisted of 1/2 inch ID pipes, with an active length of three feet. The walls of the test section were heated by electrical resistance heating. Test sections having both uniform and non-uniform axial heat generation were employed.

The range of variables were representative of that in a PWR when the core is uncovered and is as follows:

Mass Velocity	$G = 4 \times 10^3$ to 4×10^4 lb/hr-ft^2
Temperature	T_{in} = 300 to 100°F
Wall Temperature	400 to 1800°F
Pressure	P_{in} = 25 to 75 psia
Inlet Reynolds Number	1900 to 35000

The results of the low pressure heat transfer test yield the following conclusions:

1. The McEligot et. al. correlation realistically predicts the convective heat transfer coefficients in turbulent flow.

2. In turbulent flow the radiant heat transfer contribution to the total heat transfer coefficient is adequately predicted by Hottel's technique.

3. The total heat transfer coefficient to steam in turbulent flow may be calculated by adding the convective term determined by McEligot's correlation and radiative term determined by Hottel's technique. A comparison of predicted versus measured total turbulent heat transfer coefficient is shown in Figure A-2 and excellent agreement can be seen.

4. In laminar flow the total heat transfer coefficient is conservatively predicted by using the correlation of Hausen and Kays for the convective contribution and the method of Hottel for the radiant contribution.

5. The prediction of laminar heat transfer coefficient can be improved by evaluating the steam properties at film conditions instead of bulk conditions.

6. The effect of a non-uniform heat flux on the heat transfer coefficient is negligible for the conditions which exist during a LOCA.

Recovery phase of the accident

After entrainment has been initiated, heat transfer coefficients obtained from the FLECHT Program(10) are used. Detailed results of the Group I and Group II test series can be found in Reference 11 and 12, respectively. In summary, the current test results verified the ability of a bottom flooding SIS design to terminate the temperature increase during a LOCA. In particular, it has been shown that the effects of a variable flooding rate can be predicted, using constant flooding rate data and that complete blockage of as many as sixteen adjacent channels will not impair bottom flooding core cooling effectiveness.

NOMENCLATURE

SYMBOLS

D_e - equivalent diameter

G - mass velocity

L - length of heat source

N_u - Nusselt number

P - system pressure

Pr - Prandtl number

Re - Reynolds number

T - temperature

g - gravity constant

h - heat transfer coefficient

k - thermal conductivity

q" - heat flux

$\bar{\varepsilon}$ - effective emissivity

μ - dynamic viscosity

ρ - density

SUBSCRIPTS

b - quantities evaluated at bulk fluid temperature

iso - evaluation of the parameter when the temperature difference $(T_w - T_b)$ is small

sat - refers to saturated condition

v - saturated vapor

L - refers to liquid

w - wall

REFERENCES

1. W. H. Jens and P. A. Lottes, "Analyses of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water," USAEC Report ANL-4627, 1951.

2. R. F. Farman and J. O. Cermak, "Post DNB Heat Transfer During Blowdown," WCAP-9005, October 1968, Proprietary.

3. D. M. McEligot, P. M. Magee, and G. Leppert, "Effect of Large Temperature Gradients on Convective Heat Transfer: The Downstream Region," J. of Heat Transfer, Vol. 87, 1965, pp. 67-76.

4. D. M. McEligot, L. W. Ormand, and H. C. Perkins, "Internal Low Reynolds - Number Turbulent and Transitional Gas Flow with Heat Transfer," J. of Heat Transfer, Vol. 88, 1966, pp. 239-245.

5. J. Weisman, "Heat Transfer to Water Flowing Parallel to Tube Bundles," Nuclear Science and Engineering 6, 79, 1969.

6. H. Hausen, "Darstellung des Warmeuberganges in Rohren durch verall gemeinerte Potenzbezienhungen," VDI Zeit., No. 4, p. 91, 1943.

7. W. M. Kays, "Numerical Solutions for Laminar-Flow Heat Transfer in Circular Tubes," Trans. ASME, Vol. 77, 1955, pp. 1265-2374.

8. H. C. Hottel, "Radiation Heat Transmission," Ch. 4 of Heat Transmission by W. H. McAdams, McGraw-Hill, 1954.

9. R. M. Hunt (Editor) "Safety Related Research and Development for Westinghouse Pressurized Water Reactors - A Program Outline Fall 1969," WCAP-7396-L, November 1969, Proprietary.

References (Cont.)

10. R. M. Hunt (Editor), "Safety Related Research and Development for Westinghouse Pressurized Water Reactors - A Program Outline Spring 1969," WCAP-7304-L, April 1969, Proprietary.

11. J. O. Cermak, et.al., "Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Tests," WCAP-7435, January 1970.

12. F. F. Cadek, et.al., "Full Length Emergency Cooling Heat Transfer (FLECHT) Group II Test Report," WCAP-7544, September, 1970.

APPENDIX B

RESIDUAL DECAY HEAT

In the analysis of loss of coolant transients, the residual decay heat during the period immediately after shutdown is important in determining peak clad temperatures. The decay heat data used as input to these analyses is described in this section.

Residual heat in a subcritical core consists of (a) fission product decay energy, (b) decay of neutron capture products, and (c) residual fissions due to the effect of delayed neutrons. These constituents are discussed separately in the following paragraphs.

Fission Product Decay

For short times ($<10^3$ seconds) after shutdown, data on yields of short half-life isotopes is sparse. Very little experimental data is available for the γ-ray contributions and even less for the β-ray contribution. Several authors have compiled the available data into a conservative estimate of fission product decay energy for short times after shutdown, notably Shure[1], Dudziak[2], and Teage[3]. Of these three selections, Shure's curve is the highest, and it is based on the data of Stehn and Clancy[4] and Obenshain and Foderaro[5].

The fission product contribution to decay heat which has been assumed for the loss of coolant analyses in this application is the curve of Shure increased by 20% for conservatism. This curve with the 20% factor included is shown in Figure B-1. The curve of Shure coincides with the recommendation in the proposed ANS standard on fission product decay heat (unpublished). The proposed standard also recommends that 20% be added for conservative design.

Decay of U-238 Capture Products

Betas and gammas from the decay of U-239 (23.5 min. half-life) and Np-239 (2.35 day half-life) contribute significantly to the heat generation after shutdown. The cross-section for production of these isotopes and their decay schemes are relatively well known. For long irradiation times their contribution can be written as:

$$P_1/P_o = \frac{\gamma 1 + E_{\beta 1}}{200 \text{ Mev}} \; c(1+\alpha) \; e^{-\lambda_1 t} \text{ watts/watt}$$

$$P_2/P_o = \frac{E_{\gamma 2} + E_{\beta 2}}{200 \text{ Mev}} \; c(1+\alpha) \left[\frac{\lambda_2}{\lambda_1 - \lambda_2} \left(e^{-\lambda_2 t} - e^{-\lambda_1 t}\right) + e^{-\lambda_2 t}\right] \text{watts/watt}$$

where: P_1/P_o is the energy from U-239 decay

P_2/P_o is the energy from Np-239 decay

t is the time after shutdown (secs)

$c(1+\alpha)$ is the ratio of U-238 captures to total fissions = .6(1+.2)

λ_1 = the decay constant of U-239 = 4.91×10^{-4} secs^{-1}

λ_2 = the decay constant of Np-239 = 3.41×10^{-6} secs^{-1}

$E_{\gamma 1}$ = total γ-ray energy from U-239 decay = .074 MeV

$E_{\gamma 2}$ = total γ-ray energy from Np-239 decay = .30 MeV

$E_{\beta 1}$ = total β-ray energy from U-239 decay = 1/3 x 1.18 MeV

$E_{\beta 2}$ = total β-ray energy from Np-239 decay = 1/3 x .43 MeV

(Two-thirds of the potential β-energy is assumed to escape by the accompanying neutrinos.)

This expression with a margin of 10% is shown in Figure B-1. The 10% margin, compared to 20% for fission product decay, is justified by the availability of the basic data required for this analysis. The constants given above are in agreement with the coefficients of the proposed ANS standard. The decay of other isotopes, produced by neutron reactions other than fission, is neglected.

Residual Fissions

The time dependence of residual fission power after shutdown depends on core properties throughout the transient. Spatially dependent kinetics calculations have not been performed for these transients. It is assumed that core average conditions are more conservative for the calculation of reactivity and power level than actual local conditions as they would exist in hot areas of the core. Thus, static power shapes are assumed (with the exception of the effect discussed in the next sectio and these are factored by the time behavior of core average fission power calculated by a point model kinetics calculation with six delayed neutron groups.

$E_{\gamma 1}$ = total γ-ray energy from U-239 decay = .074 MeV

$E_{\gamma 2}$ = total γ-ray energy from Np-239 decay = .30 MeV

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$E_{\beta 2}$ = total β-ray energy from Np-239 decay =1/3 x .43 MeV

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For the purpose of illustration, only a one delayed neutron group calculation, with a constant reactivity of -4%Δρ, is shown in Figure B-1.

Distribution of Decay Heat

During a loss of coolant accident the core is rapidly shut down by void formation or control rods, or both, and a large fraction of the heat generation to be considered comes from fission product decay gamma rays. This heat is not distributed in the same manner as steady state fission power. Local peaking effects which are important for the neutron dependent part of the heat generation do not apply to the gamma ray contribution. The steady state factor of 97.4% which represents the fraction of heat generated within the fuel, drops to 95% for the hot rod in a loss of coolant accident.

For example, consider the transient resulting from the double ended break of the largest primary circuit pipe; 1/2 second after the rupture about 30% of the heat generated in the fuel rods is from gamma-ray absorption. The gamma power shape is less peaked than the steady state fission power shape, reducing the energy deposited in the hot rod at the expense of adjacent colder rods. A conservative estimate of this effect is a reduction of 10% of the gamma-ray contribution of 3% of the total. Since the water density is considerably reduced at this time, an average of 98% of the available heat is deposited in the fuel rods, the remaining 2% being absorbed by water, thimbles, sleeves and grids. The net effect is a factor of .95 rather than .974, to be applied to the heat production in the hot rod.

References

1. K. Shure, "Fission Prod-ct Decay Energy," Bettis Technical Review, WAPD-BT-24, December, 1961.

2. K. Shure and D. J. Dudziak, "Calculating Energy Release by Fission Products," Trans. Am. Nucl. Soc. 4 (1) 30,1961.

3. U.K.A.E.A. Decay Heat Standard (Reference unavailable).

4. J. R. Stehn and E. F. Clancy, "Fission-Product Radioactivity and Heat Generation","Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958,"Vol. 13, pp. 49-54, United Nations, Geneva, 1958.

5. F. E. Obenshain and A. H. Foderaro, "Energy from Fission Product Decay" WAPD-P-652, 1955.

PEAK CLAD TEMPERATURE AND METAL-WATER REACTION
FOR A LOSS OF COOLANT ACCIDENT

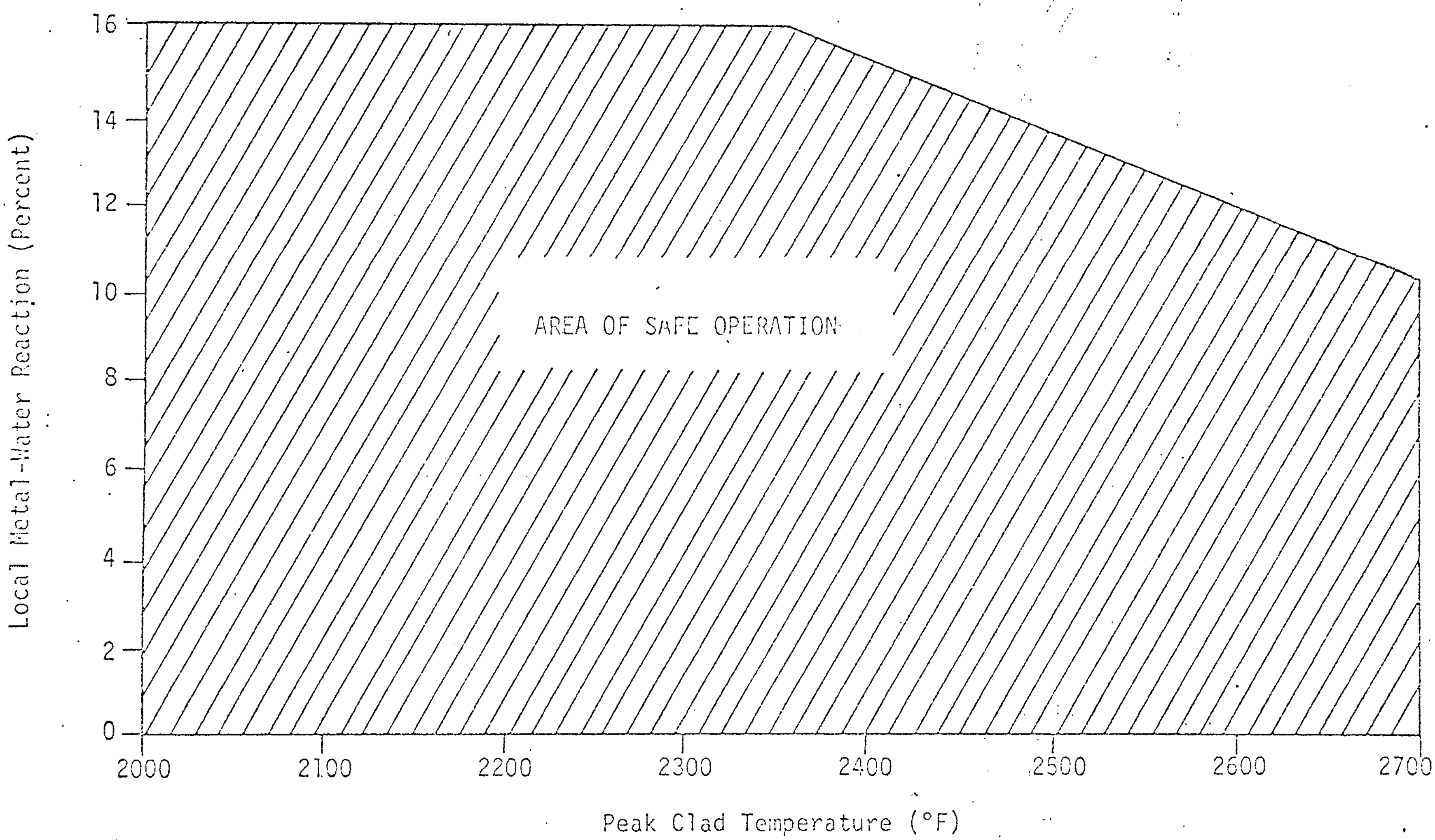


FIG. 1

A BLOCK DIAGRAM OF THE CONTINUOUS FEEDBACK MODEL CHIC-SIN CODE

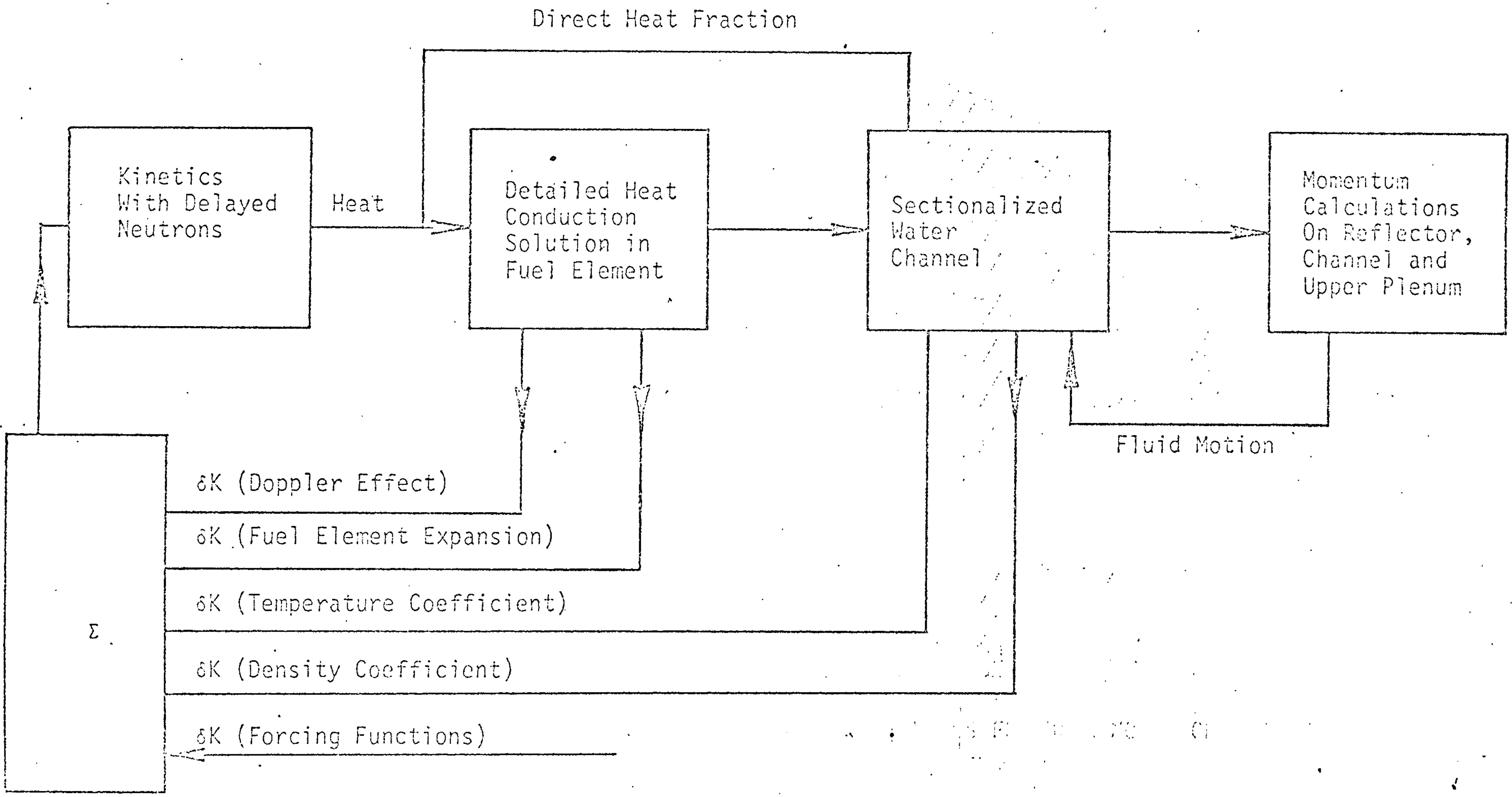
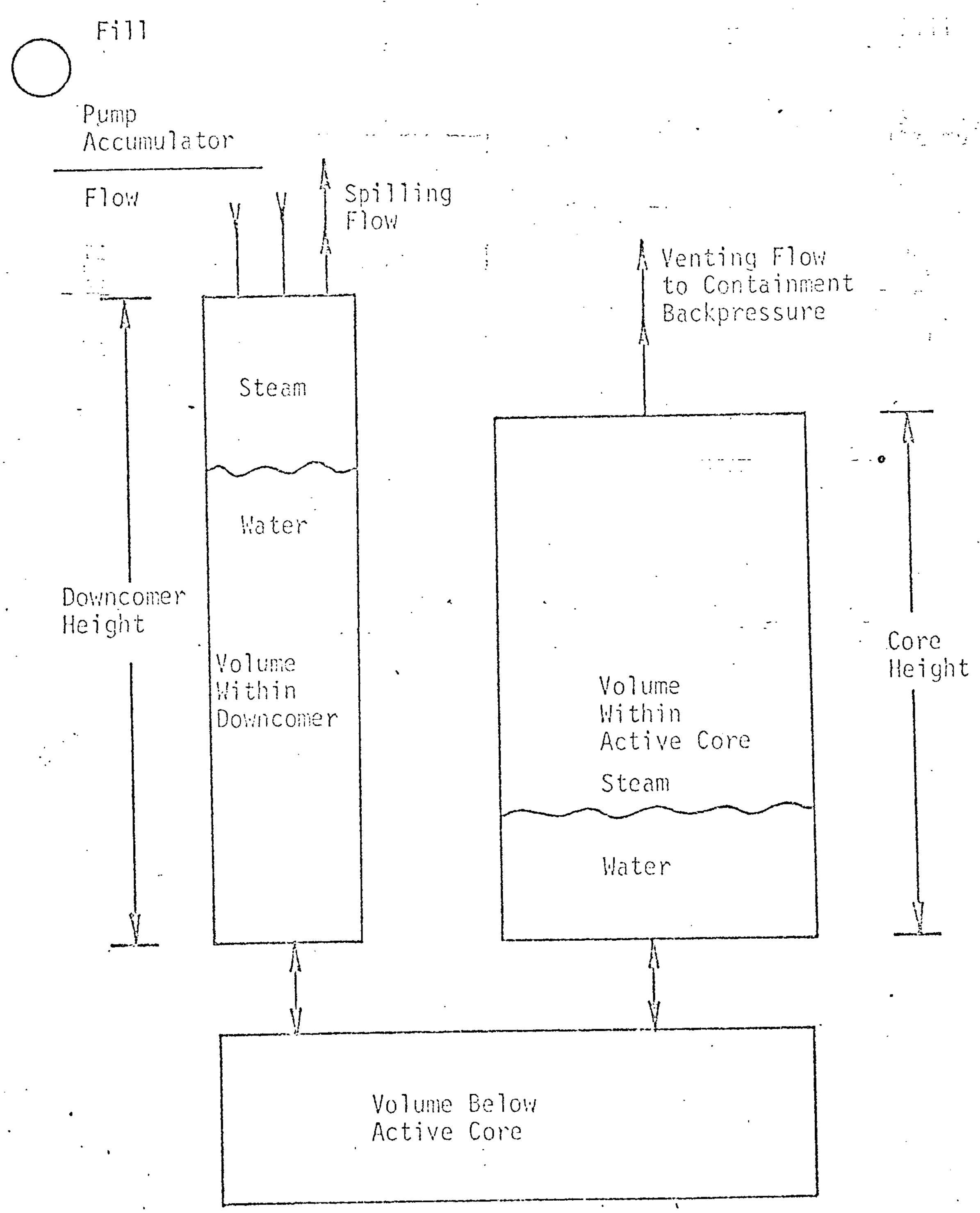
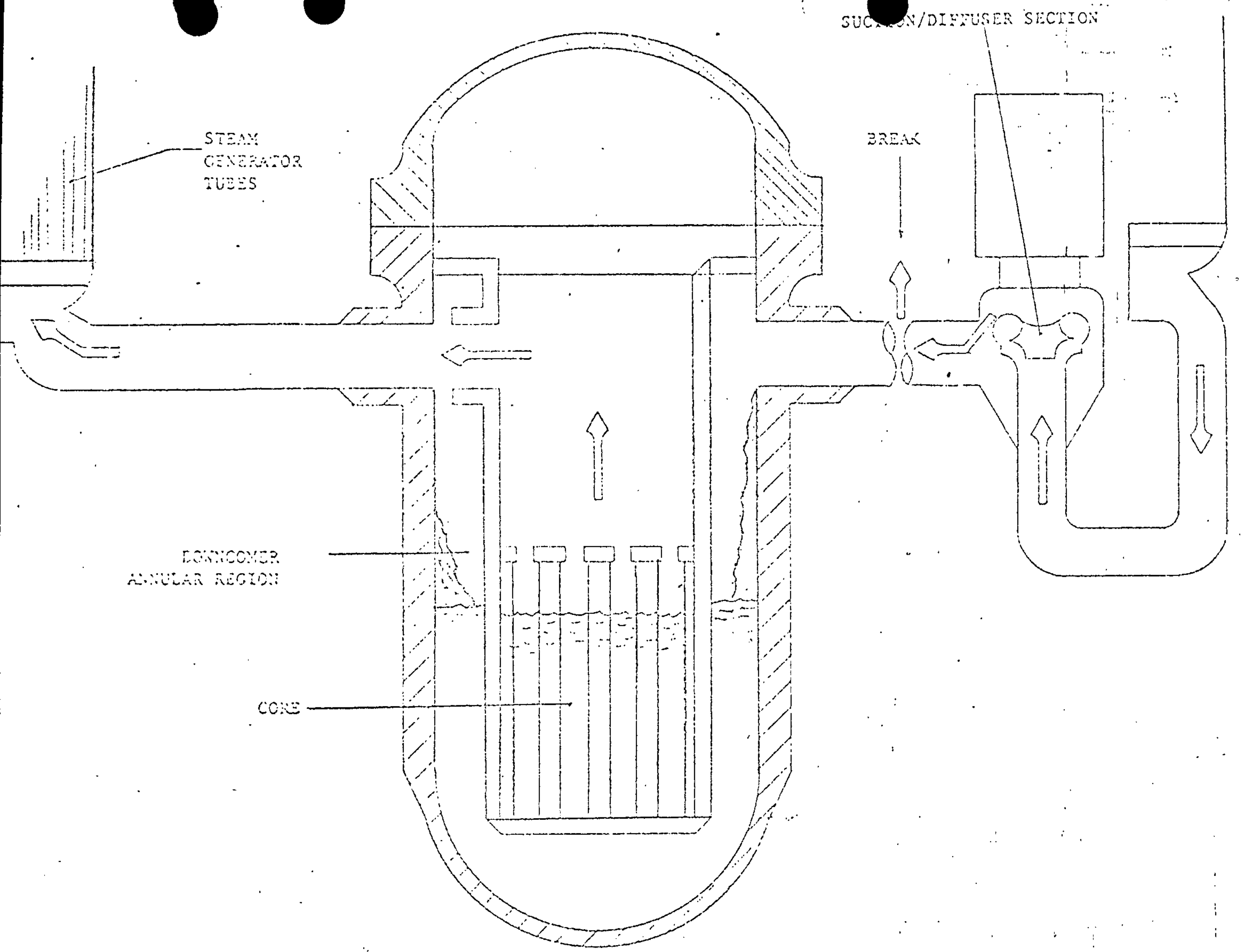


FIG. 2

FIG. 3

BLOCK DIAGRAM FOR SATAN-V REFILL CALCULATION





DIFFUSER SECTION
STEAM
GENERATOR
TUBES
BREAK
DOWNCOMER
ANNULAR REGION
CORE

FIGURE 4

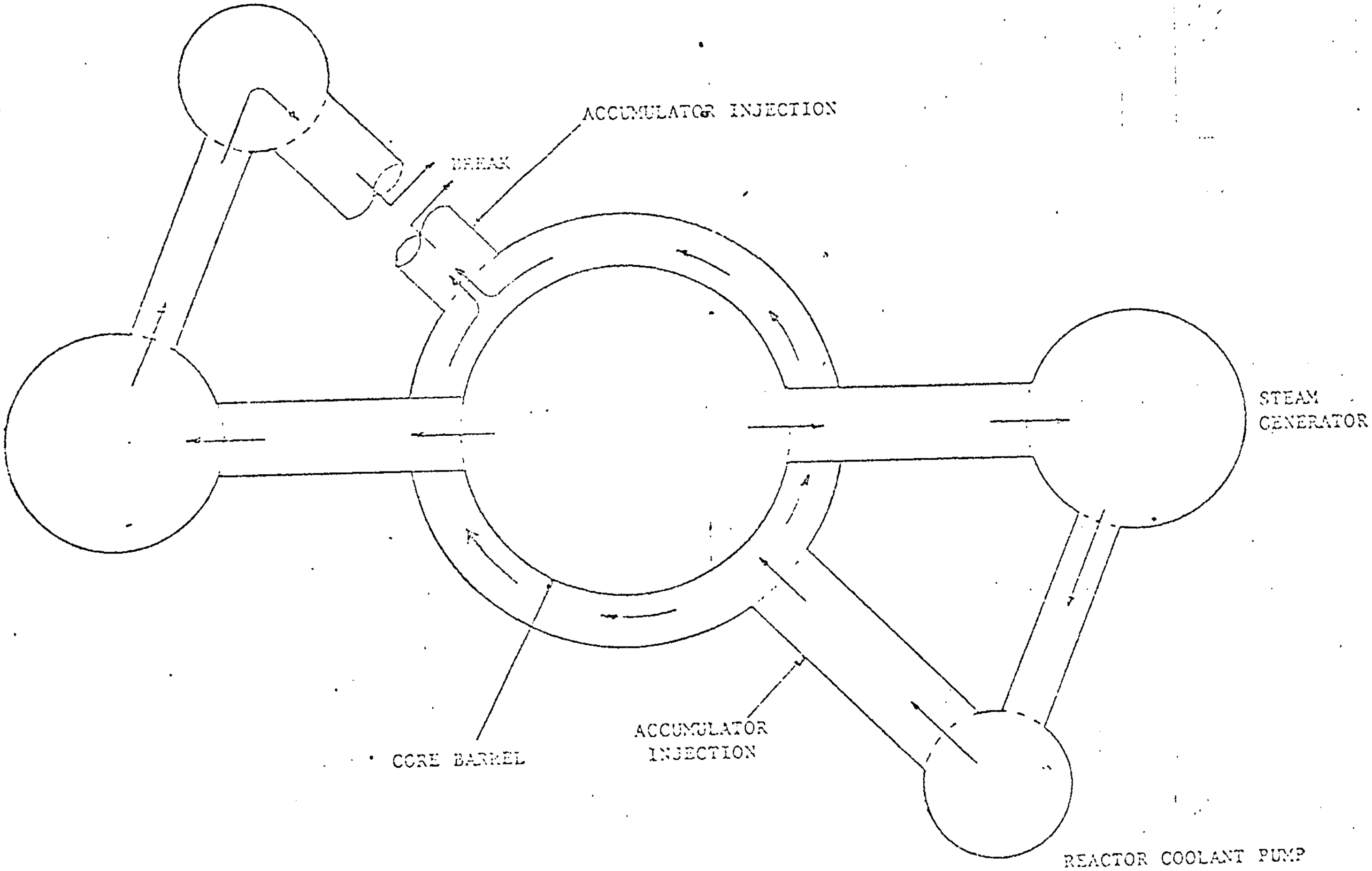

ACCUMULATOR INJECTION
BREAK
STEAM GENERATOR
CORE BARREL
ACCUMULATOR INJECTION
REACTOR COOLANT PUMP

FIGURE 5

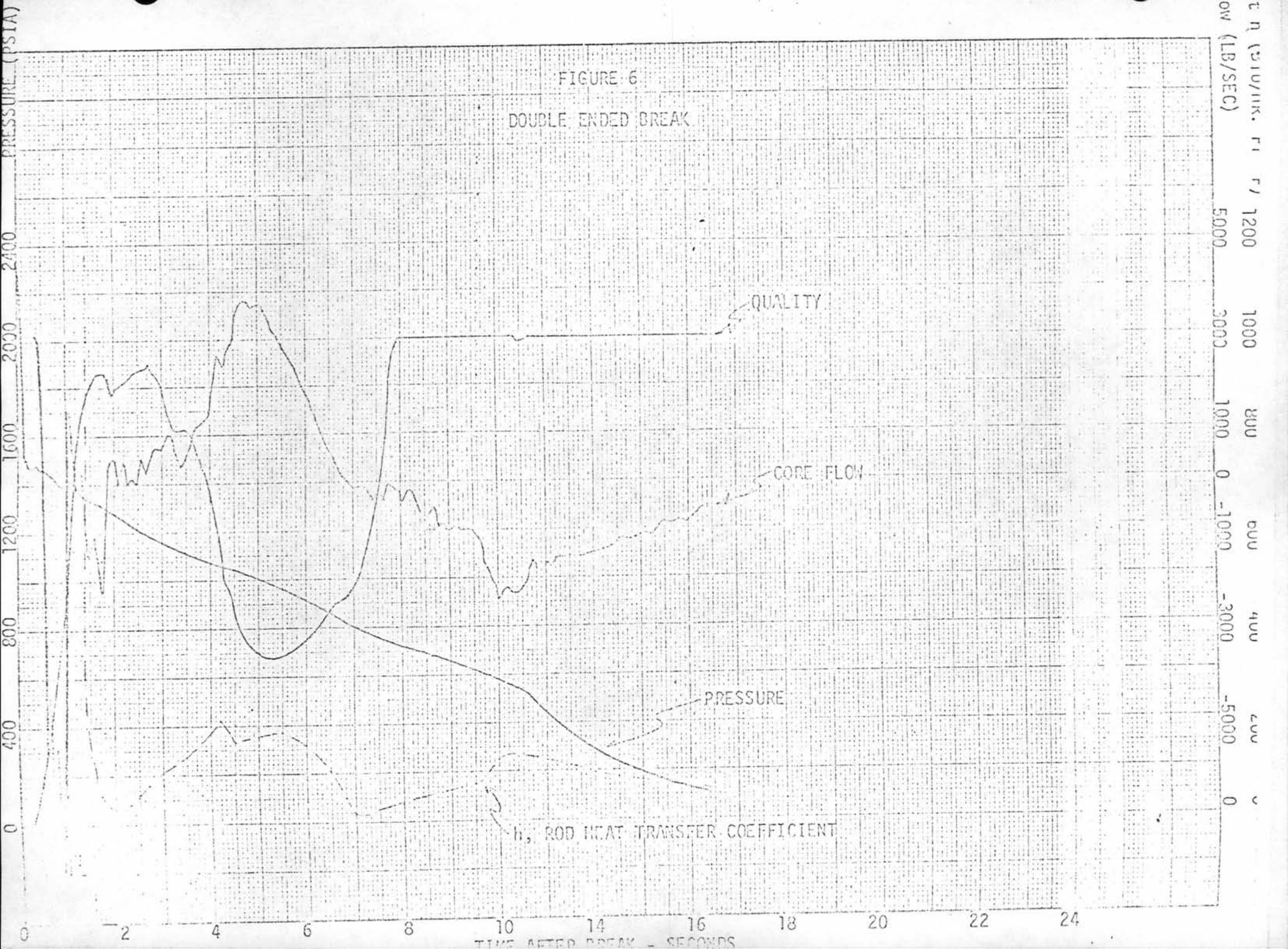


FIGURE 6
DOUBLE ENDED BREAK
PRESSURE (PSIA)
2400
2000
1600
1200
800
400
0
QUALITY
CORE FLOW
PRESSURE
h, ROD HEAT TRANSFER COEFFICIENT
5000
3000
1000
0
-1000
-3000
-5000
1200
1000
800
600
400
200
0
(LB/SEC)
0
2
4
6
8
10
14
16
18
20
22
24
TIME AFTER BREAK - SECONDS

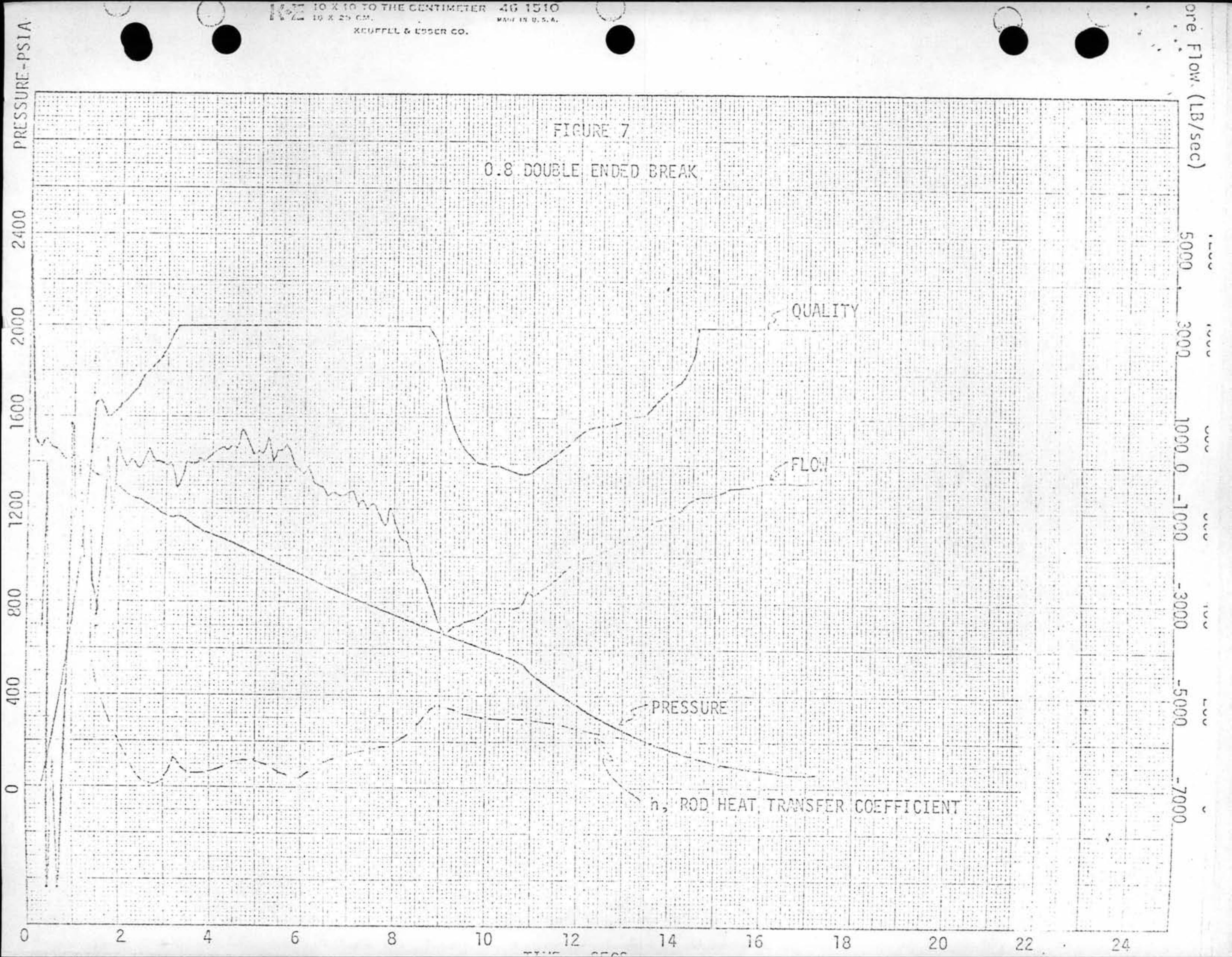


FIGURE 7
0.8 DOUBLE ENDED BREAK
PRESSURE-PSIA
Core Flow (LB/sec)
QUALITY
FLOW
PRESSURE
h, ROD HEAT TRANSFER COEFFICIENT
0
400
800
1200
1600
2000
2400
5000
3000
1000
0
-1000
-3000
-5000
-7000
2
4
6
8
10
12
14
16
18
20
22
24
TIME - SECS

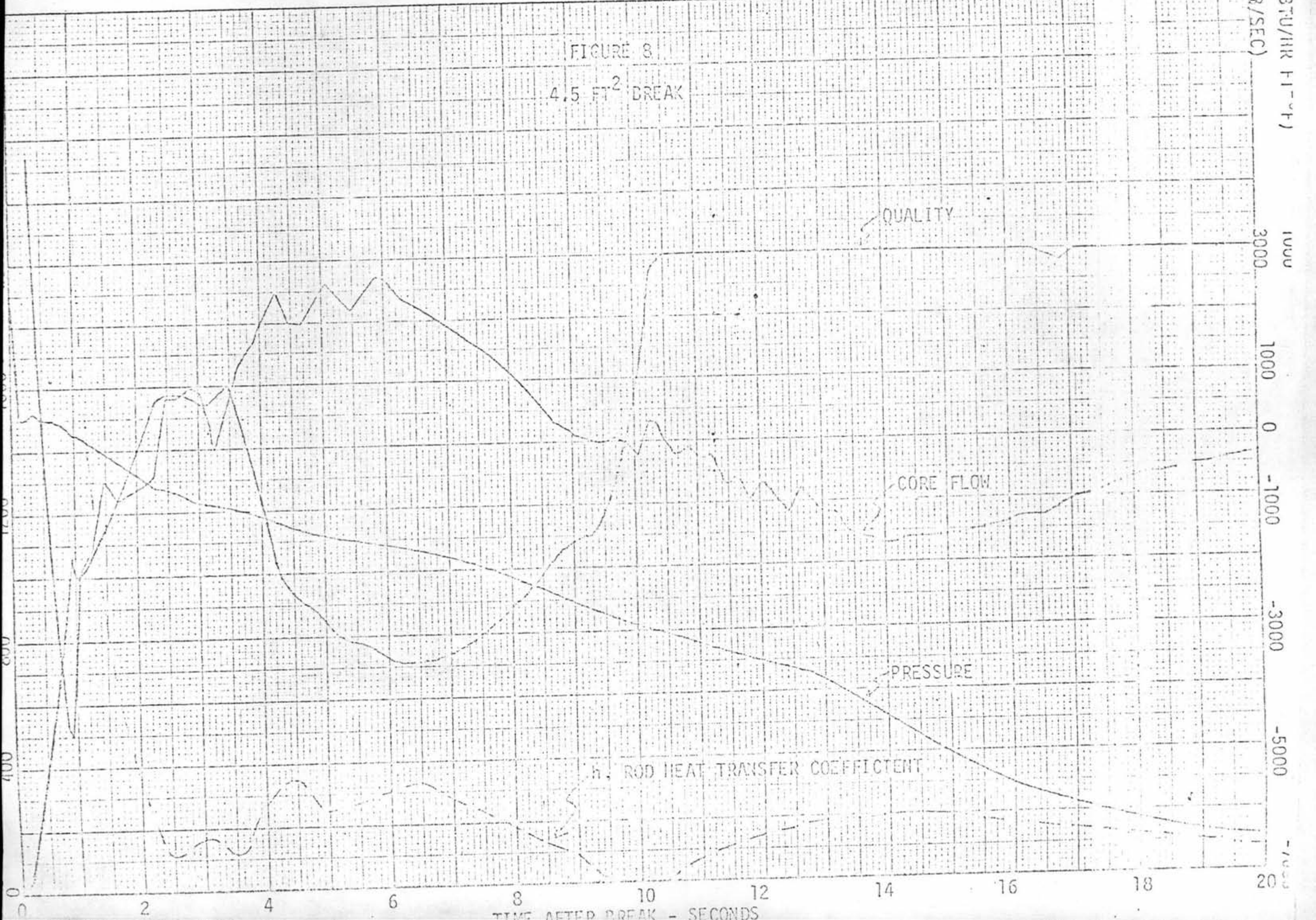


FIGURE 8
4.5 FT^2 BREAK
QUALITY
CORE FLOW
PRESSURE
h, ROD HEAT TRANSFER COEFFICIENT
Spot h (BTU/HR FT^2°F)
Flow (LB/SEC)
3000
1000
0
-1000
-3000
-5000
0
2
4
6
8
10
12
14
16
18
20
TIME AFTER BREAK - SECONDS

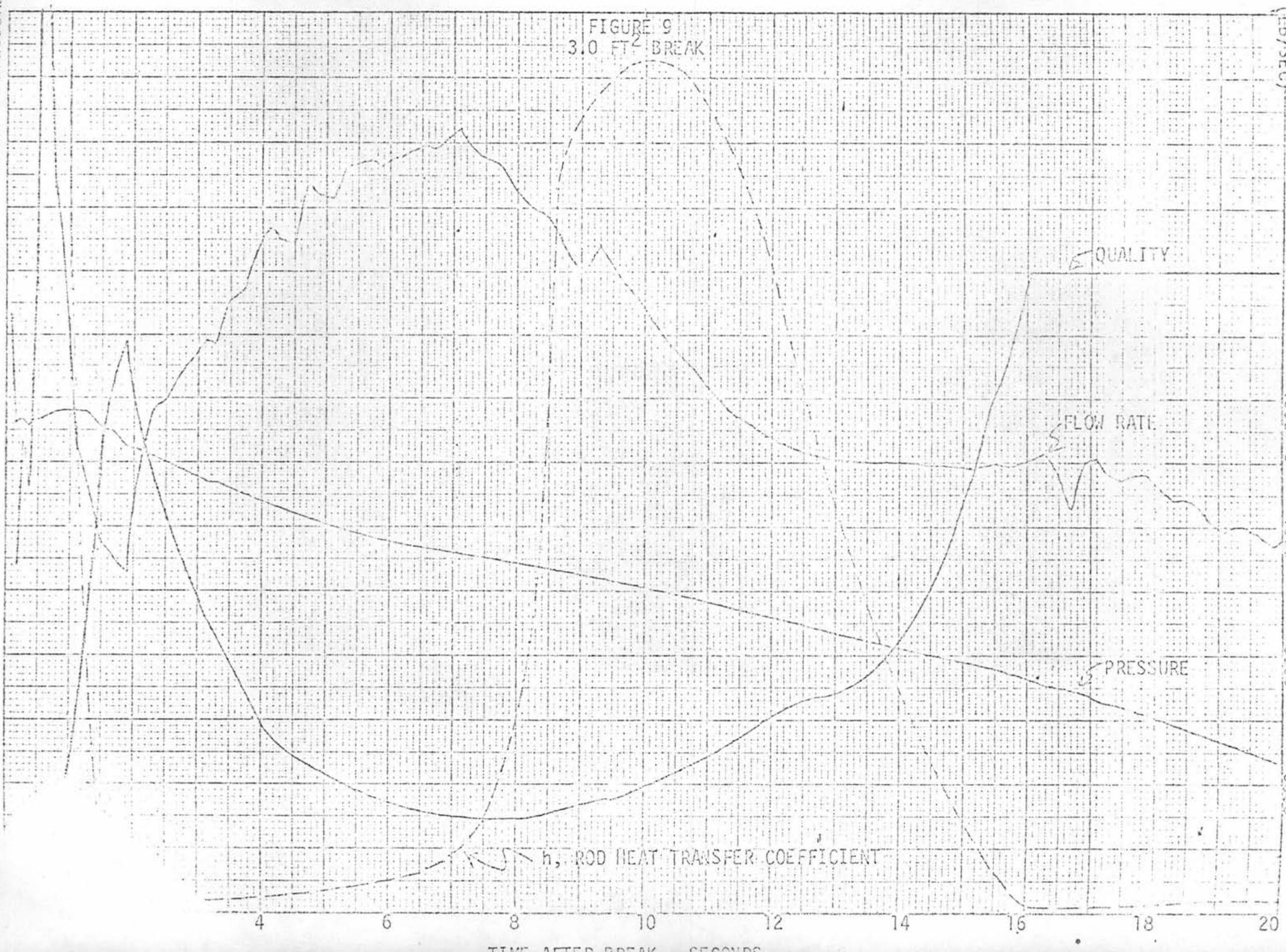


FIGURE 9
3.0 FT^2 BREAK
QUALITY
FLOW RATE
PRESSURE
h, ROD HEAT TRANSFER COEFFICIENT
5000
3000
1000
0
-1000
-3000
-5000
4
6
8
10
12
14
16
18
20

FIGURE 10

PEAK CLAD TEMPERATURE vs TIME AFTER BREAK

DOUBLE ENDED BREAK

FIGURE 11

PEAK CLAD TEMPERATURE vs TIME AFTER BREAK

.8 DOUBLE ENDED BREAK

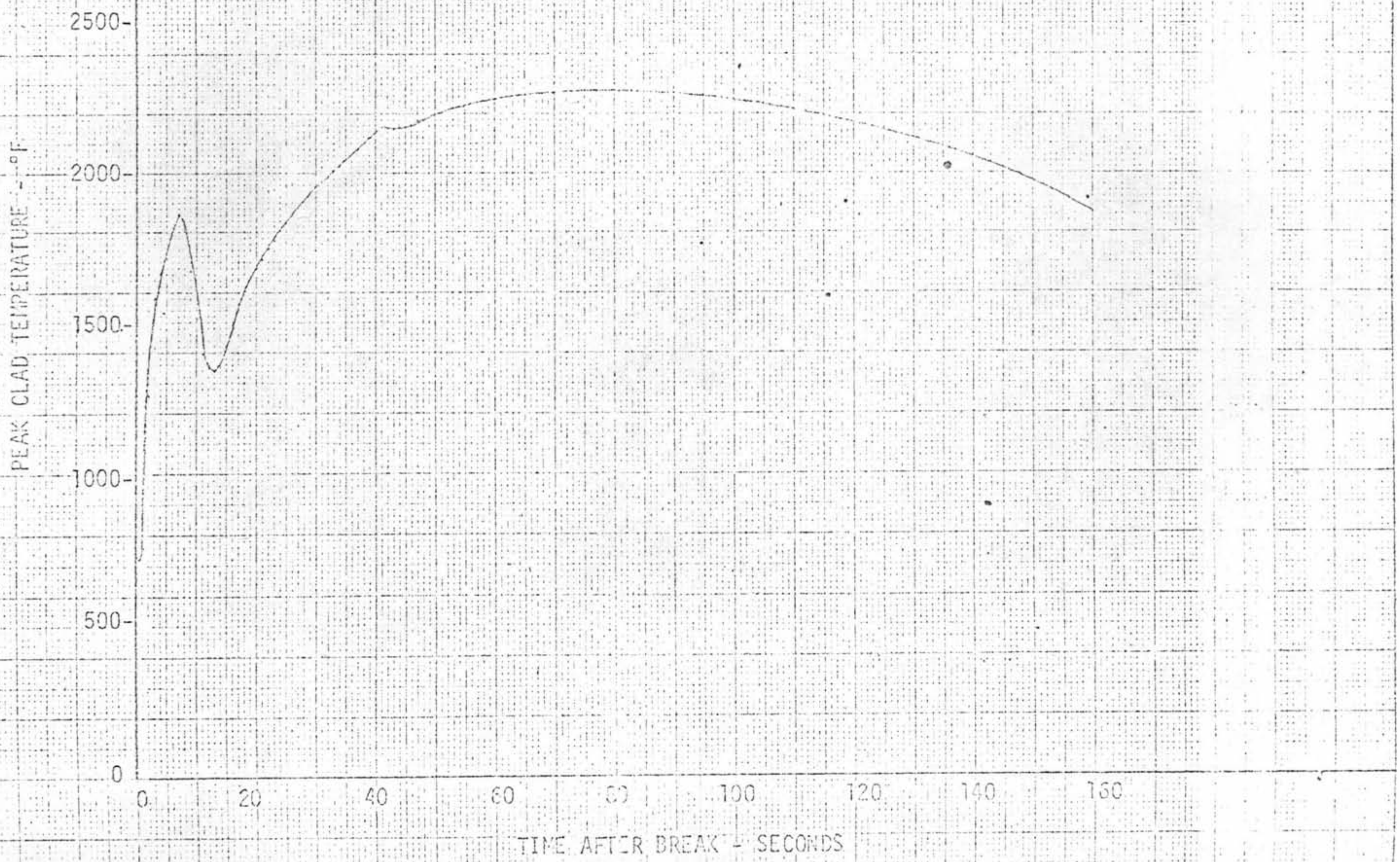


FIGURE 12

PEAK CLAD TEMPERATURE vs TIME AFTER BREAK

4.5 FT^2 BREAK

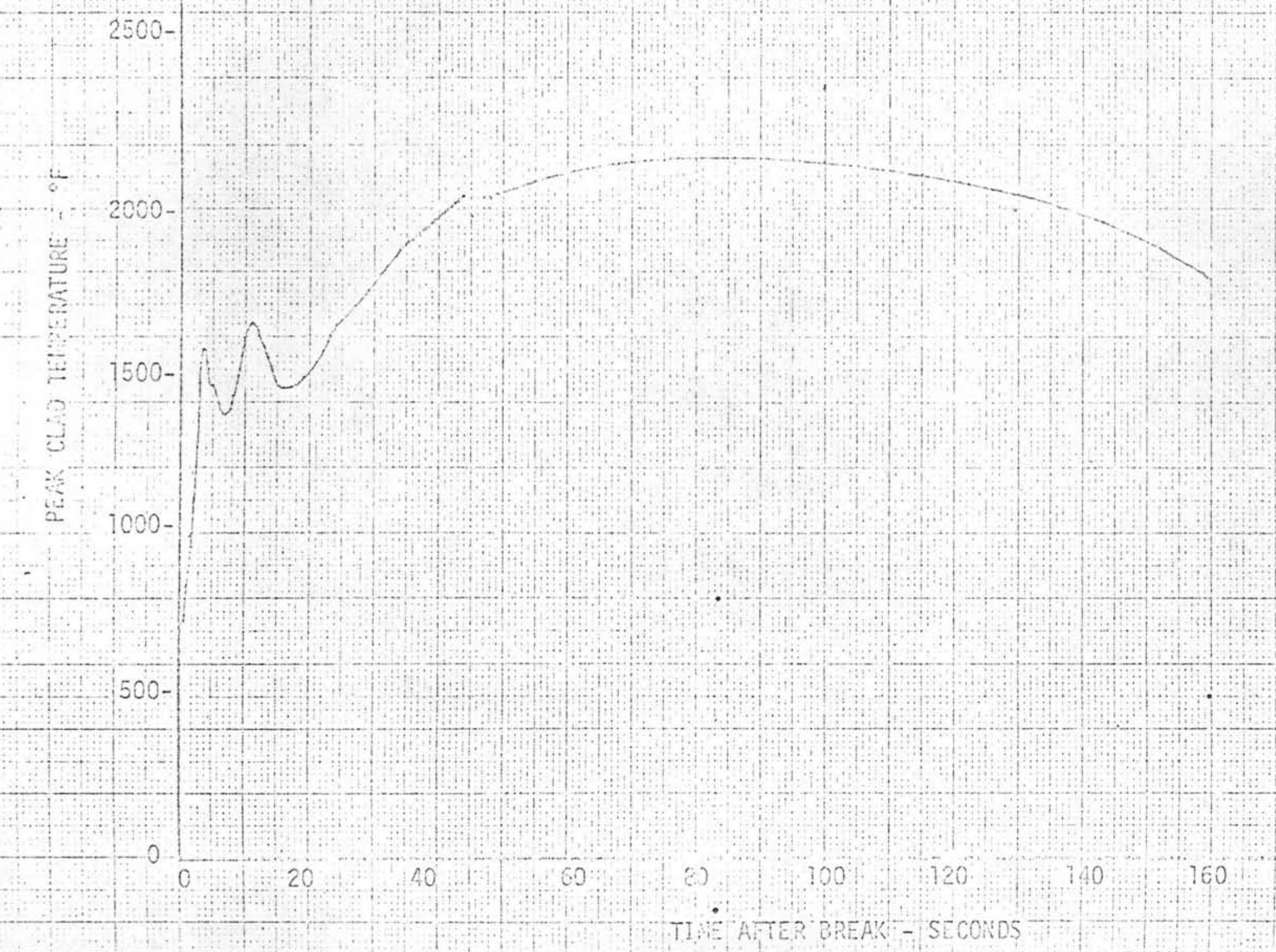
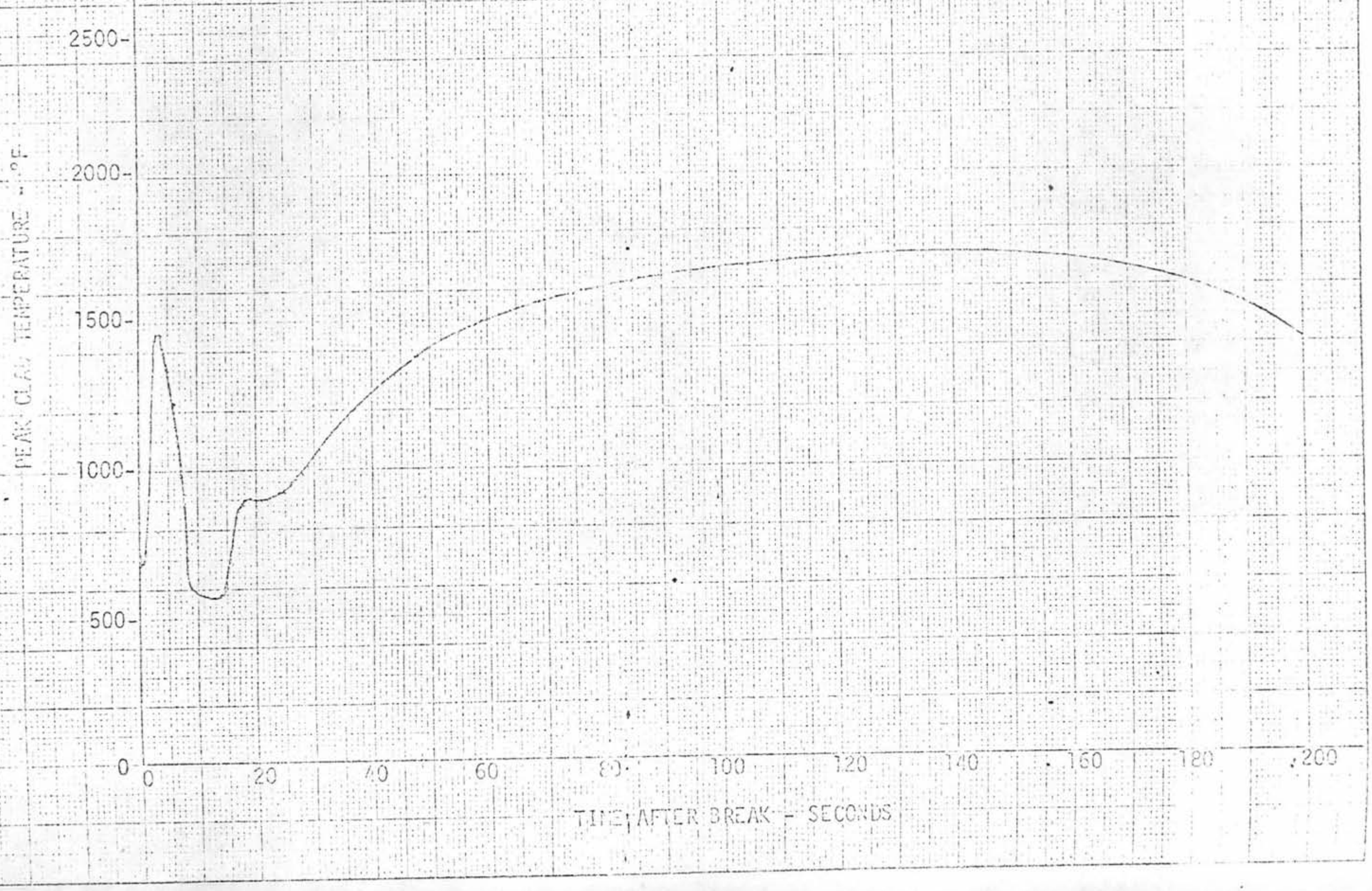


FIGURE 13

PEAK CLAD TEMPERATURE vs TIME AFTER BREAK

3 FT^2 BREAK



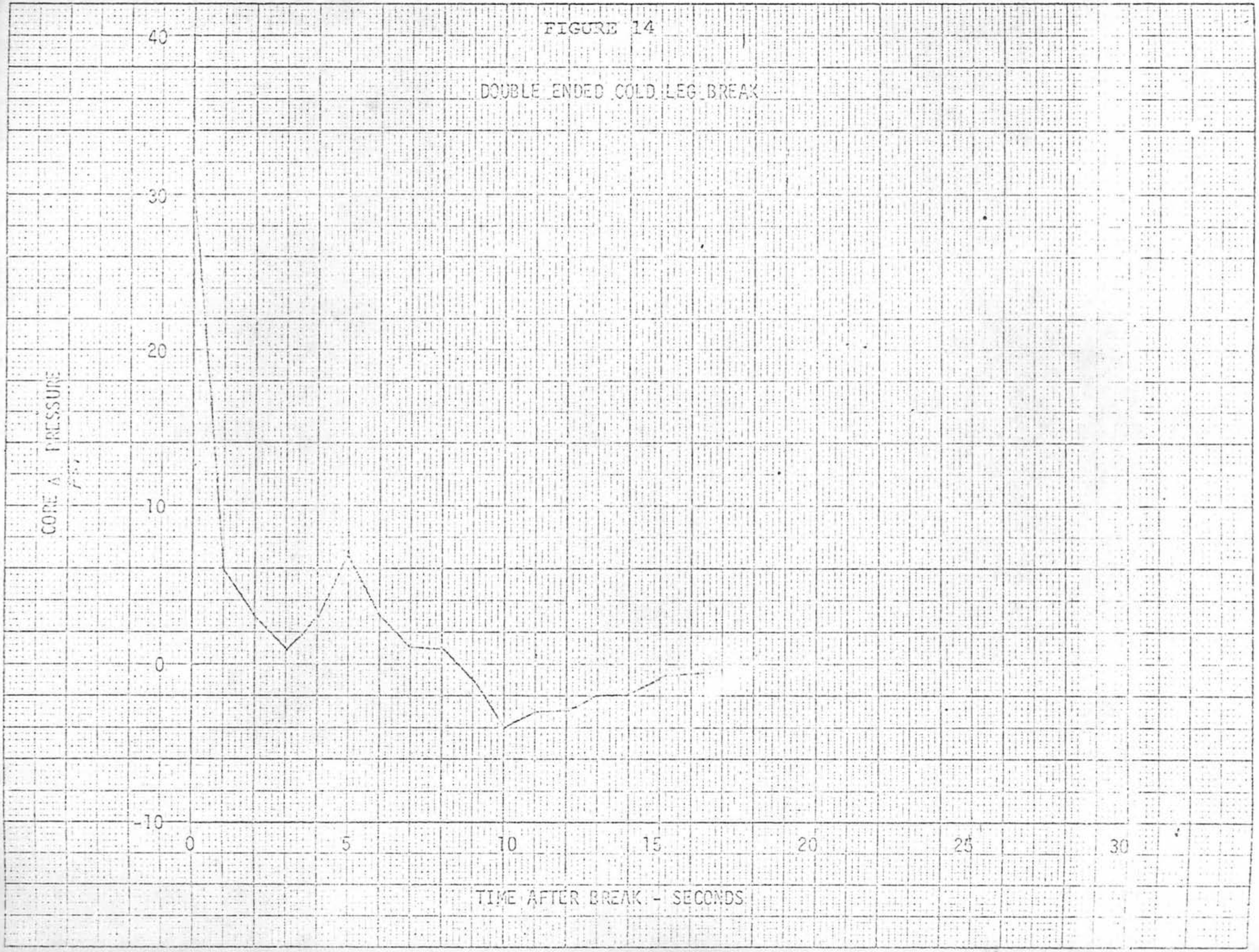
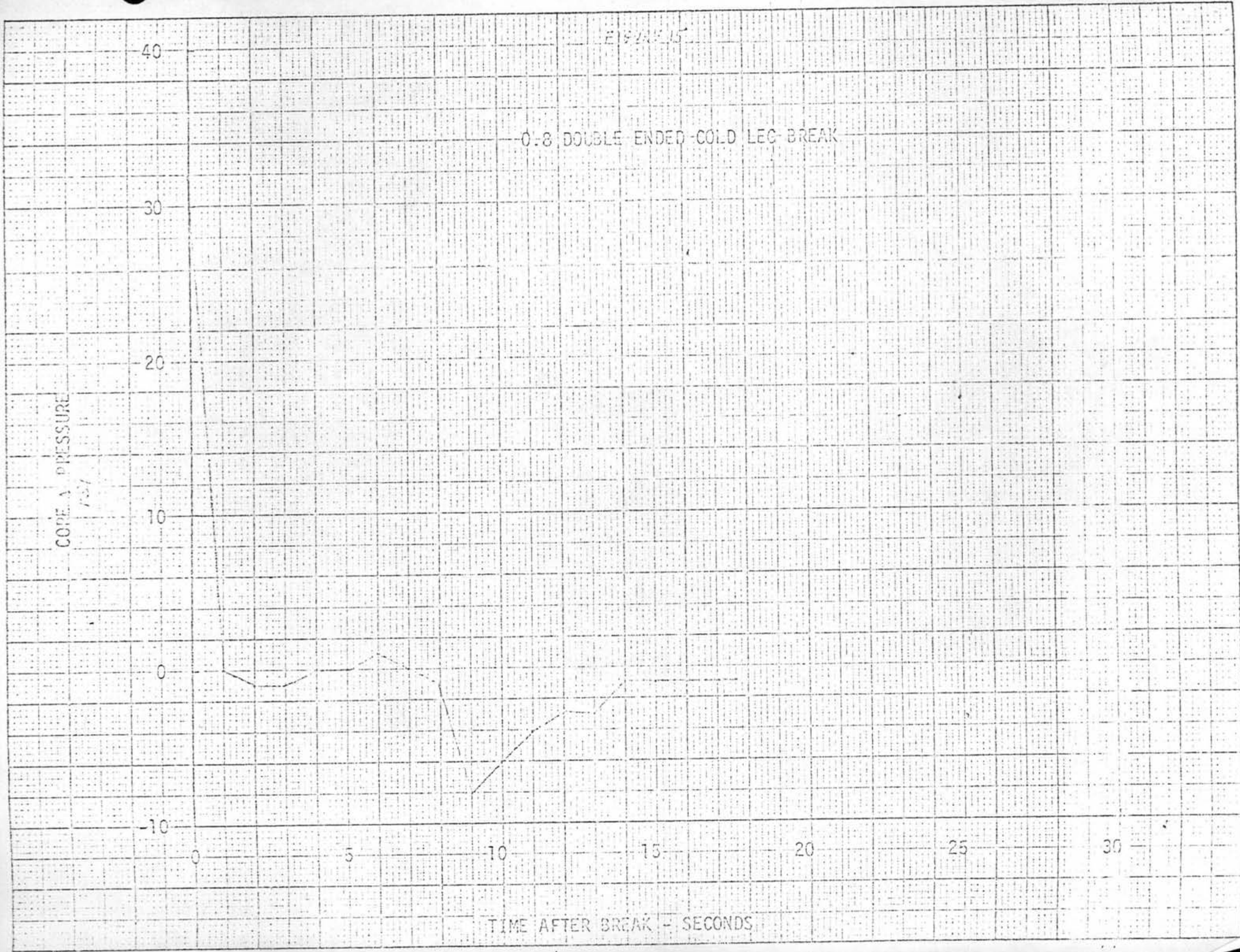


FIGURE 14
DOUBLE ENDED COLD LEG BREAK
CORE A PRESSURE
40
30
20
10
0
-10
0
5
10
15
20
25
30
TIME AFTER BREAK - SECONDS


0.8 DOUBLE ENDED COLD LEG BREAK
CORE Δ PRESSURE
PSI
40
30
20
10
0
-10
0
5
10
15
20
25
30
TIME AFTER BREAK - SECONDS

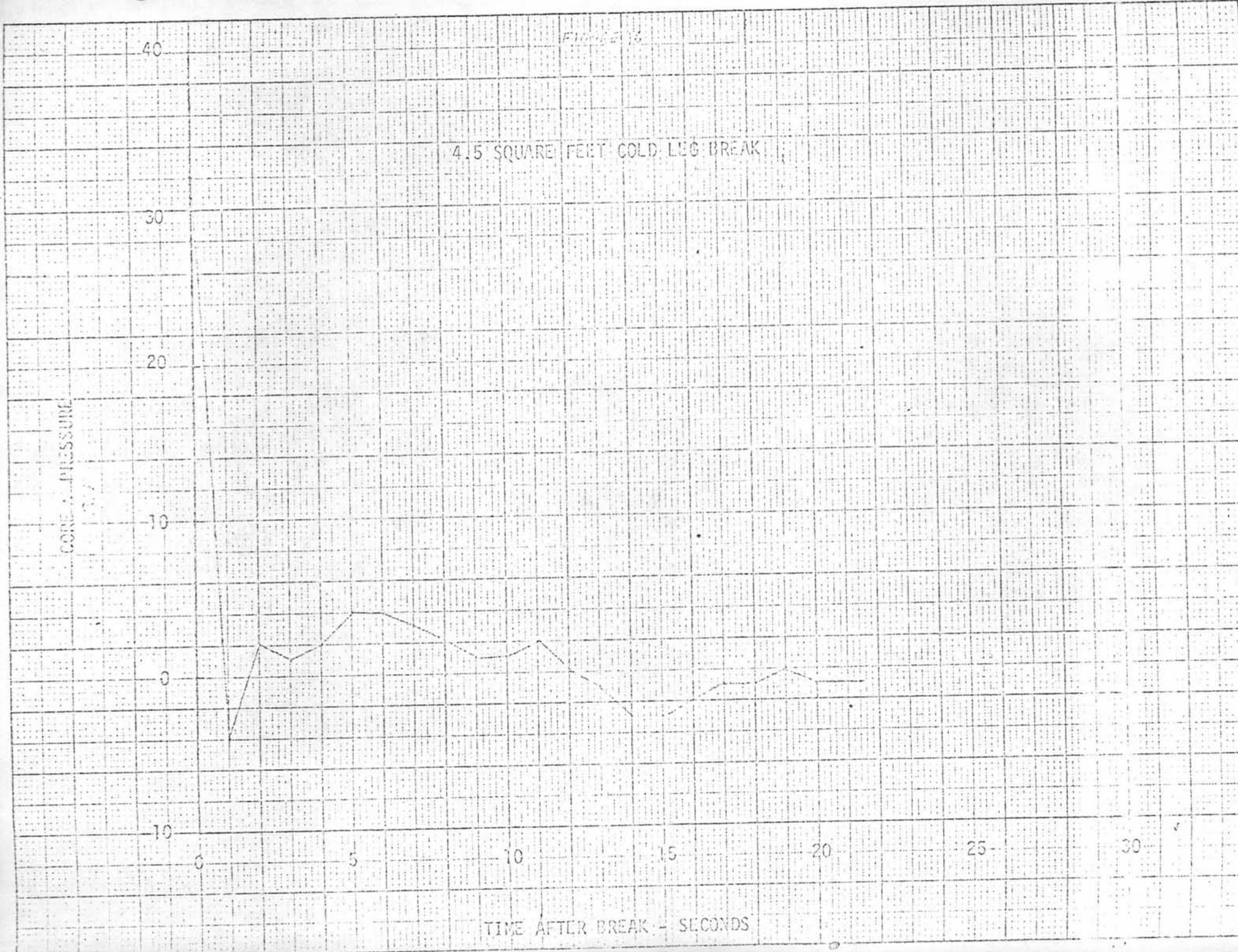

4.5 SQUARE FEET COLD LEG BREAK
40
30
20
10
0
-10
0
5
10
15
20
25
30
TIME AFTER BREAK - SECONDS

FIGURE 17

3 SQUARE FEET COLD LEG BREAK

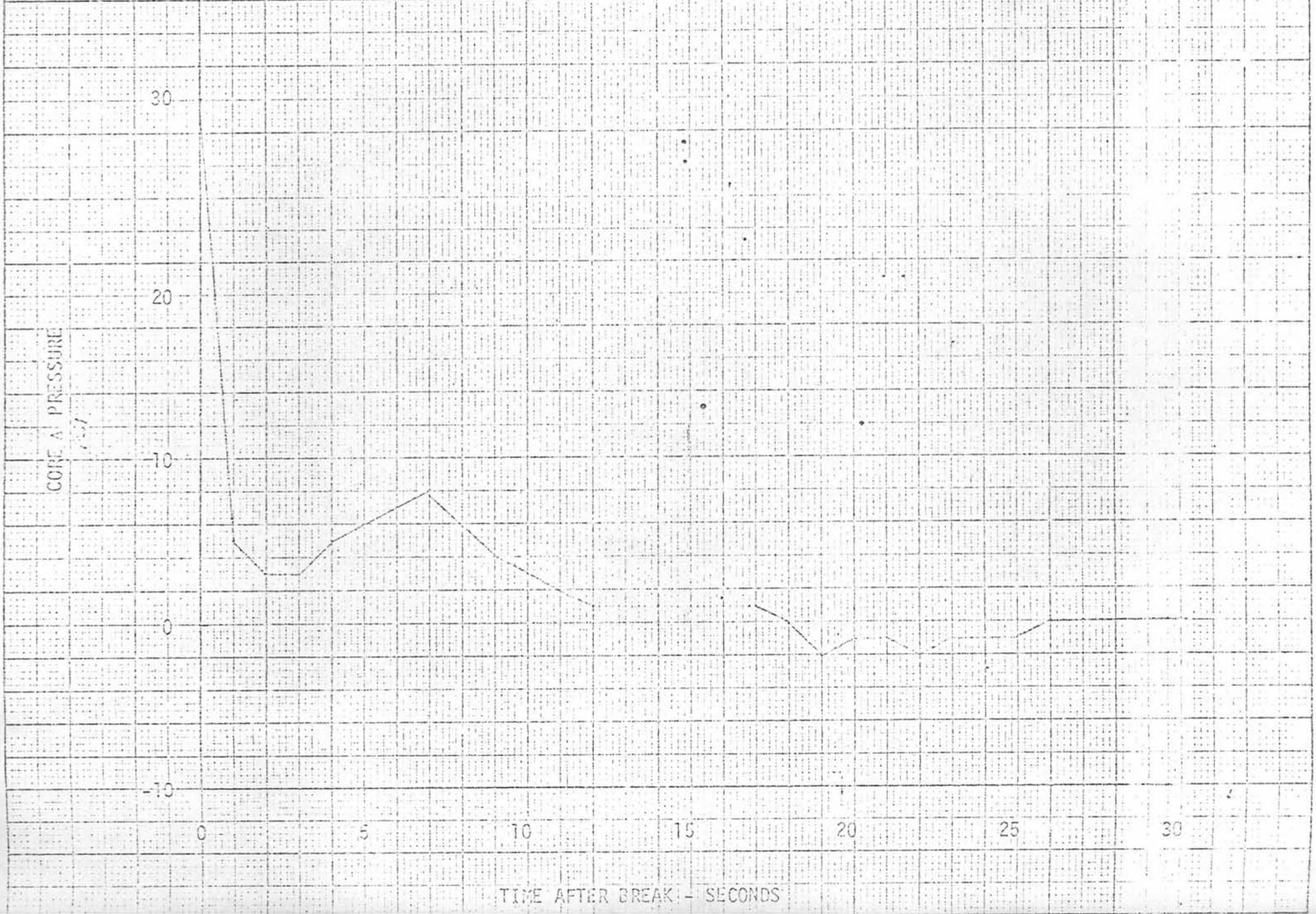


FIGURE 18

DOUBLE ENDED

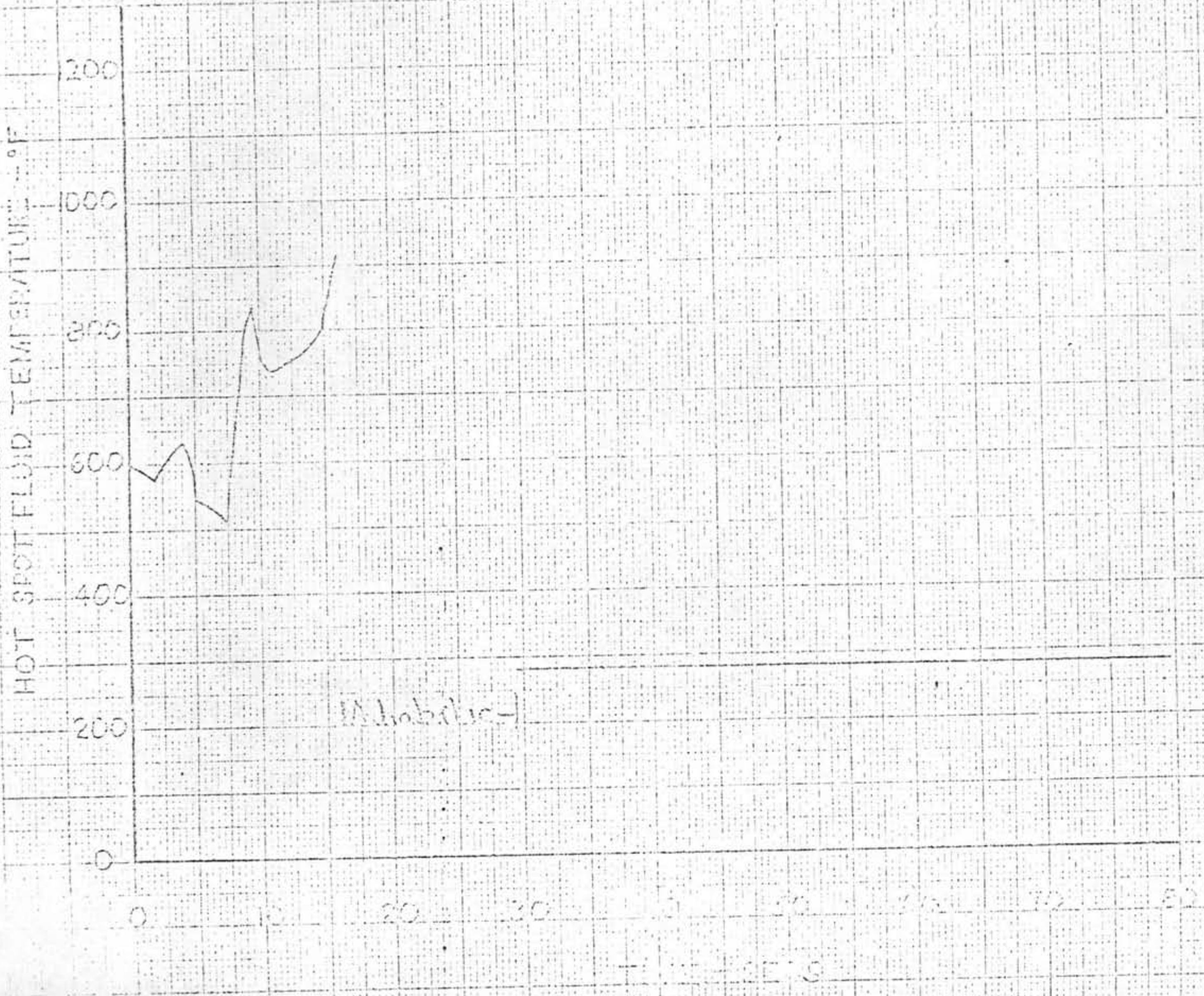
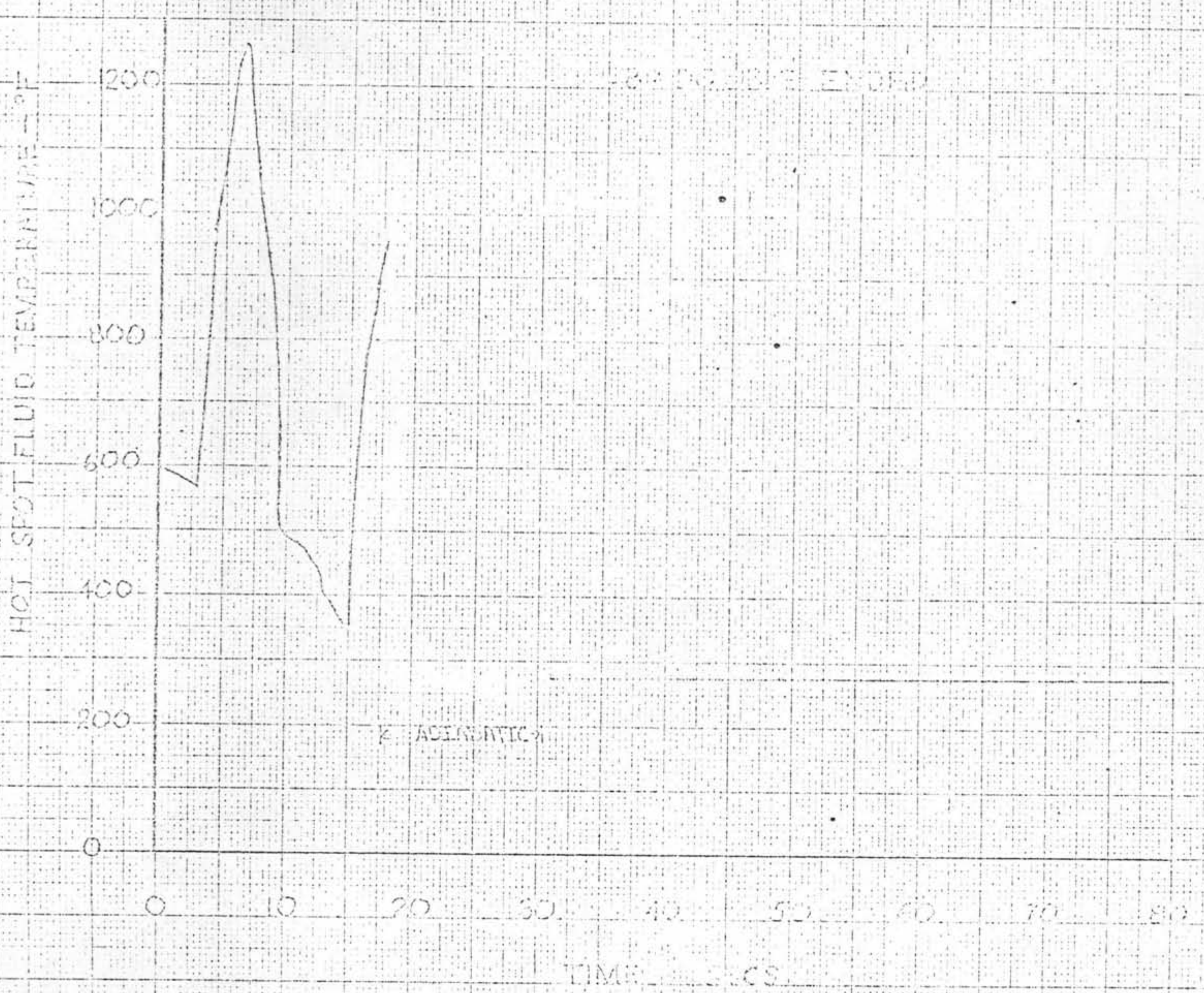


FIGURE 19


HOT SPOT FLUID TEMPERATURE — °F
1200
1000
800
600
400
200
0
0
10
20
30
40
50
60
70
80
TIME

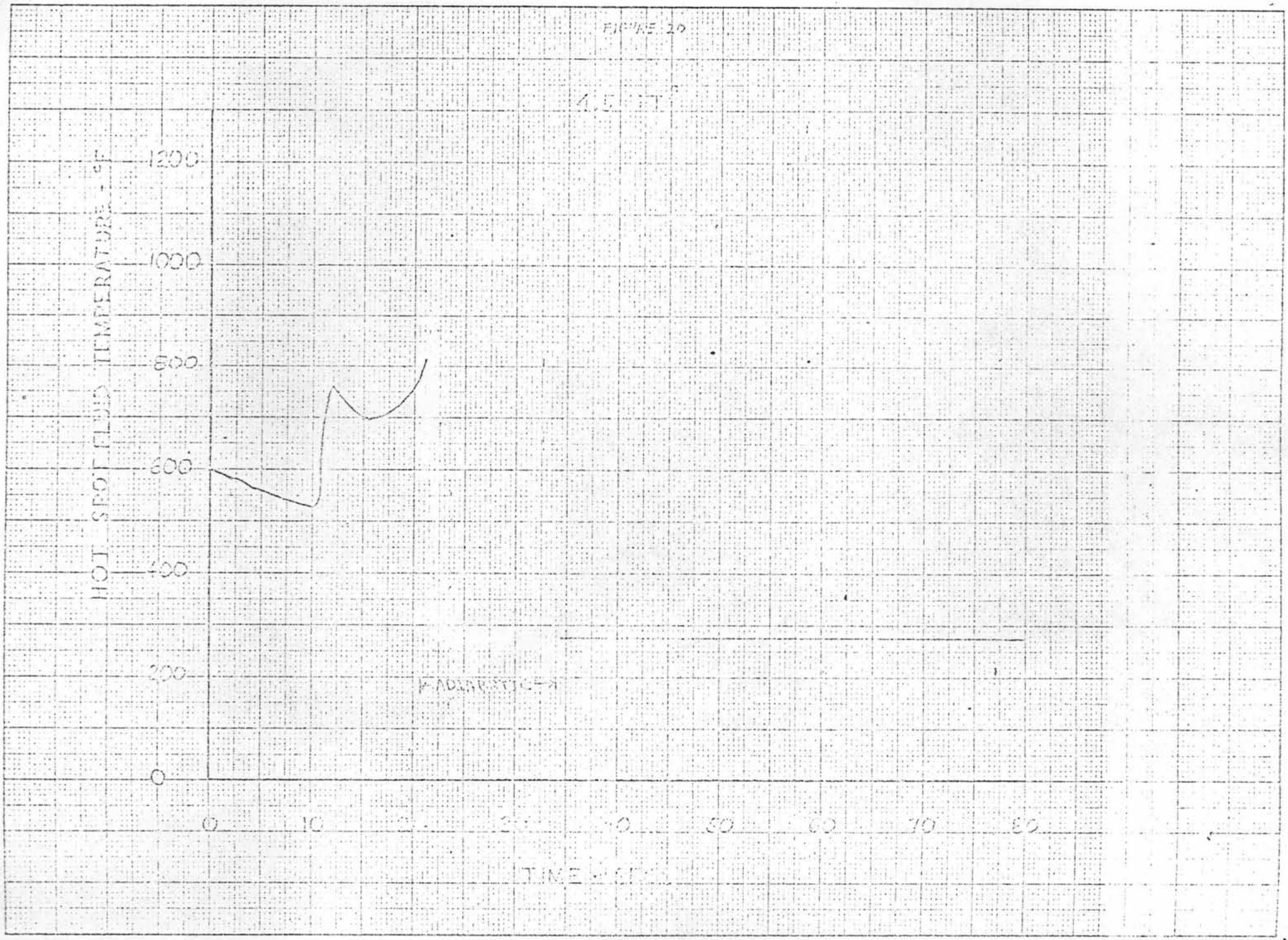
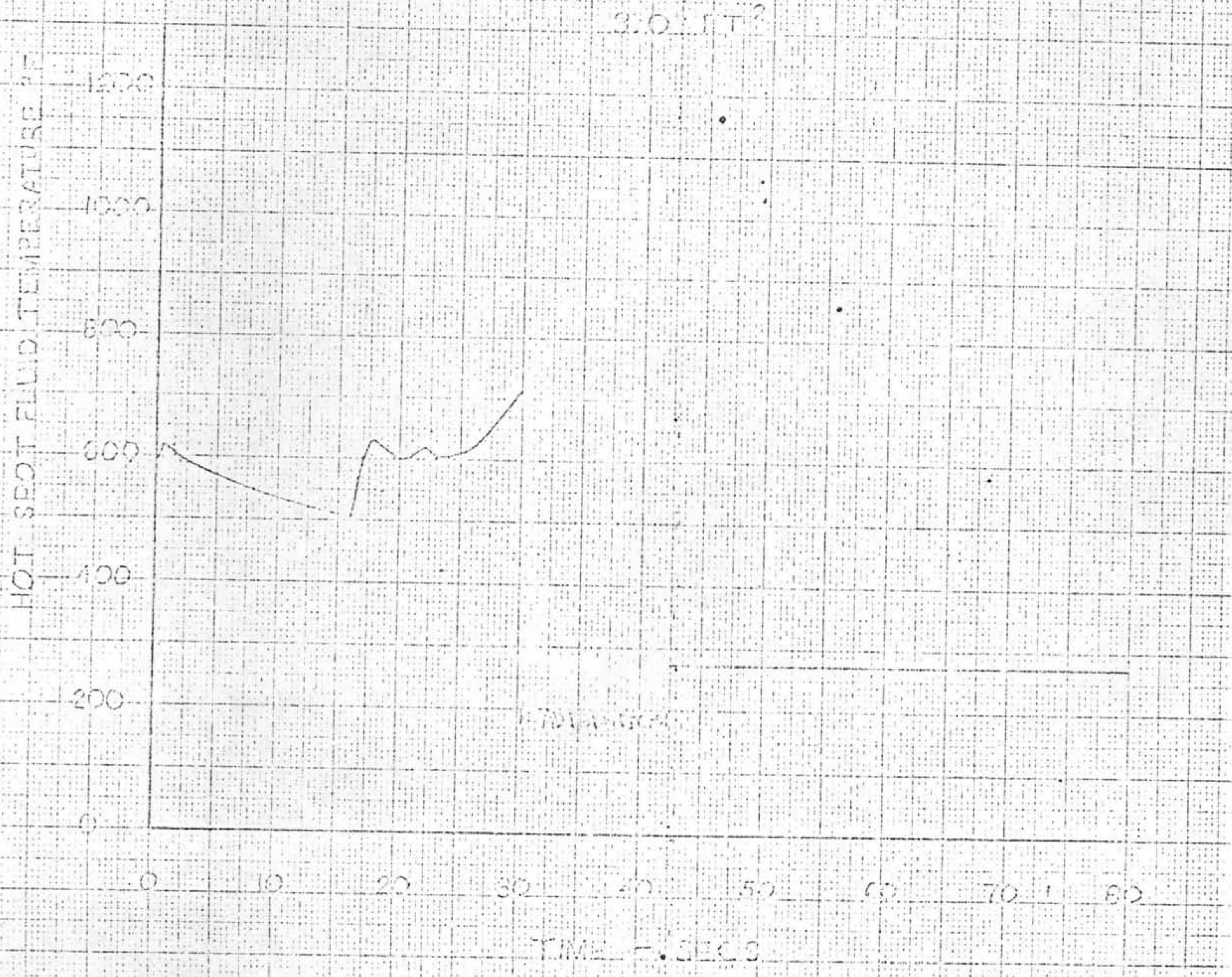


FIGURE 20
HOT SPOT FLUID TEMPERATURE - °F
1200
1000
800
600
400
200
0
ADIABATIC
0
10
50
60
70
80
TIME

FIGURE 21


3.0 FT²
HOT SPOT FLUID TEMPERATURE °F
1200
1000
800
600
400
200
0
0
10
20
30
40
50
60
70
80
TIME - SECS

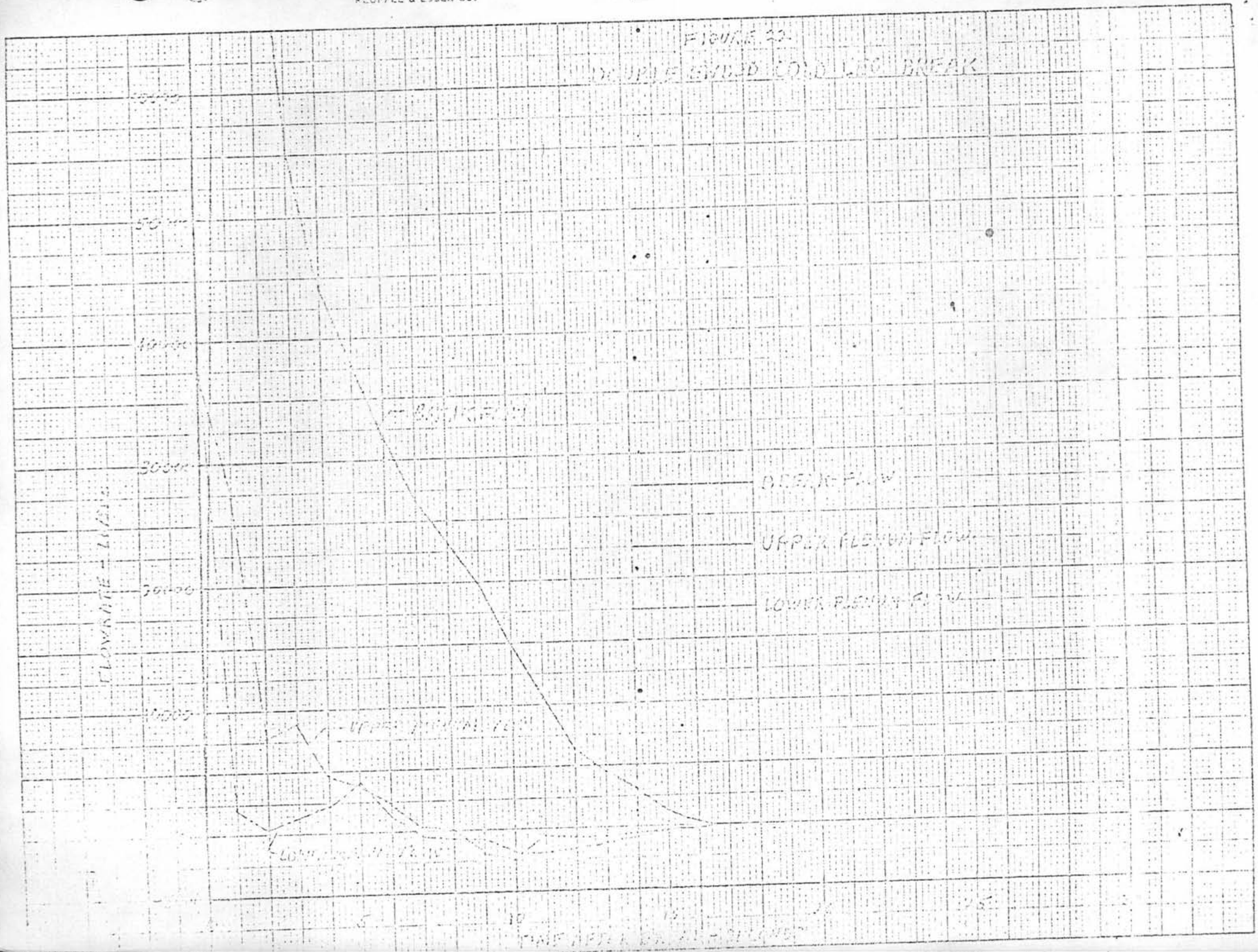
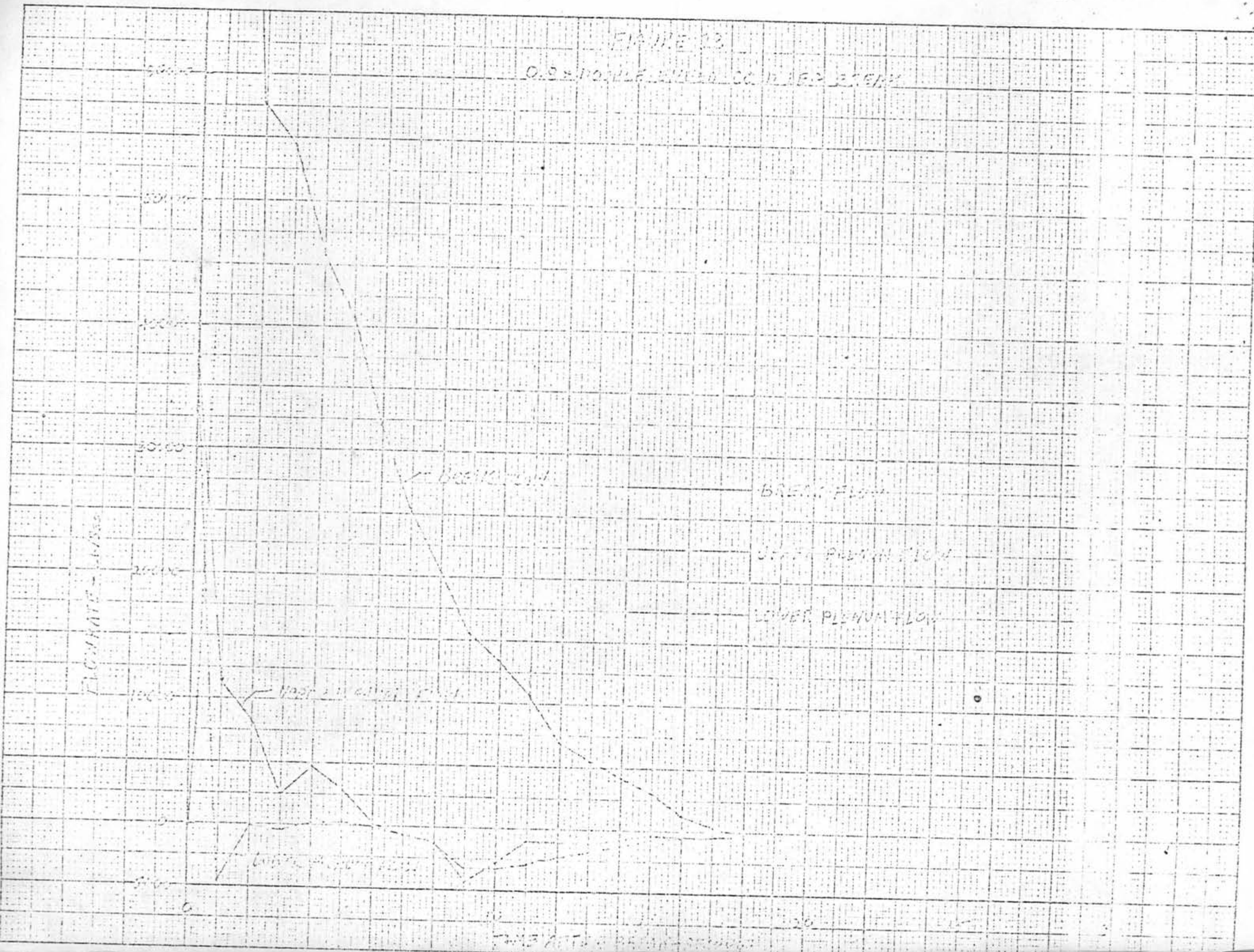
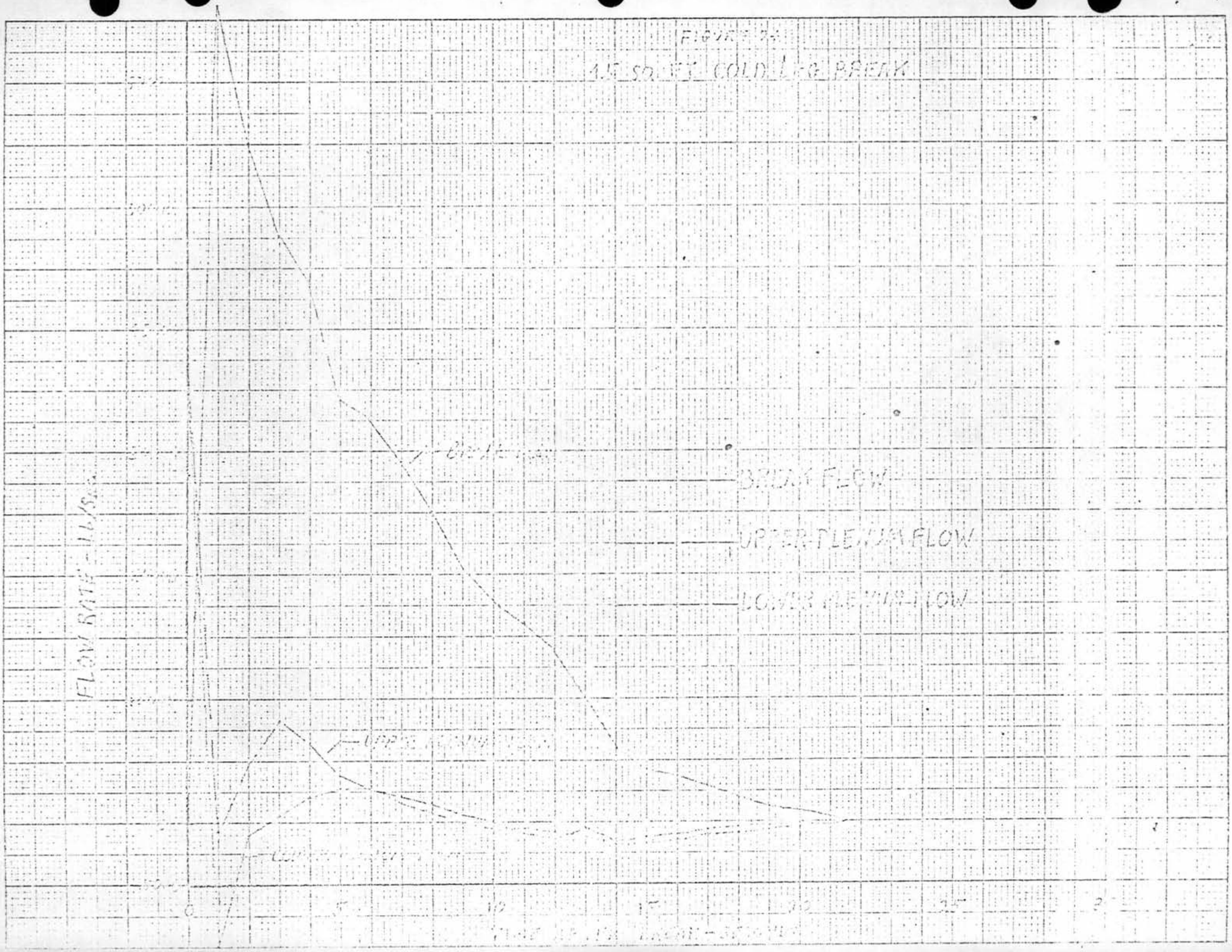


FIGURE 22
DOUBLE ENDED COLD LEG BREAK
BREAK FLOW
UPPER PLENUM FLOW
LOWER PLENUM FLOW
FLOWRATE
3000
5
10
15
25

K+E 10 X 10 TO ½ INCH 46 1470
7 X 10 INCHES MADE IN U.S.A.
KEUFFEL & ESSER CO.




COLD LEG BREAK
BREAK FLOW
UPPER PLENUM FLOW
0
5
10
15
20
25

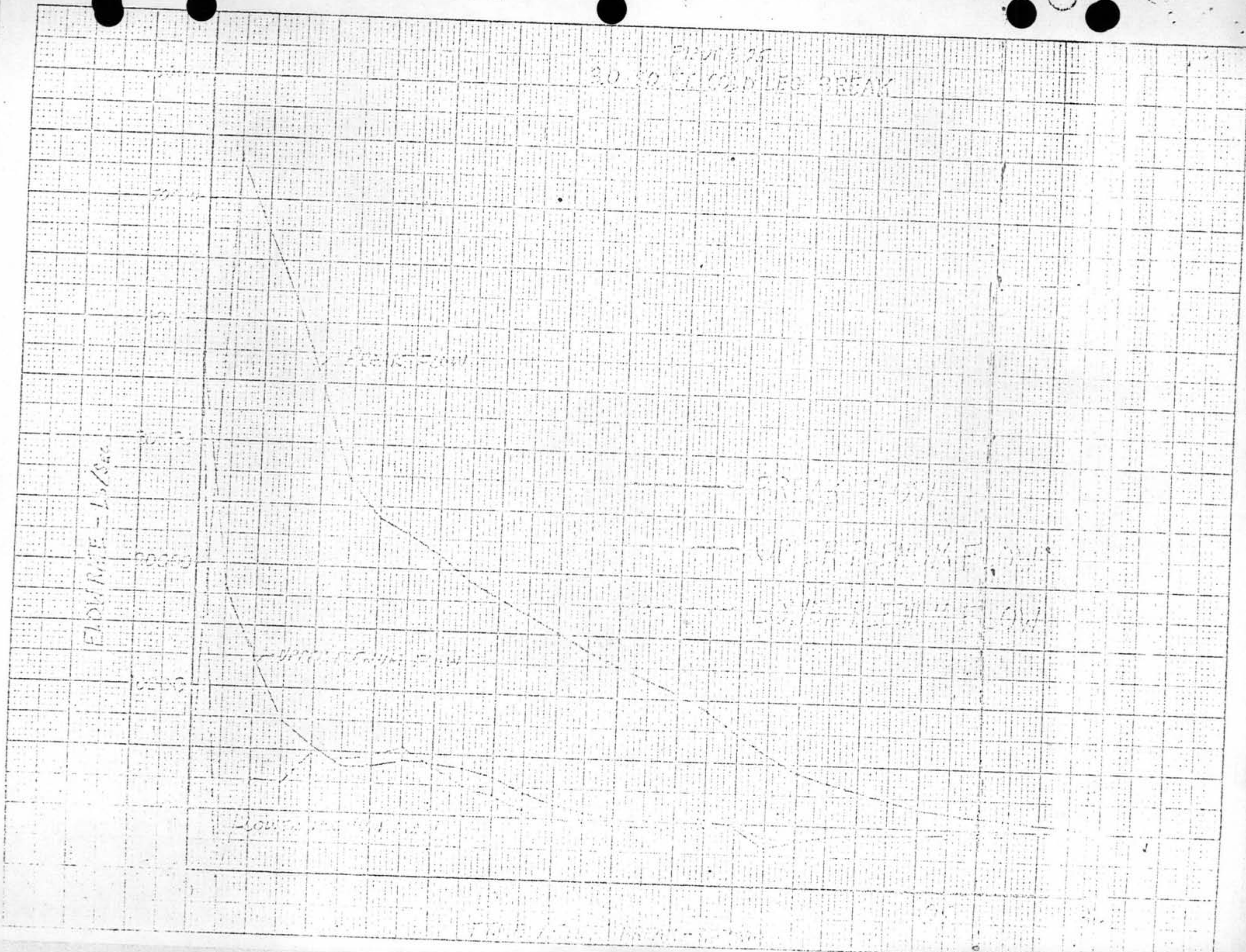
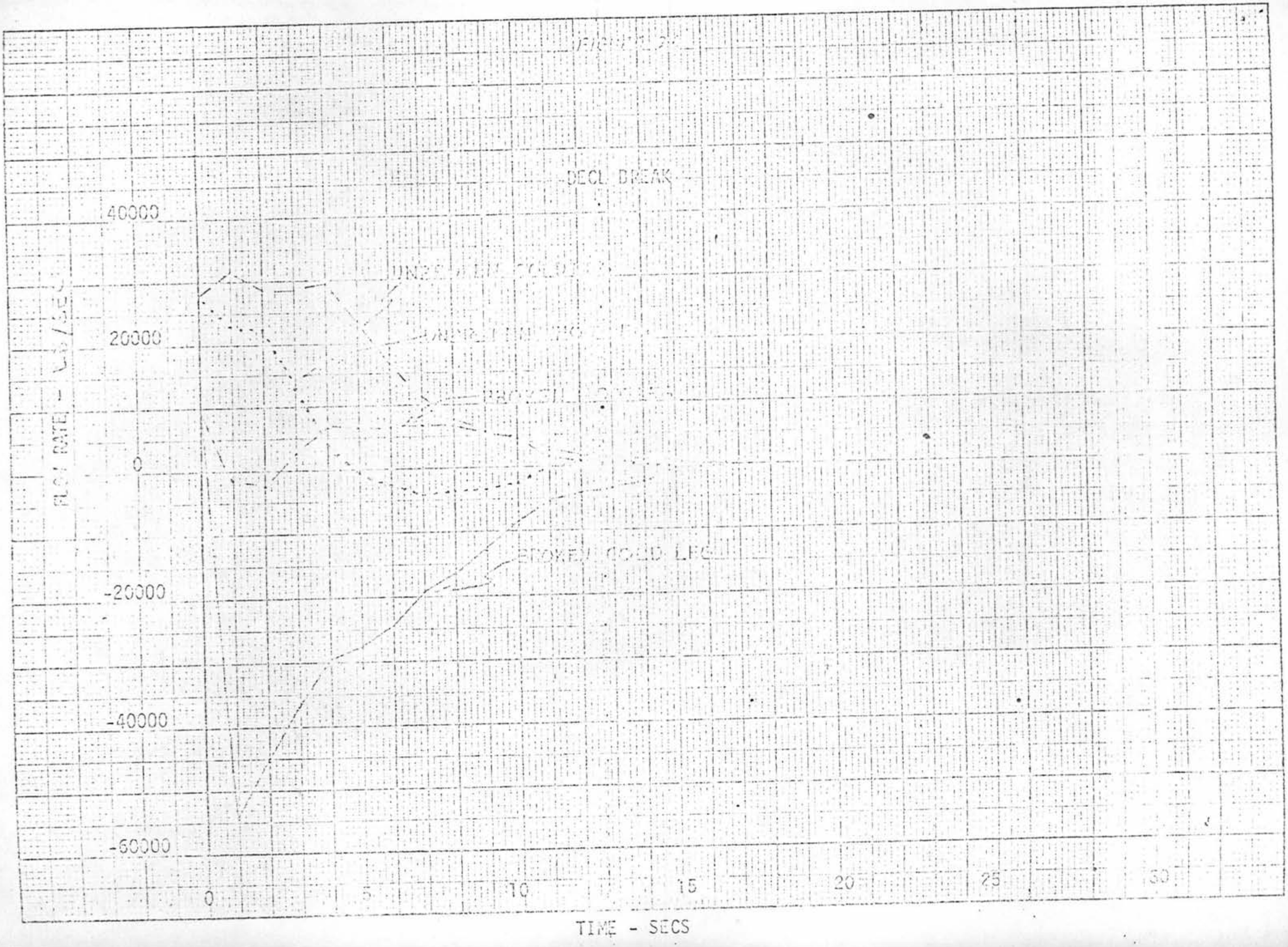
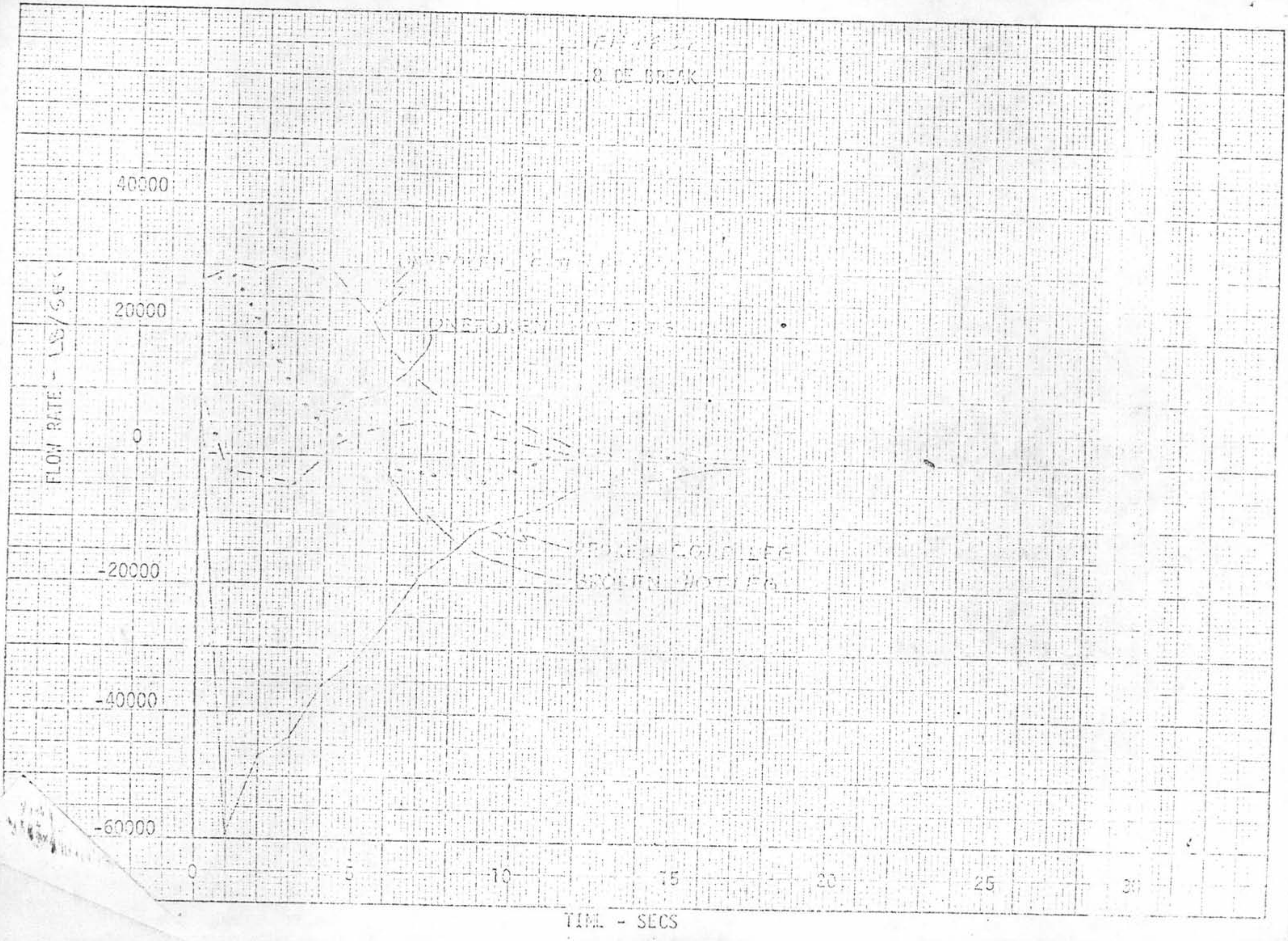
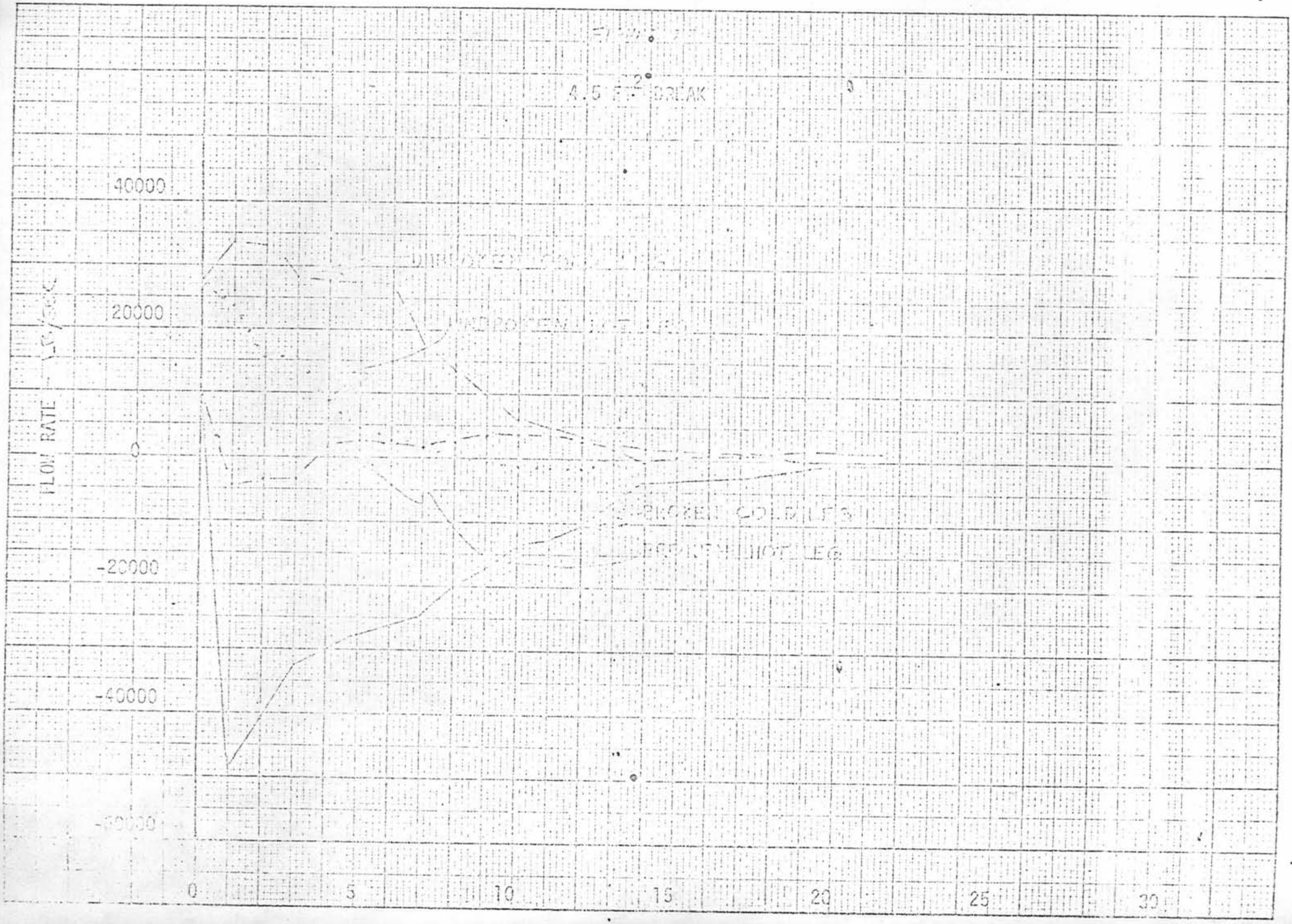
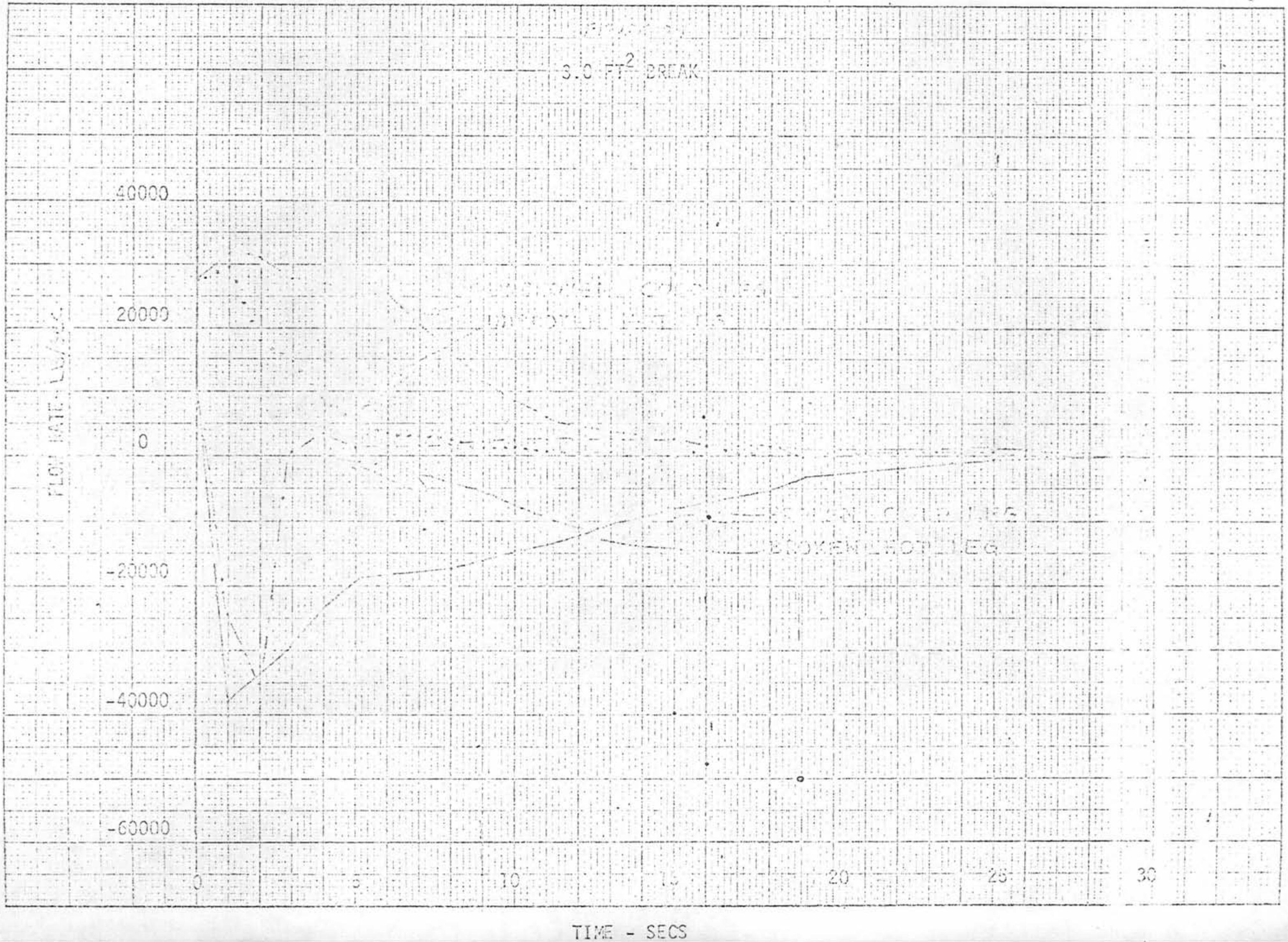


FIGURE 25
3.0 SQ FT COLD LEG BREAK
BREAK FLOW
UPPER PLENUM FLOW
LOWER PLENUM FLOW
FLOW RATE - Lb/Sec


DECL BREAK
40000
20000
0
-20000
-40000
-60000
0
5
10
15
20
25
30
TIME - SECS


8 DE BREAK
BROKEN COLD LEG
BROKEN HOT LEG
FLOW RATE - LB/SEC
40000
20000
0
-20000
-40000
-60000
0
5
10
15
20
25
30
TIME - SECS


4.5 FT² BREAK
FLOW RATE
40000
20000
0
-20000
-40000
-60000
0
5
10
15
20
25
30


3.0 FT^2 BREAK
40000
20000
0
-20000
-40000
-60000
0
5
10
15
20
25
30
BROKEN HOT LEG
TIME - SECS

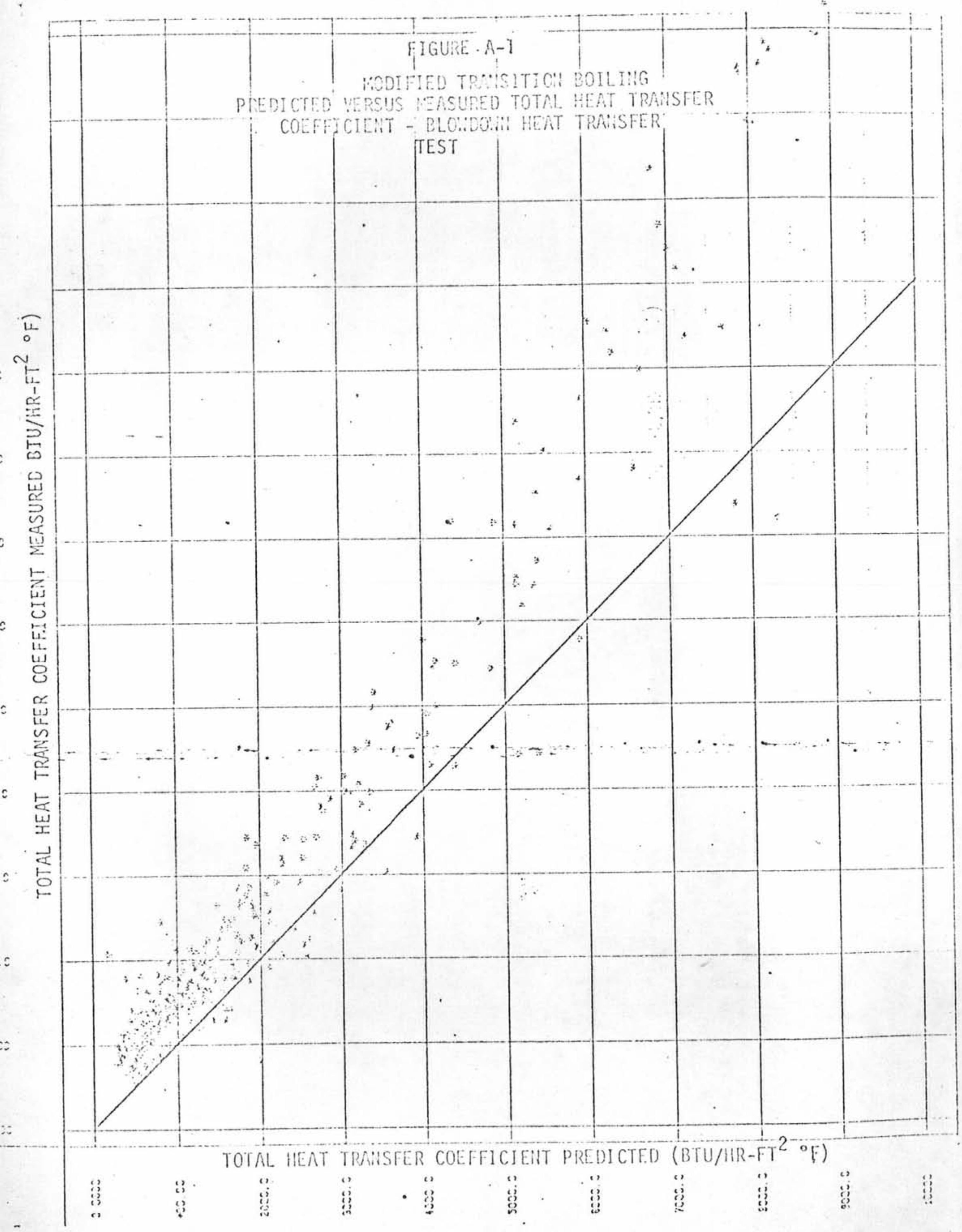
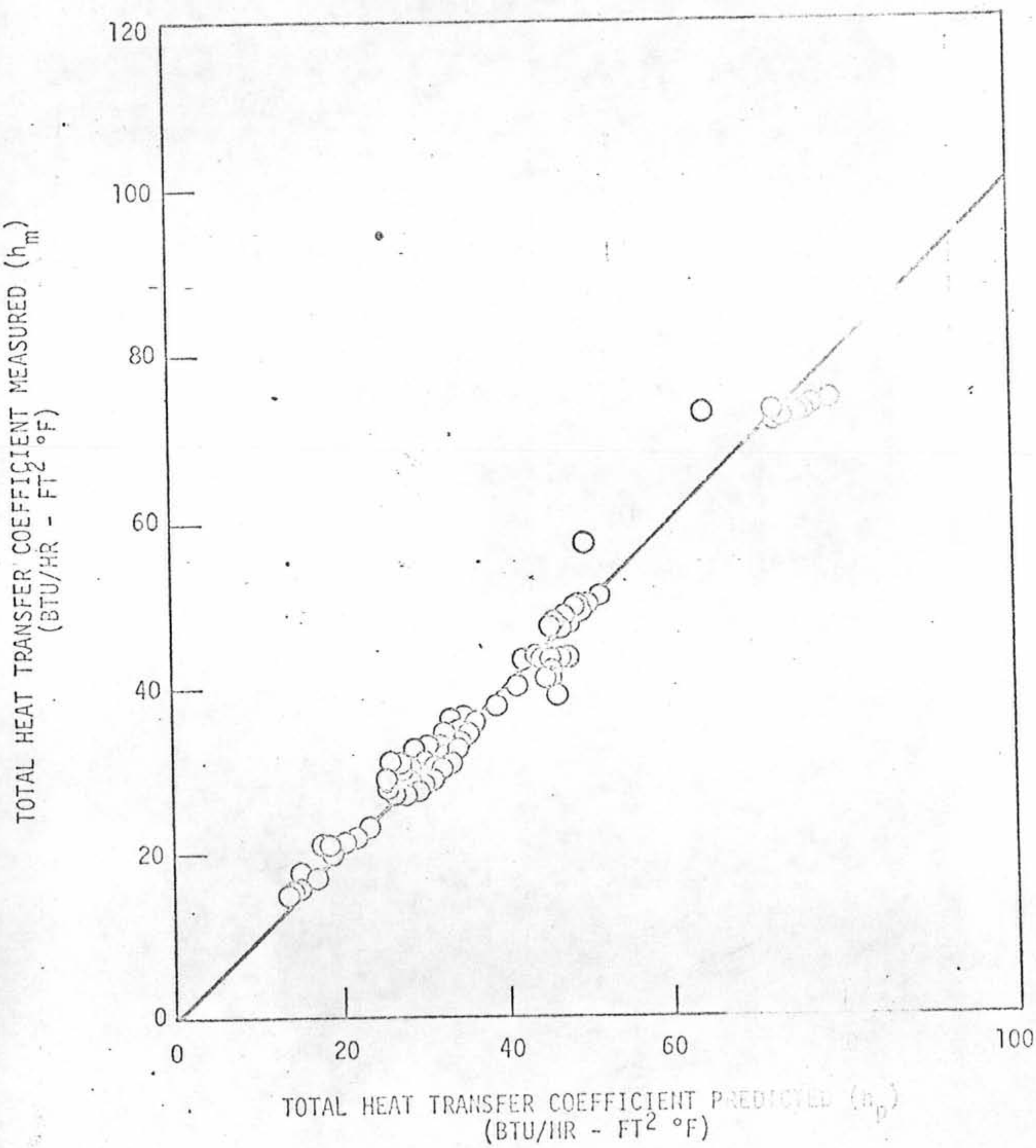


FIGURE A-1
MODIFIED TRANSITION BOILING
PREDICTED VERSUS MEASURED TOTAL HEAT TRANSFER
COEFFICIENT - BLOWDOWN HEAT TRANSFER
TEST
TOTAL HEAT TRANSFER COEFFICIENT MEASURED BTU/HR-FT2 °F)
TOTAL HEAT TRANSFER COEFFICIENT PREDICTED (BTU/HR-FT2 °F)

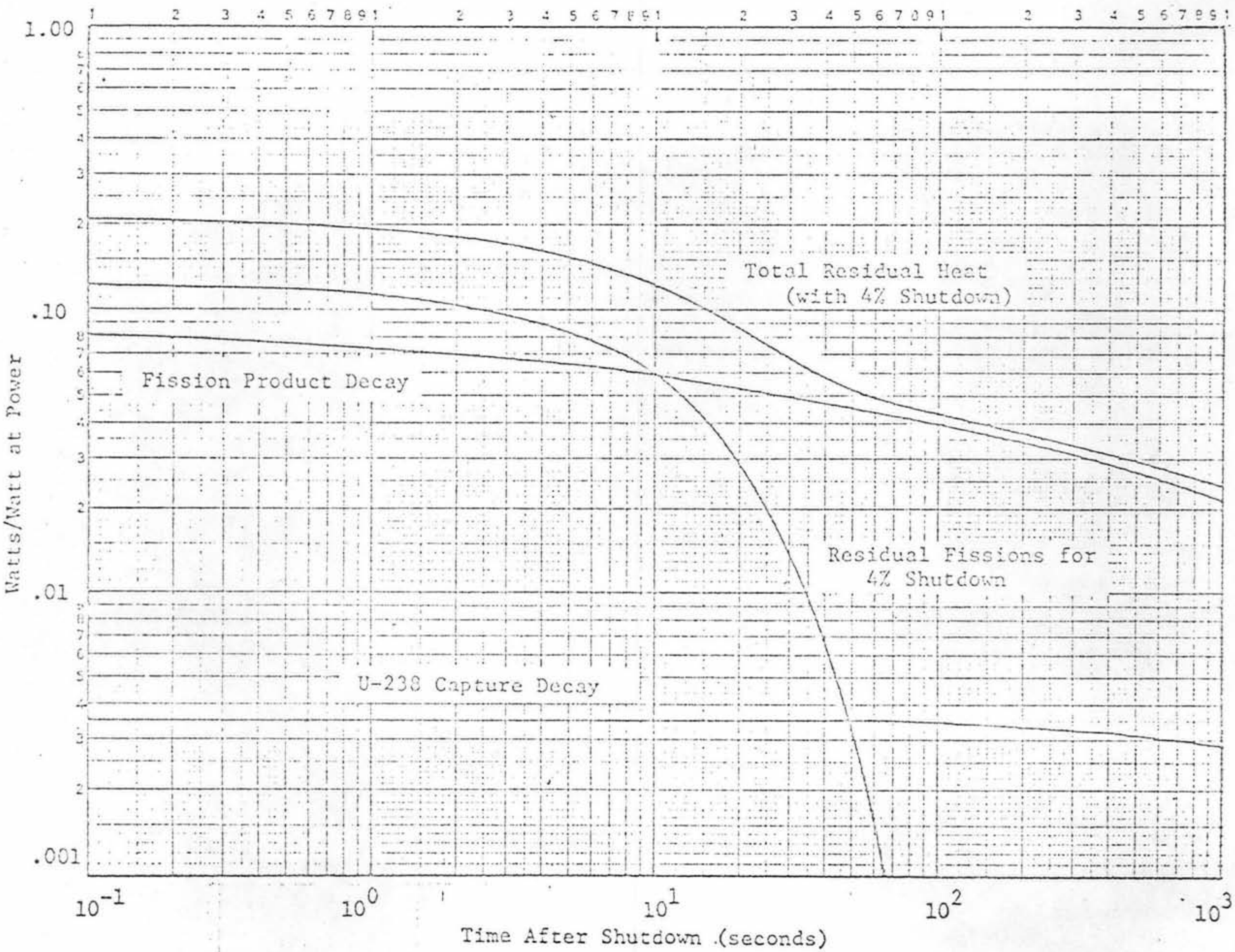
FIG. A-2



UNIVERSITY OF MICHIGAN HEAT TRANSFER TEST - TURBULENT FLOW DATA - TOTAL HEAT TRANSFER COEFFICIENT - LOW PRESSURE HEAT TRANSFER

FIG. B-1.22

RESIDUAL DECAY HEAT



APPENDIX C

Response to letter from Division of Reactor Licensing dated July 7, 1971

By a letter of July 7, 1971 the Division of Reactor Licensing requested the Applicant to provide certain information required for its evaluation of the emergency core cooling system for Indian Point Unit No. 2. The following indicates the location in the preceding report of specific items of information requested, and in addition provides the requested information not covered by the report.

Item 1 (a) System Pressure Transients
Figures 6, 7, 8, 9

1 (b) Hot Spot Clad Temperature Transients
Figures 10, 11, 12, 13

Hot Spot Local Mass Velocity
Can be obtained by using 80 percent of the core flow on Figures 6, 7, 8, 9

Hot Spot Fluid Temperatures
Figures 18, 19, 20, 21

Hot Spot Heat Transfer Coefficient
Figures 6, 7, 8, 9

1 (c) Core Pressure Drop
Figures 14, 15, 16, 17

Hot Spot Quality
Figures 6, 7, 8, 9

Average Mass Velocity at Core Midplane
Figures 6, 7, 8, 9 (From Core Flow)

1 (d) Heat Flux Distribution in the Hot Channel was a 1.79 Peak to Average Cosine

1 (e) Flow Rates in Upper and Lower Plenums
Figures 22, 23, 24, 25

1 (f) Flow Rates in the Broken and Intact Cold-Leg and Hot-Leg Piping
Figures 26, 27, 28, 29

1 (g) Flow Rate Out of the Break
Figures 22, 23, 24, 25

1 (h) Hot Spot Local Clad Metal-Water Reaction
Table 2

Item 2 Description of the Core Reflood Model
Page 10 thru 17

Item 3 There were no deviations from Appendix A, Part 3 of the interim policy statement

The core cooling analysis was performed on the basis of a peak power density of 17.4 kw/ft at 102 percent power. The appropriate Technical Specification reference to power distribution will be revised as follows:

Page 3.10-2

Section 3.10.2.2(b)

The hot channel factors shall be determined and maximum allowable power shall be reduced one percent for each percent the hot channel factors exceed the design values of

$$F_q^N = 2.90 \quad \text{or} \quad F_{\Delta H}^N = 1.66$$

8/16/71

ADDITIONAL INFORMATION ON

EMERGENCY CORE COOLING ANALYSIS

This information should be added to the discussion entitled, "Results" appearing on Page 18 of the document entitled, "Additional Testimony of Applicant Concerning Emergency Core Cooling System Performance", dated July 13, 1971.

1. Details of the Reflooding Analysis

Figure 1 is a plot of both the core flooding rate and the hot spot film heat transfer coefficient as a function of time used in the double-ended cold leg break Loss of Coolant Analysis. The time that the accumulators empty is also indicated on the graph.

2. 0.5 Square Foot Cold Leg Break

An analysis has been performed for the 0.5 ft^2 cold leg break using the interim criteria. The peak clad temperature was 2185°F. This is greater than the peak clad temperature calculated for the 3.0 ft^2 cold leg break, which is not surprising because a decrease in break size increased the amount of the accumulator water than can be injected before the end of blowdown. In fact, for these smaller breaks the clad temperature rise is stabilized before the end of blowdown, but on the other hand, these smaller breaks define the lower limit of the break size for which accumulator water loss need be considered because of the limit on the maximum amount of water that can be passed through the break and the much lower steam velocities in the system.

Figures 2 through 8 provide details of the 0.5 ft^2 cold leg break transient analysis.

3. Hot Leg Break Loss of Coolant Analysis

The hot leg break analysis was previously performed and presented in the Indian Point Unit No. 2 FSAR, Supplement 12, July 1970, and indicated the following:

a. There is no flow reversal during hot leg break blowdown; the flow just decays smoothly, providing good heat transfer in the core.

b. In a cold leg break analysis, the accumulator and low head injection flow for the broken loop is assumed to spill to the containment. This assumption is not valid for a hot leg break; thus, there is more accumulator water mass and more low head injection flow available for cooling.

c. During the reflood phase of the accident there is no problem with steam binding since the steam generated in the core is vented directly to the containment via the broken hot leg. In addition, there is more accumulator water available, heat transfer during reflood for the hot leg break will be better than for the cold leg break. The peak clad temperature will be less than 2000°F.

4. 0.6 DE Cold Leg Break

Sensitivity to break mode for various break sizes has been studied in the past. Results showed that for the double-ended break, the temperature for the guillotine type was approximately 100°F greater than for the split type. For lesser break areas, as seen in results for the 0.8 double-ended break, the split type had a higher temperature than the guillotine. If the 0.6 double-ended area break (which is 4.9 ft^2) were analyzed, the split type would yield a higher peak clad temperature. Instead of analyzing the 0.6 double-ended area break, the 4.5 ft^2 area split type break was used and results were reported. The following table indicates the break mode for each break size.

Break Size	Mode
Double-ended	Guillotine
0.8 Double-ended	Split
4.5 ft^2	Split
3.0 ft^2	Split
0.5 ft^2	Split

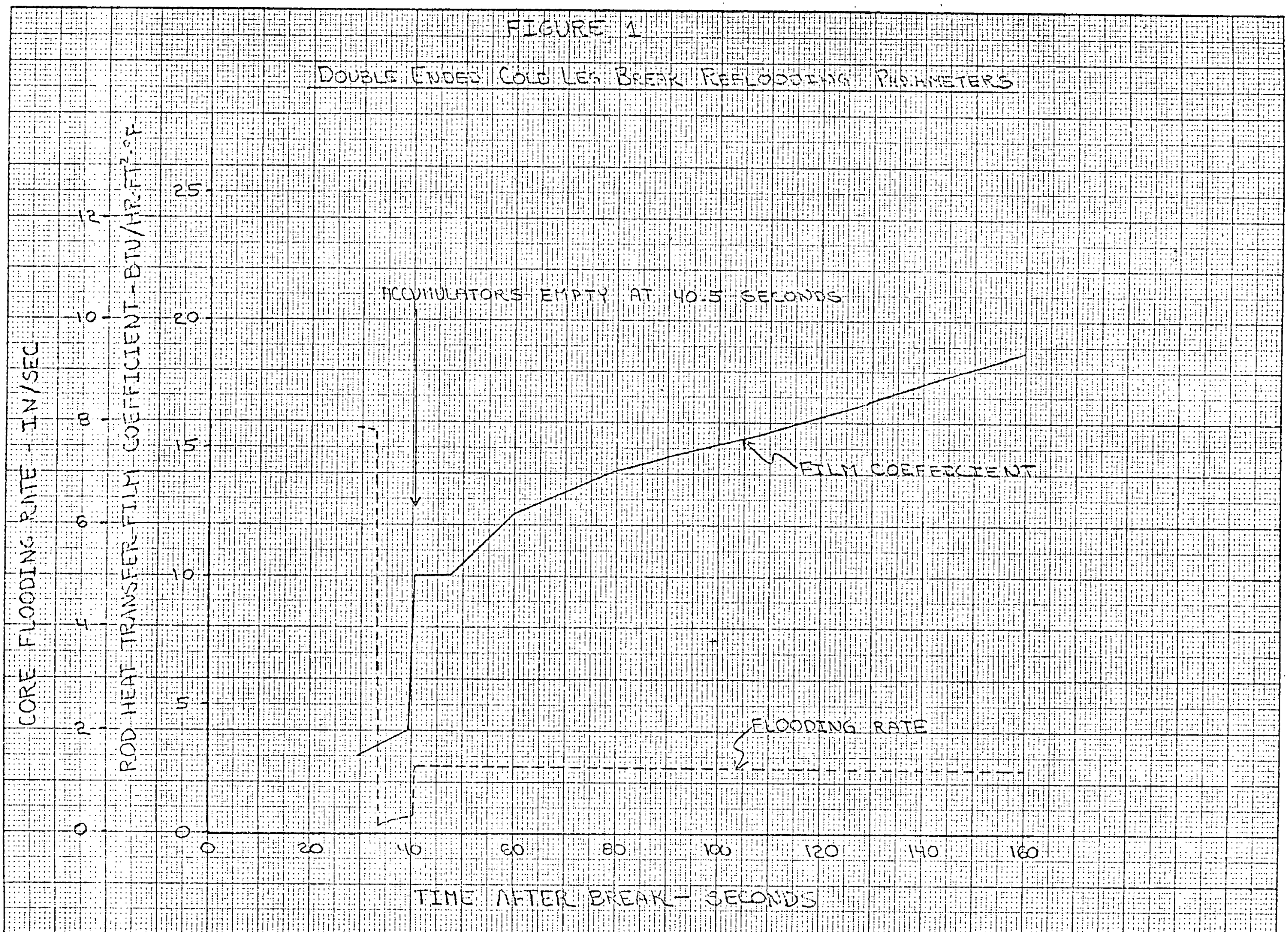


FIGURE 1
DOUBLE ENDED COLD LEG BREAK REFLOODING PARAMETERS
ACCUMULATORS EMPTY AT 40.5 SECONDS
FILM COEFFICIENT
FLOODING RATE
CORE FLOODING RATE - IN/SEC
ROD HEAT TRANSFER FILM COEFFICIENT - BTU/HR-FT²-°F
TIME AFTER BREAK - SECONDS
0
2
4
6
8
10
12
0
5
10
15
20
25
0
20
40
60
80
100
120
140
160

FIGURE 2

0.5 FT^2 BREAK

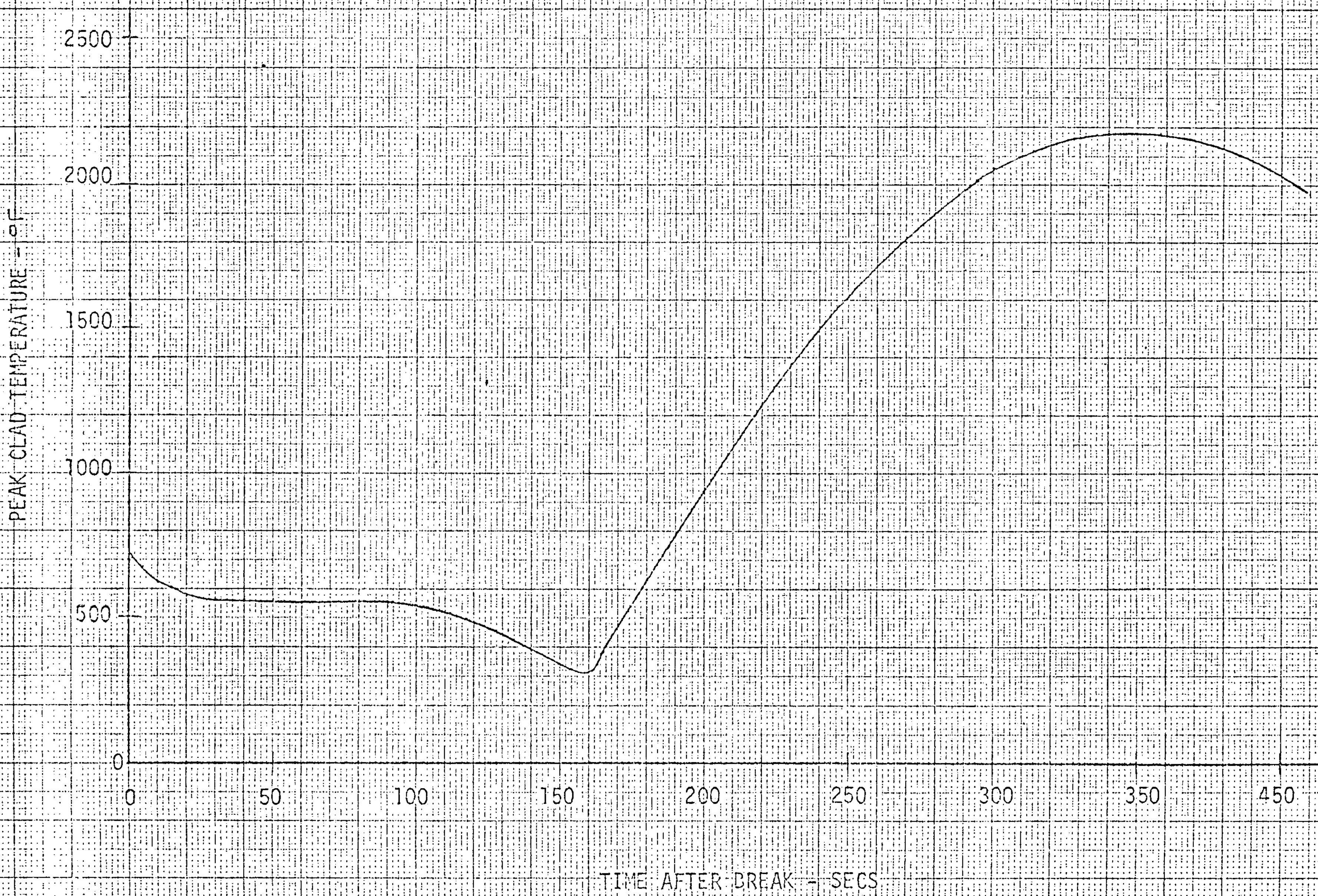
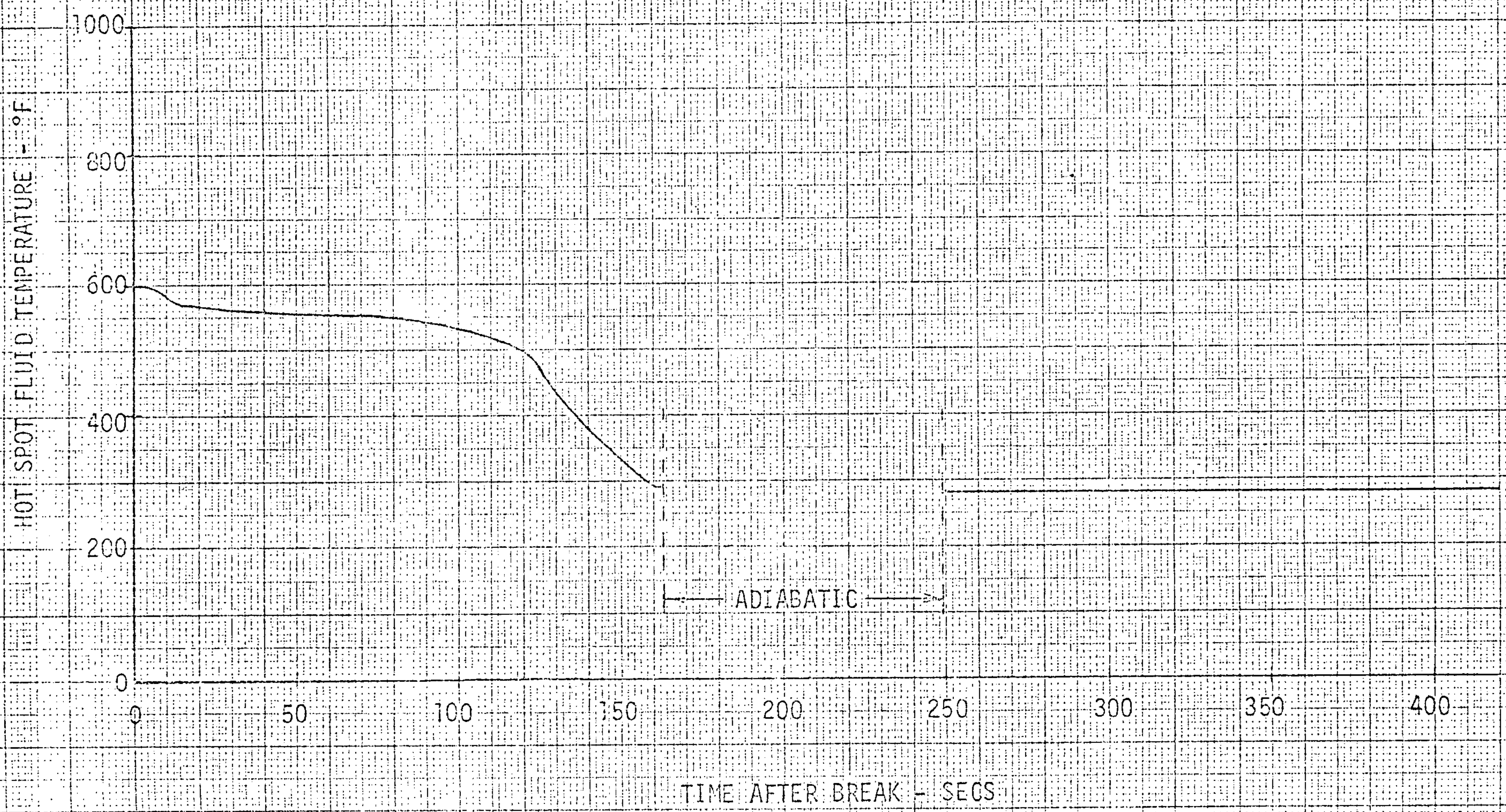


FIGURE 3

0.5 FT^2 BREAK



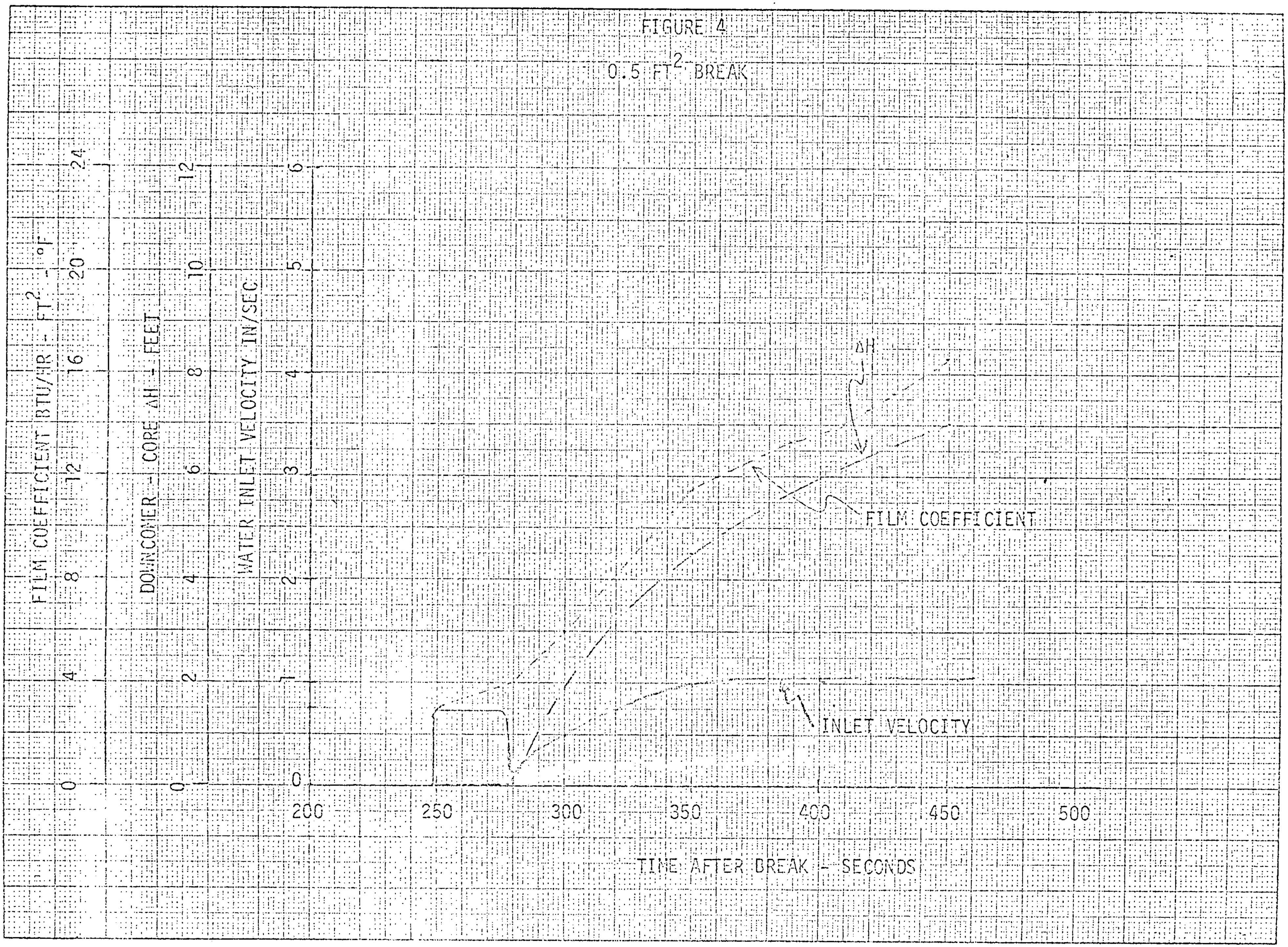


FIGURE 4
0.5 FT^2 BREAK
FILM COEFFICIENT BTU/HR - FT^2 - °F
0
4
8
12
16
20
24
DOWNCOMER - CORE ΔH - FEET
0
2
4
6
8
10
12
WATER INLET VELOCITY IN/SEC
0
1
2
3
4
5
6
ΔH
FILM COEFFICIENT
INLET VELOCITY
200
250
300
350
400
450
500
TIME AFTER BREAK - SECONDS

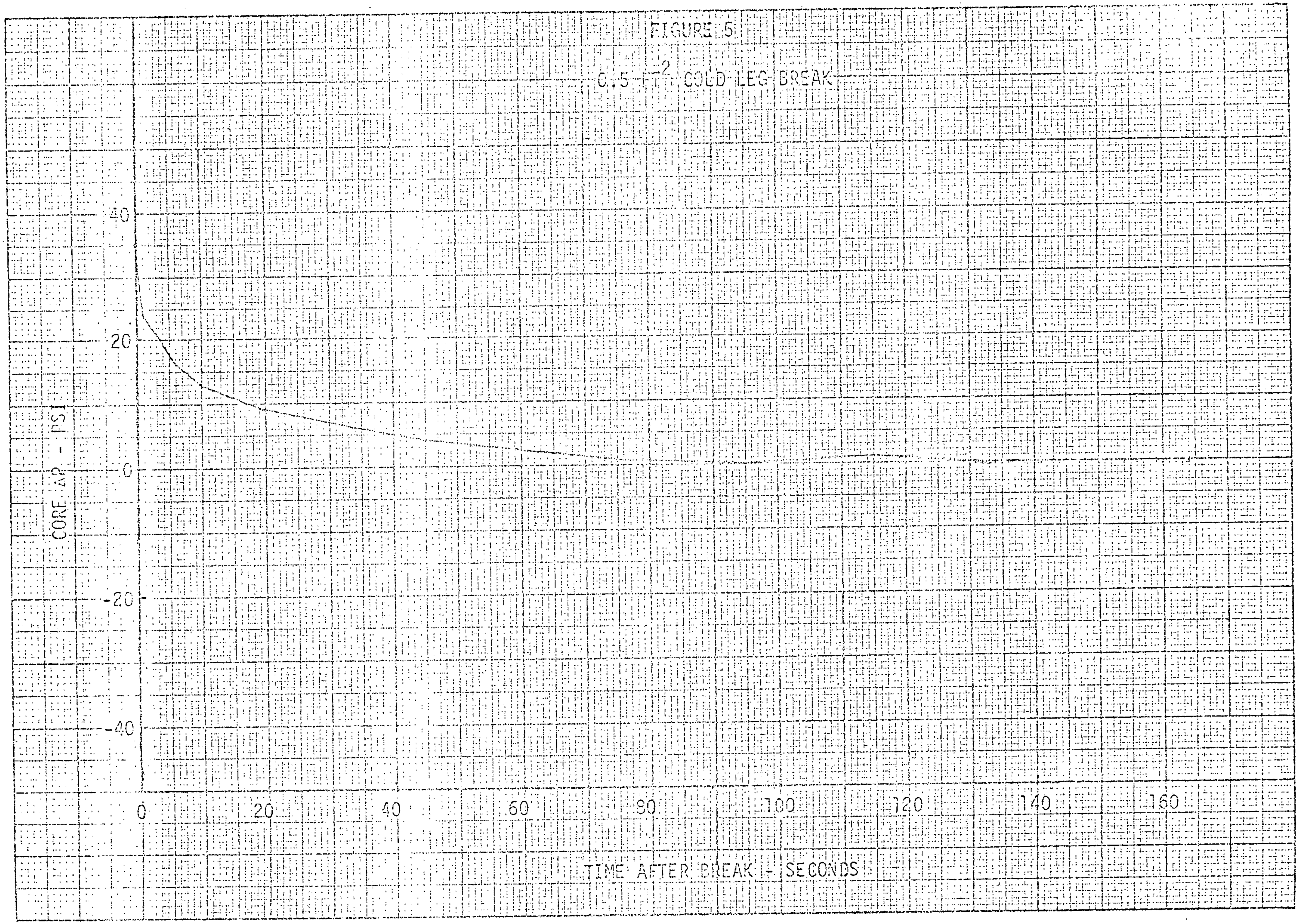
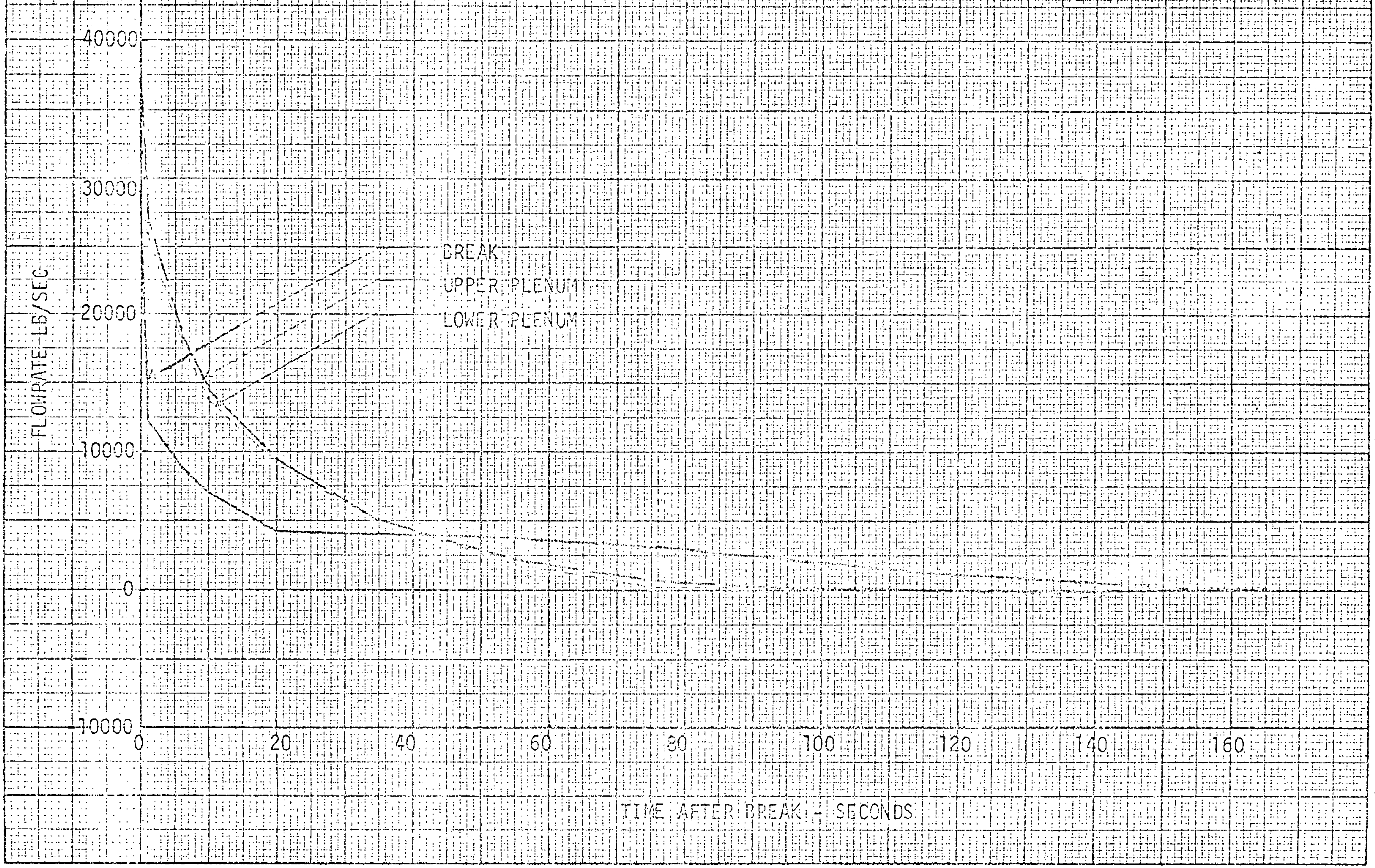
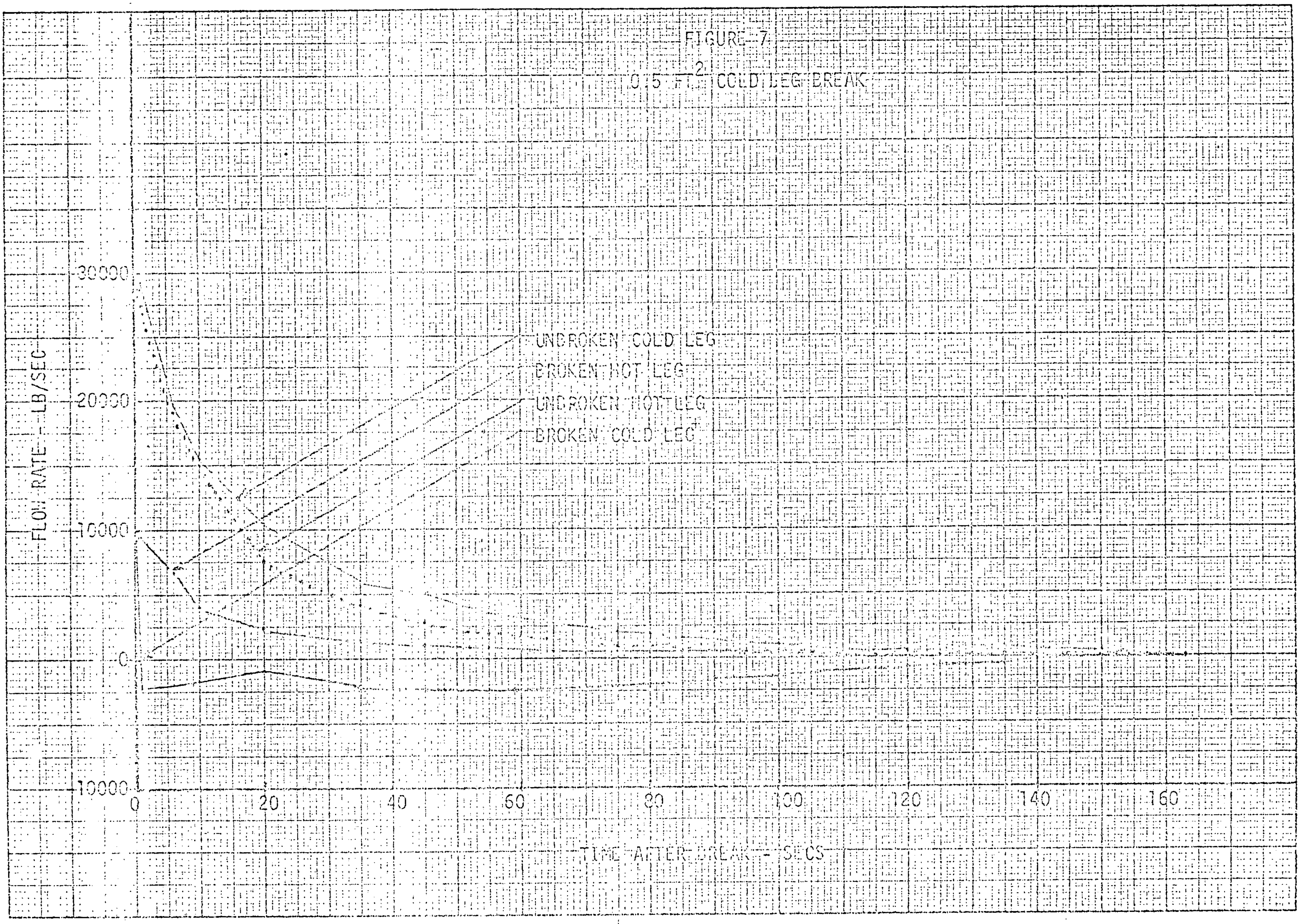


FIGURE 5
0.5 FT^2 COLD LEG BREAK
CORE ΔP - PSI
40
20
0
-20
-40
0
20
40
60
80
100
120
140
160
TIME AFTER BREAK - SECONDS

FIGURE 6

0.5 FT^2 COLD LEG BREAK





K+E 10 X 10 TO ½ INCH 46 1320
7 X 10 INCHES MADE IN U.S.A.
KEUFFEL & ESSER CO.
FIGURE 7
0.5 FT.² COLD LEG BREAK
UNBROKEN COLD LEG
BROKEN HOT LEG
UNBROKEN HOT LEG
BROKEN COLD LEG
FLOW RATE - LB/SEC
30000
20000
10000
0
-10000
0
20
40
60
80
100
120
140
160
TIME AFTER BREAK - SECS

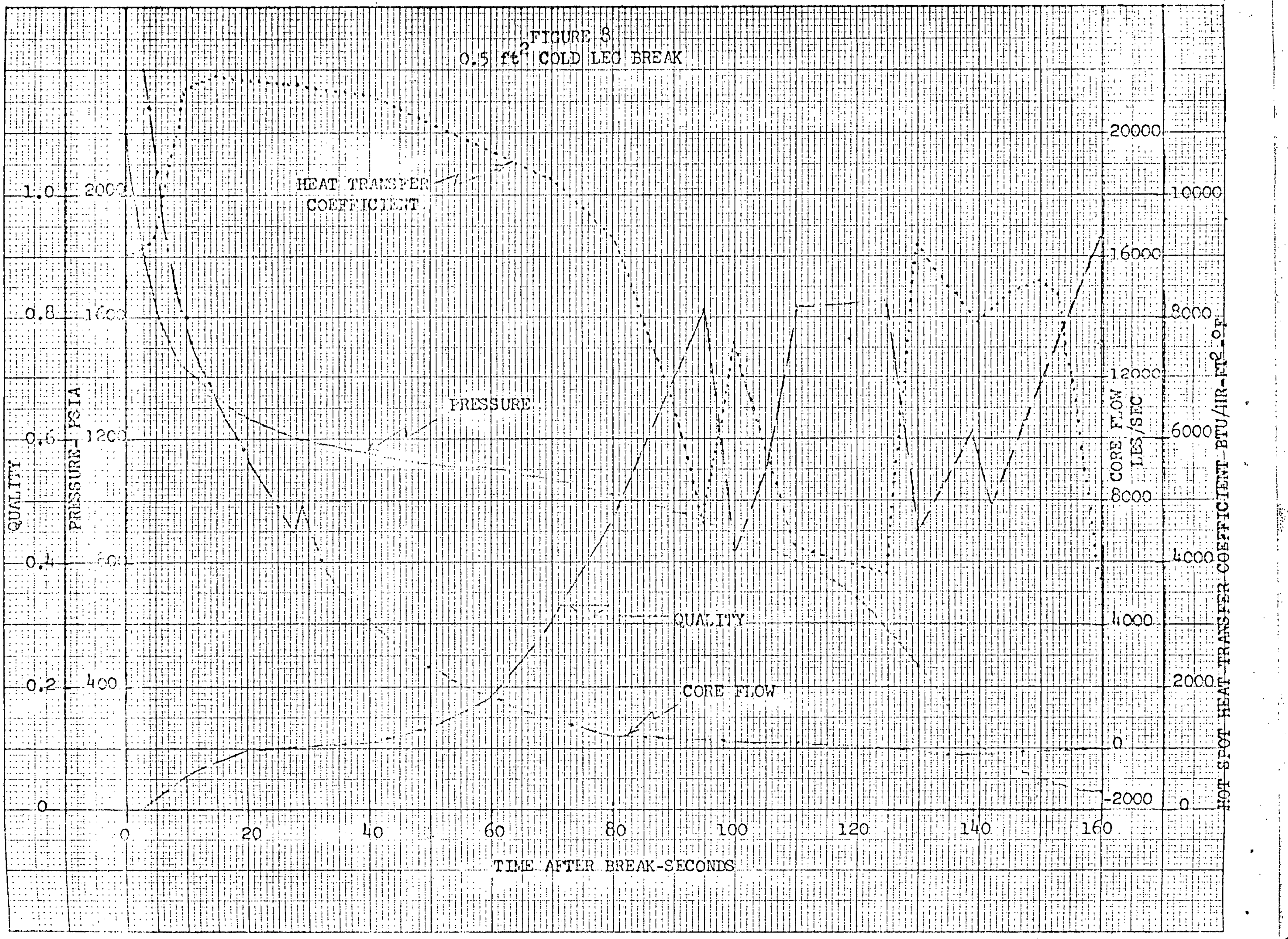


FIGURE 8
0.5 ft² COLD LEG BREAK
HEAT TRANSFER COEFFICIENT
PRESSURE
QUALITY
CORE FLOW
QUALITY
PRESSURE - PSIA
CORE FLOW LBS/SEC
HOT SPOT HEAT TRANSFER COEFFICIENT BTU/HR-FT² °F
TIME AFTER BREAK-SECONDS
0
20
40
60
80
100
120
140
160
0
0.2
0.4
0.6
0.8
1.0
400
800
1200
1600
2000
-2000
0
2000
4000
6000
8000
10000
0
4000
8000
12000
16000
20000

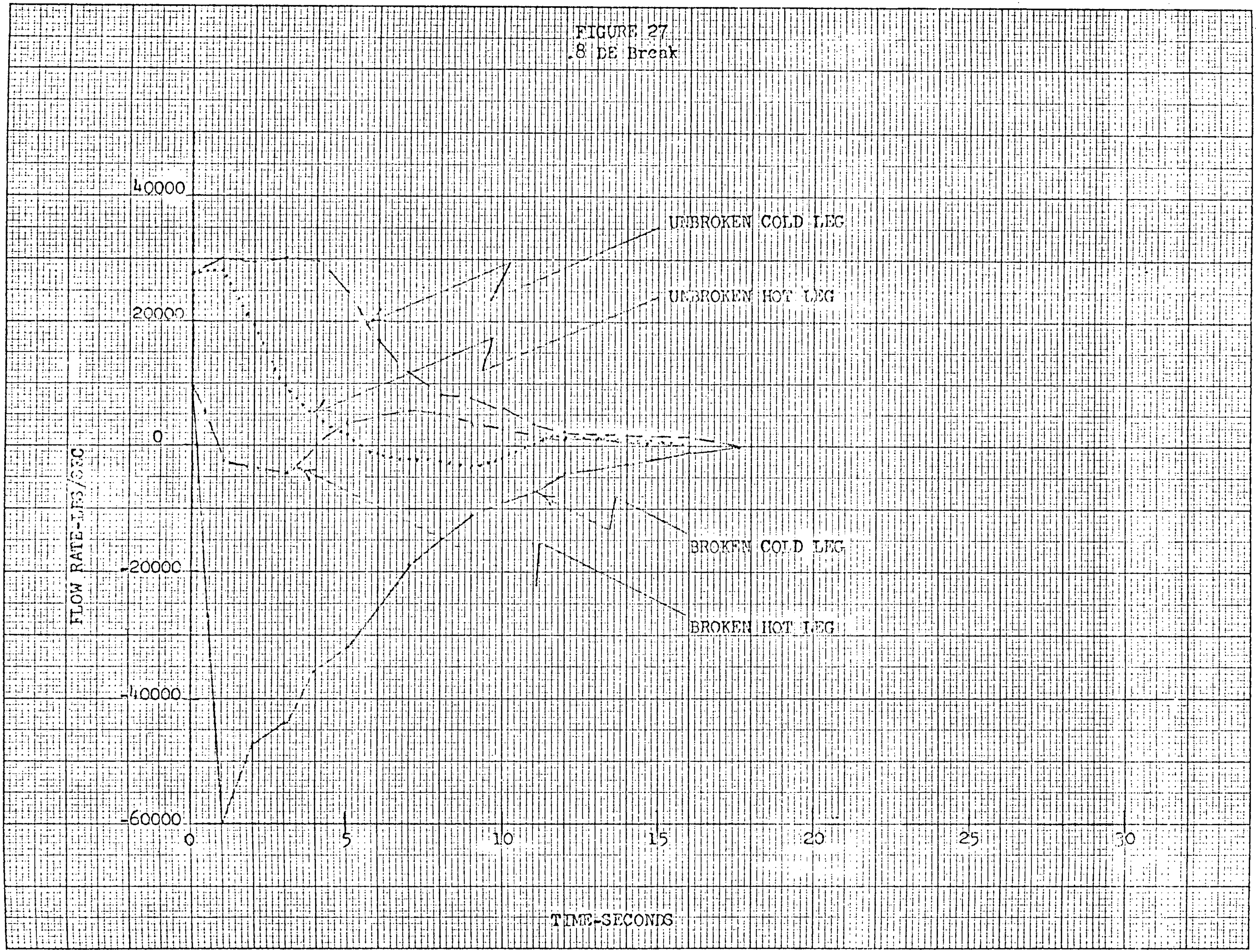


FIGURE 27
.8 DE Break
UNBROKEN COLD LEG
UNBROKEN HOT LEG
BROKEN COLD LEG
BROKEN HOT LEG
40000
20000
0
-20000
-40000
-60000
0
5
10
15
20
25
30
FLOW RATE-LBS/SEC
TIME-SECONDS

Bl wt5

MR. TROSTEN: Yes, we do.

CHAIRMAN JENSCH: The reporter is directed to incorporate into the transcript the exhibit as previously referred to.

MR. TROSTEN: I would now like to call the Board's attention to a document entitled, "Additional Testimony of Applicants Concerning Reactor Vessel Integrity" dated September 17, 1971. A copy of this document was furnished to the Board and to the parties by a letter from Applicant's counsel dated September 18, 1971. This document is to be sponsored in evidence today by a panel of seven witnesses. I might call the Chairman's attention to the fact that we have added an additional sponsoring witness to the list previously indicated to the Board by my letter of September 18th.

The panel of sponsoring witnesses consist of the following persons: Mr. Bernard F. Langer, Mr. Warren S. Hazelton, Mr. Michael J. Manjoine, Mr. Roland J. VonOsinski, Mr. Noel T. Dressel, and Mr. Robert A. Weisemann of the Westinghouse Electric Corporation, and Mr. John J. Grob of the Consolidated Edison Company.

Mr. Weisemann and Mr. Grob have previously been sworn, but I will now ask the other panel witnesses stand and be sworn.

CHAIRMAN JENSCH: Will each of the gentlemen

BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

In the Matter of	)
Consolidated Edison Company of New York, Inc.	) Docket No. 50-247
(Indian Point Station, Unit No. 2)	)

ADDITIONAL TESTIMONY OF APPLICANT CONCERNING REACTOR VESSEL INTEGRITY

September 17, 1971

1. Introduction

The Board has asked (Ref. Tr 1658, 7/16/71) that additional information be provided that supports Applicant's conclusion that the Indian Point Unit No. 2 reactor vessel will not fail (i.e., rupture).

The scope of this information encompasses not only the design, fabrication, and inspection requirements imposed by the 1965 Edition of Section III of the ASME Code, the 1965 Summer Addenda and Code Cases, and the Westinghouse equipment specification in accordance with which this work was carried out, but also the operational requirements (including provisions for in-service inspection) imposed by the Technical Specifications, the special analyses carried out above and beyond Code requirements to demonstrate that failure will not occur, the industry-wide experience that supports the conclusion that failure will not occur, and the systems, components and procedures that provide assurance that the vessel will, in fact, be operated in accordance with the Technical Specifications.

In developing the information requested by the Board, Applicant is presenting not only what was done, but, more important, the significance and reason for what was done, all of which leads to the conclusion that this reactor vessel will not fail. The information is presented in accordance with the following outline:

Section 2	Assurance Provided by Design in Accordance with ASME Code and the Equipment Specification
Section 3	Assurance Provided by Compliance with ASME Code and Equipment Specification Materials, Fabrication, and Inspection Requirements

Section 4 Assurance Provided by Operation in Accordance with the Technical Specification

Section 5 Assurance Provided That Brittle Failure Will not Occur

Section 6 A Discussion of Failures in High Pressure Steam Piping

Section 7 Summary and Conclusions

Appendix A Steps taken in the manufacture of the Indian Point Unit No. 2 reactor vessel that resulted from compliance with ASME Code Section III and the Equipment Specification

B Systems, Components, and Procedures Provided to Assure Operation in Accordance with the Technical Specifications (a discussion of what is provided, how it will function, what it will accomplish, and why there is assurance it will continue to function as required)

C Recommendations of Pressure Vessel Research Committee (PVRC) -- Toughness Requirements for Ferritic Materials (the recommendations upon which the calculations of safety margins presented in Section 5 are based)

D Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2 (a discussion of the philosophy and history of the development of Section III)

2. Assurance Provided by Design in Accordance with the ASME Code and the Equipment Specification

The Indian Point Unit No. 2 Reactor Vessel was designed and fabricated in accordance with the requirements of the 1965 Edition of Section III of the ASME Code, the 1965 Summer Addenda and Code Cases, as implemented and augmented by the requirements of the Westinghouse equipment specification. It was so designed and fabricated by Combustion Engineering and confirmed by review and approval by Westinghouse.

The Applicant's conclusion that the Indian Point Unit No. 2 reactor vessel will not fail is based on the combination of the extensive precautions which have been taken in the conception, design, manufacture, examination and testing of the vessel and the precautions which will be taken in its operation.

A large part (but not all) of the combined precautions which have been and will be taken are those required by Sections III (Nuclear Vessels and XI (In-service Inspection) of the ASME Code. Prior to the growth of the nuclear power industry, vessels were constructed to Section I (Power Boilers) or Section VIII (Pressure Vessels) of the ASME Code or equivalent rules in other countries. These Codes gave formulas for the major, widely distributed stresses and assigned allowable values to them which had sufficient margin to cover the localized stresses, which were not calculated at all. Qualification of manufacturers, quality of material and workmanship, and final inspection were controlled on a statistical basis which tended to provide, but did not always ensure that the Code rules were followed. Service conditions were not controlled at all except insofar as maximum allowable pressure and temperature were concerned. These manufacturing requirements were not as rigorous as those employed in the manufacture of the Indian Point reactor vessel but the results were good.

Section III, in accordance with which this vessel was manufactured, together with controls over operation, provides a combination of tighter controls which eliminates the probability of failure.

These improved controls provided by Section III consist of:

(1) Accurate calculation of both general and local stresses and hence accurate definition of safety margins. Allowable stress values are used for primary, secondary, and local stresses which take account of the significance of each stress category relative to the possibility of failure. In addition, use is made of the Tresca (maximum shear stress) theory of failure, which is known to be more accurate than the maximum stress theory used by older Codes.

(2) Calculation of thermal stresses and the imposition of allowable limits for them. Thermal stresses were given less consideration in

older Codes, but it is known that they can seriously affect fatigue and progressive distortion. Section III provides limits on thermal stress which are based on conservative application of the principles of Limit Analysis.

(3) Analysis for cyclic operation and quantitative evaluation of the safety margin against fatigue failure. Much better data are now available on low-cycle fatigue, in the area of both crack initiation and crack growth than were known before Section III was written. This new data is used as described in Section 5 to provide quantitative information on safety margins.

(4) Detailed listing of the service conditions, including both normal operating cycles and postulated accidents. Thus a plan for operation throughout the life of the vessel is available and the prevention of abuse during service does not depend on the mere limitation of maximum pressure and temperature.

(5) Tighter controls on the quality of materials and workmanship. Section III imposes many supplementary materials requirements more restrictive than those in the basic materials specifications and many nondestructive examination requirements more stringent than are found in older Codes. For example, Section III requires ultrasonic examination of all reactor vessel plates and Section XI requires periodic ultrasonic mapping of the strength welds.

(6) Quantitative evaluation of the margin of safety against failure by brittle fracture, as described in the PVRC recommendations.

The significance of those paragraphs of the Code pertaining directly to the design of the reactor vessel is summarized below and presented in detail in Appendix D, Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analyses. This information is pertinent to the Board's inquiry about the features of the ASME Code and the design basis that provide the high degree of assurance.

Paragraphs N-110, N-130 and N-150

These paragraphs are closely related and will be considered as one subject whose significance may be itemized as follows:

(a) The definition of the scope of Section III as covering the construction requirements for materials, design fabrication, inspection, testing and certification of nuclear vessels.

(b) The Code rules provide minimum safety requirements for new construction and for mechanical and thermal stresses due to cyclic operation.

(c) The Code rules do not cover deterioration in service due to radiation, instability of material, or mechanical shock or vibratory loading. (These are covered by additional requirements imposed by the equipment specification and the Technical Specifications which are covered in more detail in Section 5).

(d) The Code recommends a surveillance program to check periodically the effect of neutron irradiation on the brittle fracture transition temperature.

(e) Defines a reactor vessel as a Section III Class A vessel thus imposing the most stringent requirements of the Code.

(f) Defines the boundaries of vessels and their appurtenances intended to be covered by this Code.

The ASME Boiler and Pressure Vessel Code has for many years successfully provided rules for construction of boilers and pressure vessels. The requirements of Section I (Power Boilers) and Division 1 of Section VIII (Pressure Vessels) do not call for a detailed stress analysis but merely set the wall thickness necessary to keep the basic hoop stress below the tabulated allowable stress. They do not require a detailed evaluation of the higher, more localized stresses which are known to exist, but instead allow for these by the safety factor and a set of design rules.

The simplified procedures of Division 1 of Section VIII are for the most part conservative for pressure vessels in conventional service and a detailed analysis of many pressure vessels constructed to the rules of Division 1 of Section VIII would show where the design could be optimized to conserve metal. However, it is recognized that the designer may be required to provide additional design considerations for pressure vessels to be used in severe types of service such as vessels for highly cyclic types of operation and for services which require superior reliability. The need for design rules for such vessels led to the preparation of Section III.

Reference: "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2". The American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017. (See Appendix D).

Paragraphs N-410, 411, 413, 414, 415, 416, 417, 430 and 448, Fig. N-414 and Tables N-421 and 424

These paragraphs are all closely related and will be treated as basically one subject. These paragraphs, figures and tables are primarily concerned with the requirements for the acceptability of a design with respect to the defined stresses being within the allowable limits prescribed. Developments of analytical and experimental techniques has made it possible to determine stresses in considerable detail. Allowable stress values are prescribed for primary, secondary and local stresses which take into account the significance of each stress category. A calculated value of stress means little until it is associated with its location and distribution in the structure and with the type of loading which produced it. Different types of stress have different degrees of significance and must, therefore, be assigned different allowable values. For example, the average hoop stress through the thickness of the wall of a vessel due to internal pressure must be held to a lower value than the stress at the root of a notch in the wall. Likewise, a thermal stress can often be allowed to reach a higher value than one which is produced by dead weight or pressure.

Section III of the Code utilizes the developments of numerical and experimental stress analysis methods to permit "design by analysis" rather than "design by formula." As a result, the design criteria of Section III differ from those of Section I and Division 1 of Section VIII in the following respects:

(a) Section III uses the maximum shear stress (Tresca) theory of failure instead of the maximum stress theory.

(b) Section III requires the detailed calculation and classification of all stresses and the application of different stress limits to different classes of stress, whereas Section I and Division 1 of Section VIII give formulas for minimum allowable wall thickness.

(c) Section III requires the calculation of thermal stresses and gives allowable values for them, whereas Section I and Division 1 of Section VIII do not.

(d) Section III considers the possibility of fatigue failure and gives rules for its prevention, whereas Section I and Division 1 of Section VIII do not.

The stress limits of Section III are intended to prevent three different types of failure, as follows:

(a) Bursting and gross distortion from a single application of pressure are prevented by the limits placed on primary stresses.

(b) Progressive distortion is prevented by the limits placed on primary-plus-secondary stresses. These limits assure shakedown to elastic action after a few repetitions of the loading.

(c) Fatigue failure is prevented by the limits placed on peak and alternating stresses.

Paragraph N-416 of Section III, 1965 Edition, gives rules for allowable stresses and analysis of bolting. Since the publication of the 1965 Edition, studies[1] have demonstrated that these rules are excessively conservative [See Fig. 10 of ref. (1)] and subsequent editions of Section III have acknowledged this by adopting the results of their studies.

Paragraph N-417 of Section III contains special stress limits to cover special operating conditions or configurations. Some of these deviations are less restrictive and some more restrictive than the basic stress limits. In cases of conflict, the special stress limits take precedence for the

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[1] A. L. Snow and F. F. Langer "Low-cycle Fatigue of Large Diameter Bolts" Journal of Engineering for Industry, February 1967

particular situations to which they apply. The common coverage of the Code includes:

(a) A modified Poisson's ratio value to be used when computing local thermal stresses.

(b) Provisions for waiving certain stress limits if a plastic analysis is performed and shakedown is demonstrated.

(c) Provisions for Limit Analysis as a substitute for meeting the prescribed basic limits on local membrane stresses and on primary membrane plus primary bending stresses.

(d) A limit on the sum of the three principal stresses.

(e) Special rules to be applied at the transition between a vessel nozzle and the attached piping.

(f) Requirements to prevent thermal stress ratchet growth of a shell subjected to thermal cycling in the presence of a static mechanical load.

(g) Requirements to prevent progressive distortion on non-integral connections.

In addition, Paragraphs N-417.1 and N-417.2 provide rules for Bearing Loads and Pure Shear, respectively.

The first three of these special rules and the rules associated with item (f) above provide recognition of the growing significance of plastic analyses to the evaluation of pressure components. The shakedown analysis provides a means whereby the limit on primary plus secondary stress limits may be exceeded. This particular limit is the one with which most difficulty has been experienced in vessels subject to severe thermal transients. Unfortunately, the slow progress in developing practical methods of shakedown analysis has made this provision difficult to apply, and alternate methods are under study.

The Limit Analysis provision is essential when evaluating formed heads of large diameter to thickness ratio. Such heads develop significant hoop compressive stresses and meridional tensile stresses in the knuckle regions over an area which makes it unappropriate to apply the rules for classification as local membrane stresses.

Tables N-421 and N-424 list values of allowable stresses and yield strengths based on test data in support of the basis for stress criteria discussed previously.

Table N-421 gives design stress intensity values grouped according to temperature, and in every case the temperature is understood to be the actual metal temperature.

The values are obtained by applying factors to the mechanical properties of the materials. Consideration is given to the minimum properties specified, and the properties at various temperatures as determined by tests on specimens of the material. On all materials, and especially those whose properties are affected by heat treatment, the data for establishing allowable stresses are from test specimens that are representative of the material supplied in accordance with the specification including the specified heat treatment.

The mechanical properties considered, and factors applied for Table N-421 are as given below. At any temperature listed, the design stress value for ferritic steels and non-ferrous metals and alloys except those covered below is the lowest of these:

(1) 1/3 of the specified minimum tensile strength at room temperature,

(2) 1/3 of the tensile strength at temperature,

(3) 2/3 of the specified minimum yield strength at room temperature,

(4) 2/3 of the yield strength at temperature.

The design stress for austenitic steels, nickel-chromium-iron alloys and nickel-iron-chromium alloys is the lowest of these:

(1) 1/3 of the specified minimum tensile strength at room temperature,

(2) 1/3 of the tensile strength at temperature,

(3) 2/3 of the specified minimum yield strength at room temperature,

(4) 90 percent of the yield strength at temperature but not to exceed 2/3 of the specified minimum yield strength.

The tensile strength and yield strength at temperature used in applying the above criteria to obtain the stress-intensity values in Table N-421 are obtained from data collected by the ASME Boiler and Pressure Vessel Committee over many years, including assistance from the Metals Properties Council. The method used was to determine trend curves of strength properties versus temperature and then reduce these trend curves to correspond to the minimum acceptable properties for each specification as shown in Figure 2-1.

Paragraphs I-610, 612, 620 and 622

These paragraphs provide for the conservative design of nozzles by using the ratio of the stress components under consideration to the computed membrane stress intensity in the unpenetrated and unreinforced vessel material.

Detailed information concerning the validity of this design method may be found in the following documents:

(1) Mershon, J. L., "PVRC Research and Reinforcement of Openings in Pressure Vessels", Welding Research Council Bulletin No. 77, (May, 1962).

(2) Mershon, J. L., "PVRC Interpretive Report of Pressure Vessel Research, Section I, Design Consideration, Part 1.6, Reinforcement of Openings Under Internal Pressure", Welding Research Council Bulletin No. 95 (April, 1964).

(3) Taylor, C. E. and Lind, N.C., "Photoelastic Study of the Stresses Near Openings in Pressure Vessels", University of Illinois, T & A.M. Report No. 270, (March, 1965).

(4) Leven, M. M., "Photoelastic Determination of the Stresses in Reinforced Openings in Pressure Vessels", Westinghouse Research Laboratories Research Report 64-9D7-541-R1, (October 30, 1964).

(5) Riley, W. F., "Experimental Determination of Stress Distributions in Thin-Walled Cylindrical and Spherical Pressure Vessels with Circular Nozzles", Welding Research Council Bulletin No. 108, pages 1-11, (September, 1965).

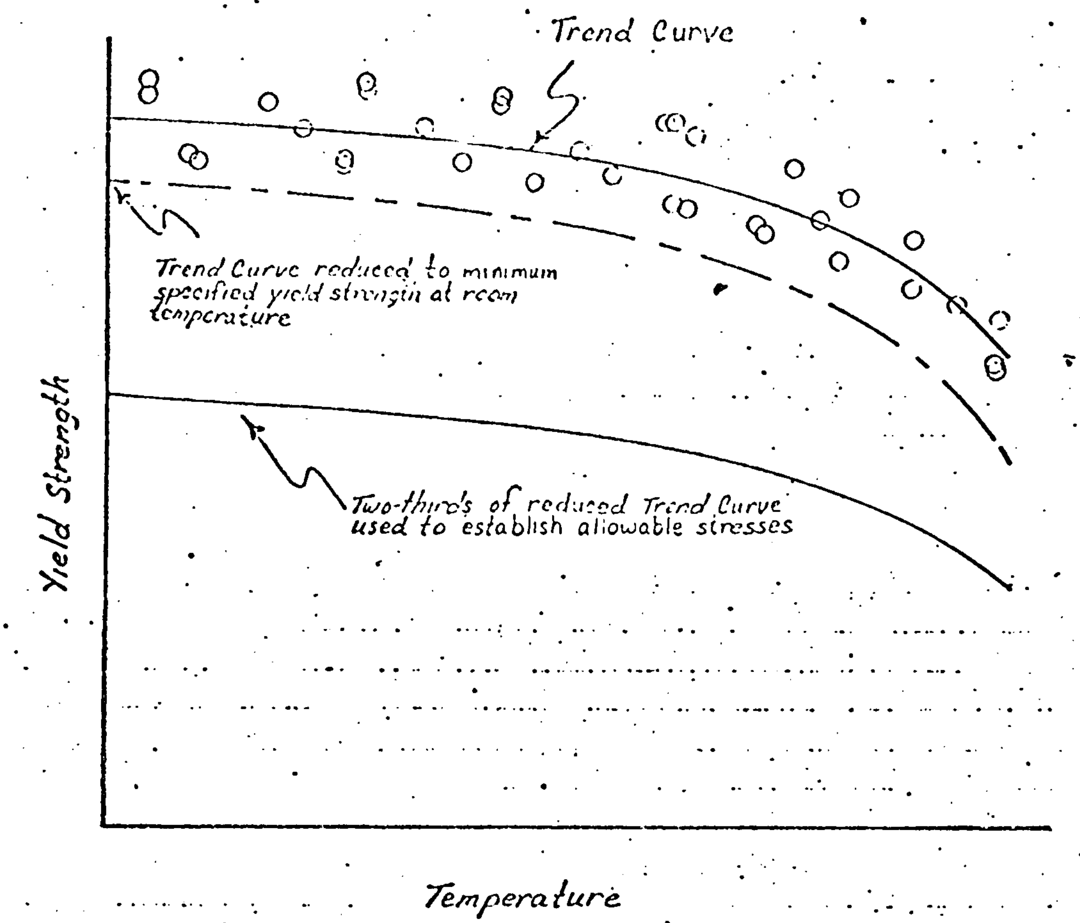
(6) Maxwell, R. L., Holland, R. W., and Cofer, J. A., "Experimental Stress Analysis of the Attachment Region of Hemispherical Shells with Radially Attached Nozzles," University of Tennessee, Engineering Experiment Station, Knoxville, Tennessee, Report ME-7-65-1.

(7) Lind, N. C., Hradek, R. W., and Cook, R. D. "Influence of Fillet Radii on Stresses Near Outlets in Pressure Vessels", University of Illinois, T. & A.M. Report No. 167 (March, 1961)

(8) Mershon, J. L., "Preliminary Evaluation of PVRC Photoelastic Test Data on Reinforced Openings in Pressure Vessels", Welding Research Council Bulletin No. 113 (April, 1966) pp. 53-70. This bulletin also contains references 3 and 4 above.

The drawings for the reactor vessel, and the stress analysis report were reviewed by Westinghouse for compliance with Code and equipment specification requirements and approved.

As a consequence, because of the extensive and technically sound requirements imposed on the design of the vessel in accordance with the Code, because the significance of these requirements was known and understood so that they could be implemented properly in the design, and because evidence of compliance was obtained, there is assurance that the Indian Point Unit No. 2 reactor vessel will not fail by overstress, creep rupture, or in fatigue.



References: "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2."

Figure 2-1

3. Assurance Provided by Compliance with ASME Code and Equipment Specification, Materials, Fabrication, and Inspection Requirements

In designing the Indian Point Unit No. 2 reactor vessel, it was assumed that the materials to be used, and the fabrication, and inspection techniques employed would conform to the requirements of the 1965 Edition of Section III of the ASME Code, the Summer 1965 Addenda and Code Cases, as implemented and augmented by the Westinghouse equipment specification.

Section III of the ASME Code requires that the materials employed shall have tensile properties that conform to applicable materials specifications (N-310). It further requires that the materials meet minimum impact test requirements (N-330). It defines the non-destructive testing requirements for the materials to be employed (N-320); fabrication techniques including forming, welding, and heat treating (N-520 et sequi); and the inspection and testing requirements and procedures (N-610 et seq). A discussion of what was done that resulted in compliance with the code is presented in Appendix A. The Westinghouse equipment specification implements and augments the Code. The materials acceptable for use are the fabrication requirements (including requirements for Westinghouse approval of welding and heat treating procedures), the test and inspection requirements (including the conduct of Charpy and drop weight tests, the maintenance of records, and the witnessing of tests, inspections, work in progress, and the review of results by Westinghouse) are specified in detail in thes specification.

Imposing these requirements provides assurance that the materials employed have adequate strength, ductility, and toughness; that the materials employed are free of injurious defects; that good workmanship was employed, and that fabrication was carried out in accordance with Code and design requirements. Imposing these (i.e. the requirement over and above Section III requirements for the conduct of drop weight and Charpy tests on core region plates in equipment specification) also provides the fracture toughness data required to establish operating limits for the reactor vessel that assure that brittle fracture will not occur. In addition, a reactor vessel surveillance program was established which assures that data necessary for continued adequacy of these operating limits throughout the vessel service life will be available.

Procurement of the Indian Point Unit No. 2 reactor vessel in accordance with applicable sections of the Code, and with the Equipment Specification, was monitored by Westinghouse PWR Systems Division Quality Control Department personnel. Evidence of compliance with all requirements is available in the "Inspection Report for the 173 inch I. D. Reactor Vessel and Components", and in voluminous reports and records on file at Combustion Engineering and at Westinghouse.

Compliance with Code and Equipment Specification requirements involved among other things ultrasonic testing of steel plates for the reactor vessel shell and closure heads, and on flange and nozzle forgings prior to fabrication. Subsequently, after hydrostatic testing of the reactor vessel, an ultrasonic mapping of plate forgings and welds was again carried out and permissible indications estimated by ultrasonic testing techniques to be 3-5% of the thickness were identified. The validity of these techniques and results was demonstrated by destructive analysis subsequent to a post hydrostatic test ultrasonic mapping of another vessel. Compliance also involved magnetic particle testing of the base plate for the reactor vessel shell and closure heads in accordance with Westinghouse approved Combustion Engineering Procedure . The relevant linear indications considered by this procedure are those for which 1 = 2W; the Code defines relevant linear indications as those for which 1 = 3W. Magnetic particle inspection was, therefore, carried out in accordance with requirements more stringent than those imposed by the Code.

The reactor vessel fabrication, testing and installation processes are included within the scope of the Applicants' monitoring activities delineated in the Quality Assurance Program Appendix to the FSAR.

In general, the reactor vessel materials, fabrication, testing, and installation were monitored via a surveillance program at the fabricators' shops by an independent agency, the United States Testing Company (USTC), and a monitoring program at the construction site by a Consolidated Edison construction group.

The activities of the Consolidated Edison agent, USTC, have included:

1. Obtaining information from Westinghouse on manufacturing and quality control specification requirements of the reactor vessel.

2. Reviewing Combustion Engineering quality control procedures covering: materials, fabrication processes, nondestructive tests, and associated records.

3. Witnessing hydrostatic pressure test of the reactor vessel.

4. Reviewing, and witnessing reactor vessel nozzle weld overlay, and reveiwing associated documentation.

5. Reviewing documentation on liquid penetrant test of instrument tubes.

The Consolidated Edison site construction group has monitored or participated in the following activities:

1. Witnessing vessel installation.

2. Witnessing ultrasonic and liquid penetrant testing of nozzle weld overlay.

3. Witnessing and reviewing documentation on primary system hydrostatic test including the reactor vessel.

4. Witnessing and conducting hot functional tests of the plant.

As a result of the combined efforts of our agent, USTC, and the Consolidated Edison on-site construction group, Applicant is confident that the vessel has been fabricated, installed, and tested in a manner consistent with applicable code requirements.

By virtue of its compliance with the Code and the equipment specification the Indian Point Unit No. 2 reactor vessel was, therefore, fabricated with materials and by techniques, and inspected, in accordance with extensive technically sound requirements. Evidence of compliance with requirements was obtained and is on file. Thus, there is assurance that the materials employed are well known with extensive experience in their use; they have the properties assumed by the designer; they are free of injurious defects; and good workmanship was employed and fabrication was properly carried out. Hence there is assurance that the Indian Point Unit No. 2 reactor vessel will not fail because of material or fabrication deficiencies.

4. Assurance Provided by Operation in Accordance with the Technical Specifications

The Technical Specifications describe in detail how the plant and vessel will be operated. When operated in this manner, all conditions assumed as design bases will be valid. Another way to state this relationship is that the pressures, temperatures, cycles of operation, etc., permitted by Technical Specifications are used as the basis for design, with a wide safety factor.

Examples of this are:

1) The vessel operates at a maximum temperature of 554.8°F, but the design temperature is 650°F.

2) The vessel, when operated in accordance with the Technical Specifications, will see only a fraction of the fatigue cycles used as the design basis.

3) Design pressure is 2485 psig using the code-imposed safety factors, whereas the operating pressure according to Technical Specifications is 2235 psig.

The conservatism of the Technical Specifications operating limits is discussed in Section 5.

Another function of the Technical Specifications is to control environmental effects inside the reactor vessel. The code cautions that these be taken account, but cannot adequately cover all variables. Therefore, this has to be taken into account by the designer by selecting suitable materials, and the user by means of controlling the environment in accordance with the designer's specifications. The action to control environmental effects is discussed below.

The pressure vessel is basically made of low alloy steels, SA508 Class 2 and SA533 Grade B. These are considered low to medium strength steels, and years of experience in many applications has proven that no stress corrosion, hydrogen embrittlement, or severe corrosion problems result from exposure of these materials to high temperature water or steam. In addition, results of many tests prove that the addition of boric acid and hydroxide at and well above the levels used in PWR systems cause no additional problems. Results of tests on irradiated material, also show that irradiation of these steels does not change their response to corrosion and corrosion-related phenomena.

The vessel is clad with stainless steel to reduce the amount of corrosion products that have to be handled by the clean-up systems, and to reduce the radioactivity caused by such products being circulated through the plant. (The cladding is not necessary for protection of the base metal to assure integrity of the vessel.

Even though cladding defects caused by corrosion or stress corrosion will not affect the integrity of the vessel (stress corrosion cracks in the stainless steel will not progress into the base metal) stringent controls are provided to assure that contaminants detrimental to stainless steel are kept to vanishingly low levels in the primary coolant. This is primarily to assure freedom from problems in the stainless steel in other components, and in minor portions of the reactor vessel, such as nozzle safe ends.

Primary coolant chemistry can be very closely controlled in the closed PWR system. Hydrogen overpressure can also be used to practically eliminate oxygen from the coolant. The control over contaminants provided by the systems furnished for this purpose gives positive assurance that the Technical Specifications limits on undesirable contaminants are met.

In addition to control of contaminants such as fluoride and chloride during normal operation, effects of the environmental conditions during shutdown, highly borated (2500 ppm B) aerated water, on the corrosion of stainless steels and Inconel in both the severely sensitized and non-sensitized condition were evaluated. No adverse effects were found.

For all of the above reasons, there is assurance that the chemical environment in which the reactor vessel will operate will be controlled in accordance with the Technical Specifications, and that the materials of construction will be completely compatible with this environment. As a consequence, there is assurance that the integrity of the reactor vessel will not deteriorate in the environment in which it is to operate.

5. Assurance Provided That Brittle Failure Will Not Occur

As shown in preceding sections, the vessel is designed according to technically proven, sound, conservative methods to perform satisfactorily under all conditions to which it will be subjected in operation. Assurance that the actual vessel is made in conformance with this design and that the designers' assumptions as to material properties and freedom from injurious defects is provided by Code, Westinghouse, and Combustion Engineering quality assurance procedures and records.

Further, the chemical environment to which the vessel is exposed is known not to cause degradation of properties or the development of defects not originally there.

Safety margins were evaluated using the latest methods of failure analysis by calculating the expected conditions that cause failure, and comparing these conditions with the actual conditions that will be seen by the vessel. This section describes such evaluations.

Failure of the pressure vessel must be considered for the two possible failure modes: ductile yielding mode and brittle fracture mode. A third possible failure mode, fatigue, must be also considered. However, while ductile yielding and brittle fracture can occur independent of fatigue, the fatigue mode failure must be in conjunction with either of the above two possible failure modes. That is, during fatigue, a crack may initiate and ductile yielding may occur; or a crack may initiate or be present and grow to a critical size where the remaining ligament fails under load limiting ductile failure; or a crack may initiate or be present and grow during fatigue from subcritical to critical crack size resulting in brittle fracture. In general, the term "brittle fracture" describes a fracture which occurs suddenly, without warning, and propagates radially to a point at which the integrity of a structure is seriously impaired.

Present design against brittle fracture of reactor pressure vessels is based on the transition temperature philosophy (Fracture Analysis Diagram, etc.). Although the transition temperature approach is deemed conservative when applied to reactor pressure vessels, it does not provide the basis for quantitative evaluation of the many aspects involved in a thorough evaluation of brittle failure potentials. A much more rigorous and sophisticated method of analyzing large, heavy-walled pressure vessels of the type employed in the nuclear industry to determine the safety margin relative to brittle failure has therefore been developed. Linear-elastic fracture mechanics provides a quantitative approach to brittle failure potentials, and therefore can be used to evaluate safety margins. Linear-elastic fracture mechanics (LEFM) is basically a stress intensity consideration in which criteria are established for fracture instability in the presence of a crack. Consequently, a basic assumption in employing LEFM is that a crack or crack-like defect exists in the structure. The essence of the approach is to relate the stress field developed in the vicinity of the crack-tip to the applied nominal stress on the structure, the material properties, and the size of defect necessary to cause failure. From the inspections performed on the Indian Point Unit 2 reactor vessel, with the fatigue crack growth data, the fracture toughness properties, and the transients to which this vessel may be subjected, the largest flaw present is not of and will not grow to a critical size. This is discussed below.

Nuclear reactor pressure vessels are designed and manufactured to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Section III of the ASME Code requires that the supplier perform a complete fatigue analysis of the vessel for all specified standard design transients. The fatigue analysis is performed using Miner's hypothesis of linear cumulative damage in conjunction with fatigue data from constant stress or strain fatigue tests. The cumulative usage factor, defined as the sum of the ratios of the number of cycles of design transients (n) to the allowable number of cycles for the stress range associated with the transient (N), must not exceed one.

The design transients which are used to evaluate the integrity of the pressure vessel in a Westinghouse Nuclear Steam Supply System are given in Table 5-1. The transients in Table 5-1 are classified as (1) normal conditions, which include any condition expected to occur in the course of system startup, operation, and shutdown; (2) upset conditions, which are deviations from normal conditions anticipated to occur often enough that the design should include capability to withstand them; (3) test conditions, which occur in the course of testing the system both prior to and following initial startup;

and (4) faulted conditions, which refer to those combinations of conditions associated with extremely low probability postulated accidents whose consequences are such that the integrity of the nuclear energy system may be impaired to the extent where considerations of public safety are involved.

As stated above, a complete fatigue analysis of the pressure vessel is performed using Miner's hypothesis. This method of fatigue analysis has been the most widely used to assure safe operation under cyclic loading conditions. Thus, a complete fatigue analysis of the Indian Point Unit No. 2 reactor vessel has been performed by Combustion Engineering using Miner's hypothesis. In addition, Westinghouse performed a fatigue evaluation of the reactor vessel using a fracture mechanics approach. The fatigue crack growth analysis utilizing fracture mechanics was performed on the most critically-stressed location of the reactor vessel. As noted in Section 3, a conservative estimate of the largest flaw present in the reactor vessel would be 5 per cent of the wall thickness (outlet nozzle, 0.55 inches; closure head, 0.46 inches; vessel shell, 0.43 inches; and the lower head, 0.25 inches).

For the fatigue crack growth analysis utilizing the fracture mechanics concept, a conservative (at least a factor of two, more likely a factor of 4) initial flaw size of 1 inch in depth was assumed to be present. The results of the fatigue crack growth analysis show that an initial assumed flaw of 1 inch would only grow 0.37 inches in the 40 year life of the reactor vessel. The maximum growth of 0.37 inches would occur at the outlet nozzles. For the beltline region which is considered the most critical region from the standpoint of irradiation embrittlement of the material, the fatigue crack growth was only 0.012 inches in the 40 year life of the vessel. It was apparent from the fatigue analysis of the pressure vessel that fatigue crack growth of flaws up to 1 inch in depth is not one of the major factors controlling the vessel integrity. Thus, the vessel will not fail in the fatigue failure mode.

The Technical Specifications for the plant provide temperature and pressure limitations to assure that brittle fracture will not occur. These were developed using the very conservative assumptions of the AEC proposed rules as subsequently published in the Federal Register, July 3, 1971, and Westinghouse criteria where they are even more conservative. They represent the most conservative limits, as a function of temperature.

Despite the conservative nature of these approaches, they do not give quantitative measures of safety margins. Recently the Pressure Vessel Research Committee (PVRC) has issued recommendations on fracture toughness (Appendix C) that include methods for determining safe operating limits. These methods are quantitative, and so can be used directly to compare operation with conservatively determined failure conditions.

Figure 5-1 shows the limits of operation based on the Technical Specifications, from the first refueling. Temperatures will be shifted after additional irradiation, in line with expected changes in material properties, as indicated in the Technical Specifications.

As can be seen, the lower lines are the upper limits permitted. For reference, the allowable pressure-temperature curve recommended by the PVRC is also shown. The PVRC criterion explicitly has a safety factor of two already incorporated in this curve. So, it can be shown that at the temperature for full pressurization, a 35°F margin exists, and further at this temperature, a margin of 2 on pressure exists. Even greater margins are shown for lower temperatures.

Actually, more margin is available than indicated by using the PVRC criterion directly. The PVRC assumption as to flaw size is extremely conservative. It assumes that a flaw one quarter of the wall thickness and six times this depth in length exists in the most critical orientation. As discussed above, there is assurance that no flaws larger than about 1/2 inch can be present in this vessel. If the methods of the PVRC criterion are used to evaluate the effect of a flaw 0.6" deep, one finds that there is an additional factor of safety of two on pressure, giving a total factor of safety of 4 on pressure, using a realistic flaw size.

If it can be shown that the pressure vessel integrity will be maintained during a faulted condition such as a steam pipe break or a reactor coolant pipe break, then it follows that the vessel's integrity will be maintained during normal, upset, and test conditions.

In the event of reactor coolant pipe break, the Reactor Coolant System will be rapidly depressurized and the loss of coolant may partially empty the reactor vessel. If the reactor was at normal operating conditions before the accident, the reactor vessel will be hot; and if the plant has been in operation for some time, the beltline region of the reactor vessel will be irradiated. During the depressurization transient, the Emergency Core Cooling System will rapidly inject cold coolant into the reactor vessel. This will thermally shock the hot vessel, resulting in high thermal stresses. A comparison of the material yield stress to calculated stresses in the vessel wall demonstrates that 82 per cent of the wall thickness remains below the minimum ASME Section III material yield strength (50,000 psi) at all times during a safety injection transient. If the vessel has been in operation for some time, the complete wall thickness will be below the actual material yield strength. However, using the conservativism of the minimum yield strength (50,000 psi), local yielding may occur only in the inner 18 percent of the metal and in the cladding. Therefore, the vessel will not fail due to ductile yielding mode.

A rigorous analysis of the reactor coolant pipe break problem was performed using linear-elastic fracture mechanics. The detailed analysis is summarized in Attachment 1. It shows that under the postulated accident conditions the integrity of the reactor vessel will be maintained throughout the life of the plant.

An analysis of the steam pipe break problem was also performed using linear elastic fracture mechanics. In this analysis, subsequent to the break it is assumed that the reactor coolant temperature is reduced to 320°F and that there is no return to power. The safety injection pump flow is conservatively assumed to restore system pressure to 2250 psia. The most severe combination of vessel temperature and stress which the pressure vessel might encounter are assumed. The case of the break occurring at no load conditions was considered, as it is more severe than if the break occurs at full power.

The result of the analysis is shown in Figure 5-3, and the expected material properties are shown in Figure 5-2. Figure 5-2 is developed using the very conservative PVRC method of estimating the fracture toughness of the material as a function of distance through the wall at end of life.

Figure 5-3 shows the stress intensity (K_I) that would be imposed in a continuous crack of the depth indicated, considering the instantaneous temperature at that depth. It also shows the K_I available by the toughness of the material considering radiation effects at end of life. It is clear that much more toughness is available than necessary to prevent failure. This represents calculations at the worst condition during the transient, and the worst type of flaw imaginable to the specified depth.

As a consequence, there is assurance that the reactor vessel will not fail in the ductile yielding or brittle fracture modes.

TABLE 5-1

SYSTEM DESIGN TRANSIENTS

NORMAL CONDITIONS

- Heatup and Cooldown
- Plant Loading and Unloading
- Step Change In Power
- Steam Dump (Large Step Load Decrease)
- Steady State Fluctuations

UPSET CONDITIONS

- Loss of Load
- Loss of Flow
- Loss of Power
- Reactor Trip

TEST CONDITIONS

- Turbine Roll Test
- Cold Hydrostatic Test
- Hot Hydrostatic Test

FAULTED CONDITIONS

- Reactor Coolant Pipe Break
- Steam Pipe Break

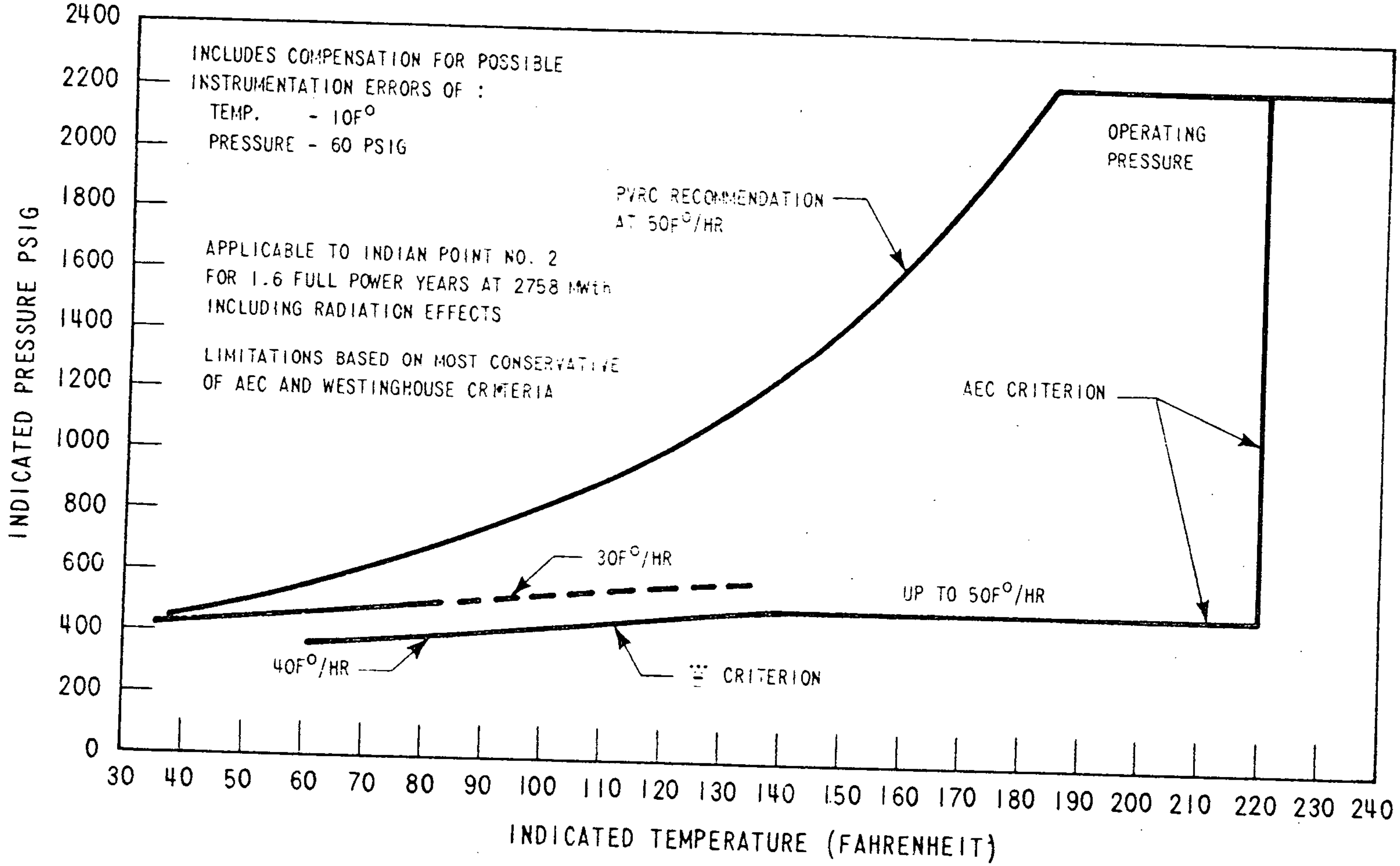


Figure 5-1. Reactor Coolant System Cooldown Limitations

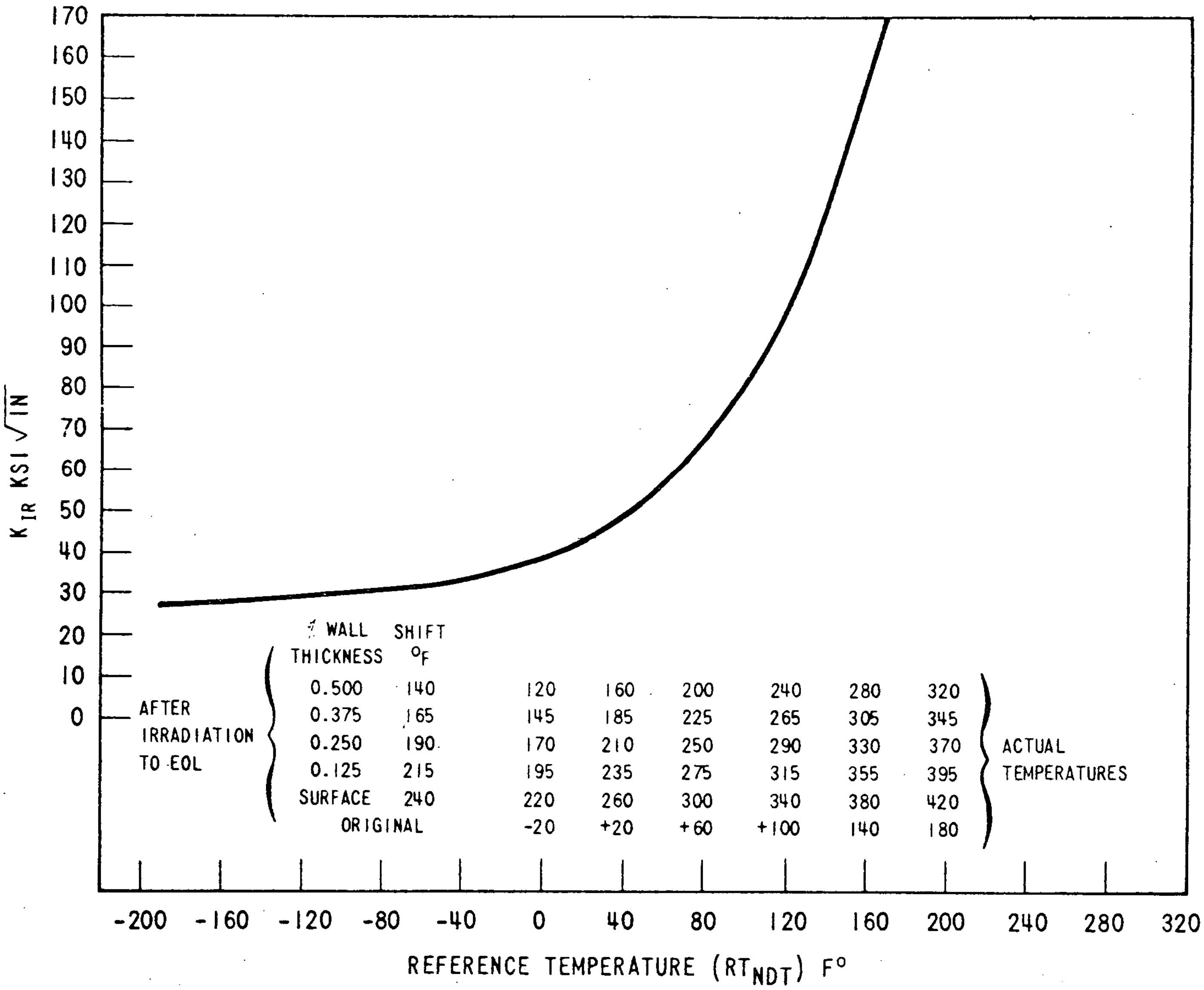


Figure 5-2. Estimated Material Toughness Thru the Vessel Wall at End Of Life Based on Highest Copper Content Plate in Indian Point Vessel

2710-6

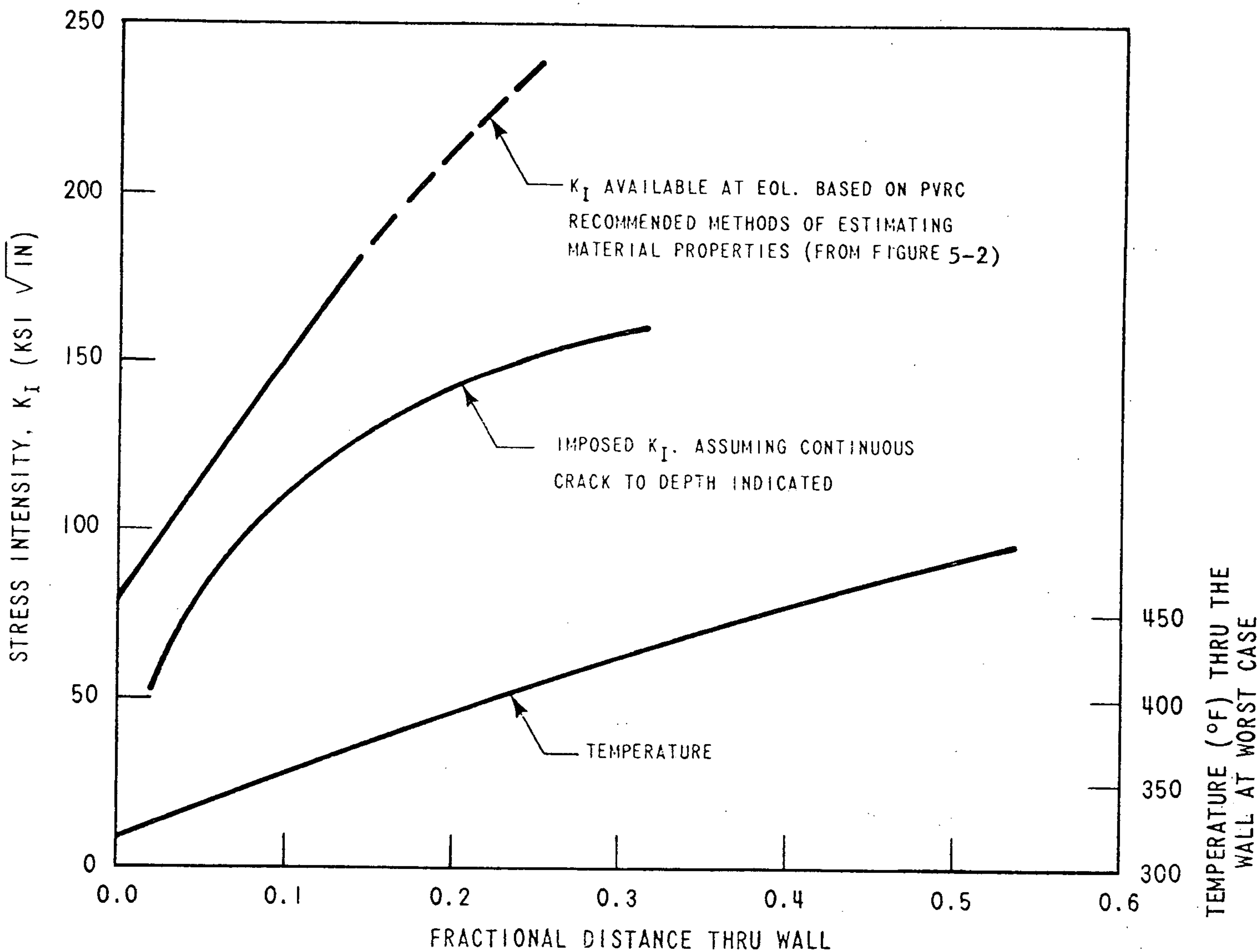


Figure 5-3. Steam Break from No Load Condition

2710-4

Section 5 - Attachment 1

A fracture mechanics analysis of the effects of safety injection water into the reactor pressure vessel following a postulated loss of coolant accident has shown that the integrity of the vessel is maintained.

With the postulation of a loss-of-coolant accident in the reactor coolant system and the subsequent automatic injection of water to prevent overheating of the core, the reactor vessel would experience a thermal transient that causes large thermal stresses to be produced in the vessel wall. The integrity of the reactor vessel is evaluated utilizing the methods of linear elastic fracture mechanics, including the latest available fracture toughness properties of A 533 Grade B reactor vessel steel as a function of temperature and irradiation.

The temperature gradients in the reactor vessel wall were calculated for different time periods following the postulated accident assuming one dimensional (radial) heat flow and utilizing a finite difference computer program to solve the resulting partial differential equation. The thermal stresses resulting from the temperature gradients were then determined from the thermal stress formulae of Timoshenko and Goodier(1).

The fracture mechanics analysis was made to determine the critical crack depths by the method of linear elastic fracture mechanics. Stress intensity factors as a function of assumed crack depth were determined. The resulting stress intensity factors for the assumed cracks or flaws are compared to the best estimate of the fracture toughness properties of the material.

For the analysis the equation for a continuous crack in an infinite media subjected to an arbitrary nominal stress field[(2)] was converted to one which applies to a continuous crack in a semi-infinite media subjected to an arbitrary nominal stress field by applying a symmetry condition and a finite thickness correction factor. The thermal stresses, which were calculated as described above, were applied in the form of third order polynomials, and the stress intensity factor was calculated as a function of assumed crack depth and time.

Fracture toughness data[(3)] has been obtained for A 533, Grade B, Class 1 steel plate and submerged arc weldment material in both the unirradiated and irradiated condition. The data indicate that the fracture toughness properties are highly temperature dependent with a rapid increase in toughness between 0°F and room temperature for the material in the pre-irradiated condition. Post-irradiation data also indicates that the fracture toughness is highly temperature dependent with a rapid increase in toughness at approximately the irradiated NDTT of the material. In the temperature range of interest, the high levels of toughness provide assurance of a high degree of fracture safety in heavy section welded structures.

The results of the analysis with the corresponding fracture toughness properties through the vessel wall at 2000 seconds following the postulated loss-of-coolant accident are shown on Figure 5-A-1. This time is representative of the most severe condition for the transient.

Thus even with the conservative assumptions inherent in the analysis and in the use of linear elastic fracture mechanics for this postulated condition the integrity of the reactor vessel will be maintained throughout the life of the plant.

REFERENCES

1. Timoshenko and Goodier, Theory of Elasticity, McGraw-Hill Book Co., 1951.

2. Paris, P. C. and Sih, G. C., "Stress Analysis of Cracks", ASTM STP-381.

3. Mager, T. R., and Yanichko, S. E., "Use of Fracture Mechanics in Reactor Vessel Surveillance," WCAP-7538, July, 1970.

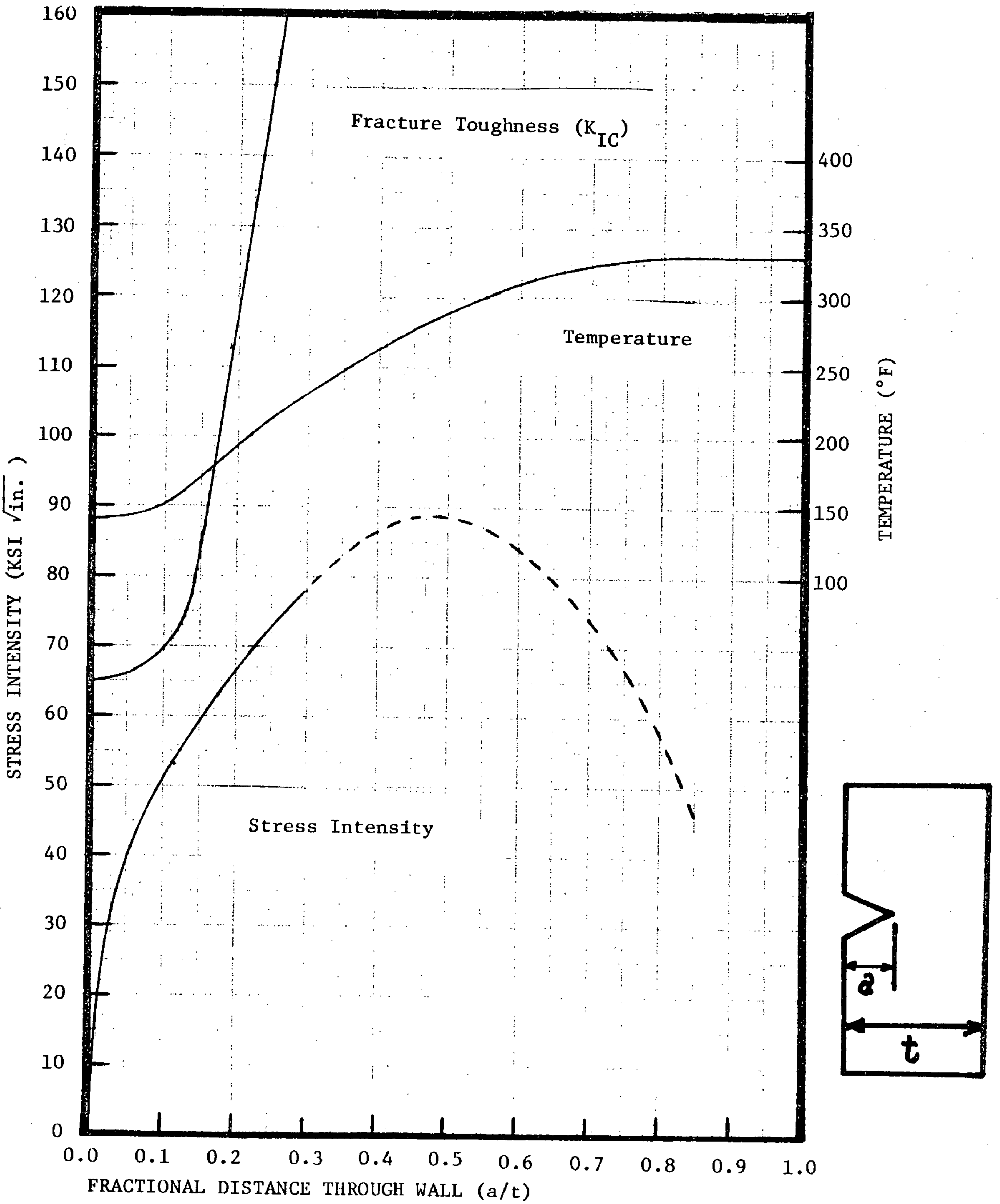


Figure 5-A-1. Stress Intensity and Fracture Toughness Versus Fractional Distance Through Wall at 2000 Seconds

6. A Discussion of Failures in High Pressure Steam Piping

In three large fossile fuel plants there have been failures reported of the high pressure steam piping. Although Applicant is not familiar with the particulars of the systems involved nor with the specifications of their design or installation, two reports on these failures have been published by the ASME in ASME publications 71-PVP-32 and 71-PVP-66. The former report, entitled "Analysis of Cracks in Elbows" covers the investigation of the elbow failures, the elbow repairs and certain welding defects. The later publication, entitled "Analysis of Stresses in Pressurized Welded Pipe in the Creep Range" discusses the stresses involved.

Based upon the reports, it appears that the failures involved occurred in welded elbows made from a newly developed austenitic steel alloy material. The material used in the Indian Point Unit #2 reactor vessel does not utilize that alloy. In addition, a vast amount of information is available for evaluating any possible effects on the reactor vessel material caused by the vessel operating environment or by conditions of fabrication. It is axiomatic, of course, that the design of nuclear pressure vessels is within the scope of ASME Section III code requirements while the design and installation of the high pressure steam piping referred to above is not so governed.

In addition to the above, the H. B. Robinson Unit #2 plant experienced a failure of a pipe nozzle which connected a valve to the 26" steam line in the secondary system. This incident apparently occurred during a start-up test program when pre-operational tests were being conducted. Report of this failure are contained in Incident Report, H. B. Robinson Unit No. 2, Steam Pipe Break, June 1970, and in ROE No. 71-12, Operating Experiences, USAEC. The investigation indicated that the failure in the material utilized occurred as a result of an overload condition.

Because of the design requirements for the Indian Point Unit No. 2 vessel resulting from complying with Section III of the ASME code and the equipment specification and because of the detailed requirements with respect to material specification, fabrication and inspection, and because of the requirements of operation for reactor vessels, and because of the abnormal situations involved in each of the steam piping failures referred to above, the failures in the high pressure steam piping do not lessen the assurances that a failure will not occur in the reactor vessel in Indian Point Unit #2.

7. Summary and Conclusions

The Indian Point Unit No. 2 reactor vessel was designed in accordance with the 1965 Edition of Section III of the ASME Code, the 1965 Summer Addenda and Code Cases, as implemented by the Westinghouse equipment specification. The significance of these requirements which govern the materials employed, and the design, fabrication and inspection techniques embodied in the construction of this vessel have been discussed. These requirements are imposed in order to provide assurance that the reactor vessel will not fail because of material or fabrication deficiencies, and that it will not fail in fatigue, or because of overstress, or creep rupture. All requirements were complied with, and an N-stamp was issued as evidence of compliance in accordance with Paragraph 812 of the Code.

The conservatism inherent in the Technical Specifications which governs the operation of the reactor vessel have also been discussed. Because it will be operated in accordance with this specification, there is assurance that it will be operated in a controlled environment designed to prevent deterioration of vessel integrity in a manner consistent with design assumptions.

Additional analyses which embody the most advanced methods of failure analysis were described which define the safety margins inherent in the design of the reactor vessel. The results of these analyses show that the safety margins are more than adequate and that the vessel will not fail in the ductile yielding or brittle fracture modes.

Failures in high pressure steam piping were of concern to the Board and were evaluated. It was concluded that these failures were atypical, and are not relevant to concerns about the integrity of the Indian Point Unit No. 2 reactor vessel.

Therefore, for all of the above reasons, there is assurance that the Indian Point Unit No. 2 reactor vessel will not fail.

Appendices

APPENDIX A

Steps Taken In The Manufacture Of The Indian Point Unit No. 2 Reactor Vessel That Resulted From Compliance With The ASME Code and The Equipment Specification

APPENDIX A

INTRODUCTION

The requirements for the Indian Point - 2 reactor vessel have taken the form of applicable portions of the ASME Code Section III (1965 Edition), 1965 Summer Addenda and Code Cases 1332, 1335, 1339 and 1359 plus additional requirements established in the Westinghouse Equipment Specification. These additional requirements provide an added assurance of reactor vessel integrity and provide adequate consideration to items not within the scope of the Code. The ASME Code establishes minimum requirements for materials, design, fabrication, inspection and testing. The additional requirements placed by Westinghouse on the reactor vessel manufacture have been in the areas of analysis and inspection. The additional items not within the scope of the Code but required during vessel manufacture were the consideration of mechanical shock loadings and instability of the material. Consideration of radiation effects on the vessel material and the shift in brittle fracture transition temperature to be experienced during the life of the vessel was also required during manufacture; these pre-operational requirements are discussed as part of the overall vessel surveillance program. There follows a discussion of the particular steps taken in the manufacture which provides assurance that the vessel will not fail. For case of reference there is set forth in the left hand column on each page the pertinent Code paragraph. Documents referenced in the attached are on file at manufacturer's facility.

Code Paragraph	Steps Taken
Article 1 General Requirements	
N-110	A reactor vessel surveillance program was established to evaluate the effects of neutron irradiation of materials based on pre-irradiation and post-irradiation testing or speciments. The Technical Specifications indicate examination intervals. Surveillance specimens were prepared from vessel material.
	Instability of material was taken into account in the selection of the vessel material and processes, cladding of surfaces in contact with primary coolant, and provisions for cleanliness throughout manufacturing stages. The equipment specification listed acceptable materials, and requirements for non-destructive testing, cleaning, and plating.
	Mechanical shock loadings due to seismic occurrances or from pipe rupture were established by detailed computer analysis. Vibratory loadings in the analysis were not considered necessary because of the natural frequency of the vessel and support. Vibration measurements on Indian Point Unit No. 2 internals utilizing vessel mounted accelerometers confirmed this decision. The equipment specification listed seismic loads and pipe rupture loads used in the vessel design and analysis.
N-131	The reactor vessel was classified in the equipment specification as a Class A vessel, and therefore there were imposed the most stringent requirements on its design.

Code Paragraph	Steps Taken
N-141	Westinghouse prepared an equipment specification as a complete basis for the design, fabrication and inspection. The specification was certified by registered professional engineers.
N-142	A detailed structural and thermal analysis was done to substantiate the adequacy of the Reactor Vessel for the design and operating conditions specified. The analytical report established the structural integrity of the Indian Point Unit No. 2 Reactor Vessel. Form N-1A, Manufacturers Data Report for Nuclear Vessels was completed certifying that design and shop inspection were in accordance with ASME Section III Code.
N-143	The authorized Code inspector performed inspection necessary to verify that the reactor vessel was fabricated in accordance with Code requirements. The authorized Code Inspector completed Form N-1A certifying that the Reactor Vessel fabrication was in accordance with ASME Section III Code.
N-144	The manufacturer required that all work by others would be in accordance with his requirements and with Code requirements. The manufacturer certified in Form N-1A that he had completed the vessel in accordance with Code requirements.
N-150 N-151 N-152	A general arrangement drawing of the Reactor Vessel showing specific dimensional locations of boundaries as defined by the Code was prepared and made a part of the equipment specification. Forces and moments on vessel nozzles and loadings on the vessel supports were established by a detailed computerized stress analysis considering thermal, seismic, and pipe

Code Paragraph	Steps Taken
(continued)	rupture actions. Forces and moments on the nozzles (vessel boundary) and support block loads were listed in the equipment specification.
Article 2 Requirements For Class A Vessels - Introduction	
N-220	The design temperature for the Reactor Vessel was established in a range where creep and stress-rupture effects are not significant factors. The equipment specification specified the design temperature. Operating restrictions on the pressure/temperature conditions of the Reactor Vessel were established to assure that the Reactor Vessel would never be operated where brittle fracture can occur. The Technical Specifications specified the conditions for safe operation of the Reactor Vessel to be always above the nil ductility transition temperature plus 60°F.
Article 3 Materials	
N-310	The pressure-boundary materials in the Reactor Vessel are established by the design specification and are included in Tables N-421 and N-422. The equipment Specification listed the acceptance materials for use as forgings, cladding, plate, closure studs and miscellaneous components. The vessel manufacturer's procurement control procedures required material supplied by suppliers to meet specifications in Tables N-421 and N-422. Material certifications provided by material suppliers included chemistry, mechanical properties, non-destructive testing certification, and impact test data with Charpy curves as required.

Code Paragraph	Steps Taken
N-311	The vessel manufacturer followed special requirements (materials) established in the Article 3 Code paragraphs. All examinations, tests, or heat treatments were documented by vessel manufacturer or material supplier.
N-312	The vessel manufacturer established with material supplier the requirements for examination and certification. All material certifications by material supplier are on file at vessel manufacturer facility. Vessel manufacturer certified on Form N-1A that material documentation is complete.
N-313.1 N-313.2	The vessel manufacturer was required to meet Code paragraphs relative to material test coupon heat treatment and location of specimen. Form N-1A certified vessel manufacturer met these Code requirements. The equipment specification required that samples for surveillance tests be metallurgically similar (thermal history) to the test specimens used in material qualification tests and established dimensional data.
N-321.1	The vessel manufacturer was required to ultrasonic test examine plates in accordance with Code procedures and acceptance standards. The equipment specification required 100% ultrasonic test volumetric coverage and required manufacturer to submit for approval his ultrasonic procedures and test reports.
N-321.2 N-321.3	Repairs to plates or cladding by welding were permitted subject to standards and limitations for maximum defect size, repair procedures, heat treatment, examination and recordation. Records of repairs, repair procedures and examination results were certified by vessel manufacturer as in accordance with Code Section III Standards.

Code Paragraph	Steps Taken
N-322	The vessel manufacturer was required to ultrasonic test plus magnetic particle or liquid penetrant examine the forgings and bars in accordance with Code procedures and acceptance standards. Repairs by welding were permitted subject to limitations for maximum defect size, repair procedures, heat treatment, examination and recordation. The equipment specification required manufacturer to submit for approval his ultrasonic test procedures and test reports and required manufacturer to magnetic particle or liquid penetrant examine.
N-324	The vessel manufacturer was required by the equipment specification to examine by either ultrasonic test, penetrant test, or magnetic particle all tubular products that confine reactor coolant during operation in accordance with Code procedures and acceptance standards. Defect elimination or repairs by welding were permitted subject to standards and limitations for maximum defect size, repair procedures, heat treatment, examination, and recordation.
N-325	The vessel manufacturer was required to examine by ultrasonic test and magnetic particle the closure studs and nuts in accordance with Code procedures and acceptance standards. Products with defects greater than acceptance standards were considered non-reparable and were unacceptable. The equipment specification required submittal for approval of procedures and test results, and required magnetic particle examination of closure stud surfaces before and after threading utilizing circular and longitudinal magnetization.
N-326	For steel products that had their properties enhanced by accelerated cooling, manufacturer was required to examine by ultrasonic test or magnetic particle method in accordance with Code procedures and acceptance standards. The equipment specification required submittal for approval of procedures and test results.

Code Paragraph	Steps Taken
N-331	Charpy V-notch impact tests on steel materials were required in accordance with Code procedures and acceptance standards for determination of nil-ductility transition temperatures. The equipment specification required vessel material to undergo a Charpy V-notch impact test to Section III rules, established requirements for Charpy specimens from flange forgings, established requirements for Charpy specimens from bolt material, established heat treatment requirements for the Charpy specimens, established number (3) and test temperature of specimens, and required full data recording and Charpy curves for forgings and plates where the test temperature was not attainable. The equipment specification required the nil-ductility transition temperature be determined for the plates and forgings using drop weight data and Charpy V-notch curves.
N-333	Westinghouse specified to the vessel manufacturer that vessel materials be heat treated in accordance with approved procedures to enhance impact properties. (Note: This paragraph is optional under Code rules.)
N-334.1	Check analyses to confirm ladle analyses were required of the materials in accordance with the frequency standards established by the Code (the number of check analyses was required to be not less than the number of tensile tests specified in the material specification.) Material certification by material suppliers show check analysis. The vessel manufacturer certified in Form N-1A that this requirement was met.
N-334.3	Identification of pressure boundary materials as well as limitation on techniques of marking as established in the Section III Code rules was required for the vessel materials. The vessel manufacturer certified in Form N-1A that this requirement was met.

Code Paragraph	Steps Taken
Article 4 Design	
N-410	The vessel manufacturer was required to analyze the design to show that stresses were within the limits established in the Section III rules. The equipment specification required that calculations be made to prove the adequacy of the pressure containing parts and be submitted for approval. Form N-1A Manufacturer Data Report noted that the Stress Analysis Report was completed.
N-411	The methods for calculating equivalent stresses, maximum shear stress, and combination of stresses have been established in subsequent sections of the Code rules based on these definitions of stress determination. The equipment specification required the vessel design to be in accordance with Section III Code rules. The analytical report utilized calculational methods that used these definitions as a basis for determining stresses. The resulting stresses were determined as within the limits established in the Section III rules.
N-413 N-414	The vessel manufacturer was required to calculate stress intensities for general primary membrane stress, local primary membrane stress, primary bending stress, secondary stress, and peak stress by the methods and procedures established in the Section III rules to show structural adequacy of the vessel. The analytical report incorporated these techniques; the calculated stress intensities were determined as within the stress limits established in the Section III rules. The equipment specification specified loadings on the reactor vessel.

Code Paragraph	Steps Taken
N-415 N-415.2 N-415.3	The vessel manufacturer was required to perform an analysis of cyclic operations (fatigue analysis) for conditions involving cyclic loads and thermal conditions in accordance with methods and procedures established in Section III rules to show the suitability of the vessel for specified operating conditions. The equipment specification specified vessel operating and design pressure and temperature, water inlet and outlet temperatures, pressure and thermal transient conditions for various normal operating modes, power steady state temperature and pressure fluctuations, and abnormal thermal and pressure transients for postulated conditions. The equipment specification specified radiation heating in the vessel, thermal and seismic loading on the vessel, and required the design of the vessel to be in accordance with Section III Code rules. The analytical report incorporated these techniques; the calculated cyclic stress intensities were determined as within the limits established in the Section III rules.
N-416 N-416.1 N-416.2 N-416.3 N-416.4	The vessel manufacturer was required to perform a fatigue analysis for cyclic operations to establish suitability of the bolts in accordance with methods and procedures established in Section III rules. The vessel manufacturer was required to use bolt material having minimum tensile strength greater than 100,000 psi in the bolt design. The equipment specification specified the design and transient conditions for steady state, normal operating modes, and for postulated abnormal transients of the vessel, specified bolting studs to be SA-320, and required the vessel design to be in accordance with Section III Code rules. Manufacturer's Inspection Report has certified that bolting material has met the requirements of SA-320 Grade L43 to Code Case 1335-1.

Code Paragraph	Steps Taken
(Continued)	Form N-1A Manufacturers Data Report has certified that bolting material has a tensile strength of 145,000 psi. The analytical report incorporated these techniques, and the cyclic and thermal stresses. The calculated stress intensities were determined as within the stress limits established by the Section III rules.
N-417 N-417.1 N-417.2 N-417.3 N-417.4 N-417.5 N-417.6	The vessel designer was required by the equipment specification to evaluate and to place special stress limits differing from the required basic stress intensity limits to cover special operating conditions or configurations relative to bearing stress, pure shear, thermal stress ratchet, stresses beyond yield strength, and combination of triaxial stresses in accordance with methods and procedures established in the Section III rules. The analytical report evaluated for the special operating conditions and configurations; calculated stresses were determined as within the limits established in the Section III rules.
Table N-421 Table N-424 Table N-426 Table N-427	The vessel manufacturer was required to use the values for design stress intensity, yield strength, coefficient of thermal expansion, and moduli of elasticity as tabulated in the Section III rules for the materials specified in the vessel design. The equipment specification required the vessel design to be in accordance with Section III Code rules, specified the required materials, and the design temperature. Stress limits included in the analytical report were established from the tabulated values in the Section III rules.

Code Paragraph	Steps Taken
N-430	The vessel manufacturer was required to analyze and compute vessel stress intensities in accordance with formulas and methods established in the Section III rules with minimum wall thicknesses in accordance with applicable formula. The equipment specification required the vessel design to be in accordance with Section III Code rules. The vessel manufacturer has certified in the analytical report that stresses were evaluated in light of the strength and fatigue requirements of the Section III Code rules.
N-441	The vessel manufacturer was required (for the analyses of stress intensities) to design for the most severe condition of coincident operating pressure and temperature as well as for the specified maximum design pressure which exists under the design conditions. The equipment specification specified design and operational water temperature and pressure variations for various plant transient conditions. The analytical report listed and used these parameters as part of the vessel design criteria.
N-442	The vessel manufacturer was required (in the analysis involving vessel pressures and loadings) to use the actual metal temperature for each area of the vessel considered where the use of the actual operating pressure was required. The equipment specification specified design and normal water operating temperature variations for plant transients. The analytical report assumed surfaces in contact with primary coolant water to have an infinite heat transfer coefficient (that is, the metal surface and water in contact to be at same temperature.)

Code Paragraph	Steps Taken
N-444	The vessel manufacturer was required to disregard any structural contribution to vessel strength because of the presence of vessel cladding. (When the cladding is less than 10% of component thickness, the presence of cladding may be considered in performing the fatigue analysis in accordance with the Section III procedures.) Clad thickness on inside surfaces of vessel wall and nozzles is 7/32". (Vessel shell thickness is 10.75" and 8.625".) The analytical report considered the effects of the cladding on thermal transients in the fatigue analysis of vessel components. The structural evaluation of vessel components did not assume any contribution from the interior cladding.
N-446	Nozzles and vessel discontinuities were located outside of the areas of highest neutron flux. Additional samples of vessel material were taken from the core region (highest neutron flux area) for use in the vessel surveillance program. The equipment specification specified dimensional details of the reactor vessel in the outline drawing and specified the taking and preparation of appropriate samples for use in the surveillance program. The Technical Specifications indicate the withdrawal intervals of capsules and evaluation of the test specimens.
N-447	The vessel designer was required to consider all loads and loading conditions in the design of the vessel. The equipment specification established for the vessel designer loading conditions covering design, normal operation, steady state fluctuations, normal transients, postulated abnormal transients, seismic occurrences, piping rupture, test, and installation.

Code Paragraph	Steps Taken
N-448	Westinghouse, in considering the design, operating, and abnormal conditions of the reactor vessel, established the mechanical loads and combinations thereof for use in the calculation of stress intensities. The equipment specification specified the actual mechanical loads and loading combinations.
N-451 N-452 N-454 N-455	The vessel manufacturer was required to perform specific structural and fatigue analysis in the vicinity of openings to show adequacy of the nozzles and vessel shell in accordance with methods and procedures established in the Section III rules. The analytical report included the detailed analyses of openings with evaluation made of the amount and distribution of compensation required for openings, and of the effect on ligament strength due to nozzle spacing. Stress intensities were determined as within the limits of acceptance in the Section III Code rules. The equipment specification specified that calculations proving adequacy of pressure containing parts were to be submitted prior to fabrication.
N-457 N-461 N-462 N-465	The vessel manufacturer was required to design for and attach nozzle connections in accordance with methods provided in Section III Code rules to achieve continuity of metal and facilitate required radiographic or other types of examination. The equipment specification specified that detail drawings were to be submitted prior to material procurement and welding procedures were to be submitted for review and approval prior to fabrication. The vessel manufacturer certified that all full penetration welds were radiographed and were acceptable in accordance with Section III Code standards and that results of other methods of examination (ultrasonic test, penetrant, magnaflux) were acceptable in accordance with Section III Code standards.

Code Paragraph	Steps Taken
N-466	The vessel manufacturer was required to make transition joints between sections of different thicknesses in accordance with dimensional and taper limitations in the Section III Code rules. The equipment specification specified that detail drawings were to be submitted prior to material procurement. Form N-1A Manufacturers Data Report certified that design of the vessel was in accordance with Section III Code rules.
N-468	The vessel manufacturer was required to perform post weld heat treatment in accordance with methods established in the Section III Code rules. Vessel manufacturer was required to prepare welding procedures for review and approval, such procedures to include preheat and postheat provisions. The equipment specification specified that welding procedures, including preheat and interpass temperatures and postheating procedures, were to be submitted for review and approval, heat treatments were to be performed per detailed procedures, and the fabrication to be in accordance with Section III code rules.
N-469	The vessel manufacturer was required to radiograph all pressure stressed full penetration welds in accordance with methods and standards established in the Section III Code rules. The equipment specification specified that radiography of welds were to conform to Section III Code rules and that the vessel manufacturer was to keep a permanent record of all inspection data.
N-473	The vessel manufacturer was required to design and analyze for stresses produced from loads imposed by vessel supports. The equipment specification specified that vessel supports were to permit axial and

Code Paragraph	Steps Taken
(continued)	radial expansion of the vessel for normal operation and transient conditions, and specified loadings resulting from thermal, seismic, static and dynamic forces for evaluating the vessel supports. The analytical report analyzed for stresses within the vessel and its supports from the specified loads. The computed stress intensities were determined as within the limits established in the Section III Code rules.
N-474	The vessel manufacturer was required to consider the effects on vessel design of various attachments to the vessel. The equipment specification listed the attachments to be made on the reactor vessel and established operational loads for those to be analyzed. Included in the list were loads to be supported within the vessel interior, clips for holding thermal insulation, shroud loads on the vessel head and the head lifting lugs. The analytical report included detailed calculations for effects on vessel stress intensities from interior core support brackets. The equipment specification specified magnaflux and dye penetrant check on weld surfaces.

Code Paragraph	Steps Taken
Article 5 Fabrication	
N-511.1	The vessel manufacturer certified he complied with the material treatments and examinations specified, and provided certified results of all tests for vessel material as well as test specimens, all in accordance with Section III Code procedures and equipment specification requirements. The equipment specification specified vessel manufacturer submit the certifications covering material evaluations, mill test data, supplier test data, and inspection reports prior to fabrication, and required the vessel manufacturer to perform chemical analysis of cladding material each shift as well as separate material qualification tests for each combination of filler wire and flux.
N-511.2	Test coupons were required to have received heat treatment equivalent to that given the material used in the vessel in accordance with Section III Code rules. The equipment specification specified Charpy impact specimens to receive heat treatments equivalent to vessel material heat treatment and material test results and evaluation information be submitted for approval. Test specimens were heat treated to 40 hours with processes representative of vessel material heat treatment conditions.
N-512.1 N-512.2	The vessel manufacturer was required to identify pressure containing parts of the vessel through all stages of fabrication and prepare a tabulation of materials used noting each location, mill test reports, and markings in accordance with Section III Code techniques.

Code Paragraph	Steps Taken
(continued)	The vessel manufacturer through his procurement procedures required the material suppliers to identify vessel components in accordance with Section III Code. All vessel materials were identified with piece mark and heat number. Material identification was transferred to the material identification drawing.
N-513.1 N-513.2	The vessel manufacturer was required to magnaflux for cracks, laminations and discontinuities on all surfaces to be clad or welded in accordance with acceptance standards established in the Section III Code rules. (Note: Section III Code rules permitted either magnaflux or dye penetrant check for this examination of cut edges.) Evaluation of extent of defect was required to be by ultrasonic test method with acceptance standards as established in Section III Code rules. The equipment specification specified magnaflux of surfaces with acceptance limits in accordance with Section III Code, specified that manufacturer fabrication process be submitted for approval, specified that detail drawings be submitted for approval prior to material procurement and specified that a permanent record be kept of all inspection data. Acceptance ticket records confirmed examination of cut edges during the manufacturing stages.
N-513.3	The vessel manufacturer established minimum thickness of vessel components on detail drawings. Acceptance tickets maintained during fabrication verified that component wall whickness were within required limits. The equipment specification specified a permanent

Code Paragraph	Steps Taken
(continued)	record be kept of all inspection data and that detail drawings be submitted for approval prior to material procurement. The analytical report determined adequacy of reactor vessel based on dimensional data shown on detail drawings. No under-toleranced areas existed requiring additional analysis.
N-514.1 N-514.2	The vessel manufacturer was required to follow procedures established in the Section III Code rules for repair, examination, and recordation of defects in materials. Repairs, when necessary, employed approved weld procedures. Repairs were documented on material supplier reject tickets with final acceptance clearing the reject documented in supplier inspection acceptance tickets. The equipment specification specified that acceptance limits for magnaflux and dye penetrant inspection be in accordance with Section III Code rules and welding procedures be submitted for approval.
N-515	For forming processes, the vessel manufacturer was required to conduct procedure qualification tests on material test specimens to verify tensile and impact properties are met in accordance with Section III Code rules. The equipment specification established the use of manufacturer process specifications in forming and heat treatment of vessel plates, and specified process specifications be submitted for approval prior to material procurement or fabrication. Material qualification was based on specimen coupons removed from material after forming and stress relieved to 1150° $\pm$ 25°F for 40 hours.

Code Paragraph	Steps Taken
N-516 N-517	The vessel manufacturer was required to meet shell and head dimensions within the deviations established in Section III Code rules. Fit-up tolerances prior to welding were confirmed using the identical template. Manufacturers standard shop practices employed use of template based on Section III Code rules. Inspection reports by code inspector verify tolerances. The equipment specification specified vessel dimensions including tolerances.
N-518 N-518.1 N-518.2 N-518.3 N-518.4 N-518.5	The vessel manufacturer was required in the design and fabrication of attachments to meet the material, design and inspection criteria established in the Section III Code rules. Manufacturer detail drawings of attachments were submitted for approval. Information included material designation, weld procedures, stress relief, and inspection methods. The equipment specification specified the welded attachments, submittal of drawings, material and process specifications for approval prior to procurement or fabrication, and magnaflux and dye penetrant examinations in accordance with Section III Code acceptance limits. Manufacturer, and Code inspection verified compliance with detail drawings.
N-519	The vessel manufacturer was permitted to use mechanical means to cut material but was required to examine cut edges in accordance with Section III Code procedures. The vessel manufacturer established cutting and inspection methods in the manufacturing plan. The equipment specification specified magnaflux and ultrasonic test inspections with acceptance limits in accordance with Section III Code rules. Examination results and inspection acceptance tickets verified compliance.

Code Paragraph	Steps Taken
N-521	The vessel manufacturer was required to weld pressure containing members by arc or gas welding processes in accordance with qualification requirements in Section IX Code. Westinghouse in the equipment specification established welding processes for use in pressure containing joints and cladding, specified all welding procedures to be submitted for approval, specified welding by electric arc methods and specified attachment of internal cladding by arc welding. Form N-1A, Manufacturers Data Report certified that fabrication was in accordance with Section III Code rules.
N-522 N-522.1 N-522.2 N-522.3 N-522.4 N-522.5	The vessel manufacturer was required to qualify procedures and welders in accordance with Section IX Code requirements and Section III Code rules.
N-523	The vessel manufacturer was required to prepare and submit welding procedures and specifications for approval prior to fabrication. Control of electrode and other material was established in manufacturer procedures. The equipment specification specified welding procedures to be submitted and approved.
N-524 N-525	The vessel manufacturer was required to use approved procedures for tack welds and to meet fit-up alignment tolerances established in Section III Code rules during assembly of parts. Fit-up and alignment of parts was verified on manufacturers shop traveller sheet. Magnaflux inspection of tack welds was required.

Code Paragraph	Steps Taken
N-526 N-527.1	Manufacturer weld procedures were submitted prior to fabrication and included standards for the finished butt weld bead as well as for intermediate process steps between passes. The equipment specification specified welding procedure requirements to be submitted for approval.
N-528 N-528.1 N-528.2	The vessel manufacturer was required to remove unacceptable weld defects, repair, re-inspect, and heat treat all in accordance with methods established in Section III Code rules. Manufacturer weld procedures included repair of welds. Manufacturer process specifications covered inspection by ultrasonic test, dye penetrant, radiography, and magnaflux methods and weld heat treatment procedures in accordance with methods established in Section III Code rules. The equipment specification specified welding procedures and be submitted for approval prior to fabrication, and approved and authorized use of vessel manufacturer process specifications. Records certifying re-test and acceptance of welds were made.
N-529	The vessel manufacturer was required to submit cladding procedures for approval and perform dye penetrant examination in accordance with methods extablished in Section III Code rules. The vessel manufacturer was required to perform bend test on clad specimens and perform ultrasonic test examination of cladding both before and after hydrotest. The equipment specification specified expected and acceptable values for alloy content of cladding material, specified cladding to be attached by arc welding method and the surface to be suitable

Code Paragraph	Steps Taken
(continued)	for dye penetrant inspection, and specified qualification and process testing of cladding in accordance with Section III Code rules and requirements.
N-530 N-531	When preheat of welds was utilized the vessel manufacturer was required to include such requirements in the welding procedure in accordance with Section IX Code requirements. The equipment specification specified information on weld preheat to be contained in welding procedures when submitted for approval, and specified submittal to be made prior to fabrication.
N-532 N-532.1 N-532.2 N-532.3 N-532.4 N-533	The vessel manufacturer was required to prepare procedures and perform post weld heat treatment. The equipment specification specified heat treatment in accordance with vessel manufacturer procedure, and specified processes to be submitted for approval prior to fabrication.
N-542 N-542.1 N-542.2 N-542.3 N-542.4	The vessel manufacturer was required to prepare weld cladding procedures, qualify welders and procedures, and examine overlay surface in accordance with criteria established in Section IX, Code and Section III Code rules. The equipment specification specified cladding to be by arc welding methods and suitable for dye penetrant inspection, specified weld procedures to be submitted for review and approval, specified dye penetrant examination of cladding and overlay, established acceptance standards per Section III Code rules, specified process specifications to be submitted for approval prior to fabrication, specified expected and

Code Paragraph	Steps Taken
(continued)	acceptable values for alloy content of cladding material, specified qualification and process testing of cladding in accordance with Section III Code rules and Westinghouse requirements, specified ultrasonic test examination and established acceptance standards for areas of the vessel and head after cladding both before and after hydrotest, and specified bend tests on clad samples and established acceptance standards.

Code Paragraph	Steps Taken
Article 6	
Inspection	
N-611.1 N-611.2	The authorized Code inspector certified the materials and dimensions complied with the requirements of the approved design and that other necessary detailed inspections were performed to verify that the Reactor Vessel was fabricated in accordance with Code requirements. The Code Inspector initialed the traveller accompanying the vessel during the inspections. Form N-1A Manufacturers Data Report for Nuclear Vessels was completed certifying that materials, material dimensions and inspections met the requirements of ASME Section III Code. The completed form was signed by the authorized Code Inspector.
N-612	A qualified inspector, as defined by this code paragraph, was retained by an insurance company to perform the inspections required by the ASME Section III Code. Form N-1A Manufacturers Data Report for Nuclear Vessels was certified by the Code inspector, who was employed by Hartford Steam Boiler Insp. & Ins. Co., Hartford, Conn., and qualified by the National Board of Boiler and Pressure Vessel Inspectors.
N-613	The vessel manufacturer was required to provide access for the inspector to perform all necessary inspections.
N-614.1 N-614.2 N-615 N-616 N-617	The authorized Code inspector certified that materials complied with the applicable requirements of the subsections of the ASME Section III Code. In addition, Westinghouse also duplicated these examinations by the Code Inspector. The manufacturer

Code Paagraph	Steps Taken
(continued)	certified in Form N-1A that all details of materials and design of the pressure vessel were in accordance with Code requirements. The authorized Code Inspector completed Form N-1A certifying that the manufacturer constructed the pressure vessel in accordance with the ASME Section III Code.
N-618.1 N-618.2	The authorized Code Inspector and the manufacturer certified in Form N-1A that the reactor vessel was successfully hydrostatically tested to 3125 psi and shop inspection were in accordance with the requirements of the ASME Code Section III. The equipment specification specified magnaflux testing of all unclad surfaces and welds, and ultrasonic testing of selected pressure vessel areas after the hydrostatic test.
N-621	The authorized Code inspector performed inspections necessary to verify that the Reactor Vessel welding procedure was in accordance with Section III Code requirements. The authorized Code Inspector and the manufacturer certified in Form N-1A that the vessel was completed in accordance with Section III Code requirements.
N-622	The authorized Code inspector performed inspections necessary to verify that the pressure vessel welding was performed by welders qualified in accordance with ASME Code Section IX. The manufacturer certified that welding on the reactor vessel and components was done by welders who were qualified in accordance with ASME Code Section IX. In addition, Westinghouse audited the certification.

Code Paragraph	Steps Taken
N-623.1	The authorized Code Inspector performed inspections necessary to verify the use of non-destructive examination methods and qualified operators in non-destructive methods in accordance with ASME Code Section III requirements. The procedures were approved for compliance to code and design specifications. The authorized Code Inspector, completed Form N-1A certifying that shop inspection was in accordance with ASME Section III Code.
N-623.2	The vessel manufacturer quality control program assured that operators of the required non-destructive examination methods met the requirements of the ASME Code Section III. Operator qualification and medical records were documented.
N-624.1 N-624.2 N-624.3 N-624.4 N-624.5 N-624.6 N-624.7 N-624.8 N-624.9	The vessel manufacturer certified that radiographic testing on the reactor vessel and components was in accordance with ASME Code Section III requirements and with the additional Westinghouse specifications. The vessel manufacturer procedures and radiographic drawings were approved by Westinghouse. The vessel manufacturer was required to notify 15 days in advance of completion of the radiograph inspection of the main seam welds. Following the vessel manufacturer and Code inspector acceptance, Westinghouse reviewed all radiographs 100%.
N-625.1 N-625.2 N-625.3 N-625.4 N-625.5 N-625.6	The vessel manufacturer prepared and submitted a report of the ultrasonic examination for acceptance by the Inspector. The manufacturer documented the test results that exceeded 60% of the full reflection from the reference plate reference hole. The equipment specification specified submission of

Code Paragraph	Steps Taken
(continued)	the test procedure prior to testing, and submission of the test reports was made as they became available. The examination was confirmed by Westinghouse surveillance. The manufacturer certified that the ultrasonic examination of welded joints on the reactor vessel and components was in accordance with the ASME Code Section III.
N-626.1 N-626.2 N-626.3 N-626.4 N-626.5	The vessel manufacturer written magnetic particle testing procedures were approved. Westinghouse maintained surveillance of the examinations. The vessel manufacturer certified that the magnetic particle testing was in accordance with the ASME Code Section III requirements.
N-627.1 N-627.2 N-627.3 N-627.4 N-627.5 N-627.6 N-627.7	The vessel manufacturer written liquid penetrant testing procedures were approved. Westinghouse maintained surveillance of the examinations. The manufacturer certified that the liquid penetrant testing was in accordance with the ASME Code Code Section III Requirements.

Code Paragraph	Steps Taken
Article 7	
Testing	
N-711 N-712 N-713	Impact tests of base metal were required and were carried out by the vessel manufacturer. The vessel manufacturer certified in Form N-1A that he completed all details of material design, construction and workmanship of the vessel in accordance with Section III Code rules. Vessel plate test data was documented. The equipment specification specified Charpy V-notch impact tests for vessel materials.
N-714	The vessel manufacturer's hydrostatic test procedures were submitted for approval. The hydrotest was a fabrication hold point for verification by Westinghouse, vessel manufacturer's inspector, and the authorized Code Inspector. The test temperature was 60°F above the highest NDTT reported on pressure boundary material. The equipment specification specified hydrostatic testing in accordance with the requirements of ASME Code Section III. The manufacturer certified satisfactory completion of hydrostatic test to 3125 psi. The authorized Code Inspector completed form N-1A, Manufacturer Data Report for Nuclear Vessels, which certified the completion of satisfactory shop inspection.
N-716	The vessel manufacturer was required to use and calibrate pressure test gages in accordance with Section III Code rules. The pressure test gages were calibrated prior to the hydrotest.

Code Paragraph	Steps Taken
Article 8	
Marking, Stamping, and Reports	
N-811	Nameplates bearing the information required by the ASME Code Section III were attached to the pressure vessel as follows: (a) Reactor Vessel, Class A; Mfg. C.E.; Design Pressure 2500 psi at 659°F; Mfg. Serial No. CE 65101; year built - 1968. (b) Closure Head, Class A; Mfg. C.E.; Design Pressure 2500 psi at 650°F; Mfg. Serial No. CE 65201; Year Built - 1968. The Code N- symbol, figure N-811 was applied to both plates. Ruboffs and photographs of the plates were made.
N-812 N-831.1 N-831.2 N-831.3	The vessel manufacturer had valid certification of authorization to use the Code N- symbol. The reactor vessel and closure head nameplates each bear National Board No. 20756, which was entered on form N-1A Manufacturer Data Report for Nuclear Vessels.
N-821	The Code required markings were applied to separate nameplates permanently attached to the reactor vessel and closure head flanges. The integrity of the vessel was not affected.
N-822	The Code requirements for stamping of nameplates were met by the manufacturer who submited details for approval reflecting Section III Code rules. The nameplate data is substantially as shown in Fig. N-822 of the ASME Code Section III.

Code Paragraph	Steps Taken
N-823	The nameplates were attached in the supplier's shop prior to shipment and the authorized code inspector in the field satisfied himself that this was done in accordance with the Section III Code rules. The vessel manufacturer entered on Form N-1A the required data which was identical to the information stamped on the nameplates. The authorized Code Inspector completed Form N-1A Manufacturer Data Report for Nuclear Vessels.
N-840	Form N-1A was completed in accordance with requirements in Section III Code rules. The form was filled out by the vessel manufacturer, signed by his authorized representative and by the authorized Code Inspector. Both signatures were dated August 26, 1968.

Code Paragraph	Steps Taken
Article 9	
Protection Against Overpressure	
N-910.1 N-910.2 N-910.3 N-910.4 N-910.5 N-910.6 N-910.7 N-910.9	The reactor vessel is protected against overpressure by three Code approved self actuated safety valves located on the top of the system pressurizer. These safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10 per cent in accordance with Section III Code rules. The capacity of the pressurizer safety valves is determined from considerations of: (1) the reactor protective system, and (2) accident or transient conditions which may potentially cause overpressure. The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or other control. No stop valve is placed between the reactor vessel and the safety valves. The safety valves are each larger than 1/2" pipe size. The Indian Point Plant Unit 2 FSAR., Section 14, reports the technical details of the overpressure analysis. The details of the plant arrangement and overall system characteristics are reported in Section 4 of the FSAR.

Code Paragraph	Steps Taken
Appendix I	
Article I-1	
Tentative Thickness of Shell and Heads	
I-100	The vessel manufacturer was required to establish minimum tentative thicknesses of shells in accordance with formula in Section III Code rules. The analytical report included analysis to determine the minimum tentative thicknesses. Detailed analysis considering fatigue, thermal, and all other loadings to determine stress intensities and show structural adequacy of the design thicknesses was included.
Article I-6	
Pressure Stresses in Openings for Fatigue Evaluation	
I-600	The vessel manufacturer utilized the stress index method and experimental stress analysis in determining peak stresses around openings in accordance with criteria established in Section III Code rules. The analytical report included these methods of analysis and results for determining peak stresses.
I-611 I-612 I-613	The vessel manufacturer utilized stress indices in the analysis for peak stresses at certain nozzle locations in accordance with methods established in Section III Code rules. In addition, the vessel manufacturer performed interaction analyses of the nozzle to vessel junctures when evaluating the structural and fatigue effects from pressure, thermal, and seismic loadings to determine peak stress values. The analytical report included the detailed analyses of vessel openings. Stress values were determined as within the allowable limits established in Section III Code rules.

Code Paragraph	Steps Taken
I-621 I-622	The vessel manufacturer utilized experimentally determined stress data in evaluating closure head penetrations for stress intensities in accordance with general criteria established in Section III Code rules. The analytical report included the analysis of the closure head penetrations. Stress values were determined as within the allowable limits established in Section III Code rules.
Article I-10 Experimental Stress Analysis	
I-1011 I-1012 I-1013 I-1020 I-1031 I-1032 I-1040 I-1062	The vessel manufacturer utilized experimental test results to establish ligament efficiency for penetrations in performing the stress analysis of the closure head penetration spacings and establish the thickness of the closure head. The analytical report showed adequacy of closure head penetration spacing and thickness.
Article I-12 Stresses In Bolting	
I-1200 I-1210 I-1220 I-1230	The vessel manufacturer utilized analytical techniques consistent with methods in Section III Code rules to establish minimum bolt area and tentative flange loads. Detailed analysis was performed by the vessel manufacturer to establish final stress values, flange movements, and verify thicknesses of components. The analytical report included the detailed analysis of bolting and flanges. Stress values were determined as within the allowable limits established in Section III Code rules.

Code Paragraph	Steps Taken
Appendix III	
Non-Mandatory Preheat Requirements	
III-100 III-110 III-120	The vessel manufacturer was required to specify in his welding procedures the weld preheat in accordance with procedure qualifications of Section IX Code rules. Preheat, interpass, and postheating temperature control were established in vessel manufacturer's welding procedures in accordance with guidelines listed in Section III Code rules. The equipment specification specified all weld procedures to include minimum preheat temperature when submitted for approval and all process specifications to be submitted prior to fabrication. Temperature maintenance was audited throughout weld cycle. Parts were given an intermediate heat test to 1150° $\pm$ 25°F. before dropping to ambient temperature after welding.

Code Paragraph	Steps Taken
Appendix IV Porosity Charts IV-100 Table IV-100	The vessel manufacturer was required to confirm to the acceptance standards established in Section III Code Rules for radiographically determined porosity in welds. Porosity charts were employed for thicknesses up to 4 inches. For material over 4 inches thick, acceptance limits for weld porosity are established in Section III Code rules. The equipment specification specified that a permanent record be kept of all inspection data and that results of all inspections to be sumbitted to Westinghouse.

Code Paragraph	Steps Taken
Appendix VII	
Manufacturer Data Report Forms	
Form N-1A	Form N-1A was completed and certified by the vessel manufacturer and Inspector.

Appendix B

Systems, Components and Procedures
Provided to Assure Operation in Accordance
With the Technical Specifications

1.0 Introduction

The purpose of this Appendix is the discussion of those features incorporated in the design of Indian Point Unit No. 2 that provide assurance that the reactor vessel will be operated in accordance with the Technical Specification and therefore will not fail. This discussion includes the following:

2.0 Assurance Provided That Abnormal Conditions Will Not Affect Reactor Vessel Integrity

3.0 Systems and Components Installed to Insure Operational Conformity With Design and Technical Specifications

4.0 Evaluations Demonstrating That Systems and Components Will Perform as Required

5.0 Testing and Surveillance Procedures Provided to Insure That the Reactor Vessel, Systems and Components Will Perform as Required

6.0 Summary

2.0 ASSURANCE PROVIDED THAT ABNORMAL CONDITIONS WILL NOT AFFECT REACTOR VESSEL INTEGRITY

In this section the action and interaction of the various systems provided to correct abnormal operating situations that might affect the reactor primary system pressure and temperature, and that might thereby deleteriously affect the integrity of the reactor vessel unless corrected is reviewed. This subject is discussed in detail in the FSAR, Chapter 14, Safety Analysis. Each abnormal situation and the corresponding protection feature is separately itemized as follows:

2.1 Uncontrolled RCCA Withdrawal From a Subcritical Condition

2.1.1 General

An RCCA withdrawal incident is defined as an uncontrolled addition of reactivity to the reactor core by withdrawal of rod cluster control assemblies resulting in a power excursion. While the probability of a transient of this type is extremely low, such a transient could be caused by a malfunction of the reactor control or control rod drive systems. This could occur with the reactor either subcritical or at power. The latter situation is discussed in the following subsection.

Reactivity is added at a prescribed and controlled rate in bringing the reactor from a shutdown condition to a low power level during startup by RCCA withdrawal. Although the initial startup procedure uses the method of boron dilution, the normal startup is with RCCA withdrawal. RCCA motion can cause much faster changes in reactivity than can be made by changing boron concentration.

The nuclear power response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a sizable prompt negative fuel temperature coefficient (Doppler effect) and is of prime importance during a startup accident since it limits the power to a tolerable level prior to external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the accident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

If the transient is permitted to continue without trip, the coolant temperature and pressure will increase and an overpressure condition with deleterious effect on vessel integrity could occur.

In order to terminate the transient before an overpressure condition is reached, RCCA design features and automatically actuated trips are provided, as discussed below:

In the RCCA design, certain mechanical and electrical power restraints are provided. Power to operate the rod cluster drive mechanisms is supplied to preselected groups, rather than to single mechanisms, and these group configurations are not altered during core life. The rods are therefore physically prevented from withdrawing in other than their respective groups. Power supplied to the rod groups is controlled such that no more than two groups can be withdrawn at any time. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is analyzed in the detailed plant analysis assuming the simultaneous withdrawal of the combination of the two rod groups with the maximum combined worth at maximum speed, which is well within the capability of the protection system.

However, assuming that a continuous RCCA withdrawal is nevertheless initiated and further assuming the source and intermediate range alarms and indications are ignored, the transient will be terminated by the following automatic safety features:

a. Source range flux level trip - actuated when either of two independent source range channels indicates a flux level above a preselected, manually adjustable value. This trip function may be manually bypassed when either intermediate range flux channel indicates a flux level above the source range cutoff power level. It is automatically reinstated when both intermediate range channels indicate a flux level below the source range cutoff power level.

b. Intermediate range rod stop - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This rod stop may be manually bypassed when two out of the four power range channels indicate a power level above approximately ten per cent power. It is automatically reinstated when three of the four power range channels are below this value.

c. Intermediate range flux level trip - actuated when either of two independent intermediate range channels indicates a flux level above a preselected, manually adjustable value. This trip function is manually bypassed when two of the four power range channels are reading above approximately ten per cent power and is automatically reinstated when three of the four channels indicate a power level below this value.

d. Power range flux level trip (low setting) - actuated when two out of the four power range channels indicate a power level above approximately 25 per cent. This trip function may be manually bypassed when two of the four power range channels indicate a power level above approximately ten per cent power and is automatically reinstated when three of the four channels indicate a power level below this value.

e. Power range flux level trip (high setting) - actuated when two out of the four power range channels indicate a power level above a preset setpoint. This trip function is always active.

2.1.2 Method of Analysis

Analysis of this transient is performed by digital computation incorporating the neutron kinetics, including six delayed groups, and the core thermal and hydraulic equations. In addition to the nuclear flux response, the average fuel, clad and water temperature, and also the heat flux response, are computed.

In order to give conservative results for a startup accident, the following additional assumptions are made concerning the initial reactor conditions:

a. Since the magnitude of the nuclear power peak reached during the initial part of the transient, for any given rate of reactivity insertion, is strongly dependent on the fuel temperature reactivity coefficient, the least negative design value is used for the startup accident.

b. The contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient.

Accordingly, a conservatively large positive value is used, since this yields the maximum rate of power increase.

c. The reactor is assumed to be at hot zero power. This is more conservative than cold zero power since the higher initial temperature causes a higher overall heat transfer coefficient; a smaller (less negative) Doppler coefficient; and an increased thermal capacity of the fuel. Initial multiplication (k_o) is assumed to be 1.0 since this results in the maximum nuclear power peak. The fuel heat capacity was 0.07 btu/lb-°F. The total fuel to water heat transfer coefficient was the full power value, 2020 btu/sec-°F. Rated coolant flow was assumed.

d. The most adverse combination of instrument and setpoint errors, as well as delays for trip signal actuation and rod release, are taken into account. Also, the rate of negative reactivity insertion corresponding to the trip action is based on the assumption that the highest worth rod is stuck in its fully withdrawn position.

In conclusion, sufficient diversity of protection exists to prevent overpressure by an uncontrolled RCCA withdrawal from a subcritical condition. In addition to design features which prevent uncontrolled withdrawal, the startup accident, should it nevertheless occur, is terminated by the trip and rod stop protection channels which prevent overpressure and reactor vessel damage. Design of the protection features is based on a conservative analysis of the transient as described above. In addition to the features described here, the high pressure reactor trip serves as a backup to terminate the accident at 2385 psi, before an overpressure condition could occur.

2.2 Uncontrolled RCCA Withdrawal at Power

2.2.1 General

An uncontrolled RCCA withdrawal at power results in an increase in core heat flux. Since the heat extraction from the steam generator remains constant, there is a net increase in reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. Therefore, the Reactor Protection System is designed to terminate any such transient with an adequate margin to DNB, primarily to prevent the possibility of damage to the fuel elements. In addition, the Reactor Protection System also prevents damage to the reactor pressure vessel by limiting the pressure excursion during such a transient.

The automatic features of the Reactor Protection System which prevent core damage in a rod withdrawal accident at power include the following:

a. Nuclear power range instrumentation actuates a reactor trip if two out of the four channels exceed an overpower setpoint.

b. Reactor trip is actuated if any two out of four ΔT channels exceed an overtemperature ΔT setpoint. This setpoint is automatically varied with power distribution, temperature and pressure to protect against DNB.

c. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This setpoint is automatically varied with power distribution to ensure that the allowable fuel power rating is not exceeded.

d. A high pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point of 2385 psig. This set pressure will be less than the set pressure of 2485 psig for the pressurizer safety valves.

e. A high pressurizer water level reactor trip, actuated from any two out of three level channels, is actuated at a fixed setpoint. This affords additional protection for RCCA withdrawal accidents.

2.2.2 Method of Analysis

The region of permissible operation (power, pressure and temperature) is completely bounded by the combination of reactor trips: nuclear overpower (fixed setpoint); high pressure (fixed setpoint); low pressure (fixed setpoint); overpower and overtemperature ΔT (variable setpoints). These trips are primarily designed to preclude a DNB ratio of less than 1.30.

The purpose of this analysis is to demonstrate the manner in which these protective systems function for various reactivity insertion rates from different initial conditions. Reactivity coefficients, initial conditions and effects of control functions govern which protective function occurs first.

The analysis is performed using several digital computer techniques. First, the actual core limits are determined employing the W-3 DNB correlation described in Section 3 of the FSAR. Protection lines, illustrated in Figure 7.2-11 of the FSAR, are then selected and incorporated in a transient analysis by a detailed digital simulation of the unit. The detailed digital simulation consists of neutron kinetics, core thermal and hydraulic equations, primary loop hydraulic equations, including pressurizer, and a detailed representation of the steam generator, both primary and secondary, including the effect of heat transfer between the two regions.

In the analysis, the effect of the RCCA movement on core power distribution is considered in its effect of causing a decrease in overtemperature ΔT and overpower ΔT trip setpoints proportionate to the decrease in margin to DNB. This has the effect of causing a reactor trip sooner in the transient.

Results of an analysis described below show that the effect of this type of accident on coolant pressure is greatest for a slow rod withdrawal starting from full load, assuming maximum power and temperature errors.

Reactor trip or overtemperature ΔT trip occurs at approximately 150 seconds. Before trip occurs, however, the rise in temperature is quite large. Also before trip, the coolant pressure rises to the relief valve setpoint of 2335 psig, but relief valve capacity is not exceeded, even if only one of the two valves open.

Starting from full power, there is no reactivity insertion rate which will exceed the capacity of the two power-operated relief valves. Hence, peak pressure is limited by the relief valve set pressure.

However, even if it is assumed that the pressure relief valves are malfunctioning, the transient is terminated and pressure rise is prevented by the trips.

2.2.3 Conclusions

In the unlikely event of a control rod withdrawal incident, whether it be from subcritical condition, from full power operation, or at any other power level between these two extremes, the reactor pressure vessel is not adversely affected. Protection is provided by the nuclear overpower reactor trips, and the overtemperature ΔT trip, as well as by the overpower ΔT trip, the fixed high and low pressure trips and high pressurizer level trips. The preceding subsection has described the effectiveness of this protection.

2.3 Chemical and Volume Control System Malfunction

2.3.1 General

Reactivity can be added to the core with the Chemical and Volume Control System (CVCS) by feeding reactor makeup water into the Reactor Coolant System via the reactor makeup control system. The normal dilution procedures call for a limit on the rate and magnitude for any individual dilution, under strict administrative controls. Boron dilution is a manual operation. A boric acid blend system is provided to permit the operator to match the concentration of reactor coolant makeup water to that existing in the coolant at the time.

In the case of an erroneous dilution of boron in the reactor coolant, the increase in reactivity would be manifested in increased coolant temperature and pressure. As is evident from the following discussion, the reactor vessel is well protected from overpressurization and rupture as a result of CVCS malfunction.

The Chemical and Volume Control System is designed to limit, even under various postulated failure modes, the potential rate of dilution to a value which, after indication through alarms and instrumentation, provides the operator sufficient time to correct the situation in a safe and orderly manner.

There is only a single, common source of reactor makeup water to the Reactor Coolant System from the reactor makeup water system, and inadvertent dilution can be readily terminated by isolating this single source. The operation of the reactor makeup water pumps which take suction from this tank provides the only supply of makeup water to the Reactor Coolant System. In order for makeup water to be added to the Reactor Coolant System the charging pumps must be running in addition to the reactor makeup water pumps.

The rate of addition of unborated water makeup to the Reactor Coolant System is limited to the capacity of the charging pumps. This limiting addition rate is 300 gpm for all three charging pumps. This is the maximum delivery rate based on a pressure drop calculation comparing the pump curve with the system resistance curve. Normally only one charging pump is operating while the others are on standby.

The boric acid from the boric acid tank is blended with the reactor makeup water in the blender and the composition is determined by the preset flow rates of boric acid and reactor makeup water on the Reactor Makeup Control. Two separate operations are required to effect dilution. First, the operator must switch from the automatic makeup mode to the dilute mode. Second, the first button must be depressed. Omitting either step would prevent dilution. This makes the possibility of inadvertent dilution very remote.

Information on the status of the reactor coolant makeup is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from preset values as a result of system malfunction.

To cover all phases of plant operation, boron dilution during refueling, startup, and power operation are considered in this analysis.

2.3.2 Method of Analysis and Results

2.3.2.1 Dilution During Refueling

During refueling the following conditions exist:

a. One residual heat removal pump is running to ensure continuous mixing in the reactor vessel,

b. The valve in the seal water header to the reactor coolant pumps is closed,

c. The valves on the suction side of the charging pumps are adjusted for addition of concentrated boric acid solution,

d. The boron concentration of the refueling water is 2000 ppm (min.), corresponding to a shutdown of 18 per cent Δk with all control rods in; periodic sampling ensures that this concentration is maintained, and

e. Neutron sources are installed in the core and BF_3 detectors connected to instrumentation giving audible count rates are installed within the reactor vessel to provide direct monitoring of the core.

A minimum water volume in the Reactor Coolant System of 5509 ft^3 is considered. This corresponds to the volume necessary to fill the reactor vessel above the nozzles to ensure mixing via the residual heat removal loop. The maximum dilution flow of 300 gpm and uniform mixing are also considered. The flow rate of the residual heat removal loop is 3000 gpm.

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate increase is proportional to the inverse multiplication factor. At 1560 ppm, for example, the core is 4.5 per cent shutdown and the count rate is increased by a factor of four over the count rate at a boron concentration of 2000 ppm (min.).

The boron concentration must be reduced from 2000 ppm (min.) to approximately 1150 ppm before the reactor will go critical. This would take at least 1.6 hours. This is ample time for the operator to recognize the audible high count rate signal and isolate the reactor makeup water source by closing valves and stopping the reactor makeup water pumps.

2.3.2.2 Dilution During Startup

Prior to refueling, the Reactor Coolant System is filled with borated [2000 ppm (min.)] water from the refueling water storage tank by the charging pumps. Core monitoring is by external BF_3 detectors. Mixing of reactor coolant is maintained by operation of the reactor coolant pumps. Again the maximum dilution flow (300 gpm) is considered. The volume of reactor coolant is approximately 10,390 ft^3 which is the volume of the Reactor Coolant System excluding the pressurizer. High source level and all reactor trip alarms are effective.

The minimum time required to reduce the reactor coolant boron concentration is 1150 ppm at which the reactor could go critical with all rods in is about 3 hours. Once again, this should be more than adequate time for operator action to the high count rate signal, and termination of dilution flow.

In any case, if continued dilution occurs, the reactivity insertion rate and consequences thereof are considerably less severe than those associated with the uncontrolled rod withdrawal analyzed in Subsection II-1, Uncontrolled RCCA Withdrawal from a Subcritical Condition.

With the reactor in automatic control, at full power, the power and temperature increase from the boron dilution results in control group insertion and a decrease in shutdown margin. A continuation of the dilution and rod insertion would cause the rods to reach the minimum rod insertion limit in approximately six minutes. Before reaching this point, however, two alarms would be actuated to warn the operator of the accident condition. The first of these, the LO (rod position) alarm, alerts the operator to initiate normal boration. The other, LO-LO alarm, alerts the operator to follow emergency boration procedures. The LO alarm is set well above the LO-LO alarm to provide for sufficient normal boration without the need for emergency procedures.

With no boration, it takes 14 minutes after the alarm before the total shutdown margin (one per cent) is lost due to dilution. Therefore, plenty of time is available following the alarms for the operator to determine the cause, isolate the reactor water makeup source, and initiate reboration.

If the reactor is in manual control, and the operator takes no action, power and temperature rise to the overtemperature ΔT trip setpoint in approximately two minutes. Prior to this the high temperature alarm would be actuated. In any case, there are approximately 14 minutes available for the operator to terminate dilution before the reactor can return to criticality following the trip.

2.3.3 Conclusions

Because of the procedures involved in the dilution process, an erroneous dilution is considered incredible. Nevertheless, if an unintentional dilution of boron in the reactor coolant does occur, numerous alarms and indications are available to alert the operator to the condition. The maximum reactivity addition due to the dilution is slow enough to allow the operator to determine the cause of the addition and take corrective action before excessive shutdown margin is lost. However, even if corrective action is not taken and the coolant temperature and pressure continue to increase as a result of the power rise, sufficient diversity exists to automatically protect the reactor vessel from overpressure. These methods include the high nuclear flux rate trip, ΔT overpower trip, ΔT overpressure trip, high pressure trip, high pressurizer water level trip, pressure relief and safety valves, feedwater actuation for core cooling, and steam dump actuation.

2.4 Loss of Reactor Coolant Flow

2.4.1 General

A loss of coolant flow incident may result from a mechanical or electrical failure in one or more reactor coolant pumps, or from a fault in the power supply to these pumps. If the reactor is at power at the time of the incident, the immediate effect of loss of coolant flow is a rapid increase in coolant temperature. This increase could result primarily in departure from nucleate boiling (DNB) with subsequent fuel damage if the reactor is not tripped promptly. The effect of this accident on coolant pressure and hence on reactor vessel integrity is also considered here.

The following trip circuits provide the necessary protection against a loss of coolant flow incident and are actuated by:

a. Low voltage or low frequency on pump power supply bus

b. Pump circuit breaker opening

c. Low reactor coolant flow

These trip circuits and their redundancy are further described in the FSAR, Section 7.2, Reactor Control and Protection System.

2.4.2 Method of Analysis

The following loss of flow cases are analyzed:

a. Loss of four pumps from 2,758 MWt during four loop operation.

b. Loss of one pump from 2,758 MWt during four loop operation.

c. Loss of three pumps from 2,068 MWt during three loop operation.

d. Loss of one pump from 2,068 MWt during three loop operation.

The normal power supplies for the pumps are the four buses connected to the generator, each of which supplies power to one of the four pumps. When a turbine trip occurs, the pumps are automatically transferred to the buses supplied from an external power line, and the pumps will continue to supply coolant flow to the core. The simultaneous loss of power to the four reactor coolant pumps is a highly unlikely event. Since the pumps are on separate buses, a single bus fault would also result in the loss of only one pump.

A full plant simulation is used in the incident analysis to compute the core average and hot spot heat flux transient responses. The model includes flow coastdown, temperature, reactivity, and control rod insertion effects. Results of the plant simulation are then used in a detailed thermal-hydraulic computation. This computation solves the continuity, momentum, and energy equations of fluid flow together with the W-3 DNB correlation discussed in Section 3.2.2 of the FSAR. The following assumptions are made in the calculations:

Initial Operating Conditions

The initial operating conditions, which are assumed to be most adverse with respect to the margin to DNB, are maximum steady state power level, minimum steady state pressure, and maximum steady state inlet temperature:

2,758 MWt - 4 loop operation:

Power	(1.02) (2,758 MWt) = 2813 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	543 + 4 = 547°F

2,068 MWt - 3 loop operation:

Power	(1.02) (2,068 MWt) = 2,110 MWt
Pressure	2250 - 30 = 2220 psia
Inlet Temperature	554 + 4 = 548°F

Reactivity Coefficients

A conservatively high absolute value of the Doppler (-1.6×10^{-5} $°F^{-1}$) and a zero moderator temperature coefficients were assumed since these result in the maximum hot spot heat flux during the transient.

Reactor Trip

For the one pump loss of flow incidents, the reactor trip is assumed to be actuated by the redundant flow monitoring channel (2/3) since this results in the largest delay to reactor trip. For the four and three pump loss of flow incidents, the reactor trip was assumed to be actuated by redundant bus undervoltage or breaker trip (1/4 or 1/3).

The low flow trip setting is 90 per cent of full flow; the trip signal is assumed to be initiated at 87 per cent of full flow, allowing 3 per cent for flow instrumentation errors. Upon reactor trip it is assumed that the most reactive RCCA is stuck in its fully withdrawn position, hence resulting in a minimum insertion of negative reactivity. The negative reactivity insertion upon trip is conservatively based on a 1 per cent shutdown margin at no load conditions.

Heat Transfer Coefficient

The overall heat conductance between the fuel and water regions varies considerably during the transient mostly as a result of the change of fuel gap conductance. The overall heat conductance was conservatively evaluated during the transient by a detailed calculation of fuel rod heat transfer. The hot spot heat transfer coefficient was increased 10% above the design value to obtain a conservatively high heat flux response.

2.4.3 Results - Most Severe Credible Loss of Flow Accident

Even for the most severe credible loss-of-coolant flow condition, which is a simultaneous loss of electrical power to all reactor coolant pumps when the reactor is operating at full power, reactor trip together with flow sustained by the inertia of the coolant and rotating pump parts will be sufficient to prevent overpressurization of the reactor vessel. (Details of the results are given in the FSAR, Section 14.1.6.).

Therefore, the vessel will not be damaged as a result of the most severe credible loss-of-coolant flow accident.

However, the locked rotor accident, a special case considered to be of low probability of occurrence but of greater potential importance to vessel integrity, is discussed below.

2.4.4 Locked Rotor Accident

2.4.4.1 General

A transient analysis is performed for the postulated instantaneous seizure of a reactor coolant pump rotor. Flow through the reactor coolant system is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the trip, heat stored in the fuel rods continues to pass into the core coolant, causing the coolant to expand. The rapid expansion of the coolant in the reactor core, combined with the reduced heat transfer to the secondary system causes an insurge into the pressurizer and a pressure increase throughout the Reactor Coolant System and consequently in the pressure vessel. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and eventually opens the pressurizer safety valves, in that sequence. The two power-operated relief valves are designed for reliable operation and would be expected to function properly during the accident. However, for conservatism, their pressure-reducing effect is not included in the analysis.

The locked rotor analysis was performed for both four loop and three loop operation.

2.4.4.2 Method of Analysis

Initial Conditions

At the beginning of the postulated locked rotor accident, i.e., at the time the shaft in one of the reactor coolant pumps is assumed to seize, the plant is assumed to be in operation under the most severe steady state operating conditions. These initial conditions are the same as the ones described above.

Evaluation of the Pressure Transient

A detailed digital code was used to determine the peak pressure in the Reactor Coolant System under the postulated accident conditions and to obtain the nuclear power as a function of time which is used elsewhere in the analysis. For four pump operation, the coolant flow through the core is conservatively assumed to be reduced from full power to 70% of its initial value at the time of the pump seizure. For three pump operation, it is reduced to 60% of its initial value (which is 71% of nominal flow).

After pump seizure, nuclear power is rapidly reduced because of the control rod insertion upon plant trip and void shutdown due to bulk boiling.

No credit was taken for the pressure-reducing effect of the pressurizer relief valves, steam dump and controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The safety valves start operating at 2485 psig and their combined capacity for steam and water relief is, respectively, 42 and 14 ft^3/sec.

The peak pressure is 2401 psig, reached at 1.6 second for 4 pump operation and 2493 psig, reached at 2.3 second for 3 pump operation.

2.4.4.3 Conclusions - Locked Rotor Accident

a. The peak pressure of (2493 psig) for the worst case is within the $\pm$ 1 percent error allowance specified in the Technical Specifications, Section 3.1 and ensures that the integrity of the primary coolant system and the reactor pressure vessel is not endangered and can be considered as an upper limit, considering the conservative assumptions used in the study:

   1) Credit was not taken for the negative moderator coefficient

   2) It was assumed that the pressurizer relief valves were inoperative

3) The coolant flow in the core was assumed to drop suddenly (0.1 sec or less) from its initial valve to its final valve

4) No DNB was assumed in the core which gives the highest average heat flux during the accident.

2.5 Loss of Normal Feedwater

2.5.1 General

A loss of normal feedwater (from a pipe break, pump failures, or valve malfunctions) results in a reduction in capability of the secondary system to remove the heat generated in the reactor core leading to possible overpressurization of coolant in the reactor vessel. If the reactor were not tripped during this accident primary plant damage including the reactor vessel could possibly occur from a sudden loss of heat sink. If an alternate supply of feedwater were not supplied, residual heat following reactor trip would heat the primary system water to the point where water relief from the pressurizer occurs.

The following provides the necessary protection against a loss of normal feedwater.

a. Reactor trip on very low water level in any steam generator

b. Reactor trip on steam flow-feedwater flow mismatch in coincidence with low water level in any steam generator

c. Two motor driven auxiliary feedwater pumps (400 gpm each) which are started on:

1) Low-low level in any steam generator
2) Trip of any main feed pump turbine
3) Any safety injection signal
4) Manually
5) Loss of outside power

d. One turbine driven pump (800 gpm) which is started on:

1) Low-low level in any two steam generators
2) Loss of outside power
3) Manually

The motor driven auxiliary feedwater pumps are supplied by the diesels if a loss of outside power occurs and the turbine-driven pump utilizes steam from the secondary system. Both type pumps start within one minute. The turbine exhausts the secondary steam to the atmosphere. The auxiliary pumps take suction from the condensate storage tank for delivery to the steam generators.

The above units provide considerable backup in equipment and control logic to ensure that reactor trip and automatic auxiliary feedwater flow will occur following any loss of normal feedwater including that followed by loss of outside power.

2.5.2 Method of Analysis

The analysis has been performed to show that following a loss of normal feedwater, the auxiliary feedwater system is adequate to remove stored and residual heat to prevent water relief through the pressurizer relief valves.

The following assumptions were made:

a. The initial steam generator water level (in all steam generators) at the time reactor trip occurs is at the lowest level which will result in reactor trip and automatic initiation of auxiliary feedwater flow.

b. The plant is initially operating at 102% of 3216.5 MWt (the maximum calculated turbine rating.

c. A conservative core residual heat generation based upon long term operation at the initial power level preceding the trip.

d. Only one motor driven auxiliary feedwater pump is available at one minute after the accident.

e. A conservatively low heat transfer coefficient in the steam generator assuming reactor coolant system natural circulation.

f. Secondary system steam relief through the self-actuated safety valves (steam relief) will, in fact, be through the power operated relief valves or condenser dump valves for most cases of loss of normal feedwater. However, these were not assumed available in the analysis.

2.5.3 Results

Following the reactor and turbine trip from full load, the water level in the steam generators will fall due to the reduction of steam generator void fraction and because steam flow continues to dissipate the stored and generated heat. One minute following the beginning of the accident the auxiliary feedwater pump is automatically started reducing the rate of water level decrease. The capacity of the auxiliary feedwater pump is such that the water level in the steam generators being fed does not recede below the lowest level at which sufficient heat transfer area is available to dissipate core residual heat without water relief from the primary system relief or safety valves.

At no time is there water relief from the pressurizer. The assumption of more auxiliary feed capacity than that of one motor driven pump, a lower reactor power (2758 MWt) or any steam generator water level initially above the low-low level trip will of course result in increased margin to the point at which reactor coolant water relief occurs.

2.5.4 Conclusion

These results show that the reactor vessel is protected against overpressurization and overtemperature from the loss of feedwater accident.

2.6 Loss of External Electrical Load

2.6.1 General

A loss of external load accident can lead to an increase in Reactor Coolant System temperature and pressure and hence reactor vessel internal pressure above design limits unless relieved by various means as discussed below.

The loss of external electrical load may result from an abnormal increase in network frequency, or an accidental opening of the main breaker from the generator which fails to cause a turbine trip but causes a rapid large load reduction by the turbine governor control.

The plant is designed to accept a 50 per cent step loss of load without actuating a reactor trip. The automatic steam bypass system with 40 per cent dump capacity is able to accommodate this abnormal load rejection by reducing the transient imposed upon the reactor coolant system. The reactor power is reduced to the new equilibrium power level at a rate consistent with the capability of the rod control system. The pressurizer relief valves may be actuated, but the pressurizer safety valves and the steam generator safety valves are not actuated in this case.

In the event the steam bypass valves fail to open following a large load loss, the steam generator safety valves are actuated and the reactor may be tripped by the high pressurizer pressure signal or the high pressurizer level signal. The steam generator shell side pressure and reactor coolant temperatures increase rapidly. The pressurizer safety valves are sized to protect the reactor coolant system against overpressure without taking credit for the steam bypass system.

The most likely source of a complete loss of load on the Nuclear Steam Supply System is a trip of the turbine-generator. In this case there is a direct reactor trip signal derived from turbine autostop oil pressure (a two out of three signal). Reactor coolant temperatures and pressure do not increase if the steam bypass system and pressurizer pressure control system are functioning properly. However, the plant behavior is also evaluated for a complete loss of load from full power without a direct reactor trip, primarily to show the adequacy of the pressure relieving devices as well as to show that no equipment damage occurs. The Reactor Coolant System and Steam System pressure relieving capacities are designed to ensure the safety of the plant without requiring the automatic rod control, pressurizer pressure control, and/or steam bypass control systems.

2.6.2 Method of Analysis

The total loss of load transients are analyzed by employing a detailed digital computer program. This code describes the neutron kinetics, decay heat, Reactor Coolant System with pressurizer, steam generators, and the associated steam bypass system and rod control system.

The objectives of this analysis are to determine margins to core protection limits and to establish pressure relieving requirements for the Reactor Coolant and Steam Systems, thereby including the reactor vessel.

2.6.3 Initial Operating Conditions

The initial reactor power, coolant temperatures and pressure are all assumed at extreme values consistent with steady state, full power operation, including allowances for calibration and instrument errors. This results in the maximum power difference for the load loss, and the minimum margin to protection limits at the initiation of the total loss of load accident.

2.6.4 Moderator and Doppler Coefficients of Reactivity

The total loss of load is analyzed for both beginning-of-life and end-of-life conditions.

At beginning-of-life the least negative value of moderator coefficient is used with the least negative value of Doppler coefficient. This results in a maximum nuclear power increase following the loss of load. At end-of-life the most negative value of moderator coefficient is used with the most negative value of Doppler coefficient. This results in a least shutdown margin following reactor trip.

2.6.5 Reactor Control

Two cases are analyzed:

a. The reactor is assumed to be in normal automatic control with the control rods in the minimum incremental worth region.

b. The reactor is assumed to be in manual control. There is no control rod insertion following the accident.

2.6.6 Steam Release

Also analyzed for two cases:

a. The steam dump system is assumed to be in normal automatic control.

b. No credit is taken for steam dump. The steam generator pressures rise toward the safety valve set point where steam release through safety valves limits secondary steam pressure at the set point.

2.6.7 Pressurizer Spray and Power Operated Relief Valves

Full credit is taken in evaluating margins to DNB for the effect of pressurizer spray and relief valves in reducing or limiting coolant pressure since this may prolong the high pressure reactor trip. A second case is analyzed where no credit is taken for pressure control, and pressurizer safety valves may be actuated during the transient.

2.6.8 Results of Transient Analysis

The transient responses for a total loss of load from full power operation are shown for four cases, two cases for beginning of core life and two cases for end of core life.

For the loss of load accident at beginning-of-life with zero moderator coefficient, 40 per cent steam bypass capacity was assumed. Full credit is taken for the effect of pressurizer spray and relief valves in reducing or limiting coolant pressure. Credit is also taken for the effect of control rods insertion in reducing the nuclear power to prolong the time to a high pressure trip. It is seen from the results that the power operated relief valve capacity is large enough to limit the pressurizer pressure at 2350 psia and prevent a high pressure trip. The peak increase in coolant average temperature is about 33°F. The high level trip will be actuated at about 35 seconds following the accident. The high ΔT trip will be actuated at about 50 seconds with a minimum DNB ratio of 1.69 which is well above the 1.3 design value.

For total loss of load at end-of-life with the most negative moderator coefficient (-3.5×10^{-4} $\delta k/°F$), the rest of the plant operating conditions are the same as the case shown before. The pressurizer pressure increases to 2330 psia initially. The spray valves are fully open to limit the pressure at this value. The pressure decreases rapidly after about 30 seconds resulting from the large reduction in nuclear power. The increase in coolant average temperature is about 16°F. No reactor trip will be actuated for this case. However, the DNB ratio is higher than at the beginning-of-life condition because of the large nuclear power reduction which results from the large negative moderator coefficient and the control rod insertion.

The total loss of load accident was also studied assuming the plant is operating at full power with manual control. There is no control rod insertion following the accident. Pressurizer spray, relief valves and steam bypass valves are all ignored. The reactor is tripped on the high pressure signal which is set at 2400 psia. The nuclear power remains at constant full power before the reactor is tripped. The peak pressure is 2450 psia and the maximum surge rate is about 23 ft^3/sec. This is compared to a pressurizer safety valve capacity of approximately 42 ft^3/sec. However, the trip prevents any safety valve actuation. At end-of-life, the nuclear power decreases before the reactor trips as a result of the large negative moderator coefficient. The peak pressurizer pressure is 2440 psia and the maximum surge rate is about 20 ft^3/sec.

2.6.9 Conclusions

The analysis indicates that a total loss of load without a direct or immediate reactor trip presents nó hazard to the integrity of the Reactor Coolant System and the Steam System and the reactor vessel. Pressure relieving devices incorporated in the two systems are more than adequate to limit the maximum pressures. The integrity of the core is maintained by the high pressurizer pressure reactor trip.

2.7 Loss of All A.C. Power To The Station Auxiliaries

2.7.1 General

In the unlikely event of a complete loss of all off-site a-c power and turbine trip while the reactor plant is at power, the Reactor Coolant System pressure could increase above the design pressure with potential effects on the reactor vessel. However, as shown below, sufficient safety measures are available to correct the situation, protecting the vessel from overpressure.

The first few seconds of the transient would be almost identical to the four pump loss of flow case presented earlier, that is, the pump coastdown inertia and reactor trip would result in a DNBR $\geq$ 1.3. After the trip, decay heat will be accommodated by the emergency feedwater system. This portion of the transient would be similar to that presented above under the loss of normal feedwater accident.

a. Plant vital instruments are supplied by the emergency power sources. (Described in detail in Section 8 of the FSAR.)

b. As the steam system pressure subsequently increases, the steam system power relief valves are automatically opened to the atmosphere. Steam bypass to the condenser is not available because of loss of the circulating water pumps.

c. As the steam flow rate through the power relief valves may not be sufficient, the steam generator self-actuated safety valves may temporarily lift to augment the steam flow until the rate of heat dissipation is sufficient to carry away the sensible heat of the fuel and coolant above no-load temperature plus the residual heat produced in the reactor.

d. As the no-load temperature is reached, the steam system power relief valves are used to dissipate the residual heat and to maintain the plant at the hot shutdown condition.

The loss of normal feedwater supply signals the start of the auxiliary feedwater pumps. The turbine driven pump utilizes steam from the secondary system to drive the feedwater pump to deliver makeup water to the steam generators. The turbine driver exhausts the secondary steam to the atmosphere. The electric motor driven auxiliary feedwater pumps are supplied with power by the diesel generators. The pumps take suction from the condensate storage tank for delivery to the steam generators.

Following the turbine trip, there is a rapid reduction of steam generator water level. This is due to the reduction of steam generator void fraction on the secondary side and because steam flow continues after normal feedwater stops. After one minute, flow is established from at least one auxiliary feedwater pump and further reduction of water level is slow. The capacity of the auxiliary feedwater pump is selected to prevent the water level in the steam generators being fed from receding below the lowest level within the indicator range during the transient. The reactor operator in the control room monitors the steam generator water level and controls the feedwater addition with remote operated auxiliary feedwater control valves.

The steam driven feedwater pump can be tested at any time by admitting steam to the turbine driver. The electrically driven auxiliary feedwater pumps also can be tested at any time. The auxiliary feedwater control valves and power relief valves can be operationally tested whenever the plant is at hot shutdown and the remaining valves in the system are operationally tested when the turbine driver and pump are tested.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant loops. The natural circulation capability for the unit has been calculated for the conditions of equilibrium flow and maximum loop flow impedence. The analytical model used to calculate the natural circulation flow has given results within 15% of the measured flow values obtained during natural circulation tests conducted at the Yankee-Rowe plant and has also been confirmed at San Onofre and Connecticut Yankee.

2.8 Rupture of a Steam Pipe

2.8.1 General

A rupture of a steam pipe is assumed to include any accident which results in an uncontrolled steam release from a steam generator. The release can occur due to a break in a pipe line or due to a valve malfunction. The steam release results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the Reactor Coolant System causes a reduction of coolant temperature and pressure. With a negative moderator temperature coefficient, the cooldown results in a reduction of core shutdown margin. If the most reactive control rod is assumed stuck in its fully withdrawn position, there is a possibility that the core will become critical and return to power even with the remaining control rods inserted. In such a case, the possibility of overpressure, with consequential deleterious effect on reactor vessel integrity, must be considered.

A return to power following a steam pipe rupture is a potential problem only because of the high hot channel factors which may exist when the most reactive rod is assumed stuck in its fully withdrawn position. Assuming the most pessimistic combination of circumstances which could lead to power generation following the steam line break, the core is ultimately shut down by the boric acid in the Safety Injection System and the reactor vessel is protected against overpressure.

The analysis of a steam pipe rupture was performed to demonstrate that with a stuck rod and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

The following systems provide the necessary protection against a steam pipe rupture:

a. Safety Injection System Actuation from any one of the following:*

   1) One out of three pressurizer coincident low pressure and low level signals

   2) Six sets of two out of three high differential pressure signals between steam lines

   3) High steam flow in two out of four lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (three out of four) or low steam line pressure (three out of four)

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* The details of the logic used to actuate Safety Injection are discussed in Section 7.2 of the FSAR.

4) Two out of three high containment pressure signals

b. The overpower reactor trips (nuclear flux and ΔT) and the reactor trip occurring upon actuation of the Safety Injection System.

c. Redundant isolation of the main feedwater lines. Sustained high feedwater flow would cause additional cooldown, thus in addition to the normal control action which will close the main feedwater valves, any safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps, and close the feedwater pump discharge valves.

d. Closing of the fast acting steam line stop valves (designed to close in less than 5 seconds even with no flow) on:

1) High steam flow in two out of four lines (one out of two per line) in coincidence with either low reactor coolant system average temperature (three out of four) or low steam line pressure (three out of four).

2) Two out of three high containment pressure signals.

Each steam line has a fast closing stop valve and a check valve. These eight valves prevent blowdown of more than one steam generator for any break location even if one valve fails to close. For example, for a break upstream of the stop valve in one line, closure of either the check valve in that line or the stop valves in the other lines will prevent blowdown of the other steam generators.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles (16" I.D. vs a pipe diameter of 28" I.D.) are located inside the containment near the steam generators and also serve to limit the maximum steam flow for any break further downstream. In particular, the nozzles limit the flow for all breaks outside the containment and those inside the containment which are downstream of the flow measuring nozzles.

2.8.2 Method of Analysis

The analysis of the steam pipe rupture has been performed to determine the core heat flux and reactor coolant system temperature and pressure resulting from the cooldown following the steam line break. A full plant digital computer simulation has been used.

The following assumptions were made:

a. The rods give 0.0195 shutdown reactivity at no load. This is the end-of-life design value including design margins with the most reactive rod stuck in its fully withdrawn position. The actual shutdown capability is expected to be significantly greater.

b. The negative moderator coefficient corresponding to the end-of-life core with all but the most reactive rod inserted. The variation of the coefficient with temperature and pressure has been included. In computing the power generation following a steam line break, the local reactivity feedback from the high neutron flux in the region of the core near the stuck control rod has been included in the overall reactivity balance. The local reactivity feedback is composed of Doppler reactivity from the high fuel temperatures near the stuck control rod and moderator feedback from the high water enthalpy near the stuck rod. For the cases analyzed where steam generation occurs in the high flux regions of the core, the effect of void formation on the reactivity has been included. The analysis assumes end-of-life core conditions with all rods in except the most reactive rod which is assumed stuck in its fully withdrawn position (completely removed from the core).

c. Minimum safety injection capability corresponding to two out of three safety injection pumps in operation and three out of four safety injection lines available for flow to the reactor coolant system. 20,000 ppm boron is assumed in the boric acid tank at the suction of the Safety Injection pumps. The time delays required to sweep the low concentration boric acid from the safety injection piping prior to the delivery of the 20,000 ppm boron have been included in the analysis.

2.8.3 Conclusions

The analysis demonstrates that consequences of steam pipe rupture can be adequately contained without damage to the primary system, consequently the pressure vessel remains intact.

2.9 Primary System Pipe Rupture

2.9.1 General

A loss-of-coolant accident may result from a rupture of the Reactor Coolant System or of any line connected to that system up to the first closed valve. Ruptures of very small cross-section will cause expulsion of coolant at a rate which can be accommodated by the charging pumps. Should such a small rupture occur, these pumps would maintain an operational level of water in the pressurizer, permitting the operator to execute an orderly shutdown.

Should a larger break occur, resultant loss of pressure and pressurizer liquid level will cause reactor trip and initiation of safety injection. These countermeasures will limit the consequences of the accident in two ways:

a. Reactor trip and borated water injection will supplement void formation in causing rapid reduction of the nuclear power to a residual level corresponding to delayed fissions and fission product decay.

b. Injection of borated water ensures sufficient flooding of the core to prevent excessive temperatures.

The major effect of such an accident on reactor pressure vessel integrity is felt during the safety injection phase. This is analyzed as follows:

2.9.2 Analysis of Effects of Loss of Coolant and Safety Injection on the Reactor Vessel

For the reactor vessel, three modes of failure are considered including the ductile mode, brittle mode and fatigue mode:

a. Ductile Mode - the failure criterion used for this evaluation is that there shall be no gross yielding across the vessel wall using the material yield stress specified in Section III of the ASME Boiler and Pressure Vessel Code. The combined pressure and thermal stresses during injection through the vessel thickness as a function of time have been calculated and compared to the material yield stress at the times during the safety injection transient.

   The results of the analyses showed that local yielding may occur only in approximately the inner 18 per cent of the base metal and in the vessel cladding, complying with the above criterion.

b. Brittle Mode - the possibility of a brittle fracture of the irradiated core region has been considered from both a transition temperature approach and a fracture mechanics approach.

The failure criteria used for the transition temperature evaluation is that a local flaw cannot propagate beyond any given point where the applied stress will remain below the critical propagation stress at the applicable temperature at that point.

The results of the transition temperature analysis showed that the stress-temperature condition in the outer 65 per cent of the base metal wall thickness remains in the crack arrest region at all times during the safety injection transient. Therefore, if a defect were present in the most detrimental location and orientation (i.e., a crack on the inside surface and circumferentially directed), it could not propagate any further than approximately 35 per cent of the wall thickness, even considering the worst case assumptions used in this analysis.

The results of the fracture mechanics analysis, considering the effects of water temperature, heat transfer coefficients and fracture toughness of the material as a function of time, temperature and irradiation are summarized in Section 5, Attachment 1 of the Additional Testimony of Applicant Concerning Reactor Vessel Integrity. Both a local crack effect and a continuous crack effect have been considered with the latter requiring the use of a rigorous finite element axisymetric code.

c. Fatigue Mode - the failure criterion used for the failure analysis was as presented in Section III of the ASME Boiler and Pressure Vessel Code. In this method the piece is assumed to fail once the combined usage factor at the most critical location for all transients applied to the vessel exceeds the code allowable usage factor of one.

The results of this analysis showed that the combined usage factor never exceeded 0.2, even after assuming that the safety injection transient occurred at the end of plant life.

In order to promote a fatigue failure during the safety injection transient at the end of plant life, it has been estimated that a wall temperature of approximately 1100°F is needed at the most critical area of the vessel (instrumentation tube welds in the bottom head).

The design basis of the Safety Injection System ensures that the maximum Zircaloy cladding temperature does not exceed the Zircaloy-4 melt temperature. This is achieved by prompt recovery of the core through flooding, with the passive accumulator and the injection systems. Under these conditions a vessel temperature of 1100°F is not considered a credible possibility and the evaluation of the vessel under such elevated temperatures is for a hypothetical case.

For the ductile failure mode, such hypothetical rise in the wall temperature would increase the depth of local yielding in the vessel wall.

2.9.3 Safety Injection Nozzles

The safety injection nozzles have been designed to withstand ten postulated safety injection transients without failure. This design and associated analytical evaluation was made in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code.

The maximum calculated pressure plus thermal stress in the safety injection nozzle during the safety injection transient was calculated to be approximately 50,900 psi. This value compares favorably with the code allowable stress of 80,000 psi.

These ten safety injection transients are considered along with all the other design transients for the vessel in the fatigue analysis of the nozzles. This analysis showed the usage factor for the safety injection nozzles was 0.47 which is well below the code allowable value of 1.0.

The safety injection nozzles are not in the highly irradiated region of the vessel and thus they are considered ductile during the safety injection transient.

Evaluations of the core barrel and thermal shield have also shown that core cooling is not jeopardized under the postulated accident conditions.

2.9.4 Conclusions

The results of these analyses show that the integrity of the reactor vessel is never violated as a result of primary system pipe rupture.

2.10 Rupture of a Control Rod Mechanism Housing-RCCA Ejection

2.10.1 General

In order for this accident to occur, a rupture of the control rod mechanism housing must be postulated creating a full system pressure differential acting on the drive shaft. The resultant core thermal power excursion is limited by the Doppler reactivity effect of the increased fuel temperature and terminated by reactor trip actuated by high nuclear power signals. However, the magnitude of the resultant pressure transient and the possibility of fuel or clad melting adding additional energy to the system and to the reactor vessel must be considered.

A failure of a control rod mechanism housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

a. Each control rod drive mechanism housing is completely assembled and shop-tested at 4100 psi.

b. The mechanism housing will be individually hydrotested to 3105 psig as they are installed on the reactor vessel head to the head adapters, and checked during the hydrotest of the completed Reactor Coolant System.

c. Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class A components.

However, a detailed analysis of the transients following such an accident have been performed to determine their effects on the core and the primary circuit, including the reactor vessel.

The operation of a chemical shim plant is such that the severity of an ejection accident is inherently limited. Since control rod clusters are used to control load variations only and core depletion is followed with boron dilution, there are only a few rods in the core at full power. Proper positioning of these rods is monitored by a control room alarm system. There are low and low-low insertion monitors (see Section 7.3 of the FSAR) with visual and audio signals. Operating instructions require boration at the low level alarm and emergency boration at the low-low alarm. The control rod position monitoring alarm systems are described in detail in the FSAR, Section 7.3. By utilizing the flexibility in the selection of control rod cluster groupings, radial locations and position as a function of load, the design minimizes the peak fuel and clad temperatures for the highest worth ejected rod. It is shown that no fuel or clad melting occurs. Overinsertion of the control rods constitutes a violation of the operating instructions. However, for completeness sensitivity to overinsertion was considered. For the worst case, which is at full power at the end-of-life, it is shown that considerable margin exists. In fact, the C4 control bank may be fully inserted without causing clad melting on ejection.

This section describes the models used and the results obtained. The worst cases are presented in detail.

Only the initial few seconds of the power transient are discussed, since the long term considerations are the same as for a loss of coolant accident.

2.10.2 Method of Analysis

The calculation of the transient is performed in two stages, first an average core calculation and then a hot region calculation. The average core is analyzed to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator density reactivity. The largest temperature rises, (and hence the largest joint Doppler feedbacks) occur in channels where the power is higher than average. Since the importance of a region is dependent on flux, these regions also have high importance. This means that the reactor Doppler feedback is larger than that predicted by a simple single channel analysis. Physics calculations have been carried out for temperature changes with a flat temperature distribution, and with a large number of axial and radial temperature distributions. Reactivity changes were compared and effective weighting factors determined. These weighting factors take the form of multipliers which, when applied to single channel Doppler feedbacks, correct them to effective whole core feedbacks for the appropriate flux shape. Enthalpy and temperature transients in the hot spot are determined by adding a multiple of the average core energy generation to the hotter rods and performing a transient heat-transfer calculation. The asymptotic power distribution calculated without feedback is pessimistically assumed to persist throughout the transient.

2.10.3 Average Core Analysis

The nuclear power transients are calculated using the CHIC-KIN code developed by the Bettis Atomic Power Laboratory for similar analyses. This code solves the point kinetics equations, with feedback from an axially and radially segmented fuel element. CHIC-KIN results have been compared with SPERT results for two dissimilar cores over a wide range of periods with good agreement.

For this study, six delayed neutron groups were used and the fuel rod was divided into eight radial increments, with a ninth increment for the clad. Five axial segments were employed. The calculation is essentially a single channel analysis representing the core average conditions.

Prompt heat generation directly in the coolant has been calculated to be 2.6 percent of the nuclear power generation using the LEOPARD code. Heat generation in the fuel pellet is assumed to occur non-uniformly radially with a slight reduction in the center due to self-shielding effects.

2.10.4 Hot Region Analysis

The average core energy addition calculated as described above is multiplied by the appropriate hot channel factors and the worst cases analyzed using a detailed heat transfer code. The code contains a representation of gap conductance as a function of fuel temperature, clad temperature and differential pressure. The zirconium water reaction is explicitly represented and all material properties are represented as functions of temperature. The Tong, Sandberg and Bishop correlation (as described in Section 3.2.2 of the FSAR) is used to determine the film boiling heat transfer coefficient. The results indicate that no clad melting occurs.

2.10.5 Selection of Input Parameters

Pessimistic values for all the input parameters were selected on the basis of calculated values and a parameter study. The parameter study indicates that the parameters to which the transient is most sensitive are the ejected rod worth, the Doppler broadening the delayed neutron fraction and transient hot channel factors. A margin of 10% has been added to the ejected rod worth. The Doppler broadening has been reduced by 10% to allow for errors in the weighting calculation and a further 10% for errors in the basic Doppler coefficient. The value of the delayed neutron fraction was not reduced since the method of calculation introduced conservatism, and in any case, a 10% conservatism had already been added to the ejected rod worth. Transient hot channel factors were conservatively calculated without taking any credit for the flattening effects of feedback.

The values of the ejected rod worth and transient hot channel factors are dependent on the positioning of the part length rods. The overall ejected rod worth is arrived at by carrying out two separate calculations, one for the reactor slab containing the part length rods and one for the non-part length slab, and then determing a weighted total. The ejected rod worth and radial hot channel facts for the part length region are always higher than those for the non-part length region. Since the regional weighting increases with increasing regional flux, the worst overall ejected rod worth is obtained with the part length rods in the axial power peak. However, in general this condition does not result in the maximum value of Fq. If the part length rods are located in the axial flux peak, as they will be under normal conditions, the radial and axial peaks will be coincident. If the part length rods are moved out of the axial peak, the axial hot channel factor will increase. However, the axial peak is now in the non-part length region where the radial peaking factor is lower. This effect tends to reduce the dependence of Fq on the part length position, but the net result is often to increase Fq when the part length rods are moved away from the axial peak.

It can be seen that the worst part length rod position from the point of view of ejected rod worth probably does not correspond to the worst condition for Fq. Analyses indicate that the worst hot spot transient occurs when the part length rods are located in the axial peak. The ejected rod worth is the dominant effect. All analyses have been conducted with the worst part length rod position with respect to the hot spot transient.

2.10.6 Additional Conservatism in the Model

Apart from the conservatism resulting from the pessimistic parameter selection, considerable conservatism exists in the model for the reasons noted below.

It is assumed that the transient and steady-state hot spots are coincident. This means that the largest temperature rise is combined with the highest steady-state temperature. In practice, the transient peak occurs in the immediate area of the ejected control rod, where under steady-state conditions the power is well below normal.

The Doppler feedback calculation is based on the asymptotic power distribution with feedback (i.e., flattened power distribution), but the hot spot transient power is computed on the basis of the asymptotic power distribution without feedback.

Detailed physics calculations indicate that the total core moderator feedback will be considerably larger than that predicted on the basis of the average channel. However, the moderator feedback is pessimistically assumed equal to the average core value. Physics calculations were carried out to determine whether it is possible for local pockets of positive temperature coefficients to occur. An infinite medium moderator temperature coefficient was obtained by performing two 2-dimensional unit cell calculations with no transverse buckling. The calculations were carried out at two different temperature coefficients.

For the conditions which give the least negative temperature coefficient for the core as a whole, the above method resulted in a local infinite moderator coefficient of zero. For positions in the core which have a higher than average local power density, there will be a net leakage out of the region, and so the local temperature coefficient will be more negative than this calculation indicates.

Reactivity insertion as a result of lattice deformation was considered. In the region of the hot spot there will be a large power gradient. Since the fuel rods are free to move in a vertical direction, differential expansion between separate rods cannot produce distortion. However, the temperature gradient across individual rods may produce a force tending to bow the midpoint of the rods toward the hot spot. Physics calculations indicate that the net result of this would be a negative reactivity insertion.

Boiling in the hot spot region will produce a net fluid flow away from that region. However, the fuel heat is released to the water relatively slowly, and it is considered inconceivable that cross flow will be sufficient to produce significant lattice forces. Even if massive and rapid boiling sufficient to distort the lattice is hypothetically postulated, the large void fraction in the hot spot region would produce a reduction in the total core moderator to fuel ratio and a large reduction in this ratio at the hot spot. The net effect would therefore be a negative feedback. It is concluded that no conceivable mechanism exists for a net positive feedback resulting from lattice deformation. In fact, a small negative feedback may result. The effect is ignored in the following analyses.

2.10.7 Cases Considered

In general the end of life cases are worse than the beginning of life cases. This is because the delayed neutron fraction is smaller at the end of life, and also because ejected rod worths and hot channel factors tend to be worse at the end of life. However, for completeness, the full (2758 MWt) and zero power cases for beginning of life were also considered.

At the end of life, ejection of the worst control bank rods at full and zero power were studied as end point cases. At the end of life, both the worst ejected rod worth and worst hot channel factor occur when two banks are fully inserted with the third bank at the bottom limit of its solo movement (i.e., immediately before the fourth control bank begins to move in). The rod program is such that the maximum power at which this situation may occur is 4% of full power. An analysis has been carried out for this condition. The ejection of a part length rod was also considered.

2.10.8 Beginning of Life Full Power

The rod program limits the control bank holding to 0.5% ΔK for this condition. The reactor is sub-prompt critical with the worst ejected rod worth of 0.156%. The peak power reached is 1.27 times normal full power, and the peak hot spot clad and center fuel temperatures are respectively 1700°F and 4410°F.

2.10.9 Beginning of Life Zero Power

For this condition there will be one control bank fully inserted, and a second bank partially inserted. For conservatism the worst ejected rod from two fully inserted banks were used. The value of 0.75% ΔK results in the core becoming weakly prompt critical. At the peak hot spot heat flux, film boiling would not occur. (Even through only two of the four main coolant pumps are assumed to be running). The peak hot spot center fuel temperature is 1880°F.

2.10.10 End of Life Full Power, Ejection of the Worst Control Bank Rod

Again the rod program limits the control bank reactivity holding to 0.5% ΔK. The worst ejected rod worth is then 0.2% ΔK. This results in a peak power of 1.55 times the normal full power and the peak hot spot heat flux of 684,000 Btu/ft^2 hr (1.2 times design maximum). This heat flux will not result in film boiling. However, two hot spot cases were considered, one with DNB and one without DNB. For the case with DNB the peak clad and center fuel temperatures were respectively 1690°F and 4300°F.

Based on a steady-state hot channel factor of 3.23 (design), and a coincident transient hot channel factor of 3.8, the model indicates that an ejected rod of worth 0.3% ΔK would be required to initiate center melting. For this condition the peak cladding temperature would be less than 2100°F. In practice, the transient and steady-state hot spots cannot be coincident. It is therefore, concluded that a 0.3% ΔK ejected rod can be tolerated with some considerable margin. The ejected rod worth for full power operation with the C4 control bank fully inserted in 0.28 ΔK. This condition constitutes a large deviation from the operating instructions. A further insertion of the control rods, beyond this condition, would of course increase the reactivity of the accident, and would result in a limited amount of fuel melting on the pellet center line at the leak spot. This would not produce fuel dispersal in the coolant. A considerable margin to clad melting would still exist.

2.10.11 Part Power End of Life

The worst part power ejected rod worth and hot channel factors occur when the first two control banks are fully inserted, and the third bank almost fully inserted. Physics calculations indicate that the maximum possible power at which this condition can occur is 4% of full power. For an ejected rod worth of 0.8% and assuming that only two main coolant pumps are running, this results in a peak cladding temperature of 2150°F and a peak center fuel temperature of 3320°F.

2.10.12 Zero Power End of Life

The worst ejected rod worth at zero power is less than the value of 0.8% used in the part power case above. For conservatism, the value of 0.8% has been used. The peak cladding temperature is 1830°F, and the peak center fuel temperature is 2780°F.

2.10.13 Fission Product Release

It is assumed that fission products are released from the gaps of all rods entering DNB. In all cases considered only a small fraction of the core (less than 2%) enters DNB. The position with regard to fission product release is therefore much better than for the double ended coolant pipe break (the maximum hypothetical accident) for which the majority of the core enters DNB.

2.10.14 Pressure Surge

Because there is no fuel or clad melting, even in the worst case with the most pessimistic assumptions, there is no danger of a sudden pressure rise due to heat transfer from dispersed molten fuel or from massive sudden metal-water reaction. Thus, the pressure conditions can be judged on the basis of relatively conventional heat transfer rates. The most severe excess addition of energy to the coolant occurs for the full power end of life case, and so this case results in the worst pressure transient, average channel and hot spot heat transfer calculations were performed using a high gap conductance and without assuming DNB. The power curves used for these calculations represented a limiting case which almost initiated center melting at the hot spot. Using the heat flux data obtained above, a THINC 3 run was conducted to determine the volume surge (without the benefit of pressure feedback). This volume surge was subsequently used as the basis for a pressure calculation. The results indicated that starting at 2250 psi a peak pressure of about 2340 psi occurs some 1.5 seconds after rod ejection.

2.10.15 Conclusions

Even on the most pessimistic bases, the analyses indicate no fuel or clad melting. Furthermore, the resulting pressure surge is insufficient to produce consequential damage to the primary circuit, including the reactor vessel.

3.0 SYSTEMS AND EQUIPMENT INSTALLED TO INSURE OPERATIONAL CONFORMITY WITH DESIGN AND TECHNICAL SPECIFICATIONS

A number of systems and various types of equipment are available to insure that the plant is operated in conformity with design and Technical Specifications. Those systems and equipment which either directly or indirectly assure maintenance of pressure vessel integrity during operation are briefly described as follows, together with pertinent documentation references:

3.1 Pressure Control

The control of reactor vessel pressure is one of the vital areas of concern for the assurance of safe operation of the vessel within the design pressure limits. Several general methods are provided for this purpose. This section discusses protection of the Reactor Coolant System, and thereby the reactor pressure vessel, specifically against overpressure by means of specialized equipment and circuits such as the pressurizer, relief and safety valving, the high pressure trip, and a high pressure alarm. In addition to these means, the Reactor Control and Protection Systems also limit the Reactor Coolant System pressure to below 2735 psig as required by the Technical Specifications Section 2.2. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System and reactor vessel is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established and hence also the reactor vessel pressure. However, since the Protection System specifically operates on neutron flux, (by negative reactivity insertion) thereby limiting coolant (and pressure vessel) temperature and pressure, and does not act on pressure uniquely, this System will be discussed separately.

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valves set points, and the protection system set point pressures are listed below as extracted from Table 4.1-1 of the FSAR and Section 2.2 of the Technical Specification.

Table B-1

Reactor Coolant System Pressure Settings

	Pressure, psig
Operating Pressure Limit	2235
Pressurizer Spray Valves (open)	2260
High Pressure Alarm	2335
Power Relief Valves	2335
High Pressure Trip	2385
Safety Valves	2485
Design Pressure	2485

As can be seen by the listing in Table B-1, provisions are made for operation of five different sets of pressure relief equipment at four increasing values of pressure between the operating & design values, resulting in a multiple over-pressure protection system.

The setting of the power operated relief valves (2335 psig) and the reactor high pressure trip (2385 psig) have been established to assure that the Reactor Coolant System pressure limit is never reached and that the system pressure does not exceed the design limits of the fuel cladding.

In addition, the Reactor Coolant System safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10 percent (2735 psig) in accordance with Section III of the ASME Boiler and Pressure Vessel Code, assuming complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valves settings.

As an assurance of system integrity, all components in the system are hydrotested at 3110 psig prior to initial operation.

3.1.1 Valving

As part of the pressure control equipment, power operated relief valves, set at 2335 psig, are provided to assure that the Reactor Coolant

System design pressure limit is never reached.

The relief valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1 (FSAR) and the valve design parameters are given in Table 4.1-3 (FSAR), valve sizes are determined as indicated in Section 4.3.4 (FSAR). In addition, self-actuating safety valves are provided to prevent the Reactor Coolant system pressure from exceeding design pressure (2485 psig) without any other operative control. Both types of valves, therefore, directly assure that the pressure vessel will not be operated at excessive pressures. Discussion of this equipment is included in the FSAR Section 4.3.4.

3.1.2 Pressurizer

The pressurizer maintains the required reactor coolant design pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure of 2485 psig, thereby also directly ensuring the integrity of the reactor vessel.

The pressurizer contains multiple safety and relief valves as described above, a spray nozzle and interconnecting piping, valves and instrumentation.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects the pressurizer to the hot leg of a reactor coolant loop. During a positive pressure surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, is set to operate at 2260 psig and condenses steam in the vessel to prevent the pressurizer pressure from reaching 2335 psig, the set point of the power operated relief valves. The spray valves on the pressurizer are power operated. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

This information is further discussed in the FSAR, Sections 4.2.2 and 7.3.

3.1.3 Alarm and Trips

In addition to the equipment discussed above, a high pressure alarm is provided when the system pressure reaches 2335 psig, a point below design pressure, to allow for corrective action by an operator, e.g. by manually opening the power relief valves, inserting rods, etc. The alarm is discussed in the FSAR, Section 7.2-2.

The reactor high pressure trip (2385 psig) is also provided to assure that the Reactor Coolant System pressure limit is never reached. This trip is discussed in the FSAR Section 7.2-2 and Table 7.2-1, and in the Technical Specifications, Section 2.3.

The alarm and trip thus provide direct assurance that the reactor vessel integrity is assured by controlling the maximum pressure of the coolant system through either manual or automatic means as described above.

3.2 Reactor Protection System

The Reactor Protection System provides assurance against failure of the reactor vessel during operation by suppressing or preventing abnormal power conditions, and by maintaining temperature and pressure conditions within the safety margins given in the Technical Specifications, Section 2.

3.2.1 Design Temperature

The design temperature of the reactor pressure vessel is 650°F and is selected to be above the maximum coolant temperature in the vessel under all normal and anticipated transient load conditions, as stated in the FSAR, Section 4.1.4.

3.2.3 General Reactor Protection System Description

The Reactor Protection System is provided to prevent abnormal power conditions, thereby limiting the reactor vessel temperature and pressure to acceptable limits include various tripping functions, rod stops, and alarms.

3.2.4 Tripping Functions

The Reactor Protection System automatically trips the reactor to protect the reactor core under the following conditions as specified in the Technical Specifications, Section 2.3.

i. The reactor power, as measured by neutron flux, reaches a preset limit during starting of 25% of rated power, and 109% of rated power during normal operation.

ii. The temperature rise across the core as automatically determined from loop ΔT reaches a limit from a variable ΔT set point analog automatically compensated for neutron flux distribution. This is the Overpower ΔT trip discussed in the FSAR, Section 7.2.

iii. The temperature rise across the core as automatically determined from loop ΔT reaches a limit from a variable ΔT setpoint as a function of T_{avg} and pressurizer pressure, also adjusted by neutron flux distribution. This is the Overtemperature ΔT trip discussed in Section 7.2 of the FSAR.

iv. The pressurizer pressure reaches an established limit of 2385 psig.

v. Loss of reactor coolant flow as sensed by low flow, loss of pump power or pump breakers opening.

The basic reactor tripping philosophy is to define a region of power and coolant temperature conditions allowed by the primary tripping functions, namely the overpower high ΔT trip the over temperature high ΔT trip, and the nuclear overpower trip. The allowable operating region within these trip settings is provided to prevent any combination of power, temperatures and pressure which would result in departure from nucleate boiling (DNB) with all reactor coolant pumps in operation. Additional tripping functions such as a high pressurizer pressure trip, discussed in Ia, low pressurizer pressure trip, high pressurizer water level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low water level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

The reactor protection system operates analogously to turning off the fire on a pressure cooker. With the exception of the high pressure trip discussed previously, which has a direct influence on the reactor vessel pressure, these tripping functions have an indirect influence on insuring reactor vessel integrity since they serve primarily to maintain the integrity of the core. In performing this function however, the reactor vessel is also protected from experiencing abnormal temperature and pressure conditions which could lead to rupture.

A detailed discussion of these tripping functions is provided in the FSAR, Sections 7.2.2 and 7.2.3, and the Technical Specifications, Section 2.3.

After trip, interlocking functions of the Reactor Protective System inhibit control rod withdrawal on the occurrence of a specified parameter reaching a value lower than the value at which reactor trip is initiated.

3.2.5 Rod Stops

Nuclear overpower, high ΔT and T_{avg} deviation mechanical rod stops are automatically actuated to prevent abnormal power conditions beyond the limits given in the Technical Specifications, Section 2, which could result from excessive control rod withdrawal initiated by malfunction of the reactor control system or by operator violation of administrative procedures. By directly preventing abnormal power conditions, the rod stops prevent abnormal temperature and pressure conditions and therefore indirectly, but importantly, serve to insure operation of the reactor vessel within design limits.

A list of rod stops, actuation signals and rod motion to be blocked is given below:

Table B-2

Rod Stops

Rod Stop	Actuation Signal	Rod Motion to be blocked
Nuclear Overpower	1/4 high power range nuclear flux or 1/2 high intermediate	Automatic and Manual Withdrawal
High ΔT	1/4 overpower ΔT or 1/4 over-temperature ΔT	Automatic and Manual Withdrawal
T_{avg} Deviation	1/4 T_{avg} deviation from average T_{avg}	Automatic Withdrawal and Insertion

3.2.6 Alarms

As listed in the FSAR, Section 7.2.2, any of the following conditions actuate an alarm:

a) Reactor trip (first-out annunciator)

b) Trip of any reactor trip channel

c) Actuation of any permissive circuit or override

d) Significant deviation of any major control variable (pressure, T_{avg}, pressurizer water level, and steam generator water level) beyond the limits set in the Technical Specifications, Section 2.3.

The operators are thereby notified of the abnormal condition and are therefore able to provide for manual initiation of protective system action and to safeguard the reactor vessel in the event of failure in the automatic system.

3.3 Regulating Systems

Regulating Systems are provided for the primary purpose of limiting transients. However, these systems also indirectly protect the reactor vessel against overpressure by their primary action, which controls the coolant temperature and hence, pressure. Overall reactivity control is achieved by the combination of rod cluster control and chemical shim. The means by which these systems function are briefly described as follows:

3.3.1 Reactor Control System

The primary function of the Reactor Control System is to provide automatic control of the rod clusters during power operation of the reactor. The system uses input signals including neutron flux; coolant temperature and pressure; and plant turbine load.

3.3.2 Chemical and Volume Control System

The Chemical and Volume Control System serves as a secondary reactor control system by the addition and removal of varying amounts of boron chemical shim. In addition, this system provides corrosion protection for the reactor vessel, as stated in the Auxiliary System Description below.

Additional discussion of these systems is provided in the FSAR, Sections 7 and 9 and in the Technical Specifications, Sections 3.10 and 4.9.

3.4 Engineered Safety Features

Engineered Safety Features, including the Residual Heat Removal Systems (RHRS), and Component Cooling Systems (CCS), are provided to remove decay heat from the core from the Reactor Coolant System components in normal and emergency shutdown situations. These systems thereby also protect the reactor vessel from possibly exceeding design temperature during shutdown as a result of decay heat.

Details pertaining to the operation and design of these systems are given in the FSAR, Sections 9.1 and 9.3.

The RHRS directly protects the reactor vessel from exceeding the Technical Specifications limits (Section 2.2) by removing decay heat from the core and hence the interior of the reactor vessel. The CCS removes the residual and sensible heat from the Reactor Coolant System via the RHRS, and makes possible the operation of important components of the Reactor Coolant System such as the reactor coolant pumps below their design limits as specified in the Technical Specifications, Sections 2 and 3.3.

3.5 Instrumentation Systems

Plant Instrumentation Systems perform the following functions in order to assure safe and orderly operation of the plant, thereby maintaining reactor vessel integrity:

a. Input signals for reactor control

b. Input for Protection Systems

c. Automatic actuation of Protection Systems

d. Timely warning to operator of the onset of unsafe situations, so that manual safeguards can be initiated

e. Information for records and data collection

These functions are provided by the Nuclear Instrumentation and Process Instrumentation Systems and are discussed in the Technical Specifications, Section 3.5, and in the FSAR, Sections 7.4 and 7.5.

3.5.1 Nuclear Instrumentation System

The Nuclear Instrumentation System indirectly aids in preventing reactor vessel failure by monitoring the reactor power from source range through the intermediate range and power range up to 120 percent full power. The system provides indication, control, and alarm signals for reactor operation and protection.

3.5.2 Process Instrumentation System

The non-nuclear process instrumentation measures temperatures, pressures, flows, and levels in the reactor coolant system, steam system, reactor containment and auxiliary systems. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded and controlled from the control room. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

The monitoring of pressure thus provides a direct means and the monitoring of the other process variables provides indirect, but necessary, means of insuring operation of the reactor vessel within safe limit.

3.6 Chemical and Volume Control System (CVCS)

In addition to the use of this system in adjusting the concentration of chemical reactivity control, as discussed separately in 3.3.2 above, the CVCS is also provided to maintain the proper concentration of corrosion inhibiting chemicals in the reactor coolant, thereby directly influencing the maintenance of the reactor vessel by preventing corrosion which might otherwise weaken the vessel by producing pitting, or by metallic grain boundary penetration. A detailed discussion of the operation of this system is given in the FSAR, Section 9.2.

3.7 Sampling System

This system provides the representative samples for chemical analysis. Typical information obtained which is pertinent to reactor vessel protection includes reactor coolant boron, halide, hydrogen, oxygen, corrosion product and chemical additive concentrations.

By means of this system therefore, the chemistry of the Reactor Coolant System can be monitored and proper feedback given to the CVCS for its functional role as discussed above in 3.3.2 and 3.6.

Thereby, the correct chemical conditions required for reactor vessel corrosion control can be monitored and maintained.

Additional details of the design and operation of this system are given in the FSAR, Section 9.4.

3.8 Steam and Power Conversion Systems

In addition to the components of the Engineered Safety Features discussed in 3.4 above, certain portions of the Steam and Power Conversion Systems are provided to remove heat under power operating conditions, thereby maintaining the reactor vessel temperature and pressure below the design limits.

These are the Auxiliary Feedwater System and the Service Water System.

3.8.1 Auxiliary Feedwater System

Three auxiliary feedwater pumps are available to meet decay heat removal requirements.

Details of this system are discussed in the FSAR, Sections 10.2.6, 14.1.9 and 14.2.5.

3.8.2 Service Water System

The Service Water System is designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. Provision is made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety either during normal operation or under abnormal and accident conditions. Design and operational details are given in the FSAR, Section 9.6.1.

In addition to the river, an adequate supply of water is available in the condensate storage tanks with an available backup supply from the city water systems. Thus a diverse water supply is assured for plant equipment cooling purposes.

3.8.3 Steam Line Isolation System

In addition to the above systems, the Steam Line Isolation System indirectly protects the reactor vessel from potentially harmful pressure transients by preventing the blowdown of more than one steam generator due to steam line failure. This system is discussed in the FSAR, Section 10.3.

3.8.4 Steam Dump

In addition to the systems discussed above, an additional means of cooling the primary systems, thereby lowering the Reactor Coolant System pressure and assisting in maintaining reactor vessel integrity, is provided by steam dump to the condenser and to the atmosphere as described in the FSAR, Section 10.2.

3.9 Auxiliary Electrical System

An auxiliary electrical power system is provided to assure safe reactor operation and to assure the continuing availability of engineered safety features so that the latter may be available to protect the reactor vessel as discussed above and as required by the Technical Specifications Section 3.7. Design and operation details are given in the FSAR, Section 8.2.3.

4.0 EVALUATIONS DEMONSTRATING THAT SYSTEMS AND EQUIPMENT WILL PERFORM AS REQUIRED BY THE TECHNICAL SPECIFICATIONS

This section describes evaluations that show why various systems and equipment identified in Section III above can be depended upon to operate as required. Since the systems and equipment discussed in this section are those considered to directly affect reactor vessel integrity, assurance of operation of those systems is also an assurance that the reactor vessel will not experience abnormal conditions which could result in vessel failure.

Additional assurance is provided by testing and surveillance of the various systems as described in Section 5.0 of this Appendix.

4.1 Pressure Control

The reactor vessel is protected from overpressure conditions by the following assurances that the pressure control equipment will function properly:

4.1.1 <u>Pressurizer Heaters</u>

The pressurizer contains electric heaters to regulate the Reactor Coolant System pressure by keeping the water and steam in the pressurizer at saturation temperature.

During a negative pressure surge, caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keeps the pressure above the minimum allowable limit. Heaters are also energized on high water level during positive surges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop.

The pressurizer heaters are incapable of overpressurizing the reactor coolant system. Maximum steam generation rate with heaters is about 15,000 lbs/hr., compared with a total capacity of 1,224,000 lbs/hr.

for the three safety valves and a total capacity of 358,000 lbs/hr. for the two power-operated relief valves.

Finally, the rate of pressure rise achievable with heaters is slow, and ample time is available for the operator to take manual action in tripping the reactor.

4.1.2 Pressurizer Spray

During a positive pressure surge, the spray system condenses steam in the pressurizer to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. The spray valves on the pressurizer are power operated and open at 2260 psig (normal operating pressure is 2235 psig). In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

4.1.3 Power Operated Relief Valves

Either of the two relief valves can easily maintain pressure below the high pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. The set point for actuating the relief valves is 2335 psig. An alarm is also actuated at the same time, announcing the high pressure condition.

4.1.4 High Pressure Trip

The set point for high-pressure reactor trip is 2385 psig. This protection function is discussed in Sections 4.2, 4.2.8.4 and 4.2.8.5 below.

4.1.5 Safety Valves

The reactor coolant system is protected against overpressure by three safety valves located on the top of the pressurizer. The safety valves on the pressurizer are sized to prevent system pressure from exceeding the design pressure by more than 10 per cent, in accordance with Section III of the ASME Boiler and Pressure Vessel Code. The capacity of the pressurizer safety valves is determined from consideration of: (1) the reactor protection system characteristics, and (2) accident or transient conditions which may potentially cause overpressure.

Each of the three pressurizer safety valves is designed to relieve 408,000 lbs. per hr. of saturated steam at the valve set point. The combined capacity of the safety valves is equal to or greater than the maximum surge rate resulting from complete loss of load without a direct reactor trip or any other control, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve setting.

Details of the analysis are reported in the FSAR, Section 14.1.8. Experience has shown that the safety valve capacity so determined is adequate for all the other transients as shown by the results of Section 14.1 of the FSAR.

4.1.6 General Assurance that the Pressure Control Devices are Properly Designed

The overall pressure relieving requirements for the Reactor Coolant and Steam Systems are determined by a detailed digital computer program, discussed in Section 14.1.8 of the FSAR, which describes the neutron kinetics, decay heat, Reactor Coolant System with pressurizer, steam generators, and the associated steam bypass system and Rod Control System.

The analysis shows that the capacity of the pressure control device is sufficient for protection of the Reactor Coolant System.

4.2 Reactor Protection Systems

4.2.1 General Evaluation Providing Assurance of Operation

The Reactor Protection Systems are designed to provide direct assurance of overall plant safety and are therefore considered to be one of the two most important systems for the maintenance of reactor vessel integrity. The other is the Control System, since reliable operational control is obviously the best approach to plant safety and reactor vessel integrity, where excursions leading to high coolant pressures and possible reactor vessel damage can be limited before protective action is necessary. Assurance of proper operation of the Control System will be discussed in the following section, 4.3.

Westinghouse design philosophy for Reactor Protection and Control Systems is to make maximum use, for both protection and control functions, of a wide range of measurements. This results in a broad spectrum of redundant protection and control functions. The design approach used permits all equipment components to be identified as protection or control and located accordingly, with electrical isolation and physical separation between them. The design approach thus permits not only redundancy of control, providing a significant and desirable increment to overall plant safety, but also provides a Protection System which continuously monitors numerous system variables by different means; i.e., Protection System diversity.

Although the Protection System design basis requires only that random single failures not negate the Protection System, a considerable depth of protection is achieved by the Westinghouse design approach. Systems designers and reviewers have emphasized the importance of achieving a suitable balance of design objectives in regard to functional and equipment diversity, interaction of control and protection functions, testing, and surveillance to achieve a Protection System design that has adequate capability to cope with both random and systematic failure modes. (Systematic failures are also known as common-mode, or nonrandom failures.)

4.2.2 Common-Mode Failures and Diversity

Common-mode, or systematic failures, are those that partially or completely prevent identical instrument channels from performing their function.

Redundancy is not an answer to this type of failure, since all channels are assumed to be affected. Further, these failures cannot be evaluated by probability analysis or reliability data; indeed, they are characterized by oversights or deficiencies which presumably would be corrected when first detected.

The general categories of common-mode failures are:

a) Functional deficiency - The variable being monitored does not provide the information intended during the course of an accident.

This deficiency could be caused by the accident's following a different course than predicted by the designers, or by a change in the plant characteristics which changes the relation between the process and the variable being monitored.

b) Maintenance error - This failure includes consistent miscalibration of all channels of a type, and also circuit modification or repair which inadvertently renders the channels functionally inoperative.

c) Design deficiency - Failure of the equipment as installed to meet functional requirements. This could arise through unrecognized dependence on a single, common element, such as ventilation; by an unexpected characteristic (such as saturation or slow response) in all controllers of a type; or by the instrumentation being disabled as a result of the accident.

d) External catastrophe - With proper isolation and separation between redundant channels, this is confined to major disasters such as flood, earthquake, fire, etc.

Considerable effort is made in Reactor Protection Systems design to prevent these common-mode failures. However remote, the possibility of a common-mode failure must nevertheless be considered. The likelihood of maintenance errors can be minimized by proper administrative procedures, identification of Protection System components, and complete documentation of the as-supplied Protection System, including the design basis. Design deficiencies can be largely eliminated by equipment qualification testing and by careful review of all potential common elements.

Redundancy is an accepted defense against random failures which affect only one component or channel at a time. Similarly, diversity is a defense against common-mode failures which could affect multiple channels.

Such protective diversity can be achieved in either of two ways: equipment diversity, by providing different types of instrumentation to monitor the same variable, or functional diversity, by monitoring different plant variables. Functional diversity entails some degree of equipment diversity, primarily with respect to sensors and set-points. More importantly, however, functional diversity is not dependent on the calculated response of any one variable during an accident. As a converse of this, functional diversity is more complex to demonstrate since the response of several variables must be analyzed for each type of accident evaluated.

To demonstrate diversity where protective action is needed, it is necessary to show combinations of two or more of the following "barriers" for each accident. Some of these are addressed to the probable need for protective action, rather than to the Instrumentation System itself. This is considered a reasonable approach to judging the adequacy of a Protective System.

a) Tolerable consequences for expected conditions - Although "worst case" analysis might fail to prove that protection is not needed, the vast majority of cases may have acceptable consequences. Whether or not this is a suitable barrier depends on the probability of adverse conditions (such as excessive inserted rod worth) and the design and operating precautions taken to prevent them.

b) Low probability of accident - Probability of the initiating fault might be considered, but only in conjunction with the probable consequences. That is, a loss-of-coolant accident does not require less protection than a loss of flow accident simply because it is less likely to occur.

c) Control interlocks - Rod stops or other devices which arrest or modify spurious control action short of reactor trip can be part of the Protection System. Protection System design standards, equipment testing, and Technical Specifications limits would therefore be applied.

d) Manual action - Manual action can be considered a reliable backup to automatic protection, depending on the accident rate, the complexity of the problem and corrective action, and the alarms and indication provided.

e) Automatic reactor trip - Each accident may have a "principle" reactor trip associated with it.

f) Backup reactor trip - A second reactor trip function, of a diverse type, is an additional barrier.

The following subsections present discussions pertinent to the assurance that the Protection Systems can be depended upon to insure plant and reactor vessel safety. Further discussion is available in the FSAR, Section 7.2.

4.2.3 Protection Systems Reliability

The reactor uses a higher speed version of the Westinghouse magnetic-type control rod drive mechanisms used in the San Onofre, and Connecticut Yankee plants. Upon a loss of power to the coils, the rod cluster control assemblies with full length absorber rods are released and fall by gravity into the core.

The reactor internals, fuel assemblies, RCC assemblies and drive system components are designed as Seismic Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly. Due to this and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

The Rod Cluster Control (RCC) assembly guide system is locked together with pins throughout its length to ensure against misalignments which might impair control rod movement under normal operating conditions and credible accident conditions. An analogous system has successfully undergone 4132 hours of testing in the Westinghouse Reactor Evaluation Channel during which about 27,200 feet of step-driven travel and 1461 trips were accomplished with test misalignments in excess of the maximum possible misalignment that may be experienced when installed in the plant.

All reactor trip protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit is accomplished by placing that circuit in a half-tripped mode; i.e., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Reliability and independence is obtained by redundancy within each tripping function. In a two-out-of-three circuit, for example, the three channels are equipped with separate primary sensors. Each channel is continuously fed from its own independent electrical source. Failure to de-energize a channel when required would be a mode of malfunction that would affect only that channel. The trip signal furnished by the two remaining channels would be unimpaired in this event.

4.2.4 Protection Systems Redundancy and Independence

The reactor protection systems are designed so that the most probable modes of failure in each protection channel result in a signal calling for the protective trip. Each protection system design combines redundant sensors and channel independence with coincident trip philosophy so that a safe and reliable system is provided in which a single failure will not defeat the channel function, cause a spurious plant trip, or violate reactor protection criteria.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions when combined are combined only at the sensor. Both of these functions are fully isolated in the remaining part of the channel, control being derived from the primary protection signal path through an isolation amplifier. As such, a failure in the control circuitry does not affect the protection channel. This approach is used for pressurizer pressure and water level channels, steam generator water level, T_{avg} and ΔT channels, steam flow-feedwater flow and nuclear source, power range channels.

In the Reactor Protection System, two reactor trip breakers are provided to interrupt power to the full length rod drive mechanisms. The breaker main contacts are connected in series (with power supply) so that opening either breaker interrupts power to all full length rod mechanisms, permitting them to fall by gravity into the core. In the event of a loss of rod control power reactor trip breaker is de-energized and trips to an open mode.

Further detail on redundancy is provided through the detail descriptions of the respective systems covered by the various sections in this chapter. In summary, reactor protection is designed to meet all presently defined reactor protection criteria and is in accordance with the IEEE-279 "Proposed Standard for Nuclear Power Generation Station Protection Systems". Redundancy and independence are achieved by protection channel designs which combine more than one sensor and parameter measurement with coincident trip circuitry (e.g., pressure coincident with level and interlocked with flow or nuclear flux).

4.2.5 Protection Against Multiple Disability for Protection Systems

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Separation of redundant analog protection channels originates at the process sensors and continues back through the field wiring and containment penetrations to the analog protection racks. Physical separation is used to the maximum practical extent to achieve separation of redundant transmitters. Separation of field wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Redundant analog equipment is separated by locating redundant components in different protection racks. Each channel is energized from a separate a-c instrument bus.

4.2.6 Multiple Trip Settings

Where it is necessary to change to a more restrictive trip setting to provide adequate protection for a particular mode of operation or set of operating conditions, the design provides positive means of assuring that the more restrictive trip setting is used. The devices used to prevent improper use of less restrictive trip settings are considered a part of the protective system and are designed in accordance with the other provisions of these criteria.

4.2.7 Information Readout and Indication of By-Pass

The protective systems are designed to provide the operator with accurate, complete, and timely information pertinent to their own status and to plant safety.

Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service.

Trips are indicated and identified down to the channel level.

4.2.8 Specific Control and Protection Interactions

4.2.8.1 Nuclear Flux

Four power-range nuclear flux channels are provided for overpower protection. Isolated outputs from all four channels are averaged for automatic control rod regulation of power. If any channel fails in such a way as to produce a low output, that channel is incapable of proper overpower protection. In principle, the same failure would cause rod withdrawal and overpower. Two-out-of-four overpower trip logic will ensure an overpower trip if needed even with an independent failure in another channel. Finally, an overpower signal from any nuclear channel will block automatic rod withdrawal. The set point for this rod stop is below the reactor trip set point.

4.2.8.2 Coolant Temperature

Four T_{avg} channels are used for overtemperature-overpower protection. Isolated output signals from all four channels are also averaged for automatic control rod regulation of power and temperature since in principle, a spuriously low temperature signal from one sensor would partially defeat this protection function and cause rod withdrawal and overtemperature.

In addition, channel deviation alarms in the control system will block automatic rod motion (insertion or withdrawal) if any temperature channel deviates significantly from the others. Automatic rod withdrawal blocks will also occur if any one of four nuclear channels indicates an overpower condition or if any one of four temperature channels indicates an overtemperature condition. Two-out-of-four trip logic is used to ensure that an overtemperature trip will occur if needed even with an independent failure in another channel. Finally, as shown in the FSAR Section 14.1, the combination of trips on nuclear overpower, high pressurizer water level, and high pressurizer pressure also serve to limit an excursion for any rate of reactivity insertion.

4.2.8.3 Pressurizer Pressure

Four pressure channels are used for high and low pressure protection and for overpower-overtemperature protection. Isolated output signals from these channels also are used for pressure control and compensation signals for rod control. These are discussed separately below:

a) Control of rod motion: one of the pressure channels is used for rod control with a low pressure signal acting to withdraw rods. The discussion for coolant temperature is applicable, i.e., two-out-of-four logic for overpower-overtemperature protection as the primary protection, with backup from multiple rod stops and "backup" trip circuits. In addition, the pressure compensation signal is limited in the control system such that failure of the pressure signal cannot cause more than about a 10°F change in T_{avg}. This change can be accommodated at full power without a DNBR less than 1.30.

b) High pressure control: This is discussed in 4.1.5 above.

4.2.8.4 Pressurizer Level

Three pressurizer level channels are used for high level reactor trip (2/3) and low level safety injection (1/3 logic level coincident with pressure). Isolated output signals from these channels are used for volume control, increasing or decreasing water level.

a) High Level

A reactor trip on pressurizer high level is provided to prevent rapid thermal expansions of reactor coolant fluid from filling the pressurizer.

b) Low Level

For control failures which tend to empty the pressurizer, one-out-of-three logic for safety injection actuation on low level coincident with low pressure ensures that the protection system can withstand an independent failure in another channel.

In addition, a signal of low level from either of two independent level control channels will isolate letdown, thus preventing the loss of coolant. Also, ample time and alarms exist for operator action.

4.2.8.5 Steam Generator Water Level; Feedwater Flow

The basic function of the reactor protection circuits associated with low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature-overpressure excursion in the reactor coolant. Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, residual heat after trip would cause thermal expansion and discharge of the reactor coolant to containment through the pressurizer relief valves. Redundant emergency feedwater pumps are provided to prevent this.

4.2.9 Conclusions

The diversity and redundancy of safeguards and components in the Reactor Protection System provides assurance that this System will function as required in meeting the Safety Limits specified. The Technical Specifications, Sections 2.1 and 2.2, will thereby maintain the reactor vessel integrity.

4.3 Regulating Systems

4.3.1 General

Two independent reactivity control systems are provided. One of the two reactivity control systems employs rod cluster control assemblies within the reactor core. The other reactivity control system employs the Chemical and Volume Control System to regulate the concentration of boric acid solution neutron absorber in the Reactor Coolant System.

A redundancy of reactivity control is therefore built into the Regulating Systems to insure operation within the limits of the Technical Specifications, Section 2, thereby maintaining the integrity of the plant, including the reactor vessel.

Each regulating system is discussed separately with respect to evaluations which ensure their operation as required.

4.3.2 Reactor Control System

The primary function of the Reactor Control System is to provide automatic control of the rod clusters during power operation of the reactor. The system uses input signals including neutron flux; coolant temperature and pressure; and plant turbine load.

The Reactor Control System is designed to enable the reactor to follow load changes automatically when the plant output is above 15% of nominal power. Control rod positioning may be performed automatically when plant output is above this value, and manually at any time.

The operator is able to select any single bank of rods (shutdown or control) for manual operation. Using a single switch, he may not select more than one bank from these two groups. He may also select automatic reactor control, in which case, the control banks can be moved only in their normal sequence with some overlap as one bank reaches its full withdrawal position and the next bank begins to withdraw. Power supplied to the rod banks is controlled such that no more than two banks can be withdrawn simultaneously.

The control system is capable of restoring coolant average temperature to within the programmed temperature deadband, following a scheduled or transient change in load.

The control system is designed to operate as a stable system over the full range of automatic control throughout core life without requiring operator adjustment of set points other than normal calibration procedures.

The following specific evaluations, from the FSAR, Section 7.3, provide assurance of operation as required by means of redundancies.

4.3.2.1 Rod Control

There are 61 total RCC assemblies of which 53 are full length and 8 are part length rods. The full length rods are divided into (1) a shutdown group comprising two shutdown banks of 8 rod clusters each and two shutdown banks of 4 rod clusters each, and (2) a control group comprising 4 control banks containing 8, 4, 8 and 9 rod clusters.

The four banks of the control group are the only rods that can be manipulated under automatic control. The banks are divided into subgroups to obtain smaller incremental reactivity changes. All RCC assemblies in a subgroup are electrically paralleled to step simultaneously. Position indication for each RCC assembly type is the same. There are two types of drive mechanisms for the RCC assemblies, those for the control and shutdown groups and those for the part length rod group.

4.3.2.2 Control Group Rod Control

The automatic rod control system maintains a group programmed reactor coolant average temperature with adjustments of control group rod position for equilibrium plant conditions. The reactor control system is capable of restoring programmed average temperature following a scheduled or transient change in load. The coolant average temperature increases linearly from zero power to the full power conditions.

The rod control group is divided into four banks comprising 8, 4, 8 and 9 RCC's respectively, to follow load changes over the full range of power operation. Each rod control bank is driven by a sequencing, variable speed rod drive control unit. The rods in each control bank are divided into two subgroups; the subgroups are moved sequentially one step at a time. The sequence of motion is reversible; that is, a withdrawal sequence is the reverse of the insertion sequence. The variable speed sequential rod control affords the ability to insert a small amount of reactivity at low speed to accomplish fine control of reactor coolant average temperature about a small temperature deadband. Any reactor trip signal causes the rods to insert by gravity into the core.

Manual control is provided to manually move a control bank in or out at a preselected fixed speed.

Proper sequencing of banks of RCCA's is assured first, by fixed programming equipment in the Rod Control System, and second, through administrative control of the reactor plant operator. Startup of the plant is accomplished by first manually withdrawing the shutdown rods to the full out position. This action requires that the operator select the SHUTDOWN BANK position on a control board mounted selector switch and then position the IN-HOLD-OUT level (which is spring return to the HOLD position) to the OUT position.

RCCA are then withdrawn under manual control of the operator by first selecting the MANUAL position on the control board mounted selector switch and then positioning the IN-HOLD-OUT lever to the OUT position. In the MANUAL selector switch position, the rods are withdrawn (or inserted) in a predetermined programmed sequence by the automatic programming equipment.

When the reactor power reaches approximately 15% of rated full power, the operator may select the AUTOMATIC position, where the IN-HOLD-OUT lever is out of service, and rod motion is controlled by the Reactor Control and Protection Systems. A permissive interlock limits automatic control to reactor power levels above 15%. In the AUTOMATIC position, the rods are again withdrawn (or inserted) in a pre-determined programmed sequence by the automatic programming equipment.

Programming is set so that as the first bank out (control bank C-2) reaches a preset position near the top of the core, the second bank out (control bank C-3) begins to move out simultaneously with the first bank. When control bank C-2 reaches the top of the core, it stops, and control bank C-3 continues until it reaches a preset position near the top of the core where control bank C-4 motion begins. This withdrawal sequence continues until the plant reaches the desired power level. The programmed insertion sequence is the opposite of the withdrawal sequence, i.e., the last control bank out is the first control bank in.

With the simplicity of the rod program, the minimal amount of operator selection, and two separate direct position indications available to the operator, there is very little possibility that inadvertent rearrangement of the control rod sequencing could be made.

4.3.2.3 Shutdown Rod Group Control

The shutdown group of control rods together with the control group are capable of shutting the reactor down. They are used in conjunction with the adjustment of chemical shim and the control group to provide shutdown margin of at least one per cent following reactor trip with the most reactive control rod in the fully withdrawn position for all normal operating conditions. The shutdown banks are manually controlled during normal operation and are moved at a constant speed with staggered stepping of the subgroups within the banks. Any reactor trip signal causes them to insert by gravity into the core. They are fully withdrawn during power operation and are withdrawn first during startup. Criticality is always approached with the control group after withdrawal of shutdown banks. For shutdown banks with a total of 24 clusters are provided.

4.3.2.4 Part Length Rod Control

Eight part length rods are provided in the reactor in addition to the normal control rods. The function of these rods, which have neutron absorber material in only the bottom one quarter of the length (three feet), is primarily to shape the axial power distribution and thus stabilize axial xenon oscillations. In addition, they are beneficial in flattening the axial power distribution and thus in reducing hot channel factors.

The part length rods are operated only by manual control by the operator from the control console. They are moved together as a bank to make the upper and lower ion chamber readings approach a prescribed relationship within a prescribed allowable region of travel.

The part length rods do not trip since power is required to change their position. Instead, upon loss of power they are locked in place by mechanical brakes.

4.3.2.5 Interlocks

The rod control group is used for automatic control and is interlocked with measurements of turbine-generator load and reactor power to prevent automatic control rod withdrawal below 15% of nominal power. The manual and automatic controls are further interlocked with measurements of nuclear flux, ΔT and rod drop indication to prevent approach to an overpower condition.

4.3.2.6 RCCA Position Indication

Two separate systems are provided to sense and display control rod position as described below:

a) Analog System

An analog signal is produced for each individual rod by a linear position transmitter.

An electrical coil stack is located above the stepping mechanisms of the control rod magnetic jacks, external to the pressure housing, but concentric with the rod travel. When the associated control rod is at the bottom of the core, the magnetic coupling between the primary and secondary coil winding of the detector is small and there is a small voltage induced in the secondary. As the control rod is raised by the magnetic jacks, the relatively high permeability of the lift rod causes an increase in magnetic coupling. Thus, an analog signal proportional to rod position is obtained.

Direct, continuous readout of every control rod is presented to the operator on individual indicators.

A deviation monitor alarm is actuated if an individual rod position deviates from its group position by a preselected distance.

Lights are provided for rod bottom positions for each rod. The lights are operated by bistable devices in the analog system.

b) Digital System

The digital system counts pulses generated in the rod drive control system. Readout of the digital system is in the form of electromechanical add-subtract counters reading the number of steps or rod withdrawal with one display for each subgroup. These readouts are mounted on the control panel.

The digital and analog systems are separate systems; each serves as backup for the other. Operating procedures require the reactor operator to compare the digital and analog readings upon recognition of any apparent malfunction. Therefore, a single failure in rod position indication does not in itself lead the operator to take erroneous action in the operation of the reactor.

4.3.2.7 Full Length Rod Drive Power Supply

The full length control rod drive power supply concept using a single scram bus system has been successfully employed on all Westinghouse PWR plants. Potential fault conditions with a single scram bus system are discussed in this section. The unique characteristics of the latch type mechanisms with its relatively large power requirements makes this system with the redundant series trip breakers particularly desirable.

The solid state rod control system is operated from two parallel connected 400 KVA generators which provide 260 volt line to line, three phase, four wire power to the rod control circuits through two series connected reactor trip breakers. This AC power is distributed from the trip breakers to a line-up of identical solid state power cabinets using a single overhead run of enclosed bus duct which is bolted to and therefore comprises part of the power cabinet arrangement. Alternating current from the motor-generator sets is converted to a profiled direct current by the power cabinet and is then distributed to the mechanism coils. Each complete rod control system includes a single 70 volt DC power supply which is used for holding the mechanisms in position during maintenance of normal power supply and is provided to avoid the need for bringing a separate outside DC source to its system.

This 70 volt supply, which receives its input from the AC power source downstream of the reactor trip breakers, is distributed to each power cabinet and permits holding mechanisms in groups of four by manually positioning switches located in the power cabinets. The 30 ampere output capacity limits the holding capability to eight rods.

Current to the mechanisms is interrupted by opening either of the reactor trip breakers. The 70 volt DC maintenance supply will also be interrupted since this supply receives its input power through the reactor trip breakers.

The trip breakers are arranged in the reactor trip switchgear in individual metal enclosed compartments. The 1000 ampere bus work, making up the connections between scram breakers will be separated by metal barriers to prevent the possibility that any conducting object could short circuit, or bypass, scram breaker contacts.

4.3.2.8 Power Connections

Safeguards are provided to insure against serious external power connections which might incapacitate the reactor control system. These are discussed separately under AC and DC Power Connections.

a) AC Power Connections

The three phase four wire supply voltage required to energize the equipment is 260 volts line to line, 58.3 Hz, 438 KVA capacity, zig-zag connected. It is unlikely that any power supply, and in particular one as unusual as this four wire power source could be accidently connected, in phase, in the required configuration. Also it should be noted that this requires multiple connections, not single connections. The closest outside sources available in the plants are 480 volt auxiliary power sources and 208 volt lighting sources.

Connections of either a 480 or 208 volt, 60 Hz source to the single AC bus supplying the rod control system causes currents to flow between the sources due to an out of phase condition. These currents flow until the generator accelerates to a speed synchronous with the 60 Hz outside source, a time sufficient to trip the generator breakers. The out-of-phase currents for an unlimited capacity outside source, an outside source with a capacity equivalent to the normal generator KVA, and for either one or two M-G sets in service are tabulated below:

TABLE B-3

<u>Out of Phase Currents (Amperes)</u>

		One M-G Set in Service	Two M-G Sets in Service
480V	Unlimited Capacity	25,000	50,000
	438 KVA Capacity	12,000	25,000
208V	Unlimited Capacity	16,000	32,000
	438 KVA Capacity	8,000	16,000

All of the above currents are sufficiently high to trip out the generator breakers on either overcurrent or reverse current. This trip-out is detectable by annunciation in the control room. If the outside power source trips, the connection is of no concern.

Each solid state power cabinet is tied to the main AC bus through three fused disconnect switches; one for the stationary gripper coil circuits, one for the movable gripper coil circuits, and one for the lift coil circuits. Reference voltages to operate the control circuits for all three coil circuits must be in phase with the supply to all coil circuits for proper operation of the system. If the outside power source were brought into an individual cabinet, nine (9) normal source connections would have to be disconnected and the outside source would have to be tied in phase to the proper nine (9) points plus one (1) neutral point to allow movement of the rods. This is not considered credible.

Connection of a single phase AC source (i.e., one line to neutral) is also considered improbable. This would again require a high capacity source which would have to be connected in-phase with the non-synchronous M-G set supply. Again more than one connection is needed to achieve this condition. Each power cabinet contains three alarm circuits (stationary, movable, and lift) that would annunciate the condition to the operator. In addition, calculations show that a single phase source of 208 volts, 260 volts, or 480 volts will not supply enough current to hold the rods. Therefore, a jumper across two trip circuit breaker contacts in series which results in a single phase remaining closed would not provide sufficient current to hold-up the rods.

The normal source generators are connected in a zig-zag winding configuration to eliminate the effects of direct current saturation of the machines resulting from the direct currents that flow in the half wave bridge rectifier circuits. If this connection were not used, the generator core would saturate and loss of generating action would occur. This condition would also occur in a transformer. An outside source not having the zig-zag configuration would have to have a large capacity (>400 KVA) to avoid the loss of transformer action from saturation.

b) <u>DC Power Connections</u>

An external DC source could, if connected inside the power cabinet, hold the rods in position. This would require a minimum supply voltage of 50 volts. Since the holding current for each mechanism coil is 4 amperes, the DC current capacity would have to be approximately 180 amperes to hold all rods. Achieving this situation would require several acts - bringing in a power source which is not required forjny type of operation in the rod control system, preferentially connecting it into the system at the correct points, and actuating specific holding switches so as to interconnect all rods. Closure of twelve switches in four separate cabinets would be required to hold all rods. One switch could hold as many as four rods.

The application of a DC voltage to an individual rod external to the power cabinet would affect only a single rod connection with other rods in the group being prevented by the blocking diodes in the power circuits.

Should an external DC source be connected to the system, the system is provided with features to permit its detection.

Each solid state power cabinet contains circuitry which compares the actual currents in the stationary and movable gripper coils with the reference signals from the step sequencing unit (slave cycler). In taking a single step, the current to the stationary gripper coil will be profiled from the holding value to the maximum, to zero and return to holding level. Correspondingly, the movable gripper coil must change from zero to maximum and return to zero. The presence of an external DC source on either the stationary or movable coils would prevent the related currents from returning to zero.

This situation would be instantaneously annunciated by way of the comparison circuit. Therefore, any rod motion would actuate an alarm indicating the presence of an external DC source. In addition, an external DC source would prevent rods from stepping. Thus, an external source could be detected by the rod position indication system indicating failure of the rod(s) to move.

Connection of an external DC power source to the output lines of the 70 volt DC power supply can be detected by opening the three phase primary input of the supply and checking the output with a built-in voltmeter.

c) Control System Construction

The rod control system electrical equipment is assembled in enclosed steel cabinets. Three phase power is distributed to the equipment through a steel enclosed bus duct, bolted to the cabinets. DC power connections to the individual mechanisms are routed to the reactor head area from the solid state cabinets through insulated cables, enclosed junction boxes, enclosed reactor containment penetrations, and sealed connectors. In view of this type of construction, any accidental connections of either an AC or DC power source, either internal or external to the cabinets, is not considered credible.

d) Power Connections Evaluation Summary

In view of the proceding discussion, the postulated connection of an external power source (either AC or DC) or occurrence of short circuits that could prevent dropping of the rods is not considered credible. Specifically:

a. The need for an outside power source has been eliminated by incorporating built-in holding sources as part of the rod control system and by providing two M-G sets.

b. The equipment is contained within enclosed steel cabinets precluding the possibility of an accidental connection of either AC or DC power in the cabinets.

c. AC power distribution is accomplished using steel enclosed bus duct. The high capacity (400 KVA) AC power source is unique and not readily available. Multiple connections are required.

d. DC power is distributed to the individual mechanisms through insulated cables and enclosed electrical connections precluding the accidental connection of an outside DC source external to the cabinets. The high capacity DC source required to hold rods is not readily available in the rod control system, would require multiple connections, and would require deliberate positioning of switches within the enclosed cabinets.

e. Provisions are made in the system to permit detection of an external DC source which could preclude a rod release.

The total capacity of the system including the overload capability of each motor generator set is such that a single set out of service does not cause limitations in rod motion during normal plant operation. In order to minimize reactor trip as a result of a unit malfunction, the power system is normally operated with both units in service.

4.3.2.9 Reactivity Control Systems Malfunction

Reactor shutdown with rods is completely independent of the normal control functions since the trip breakers completely interrupt the power to the full length rod mechanisms regardless of existing control signals.

4.3.3 Chemical and Volume Control System

4.3.3.1 General

The Chemical and Volume Control System serves as a secondary reactor control system by the addition and removal of varying amounts of boric acid solution.

When the reactor is critical, the best indication of reactivity status in the core is the position of the control group in relation to plant power and average coolant temperature. There is a direct, predictable, and reproducible relationship between control rod position and power and it is this relationship which establishes the lower insertion limit calculated by the rod insertion limit monitor. There are two alarm setpoints to alert the operator to take corrective action in the event a control bank approaches or reaches its lower limit.

Any unexpected change in the position of the control group when under automatic control or a change in coolant temperature when under manual control provides a direct and immediate indication of a change in the reactivity status of the reactor. In addition, periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

4.3.3.2 Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing with alarm protection of lines, valves, and components normally containing concentrated boric acid.

The system has three high pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses. Each of the three charging pumps are powered from separate 480 volt buses. The two boric acid transfer pumps are also powered from separate 480 volt buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of a-c power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels if necessary.

4.3.4 Conclusion

The redundancy of regulation by two means of control, rod and chemical shim, and the redundancy and diversity of equipment safeguards including interlocks and manual and automatic controls and alarms, as discussed above, provide ample assurance that the Regulatory Systems will operate as required by the Technical Specifications, Section 2, in maintaining both safe plant operation and integrity of Primary Coolant System, including reactor vessel.

4.4 Instrumentation Systems

4.4.1 General

The evaluation of the Instrumentation Systems providing assurance of operation as required is discussed separately for the Nuclear Instrumentation and Process Instrumentation System.

4.4.2 Nuclear Instrumentation System

Instrumentation is provided to monitor neutron flux so that the control system can maintain the flux within operating ranges as prescribed in the Technical Specifications, Section 2.3. This equipment indirectly insures reactor vessel integrity by making possible proper operation of the reactor.

4.4.2.1 Reliability and Redundancy

The requirements established for the reactor protective system apply to the nuclear instrumentation. Therefore, the discussions given in Section IV.2 of the Appendix are also pertinent here. All channel functions are independent of every other channel.

4.4.2.2 Power Supply

The nuclear instrumentation system is powered by four independent vital bus circuits (Details are provided in Section 8 of the FSAR).

Loss of nuclear instrumentation power would result in the initiation of all reactor trips associated with the channel power failure. In addition, all trips which were blocked prior to loss would be unblocked and initiated.

4.4.2.3 Safety Factors

The relation of the power range channels to the Reactor Protective System is described in detail in Section 7.2 of the FSAR. To maintain the desired accuracy in trip action, the total error from drift in the power range channels is held to ±1 per cent at full power. Routine tests and recalibration ensure that this degree of deviation is not exceeded. Bistable trip set points of the power range channels are also held to an accuracy of ±1 per cent of full power. The accuracy and stability of the equipment are verified by vendor tests.

4.4.3 Process Instrumentation System

The non-nuclear regulating process instrumentation measures temperatures, pressure, flow, and levels in the Reactor Coolant System, Steam Systems, and other Auxiliary Systems and thus contributes to the assurance of reactor vessel integrity both directly and indirectly by providing data to the Reactor Protective and Control Systems.

Process variables required on a continuous basis for the startup, power operation, and shutdown of the plant are controlled and indicated or recorded from the control room, access to which is supervised. The quantity and types of process instrumentation provided ensures safe and orderly operation of all systems and processes over the full operating range of the plant.

A general discussion of the evaluation of this system is given in the FSAR, Section 7.5.3.

4.4.3.1 System Evaluation

Redundant instrumentation has been provided for all inputs to the protective systems and vital control circuits.

Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation is provided.

Instrumentation components are selected from standard commercially available products.

All electrical and electronic instrumentation required for safe and reliable operation is supplied from four redundant instrument buses.

Additional detail is presented in the FSAR, Section 7.2, Protective Systems.

4.4.4 Conclusion

The redundancy and diversity within the systems and the independence of channel functions provide assurance that the Instrumentation Systems will function properly in providing data required for the proper operation of the other Reactor Systems.

4.5 Auxiliary Systems

4.5.1 General

Only those auxiliary systems considered to be most directly concerned with assurance of continued reactor pressure vessel integrity are discussed in this section. These are the Chemical and Volume Control System (in its function for corrosion control), Residual Heat Removal System, Sampling System, and Steam and Power Conversion System. These systems are considered to have direct application to the assurance of reactor vessel integrity since they either control corrosion or help prevent the Reactor Coolant System, of which the reactor vessel is a part, from exceeding design pressure.

4.5.2 Chemical and Volume Control System (CVCS)

The CVCS, in its function for corrosion control, can be considered as an auxiliary system. However, the evaluation of the functional reliability of the system is best discussed in a single context, and is presented in the discussion of this system in Section 4.3.3 of this Appendix.

4.5.3 Residual Heat Removal System (RHRS)

Assurance that the RHRS will function to protect the Reactor Coolant System from overheating and thereby protect the reactor vessel from overpressurization, according to the provisions of the Technical Specification, Section 3.3, is provided by redundancy of all active loop components.

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation or safe cooldown of the plant is not affected, however, the time for cooldown is extended.

4.5.4 Sampling Systems

4.5.4.1 Materials

All sample lines and valves are constructed of austenitic stainless steel or equivalent corrosion resistant material.

4.5.4.2 Availability and Reliability

The system operates on an intermittent basis, and under administrative manual control.

Neither automatic nor operator action is required of the Sampling System during an emergency or to prevent an emergency condition. The system is therefore designed in accordance with standard practices of the chemical processing industry.

4.5.4.3 Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a loss-of-coolant accident, and the consequences analyzed. The results are presented in Table B-4. From this evaluation it is concluded that proper consideration has been given to station safety in the design of the system.

TABLE B-4

MALFUNCTION ANALYSIS OF SAMPLING SYSTEM

Sample Chains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, or hot leg sample	Remote operated sampling valve inside reactor containment fails to close	Diaphragm - operated valve outside the reactor containment closes on containment isolation signal
Any sample chain	Sample line break inside containment	Same as above

4.6 Steam and Power Conversion System

4.6.1 General

Section 10.3 of the FSAR presents an evaluation of the capability of the Steam and Power Conversion System to respond to the temperature and pressure conditions following a reactor or turbine trip by permitting the dissipation of decay heat. The principle portions of the evaluation are presented below.

4.6.2 Evaluation of System

A reactor trip from power requires subsequent removal of core decay heat. Immediate decay heat removal requirements are satisfied by steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser. Decay heat is transferred to feedwater in the steam generator where it is converted to steam.

Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of a complete loss of offsite electrical power to the station and concurrent reactor trips decay heat removal would continue to be assured by the steam-driven, and two motor-driven (via emergency generator) auxiliary steam generator feed-water pumps, and steam dumped to atmosphere via the main steam safety and power relief valves. In this case feedwater is available from the condensate storage tank by gravity feed to the auxiliary feedwater pumps. The minimum 360,000 gallons of water in the condensate storage tank is adequate for decay heat removal for a period of at least 24 hours. A back-up source of feedwater is available from the city water storage tank.

Following a turbine trip, the control system reduces reactor power output immediately by a reactor trip. Steam is bypassed to the condenser and there is no lifting of the main safety valves. In the event of failure of a main feedwater pump, the auxiliary feedwater pumps are automatically started and the second main feedwater pump remaining in service will carry approximately 65 per cent of full load feedwater flow. If both main feedwater pumps fail, the reactor will be tripped, as a result of steam generator low-low level or steam-feedwater flow mismatch and the auxiliary feedwater pumps started. If Reactor Coolant System conditions reach trip limits, the reactor will trip.

Pressure relief is required at the system design pressure of 1085 psig. The first safety valve is set to relieve at 1065 psig. Additional safety valves are set at pressures up to 1120 psig, as allowed by the ASME Code. The pressure relief capacity is equal to the steam generation rate at maximum calculated conditions.

A single failure analysis has been made for all active components of the system which have an emergency function. The analysis, which is presented in Table B-5 shows that the failure or malfunction of any single active component will not reduce the capability of the system to perform its emergency function.

TABLE B-5

SINGLE FAILURE ANALYSIS

Component or System	Malfunction	Comments and Consequences
Auxiliary Feedwater System	Auxiliary Feedwater pump fails to start (following loss of main feedwater)	The Auxiliary Feedwater System comprises one turbine driven and two motor driven pumps. The turbine pump is twice the capacity of a motor driven pump and one motor driven pump has sufficient capacity to prevent release of coolant through the pressurizer relief valve. Thus adequate redundancy of auxiliary feedwater pumps is provided.
Steam Line Isolation System	Failure of Steam isolation valve to close (following a main steam line rupture)	Each steam line contains an isolation valve and a non-return valve in series. Hence a failure of an isolation (or non-return) valve will not permit the blowdown of more than one steam generator irrespective of the steamline rupture location.
Turbine Bypass System	Bypass valve sticks open (following operation of the bypass system resulting from a turbine trip)	The turbine bypass system is rated at 40% of the maximum calculated steam flow at the maximum guarantee pressure in the nuclear steam supply system and includes 12 bypass valves. Hence one valve can only pass < 4% of the steam generator steam flow and there is no hazard in the form of an uncontrolled plant cooldown if a bypass valve sticks open.

5.0 TESTING AND SURVEILLANCE PROCEDURES PROVIDED TO INSURE THAT THE REACTOR VESSEL, SYSTEMS AND EQUIPMENT WILL PERFORM AS REQUIRED.

Comprehensive testing and surveillance programs have been established and adopted to insure the continued integrity and performance to design requirements of the reactor vessel and the systems and equipment whose functions are vital to proper operation of the vessel. This section describes these programs in detail.

5.1 Reactor Vessel

A comprehensive and well-detailed description of pre-operational testing and in-service structural surveillance of the reactor vessel is given in Section 4.2 of the Technical Specifications, which also refers to pertinent sections of Section XI of the ASME Code for In-Service Inspection of Nuclear Reactor Coolant Systems. Additional information is provided in the FSAR, pages 4.1.9 - 4.1.10, 4.3, 4.5.1 - 4.5.8, 4D.6 - 4D.7, and Table 4.5.1 and 4.5.2. This section also includes a discussion of the reactor vessel materials irradiation surveillance program in the Technical Specifications, page 4.2-16.

The information is to be discussed in two parts:

a. Pre-Operational Testing, including materials surveillance and testing and hydrostatic testing to ASTM Code Section III and additional Westinghouse requirements.

b. In-Service Inspection, including detection of reactor vessel head flange seal leakage as discussed in the FSAR page 4.2-16.

5.2 Primary System Pressure Boundary

Testing and surveillance information pertinent to those portions of the Primary System outside of the pressure vessel, such as the pressurizer, primary pump, and primary piping and valves are contained in the Technical Specifications pages 4.2.1-4, 4.2.9 - 4.2.17, and Table 4.2.1, and in the FSAR pages 4.1.9 -4.1.10, 4.5.1 - 4.5.8, 4.3, 4D.6 - 4D.7, and Tables 4.5.1 and 4.5.2. This section is given in two parts:

a. Pre-Operational Testing including materials surveillance and testing to ASTM Code Section III.

   This section also discusses pre-operational testing to verify performance of the humidity detector and the condensate measuring system (cf. AEC Q.4.4.2 and 4.4.4).

b. In-Service Inspection, including discussion of Primary System leakage as given in the FSAR page 4.2.16.

   AEC Q.4.4.2 and 4.4.3 discuss leakage from the coolant system.

5.3 Reactor Protection Systems

Reactor trip System testing is discussed in the FSAR pages 7.2.17 - 7.2.19, which includes a detailed description of Analog and Logic Channel Testing. Demonstration of functional operability of Protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred is discussed in FSAR pages 7.2.7 - 7.2.8.

5.4 Regulating System

5.4.1 <u>Reactor Control System</u>

Frequencies for equipment tests are given in Table 4.1.3 of the Technical Specifications.

Additional information concerning evaluation of reactivity anomalies within the reactor are discussed in the Technical Specifications, Section 4.9, and the FSAR page 3.3.1.

5.4.2 <u>Chemical & Volume Control System (CVCS)</u>

Chemistry Checks of the regulatory functions of this system are tabulated in the Technical Specifications, Table 4.1.2.

Testing of instrument channels in the CVCS are given in the Technical Specifications 4.1-1 and in the FSAR Table 9.2-8.

5.6 Instrumentation System

The minimum frequency and types of surveillance applied to instrument channels are discussed in detail in Technical Specifications Section 4 and Table 4.1.1 and in general in the FSAR page 7.9.1.

5.7 Auxiliary Systems

5.7.1 Chemical & Volume Control System (CVCS)

Checks of the non-regulatory functions of the CVCS are listed in the Technical Specifications, Table 4.1.1 and 4.1.2, and FSAR Table 9.2-8.

5.7.2 Auxiliary Coolant System

Testing of the Auxiliary Coolant System is discussed in the FSAR, pages 9.3.18 - 9.3.19, and in the Technical Specifications, pages 4.4.5 - 4.4.9.

5.7.3 Sampling System

Frequencies for sampling tests are given in Technical Specifications Table 4.1-2, typical examples are listed in the FSAR, page 9.4.9.

5.8 Emergency Systems

5.8.1 Safety Injection System

The Technical Specifications, pages 4.5.1, 4.5.3, 4.5.5 - 4.5.6 and FSAR pages 6.2.3 - 6.2.4 and 6.2.46 - 6.2.51 discuss testing of the Safety Injection Systems.

5.8.2 Auxiliary Feedwater System

Periodic testing requirements are discussed in Technical Specifications pages 4.8.1 - 4.8.2 and the FSAR page 10.4.1.

5.8.3 Emergency Power System

Periodic testing and surveillance of the Emergency Power System are discussed in detail in Technical Specifications page 4.6 and in general in the FSAR, page 8.4.1.

6.0 SUMMARY AND CONCLUSION

This Appendix presents a discussion of the system, components and procedures provided in the design of Indian Point that provide assurance that the reactor vessel will be operated in accordance with the Technical Specifications. Included in the discussion are:

a. Assurances that abnormal conditions will not affect reactor vessel integrity,

b. A listing and brief description of the systems and components installed to insure conformity of operation with design and Technical Specifications requirements,

c. Evaluations demonstrating that systems and components will perform as required by the design and Technical Specifications requirements,

d. A listing of the testing and surveillance procedures provided to insure that the reactor vessel, systems and components will perform as required by the Technical Specifications, and

It is concluded that as a result of the diversity and redundancy of the systems and components, and the testing and surveillance procedures provided for Indian Point Unit No. 2, plant operation within the safety limits specified in the Technical Specifications is assured. As a consequence, there is assurance that the reactor vessel will not fail in service.

Appendix C

Recommendations of PVRC*

Toughness Requirements for Ferritic Material

*Pressure Vessel Research Committee

RECOMMENDATIONS OF PVRC
TOUGHNESS REQUIREMENTS FOR FERRITIC MATERIALS

Purpose: To recommend, on the basis of current knowledge, criteria for ferritic material toughness requirements for pressure-retaining components of the reactor coolant pressure boundary operating below 700F. These criteria, when used in addition to the stress limits allowed by the ASME Code, should permit the establishment of safe procedures for operating nuclear reactor components under normal, upset and testing conditions; emergency and faulted conditions should be considered on a case basis.

Major Regions Requiring Criteria: For the purpose of this study, the requirements were considered in the following four major regions of the reactor coolant boundary:

A. Vessel shell and head regions remote from discontinuities. These may be subjected to sufficient radiation to affect the mechanical and toughness properties.

B. Vessels, pumps and valve bodies at or near nozzle and flange regions if they are not subjected to significant radiation.

C. Piping

D. Bolting

In the following discussion, each of the four regions will be considered from the standpoint of (1) acceptance tests for materials, (2) methods for determining safe operating procedures, and (3) the severity of the defects for which the specified properties will provide protection against failure. In this summary the basis for the recommendations is given in brief general terms and more detailed justification is given in appendices.

A. Shell or Head Regions Remote from Geometric Discontinuities

A(1) Acceptance Tests

(a) Tests are required for materials having a nominal section thickness greater than 1/2 inch.

(b) Properties are to be determined in the direction of the maximum general primary membrane stress. This is the hoop direction in a cylindrical shell, but in a spherical shell or head the specified properties are required in both tangential-longitudinal and tangential-transverse directions. Therefore, for spherical shells or heads test specimens should be oriented in the direction normal to the principal direction in which the metal was worked (other than thickness direction), so that the specimens represent the generally lower toughness of that orientation.

(c) Determine the NDT Temperature, T_{NDT}, by drop weight tests (ASTM E-208-69) using P-1, P-2 or P-3 type specimens from the 1/4 thickness location. At the temperature T_{NDT} + 60F conduct three C_v tests (SA-370) using specimens from the 1/4 thickness location. If all C_v values are $\gtrsim$ 40 mils lateral expansion, then T_{NDT} is the reference temperature, RT_{NDT}. If any value is less than 40 mils, conduct C_v tests to determine the temperature, T_{40}, at which all C_v values are $\gtrsim$ 40 mils. Then the reference temperature $RT_{NDT} = T_{40}$ - 60F. Thus RT_{NDT} is the higher of T_{NDT} and T_{40} - 60F.

If the material is to be subjected to significant radiation it will be found that a complete C_v versus temperature curve must be obtained to permit interpretation of the surveillance specimen data (see (e) below).

(d) Apply the procedure to specimens representing base material, heat-affected zone and weld metal.

(e) The impact properties of (c) above must be met throughout the life of the component. Various data and methods are available for determination of the margin between initial properties and irradiated end-of-life properties to provide reasonable assurance that the required properties will be obtained at the 1/4 thickness location throughout life. Maintenance of the required properties in the materials of lowest toughness in (d) above, and with due consideration to the effect of copper and phosphorus content from radiation, is to be verified by surveillance specimens (see Appendix VIII). The radiation-induced temperature change ΔT_{NDT} and the maintenance of the required impact properties at RT_{NDT} + 60F at the 1/4 thickness location is to be established by tests of surveillance specimens conducted in accordance with the methods of ASTM. It is not feasible to name a specific fluence above which a surveillance program is mandatory. Available data show, however, that if the fluence at the inner wall is less than 10^{18}nvt(>1Mev) no significant radiation damage is to be expected. For higher values of end-of-life fluence the omission of a surveillance program should be justified by showing that for the particular lots of base and weld metal being used, the reactor vessel shell will not become more limiting than other parts of the vessel.

The recommended tests and acceptance standards given above were based on the following considerations:

1. The use of both dropweight and C_v tests gives protection against the possibility of errors in the conducting of tests or the reporting of test results.
2. The C_v requirements are given in terms of lateral expansion rather than absorbed energy because this provides protection from variation in yield strength from initial heat treatment and the change in yield strength produced by irradiation. (See App. I).
3. The requirement of 40 mils lateral expansion at RT_{NDT} + 60F throughout the life of the component provides assurance of adequate ductility at "upper shelf" temperatures.
4. The C_v test at T_{NDT} + 60F serves to weed out non-typical materials such as those which might have low transition temperature but abnormally low energy absorption on the upper shelf.

A(2) <u>Allowable Loadings for Normal Operation</u>

The procedure for obtaining allowable loadings during normal, upset and testing conditions as defined in the ASME Code (Section III, NB-3113), is based on the principles of linear elastic fracture mechanics. A maximum credible flaw which is proportional in size to the section thickness is assumed, as discussed in A(3) below. Fig. 1 is a curve which shows the relationship which can be conservatively expected between the critical (or reference) stress intensity factor, K_{IR}, and a temperature which is related to the reference nil-ductility temperature, RT_{NDT}, as determined in A(1). This curve is based on the lower bound of dynamic and crack arrest K_{Id} and K_{Ia} values measured as a function of temperature on specimens of SA-533 B-1, SA-508-2 and SA-508-3 steels. The data used for the derivation of Fig. 1 are described in App. II. No available data points for static, dynamic or arrest tests fall below the curve. An analytical approximation to the curve in Fig. 1 is:

$$K_{IR} = 1.223 \exp \left\{0.0145 \left[T - (RT_{NDT} - 160)\right]\right\} + 26.777 \quad \text{- - (1)}$$

Curves similar to Fig. 1 are needed for other materials, such as SA-516 Gr 70 and SA-216-WCB, but the data are not available in sufficient quantity. In the interim, until data become available, it is recommended and considered safe to use the curve of Fig. 1 for steels with minimum specified yield strengths lower than 50000 psi, but not for steels with higher specified yield strengths.

In a vessel shell or head remote from discontinuities the significant loadings are (1) general primary membrane stress due to pressure and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Effects of residual stress are not included in the recommended procedure because:

1. Peak values in a postweld heat treated component are less than 20% of the yield strength.

2. Service stresses and radiation effects both tend to reduce residual stresses during the life of the component.

3. Conservatisms throughout the whole recommended procedure and the safety factor applied to the allowable pressure appear to be ample to cover any incalculable adverse effects.

Therefore the procedure for calculating the allowable system pressure during startup consists of adding the K_I corresponding to the membrane tension produced by pressure to the K_I produced by thermal gradient at a desired startup and shutdown rate and requiring that the sum of these K_I values not exceed the available K_{IR} of Fig. 1 at each temperature. For conservatism, a safety factor of 2 is applied to the K_I produced by pressure during startup, shutdown and normal and upset operating conditions. Methods for calculating these two K_I values for the assumed maximum credible defect are described in Appendices III and IV. These methods result in the curves shown in Figs. 2 and 3. The K_I produced by pressure is:

K_I pressure = M_t x general primary membrane stress ----- (2)

Figure 2 shows M_t as a function of wall thickness. The three lines shown for the M values are to be used for stress values 0.1, 0.5, 0.7 and 1.0 times the material yield strength. The M values for other stress levels can be interpolated linearly with sufficient accuracy since they are not strongly affected by the stress level.

The K_I produced by thermal gradient is:

$K_{Itherm.} = M_\theta$ x temp. difference through wall ---------- (3)

Figure 3 shows M_θ as a function of wall thickness, based on the following assumptions:

1. An assumed shape of the temperature gradient, as shown in App. IV.
2. The shutdown starts after a steady-state temperature has been attained.
3. The rate of change of temperature is of the magnitude associated with startup and shutdown, i.e. less than about 100°F per hour. The result would be overly conservative if applied to rapid temperature changes.
4. The postulated defect has a depth 1/4 of the wall thickness and infinite length. (The values of M_θ for 1/8 thickness crack depth are also shown in Figure 3).

For some purposes, Figure 3a may be found more convenient to use than Fig. 3. It is based on a radius-to-thickness ratio of 10 and uses cooling rate directly, thus avoiding the need to calculate the temperature difference across the wall.

The governing operating condition is most apt to be shutdown because decreasing temperature produces tensile stress in the portion of the reactor vessel wall which is subjected to the greatest radiation damage.

The requirement for startup, shutdown and normal and upset operating conditions is that:

$2K_{Ipressure} + K_{Itherm.} \leq K_{IR}$ from Figure 1 ------------- (4)

throughout the life of the component at each temperature.

During hydrotest of the vessel and/or system there is no thermal stress and there is less uncertainty regarding the stress values. Also the consequences of failure are less severe. Therefore less conservatism is justifiable.

The recommended requirement for the system hydrotest after installation is:

$$1.5 K_{I_{pressure}} \leq K_{IR} \text{ from Fig. 1} \text{ - - - - - - - - - } (4)$$

at the hydrotest temperature.

The recommended requirement for a shop hydrotest of an individual component is that the test be made at a temperature 60F higher than the reference temperature RT_{NDT} as determined in A(1)(c).

A(3) Severity of Defects

Except for some special situations which will be discussed, the postulated maximum credible defect which was used in the development of the foregoing material acceptance standards and allowable pressure loading was a semi-elliptic surface flaw, perpendicular to the direction of maximum stress, having a depth of 1/4 of the section thickness and a length six times its depth. For the calculation of $K_{I therm}$ in eq. (3) the defect was assumed to be infinitely long. This assumed defect provided the rationale for using the 1/4 thickness location for test specimens and the same location for the radiation damage calculations.

If the ASME Code's 2% radiography standard is taken at face value, it may be noted that the postulated defect is 12.5 times as large in linear dimensions as the Code allowable and thus has over 150 times the area. It is not safe to assume that no defects larger than the Code allowable will ever occur, but it does seem reasonable to assume that with the combination of radiography required by Section III and the ultrasonic mapping required by Section XI there is a very low probability that a defect larger than about four times the Code allowable will escape detection. The postulated defect is about three times as large in linear dimensions and has about 10 times the area of even that conservative value.

Other defect shapes and locations might have been chosen to develop the recommended criteria. Buried defects are more apt to escape detection than surface defects, but produce only about half the K_I value for a given size, so the use of a buried defect would not be conservative. Surface defects longer than six times their depth would produce higher K_I values, but even if an infinitely long defect had been postulated the recommended properties and calculation methods would still provide protection against a defect having a depth 1/6 the section thickness.

The postulation of the large defect size, the use of dynamic test data and the safety factor of 2 on the allowable pressure stress combine to provide a large degree of conservatism. It is interesting to note how large the defects would have to be before fast fracture could occur even if the critical stress intensity factor of the material was really as low as the K_{IR} of Fig. 1. For a surface crack perpendicular to the maximum pressure stress direction in a section T inches thick, the safety factor is still not lower than unity even if the defect becomes:

1. T/2 deep and 3T long, or
2. T/3 deep and infinitely long, or
3. Through-the-wall and 2T long

B. <u>Nozzles, Flanges and Shell Regions Near Geometric Discontinuities</u>

B(1) <u>Acceptance Tests</u>

The recommended tests, acceptance standards and discussions are the same as given in A(1) above, except that:

1. Since the direction of maximum stress varies from one location to another, the specified properties must be determined in the less favorable direction, normal to the principal direction in which the metal was worked (other than the thickness direction).

2. The provisions of A(1)(e) on radiation damage are not applicable because it is assumed that the component

will experience no significant fast neutron exposure in the regions near geometric discontinuities.

The location of test specimens should be the same as given in the ASME Code, Section III, 1971 Edition, Article NB-2000.

B(2) Allowable Loadings for Normal Operation

The same general procedure as was used for the shell region in A(2) can be used with some modifications for areas where more complicated stress distributions occur. The sum of the K_I values at each point must not exceed the K_{IR} from Fig. 1. Equation (3) requires additional terms to represent the K_I's contributed by the possible combinations of primary and secondary, membrane and bending loads. The $K_{I\,therm}$ of Fig. 3 is not used here because these more complicated stress distributions are not adequately represented by this simplified approach.

The K_I produced by bending is:

$$K_{Ib} = M_b \times \text{bending component of stress} \quad --- \quad (5)$$

M_b can be obtained from Fig. 2 by taking 2/3 of the M_t value, as described in App. III.

If a membrane or bending stress is produced by a primary (load controlled) loading the K_I is included at twice its calculated value to provide the chosen safety factor. For purposes of this evaluation, stresses which result from bolt preloading shall be considered as primary.

If a membrane or bending stress is produced by a secondary (strain controlled) loading the K_I is included at its calculated value. The K_I's produced by thermal stress should be calculated by appropriate methods and considered as secondary.

The terms whose sum must be $\leq K_{IR}$ for startup, shutdown, and normal and upset operating conditions are:

$2\ M_t$ x primary membrane stress

$2\ M_b$ x primary bending stress

M_t x secondary membrane stress

M_b x secondary bending stress - - - - - - - - - - (6)

For the system hydrotest after installation the terms to be added are:

$1.5\ M_t$ x primary membrane stress

$1.5\ M_b$ x primary bending stress

M_t x secondary membrane stress

M_b x secondary bending stress - - - - - - - - - - (7)

For the shop hydrotest, as in A(2) above, the only recommended requirement is that the test required by the ASME Code, Section III, be made at a temperature not lower than RT_{NDT} + 60F.

B(3) Severity of Defects

The ASME Code requires that for most product forms the impact test specimens be taken at a depth of 1/4 of the section thickness. Therefore for these cases the properties required by this set of recommendations will furnish protection against brittle failure for the postulated maximum credible defect with a depth of 1/4 the wall thickness. There are exceptions to this general rule, however, and separate consideration must be given to some special situations, as follows:

(a) Fig. 2 provides values for the calculation of K_I for section thickness up to about 12 inches. For sections greater than 12 inches thick it does not seem reasonable to postulate a credible defect more than 3 inches deep and 18 inches long. Therefore, when using Figs. 2 and 3 for sections greater than 12 inches thick, the M_t, M_b and M_θ values for 12 inches should be used.

(b) The ASME Code specifies that for "very thick and complex forgings" (NB-2223.3) test coupons may be taken from material as close as 3/4 inch from a heat treated surface instead of at the 1/4 thickness location. Therefore there is an inconsistency in the recommended procedure since the toughness properties of the test coupons are probably better than the properties of the actual forging at the root of the postulated 1/4 thickness deep defect. It is believed that this inconsistency is amply compensated by the ultrasonic examination requirements for heavy forgings which make it highly improbable for a surface defect greater than 1 inch deep to escape detection. The recommended use of Fig. 2 for obtaining M_t and M_b values for use in equations (6) and (7) assumes defects as deep as 3 inches, which compensates for any possible lack of conservatism involved in the location of the impact test specimens.

(c) The inside corner of the juncture between a nozzle and a cylindrical shell is a critical location because at this point the local shell hoop stress produced by pressure can be two or three times the membrane stress, σ_h, in other parts of the shell. Accurate values of K_I as a function of pressure, specific geometry and crack depth are not available, but App. V describes a method of approximation based on analysis, fatigue data, and burst tests of epoxy models. Fig. 4 shows a comparison of the influence of flaw size at this critical location with the influence of the postulated flaw used for the vessel shell. It may be seen that for K_I/σ_h in the range of 2 to 3 the critical flaw size in a nozzle corner may be as small as 1/4 that for the shell. Since it is not reasonable to require material with a K_{Id} value double that in the shell, extra precautions must be taken to assure freedom from defects at this critical region. Since the location of the critical region is known, since its extent is small, and since it need not contain a strength weld, this additional precaution appears to be feasible.

(d) For the hydro test performed in the manufacturer's shop, the recommendation given above that the test be performed at 60F above RT_{NDT} is less conservative as far as protection against failure is concerned than the recommended procedures for system hydro and operation. This is based on the idea that if the component is going to fail, this is the best place to have it happen. The 60F margin does provide protection against failure for a surface flaw with a depth equal to the lesser of 1/4 thickness or 1 inch.

C. Piping

As described in App. VI there is a possibility of yield level bending stresses in piping. The Code allows these to be applied during service and they may also occur as residual stresses applied during construction. Loading can occur at temperatures as low as 40F. If it were not for the possibility of these yield-level stresses the toughness requirements for piping in the wall thickness and sizes used in water-cooled reactors would be so modest that little or no testing would be required. However, in view of these common practices it seems prudent to require the following acceptance standards:

C(1) Acceptance Standards

The required C_v values, in the axial direction (through-thickness notch), at 40°F, for pipes with wall thickness greater than 1/2 inch are:

Wall Thickness, inches	Mils Lateral Exp.
Over 1/2 to 3/4	15
Over 3/4 to 1	20
Over 1 to 1 1/2	30
Over 1 1/2 to 2 1/2	40

For wall thicknesses greater than 2 1/2 in., the vessel criteria of B(1) should be used.

C(2) Allowable Loadings

For material which meets the requirements of C(1) above, no special design considerations beyond those of ASME Code Section III, 1971 Edition, are necessary. The assumed loadings are pressure plus yield stress bending at 40°F.

C(3) Severity of Defects

The postulated defects considered were a 360° circumferential crack with a depth 1/4 of the wall thickness and a circumferential through-wall crack of a length 2 times the wall thickness. The C_v values required in C(1) are a mean between those needed to protect against these two assumed defects at the loadings specified in C(2) above.

D. Bolting

Details of the studies made for bolting criteria are given in App. VII.

D(1) Acceptance Standards

For bolting, including studs, nuts and bars, with a nominal size greater than 1 inch diameter, conduct 3 C_v tests at the lowest service metal temperature using specimens from the 1/4 diameter location, in the axial direction. All C_v values shall be in accordance with the following:

Nominal Diam., In.	Mils Lateral Exp.
1 or less	No test required
Over 1 to 3	30
Over 3	35

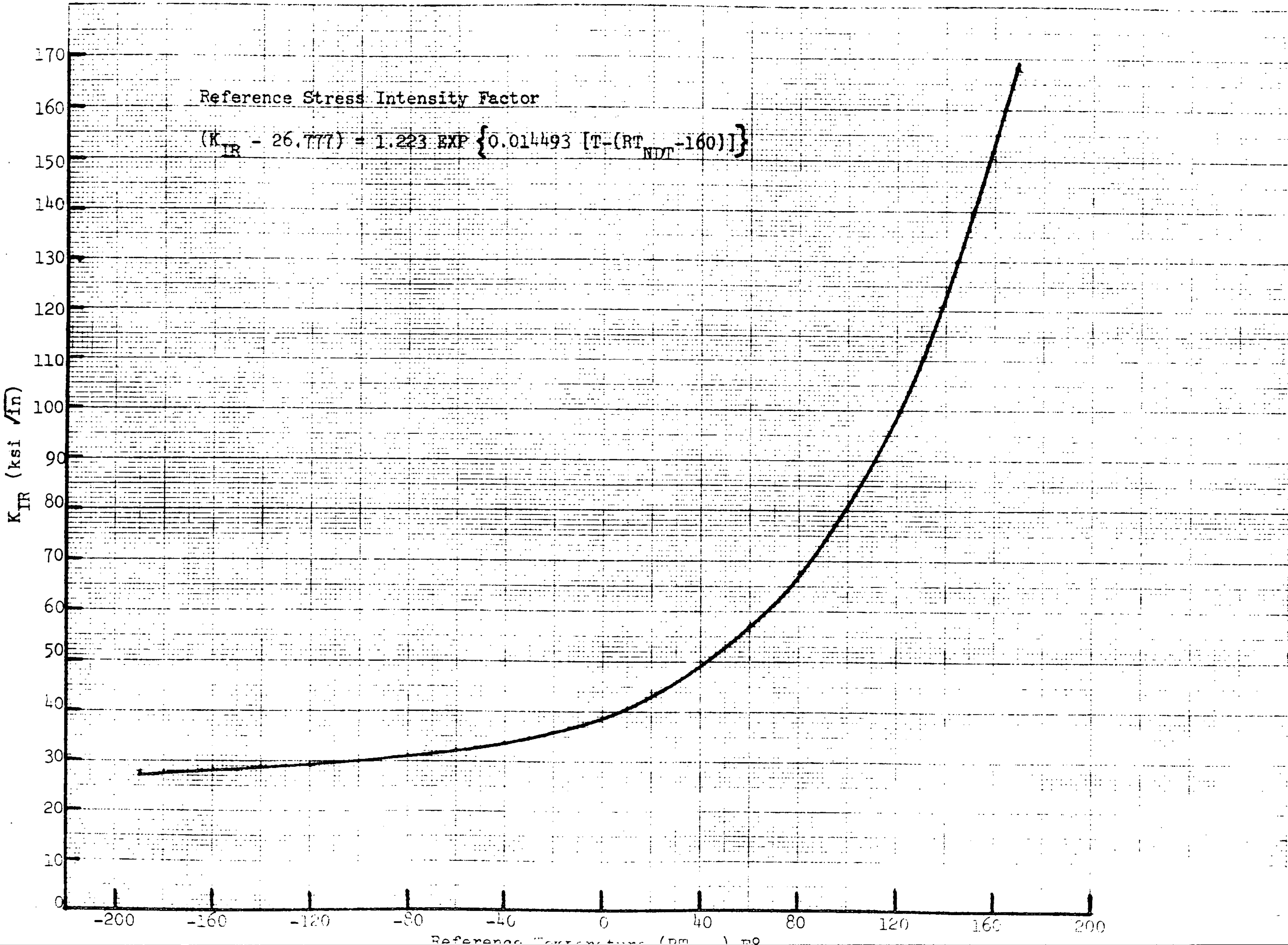
D(2) Allowable Loading

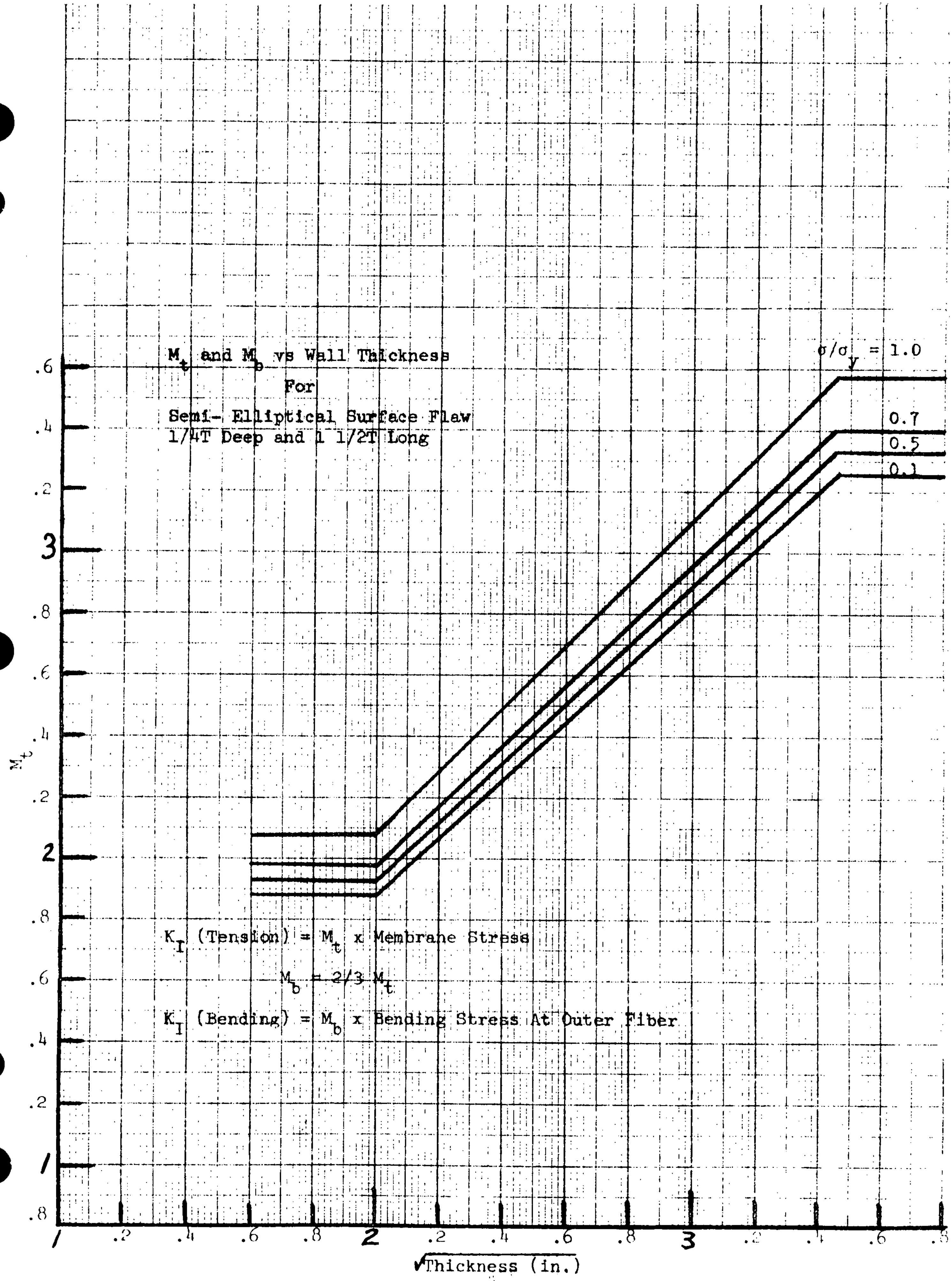
The loading allowed by Section III of the ASME Code is 2/3 yield stress averaged across the section plus an additional 1/3 yield stress in bending.

D(3) Severity of Defects

As shown in App. VII, the specified properties will protect against failure with a 360° circumferential crack of a depth 10% of the diameter up to 3 inch diameter and 0.3 inch depth for larger sizes.

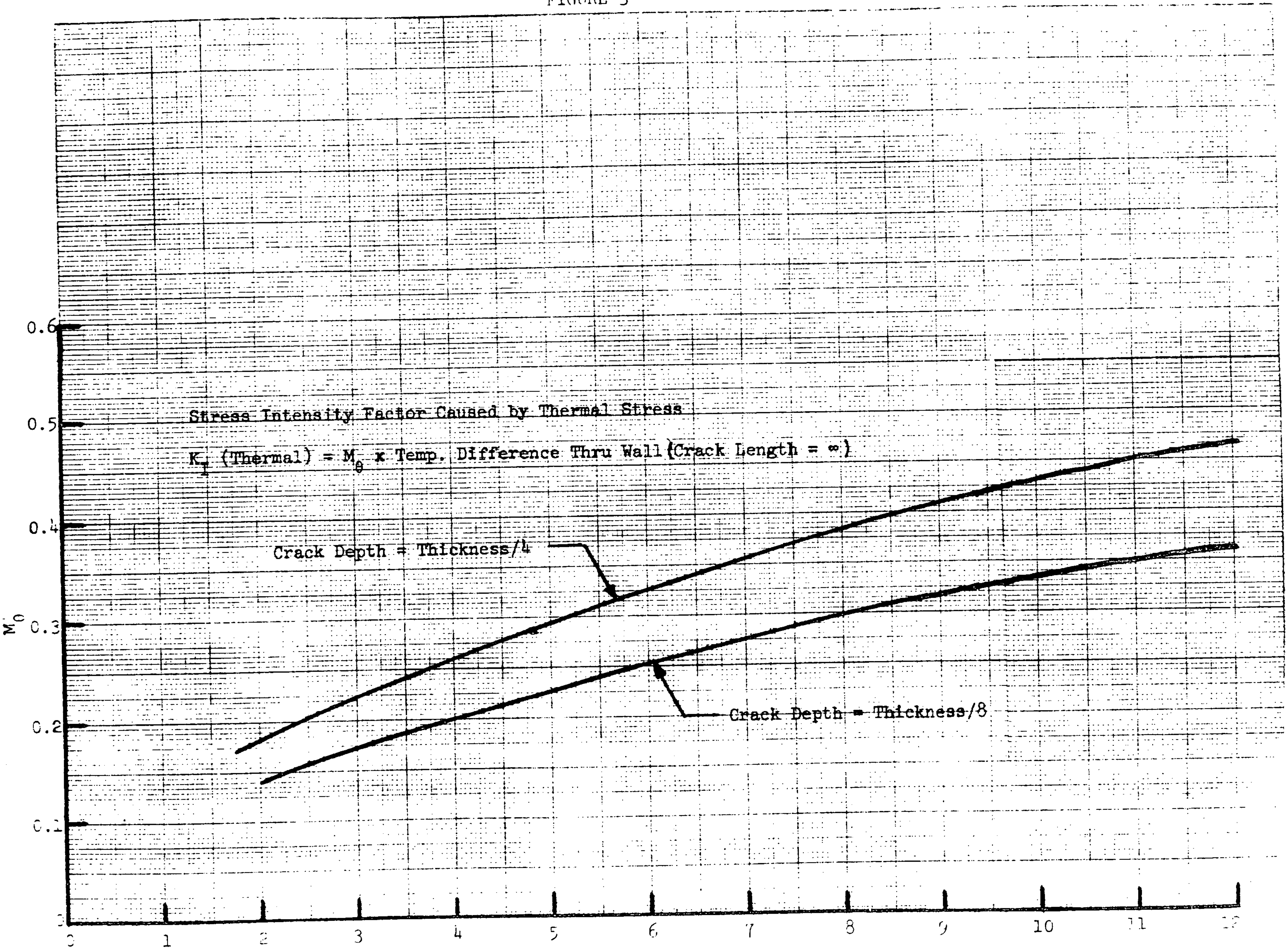
FIGURE 1

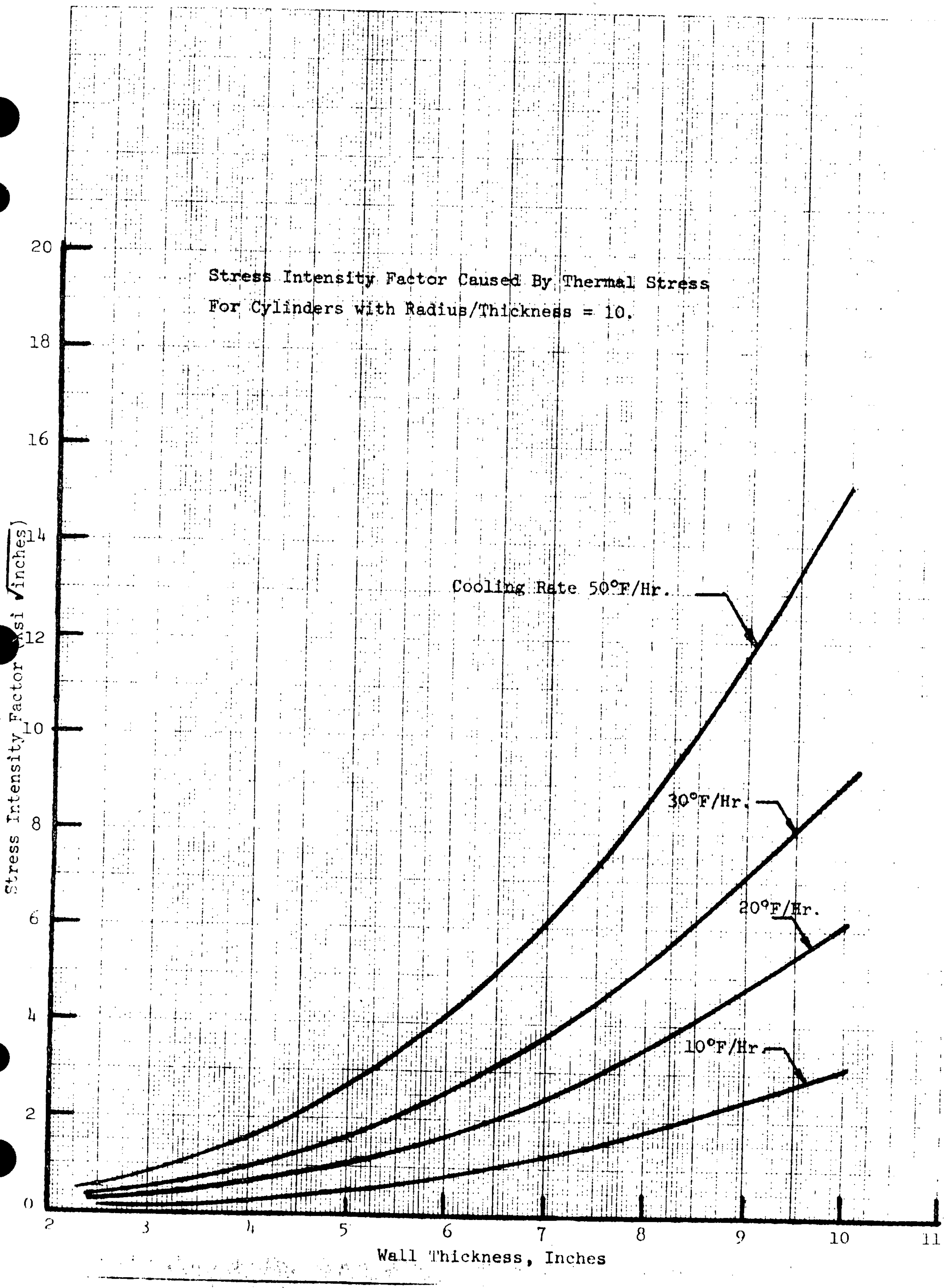

Reference Stress Intensity Factor
$(K_{IR} - 26.777) = 1.223\ \mathrm{EXP}\left\{0.014493\ [T-(RT_{NDT}-160)]\right\}$
K_{IR} (ksi $\sqrt{in}$)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
-200
-160
-120
-80
-40
0
40
80
120
160
200



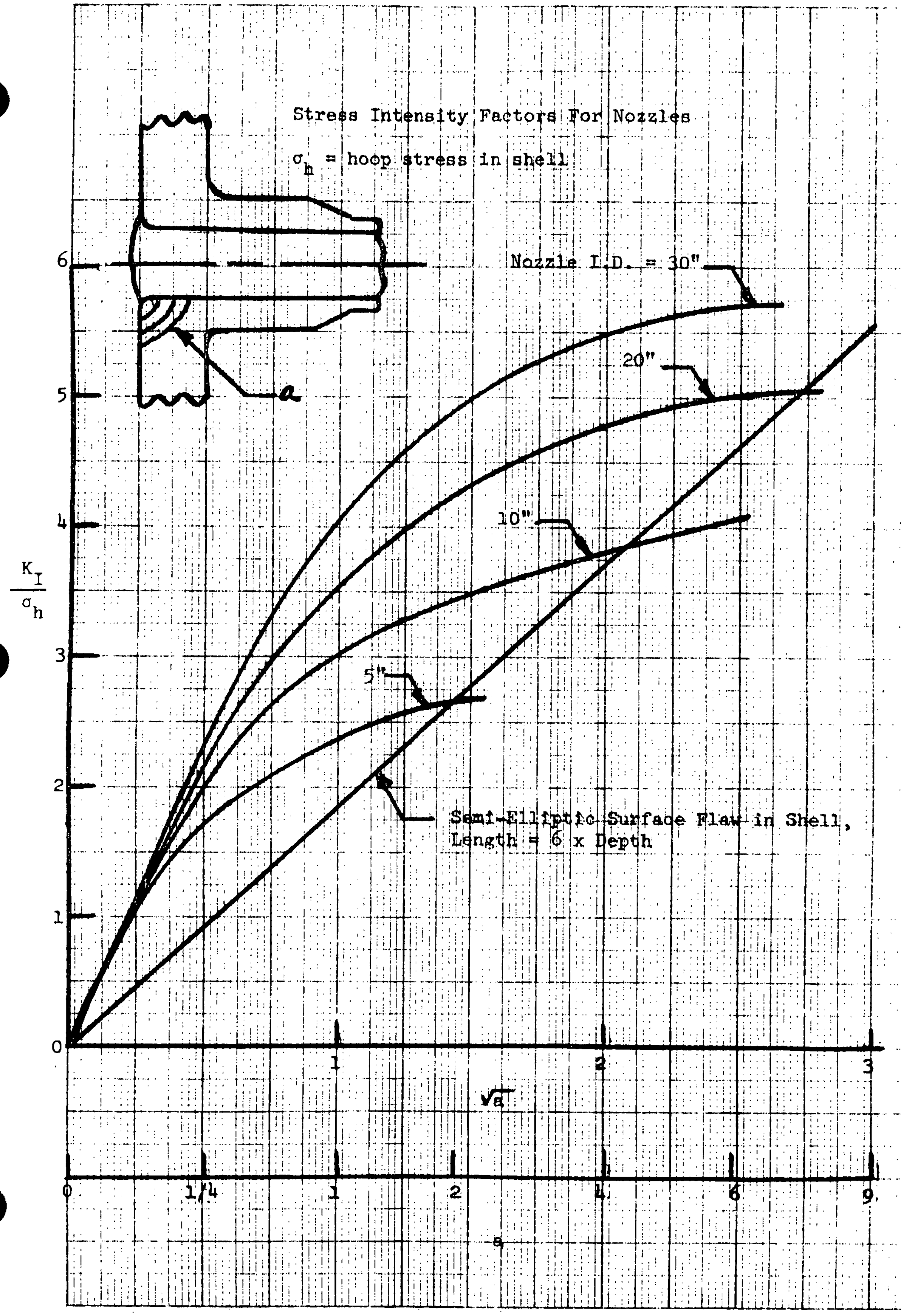
M_t and M_b vs Wall Thickness
For
Semi- Elliptical Surface Flaw
1/4T Deep and 1 1/2T Long
σ/σ_y = 1.0
0.7
0.5
0.1
K_I (Tension) = M_t x Membrane Stress
M_b = 2/3 M_t
K_I (Bending) = M_b x Bending Stress At Outer Fiber
M_t
.6
.4
.2
3
.8
.6
.4
.2
2
.8
.6
.4
.2
1
.8
1
.2
.4
.6
.8
2
.2
.4
.6
.8
3
.2
.4
.6
.8
$\sqrt{\text{Thickness}}$ (in.)

FIGURE 3


Stress Intensity Factor Caused by Thermal Stress
K_I (Thermal) = M_θ x Temp. Difference Thru Wall (Crack Length = ∞)
Crack Depth = Thickness/4
Crack Depth = Thickness/8
M_0
0.6
0.5
0.4
0.3
0.2
0.1
0
1
2
3
4
5
6
7
8
9
10
11
12



Stress Intensity Factor Caused By Thermal Stress
For Cylinders with Radius/Thickness = 10.
Cooling Rate 50°F/Hr.
30°F/Hr.
20°F/Hr.
10°F/Hr.
Stress Intensity Factor (ksi √inches)
Wall Thickness, Inches
0
2
4
6
8
10
12
14
16
18
20
2
3
4
5
6
7
8
9
10
11


Stress Intensity Factors For Nozzles
σ_h = hoop stress in shell
a
Nozzle I.D. = 30"
20"
10"
5"
Semi-Elliptic Surface Flaw in Shell,
Length = 6 x Depth
$\frac{K_I}{\sigma_h}$
0
1
2
3
4
5
6
$\sqrt{a}$
1/4
9
a

APPENDIX I

The Use of Lateral Expansion in the Charpy V-Notch Impact Test

For many years, standards and code-writing bodies specified a constant value for minimum energy absorption as the criterion for toughness when judged by the Charpy impact test. The energy absorption value was selected arbitrarily without consideration of the strength level of the steel.

It is now widely recognized that energy absorption must be increased with strength in order to maintain a constant fracture toughness. Energy absorption is controlled by two factors—the strength of the steel which regulates the force required to deform the Charpy specimen, and the ductility of the steel which determines the distance through which the force acts during testing. Since loss of fracture toughness is due to loss in ductility rather than strength, a criterion which evaluates notch ductility rather than energy is a more significant and universal index.

An index of ductility can readily be obtained from Charpy specimens my measuring lateral expansion of the specimen at the compression side directly opposite the notch. The procedure is described in ASTM specification A370-68. The ductility and energy relations in Charpy tests of carbon and alloy steels with yield strengths in the range of 30 ksi to 150 ksi were first described by Gross and Stout in 1958.[1)*] More extensive information was recently presented by Gross.[2,3)] The results showed, for example, that 15 mils lateral expansion corresponds to about 11 ft-lb for a 35 ksi yield strength (60 ksi tensile strength) steel, to about 15 ft-lb for a 55 ksi yield strength (85 ksi tensile strength) steel, and to about 22 ft-lb for a 120 ksi yield strength (133 ksi tensile strength) steel. The results supported the adoption by the ASME Boiler and Pressure Vessel Committee of a lateral-expansion requirement for quenched and tempered steels.

The PVRC Task Group on Toughness Requirements for Ferritic Materials determined, at 25, 35, 40, 45, and 55 mils lateral expansion, the relationship between energy absorption and yield strength, as shown in Figure I-1. The relationship is based on Charpy impact and tensile data for specimens from the surface, quarter-thickness, and center locations in thick sections of plate steel (A533B, A212B, A517-F and A543), forging steel (A508-2),

* See References.

weld metal joining A533B steel, and the weld heat-affected zone of A533B steel. Figure I-2 shows a plot of the data at 40 mils lateral expansion. Each data point was obtained from a singular relation between lateral expansion and energy absorption for a given steel with a known strength at the thickness location for the Charpy specimens.

In addition to the above mentioned graphs, a plot was made of lateral expansion vs energy absorption based on Charpy data obtained for the qualification of many plates of A533B steel used in actual vessel construction. The data were for plates with a yield strength of about 65 ksi at the quarter-thickness test location. The plot showed that 50 ft-lb corresponded to 40 mils lateral expansion for 65 ksi yield strength material.

The use of a Charpy V-notch requirement based on lateral expansion rather than energy absorption provides an automatic means for obtaining a constant level of fracture toughness irrespective of strength variations for material within a range of tensile properties in a specification or irrespective of deliberate changes in the specified tensile properties.

Only limited data are currently available for determining the relation between lateral expansion and energy absorption for steel with increased yield strength produced by irradiation. However, these data indicated, as expected, that a Charpy V-notch requirement based on lateral expansion provides an automatic means for obtaining a constant level of fracture toughness.

References

1. J. H. Gross and R. D. Stout, "Ductility and Energy Relations in Charpy Tests of Structural Steels," Welding Journal, 37 (4), Research Suppl., 151-s to 159-s (1958).

2. J. H. Gross, "The Effect of Strength and Toughness on Notch Ductility," Welding Journal, 48 (10), Research Suppl., 441-s to 453-s (1969).

3. J. H. Gross, "Transition-Temperature Data for Five Structural Steels," Welding Research Council Bulletin, No. 155, October 1970.

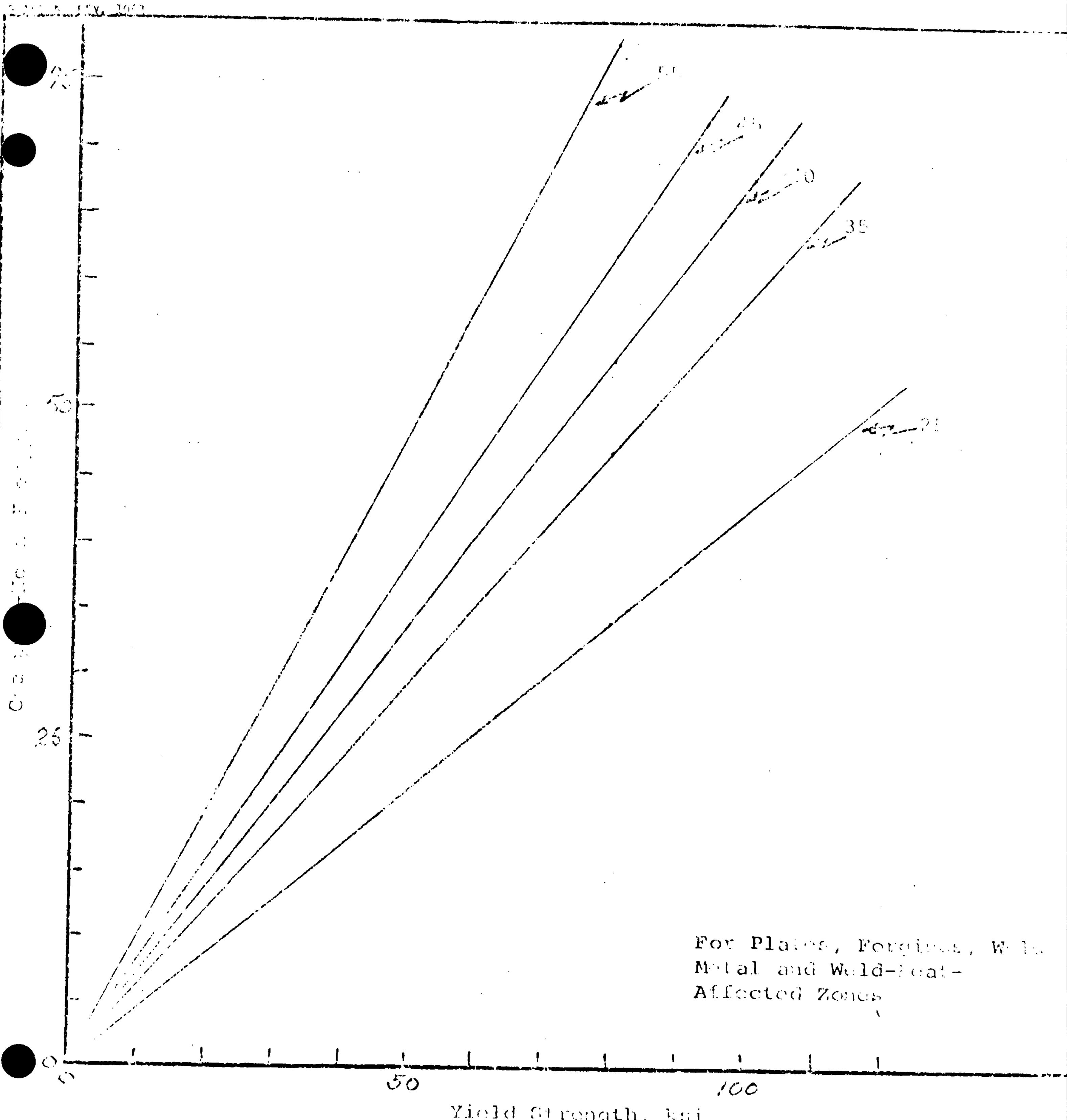


Figure J-1. Relationship, at 25, 35, 40, 45, and 50 mils Lateral Expansion, Between Energy and Yield Strength.

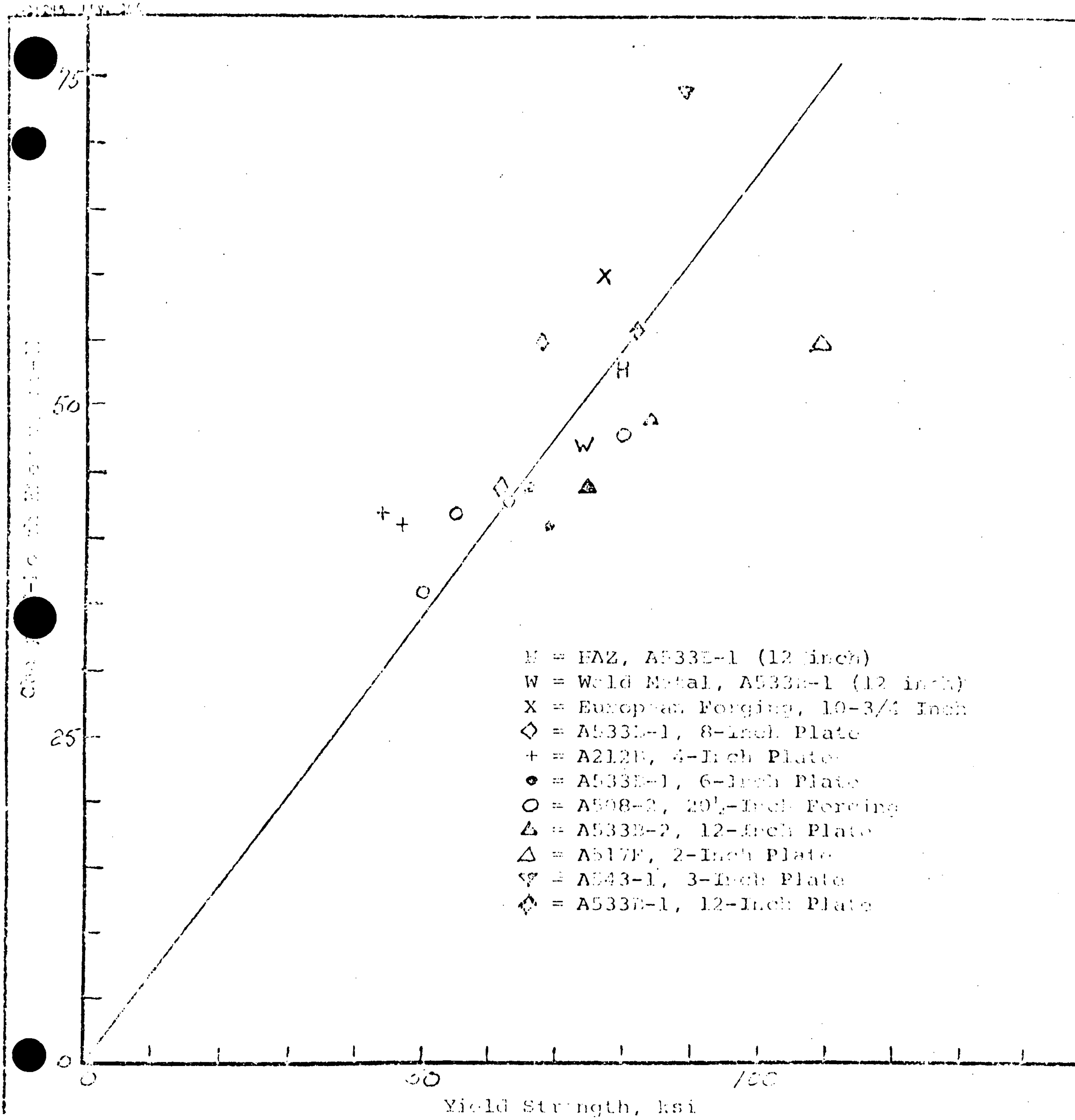


Figure I-2. Relationship at 40 mils Lateral Expansion [illegible] Energy, and Yield Strength

APPENDIX II

DERIVATION OF K_{IR} CURVE

In general, tests to determine K_{Ic}, K_{Id} and K_{Ia} are expensive and time consuming, but the advantages of using linear elastic fracture mechanics for fracture analysis are so strong that many ways have been developed to estimate critical stress intensity factors from data from other tests. Enough data exist on A533, Grade B, Class 1 and A508 to relate actual measured K_{Ic} values to results from other tests with a very high degree of confidence.

One of the more fundamental tests commonly used is the drop weight test. The NDTT so determined represents the upper temperature where brittle fracture occurs under defined conditions of strain and flaw size. Because conditions and geometry are defined, the critical stress intensity factor can be calculated, and this has been done by Irwin and others. The relationship between dynamic yield strength (Y_d) and K_{Id} resulting from this work is:

$$K_{Id} = (\text{Factor}) \times Y_d$$

where the factor varies from 0.634 to 0.78.

The development of the K_{IR} - temperature curve (where temperature is relative to the NDTT) was done by plotting all known data - K_{Id}, K_{Ia}, and low temperature K_{Ic} versus the drop weight NDTT of the same plate or forging. A lower bound curve was drawn and the values at NDTT checked against the analytical relationships discussed above. The value of K_{IR} at NDTT on this curve is .55 of the dynamic yield of the steels tested. The resulting curve, therefore, represents a very conservative assumption as to the critical stress intensity - temperature properties of materials similar to those tested, as related to the measured NDTT.

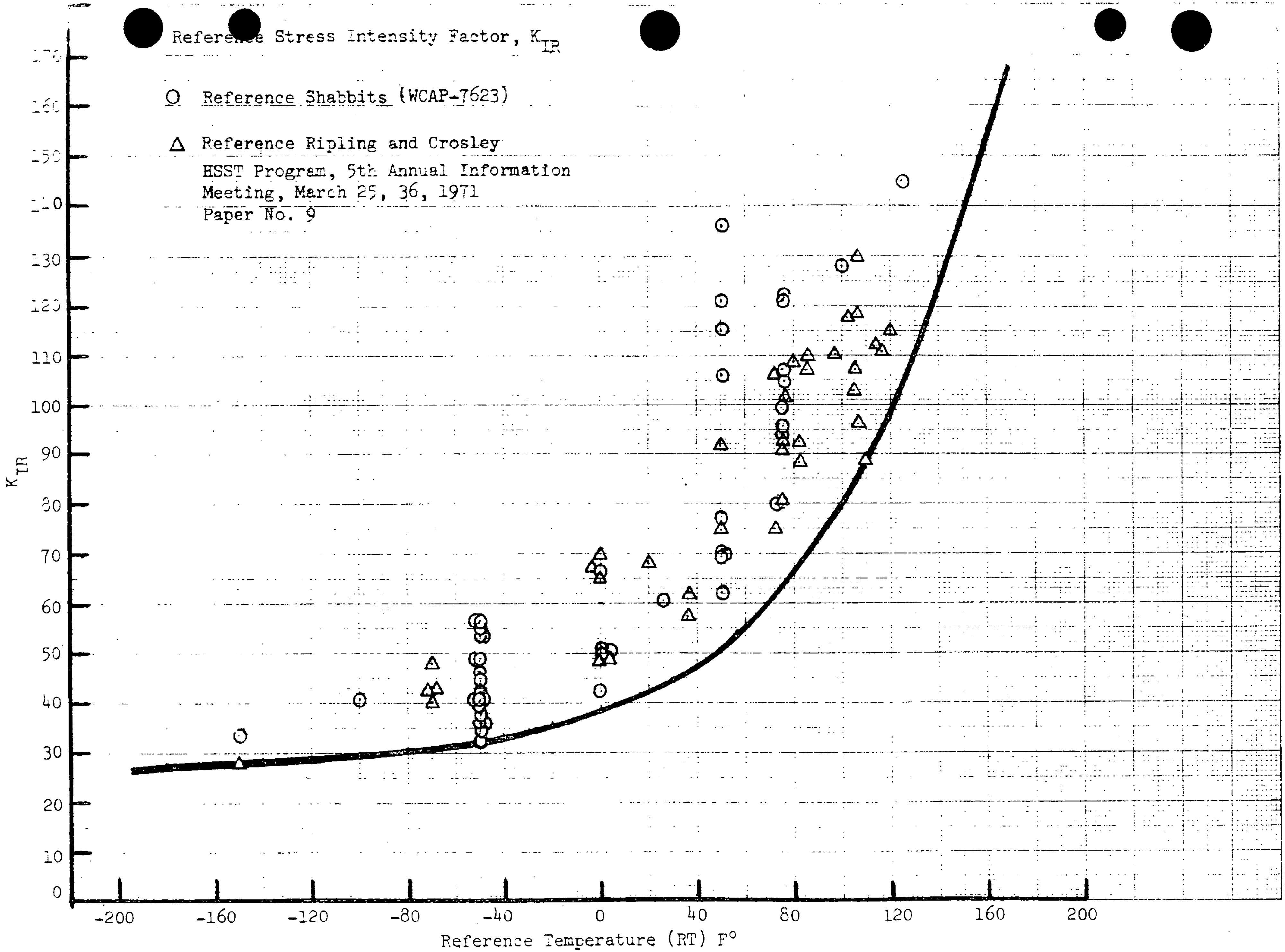
The use of all dynamic and arrest data, even though vessel loading is essentially static, also represents a large degree of conservatism.

Figure II-1 shows the K_{IR} - temperature curve, on which are plotted all pertinent data available at present. Only K_{Id} and K_{Ia} data are shown, because K_{Ic} values are invariably much higher, so they do not affect the lower bound.

To facilitate computer calculations, the equation representing this curve is also shown, and actual values for reference areas follows.

$$K_{IR} = 26.777 + 1.223 \exp \left\{0.014493 \, [T-(NDT-160)]\right\}$$

RT_{NDT}	K_{IR} (ksi $\sqrt{\text{inch}}$)
NDT - 160	28.00
NDT - 140	28.41
NDT - 120	28.96
NDT - 100	29.7
NDT - 80	30.68
NDT - 60	31.99
NDT - 40	33.74
NDT - 20	36.08
NDT	39.21
NDT + 20	43.39
NDT + 40	48.97
NDT + 60	56.44
NDT + 80	66.41
NDT + 100	79.73
NDT + 120	97.54
NDT + 140	121.33
NDT + 160	153.13
NDT + 180	195.61


Reference Stress Intensity Factor, K_{IR}
O Reference Shabbits (WCAP-7623)
△ Reference Ripling and Crosley
HSST Program, 5th Annual Information
Meeting, March 25, 36, 1971
Paper No. 9
K_{IR}
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
-200
-160
-120
-80
-40
0
40
80
120
160
200
Reference Temperature (RT) F°

APPENDIX III

DERIVATION OF FIGURE 2

Figure 2 presents a simple way to determine the K_I resulting from either a tensile or bending stress in a plate or cylinder, using the specific flaw size and shape chosen for reference. All correction and shape factors required to calculate K_I have been combined into a single term. (M_t and M_b as the case may be). Because the absolute size of the "reference" flaw varies with thickness, M_t and M_b are shown as a function of the square root of the thickness of the components.

The analytical basis for the calculated values is presented here to permit calculations of a similar nature using flaw sizes and shapes different from that used as the basic reference.

Although there are several expressions that have been developed for the K_I of part-through surface cracks, the differences between them are relatively minor for crack sizes and shapes similar to that chosen as a reference. The particular expression chosen for derivation of Figure 2 is recommended in Reference (1), and represents the cumulative work of some of the most active analysts. Those interested in details of the development of the expression and correction factors applied should refer to this original work.

The basic expression for K_I in tension for a semi-elliptical part through surface crack used here is:

(1) $$K_I = 1.1 \sqrt{\pi} M_k \sigma \sqrt{a/Q}$$

where: a is crack depth

Q is the flaw shape factor modified for plastic zone size

M_k is a correction factor dependent on the relative depth of the crack

Values of Q for the 1/4 thickness flaw of depth to length ratio of .167 at various stress levels are:

$\sigma/\sigma y$ ratio	Q	Basic Expression for Q
.1	1.235	$Q = [\Phi^2 - .212\ (\sigma/\sigma ys)^2]$
.3	1.215	
.5	1.190	where: $\Phi = \int_0^{\pi/2} \sqrt{1 - [\frac{b^2 - a^2}{b^2}] \cos^2 \phi}\ d\phi$
.7	1.135	
1.0	1.030	

M_K as a function of relative flaw depth is shown in Figure III-1. The value for the 1/4 thickness flaw (a/t - .25) is 1.075.

The expression for K_I in bending is:

(2) $$K_I = M_b\ \sigma\sqrt{\pi}\ \sqrt{a/Q}$$

where: M_b is shown as a function of relative flaw depth in Figure III-2.

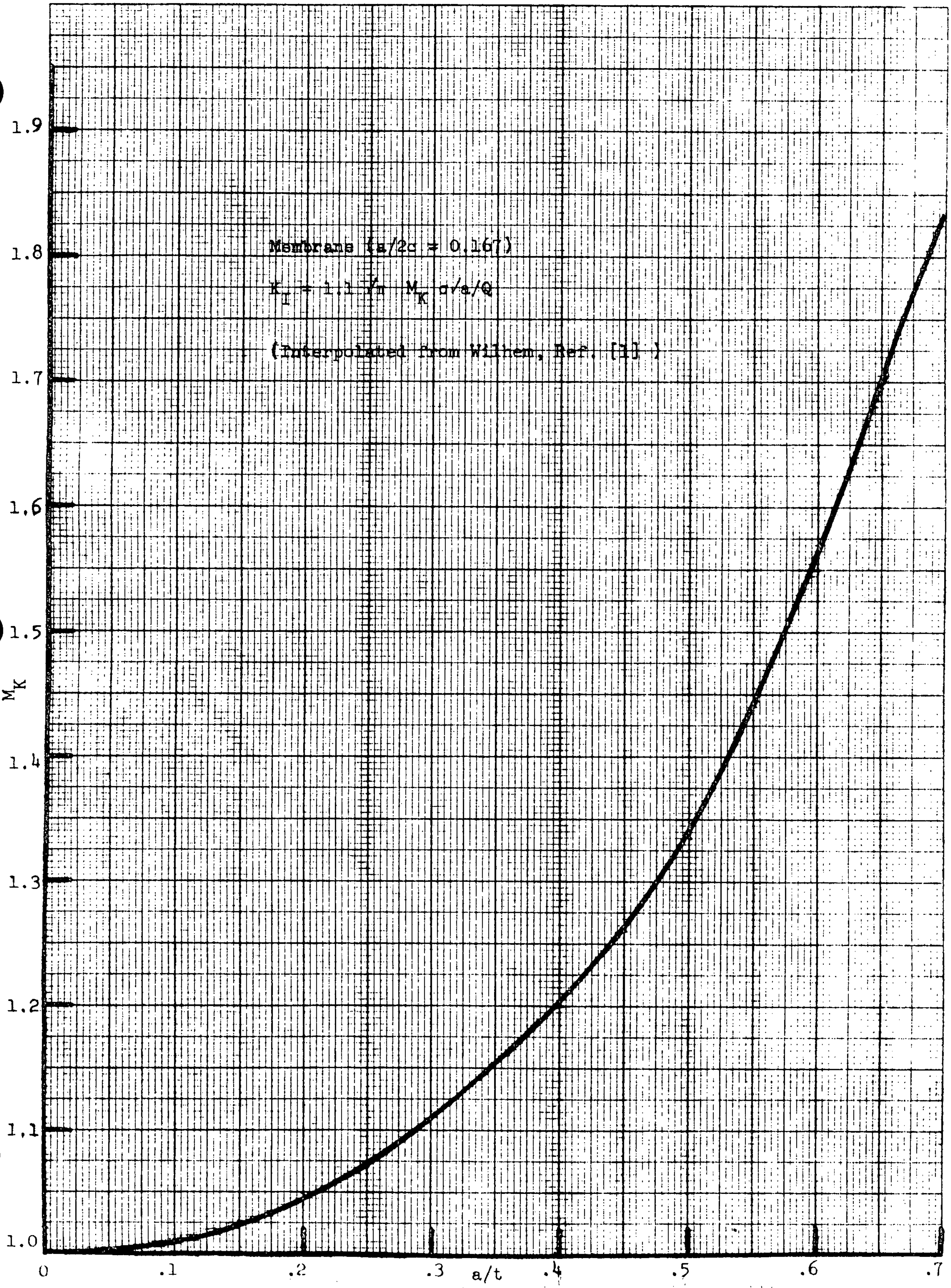
For the reference flaw selected, the value of M_b is .67 (or 2/3) of the value of M_I, so separate curves for M_b are not shown.

Because non-destructive test procedures used are not strongly dependent on thickness above 12 inches or under 4 inches, M_t is shown as constant beyond these values.

It is felt that the size of the flaw selected for reference purposes is very large in comparison with any flaw that could possibly be missed by the inspection procedures used. Calculations of K_I caused by flaws of more realistic size can be performed using the basic information presented in this Appendix, or by using other appropriate expressions for the type of flaw considered.

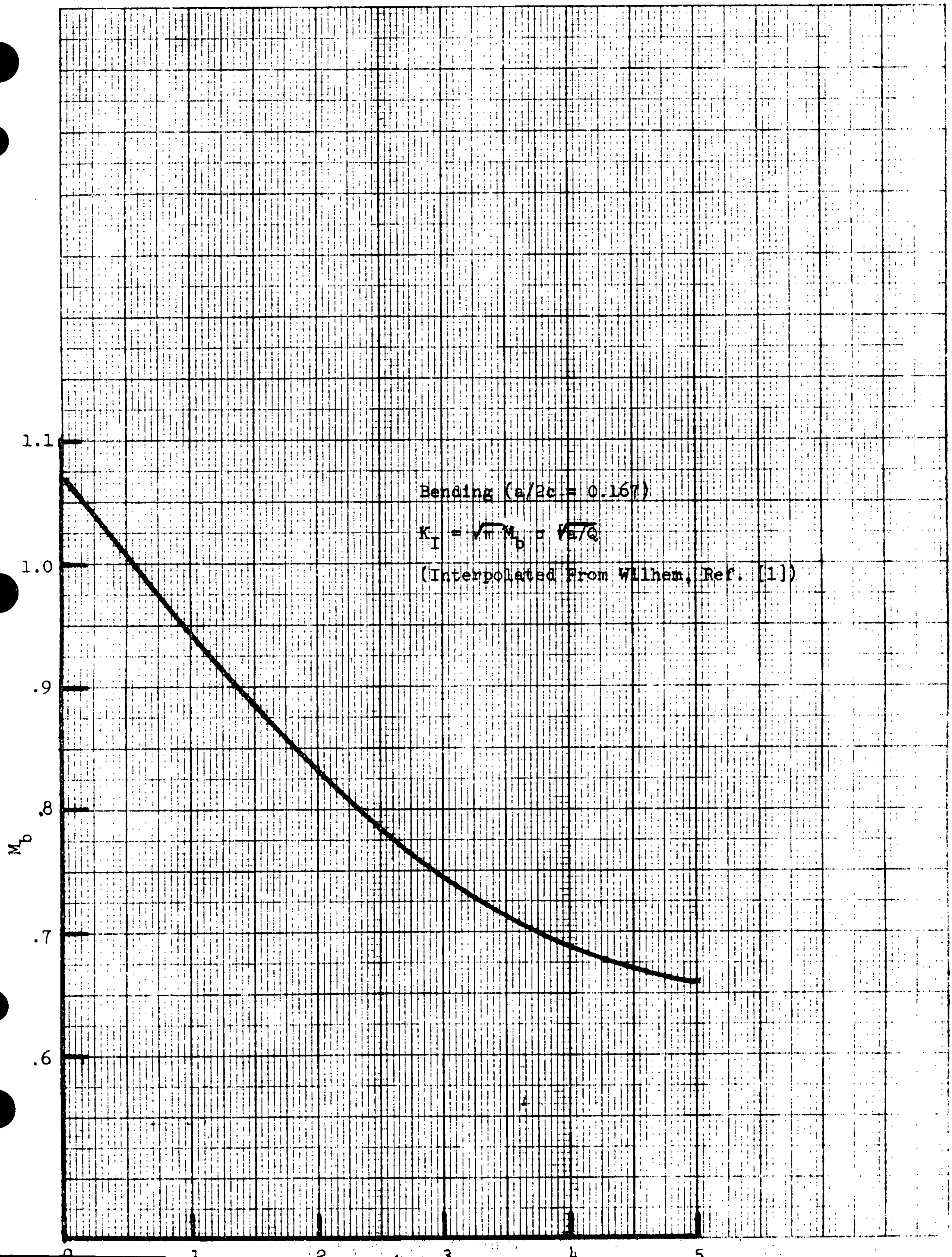
Reference (1) - AFFDL-TR-69-11, Fracture Mechanics Guideline for Aircraft Structural Applications, D. P. Wilhem, Northrup Corporation, February, 1970

Figure III-1



Membrane (a/2c = 0.167)
$K_I = 1.1 \sqrt{\pi}\ M_K\ \sigma \sqrt{a/Q}$
(Interpolated from Wilhem, Ref. [1])
M_K
1.9
1.8
1.7
1.6
1.5
1.4
1.3
1.2
1.1
1.0
0
.1
.2
.3
.4
.5
.6
.7
a/t

FIGURE III-2



Bending (a/2c = 0.167)
$K_I = \sqrt{\pi}\, M_b\, \sigma \sqrt{a/Q}$
(Interpolated From Wilhem, Ref. [1])
1.1
1.0
.9
.8
.7
.6
M_b

APPENDIX IV

COMPUTATION OF THE STRESS INTENSITY FACTOR AT THE TIP OF A CRACK IN A THERMAL STRESS FIELD

The computation of the stress intensity factor at the tip of a crack in a thermal stress field has been discussed in previous reports.[1,2] The values predicted by the closed form solution usually referred to as the Irwin method were compared to the results of a more complex and detailed numerical finite element analysis.[3] The Irwin method was found to give adequate results for cracks less than thirty percent through the vessel wall. This method, which is derived in Appendix IVA, computes the stress intensity factor from the thermal stress distribution in the uncracked wall. A FORTRAN program which computes the stress distribution and the stress intensity factor from the temperature distribution is presented in Appendix IVB. Using this program, the stress intensity factor for a long edge crack in a plate or vessel can be computed for all transient and steady-state temperature situations.

In order to illustrate typical values of the stress intensity factor caused by thermal stress, some assumptions have been made. To obtain Figure 3 of the Proposed Criteria, the shape of the steady-state temperature distribution was assumed to be as shown in Figure IV-1. Using the computer program in Appendix IVB, the stress intensity factors for various thicknesses and various temperature differences through the vessel wall were computed.

Other simplifying assumptions can lead to approximate expressions. For example, if the temperature distribution is assumed to be parabolic, the stress intensity factor can be expressed by

$$K_I = 0.823 \left[\frac{2}{3} \frac{E \alpha \Delta T}{(1 - \nu)} \right] \sqrt{B}$$

where: ΔT = temperature difference through wall

B = wall thickness

for the case where the crack depth is one-quarter of the thickness.

The computer program in Appendix IVB can be used to construct designer-oriented charts and graphs for most design and analysis applications.

REFERENCES

1. Tuppeny, W. H., Jr., Siddall, W. F., Jr. and Hsu, L. C., "Thermal Shock Analysis on Reactor Vessels Due to Emergency Core Cooling System Operation," Combustion Engineering, Inc., Report A-68-9-1, March 15, 1968.

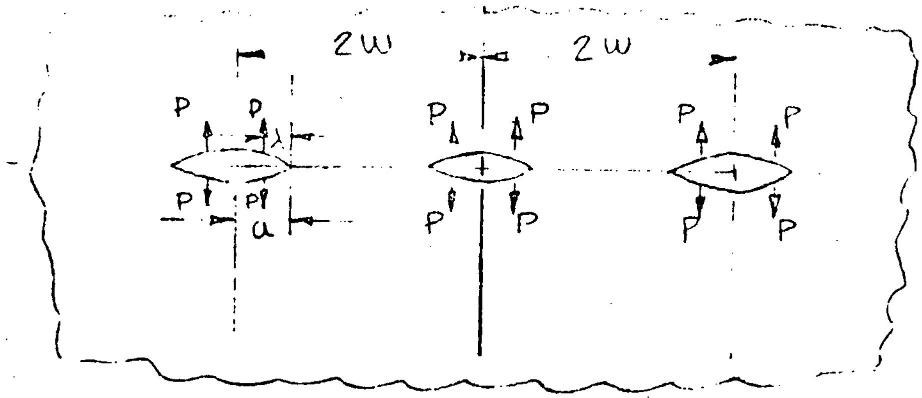
2. Hutto, R. C., Morgan, C. D., Van Der Sluys, W. A., "Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock," Babcock & Wilcox Power Generation Division, Topical Report BAW10018, May 1969.

3. Ayres, D. J., Siddall, W. F., Jr., "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel During Emergency Core Cooling," Combustion Engineering Report A-70-19-2, January 1970, also presented at Petroleum Mechanical Engineering Pressure Vessel and Piping Conference, Denver, Colorado, September 1970.

APPENDIX IVA

DERIVATION OF IRWIN METHOD

Consider the colinear array of cracks with boundary forces shown in the following sketch.



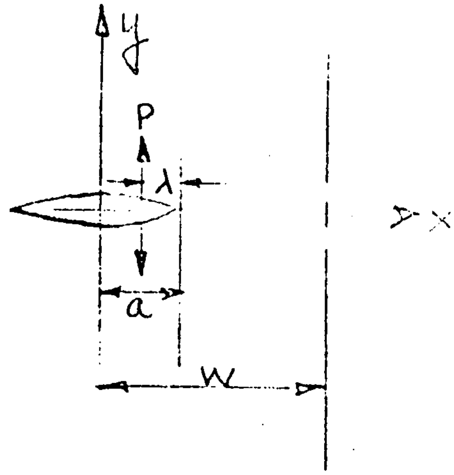
Let $\bar{\bar{Z}}$, $\bar{Z}$ and Z' represent successive derivatives with respect to ζ of a function $\bar{\bar{Z}}(\zeta)$ where ζ is (x + i y). Assume that the Airy stress function may be represented by

$$F = Re\,\bar{\bar{Z}} + y\, I_m\, \bar{Z}$$

$$\sigma_x = \frac{\partial^2 F}{\partial y^2} = Re\, Z - y\, I_m Z'$$

$$\sigma_y = \frac{\partial^2 F}{\partial^2 x} = Re\, Z + y\, I_m Z'$$

$$\tau_{xy} = \frac{\partial^2 F}{\partial x \partial y} = -y\, Re\, Z'$$



This semi-inverse method for solving stress distribution in the neighborhood of a crack was first shown by Westergaard. Westergaard's method has been found to be useful to the class of problems where $\tau_{xy} = 0$ along the x-axis.

It can be shown for a large plate containing a series of colinear symmetric cracks along the x axis with

a) zero shear stress along the x axis

b) self-equilibrating type stress field

c) crack lengths which are small compared to the plate dimensions

that:

$$Z = \frac{P}{W}\,\frac{\cos\frac{\pi\lambda}{2W}\,\sin\frac{\pi J}{2W}}{\left\{\sin^2\frac{\pi J}{2W} - \sin^2\frac{\pi\lambda}{2W}\right\}}\left\{\frac{\sin^2\frac{\pi a}{2W} - \sin^2\frac{\pi\lambda}{2W}}{\sin^2\frac{\pi J}{2W} - \sin^2\frac{\pi a}{2W}}\right\}^{1/2}$$

$$\bar{\bar{Z}} = \int_0^J Z\,dJ = -\frac{2P}{\pi}\tan^{-1}\left\{\frac{1-\left(\frac{\cos\pi a/2W}{\cos\pi J/2W}\right)^2}{\left(\frac{\cos\pi a/2W}{\cos\pi J/2W}\right)^2 - 1}\right\}^{1/2}$$

Near The Leading Edge of Crack

$$Z\Big|_{J=a+r} = \frac{P}{W}\,\frac{\cos\frac{\pi\lambda}{2W}\sin\frac{\pi a}{2W}}{\left\{\sin^2\frac{\pi a}{2W} - \sin^2\frac{\pi\lambda}{2W}\right\}}\;\frac{\left\{\sin^2\frac{\pi a}{2W} - \sin^2\frac{\pi\lambda}{2W}\right\}^{1/2}}{\left\{\left(2\sin\frac{\pi a}{2W}\cos\frac{\pi a}{2W}\right)\frac{\pi r}{2W}\right\}^{1/2}}$$

$$= \frac{K_I}{\sqrt{2\pi r}}$$

Note $$\left\{\sin\pi\frac{(a+r)}{2W}\right\}^2 \simeq \sin^2\frac{\pi a}{2W} + 2\left(\sin\frac{\pi a}{2W}\cos\frac{\pi a}{2W}\right)\frac{\pi r}{2W}$$

$$\therefore\; K_I = P\left(\frac{3}{W}\right)^{1/2}\frac{\left(\tan\frac{\pi a}{2W}\right)^{1/2}\cos\frac{\pi\lambda}{2W}}{\left(\sin^2\frac{\pi a}{2W} - \sin^2\frac{\pi\lambda}{2W}\right)^{1/2}}$$

K_I can be modified for a free edge condition by the following

$$K_I' = 1.127\,K_I$$

A P P E N D I X IVB

FORTRAN PROGRAM OF IRWIN METHOD FOR COMPUTING STRESS INTENSITY FACTOR DUE TO THERMAL STRESS

```
SEND     14:38   26 MON 07/26/71

DIMENSION T(50),R(50),TITLE(17),RA(50),ELE(50)
DIMENSION DR(50),TREM(50),RAV(50),CORR(11),SETC(50)
DIMENSION COMN(20,20)
DIMENSION  EEF(50)
DIMENSION TAX(50)
CORR(1)=1.12
CORR(2)=1.14
CORR(3)=1.19
CORR(4)=1.[illegible]
CORR(5)=1.37
CORR(6)=1.51
CORR(7)=1.66
CORR(8)=1.87
CORR(9)=2.12
CORR(10)=2.44
CORR(11)=2.80
READ, ST,SR,XNC
100 FORMAT(3F10.0)
6 READ, N,E,ALP,PR,A,B,WK,RES
10 FORMAT(I9,7F9.0)
READ, (R(I),I=1,N)
20 FORMAT(7F10.0)
1 READ 2,(TITLE(I),I=1,17),NCOM
2 FORMAT(4X,17A3,F4.0)
5 FORMAT (4X,17A3)
79 FORMAT(1X,F4.0,2H #,7X,F10.3,F13.0,F15.0)
PRINT 8,TITLE
8 FORMAT(4X,17A3)
IF(NCOM)3,215,3
3 DO 4 J=1,NCOM
READ 5,(COMN(J,I),I=1,17)
PRINT 5,(COMN(J,I),I=1,17)
4 CONTINUE
215 CONTINUE
PRINT 13,A,B,E,ALP,PR,WK,RES
13 FORMAT( 8X,14HINSIDE RADIUS=,F10.3,14X,15HOUTSIDE RADIUS=,
+ F10.3//1X,21HMODULUS OF ELASTICITY,F10.3,4X,25HCOEFFICIENT OF
+ EXPANSION=,F10.3//5X,17HPOISSON'S RATIO=,F10.3,11X,18HMECHANICAL
+ STRESS=,F10.3//8X,26HRESIDUAL STRESS AMPLITUDE=,F6.0//1X,
+ 9HTHICKNESS,8X,6HRADIUS,5X,11HTEMPERATURE,3X,14HTHERMAL STRESS
+ 3X,11HS.I. FACTOR/)
24 CONTINUE
SIMA=WK
C12=E*ALP/(1.-PR)
PI=3.14159
CALL=B-A
N1=2.
JA=1
```

```
READ,(T(I),I=1,11)
DO 45 I=1,6
TAR(JA)=T(I)
45 JA=JA+5
JA=[illegible]
DO 46 I=7,11
TAR(JA)=T(I)
46 JA=JA+2
DO 44 JA=1,5
LA=JA+1
SU=(T(LA)-T(JA))/5.
NAT=JA*5-3
DO 44 I=1,4
TAR(NAT)=TAR(NAT-1)+SU
44 NAT=NAT+1
NAT=27
DO 53 JA=6,10
LA=JA+1
SU=(T(LA)-T(JA))/2.
TAR(NAT)=TAR(NAT-1)+SU
53 NAT=NAT+2
DO 57 I=1,36
57 T(I)=TAR(I)
15 FORMAT(7F10.0)
DO 25 I=1,N
25 RA(I)=(B-A)*R(I)/100.+A
A2=A**2
B2=B**2
FACT=B2-A2
K=N-1
SUM=0.
DO 50 I=1,K
J=I+1
DR(I)=RA(J)-RA(I)
RAV(I)=(RA(J)+RA(I))/2
DRTC(I)=RES*COS(2.*PI*(RAV(I)-A)/WALL)
TAV=(T(J)+T(I))/2
ELE(I)=RAV(I)*TAV*DR(I)
50 SUM=SUM+ELE(I)
TOT=0
DO 75 I=1,K
J=I+1
75 EEE(J)=ELE(I)
EEE(1)=0.
TOT=0.
DO 80 I=1,N
GEOM=(RA(I)**2+A2)/FACT
RAT=GEOM*SUM
TOT = TOT + EEE(I)
SAT=-T(I)*RA(I)**2
TSE(I)=(RAT+TOT+SAT)*C12/RA(I)**2
80 CONTINUE
```

```
M=1
PRINT 78,R(1),RA(1),T(1),TREM(1)
110 DO 115 I=1,N
IF(R(I)-ST)115,120,120
115 CONTINUE
120 KK=I-1
XL=RA(I)
ACC=0.
[illegible]=ST/2.
DO 130 [illegible]=1,11
[illegible]1
IF(FA-ACC) 132,132,130
130 CONTINUE
132 [illegible]=I+1
FACTOR=(1.+(FA-ACE)*(CONS(K)-CONS(I))+CONS(I))
FKAB=SIMA*SQRT((AL-A)*PI)*FACTOR
SA=SIN(PI*(XL-A)/2./WALL)
CA=COS(PI*(XL-A)/2./WALL)
GAM=SA*SQRT(2./WALL)/SQRT(SA*CA)
SA2=SA**2
DO 150 I=1,KK
J=I+1
TRAV=(TREM(J)+TREM(I))/2.+SRTC(I)
C=XL-RAV(I)
CC=COS(PI*C/2./WALL)
SC=SIN(PI*C/2./WALL)
SC2=SC**2
PROP=CC*TRAV*DR(I)/SQRT(SA2-SC2)
ACC=ACC+PROP
150 CONTINUE
M=M+1
ACCT=1.127*ACC*GAM+FKAB
ACC=ACCT/1000.
IF(ACC)160,161,161
160 ACC=0.
161 CONTINUE
PRINT 77,R(M),RA(M),T(M),TREM(M),ACC
77 FORMAT(1X,F4.0,2E #,7X,F10.3,F13.1,F15.0,7X,F10.1)
IF(SP-ST)175,175,170
170 ST=ST+XNC
GO TO 110
175 M=M+1
DO 180 I=M,N
PRINT 78,R(I),RA(I),T(I),TREM(I)
180 CONTINUE
GO TO 1
200 STOP
END
```

PROGRAM DATA

```
1200 [illegible]
1201 [illegible]
1202 26, [illegible]6.5, 7.5, .3, 100., 109.5, [illegible]
1203 0, 2, 4, 6, 8, 10, 1[illegible], 14, 16, 1[illegible], [illegible], 22, [illegible]4, 26, 2[illegible], 30
1204 32, [illegible]4, 36, 3[illegible], 40, [illegible], 44, 46, 48, 50, 55, 60, 65, 70, 75,
1205 [illegible]0, [illegible]5, 90, 95, 100
1209 .01 [illegible] TEMPERATURE DIFFERENCE
1212 [illegible], [illegible]9.3, 3[illegible], 51.5, 64.5, 75.5, [illegible]4.[illegible], 91.[illegible], 95.[illegible], 9[illegible].0, 100.
```

LINE	DATA
1201	CRACK DEPTH AS A PERCENT OF WALL THICKNESS FOR FIRST K_I, LAST K_I, AND INCREMENT FOR INTERMEDIATE VALUES
1202	NUMBER OF POINTS FOR THERMAL STRESS OUTPUT, YOUNGS MODULUS $*10^{-6}$, COEFFICIENT OF EXPANSION X 10^6, POISSON'S RATIO, INSIDE RADIUS, OUTSIDE RADIUS, MECHANICAL STRESS AVERAGE, RESIDUAL STRESS AMPLITUDE (COSINE DISTRIBUTION ASSUMED)
1203	PERCENT OF WALL THICKNESS AT WHICH
1204	THERMAL STRESS IS OUTPUT
1205	
1209	TITLE
1212	TEMPERATURES AT INSIDE WALL, 10%, 20% ... 100% OF WALL THICKNESS

SAMPLE OUTPUT

RUN

GEND 14:46 [illegible] MON 07/26/71

NOMINAL TEMPERATURE DIFFERENCE

INSIDE RADIUS= 100.000 OUTSIDE RADIUS= 109.500

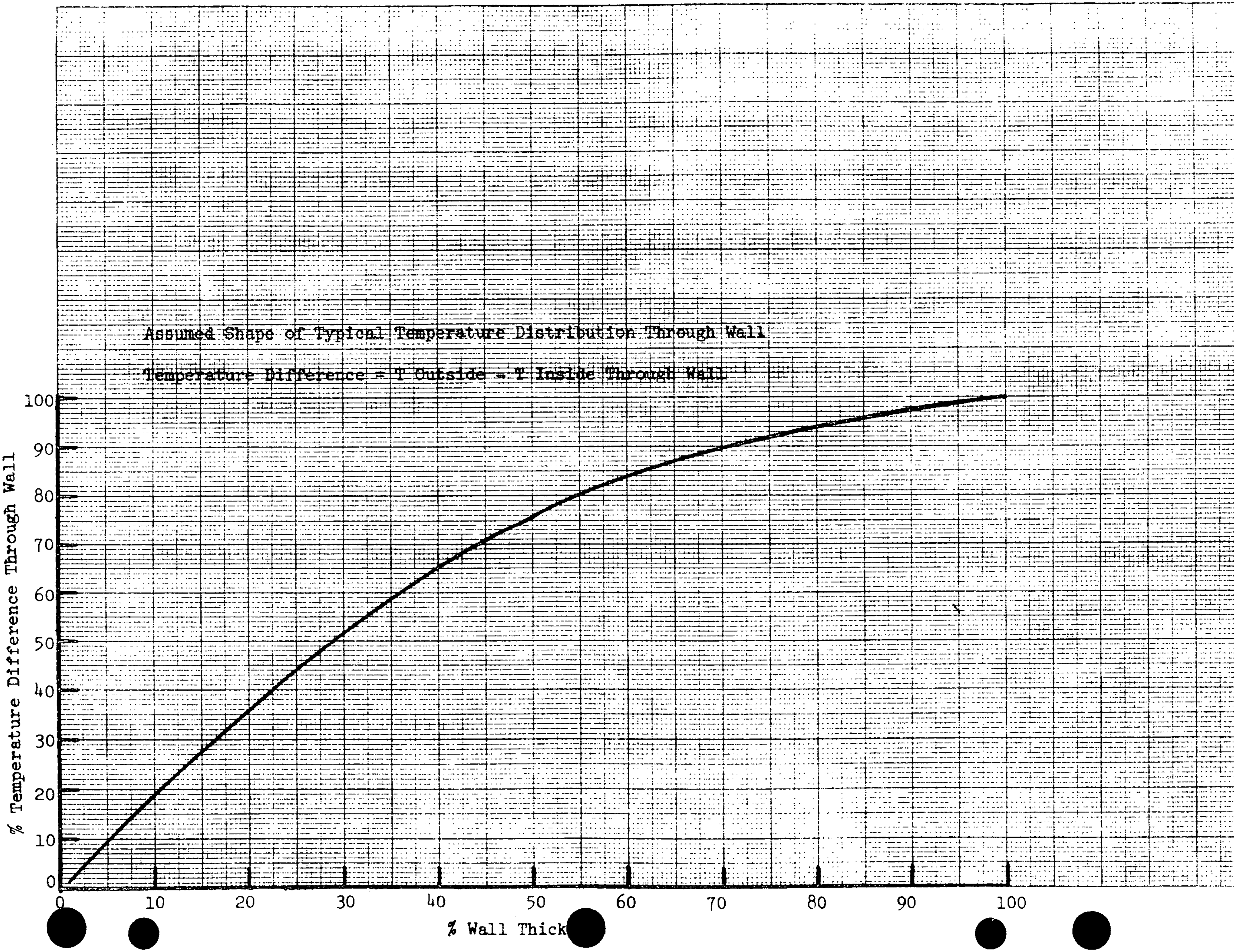
MODULUS OF ELASTICITY 26.500 $\times 10^6$ psi COEFFICIENT OF EXPANSION= 7.500 $\times 10^{-6}$ in/[illegible]

POISSON'S RATIO= .300 MECHANICAL STRESS= .000

RESIDUAL STRESS AMPLITUDE= .

CRACK DEPTH % THICKNESS	RADIUS	TEMPERATURE	UNCRACKED WALL THERMAL STRESS psi	S.I. FACTOR ksi $\sqrt{in}$
. %	100.000	.	19694.	
2. %	100.190	3.9	17963.	11.9
4. %	100.380	7.7	16834.	18.0
6. %	100.570	11.6	15707.	22.5
8. %	100.760	15.4	14583.	26.4
10. %	100.950	19.3	13461.	28.8
12. %	101.140	22.5	12517.	31.2
14. %	101.330	25.8	11575.	33.2
16. %	101.520	29.0	10634.	35.0
18. %	101.710	32.3	9696.	36.5
20. %	101.900	35.5	8759.	37.8
22. %	102.090	38.7	7836.	38.9
24. %	102.280	41.9	6914.	39.9
26. %	102.470	45.1	5994.	40.7
28. %	102.660	48.3	5076.	41.4
30. %	102.850	51.5	4159.	41.9
32. %	103.040	54.1	3414.	42.[illegible]
34. %	103.230	56.7	2671.	42.[illegible]
36. %	103.420	59.3	1929.	43.[illegible]
38. %	103.610	61.9	1189.	43.[illegible]
40. %	103.800	64.5	450.	43.[illegible]
42. %	103.990	66.7	-174.	43.[illegible]
44. %	104.180	68.9	-798.	43.[illegible]
46. %	104.370	71.1	-1420.	43.[illegible]
48. %	104.560	73.3	-2041.	43.[illegible]
50. %	104.750	75.5	-2660.	42.[illegible]
55. %	105.225	80.	-3879.	
60. %	105.700	84.	-5093.	
65. %	106.175	88.	-6032.	
70. %	106.650	91.	-6967.	
75. %	107.125	93.	-7573.	
80. %	107.600	95.	-8177.	
85. %	108.075	97.	-8494.	
90. %	108.550	98.	-8810.	
95. %	109.025	99.	-9055.	
100. %	109.500	100.	-9299.	

FIGURE IV-1



Assumed Shape of Typical Temperature Distribution Through Wall
Temperature Difference = T Outside - T Inside Through Wall
% Temperature Difference Through Wall
0
10
20
30
40
50
60
70
80
90
100
% Wall Thick

APPENDIX V

ESTIMATED K_I FOR FLAW AT INSIDE NOZZLE CORNER

Introduction

A complete and detailed analysis for the calculation of the stress intensity factor, K_I, for a flaw at the inside corner of a nozzle is not available. This is a difficult flaw geometry to analyze because of the complex stress distributions and gradients present in this region. However, an approximate analysis can be made and also, some limited results are available from recent and current finite element and experimental studies. These are described in this Appendix.

Estimates of K_I For Nozzle Corner Flaw

For a very simplified analysis, a nozzle in a pressure vessel can be considered as a hole in the shell. By further assuming a large shell radius, it can be considered as a hole in a flat plate. K_I analyses have been made for the case of cracks emanating from a hole in a flat plate. Paris and Sih[1] give the results for this geometry in the form:

$$\frac{K_I}{\sigma\sqrt{\pi a}} = F(a/r) \qquad \text{Eq. (V-1)}$$

where σ = stress

a = crack length

r = hole radius

Values of $F(a/r)$ are given by Paris and Sih for one crack and for two diametrically opposite cracks under uniaxial tension and equal biaxial tension. In the Eq. V-1 formulation, the crack extends completely through the plate thickness.

Eq. V-1 can be adapted to the pressure vessel case where a 2:1 stress biaxiality exists by assuming that the values of $F(a/r)$ are halfway between the uniaxial and the equibiaxial values. The shell hoop stress, σ_h, is used as the value of stress in the equation. The resulting values of $F(a/r)$ versus a/r for the single crack geometry are as shown by the solid curve in Fig. V-1. As mentioned earlier, this curve applies for a crack completely through the plate thickness. Analysis of K_I for related geometries would indicate that the $F(a/r)$ values would be smaller for a crack which only subtends the corner region of the hole-plate intersection; how much smaller is not known from the available analyses.

Two current studies provide information for improving the approximate formulation outlined above. One study is by Gilman[2] who used a three-dimensional finite element stress analysis method to calculate K_I for one specific nozzle configuration. The other is Derby's current study[3] on fracture testing of epoxy models of nozzle-vessel configurations provided with flaws at the nozzle corner. The study involves an indirect method of obtaining the nozzle K_I in which the K_{Ic} of the epoxy material is measured separately and used to derive K_I values from the model test results. Only initial and very preliminary results are currently available from Derby's work now in progress.

The principal dimensions of the vessel-nozzle geometries in Gilman's analysis and in Derby's tests to date are the following:

	Gilman	Derby
Vessel ID (in.)	206	5.38
Vessel Shell Thickness (in.)	5-7/8	.59
Nozzle ID (in.)	12	1.85
Nozzle Wall Thickness (in.)	6-5/8	.56
Inside Corner Radius (in.)	2.7	5/16

Since actual nozzles have radiused corners rather than a square corner, a convention needs to be adopted for expressing the ratio of flaw size to nozzle inside radius to maintain analogy with the a/r ratio in Eq. V-1. The convention adapted is to measure flaw size as the depth along the diagonal bisecting the nozzle corner. An "apparent" radius is then defined as the dimension from the nozzle axis to the point where the nozzle corner intersects the diagonal. The relationship is:

$$r_n = ri + 0.29\ r_c \qquad \text{(Eq. V-2)}$$

where r_n = apparent radius of nozzle

r_i = actual inner radius of nozzle

r_c = nozzle corner radius

The results of the finite element calculations and the epoxy model tests are shown in Fig. V-1 for comparison with the hole-in-plate model.

The $F(a/r_n)$ derived from finite element calculations roughly parallels but is lower than the $F(a/r)$ for the hole-in-plate by a factor ranging between 0.74 and 0.82. The epoxy model results give higher $F(a/r_n)$ values than the finite element analysis and in one instance, higher than the hole-in-plate value. However, these are only preliminary test results and should be given limited emphasis until additional results are available from further work in progress.

As an interim procedure, it is proposed that the curve drawn through the finite element results be used. This is very tentative inasmuch as it is a generalization of results for only one specific nozzle geometry.

The K_I analysis described above pertains only to pressure stresses. At present, there is no available K_I analysis of a nozzle corner flaw under thermal stress conditions.

Relation to Toughness Criteria

Using the proposed curve in Fig. V-1, the effect of flaw size for various size nozzles can be represented as shown by Fig. 4 in the main section of the Proposed Criteria. In deriving the curves for various nozzle sizes, it was assumed that all had a 1" radius inside corner. For comparison, Fig. 4 also includes a curve for the reference geometry flaw in the vessel shell.

Fig. 4 indicates that for certain combinations of nozzle size and K_I/σ_h values, the critical flaw size (depth) for the nozzle corner region may be as small as 1/4 of that for a flaw in the vessel shell. Stated another way, if equal flaw sizes are assumed for the nozzle corner and the shell, the toughness required in the nozzle material can be up to about twice the toughness required in the shell material.

The Proposed Criteria requires that the toughness requirements for the nozzle material shall be the same as for the shell to which it is attached. The implied consequence of this requirement is that the reference flaw size for the nozzle corner may be as small as 1/4 of the reference size in the shell. This means extra precautions are necessary in the inspection of this critical region.

It is recognized that the toughness requirements for nozzles may result in interpretation questions in certain design analysis situations. An example would be if the nozzle material experiences transient cooling relative to the vessel material. In this and similar situations, the general method of analysis would be:

1. Determine the lowest temperature in the nozzle during the transient at a depth distance from the nozzle corner equal to 1/4 of thickness of the vessel shell.
2. The toughness required for the nozzle material would be the same as for the shell material at the temperature determined in step (1) for existing shell hoop pressure stress.

REFERENCES

1. P. C. Paris and G. C. Sih: "Stress Analysis of Cracks," Fracture Toughness Testing and Its Applications, ASTM STP-381, 1965.

2. J. D. Gilman: General Electric Company, San Jose, California - private communication.

3. R. W. Derby: Oak Ridge National Laboratory, Oak Ridge, Tennessee - private communication of Heavy Section Steel Technology Program results.

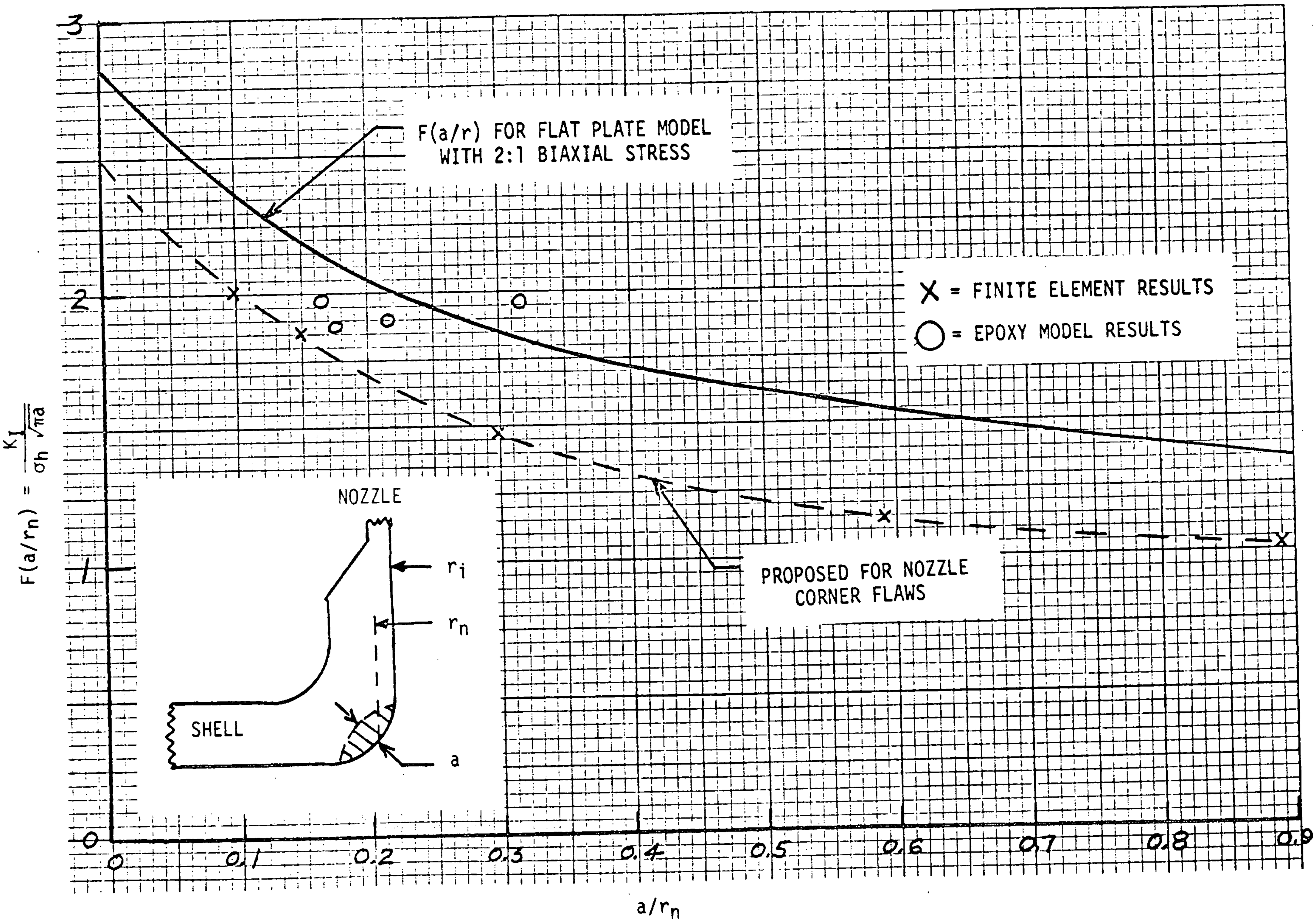


FIG. V-1 ESTIMATED STRESS INTENSITY FACTORS FOR FLAWS AT A NOZZLE CORNER

APPENDIX VI

TOUGHNESS REQUIREMENTS FOR REACTOR PIPING

Introduction

From the standpoint of primary membrane stresses alone toughness requirements for reactor piping material are modest compared to the pressure vessel material. This results from the fact that pipe wall section thicknesses are not great enough to support plane strain fracture propagation in any but the thickest of pipes at the lowest possible service or hydrotest temperature and from the fact that the size limit for flaw detection decreases with wall thickness while the load limit-flaw size relation changes but slightly. In the pressure vessel wall away from nozzles or similar fittings, the maximum principal stress is the hoop stress; in the case of a segment of pipe this may also be the case, but there is always a possibility that the maximum principal stress will lie in the axial direction and will be of yield level as a result of cold bending introduced in fabrication of the piping system.

As was the case with the pressure vessel, the relationship between material toughness and critical flaw size in piping may be estimated by application of some of the mathematical relationships of linear elastic fracture mechanics. It should be emphasized that because of the smaller section thicknesses in pipe, load limits calculated by fracture mechanics will in general be more conservative for pipe than for pressure vessels and other thick section structures in which the elastic constraint needed for plane strain crack propagation can be developed. The relationship between flaw size and load limit for some simple flaw shapes will be considered for both circumferential and axial flaws, although only the former will be of first importance in the case of piping toughness considerations.

Axial Through-Wall Flaws

Stress intensity factor calculations for axial flaws in cylindrical shells of the size of pressure vessels are reasonably straightforward. As the cylinder diameter decreases, the magnitude of the correction necessary to account for bulging of the flawed region under pressure increases and, in the wall thickness and diameter range covered by pipes, becomes a significant factor in the stress intensity factor expression. A number of workers have studied pressure load limits for axially flawed pipes and have developed relationships between flaw size and load limit from both theoretical and empirical viewpoints.[1-6] These relationships are, for the through-wall case, of the general form:

$$K = A \sigma c^{m} \tag{1}$$

in which K is a material toughness parameter, A is a geometrical factor, c is the crack half-length, and m is a number between 1/2 and 3/2. For the case of pure plane strain (brittle) fracture in a cylindrical shell of large diameter, the effect of bulging is negligible and (1) reduces to the wide-plate formula of linear elastic fracture mechanics:

$$K_{Ic} = \sigma \sqrt{\pi c}. \tag{2}$$

As cylinder diameter decreases, the effect of bulging increases and the exponent m of equation (1) approaches the value 3/2. Figures 1 and 2 give burst hoop stresses at $60^{o}F$ for ASTM A106B pipes containing axial through-wall flaws of different lengths.[5] When these data were used in an empirical expression containing a modification of the Folias bulging correction and a plastic zone correction an "effective stress intensity factor" having a value of about 95 ksi $\sqrt{in}$ was obtained.

Axial Part-Through Flaws

In the same study it was found that full-length axial part-through flaws of constant depth reduced the room temperature burst pressure of ASTM A106B pipe in approximate proportion to the depth of the flaw; in other words a long axial part-through flaw of depth equal to one-fourth the wall thickness reduced the burst pressure by one-fourth. The sharpness of the flaw tip was found to have an insignificant effect on load limit in the room temperature tests. Applicability of these results to reactor piping systems is supported by results of the work of Wellinger, Sturm, et al,[6] who also found flaw sharpness to have slight effect (5% or less) on the burst strength of flawed pipes.

Kihara, Ikeda and Iwanaga[2] carried out burst tests on flawed segments of electric-resistance welded pipe of semikilled steel* at temperatures ranging from $-70^{\circ}C$ to $+10^{\circ}C$ and found burst hoop stress to be but slightly dependent on temperature over this range.

Circumferential Flaws

Since the axial stress resulting from internal pressure in a cylindrical shell is one-half the circumferential or hoop stress, a larger circumferential flaw is required to cause a failure under pressure loading alone than is the case for an axial flaw. There are few published experimental data on the strength reduction due to circumferential flaws in pipes either for pressure loading alone or for combined pressure and bending. There are published papers on calculation of stress intensity factors applicable to circumferential cracks in pipes under combined pressure and bending loads. In the absence of substantial experimental data, about the best one can do to estimate pipe toughness requirements is to use the published elastic fracture mechanics relationships to estimate stress intensity factors for plausible flaw geometries, realizing that such calculations will be conservative in most cases.

Two ground rules must apply to this problem: (1) The lowest full-load service temperature will be $40^{\circ}F$, and (2) installation practices are such that it must be assumed that bending moments great enough to give yield level surface fiber stresses in the axial direction may exist in any pipe segment.

Through-Wall Flaw

Treating first the through-wall circumferential flaw, one may take as a "standard" flaw one of length equal to twice the pipe wall thickness. Gilman[7] gives a general expression for the stress intensity factor applicable to a circumferential through-wall crack type flaw in a pipe under a bending moment, M, plus internal pressure, P:

$$K = \sqrt{\pi c}\left[\frac{PR}{2t} F_1(\beta) + \frac{M}{\pi R^2 t} F_2(\beta)\right] \tag{3}$$

* Probably equivalent to ASTM A135.

in which c is the crack half-length, R is the pipe mean radius, t is the pipe wall thickness and F_1 and F_2 are functions of β, the half angle subtended by the crack. For the case of a crack of length equal to twice the wall thickness, it is seen that:

$$\beta = t/R \tag{4}$$

and equation (3) may be written:

$$K = \sqrt{\pi t}\,\left[\frac{PR}{2t} F_1 \left(\frac{t}{R}\right) + \frac{M}{\pi R^2 t} F_2 \left(\frac{t}{R}\right)\right] \tag{5}$$

in which t/R is simply the wall thickness to radius ratio for the pipe in question.

It will be noted in Table 1 that the largest value of t/R is 0.244. This corresponds to a half-angle β of approximately 14°. For this value of β, Gilman indicates that both $F_1(\beta)$ and $F_2(\beta)$ have values less than 1.1. This indicates that under these conditions an error of less than 10% would be made in the estimation of K if the simple wide-plate formula were used instead of equation (5), i.e.,

$$K = \sigma \sqrt{\pi t} \tag{6}$$

in which,

$$\sigma = \frac{PR}{2t} + \frac{M}{\pi R^2 t} \tag{7}$$

The value of K/σ, calculated from (6) above, is given for each pipe size in Table 1. The minimum values of K required to keep a 2t circumferential crack subcritical are plotted against wall thickness in Figure 5 for two cases; pressure stress only* and pressure plus bending at an assumed yield level of 45 ksi.

---

* The code allowable stress for A106B pipe ranges from 20 ksi at room temperature to 17.3 ksi at 600°F. Assuming this stress in the hoop direction, the axial pressure stress will be half this value; hence the plot in Figure 5 is based on a value of 9.5 ksi.

Part-Through Flaw

Harris[8] has developed an expression for the stress intensity factor for a hollow cylinder with a 360° circumferential part-through flaw under combined axial tension and bending load. This expression may be written in the form:

$$K_1 \simeq \sigma_p \sqrt{\pi a}\,[0.80 + \frac{a}{R}(4 + 1.08\,\frac{R}{x})]^{-\frac{1}{2}}$$

$$+ \sigma_b \sqrt{\pi a}\,[0.80 + \frac{a}{R}(7.12 + 1.08\,\frac{R}{x})]^{-\frac{1}{2}} \qquad (8)$$

in which:
- a = flaw depth,
- x = depth of ligament at base of flaw,
- R = mean radius,
- σ_p = axial gross section stress due to pressure,
- σ_b = maximum axial fiber stress due to bending,
- T = a + x = wall thickness.

If the "standard" flaw is assumed to be equal to T/R, equation (6) becomes:

$$K_1 \simeq \frac{1}{2}\,\sigma_p \sqrt{\pi T}\,[1.16 + \frac{T}{R}]^{-\frac{1}{2}}$$

$$+ \frac{1}{2}\,\sigma_b \sqrt{\pi T}\,[1.16 + 1.78\,\frac{T}{R}]^{-\frac{1}{2}} \qquad (9)$$

The quantity T/R is approximately constant for all pipes of the same "schedule". These constants are substituted into equation (9) giving:

Schedule 160:

$$K_1 \simeq 0.76\,\sigma_p \sqrt{T} + 0.71\,\sigma_b \sqrt{T} \qquad (10a)$$

Schedule 120:

$$K_1 \simeq 0.77\,\sigma_p \sqrt{T} + 0.74\,\sigma_b \sqrt{T} \qquad (10b)$$

Schedule 80:

$$K_1 \simeq 0.78\,\sigma_p \sqrt{T} + 0.76\,\sigma_b \sqrt{T} \qquad (10c)$$

Schedule 40:

$$K_1 \simeq 0.80\,\sigma_p \sqrt{T} + 0.79\,\sigma_b \sqrt{T} \qquad (10d)$$

one correlation between mils lateral expansion and Charpy foot pounds for A106B pipe; the 15 mil criterion should be met by this material at $40^{\circ}F$ since <u>the longitudinal direction only is of interest in the presence of large bending stresses</u>.

Calculations based on equation (6) indicate that enough toughness to sustain a 2T through-wall flaw at yield level stress can <u>not</u> be achieved in A106B type material at $40^{\circ}F$, however. Therefore if the 2T flaw condition must be met, large bending stresses <u>must</u> be avoided if the system is to be subject to full load at $40^{\circ}F$. Since the axial pressure stress contributes so little, perhaps any piping system which stays together at $40^{\circ}F$ might be assumed to be free from either large bending stresses or large flaws, although the remaining margin of safety would be uncertain.

Experience with piping systems indicates that equation (6) may give toughness requirements which are too conservative or that real piping systems are generally free of dangerous combinations of bending stress and flaw conditions. Considering the low incidence of pipe failures[9] a realistic pipe toughness requirement might be based on the geometric mean of the toughness requirements for the two "standard" flaw types. Expressed in terms of pipe wall thickness, this would take the form shown in Table 2. It should be reiterated that this table is based on a combination of less experimental data than would be desired plus fracture mechanics relationships which are quite conservative for small pipe sizes but become less conservative as pipe size increases. The critical stress level for a flawed structure is more dependent on flaw size than on structure section thickness. If the detection threshold for a flaw detection system is an approximately constant fraction of section thickness increasing care in both inspection and fabrication must be taken as section thickness increases.

References

1. A. Quirk, "Effects of Material Properties and Component Geometry on Unstable Propagation of Defects", Paper 69-SESA-2, Meeting of Society for Experimental Stress Analysis, Houston, Texas, October 1969.

Taking Schedule 80 pipe as typical and again assuming $\sigma_p = 9.5$ ksi and $\sigma_b = 35.5$ ksi, K is calculated as a function of T and plotted in Figure 6.

Toughness Requirements

When Figures 5 and 6 are compared, it is seen that from a standpoint of strength reduction, the 2T through-wall flaw is a considerably more severe defect than the T/4 part-through flaw. Also it is seen that nearly all of the toughness requirement arises from yield level or near-yield level bending stresses.

In order to translate the toughness requirements shown in Figures 5 and 6 into Charpy impact values, the Corten correlation (March 1971, HSST Meeting, Paper #3) may be used in absence of something better. Corten gives the relation:

$$K_{Id} = 1.587 (C_v)^{.375} \tag{11}$$

for the correlation between dynamic plane strain crack toughness K_{Id} and Charpy energy C_v. The pipe toughness requirements may be estimated to a good approximation by the relations:

$$K_{Id} = 1.8 \sigma \sqrt{T} \tag{12a}$$

for the 2T through-wall case and,

$$K_{Id} = 0.8 \sigma \sqrt{T} \tag{12b}$$

for the T/4 part-through case.

Charpy requirements based on (1) and (12) are plotted in Figure 7.

It will be seen from Figures 3 and 7 that if the T/4 part-through flaw is used as the reference case, there should be no difficulty in meeting the toughness requirement at 40°F. Our limited data (Figure 4) show a one to

2. H. Kihara, K. Ikeda, H. Iwanaga, "Properties of Brittle Fracture Initiation of Steel Pipe", The Sumitomo Search, No. 2, November 1969, pp 1-7.

3. R. J. Eiber, et al, "Investigation of the Initiation and Extent of Ductile Pipe Rupture", Battelle Memorial Institute, Columbus, Ohio, November 1968 (BMI-1853).

4. E. S. Folias, "An Axial Crack in a Pressurized Cylindrical Shell", Int. J. Fracture Mechanics, 1, 104 (1965).

5. M. B. Reynolds, "Failure Behavior of Flawed Carbon Steel Pipes and Fittings", General Electric Co., San Jose, Calif., October 1970 (GEAP-10236).

6. K. Wellinger, D. Sturm, P. Julisch, "Untersuchungen zur Rissansbreitung in Zylindrischen und Kugeligen Druckbehältern", Fortschr.-Bericht VDI-2 (1970) Reihe 5, Nr. 11.

7. J. D. Gilman, "Stress Intensity Factor for a Circumferential Through-wall Crack in a Straight Pipe". Report GEAP-5557, General Electric Co., APED, San Jose, Calif. (1968).

8. D. O. Harris, "Stress Intensity Factors for Hollow Circumferentially Notched Round Bars", Report UCRL-14859, Univ. of California Radiation Laboratory, Livermore, Calif.,(1966).

9. W. S. Gibbons, B. D. Hackney, "Survey of Piping Failures for the Reactor Primary Coolant Pipe Rupture Study". Report GEAP-4574, General Electric Co., APED, San Jose, Calif. (May 1964)

M. B. Reynolds

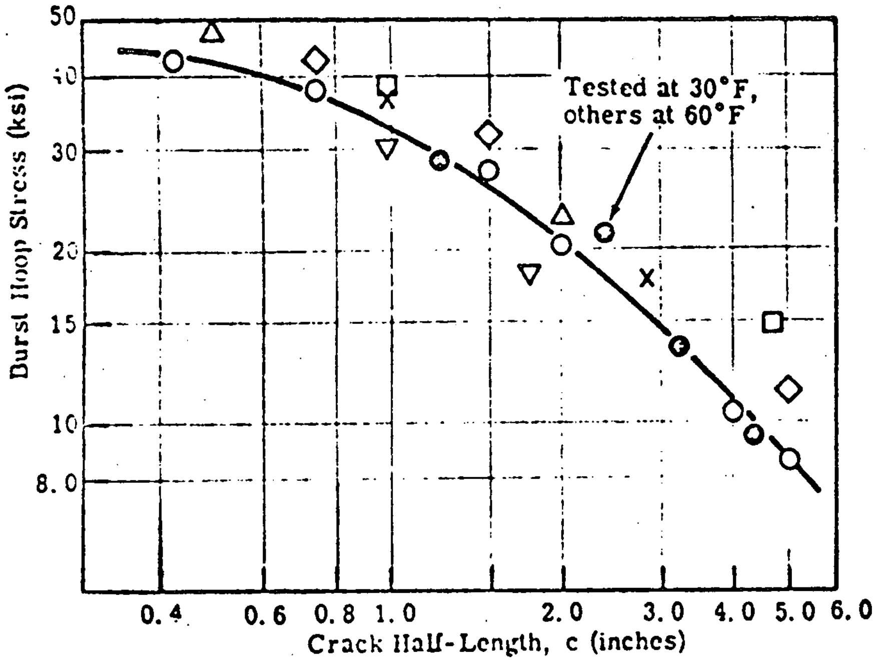
TABLE 1. Pipe Dimensions and Stress Intensity Factor/Stress Ratio for 2T Circumferential Through-wall Crack

Nominal Size, in.	Schedule	Wall Thickness	Mean Radius	t/R	K/σ
6	80	0.43	3.12	.138	1.18
6	160	0.72	2.95	.244	1.53
8	80	0.50	4.06	.123	1.27
8	160	0.91	3.86	.236	1.72
10	80	0.59	5.08	.116	1.38
10	160	1.125	4.81	.234	1.91
12	80	0.70	6.04	.116	1.51
12	160	1.31	5.72	.229	2.06
14	80	0.75	6.62	.113	1.56
14	160	1.41	6.30	.224	2.14
16	40	.50	7.75	.064	1.27
16	80	.84	7.58	.111	1.65
16	160	1.56	7.22	.216	2.25
18	40	0.56	8.72	.064	1.35
18	80	0.94	8.53	.110	1.74
18	160	1.75	8.12	.216	2.38
20	40	0.59	9.70	.061	1.38
20	80	1.03	9.48	.109	1.83
20	120	1.50	9.25	.162	2.21
20	160	1.94	9.03	.215	2.51
24	40	0.69	11.75	.059	1.50
24	80	1.22	11.39	.107	1.99
24	120	1.75	11.12	.157	2.38
24	160	2.31	10.84	.213	2.74

TABLE 2. Pipe Toughness Requirement at Lowest Full-Load Service Temperature*

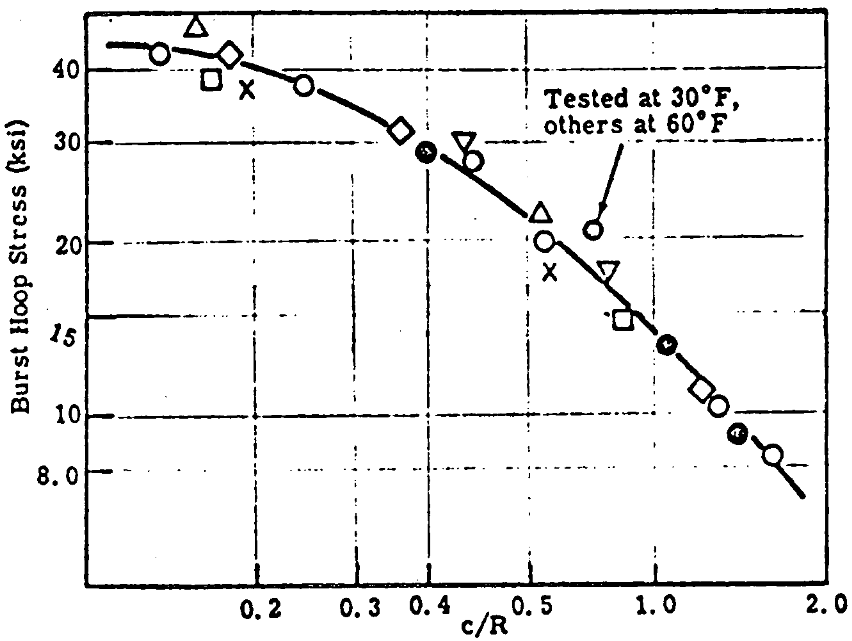
Wall Thickness Range, inches	Lateral Expansion mils
< ½	- no test required -
½ - 3/4	15
3/4 - 1	20
1 - 1½	30
1½ - 2½	40

* Longitudinal direction, notch in radial direction.



	Diameter	Schedule	R	t	R/t	Rt
▽	4	40	2.10	0.23	9.15	0.48
△	6	40	3.20	0.28	11.4	0.90
● ○	6	80	3.12	0.43	7.26	1.34
◇	8	40	4.17	0.32	12.8	1.35
X	10	40	5.22	0.36	14.5	1.88
□	12	80	6.04	0.71	8.51	4.28

Figure 1 Variation of Pressure Load Limit with Flaw Length in ASTM A106B Pipe with Axial Through-Wall Flaws



	Diameter	Schedule	R	t	R/t	Rt
▽	4	40	2.10	0.23	9.15	0.48
△	6	40	3.20	0.28	11.4	0.90
● ○	6	80	3.12	0.43	7.26	1.34
◇	8	40	4.17	0.32	12.8	1.35
X	10	40	5.22	0.36	14.5	1.88
□	12	80	6.04	0.71	8.51	4.28

Figure 2 Pressure Load Limit Versus c/R for ASTM A106B Pipe with Axial Through-Wall Flaws

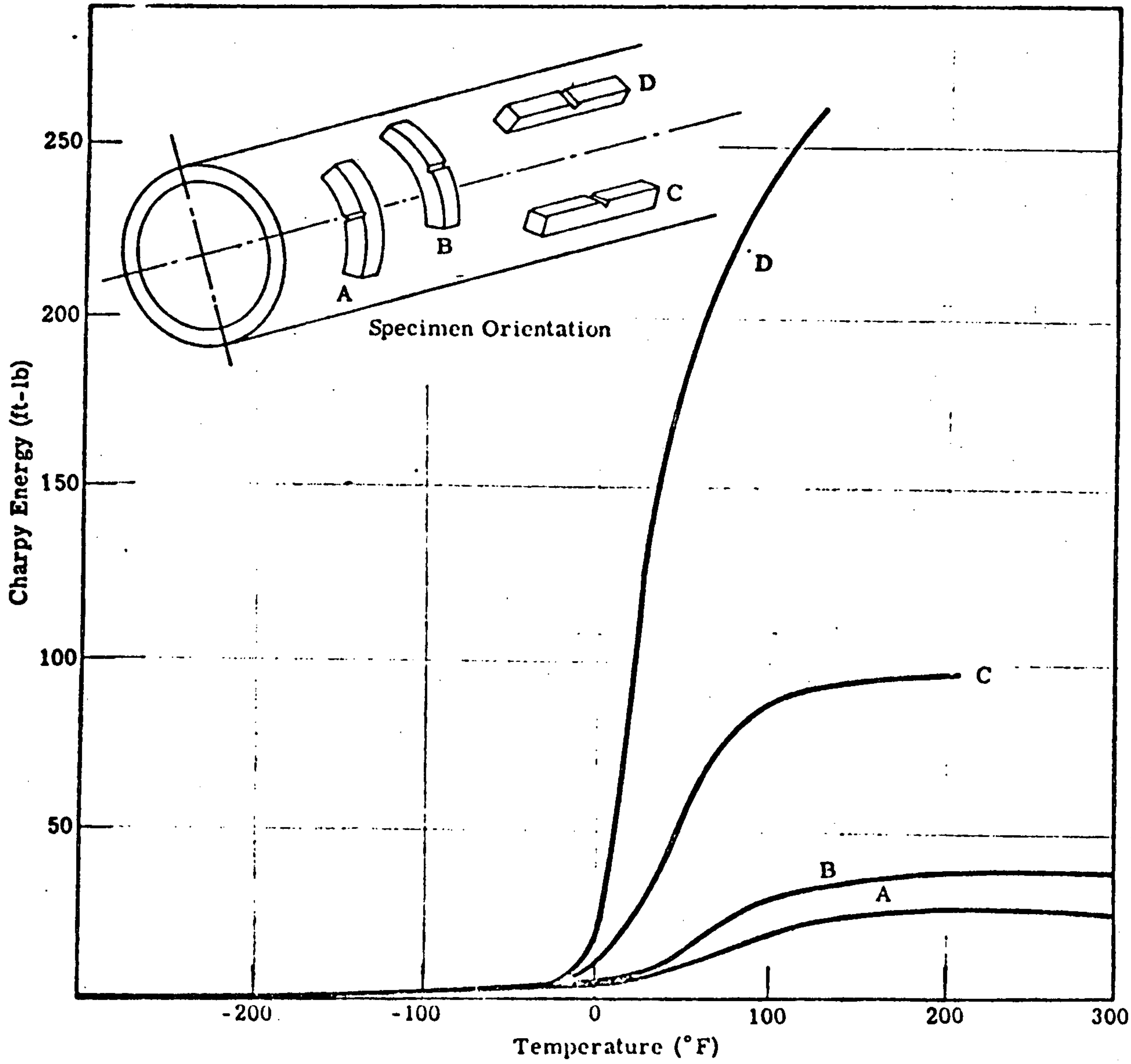


Figure 3. Toughness Anisotropy of ASTM A106B Pipe

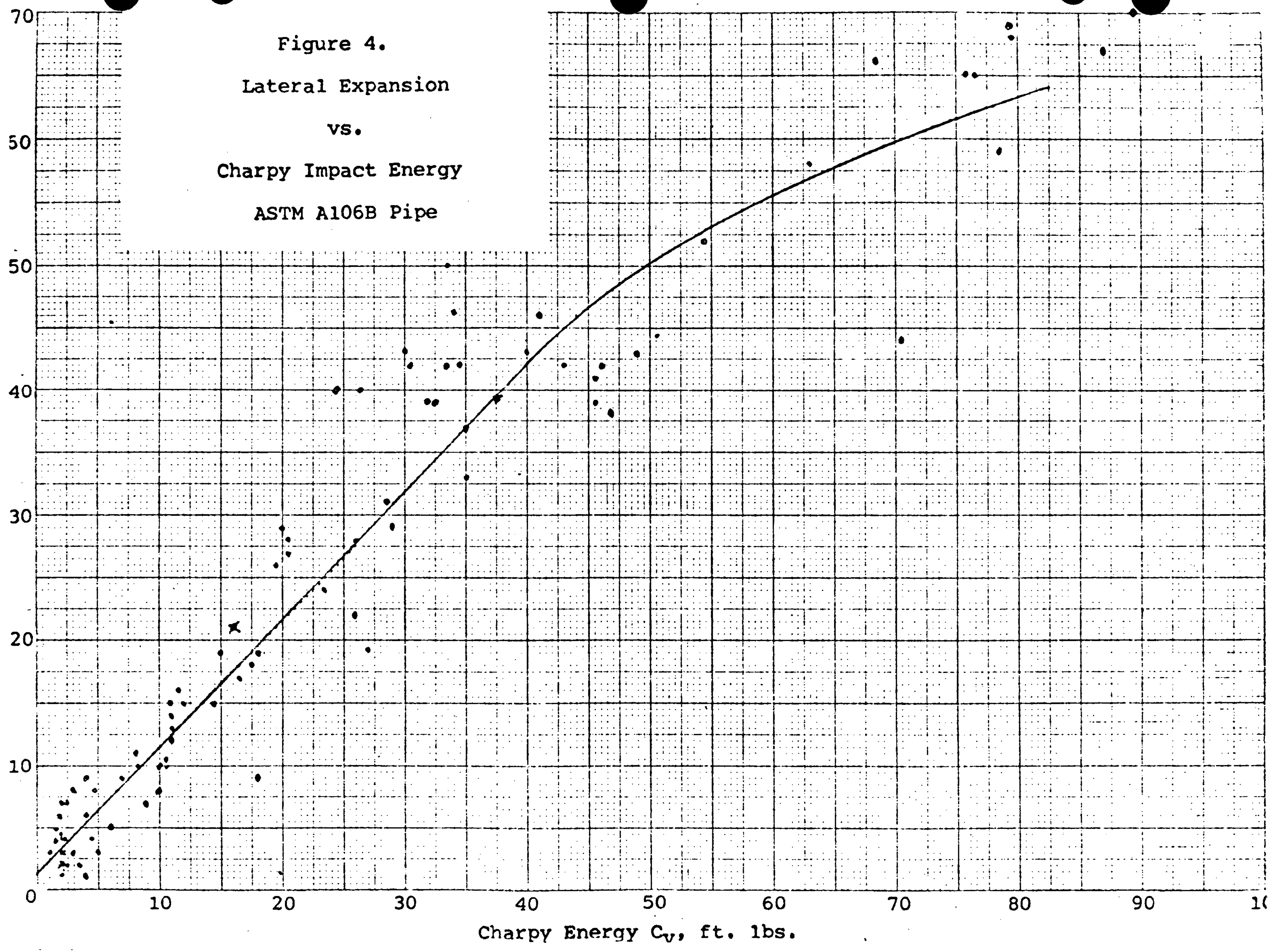


Figure 4.

Lateral Expansion

vs.

Charpy Impact Energy

ASTM A106B Pipe

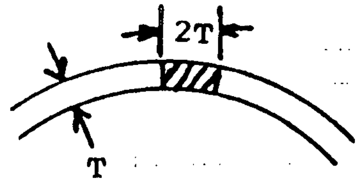
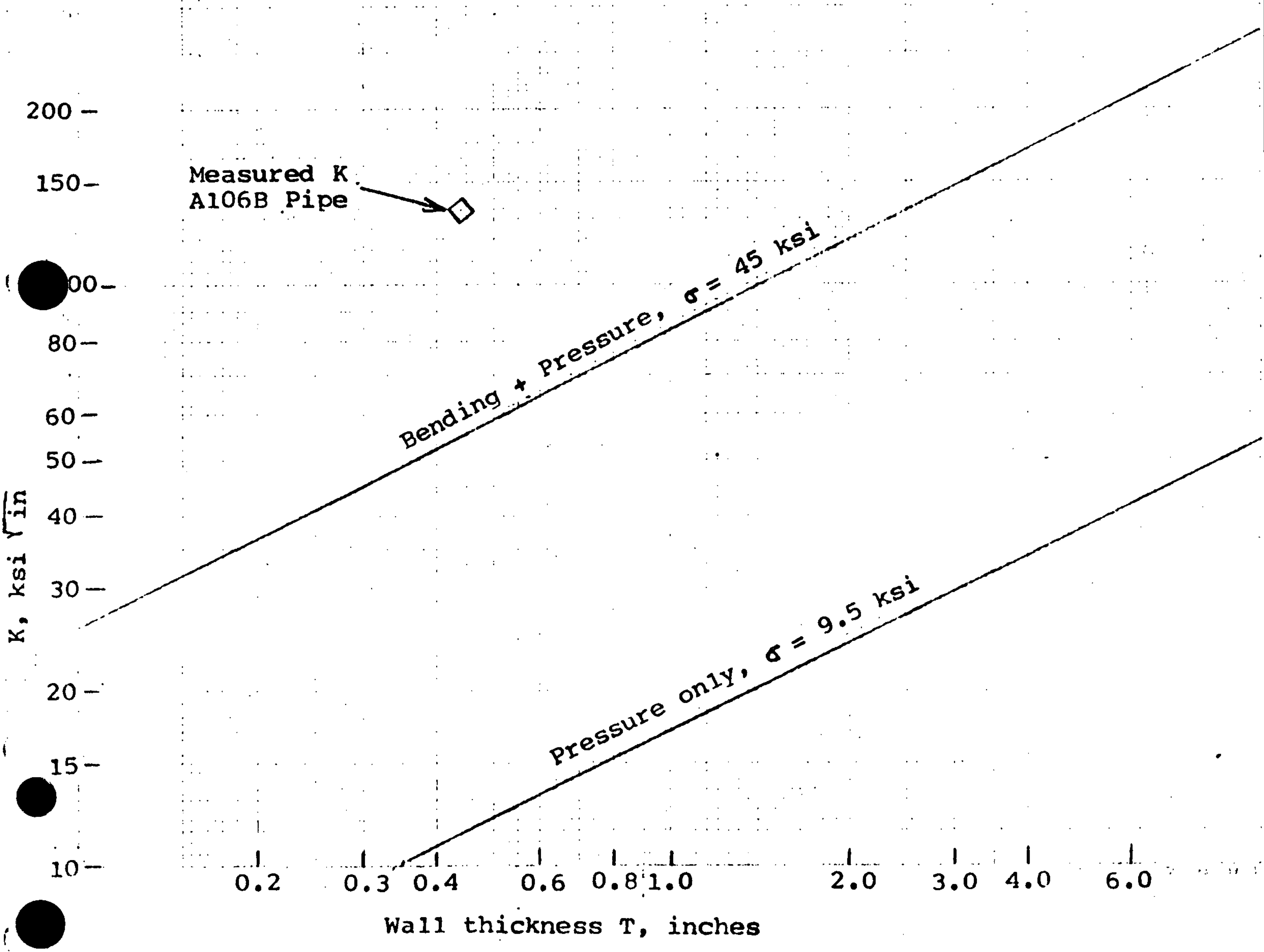


Figure 5. Required K to Support a 2T Circumferential Through-wall Crack in a Pipe under both Pressure and Bending



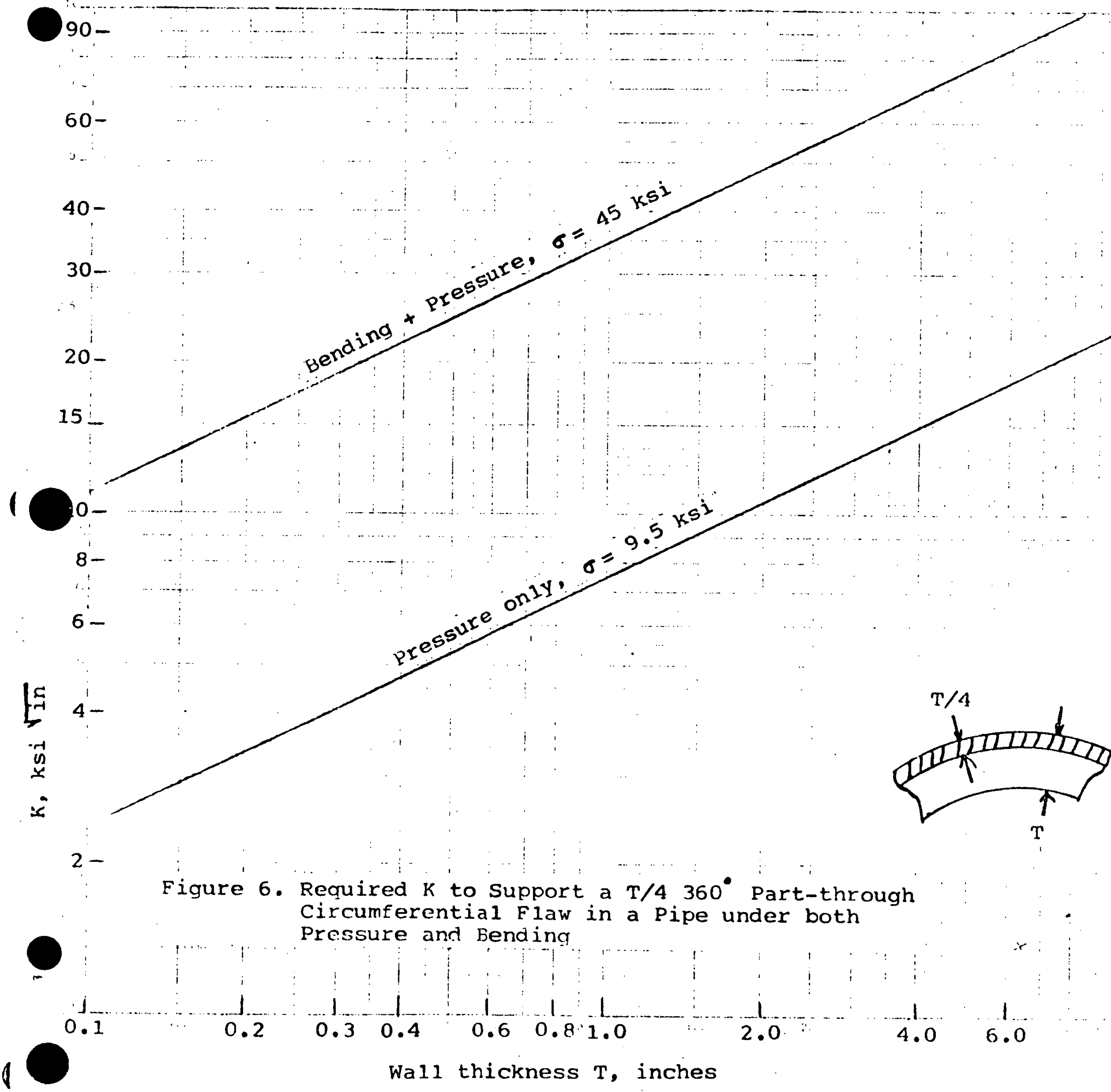
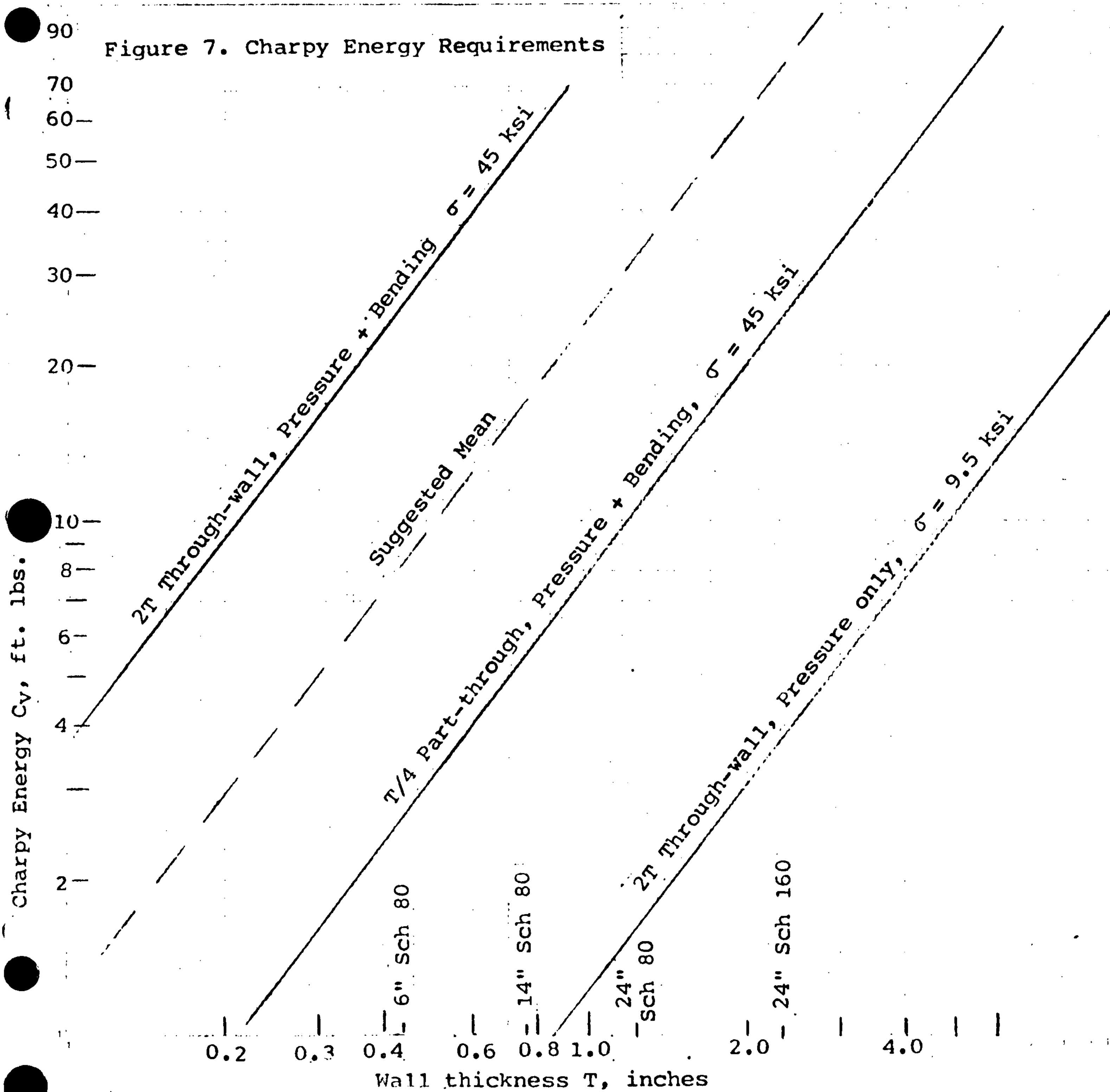


Figure 6. Required K to Support a T/4 360° Part-through Circumferential Flaw in a Pipe under both Pressure and Bending

Figure 7. Charpy Energy Requirements



APPENDIX VII

TOUGHNESS REQUIREMENTS FOR BOLTING

Introduction

Nearly all of the larger size, higher stressed low-alloy steel bolting that have been or are currently used in reactor coolant pressure boundary applications are covered by the following listing:

Specification	Minimum Specified YS (ksi)	Maximum Diameter in Current Use (in.)
SA 193, B7	105	1-7/8
SA 320, L43	105	4
SA 540, Cl. 5	110	6-1/2
SA 540, Cl. 3	130	6-1/2

The ASME Code (Section III) permits maximum service stresses in bolts equal to the combination of the following:

1. Tensile load stress averaged over the bolt minimum cross-section up to 2 S_m.
2. Bending stress at the periphery of the bolt minimum cross-section up to S_m.

Thus, the sum of tensile plus bending stresses can be up to 3 S_m under maximum allowable service conditions. For the bolting materials listed above, 3 S_m is equal to the specified minimum yield strength.

In assessing the toughness requirements for bolting materials, it is assumed that the most probable flaw location will be in the thread regions. Specifically, a fatigue crack initiating from the thread root is a possible flaw configuration.

This Appendix gives a fracture mechanics analysis for assumed flaws in bolts and toughness requirements derived from the analysis.

K_I Calculations

Two flaw geometries can be considered in deriving K_I versus flaw size relations:

1. Circumferential flaw extending symmetrically in from the periphery to a depth "a".

2. Asymmetrical circumferential flaw having a depth "a" at its deepest point from the periphery.

The K_I for the case of the uniform depth flaw can be calculated from equations given by Harris[1] for circumferentially-notched cylinders subjected to tensile and bending loads. He also considered the influence of a central axial hole. Some types of bolts for nuclear applications may have a central hole but the hole to bolt diameters in these cases are in the range that their influence on K_I values are minor.

The case of asymmetrical flaw can be approximately modeled by considering an axial slice parallel to the diametral plane of the bolt. The resulting geometry is approximately a single edge notch (SEN) geometry for which K_I formulas are available for both tension and bending.[2] In this case, the width of the SEN geometry is taken equal to the bolt diameter.

The calculation of K_I for these two flaw geometries can be summarized as follows:

	Value of $K_I/\sigma\sqrt{a}$			
	Notched Cylinder		SEN	
a/D	Tension	Bending	Tension	Bending
0	1.98	1.98	1.99	1.99
0.05	1.98	1.98	2.00	1.89
0.10	2.05	2.11	2.10	1.85
0.15	2.27	2.64	2.25	1.85
0.20	2.64	3.47	2.44	1.87
0.25	3.10	5.03	2.67	1.92

In this tabulation, "a" is the flaw depth and "D" is the gross diameter in the region of the flaw. The stress "σ" is the value based on the diameter "D". For the case of a flaw in the thread region of a bolt, the diameter is the major diameter of the thread and the stresses have to be calculated accordingly.

The preceding tabulations show that the value of $K_I/\sigma\sqrt{a}$ for bending increases rapidly at larger a/D values for the notched cylinder case, but it remains nearly constant for the SEN case. The reason is that the notched cylinder case assumes that the notched area on the compression side cannot sustain compressive stresses. It would be expected that a tight flaw such as a fatigue crack could sustain compressive stresses. Consequently, the SEN geometry is considered to be more appropriate for the bending stresses and is used in the K_I analysis for bolts. Furthermore, in the a/D range of interest, $K_I/\sigma\sqrt{a}$ = 2 will be used to simplify the calculations.

The notched cylinder geometry will be used for calculating K_I due to tensile stresses. In summary, the total K_I for a flaw in the thread region of a bolt is calculated as the sum of K_I from tension using the notched cylinder geometry plus K_I from bending using the SEN geometry.

For typical bolts, the shank is the minimum diameter section and usual dimensions are such that the stress based on the major diameter of the thread is about 90% of the stress in the shank area. Therefore, the calculation of K_I for maximum Code permitted stress conditions will be done on the basis of "σ" in the K_I equations as being 1.8 S_m in tension and 0.9 S_m in bending.

The resulting K_I versus flaw depth relation for maximum Code allowed stress conditions is shown in Fig. VII-1 for 3" and 6" diameter bolts at two levels of stresses. It should be noted that in making these K_I calculations, plasticity corrections were not included because of uncertain applicability to these geometries and also, the actual material yield strengths are usually 10-20% higher than specification minimum values.

It can be seen in Fig. VII-1 that for a fixed stress value, the K_I versus flaw depth relation is identical up to flaw depth of 0.3" in the 3-6" bolt diameter range. Although not shown, the K_I values for smaller diameters would follow the same curve up to a flaw depth equal to about 10% of the diameter and then depart towards higher K_I values.

In applying the curves in Fig. VII-1 to the case of flaws at thread roots, it should be noted that the flaw depth is the sum of the thread depth plus the actual depth of any flaw. Flange stud bolts generally use 8N threads which have a thread depth of 0.08" for all diameters.

Toughness Requirements

For purposes of defining toughness requirements, the reference flaw sizes are as follows:

1. For nominal diameters over 1" and up to and including 3", a reference flaw depth equal to 10% of the diameter.
2. For nominal diameters over 3", a reference flaw depth equal to 0.3".

For diameters over 3", Fig. VII-1 shows that the required material toughness would be about 100 ksi $\sqrt{in.}$ in the lower strength material and about 125 ksi $\sqrt{in.}$ in the higher strength material. These toughness values would be needed at the lowest service temperature at which the maximum Code allowed stresses occur. The toughness required would be proportionately less for smaller diameter bolts.

For simplification, the toughness requirements will be based on the higher strength (130 ksi minimum YS) bolt material. This will provide an extra conservative margin for the lower strength material.

The applicable toughness property for bolts should be the static fracture toughness value, K_{Ic}. Dynamic loading would not be expected to occur in bolting. Also, these higher strength steels generally exhibit very little loading rate influence on fracture toughness.

There are no K_{Ic} data available for these bolting materials. However, data are available for other varieties of high strength steels having yield strengths over 100,000 psi and K_{Ic} versus C_V energy correlations have been proposed based on these data.[3] These correlations suggest the following estimate of C_V energy to K_{Ic} relation for bolting steels:

C_V Energy (ft.-lbs.)	Lower Bound K_{Ic} (ksi $\sqrt{in.}$)
30	85
40	110
50	130

For reasons discussed in Appendix I, toughness requirements based on C_V lateral expansion provides a method of obtaining equivalent toughness at various yield strengths. Fig. VII-2 is a composite plot of C_V energy versus lateral expansion data for bolting steels of the 4340 composition. The data are from eight lots of material covering the following variations:

1. Diameters from 3 - 6-3/4".
2. Yield strengths between 140 and 150 ksi.
3. Air melted, vacuum degassed and vacuum arc remelted steels.

Fig. VII-2 shows that 30 and 35 mils lateral expansion correspond generally to at least 40 and 50 ft.-lbs. energy, respectively, Based on the preceding analysis and data, the toughness requirements derived for bolting materials are:

Nominal Dia., (in.)	Mils Lateral Exp.
1 or less	no test required
over 1 to 3	30
over 3	35

These requirements apply at the lowest metal temperature where the maximum Code allowable stresses occur. These requirements are believed to provide a conservative margin against fracture failure for flaw sizes up to the specified reference sizes.

Assessment of Recommended Criteria From Bolt Fatigue Test Data

Two publications[4,5] have presented results of fatigue tests conducted on bolts covering sizes and materials pertinent to this toughness criteria analysis. Although not included in the publications, the information necessary to analyze the final failure characteristics in these fatigue tests have been collected.

The tests of principal interest were those conducted on bolts in the 3-1/2 to 5" diameter range. The information available for these tests were all for cases where the material had greater than 50 ft.-lbs. C_V energy at the fatigue testing temperatures. In these cases, the fatigue cracks grew to depths considerably larger than the reference flaw depths described earlier. At final failure, the stress on the remaining net area was considerably above the material yield strengths. These fatigue test results are consistent with the recommended requirements and indicate a conservative adequacy of the requirements.

REFERENCES

1. D. O. Harris: "Stress Intensity Factors for Hollow Circumferentially Notched Round Bars," Trans. ASME, Jnl. Basic Eng., Vol. 89, 1967, pg. 49.

2. W. F. Brown, Jr. and J. E. Srawley: "Plane Strain Crack Toughness Testing of High Strength Metallic Materials," ASTM STP-410, 1967.

3. J. M. Barsom and S. T. Rolfe: "Correlations Between K_{IC} and Charpy V-Notch Test Results in the Transition Temperature Range," Impact Testing of Metals, ASTM STP-466, 1970, pg. 281.

4. A. L. Snow and B. F. Langer: "Low-Cycle Fatigue of Large Diameter Bolts," Trans. ASME, Jnl. Eng. for Industry, Vol. 89, 1967, pg. 53.

5. R. J. Fritz: "Cyclic Stress for Bolts and Studs," ASME Paper 67-MET-23, 1967.

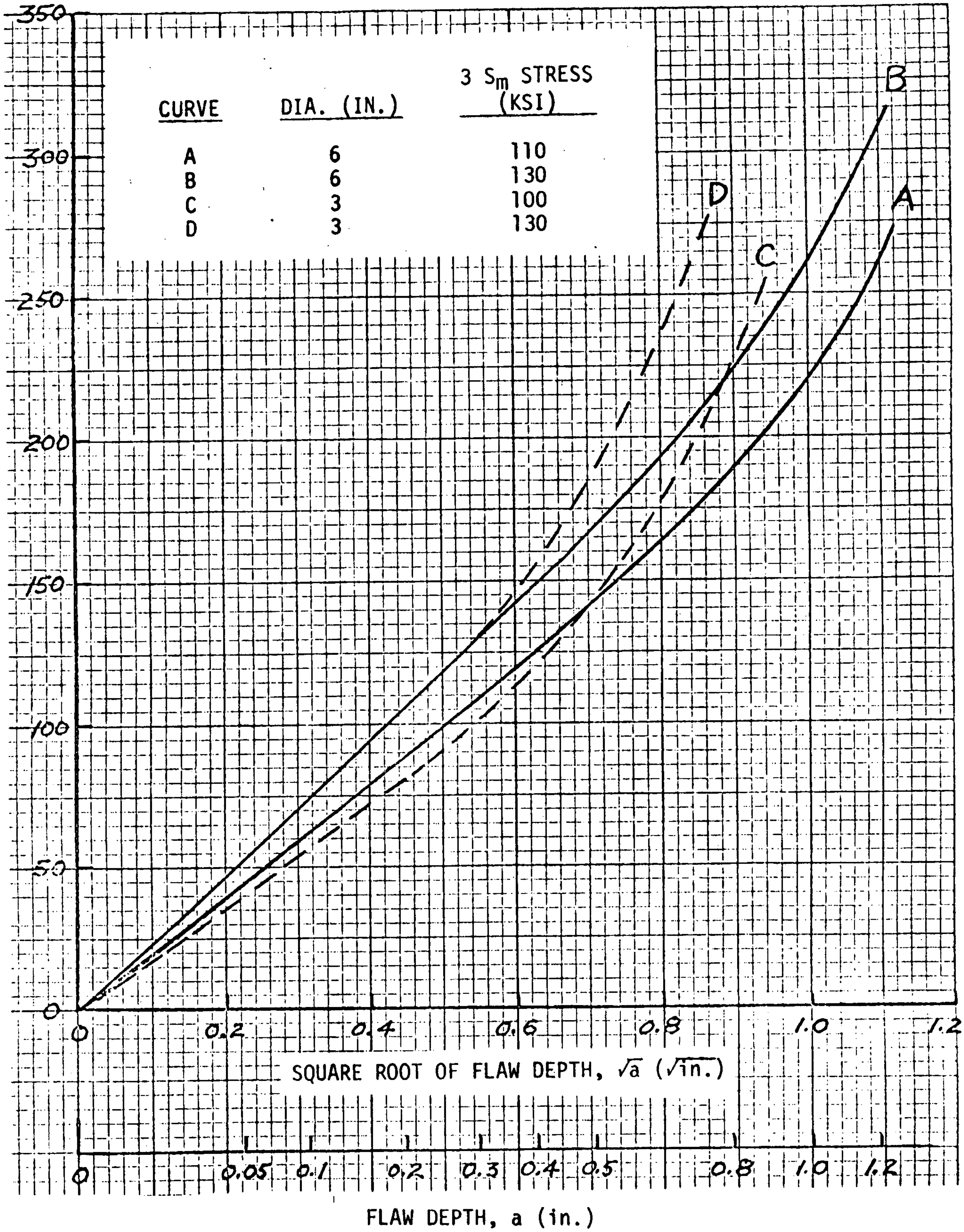


FIG. VII-1 STRESS INTENSITY FACTORS FOR CIRCUMFERENTIAL FLAWS IN BOLTS (DIA. = THREAD MAJOR DIA.; STRESS = STRESS ON MINIMUM CROSS-SECTION)

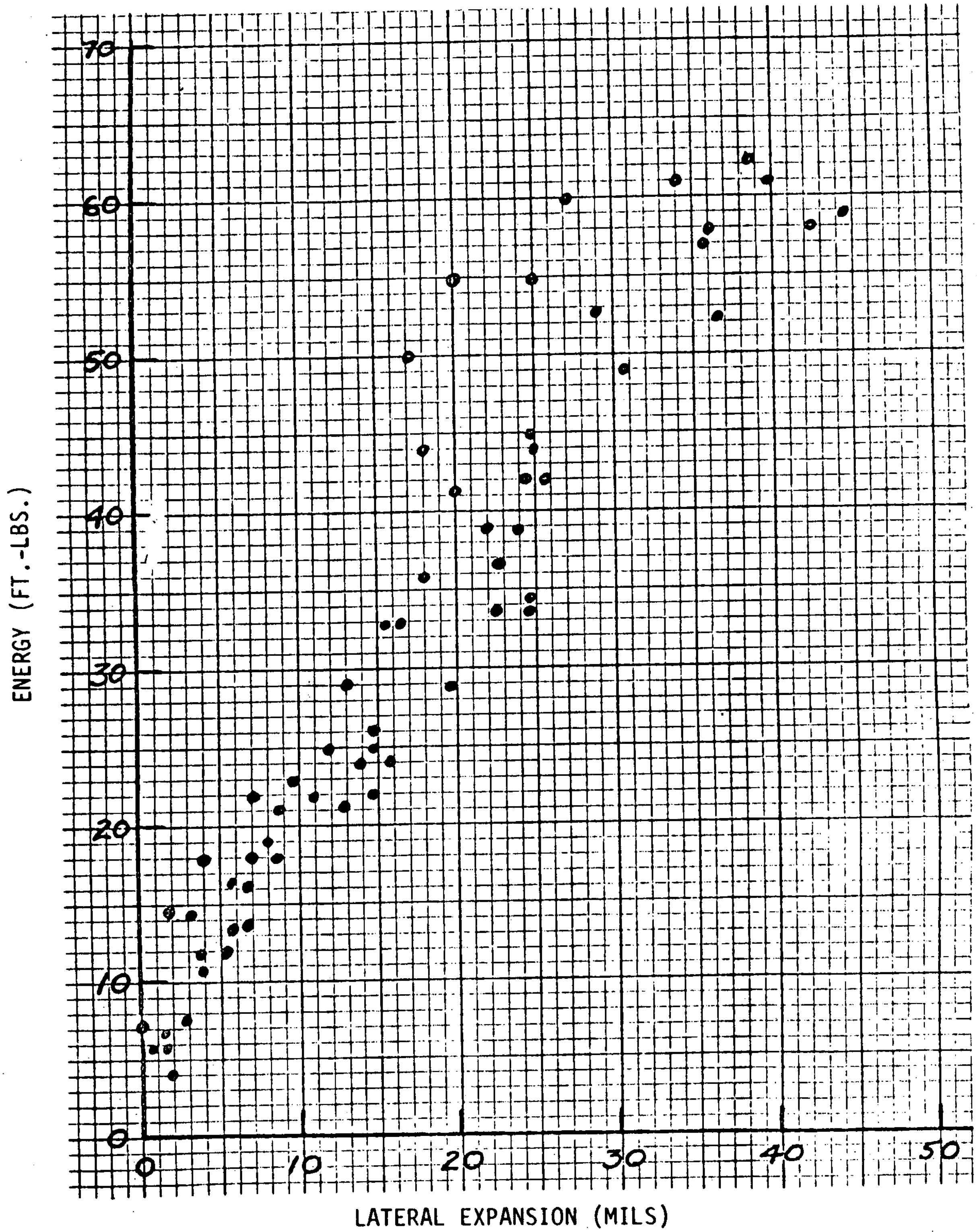


FIG. VII-2 ENERGY VERSUS LATERAL EXPANSION DATA FOR V-NOTCH CHARPY TESTS ON 4340 TYPE BOLT STEELS

APPENDIX VIII

Radiation Induced Changes - Adjustment Procedures

Neutron radiation increases the transition temperature and reduces the fracture toughness of ferritic steels. The degree of change varies widely depending upon the neutron fluence, the temperature at the point of irradiation, and the relative sensitivity to radiation-induced change of the particular steel. Because of these changes though, special provision must be made for the irradiated condition.

Materials for radiation effects surveillance purposes shall be the base material, weld metal, and heat-affected zone (HAZ) having the highest content of copper and of phosphorus. Where there is a divergence in the content of these two constituents (e.g., copper high, phosphorus low), the material highest in copper shall be selected.

For irradiated materials, use surveillance specimens of base metal, weld metal, and (HAZ) to determine the radiation induced transition temperature change, ΔT_{NDT} according to ASTM E185-70, "Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors,"*

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*ASTM E185-70 should be revised to reflect current knowledge, with particular attention to the effect of copper and phosphorus content, to the preferred use of lateral expansion as the index for toughness, and to the effect of radiation through the vessel wall.

Appendix D

Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2

CRITERIA OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR DESIGN BY ANALYSIS IN SECTIONS III AND VIII, DIVISION 2

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CRITERIA OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR DESIGN BY ANALYSIS IN SECTIONS III AND VIII, DIVISION 2

DESIGN

I. INTRODUCTION

The design philosophy of the present Section I (Power Boilers) and Division 1 of Section VIII (Pressure Vessels) of the ASME Boiler Code may be inferred from a footnote which appears in Division 1 of Section VIII on page 9 of the 1968 edition. This footnote refers to a sentence Par. UG-23 (c) which states, in effect, that the wall thickness of a vessel shall be such that the maximum hoop stress does not exceed the allowable stress. The footnote says:

> "It is recognized that high localized and secondary bending stresses may exist in vessels designed and fabricated in accordance with these rules. Insofar as practical, design rules for details have been written to hold such stresses at a safe level consistent with experience."

What this means is that Section I and Division 1 of Section VIII do not call for a detailed stress analysis but merely set the wall thickness necessary to keep the basic hoop stress below the tabulated allowable stress. They do not require a detailed evaluation of the higher, more localized stresses which are known to exist, but instead allow for these by the safety factor and a set of design rules. An example of such a rule is the minimum allowable knuckle radius for a torispherical head. Thermal stresses are given even less consideration. The only reference to them is Par. UG-22 where "the effect of temperature gradients" is listed among the loadings to be considered. There is no indication of how this consideration is to be given. In the othe hand, the Piping Code (USAS-B31.1) does give allowable values for the thermal stresses which are produced by the expansion of piping systems and even varies these allowable stresses with the number of cycles expected in the system.
lowable stresses with the number of cycles expected in the system.

The Special Committee to Review Code Stress Basis was originally established to investigate what changes in Code design philosophy might permit use of higher allowable stresses without reduction in safety. It soon became clear that one approach would be to make better use of modern methods of stress analysis. Detailed evaluation of actual stresses would permit substituting knowledge of localized stresses, and assignment of more rational margins, in place of a larger factor which really reflected lack of knowledge.

The ASME Special Committee dealt with these problems partly by the knowledge and experience of individual members and partly by the results of numerous analytical and experimental investigations. The Code Committee itself does not conduct research programs, but is able to derive much useful information from the Pressure Vessel Research Committee. PVRC is a private non-profit organization supported by subscription of interested fabricator and user groups and established to sponsor cooperative research programs aimed at improving the design, fabrication, and materials used in pressure vessels. Among other programs PVRC has sponsored considerable work on fatigue behavior in materials and vessels. Results of these experimental programs were studied by the ASME Special Committee and formed the basis for the design methods described in Section III and Appendix E of Division 2 of Section VIII for evaluation of fatigue behavior in vessels. The PVRC effort is now continuing in the even more difficult region of high temperature, in which the effects of cyclic loading are combined with the plastic deformation of creep.

The simplified procedures of Division 1 of Section VIII are for the most part conservative for pressure vessels in conventional service and a detailed analysis of many pressure vessels constructed to the rules of Division 1 of Section VIII would show where the design could be optimized to conserve metal. However, it is recognized that the designer may be required to provide additional design considerations for pressure vessels to be used in severe types of service such as vessels for highly cyclic types of operation, for services which require superior reliability, or for nuclear service where periodic inspection is usually difficult and sometimes impossible. The need for design rules for such vessels led to the preparation of Section III and Division 2 of Section VIII.

The development of analytical and experimental techniques has made it possible to determine stresses in considerable detail. When the stress picture is brought into focus, it is not reasonable to retain the same values of allowable stress for the clear detailed picture as had previously been used for the less detailed one. Neither is it sufficient merely to raise the allowable stresses to reasonable values for the peak stresses, since peak stress by itself is not an adequate criterion of safety. A calculated value of stress means little until it is associated with its location and distribution in the structure and with the type of loading which produced it. Different types of stress have different degrees of significance and must, therefore, be assigned different allowable values. For example, the average hoop stress through the thickness of the wall of a vessel due to internal pressure must be held to a lower value than the stress at the root of a notch in the wall. Likewise, a thermal stress can often be allowed to reach a higher value than one which is produced by dead weight or pressure. Therefore the Special Committee developed a new set of design criteria which shifted the emphasis away from the use of standard configurations and toward the detailed analyses of stresses. The setting of allowable stress values required dividing stresses into categories and assigning different allowable values to different groups of categories.

With its knowledge of the problems enhanced and its technical ability to solve them improved by its work on Section III, in 1963 the Special Committee returned to the objective inherent to its original assignment: the development of Alternative Rules for Pressure Vessels. More specifically, the objective was the development of rules which would be consistent with the higher stress levels of Section III but retain or enhance the degree of safety inherent in the prior rules and achieve balanced construction. The result of this effort was the publication of Division 2, Alternative Rules for Pressure Vessels, of Section VIII in 1968.

The design requirements of Division 2 consist of a text, comparable to the paragraphs on design in part UG of Division 1, and three appendices:

Appendix 4, Design Based on Stress Analysis

Appendix 5, Design Based on Fatigue Analysis

Appendix 6, Experimental Stress Analysis

These three appendices are essentially identical to the analysis requirements of Section III. They provide a means whereby one can evaluate those vessels subject to severe service stresses or which contain configurations not considered within the text, using the detailed engineering approach which modern methods of stress analysis have made possible.

For reasons discussed in Part V of this booklet, neither Section III nor Division 2 of Section VIII consider metal temperatures in the creep range, at this time.

Because of the prominent role played by stress analysis in designing vessels by the rules of Section III or by the appendices of Division 2, and because of the necessity to integrate the design and analysis efforts, the procedure may be termed *"design by analysis."* This document provides an explanation of the strength theories, stress categories, and stress limits on which these design procedures are presently based. It also provides an explanation of the methods used for determining the suitability of vessels and parts for cyclic application of loads. In these respects, this document replaces the "Criteria of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels" published by ASME in 1964.

Definitions

When discussing various combinations of stresses produced by various types of loading, it is important to use terms which are clearly defined. For example, the terms "membrane stress" and "secondary stress" are often used somewhat loosely. However, when a limit is to be placed on membrane stress, it is imperative that there must be no question about what is meant. Therefore the Special Committee spent a considerable amount of time in preparing a set of definitions. These definitions are given in Par. N-412 of Section III and Appendix 4, Par. 4-112 of Division 2.

Strength Theories

The stress state at any point in a structure may be completely defined by giving the magnitudes and directions of the three principal stresses. When two or three of these stresses are different from zero, the proximity to yielding must be determined by means of a strength theory. The theories most commonly used are the maximum stress theory, the maximum shear stress theory (also known as the Tresca criterion), and the distortion energy theory (also known as the octahedral shear theory and the Mises criterion). It has been known for many years that the maximum shear stress theory and the distortion energy theory are both much better than the maximum stress theory for predicting both yielding and fatigue failure in ductile metals. Section I and Division 1 of Section VIII use the maximum stress theory, by implication, but Section III and Division 2 use the maximum shear theory. Most experiments show that the distortion energy theory is even more accurate than the shear theory, but the shear theory was chosen because it is a little more conservative, it is easier to apply, and it offers some advantages in some applications of the fatigue analysis, as will be shown later.

The maximum shear stress at a point is defined as one-half of the algebraic difference between the largest and the smallest of the three principal stresses. Thus, if the principal stresses are σ_1, σ_2, and σ_3, and $\sigma_1 > \sigma_2 > \sigma_3$ (algebraically), the maximum shear stress is $\frac{1}{2}(\sigma_1 - \sigma_3)$. The maximum shear stress theory of failure states that yielding in a component occurs when the maximum shear stress reaches a value equal to the maximum shear stress at the yield point in a tensile test. In the tensile test, at yield, $\sigma_1 = S_y$, $\sigma_2 = 0$, and $\sigma_3 = 0$; therefore the maximum shear stress is $S_y/2$. Therefore yielding in the component occurs when

$$\frac{1}{2}(\sigma_1 - \sigma_3) = \frac{1}{2} S_y.$$

In order to avoid the unfamiliar and unnecessary operation of dividing both the calculated and the allowable stresses by two before comparing them, a new term called "equivalent intensity of combined stress" or, more briefly, "stress intensity" has been used. The stress intensity is defined as twice the maximum shear stress and is equal to the largest algebraic difference between any two of the three principal stresses. Thus the stress intensity is directly comparable to strength values found from tensile tests.

For the simple analyses on which the thickness formulas of Section I and Division 1 of Section VIII are based, it makes little difference whether the maximum stress theory or the maximum shear stress theory is used. For example, in the wall of a thin-walled cylindrical pressure vessel, remote from any discontinuities, the hoop stress is twice the axial stress and the radial stress on the inside is compressive and equal to the internal pressure, p. If the hoop stress is σ, the principal stresses are:

$$\sigma_1 = \sigma$$

$$\sigma_2 = \sigma/2$$

$$\sigma_3 = -p \ .$$

According to the maximum stress theory, the controlling stress is σ, since it is the largest of the three principal stresses. According to the maximum shear stress theory, the controlling stress is the stress intensity, which is $(\sigma + p)$. Since p is small in comparison with σ for a thin-walled vessel, there is little difference between the two theories. When a more detailed stress analysis is made, however, the difference between the two theories often becomes important.

II. STRESS CATEGORIES AND STRESS LIMITS

The various possible modes of failure which confront the pressure vessel designer are:

1. Excessive elastic deformation including elastic instability.
2. Excessive plastic deformation.
3. Brittle fracture.
4. Stress rupture/creep deformation (inelastic).
5. Plastic instability – incremental collapse.
6. High strain – low cycle fatigue.
7. Stress corrosion.
8. Corrosion fatigue.

In dealing with these various modes of failure, we will assume that the designer has at his disposal a picture of the state of stress within the part in question. This would be obtained either through calculation or measurements of both the mechanical and thermal stresses which could occur throughout the entire vessel during transient and steady state operations. The question one must ask is what do these numbers mean in relation to the adequacy of the design? Will they insure safe and satisfactory performance of a component? It is against these various failure modes that the pressure vessel designer must compare and interpret stress values. For example, elastic deformation and elastic instability (buckling) cannot be controlled by imposing upper limits to the calculated stress alone. One must consider, in addition, the geometry and stiffness of a component as well as properties of the material.

The plastic deformation mode of failure can, on the other hand, be controlled by imposing limits on calculated stress, but unlike the fatigue and stress corrosion modes of failure, peak stress does not tell the whole story. Careful consideration must be given to the consequences of yielding, and therefore the type of loading and the distribution of stress resulting therefrom must be carefully studied. The designer must consider, in addition to setting limits for allowable stress, some adequate and proper failure theory in order to define how the various stresses in a component react and contribute to the strength of that part.

As mentioned previously, different types of stress require different limits, and before establishing these limits it was necessary to choose the stress categories to which limits should be applied. The categories and sub-categories chosen were as follows:

A. Primary Stress.
   (1) General primary membrane stress.
   (2) Local primary membrane stress.
   (3) Primary bending stress.

B. Secondary Stress.

C. Peak Stress.

Definitions of these terms are given in Table N-414 of Section III and Appendix 4, Table 4-120.1 of Division 2, but some justification for the chosen categories is in order. The major stress categories are primary, secondary, and peak. Their chief characteristics may be described briefly as follows:

(a) Primary stress is a stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

(b) Secondary stress is a stress developed by the self-constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary, stress is that it is self-limiting. Local yielding and minor distortions can satisfy the discontinuity conditions or thermal expansions which cause the stress to occur.

(c) Peak stress is the highest stress in the region under consideration. The basic characteristic of a peak stress is that it causes no significant distortion and is objectionable mostly as a possible source of fatigue failure.

The need for dividing primary stress into membrane and bending components is that, as will be discussed later, limit design theory shows that the calculated value of a primary bending stress may be allowed to go higher than the calculated value of a primary membrane stress. The placing in the primary category of local membrane stress produced by mechanical loads, however, requires some explanation because this type of stress really has the basic characteristics of a secondary stress. It is self-limiting and when it exceeds yield, the external load will be resisted by other parts of the structure, but this shift may involve intolerable distortion and it was felt that it must be limited to a lower value than other secondary stresses, such as discontinuity bending stress and thermal stress.

Secondary stress could be divided into membrane and bending components, just as was done for primary stress, but after the removal of local membrane stress to the primary category, it appeared that all the remaining secondary stresses could be controlled by the same limit and this division was unnecessary.

Thermal stresses are never classed as primary stresses, but they appear in both of the other categories, secondary and peak. Thermal stresses which can produce distortion of the structure are placed in the secondary category and thermal stresses which result from almost complete suppression of the differential expansion, and thus cause no significant distortion, are classed as peak stresses.

A special exception to these general rules is the case of the stress due to a radial temperature gradient in a cylindrical shell. It is specifically stated in N-412 (m) (2) (6) of Section III, and in 4-112 (1) (2) (6) of Appendix 4 of Division 2, that this stress may be considered a local thermal stress. In reality, the linear portion of this gradient can cause deformation, but it was the opinion of the Special Committee that this exception could be safely made.

One of the commonest types of peak stress is that produced by a notch, which might be a small hole or a fillet. The phenomenon of stress concentration is well-known and requires no further explanation here.

Many cases arise in which it is not obvious which category a stress should be placed in, and considerable judgement is required. In order to standardize this procedure and use the judgement of the writers of the Code rather than the judgement of individual designers, a table was prepared covering most of the situations which arise in pressure vessel design and specifying which category each stress must be placed in. This table appears as Table N-413 of Section III and Appendix 4, Table 4-120.1 of Division 2.

The grouping of the stress categories for the purpose of applying limits to the stress intensities is illustrated in Fig. N-414 of Section III and Fig. 4-130.1 of Appendix 4 of Division 2. This diagram has been called the "hopper diagram" because it provides a hopper for each stress category. The calculated stresses are made to progress through the diagram in the direction of the arrows. Whenever a rectangular box appears, the sum of all the stress components which have entered the box are used to calculate the stress intensity, which is then compared to the allowable limit, shown in the circle adjacent to the rectangle. The following points should be noted in connection with this diagram:

(a) The symbols P_m, P_l, P_b, Q and F do not represent single quantities, but each represents a set of six quantities, three direct stress and three shear stress components. The addition of stresses from different categories must be performed at the component level, not after translating the stress components into a stress intensity. Similarly, the calculation of membrane stress intensity involves the averaging of stresses across a section, and this averaging must also be performed at the component level.

(b) The stresses in Category Q are those parts of the total stress which are categorized as secondary, and do not include primary stresses which may also exist at the same point. It should be noted, however, that a detailed stress analysis frequently gives the combination of primary and secondary stresses directly, and this calculated value represents the total of P (or P_L) + P_b + Q and not Q alone. It is not necessary to calculate Q separately since the stress limit (to be described later) applies to the total stress intensity. Similarly, if the stress in Category F is produced by a stress concentration, the quantity F is the additional stress produced by the notch, over and above the nominal stress, but it is not necessary to calculate F separately.

The potential failure modes and various stress categories are related to the Code provisions as follows:

(a) The primary stress limits are intended to prevent plastic deformation and to provide a nominal factor of safety on the ductile burst pressure.

(b) The primary plus secondary stress limits are intended to prevent excessive plastic deformation leading to incremental collapse, and to validate the application of elastic analysis when performing the fatigue evaluation.

(c) The peak stress limit is intended to prevent fatigue failure as a result of cyclic loadings.

(d) Special stress limits are provided for elastic and inelastic instability.

Protection against brittle fracture is provided by material selection, rather than by analysis. Protection against environmental conditions such as corrosion and radiation effects are the responsibility of the designer. The creep and stress rupture temperature range will be considered in later editions.

Basic Stress Intensity Limits

The choice of the basic stress intensity limits for the stress categories described above was accomplished by the application of limit design theory tempered by some engineering judgement and some conservative simplifications. The principles of limit design which were used can be described briefly as follows.

The assumption is made of perfect plasticity with no strain-hardening. This means that an idealized stress-strain curve of the type shown in Fig. 1 is assumed. Allowable stresses based on perfect plasticity and limit design theory may be considered as a floor below which a vessel made of any sufficiently ductile material will be safe. The actual strain-hardening properties of specific materials will give them larger or smaller margins above this floor.

In a structure as simple as a straight bar in tension, a load producing yield stress, S_y, results in "collapse." If the bar is loaded in bending, collapse does not occur until the load has been increased by a factor known as the "shape factor" of the cross section; at that time a "plastic hinge" is formed. The shape factor for a rectangular section in bending is 1.5. When the primary stress in a rectangular section consists of a combination of bending and axial tension, the value of the limit load depends on the ratio between the tensile and bending loads. Fig. 2 shows the value of the maximum calculated stress at the outer fiber of a rectangular section which would be required to produce a plastic hinge, plotted against the average tensile stress across the section, both values expressed as multiples of the yield stress, S_y. When the average tensile stress, P_m, is zero, the failure stress for bending is 1.5 S_y. When the average tensile stress is S_y, no additional bending stress, P_b, may be applied.

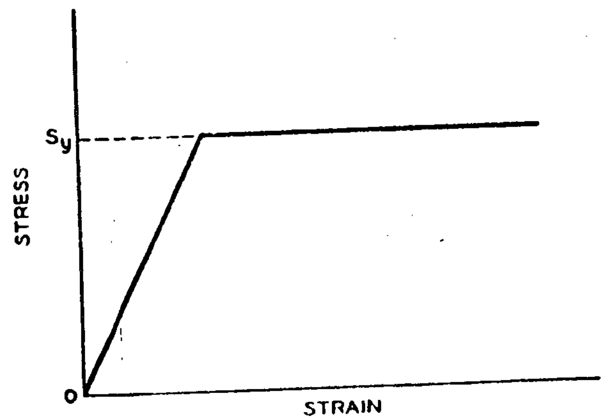


FIGURE 1. IDEALIZED STRESS – STRAIN RELATIONSHIP

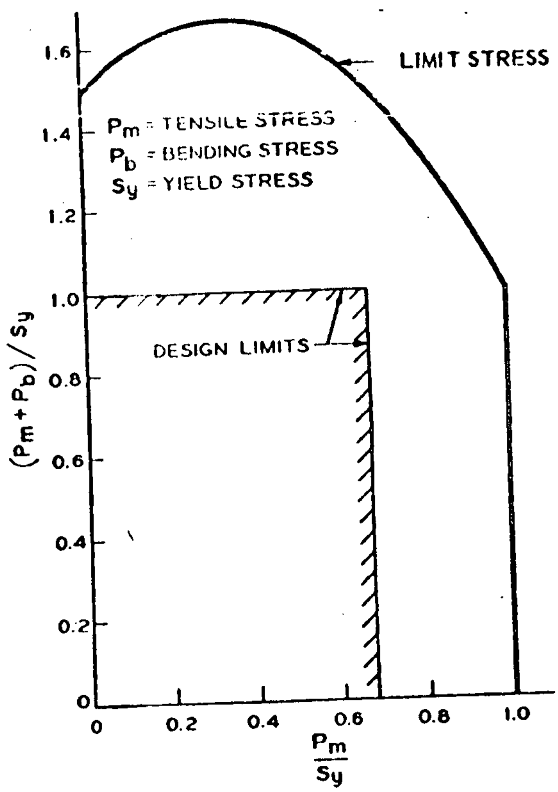
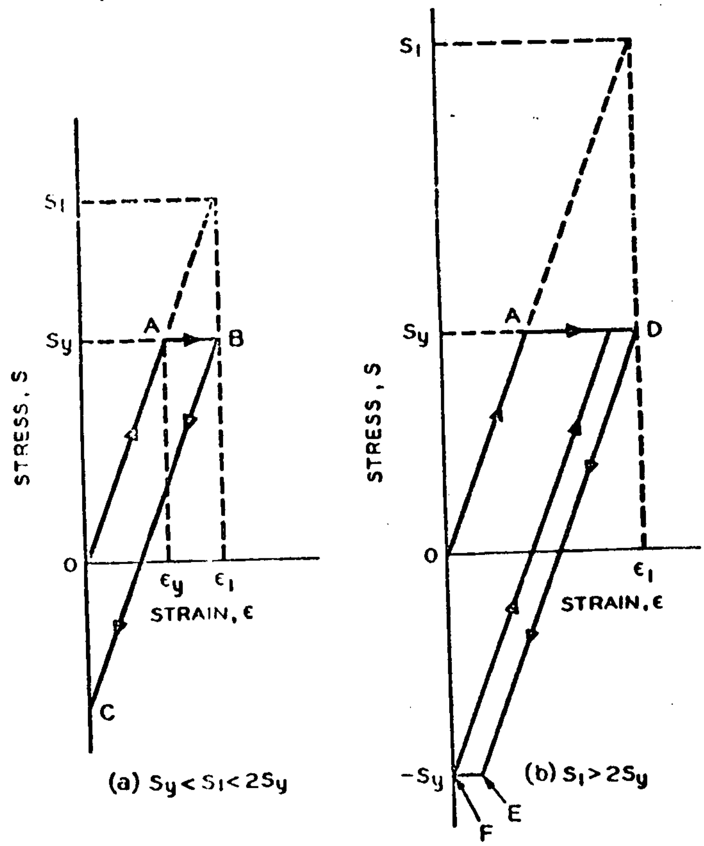


FIGURE 2. LIMIT STRESS FOR COMBINED TENSION AND BENDING (RECTANGULAR SECTION)

Figure 2 was used to choose allowable values, in terms of the yield stress, for general primary membrane stress, P_m, and primary membrane-plus-bending stress, $P_m + P_b$. It may be seen that limiting P_m to (2/3) S_y and $P_m + P_b$ to S_y provides adequate safety. The safety factor is not constant for all combinations of tension and bending, but a design rule to provide a uniform safety factor would be needlessly complicated.

In the study of allowable secondary stresses, a calculated elastic stress range equal to twice the yield stress has a very special significance. It determines the borderline between loads which, when repetitively applied, allow the structure to "shake down" to elastic action and loads which produce plastic action each time they are applied. The theory of limit design provides rigorous proof of this statement, but the validity of the concept can easily be visualized. Consider, for example, the outer fiber of a beam which is strained in tension to a strain value ϵ_1, somewhat beyond the yield strain as shown in Fig. 3 (a) by the path OAB. The calculated elastic stress would be $S = S_1 = E\epsilon_1$. Since we are considering the case of a secondary stress, we shall assume that the nature of the loading is such as to cycle the strain from zero to ϵ_1 and back to zero, rather than cycling the stress from zero to S_1, and back to zero. When the beam is returned to its undeflected position, O, the outer fiber has a residual compressive stress of magnitude $S_1 - S_y$. On any subsequent loading, this residual compression must be removed before the stress goes into tension and thus the elastic range has been increased by the quantity $S_1 - S_y$. If $S_1 = 2S_y$, the elastic range becomes $2S_y$, but if $S_1 > 2S_y$, the fiber yields in compression, as shown by EF in Fig. 3 (b) and all subsequent cycles produce plastic strain. Therefore, $2S_y$ is the maximum value of calculated secondary elastic stress which will "shake down" to purely elastic action.



STRAIN HISTORY BEYOND YIELD

FIGURE 3.

An important point to note from the foregoing discussion of primary and secondary stresses is that 1.5 S_y is the *failure* stress for primary bending, whereas for secondary bending $2S_y$ is merely the threshold beyond which some plastic action occurs. Therefore the allowable design stress for primary bending must be reduced below $1.5S_y$ to, say, $1.0S_y$, whereas $2S_y$ is a safe design value for secondary bending since a little plastic action during overloads is tolerable. The same type of analysis shows that $2S_y$ is also a safe design value for secondary membrane tension. As described previously, local membrane stress produced by mechanical load has the characteristics of a secondary stress but has been arbitrarily placed in the primary category. In order to avoid excessive distortion, it has been assigned an allowable stress level of S_y, which is 50 per cent higher than the allowable for general primary membrane stress but precludes excessive yielding.

We have now shown how the allowable stresses for the first four stress categories listed in the previous section should be related to the yield strength of the material. The last category, peak stress, is related only to fatigue, and will be discussed later. With the exception of some of the special stress limits, the allowables in Codes are not expressed in terms of the yield strength, but rather as multiples of the tabulated value S_m, which is the allowable for general primary membrane stress. In assigning allowable stress values to a variety of materials with widely varying ductilities and widely varying strain-hardening properties, the yield strength alone is not a sufficient criterion. In order to prevent unsafe designs in materials with low ductility and in materials with high yield-to-tensile ratios, the Code has always considered both the yield strength and the ultimate tensile strength in assigning allowable stresses. This principle has not been changed in Section III or Division 2 but the chosen fractions of the mechanical properties have been increased to two-thirds yield strength and one-third ultimate strength instead of five-eighths yield strength (for ferrous materials) and one-fourth ultimate strength. The Special Committee believed that this increase was quite safe because the detailed stress analysis required eliminates the need for a large safety factor to cover unanalyzed areas. The stress intensity limits for the various categories given are such that the multiples of yield strength described above are never exceeded.

The allowable stress intensity for austenitic steels and some non-ferrous materials, at temperatures above 100 F, may exceed $(2/3)S_y$ and may reach $0.9S_y$ at temperature. Some explanation of the use of up to $0.9S_y$ for these materials as a basis for S_m is needed in view of Figure 2 because this figure would imply that loads in excess of the limit load are permitted. The explanation lies in the different nature of these materials' stress strain diagram. These materials have no well-defined yield point but have strong strain-hardening capabilities so that their yield strength is effectively raised as they are highly loaded. This means that some permanent deformation during the first loading cycle may occur, however the basic structural integrity is comparable to that obtained with ferritic materials. This is equivalent to choosing a somewhat different definition of the "design yield strength" for those materials which have no sharply defined yield point and which have strong strain-hardening characteristics. Therefore, the S_m value in the code tables, regardless of material, can be thought of as being no less than 2/3 of the "design yield strength" for the material in evaluating the primary and secondary stresses.

Table I summarizes the basic stress limits and shows the multiples of yield strength and ultimate strength which these limits do not exceed.

TABLE I. BASIC STRESS INTENSITY LIMITS

Stress Intensity	Tabulated Value	Yield Strength	Ultimate Tensile Strength
General primary membrane (P_m)	S_m	$\leqslant \frac{2}{3} S_y$	$\leqslant \frac{1}{3} S_u$
Local primary membrane (P_l)	$1.5\ S_m$	$\leqslant S_y$	$\leqslant \frac{1}{2} S_u$
Primary membrane plus bending ($(P_l + P_b)$)	$1.5\ S_m$	$\leqslant S_y$	$\leqslant \frac{1}{2} S_u$
Primary plus secondary ($P_l + P_b + Q$)	$3\ S_m$	$\leqslant 2S_y$	$\leqslant S_u$

Stresses Above Yield Strength

The primary criterion of the structural adequacy of a design, is that the stresses, as determined by calculation or experimental stress analysis, shall not exceed the specified allowable limits. It frequently happens that both the calculated stress and allowable stress exceeds the yield strength of the material. Nevertheless, unless stated specifically otherwise, it is expected that calculations be made on the assumption of elastic behavior.

Allowable stresses higher than yield appear in the values for primary-plus-secondary stress and in the fatigue curves. In the case of the former, the justification for allowing calculated stresses higher than yield is that the limits are such as to assure shake-down to elastic action after repeated loading has established a favorable pattern of residual stresses. Therefore the assumption of elastic behavior is justified because it really exists in all load cycles subsequent to shake-down.

In the case of fatigue analysis, plastic action can actually persist throughout the life of the vessel, and the justification for the specified procedure is somewhat different. Repetitive plastic action occurs only as the result of peak stresses in relatively localized regions and these regions are intimately connected to larger regions of the vessel which behave elastically. A typical example is the peak stress at the root of a notch, in a fillet, or at the edge of a small hole. The material in these small regions is strain-cycled rather than stress-cycled (as will be discussed later) and the elastic calculations give numbers which have the dimensions of stress but are really proportional to the strain. The factor of proportionality for uniaxial stress is, of course, the modulus of elasticity. The fatigue design curves have been specially designed to give numbers comparable to these fictitious calculated stresses. The curves are based on strain-cycling data and the strain values have been multiplied by the modulus of elasticity. Therefore stress intensities calculated from the familiar formulas of strength-of-materials texts are directly comparable to the allowable stress values in the fatigue curves.

III. FATIGUE ANALYSIS

One of the important innovations in Section III and Division 2 as compared to Sections I and Division 1 of Section VIII, is the recognition of fatigue as a possible mode of failure and the provision of specific rules for its prevention. Fatigue has been a major consideration for many years in the design of rotating machinery and aircraft, where the expected number of cycles is in the millions and can usually be considered infinite for all practical purposes. For the case of large numbers of cycles, the primary concern is the endurance limit, which is the stress which can be applied an infinite number of times without producing failure. In pressure vessels, however, the number of stress cycles applied during the specified life seldom exceeds 10^5 and is frequently only a few thousand. Therefore, in order to make fatigue analysis practical for pressure vessels, it was necessary to develop some new concepts not previously used in machine design [1,2].

Use of Strain-Controlled Fatigue Data

The chief difference between high-cycle fatigue and low-cycle fatigue is the fact that the former involves little or no plastic action, whereas failure in a few thousand cycles can be produced only by strains in excess of the yield strain. In the plastic region large changes in strain can be produced by small changes in stress. Fatigue damage in the plastic region has been found to be a function of plastic strain and therefore fatigue curves for use in this region should be based on tests in which strain rather than stress is the controlled variable. As a matter of convenience, the strain values used in the tests are multiplied by the elastic modulus to give a fictitious stress which is not the actual stress applied but has the advantage of being directly comparable to stresses calculated on the assumption of elastic behavior.

The general procedure used in evaluating the strain-controlled fatigue data was to obtain a "best fit" for the quantities A and B in the equation

$$S = \frac{E}{4\sqrt{N}} \ln \frac{100}{100 - A} + B \quad (1)$$

where E = elastic modulus (psi)
N = number of cycles to failure
S = strain amplitude times elastic modulus

It is possible to estimate the fatigue properties by taking A as the percentage reduction of area in a tensile test, RA, and B as the endurance limit, S_e.

The use of strain instead of stress and the consideration of plastic action have necessitated some additional departures from the conventional methods of studying fatigue problems. It has been common practice in the past to use lower stress concentration factors for small numbers of cycles than for large numbers of cycles. This is reasonable when the allowable stresses are based on stress-fatigue data, but is not advisable when strain-fatigue data are used. Fig. 4 shows typical relationships between stress, S, and cycles-to-failure, N, from (A) strain cycling tests on unnotched specimens, (B) stress-cycling tests on unnotched specimens, and (C) stress-cycling tests on notched specimens. The ratio between the ordinates of curves (B) and (C) decreases with decreasing cycles-to-failure, and this is the basis for the commonly-accepted practice of using lower values

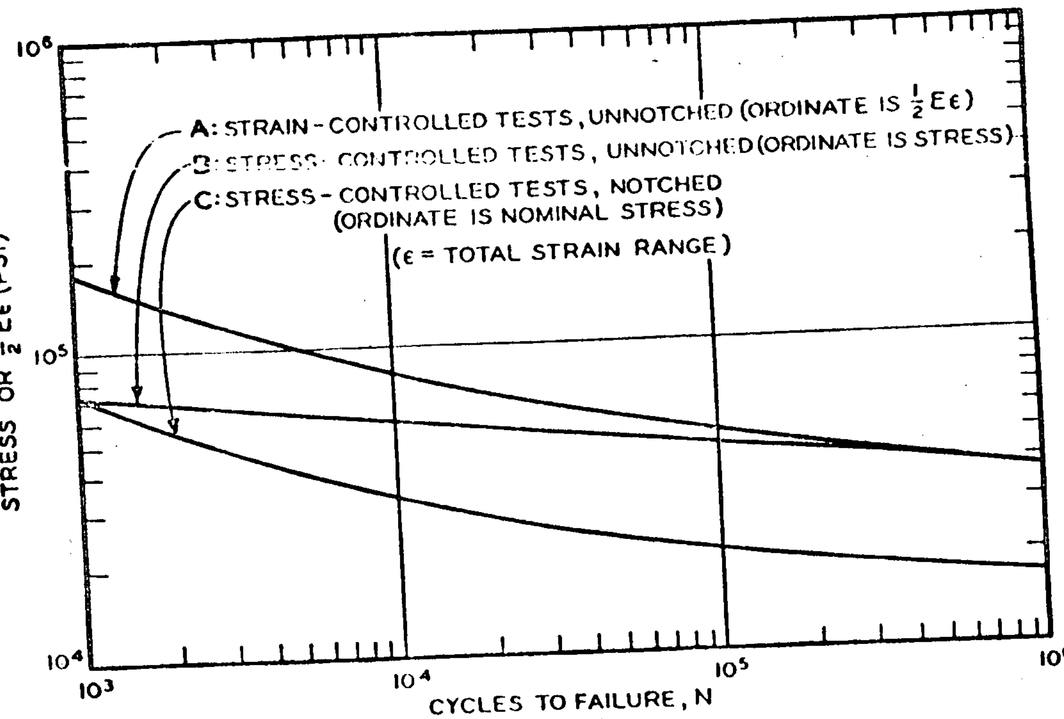


FIG. 4. TYPICAL RELATIONSHIP BETWEEN STRESS, STRAIN, AND CYCLES-TO-FAILURE.

of K (stress concentration factor) for lower values of N. In (C), however, although nominal stress is the controlled parameter, the material in the root of the notch is really being strain cycled, because the surrounding material is at a lower stress and behaves elastically. Therefore it should be expected that the ratio between curves (A) and (C) should be independent of N and equal to K. For this reason it is recommended in Section III and Division 2 of Section VIII that the same value of K be used regardless of the number of cycles involved.

The choice of an appropriate stress concentration factor is not an easy one to make. For fillets, grooves, holes, etc. of known geometry, it is safe to use the theoretical stress concentration factors found in such references as [3] and [4], even though strain concentrations can sometimes exceed the theoretical stress concentration factors. The use of the theoretical factor as a safe upper limit is justified, however, since strain concentrations significantly higher than the stress concentrations only occur when gross yielding is present in the surrounding material, and this situation is prevented by the use of basic stress limits which assure shake-down to elastic action. For very sharp notches it is well known that the theoretical factors grossly overestimate the true weakening effect of the notch in the low and medium strength materials used for pressure vessels. Therefore no factor higher than 5 need ever be used for any configuration allowed by the design rules and an upper limit of 4 is specified for some specific constructions such as fillet welds and screw threads. When fatigue tests are made to find the appropriate factor for a given material and configuration, they should be made with a material of comparable notch sensitivity and failure should occur in a reasonably large number of cycles (> 1000) so that the test does not involve gross yielding.

Effect of Mean Stress

Another deviation from common practice occurs in the consideration of fluctuating stress, which is a situation where the stress fluctuates around a mean value different from zero, as shown in Fig. 5. The evaluation of the effects of mean stress is commonly accomplished by use of the modified Goodman diagram, as shown in Fig. 6, where mean

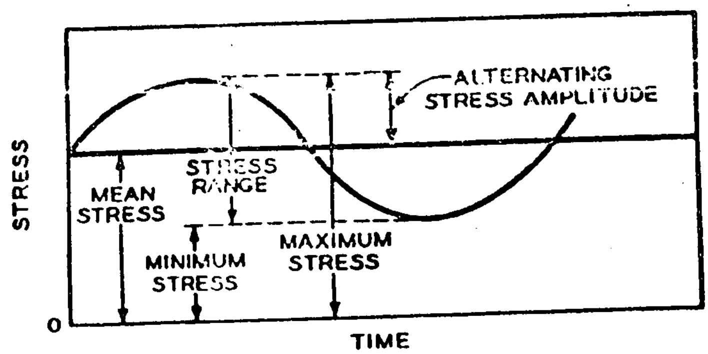


FIG. 5. STRESS FLUCTUATION AROUND A MEAN VALUE.

stress is plotted as the abscissa and the amplitude (half range) of the fluctuation is plotted as the ordinate. The straight line joining the endurance limit, S_e, (where $S_N = S_e$) on the vertical axis (point E) with the ultimate strength, S_u, on the horizontal axis (point D) is a conservative approximation of the combinations of mean and alternating stress which produce failure in large numbers of cycles. A little consideration of this diagram shows that not all points below the "failure" line, ED, are feasible. Any combination of mean and alternating stresses which results in a stress excursion above the yield strength will produce a shift in the mean stress which keeps the maximum stress during the cycle at the yield value. This shift has already been illustrated by the strain history shown in Fig. 3. The feasible combinations of mean and alternating stress are all contained within the 45 degree triangle AOB or on the vertical axis above A, where A is the yield strength on the vertical axis and B is the yield strength on the horizontal axis. Regardless of the conditions under which any test or service cycle is started, the true conditions after the application of a few cycles must fall within this region because all combinations above AB have a maximum stress above yield and there is a consequent reduction of mean stress which shifts the conditions to a point on the line AB or all the way to the vertical axis.

It may be seen from the foregoing discussion that the value of mean stress to be used in the fatigue evaluation is not always the value which is calculated directly from the imposed loading cycle. When the loading cycle produces calculated stresses which exceed

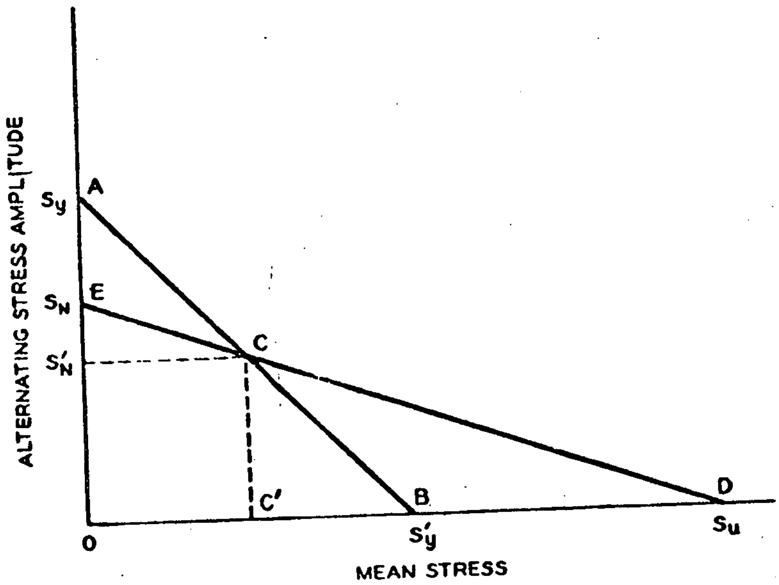


FIG. 6. MODIFIED GOODMAN DIAGRAM.

the yield strength at any time, it is necessary to calculate an adjusted value of mean stress before completing the fatigue evaluation. The rules for calculating this adjusted value when the modified Goodman diagram is applied may be summarized as follows:

Let S'_{mean} = basic value of mean stress (calculated directly from loading cycle)
S_{mean} = adjusted value of mean stress
S_{alt} = amplitude (half range) of stress fluctuation
S_y = yield strength

$$\left.\begin{array}{l} \text{If } S_{alt} + S'_{mean} \lesseqgtr S_y,\ S_{mean} = S'_{mean} \\ \text{If } S_{alt} + S'_{mean} > S_y \text{ and } S_{alt} < S_y,\ S_{mean} = S_y - S_{alt} \\ \text{If } S_{alt} \geqslant S_y,\ S_{mean} = 0. \end{array}\right\} \quad (2)$$

The fatigue curves are based on tests involving complete stress reversal, that is, $S_{mean} = 0$. Since the presence of a mean stress component detracts from the fatigue resistance of the material, it is necessary to determine the equivalent alternating stress component for zero mean stress before entering the fatigue curve. This quantity, designated S_{eq}, is the alternating stress component which produces the same fatigue damage at zero mean stress as the actual alternating stress component, S_{alt}, produces at the existing value of mean stress. It can be obtained graphically from the Goodman diagram by projecting a line as shown in Fig. 7 from S_u through the point (S_{mean}, S_{alt}) to the vertical axis. It is usually easier, however, to use the simple formula

$$S_{eq} = \frac{S_{alt}}{1 - \dfrac{S_{mean}}{S_u}}. \quad (3)$$

S_{eq} is the value of stress to be used in entering the fatigue curve to find the allowable number of cycles.

The foregoing discussion of mean stress and the shift which it undergoes when yielding occurs leads to another necessary deviation from standard procedures. In applying stress concentration factors to the case of fluctuating stress, it has been the common practice to apply the factor to only the alternating component. This is not a logical procedure, however, because the material will respond in the same way to a given load regardless of whether the load will later turn out to be steady or fluctuating. It is more logical to apply the concentration factor to both the mean and the alternating component and then consider the reduction which yielding produces in the mean component. It is important to remember that the concentration factor must be applied before the adjustment for yielding is made. The following example shows that the common practice of applying the concentration factor to only the alternating component gives a rough approximation to the real situation but can sometimes be unconservative.

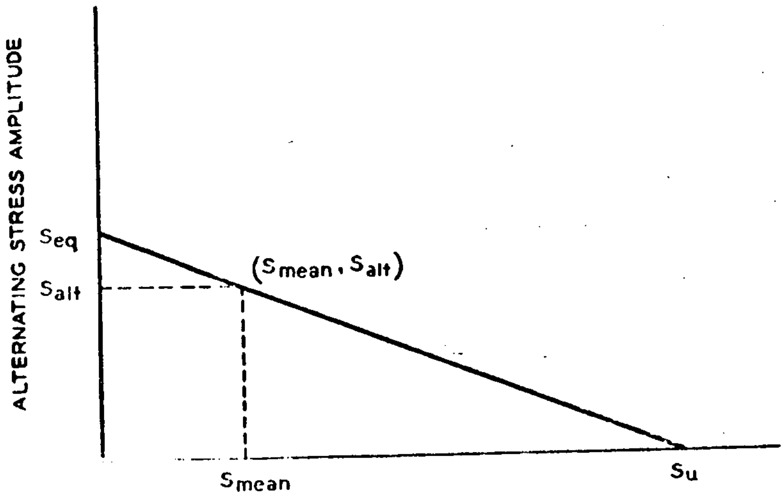


FIG. 7. GRAPHICAL DETERMINATION OF S_{eq}.

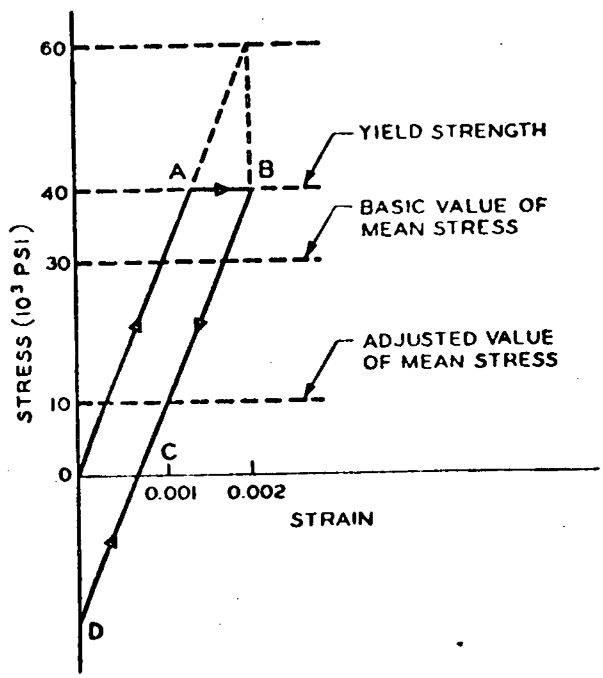


FIGURE 8. IDEALIZED STRESS VS STRAIN HISTORY

Take the case of a material with 80,000 psi tensile strength, 40,000 psi yield strength and 30×10^6 psi modulus made into a notched bar with a stress concentration factor of 3. The bar is cycled between nominal tensile stress values of 0 and 20,000 psi. Common practice would call S_{mean}, the mean stress, 10,000 psi and S_{alt}, the alternating component, $(1/2) \times 3 \times 20,000 = 30,000$ psi. The stress-strain history of the material at the root of the notch would be, in idealized form, as shown in Fig. 8. The calculated maximum stress, assuming elastic behavior, is 60,000 psi. The basic value of mean stress, S'_{mean}, is 30,000 psi, but since $S_{alt} + S'_{mean} = 60,000 \text{ psi} > S_y$ and $S_{alt} = 30,000 \text{ psi} < S_y$,

$$S_{mean} = S_y - S_{alt} = 40,000 - 30,000 = 10,000 \text{ psi}$$

and

$$S_{eq} = \frac{30,000}{1 - \dfrac{10,000}{80,000}} = 34,300 \text{ psi.}$$

It so happens that, for the case chosen, the common practice gives exactly the same result as the proposed method. Thus, the yielding during the first cycle is seen to be the justification for the common practice of ignoring the stress concentration factor when determining the mean stress component. The common practice, however, would have given the same result regardless of the yield strength of the material, whereas the proposed method gives different mean stresses for different yield strengths. For example, if the yield strength had been 50,000 psi, S_{mean} would have been 20,000 psi and S_{eq} by the proposed method would have been 40,000 psi. The common practice would have given 34,300 psi for S_{eq} and too large a number of cycles would have been allowed.

For parts of the structure, particularly if welding is used, the residual stress may produce a value of mean stress higher than that calculated by the procedure. Therefore it would be advisable and also much easier to adjust the fatigue curve downward enough to allow for the maximum possible effect of mean stress. It will be shown here that this adjustment is small for the case of low and medium-strength materials.

As a first step in finding the required adjustment of the fatigue curve, let us find how the mean stress affects the amplitude of alternating stress which is required to produce fatigue failure. In the modified Goodman diagram of Fig. 6 it may be seen that at zero mean stress the required amplitude for failure in N cycles is designated S_N. As the mean stress increases along OC', the required amplitude of alternating stress decreases along the line EC. If we try to increase the mean stress beyond C', yielding occurs and the mean stress reverts to C'. Therefore C' represents the highest value of mean stress which has any effect on fatigue life. Since S_N' in Fig. 6 is the alternating stress required to produce failure in N cycles when the mean stress is at C', S_N' is the value to which the point on the fatigue curve at N cycles must be adjusted if the effects of mean stress are to be ignored. From the geometry of Fig. 6, it can be shown that

$$S_N' = S_n \left[\frac{S_u - S_y}{S_u - S_N} \right] \quad \text{for } S_N < S_y \tag{4}$$

When N decreases to the point where $S_N \geqslant S_y$, then $S_N' = S_N$ and no adjustment of this region of the curve is required.

Figures 9, 10 and 11 show the fatigue data which were used to construct the design fatique curves for certain materials. In each case the solid line is the best-fit failure curve for zero mean stress and the dotted line is the curve adjusted in accordance with (4). Fig. 11 for stainless steel and nickel-chrome-iron alloy has no dotted line because the fatigue limit is higher than the yield strength over the whole range of cycles. A single design curve is used for carbon and low-alloy steel below 80,000 psi ultimate tensile strength because, as may be noted from Figs. 9 and 10, the adjusted curves for these classes of material were nearly identical.

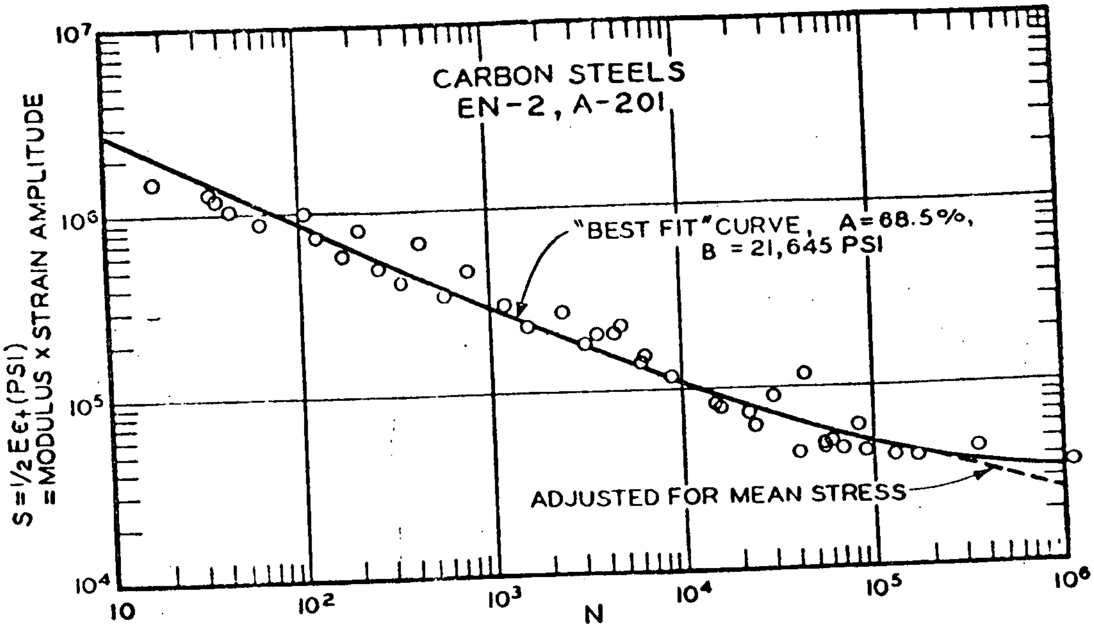


FIG. 9. FATIGUE DATA – CARBON STEELS.

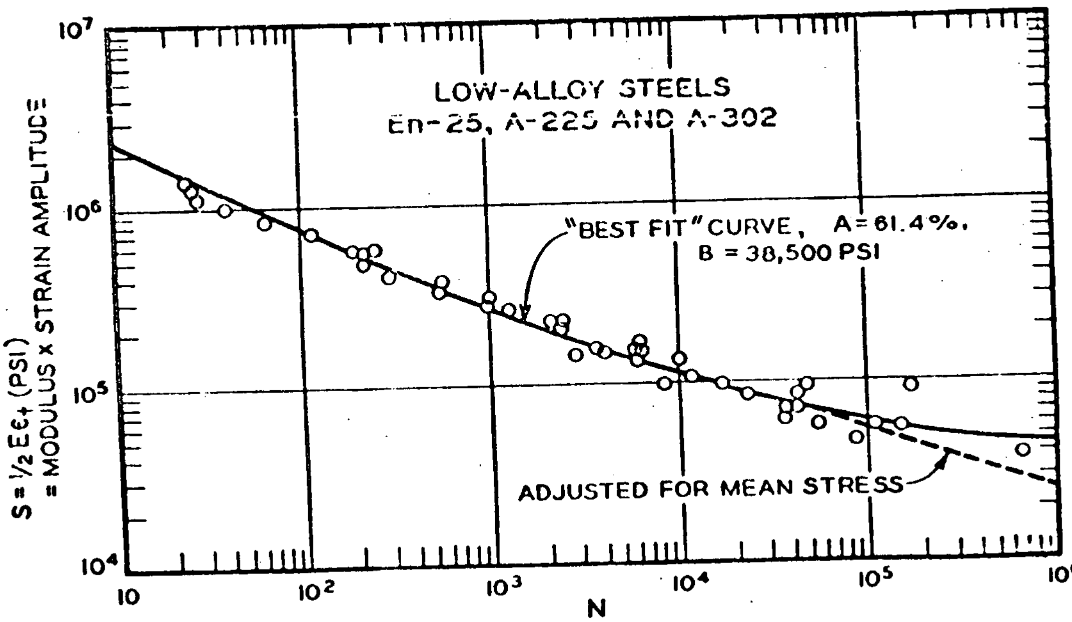


FIG. 10. FATIGUE DATA – LOW-ALLOY STEELS.

For the case of high-strength, heat-treated, bolting materials, the heat treatment increases the yield strength of the material much more than it increases either the ultimate strength, S_u, or the fatigue limit, S_N. Inspection of (4) shows that for such cases, S_N' becomes a small fraction of S_N and thus the correction for the maximum effect of mean stress becomes unduly conservative.

Test data indicate that use of the Peterson cubic equation

$$S_{eq} = \frac{7S_a}{8 - \left(1 + \frac{S_{mean}}{S_a}\right)^3} \qquad (5)$$

results in an improved method for high strength bolting materials, and this equation has been used in preparing design fatigue curves for such bolts [10].

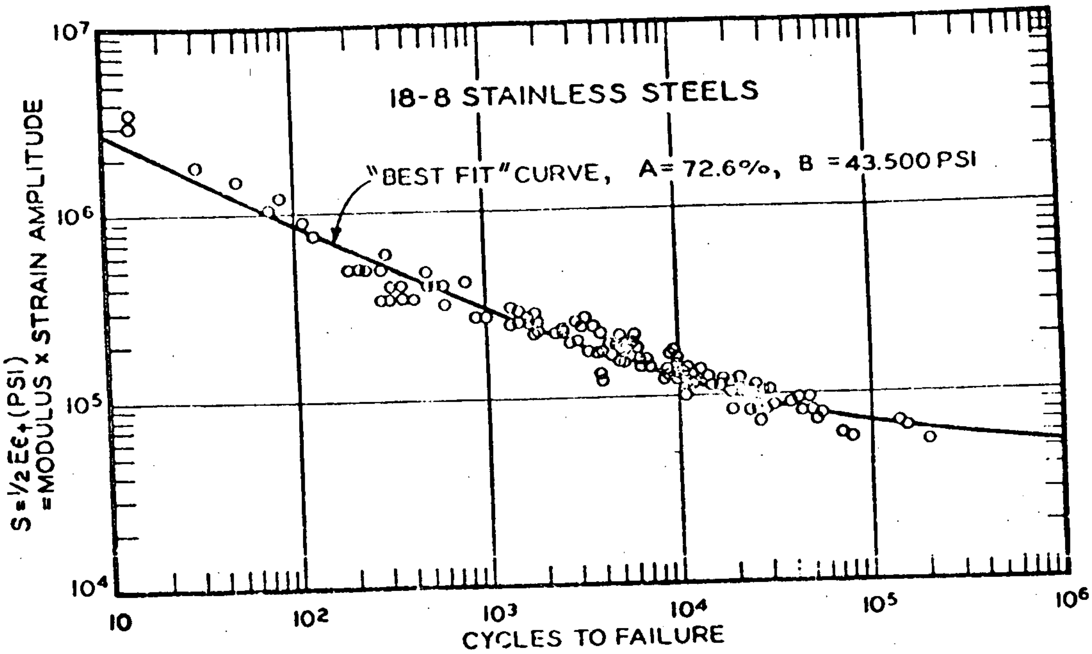


FIG. 11. FATIGUE DATA – STAINLESS STEELS.

Procedure for Fatigue Evaluation

The step-by-step procedure for determing whether or not the fluctuation of stresses at a given point is acceptable is given in detail in Par. N-415.2 of Section III and Appendix 5 of Division 2. The procedure is based on the maximum shear stress theory of failure and consists of finding the amplitude (half full range) through which the maximum shear stress fluctuates. Just as in the case of the basic stress limits, the stress differences and stress intensities (twice maximum shear stress) are used in place of the shear stress itself.

At each point on the vessel at any given time there are three principal stresses, σ_1, σ_2, and σ_3 and three stress differences, S_{12}, S_{23}, and S_{31}. The stress intensity is the largest of the three stress differences and is usually considered to have no direction or sign, just as for the strain energy of distortion. When considering fluctuating stresses, however, this concept of non-directionality can lead to errors when the sign of the shear stress changes during the cycle. Therefore the range of fluctuation must be determined from the stress differences in order to find the full algebraic range. The alternating stress intensity, S_{alt}, is the largest of the amplitudes of the three stress differences. This feature of being able to maintain directionality and thus find the algebraic range of fluctuation is one reason why the maximum shear stress theory rather then the strain energy of distortion theory was chosen.

When the directions of the principal stresses change during the cycle (regardless of whether the stress differences change sign), the non-directional strain energy of distortion theory breaks down completely. This has been demonstrated experimentally by Findley and his associates [5] who produced fatigue failures in a rotating specimen compressed across a diameter. The load was fixed while the specimen rotated. Thus the principal stresses rotated but the strain energy of distortion remained constant. The procedure outlined in Par. N-415.2(b) and 5-110(b) is consistent with the results of Findley's tests and uses the range of shear stress on a fixed place as the criterion of failure. The procedure brings in the effect of rotation of the principal stresses by considering only the *changes* in shear stress which occur in each plane between the two extremes of the stress cycle.

Cumulative Damage

In many cases a point on a vessel will be subjected to a variety of stress cycles during its lifetime. Some of these cycles will have amplitudes below the endurance limit of the material and some will have amplitudes of varying amounts above the endurance limit. The cumulative effect of these various cycles is evaluated by means of a linear damage relationship in which it is assumed that if N_1 cycles would produce failure at a stress level S_1, then n_1 cycles at the same stress level would use up the fraction n_1/N_1 of the total life. Failure occurs when the cumulative usage factor, which is the sum $n_1/N_1 + n_2/N_2 + n_3/N_3 + \ldots$ is equal to 1.0. Other hypotheses for estimating cumulative fatigue damage have been proposed and some have been shown to be more accurate than the linear damage assumption. Better accuracy could be obtained, however, only if the sequence of the stress cycles were known in considerable detail, and this information is not apt to be known with any certainty at the time the vessel is being designed. Tests have shown [6] that the linear assumption is quite good when cycles of large and small stress magnitude are fairly evenly distributed throughout the life of the member, and therefore this assumption was considered to cover the majority of cases with sufficient accuracy. It is of interest to note that a concentration of the larger stress cycles near the beginning of life tends to accelerate failure, whereas if the smaller stresses are applied first and followed by progressively higher stresses, the cumulative usage factor can be "coaxed" up to a value as high as 4 or 5.

When stress cycles of various frequencies are intermixed through the life of the vessel, it is important to identify correctly the range and number of repetitions of each type of cycle. It must be remembered that a small increase in stress range can produce a large decrease in fatigue life, and this relationship varies for different portions of the fatigue curve. Therefore the effect of superposing two stress amplitudes cannot be evaluated by adding the usage factors obtained from each amplitude by itself. The stresses must be added before calculating the usage factors. Consider, for example, the case of a thermal transient which occurs in a pressurized vessel. Suppose that at a given point the pressure stress is 20,000 psi tension and the added stress from the thermal transient is 70,000 psi tension. If the thermal cycle occurs 10,000 times during the design life and the vessel is pressurized 1000 times, the usage factor should be based on 1000 cycles with a range from zero to 90,000 psi and 9000 cycles with a range from 20,000 psi to 90,000 psi. Another example, is given in N-415.2(d)(1) and in 5-110(e).

Exemption from Fatigue Analysis

The fatigue analysis of a vessel is quite apt to be one of the most laborious and time-consuming parts of the design procedure and this engineering effort is not warranted for vessels which are not subjected to cyclic operation. However, there is no obvious borderline between cyclic and non-cyclic operation. No operation is completely non-cyclic, since startup and shutdown is itself a cycle. Therefore, fatigue cannot be completely ignored, but Par. N-415 and AD-160 gives a set of rules which may be used to justify the by-passing of the detailed fatigue analysis for vessels in which the danger of fatigue failure is remote. The application of these rules requires only that the designer know the specified pressure fluctuations and that he have some knowledge of the temperature differences which will exist between different points in the vessel. He does not need to determine stress concentration factors or to calculate cyclic thermal stress ranges. He must, however, be sure that the basic stress limits of N-414.1 to 414.4 or of 4-131 to 4-134 are met, which may involve some calculation of the most severe thermal stresses.

The rules for exemption from fatigue analysis are based on a set of assumptions, some of which are highly conservative and some of which are not conservative, but is believed that the conservatisms outweigh the unconservatisms. These assumptions are:

(1) The worst geometrical stress concentration factor to be considered is 2. This assumption is unconservative since $K = 4$ is specified for some geometries.

(2) The concentration factor of 2 occurs at a point where the nominal stress is $3S_m$, the highest allowable value of primary-plus-secondary stress. This is a conservative assumption. The net result of assumptions 1 and 2 is that the peak stress due to pressure is assumed to be $6S_m$, which appears to be a safe assumption for a good design.

(3) *All* significant pressure cycles and thermal cycles have the same stress range as the *most severe* cycle. This is a highly conservative assumption. (A "significant" cycle is defined as one which produces a stress amplitude higher than the endurance limit of the material).

(4) The highest stress produced by a pressure cycle does not coincide with the highest stress produced by a thermal cycle. This is unconservative and must be balanced against the conservatism of assumption 3.

(5) The calculated stress produced by a temperature difference ΔT between two points does not exceed $2Ea\Delta T$, but the peak stress is raised to $4Ea\Delta T$ because of the assumption that a K value of 2 is present. This assumption is conservative, as evidenced by the following examples of thermal stress:

(a) For the case of a linear thermal gradient through the thickness of a vessel wall, if the temperature difference between the inside and the outside of the wall is ΔT, the stress is

$$\sigma = \frac{Ea\Delta T}{2(1-\nu)} = .715\, Ea\Delta T \text{ (for } \nu = 0.3) .$$

(b) When a vessel wall is subjected to a sudden change of temperature, ΔT, so that the temperature change only penetrates a short distance into the wall thickness, the thermal stress is

$$\sigma = \frac{Ea\Delta T}{1-\nu} = 1.43\, Ea\Delta T \text{ (for } \nu = 0.3) .$$

(c) When the average temperature of a nozzle is ΔT degrees different from that of the rigid wall to which it is attached, the upper limit to the magnitude of the discontinuity stress is

$$\sigma = 1.83\, Ea\Delta T \text{ (for } \nu = 0.3).$$

Thus the coefficient of $Ea\Delta T$ is always less than the assumed value of 2.0.

When the two points in the vessel whose temperatures differ by ΔT are separated from each other by more than $2\sqrt{Rt}$, there is sufficient flexibility between the two points to produce a significant reduction in thermal stress. Therefore only temperature differences between "adjacent" points need be considered.

Experimental Verification of Design Fatigue Curves

The design fatigue curves are based primarily on strain-controlled fatigue tests of small polished specimens. A best-fit to the experimental data was obtained by applying the method of least squares to the logarithms of the experimental values. The design stress values were obtained from the best-fit curves by applying a factor of two on stress or a factor of twenty on cycles, whichever was more conservative at each point. These factors were intended to cover such effects as environment, size effect, and scatter of data, and thus it is not to be expected that a vessel will actually operate safely for twenty times its specified life.

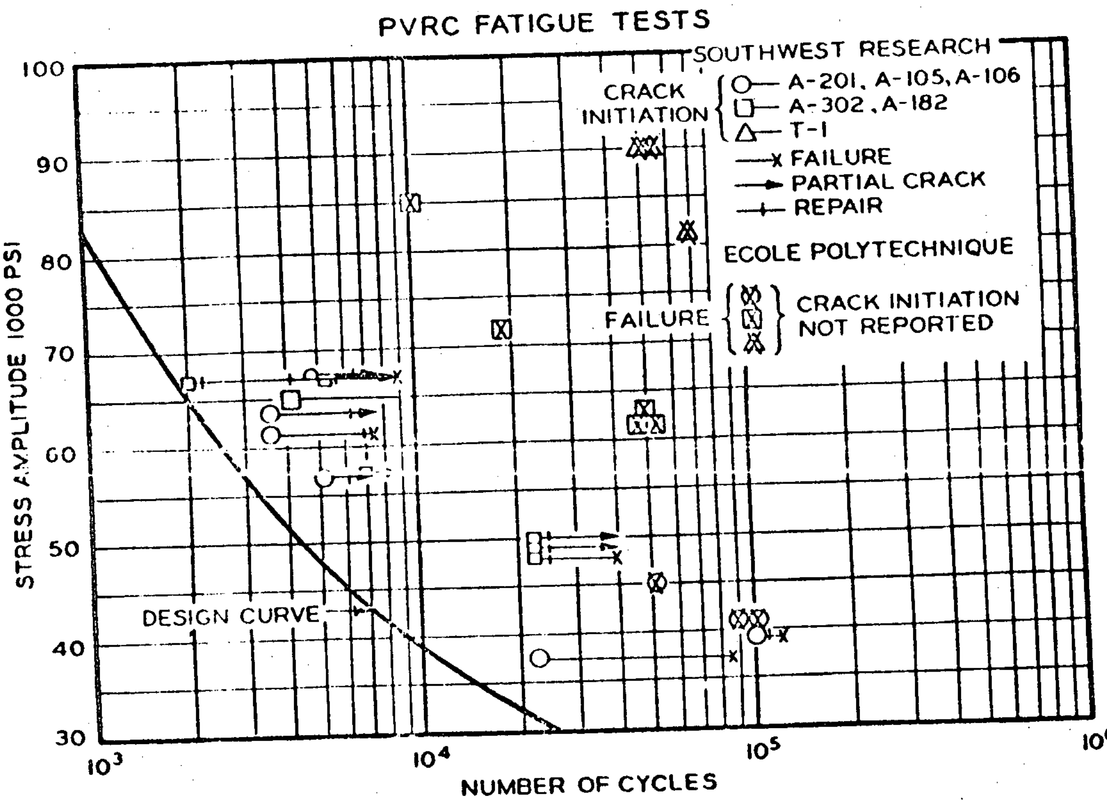


FIG. 12. PVRC FATIGUE TESTS.

The appropriateness of the chosen safety factors for fatigue has recently been demonstrated by tests conducted by the Pressure Vessel Research Committee [7,8]. In these tests 12-inch diameter model vessels and 3-foot diameter full-size vessels were tested by cyclic pressurization after a comprehensive strain gage survey was made of the peak stresses. Fig. 12 shows a summary of the PVRC test results compared to the recommended design fatigue curve of Section III for carbon and low-alloy steel. It may be seen that no crack initiation was detected at any stress level below the allowable stress, and no crack progressed through a vessel wall in less than three times the allowable number of cycles. The large scatter of the data does indicate that further research on specific materials and further studies of nozzle stresses could eventually lead to less restrictive rules for some materials and some nozzle designs. Additional data are included in Reference [9].

IV. SPECIAL STRESS LIMITS

Paragraph N-417 of Section III and Paragraphs 4-136 through 4-138 of Appendix 4 and Paragraphs 5-130 and 5-140 of Appendix 5 of Division 2 of Section VIII contain special stress limits. These deviations from the basic stress limits are provided to cover special operating conditions or configurations. Some of these deviations are less restrictive and some more restrictive than the basic stress limits. In cases of conflict, the special stress limits take precedence for the particular situations to which they apply.

The common coverage of the two Codes includes:

(a) A modified Poisson's ratio value to be used when computing local thermal stresses.

(b) Provisions for waiving certain stress limits if a plastic analysis is performed and shakedown is demonstrated.

(c) Provisions for Limit Analyses as a substitute for meeting the prescribed basic limits on local membrane stresses and on primary membrane plus primary bending stresses.

(d) A limit on the sum of the three principal stresses.

(e) Special rules to be applied at the transition between a vessel nozzle and the attached piping.

(f) Requirements to prevent thermal stress ratchet growth of a shell subjected to thermal cycling in the presence of a static mechanical load.

(g) Requirements to prevent progressive distortion on non-integral connections.

In addition, Paragraphs N-417.1 and N-417.2 of Section III and Paragraphs AD-132.1 and AD-132.2 of Div. 2 provide rules for Bearing Loads and Pure Shear, respectively.

The first three of these special rules and the rules associated with item (f) provide recognition of the growing significance of plastic analysis to the evaluation of pressure components. The shakedown analysis provides a means whereby the limit on primary plus secondary stress limits may be exceeded. This particular limit is the one with which most difficulty has been experienced in vessels subject to severe thermal transients. Unfortunately, the slow progress in developing practical methods of shakedown analysis has made this provision difficult to apply, and alternate methods are under study.

The limit analysis provision is essential when evaluating formed heads of large diameter to thickness ratio. Such heads develop significant hoop compressive stresses and meridional tensile stresses in the knuckle regions over an area in excess of that permitted by the rules for classification as local membrane stresses. A limit analysis such as that by Drucker and Shield [11] is essential and has been used to develop Figure AD-204.1 of Division 2. These techniques represent an extension to more complex geometries of the principles applied to the development of Figure 2.

The problem of potential thermal ratchet growth has been described by Miller [12], and this paper provides the basis for the Code rules.

Since the "stress intensity" limit used in these Codes is based upon the maximum shear stress criterion, there is no limit on the "hydrostatic" component of the stress. Therefore, a special limit on the algebraic sum of the three principal stress is required for completeness.

V. CREEP AND STRESS-RUPTURE

It is an observed characteristic of pressure vessel materials that in service above a certain temperature, which varies with the alloy composition, the materials undergo a continuing deformation (creep) at a rate which is strongly influenced by both stress and temperature. In order to prevent excessive deformation and possible premature rupture it is necessary to limit the allowable stresses by additional criteria on creep-rate and stress-rupture. In this creep range of temperatures these criteria may limit the allowable stress to substantially lower values than those suggested by the usual factors on short time tensile and yield strengths. Satisfactory empirical limits for creep-rate and stress-rupture have been established and used in Section I and Section VIII, Div. 1.

Creep behavior complicates the detailed stress analysis because the distribution of stress will vary with time as well as with the applied loads. The difficulties are particularly noticeable under cyclic loading. It has not yet been possible to formulate complete design criteria and rules in the creep range, and the present application of Section III and Division 2 of Section VIII is restricted to temperatures at which creep will not be significant. This has been done by limiting the tabulated allowable stress intensities to below the temperature of creep behavior. The Subgroup on Elevated Temperature is studying this problem.

VI. SUMMARY

The design criteria of Section III and Division 2 of Section VIII differ from those of Section I and Division 1 of Section VIII in the following respects:

(a) Section III and Division 2 use the maximum shear stress (Tresca) theory of failure instead of the maximum stress theory

(b) Section III and the Appendices of Division 2 require the detailed calculation and classification of all stresses and the application of different stress limits to different classes of stress, whereas Section I and Division 1 of Section VIII give formulas for minimum allowable wall thickness.

(c) Section III and Division 2 require the calculation of thermal stresses and give allowable values for them, whereas Section I and Division 1 do not.

(d) Section III and Division 2 consider the possibility of fatigue failure and give rules for its prevention, whereas Section I and Division 1 do not.

The stress limits of Section III and Division 2 are intended to prevent three different types of failure, as follows:

(a) Bursting and gross distortion from a single application of pressure are prevented by the limits placed on primary stresses.

(b) Progressive distortion is prevented by the limits placed on primary-plus-secondary stresses. These limits assure shake-down to elastic action after a few repetitions of the loading.

(c) Fatigue failure is prevented by the limits placed on peak stresses.

The design criteria described here were developed by the joint efforts of the members of the Special Committee to Review the Code Stress Basis and its Task Groups over a period of several years. It is not to be expected that this paper will answer all the questions which will be asked, but it is hoped that it will give sufficient background to justify the rules which have been given.

REFERENCES

[1] B. F. Langer, "Design Values for Thermal Stress in Ductile Materials," *Welding J. Res. Suppl.*, Vol. 37, September 1958, p. 411-s.

[2] B. F. Langer, "Design of Pressure Vessels for Low-Cycle Fatigue," *J. Basic Eng.*, Vol. 84, No. 3, September 1962, p. 389.

[3] R. E. Peterson, "Stress Concentration Design Factors," John Wiley and Sons, Inc., New York, N.Y., 1953.

[4] R. B. Heywood, "Designing by Photoelasticity," Chapman & Hall, Ltd., London, 1952.

[5] W. N. Findley, P. N. Mathur, E. Szczepanski and A. O. Temel, "Energy Versus Stress Theories for Combined Stress – A Fatigue Experiment Using a Rotating Disk," *J. Basic Eng.*, Vol. 83, No. 1, March 1961.

[6] E. E. Baldwin, G. J. Sokol, and L. F. Coffin, Jr., "Cyclic Strain Fatigue Studies on AISI Type 347 Stainless Steel," *ASTM Proc.*, Vol. 57, 1957, p. 567.

[7] L. F. Kooistra and M. M. Lemcoe, "Low Cycle Fatigue Research on Full-Size Pressure Vessels," *Welding Res. Suppl.*, July 1962, p. 297-s.

[8] G. Welter and J. Dubuc, "Fatigue Resistance of Simulated Nozzles in Model Pressure Vessels of T-1 Steel," *Welding J. Res. Suppl.*, August 1962, p. 368-s.

[9] C. W. Lawton, "High-temperature Low-cycle Fatigue: A Summary of Industry and Code Work," *Experimental Mechanics*, June, 1968, p. 264.

[10] A. L. Snow, and B. F. Langer, "Low-cycle Fatigue of Large Diameter Bolts," *Journal of Engineering for Industry*, Vol. 89, Ser. B, No. 1, Feb., 1967, p. 53.

[11] R. J. Shield and D. C. Drucker, "Design of Thin-walled Torispherical and Toriconical Pressure Vessel Heads," *Journal of Applied Mechanics*, Vol. 28, Ser. E, June, 1962, p. 262.

[12] D. R. Miller, "Thermal-stress Ratchet Mechanism in Pressure Vessels," *ASME Transactions*, Vol. 81, Ser. D, No. 2, 1959.

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Applicants' counsel have referred to please stand and raise your right hand.

Whereupon,

BERNARD F. LANGER

WARREN S. HAZELTON

MICHAEL J. MANJOINE

ROLAND J. VON OSINSKI

NOEL T. DRESSEL, JR.,

sworn.

MR. TROSTEN: Referring now to the five witnesses who have just been sworn by the Chairman, I show each of you a document entitled, "Professional Qualifications," which bears each of your names and which consists of a statement of your experience and professional qualifications. I ask you if you prepared this statement of your professional qualifications?

MR. LANGER: Yes.

MR. HAZELTON: Yes.

MR. MANJOINE: Yes.

MR. VON OSINSKI: Yes.

MR. DRESSEL: Yes.

MR. TROSTEN: Are these statements of your professional qualifications true and correct?

MR. LANGER: Yes.

MR. HAZELTON: Yes.

MR. MANJOINE: Yes.

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MR. VON OSINSKI: Yes.

MR. DRESSEL: Yes.

MR. TROSTEN: Do you desire to have these statements of your professional qualifications incorporated in this transcript as if read and received into evidence in this proceeding?

MR. LANGER: Yes.

MR. HAZELTON: Yes.

MR. MANJOINE: Yes.

MR. VON OSINSKI: Yes.

MR. DRESSEL: Yes.

MR. TROSTEN: Mr. Chairman, copies of these papers of professional qualifications will be furnished to the reporter, the Board and the parties. I ask that these statements be received in evidence in this proceeding.

end

PROFESSIONAL QUALIFICATIONS
BERNARD F. LANGER
CONSULTING ENGINEER
NUCLEAR ENERGY SYSTEMS
WESTINGHOUSE ELECTRIC CORPORATION

My name is Bernard F. Langer. My residence is 6814 Linden Lane, Pittsburgh Pennsylvania 15208. I am presently consultant to the Westinghouse Electric Corporation, Nuclear Energy Systems, in the areas of mechanical design, stress analysis and the design and safety of pressure vessels.

I was graduated from Stanford University in California with a degree of BA in mechanical engineering in 1926 and Engineer in mechanical engineering in 1928. My professional career has all been with the Westinghouse Electric Corporation. I joined the Westinghouse Research Laboratory in 1928 and I was associated with the Mechanics Department. In 1946 I transferred to the Transportation Engineering Department as manager of the Mechanical Section. In 1949 I joined the Bettis Atomic Laboratory where I held the position of manager of Structural and Heat Engineering. In 1950 I was transferred to the position of consulting engineer and have held this position at Bettis, until 1966, at the Research Laboratories until 1968 and presently at Nuclear Energy Systems. I retired from full time service in December 1970 but still spend a major portion of my time in consulting work for Westinghouse.

A large part of my activity has been connected with the Boiler and Pressure Vessel Committee of the American Society of Mechanical Engineers (ASME). Since April 1964 I have acted as chairman of the Subcommittee on Nuclear Power which is responsible for the original preparation and revisions to Section III of the ASME Boiler Code. Section III presently covers all pressure components in nuclear power plants. I am also a member of the Executive Committee of the ASME Boiler & Pressure Vessel Committee which prepares and services codes for all types of pressure retaining components.

My society memberships include:

1) American Society of Mechanical Engineers
2) Society for Experimental Stress Analysis
3) American Nuclear Society
4) American Society for Testing and Materials
5) Pressure Vessel Research Committee

I have been awarded the following honors:

1) Westinghouse Order of Merit - 1941
2) Fellow ASME - 1959
3) Murray Lecturer, SESA - 1970
4) J. Hall Taylor Medal, ASME - 1970

I have written over thirty technical papers and contributions to books including several on various aspects of pressure vessel design.

PROFESSIONAL QUALIFICATIONS
WARREN S. HAZELTON
ADVISORY ENGINEER, MATERIALS ENGINEERING
PWR SYSTEMS DIVISION
WESTINGHOUSE ELECTRIC CORPORATION

My name is Warren S. Hazelton, and my address is Box 248, R. D. #1, Jeannette, Pennsylvania, 15644. I am an Advisory Engineer in Materials Engineering at the PWR Systems Division of the Westinghouse Electric Corporation. In this position I am responsible for the technical direction of materials work primarily related to reactor safety and integrity.

I was graduated from the University of Minnesota in 1949 with a Bachelor of Metallurgical Engineering degree, with honors.

From 1949 to 1960 I was employed in the Westinghouse Aviation Gas Turbine Division, at South Philadelphia and at Kansas City, Mo. From 1954 to 1960, I was manager of the materials application and development activity, responsible for the materials aspect of design, materials properties, failure analysis, and the development of new materials.

From 1960 to 1963 I was Supervising Engineer of the Materials Development section at the Westinghouse Bettis Atomic Power Laboratory. In this capacity I was responsible for development programs in the fields of stress corrosion, brittle fracture prevention, and radiation damage.

From 1963 to the present time I have held various management positions in what is now the PWR Systems Division. My responsibilities have included the development and application of improved fracture prevention technology, evaluation of radiation damage, stress corrosion prevention, with close interface with design groups. I was responsible for the detailed failure analysis performed on the internals at the Yankee Row, Connecticut Yankee, Trino (Italy) and SENA (Franco-Belg.)

plants. I also participated actively in the redesign and repair work performed for these plants.

I am a member of the American Society on Metals and American Society for Testing and Materials Subcommittees on radiation effects. I also am a member of the Pressure Vessel Research Committee (PVRC) Subcommittee on Pressure Vessel Materials, chairman of the Task Group on Effect of Flaws and Variations in Properties, and a member of the PVRC Task Group on Fracture Toughness Requirements for Nuclear Power Reactors.

PROFESSIONAL QUALIFICATIONS
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WESTINGHOUSE RESEARCH & DEVELOPMENT CENTER
WESTINGHOUSE ELECTRIC CORPORATION

My name is Michael J. Manjoine. My address is 25 Lewin Lane, Pittsburgh, Pennsylvania 15235. I am a consultant in Mechanics of Materials and Mechanical Metallurgy at the Westinghouse Research Laboratories, Westinghouse Research & Development Center, Westinghouse Electric Corporation. In this position I act as consultant to all Divisions of the Company on the properties of materials, their application in design and design criteria. My thirty years of research and experience in the field of mechanics of materials provides a broad base for the application of materials in the design and analysis of structural components in many Westinghouse products. My current research is a study of the effect of the state of stress and strain rate on the interaction of plastic flow, creep and fatigue.

I received a B.S. degree in Mechanical Engineering and Electrical Engineering in 1937 from Iowa State University and an M.S. in Mechanical Engineering from the University of Pittsburgh in 1939. I attended the Advanced Mechanics Design School at Westinghouse Electric Corporation in 1937-1938. I was appointed Research Fellow at the University of Pittsburgh in 1938 and conducted research on the strain rate effects on materials for two years. This pioneering work is published in three papers in the American Society of Mechanical Engineers and American Society of Testing and Materials Transactions and has received international recognition.

In 1940 I returned to Westinghouse Electric Corporation as a research engineer and later as a Fellow Engineer. During the period 1950 to 1961

I was consultant in mechanical metallurgy at Westinghouse Bettis Atomic Power Laboratory during the development of the atomic reactor for submarine propulsion. In 1961 I was assigned to the Westinghouse Astronuclear Laboratories as Manager of the Engineering Mechanics Department to assist in the development of the NERVA engine for space propulsion. This department was responsible for stress analysis and component evaluation. In 1963 I was appointed Consultant for Mechanical Experimentation and Component Design.

In 1968, I returned to Westinghouse Research Laboratories as a Consultant in the Mechanics Department. The major portion of the past three years has been spent in consulting for the PWR Systems Division, Nuclear Energy Systems. My extensive work on evaluating and prediction of stresses in pressurized water nuclear reactors is reported in two papers to be presented at the first International Congress on Structural Mechanics in Reactor Technology, 1971.

I have published 22 papers in technical societies, 7 in conferences and have written chapters for 4 different books relative to material properties and their application.

In 1953 I received the ASTM Dudley Medal for the paper (in which I participated jointly with Mr. E. A. Davis) entitled "Effect of Notch Geometry on Rupture Strength at Elevated Temperature".

I am a member of the American Society of Mechanical Engineers (ASME) and received the grade of Fellow in 1967. I am a member of the American Society on Metals (ASM) and in 1970 was included in the first group of Fellows of the society. I am a member of the American Society of Testing and Materials (ASTM).

During the period 1956-1960 I was a member of the Executive Committee of ASME Metals Engineering Division and National Chairman in 1959. In 1959 I was chairman of the Basic Science Committee, ASME, Pittsburgh Section; in 1960-1963 I was national Chairman of the ASME-IME-ASTM for the 1963 Joint International Conference on Creep; in 1958-1966 I was National Secretary, ASME Research Committee on Prevention of Fracture in Metals and National Chairman of Subcommittee on Combined Stress. From 1965 to the present I have been a member of the ASME Research Committee on Effect of Radiation on Materials and Chairman of Committee on Stress Relaxation of Metals Properties Council. From 1969 to the present I have been a member of subcommittee on Elevated Temperature Design of PVRC and a member of the ASTM Committee on Stress Relaxation.

PROFESSIONAL QUALIFICATIONS
ROLAND J. VON OSINSKI
MANAGER, REACTOR VESSELS
PWR SYSTEMS DIVISION
WESTINGHOUSE ELECTRIC CORPORATION

My name is Roland J. Von Osinski. My address is 108 Alstan Court, Monroeville, Pennsylvania 15146. I am Manager, Pressure Vessels, in the Engineering Department at the PWR Systems Division, Nuclear Energy Systems of the Westinghouse Electric Corporation. In this position I am responsible for the design development and engineering procurement of reactor vessels, pressurizers and associated mechanical equipment for pressurized water nuclear plants.

I graduated in 1955 with a BSAE degree from Ohio University. I also completed graduate courses in Pressure Vessel Design from Akron University. I am a member of the American Society of Mechanical Engineers (ASME), and serve as a member of the subgroup on general requirements of the subcommittee on nuclear power plant components. Formerly I was secretary of the ASME subcommittee on nuclear power.

I have been employed by Westinghouse since 1967, and have held assignments in the Pressure Vessel and Tool Design group as a senior engineer prior to my appointment to my present position in March, 1969.

Prior to joining Westinghouse, I spent 10 years in the employ of Babcock & Wilcox Company in the Nuclear and Special Products Engineering Department. My tenure at B&W was spent in various engineering capacities all associated with Pressure Vessel design, analysis and fabrication. I spent approximately four years as a stress analyst, six months as a proposal engineer and four and one-half years as a project engineer. As a project engineer I controlled engineering contracts for commercial nuclear reactor vessels, Navy nuclear reactor vessels, Navy steam generators, complete nuclear primary power systems, ammonia converters, Isomax reactors and other pressure vessels.

PROFESSIONAL QUALIFICATIONS
NOEL T. DRESSEL, JR.
MANAGER, QUALITY ASSURANCE/PRESSURE VESSELS
PWR SYSTEMS DIVISION
WESTINGHOUSE ELECTRIC CORPORATION

My name is Noel T. Dressel, Jr. My residence is 318 Marose Drive, Pittsburgh, Pennsylvania, 15230. I am Manager of the Pressure Vessel section of the Quality Assurance Department at the PWR Systems Division, Nuclear Energy Systems of the Westinghouse Electric Corporation. In my present position, I am responsible for development of quality assurance programs and for providing an effective supplier quality surveillance program to assure that equipment is manufactured, inspected, and tested in accordance with engineering requirements.

From 1941 to 1946 I served in engineering officer capacities aboard United States merchant vessels. In the period 1946 to 1948 I was associated with the operation, maintenance, and repair of industrial power plant equipment.

From 1948 to 1955 I was employed as Inspector and subsequently Supervising Engineer of a field office of the Ocean Accident and Guarantee Corporation. In this capacity I was responsible for the manufacturing and inspection follow of pressure vessel components to assure compliance to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Codes in addition to the inspection of operating boilers and mechanical equipment. In the period 1955 to 1958 I was employed by Gulf Engineering Company, Inc. as Service Engineer dealing primarily with industrial water control systems.

I joined the Plant Apparatus Division of Westinghouse Electric Corporation in 1958 as a Quality Control Engineer and was subsequently assigned Supervisor of the Pressure Vessel section of the Quality Control Department. During this period my responsibilities required follow of various pressure vessel and mechanical equipment for pressurized water reactor systems utilized in the naval nuclear program to assure compliance to engineering specifications and participation in development of quality programs associated with military specifications. In 1968 I joined the PWR Systems Division in my present position.

I am a member of the American Society of Nondestructive Testing.

CHAIRMAN JENSCH: Is there any objection on behalf of the Regulart ory Staff?

MR. KARMAN: No objection, Mr. Chairman.

CHAIRMAN JENSCH: Any objection on behalf of the Citizens' Committee for the Protection of the Environment?

MR. ROISMAN: No objection.

CHAIRMAN JENSCH: Any objection on behalf of the New York State Atomic Energy Council?

MR. MARTIN: No objection.

CHAIRMAN JENSCH: All parties indicated no objection. The statement of the qualifications of the witnesses to whom Applicant's counsel has just referred may be physically incorporated into the transcript and placed in the transcript.

Proceed, please.

MR. TROSTEN: Referring again to the document entitled, "Additional Testimony of Applicant Concerning Reactor Vessel Integrity," and addressing my question now to all seven panel witnesses. Was this document prepared under your collective supervision and direction and by you, and are the statements contained in this document true and correct to the best of your knowledge?

MR. LANGER: Yes.

MR. FAZELTON: Yes.

MR. MANJOINE: Yes.

MR. VON OSINSKI: Yes.

MR. DRESSEL: Yes.

MR. GROB: Yes.

MR. WEISEMANN: Yes.

MR. TROSTEN: Do you desire to have the copy to which I have referred received in evidence in this proceeding as your testimony?

MR. LANGER: Yes.

MR. HAZELTON: Yes.

MR. MANJOINE: Yes.

MR. VON OSINSKI: Yes.

MR. DRESSEL: Yes.

MR. GROB: Yes.

MR. WEISEMANN: Yes.

MR. TROSTEN: Mr. Chairman, I now offer in evidence the document entitled, "Additional Testimony of Applicant concerning Reactor Vessel Integrity," dated September 17, 1971. I ask that it be physically incorporated into the transcript and received in evidence in this proceeding.

CHAIRMAN JENSCH: Rather than by an exhibit; is that correct?

MR. TROSTEN: Yes.

CHAIRMAN JENSCH: Is there any objection on behalf of the Regular Tory Staff?

MR. KARMAN: No objection, Mr. Chairman.

CHAIRMAN JENSCH: Citizens' Committee for the Protection of the Environment?

MR. ROISMAN: No objection.

CHAIRMAN JENSCH: Hudson River Fishermen's Association.

MR. MacBETH: No objection.

CHAIRMAN JENSCH: New York State Atomic Energy Council.

MR. MARTIN: No objection, Mr. Chairman.

CHAIRMAN JENSCH: Very well. The request is granted and the additional testimony in the form of the document to which Applicant's counsel has just referred is received in evidence, and the document containing that additional testimony may be physically incorporated into the transcript and placed in the transcript.

Proceed, please.

MR. TROSTEN: Thank you.

Mr. Chairman, Applicant has two other documents which it desires to have received in evidence in this proceeding. However, both of these documents which pertain to the security system for the Indian Point Station, Applicant wishes to have received in evidence in an in camera proceeding. In view of the fact that Mr. Roisman is not going to be cross-examining Applicant's witnesses today, Applicant will prefer that the formal receipt into evidence of this document be deferred until such time as an in camera session of the hearing is held, in view of the fact that Applicant does not wish to have this document formally received into the open transcript of this proceeding.

CHAIRMAN: Very well. That may be done and we will arrange an in camera session at a later time and be scheduled at a time convenient to all attorneys in the proceeding.

Do you desire to have those documents identified by any numerical identification at this time, or should that wait until the in camera proceeding?

MR. TROSTEN: I will indicate it now.

CHAIRMAN JENSCH: Indicate the next succeeding number for the exhibits, please.

MR. TROSTEN: I beg your pardon?

CHAIRMAN JENSCH: Please indicate the next succeeding number.

MR. TROSTEN: I would suggest, Mr. Chairman, that these be incorporated in the in camera transcript rather than be received as exhibits, if that is satisfactory to you.

CHAIRMAN JENSCH: Very well. If that be the case, the entire matter may wait until the in camera session.

MR. TROSTEN: Very well.

CHAIRMAN JENSCH: Does that complete all the evidence of the Applicant?

MR. TROSTEN: Yes, Mr. Chairman.

The applicant has no further evidence to offer.

CHAIRMAN JENSCH: Before we proceed to inquire as to the arrangements made, as I understand, a letter of transmittal from one of the parties in this proceeding, let me inquire generally about some matters pertaining to the Indian Point Facility.

As I recall the reference to arrangements, the attorneys have met, as I understand it, sometime within the last week, and discussed procedures that will aid it in the presentation of evidence and the manner of expediting this proceeding to bring it to a conclusion; is that correct?

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MR. TROSTEN: Yes, that is correct, Mr. Chairman. We have been meeting, Applicant's counsel has been meeting with Mr. Roisman and Mr. MacBeth and also with counsel for the other parties and we have reached, Applicant has reached certain understandings with the Intervenors concerning the consideration of Applicant's pending motion for a limited operation license, and also with respect to the case and manner of consideration of the full environmental hearing.

I have discussed this with Mr. Roisman several times, including our discussion last night, and we agreed last night that Mr. Roisman would present the general understandings to the Board at this time, subject to comment by myself and Mr. MacBeth. I would propose, Mr. Chairman, that we consider that after we had reviewed certain other matters that could be taken up today.

There is one other matter in addition to any questions that the Board may wish to raise concerning the evidence that has been introduced today, Mr. Chairman, which I think that we ought to take up, and that concerns the fuel-loading authorization for the Indian Point 2 facility. Mr. Karman has certain information which he wishes to convey to the Board, and if it is satisfactory to the Board perhaps we could consider that right now.

CHAIRMAN JENSCH: Very well. But before doing that

I should announce that Congressman John Dow, who has made a limited appearance in this proceeding by the presentation of a statement through his representative, has requested an additional opportunity to present a further limited appearance statement, and the gentleman representing the Congressman is here and we will be glad to have the statement from the Congressman at this time. Will you come forward please, sir, and give your name and address if you will, please, and identify yourself in any manner you desire.

Since this is a limited appearance statement the gentleman need not be sworn.

Any members of the public who are here probably recall our statement at the beginning of these proceedings that members of the public are invited by the Commission to present statements by way of limited appearance. Naturally, they can participate in the proceedings and express some concerns that they may have respecting those proceedings. Those statements are not sworn. Those statements are not evidence. They do not constitute any basis upon which any statement can be made in the proceeding, or rather the procedure for limited appearance statements is intended to provide a limited manner in which members of the public can participate in a proceeding of this kind. The Commission encourages the presentation of statements by way of limited appearance, and at the outset of these proceedings two opportunities, I believe, two occasions, were provided for the presentation of statements by way of limited appearance. Many members of the public did present statements, and as I have indicated, Congressman Dow through this gentleman who is here today also presented a statement by way of limited appearance. The Congressman has decided an opportunity of making an additional statement and the Board is agreeable to changing the schedule for the presentation of statements by way of limited appearances to this extent.

Will you proceed, sir.

MR. EGAN: My name is William Egan. I am district representative for Congressman John Dow, 27th Congressional District. And he has sent the following testimony.

"Mr. Examiner:

"Since the Indian Point plants of the Consolidated Edison Company are almost surrounded by my Congressional District, even though they are not located in it, I am deeply concerned to assure that they will be safe in all respects.

"I have a good many doubts from listening to members of the scientific community, about the safety of present questions raised about the radioactivity of the air and water in the vicinity of nuclear plants, about thermal pollution, and about wastes disposed of, have not been conclusively answered by adequate authorities. The most commonly noted danger of these plants seems to be a 'peace-time catastrophe whose scale might well exceed anything the nation has ever known.'

"The foregoing words are taken from a publication titled 'Nuclear Reactor Safety: An Evaluation of New Evidence.' This is a paper presented by the Union of Concerned Scientists, Cambridge, Massachusetts, July, 1971. In it, the three authorities on nuclear matters and one economist concerned with the environment have joined in a critique about the potential for catastrophe that resides in present-day nuclear reactors. I am sure that you have read the evaluation by the four authorities. I do want you to give it full weight in your deliberations. The authors recommend a total halt to the issuance of operating licenses until safeguards of assured performance can be provided and a thorough technical review to determine whether reactors now constitute an unacceptable hazard to the population.

"In order to emphasize the point I am making, I append hereto the part IV of the evaluation by the four scientists which has been title 'Conclusions and Recommendations.' It is important for you and all those at this hearing to hear their conclusions and recommendations even though this may not be the first time that you have done so."

Accordingly, I am attaching this statement which I will turn over to you.

CHAIRMAN JENSCH: If you will give it to the Reporter, and it is requested that the Reporter transmit your document to the Secretary of the Commission who will file the statement with these statements from persons making limited appearances. Will that be agreeable?

MR. EGAN: Yes. On behalf of the Congressman I wish to thank you for allowing me to make this appearance.

CHAIRMAN JENSCH: Let me inquire, sir. Your statement as I recall it said something to this effect, that the Congressman has been listening to many members of the scientific community. Has he had an opportunity to listen to any of the evidence in this case? I mean I don't know whether he has been attending any of the hearings here.

MR. EGAN: No. No, he has not attended the hearings here.

CHAIRMAN JENSCH: Would he desire to do that? Perhaps he'd like a list of some of the evidence that's going to be directed specifically to this plant. The reason I ask is that we will have a session of hearings extending for quite some considerable time starting November 1, and if there is some particular day that he could attend perhaps we could schedule some of the witnesses who would deal with the matters in which he has a particular interest. Will you inquire of the Congressman in that respect?

MR. EGAN: I will, and perhaps it can be arranged.

CHAIRMAN JENSCH: Very well. I think the Commission welcomes the attendance of persons who have concerns about the proceedings and about the actions that are under examination, and if the Congressman has been listening to some members who have concerns maybe he could bring those members of the scientific community with him so that they would have a chance to hear the evidence specifically directed to this plant. And I am sure the Board will endeavor to schedule a time and arrange with the parties to have witnesses present who can discuss in some detail the evidence of this case, because I think in that way you'd probably get several sides of the controversy, and that may be helpful to the Congressman in formulating his conclusions. We'd be glad to do that if you will inquire and let us know. Will you do that, please?

MR. EGAN: All right. We can let you know through the New York office, your New York office.

CHAIRMAN JENSCH: You can write to the Secretary of the Atomic Energy Commission so that all matters are public records available to everybody, and the Secretary, I am sure, will transmit that information to the Board by communication, which will also be available to all members of the proceeding.

MR. EGAN: Well, I will relay this and I will work on it and see what we can do. Those dates, it would seem at that time we may have some time available around that time.

CHAIRMAN JENSCH: Thank you, Mr. Egan.

end

MR. EGAN: Okay. Thank you very much.

(The following are the conclusions and the recommendations as submitted by Senator Dow.

"The grave weaknesses apparent in engineering knowledge of emergency core--cooling systems and the strong implications that these systems would fail to terminate safely a loss-of-cooolant accident makes it clear that in the event of a major reactor accident the United States might easily suffer a peace-time catastrophe whose scale as we have seen might well exceed anything the nation has ever known.

"The gaps in basic knowledge concerning the effectivensss of the safety features of large power reactors, the surprising scarcity of adequate tests--amounting nearly to a lack--has astonished our group, especially in view of the large number of apparently hazardous design that are already operating. Not until 14 years after the publication of WASH-740 do we see experimental tests of an emergency core-cooling system, tests carried out on nothing larger than a nine inch model, described by the AEC as not meant to simulate a reactor fully. It is now over eleven years since the first reactor for the commercial production of power was brought into operation.

"The hazards inherent in the present situation have not gone entirely unnoticed by the AEC; the Commission was evidently disturbed by the Idaho test results and appointed a Task Force to assess them. The Task Force report has not yet been released but 'interim' criteria for determining the adequacy of emergency core-cooling systems were published in the FEDERAL REGISTER. In an unusual move the AEC waived the normal 60-day waiting period, noting, 'In view of the public health and safety considerations... the Commission has found that the interim acceptance criteria contained herein should be promulgated without delay, that notice of proposal issuance and public procedure thereon are impracticable, and that good cause exists for making the statement policy effective upon publication in the FEDERAL REGISTER.'

"In the delicately chosen words of AEC Chairman Glenn T. Seaborg, 'The use of recently developed techniques for calculating fuel cladding temperatures following postulated loss-of-coolant accidents, and the results of recent preliminary safety research experiments, have indicated that the predicted margins of ECCS (Emergency Core-Cooling System) performance may not be as large as those predicted previously.'

"The Chairman has failed to indicate why obviously critical preliminary safety tests have not, until this year, been carried out in view of the potential hazards associated with the 21 power reactors authorized by the AEC and now operating.

"We have concluded that there are major and critical gaps in present knowledge of safety systems designed to prevent or ameliorate major reactor accidents. We have further concluded that the scanty information available indicates that presently installed emergency core-cooling systems would very likely fail to prevent such a major accident. The scale of the consequent catastrophe which might occur is such that we cannot support the licensing and operation of any additional power reactors in the United States, irrespective of the benefits they would provide to our power-shy nation.

"We do not believe it is possible to assign a reliable numerical probability to the very small likelihood of a loss-of-coolant accident in a power reactor. There are too many sources of uncertainty whose importance cannot be adequately assessed. The acquisition of this information by trial and error, in the absence of safeguards that would mitigate or prevent core meltdown, could be extremely costly to the nation.

"While it appears that the probabilities are not very large we do not believe that a major reactor accident can be totally or indefinitely avoided. The consequences of such an accident to public health are too grave to assume anything more than a very conservative position. Accordingly we have concluded that power reactors <u>must</u>

<u>have</u> <u>assured</u> <u>safeguards</u> against a core meltdown following a serious reactor accident.

"We have grave concern that reactors now operating may at present offer unacceptable risks and believe these risks must be promptly and thoroughly assessed.

"Accordingly, we recommend:

"1) A total halt to the issurance of operating licenses for nuclear power reactors presently under construction, until safeguards of assured performance can be provided.

"2) A thorough technical and engineering review, by a qualified, independent group, of the expected performance of emergency core-cooling systems installed in operating power reactors to determine whether these reactors now constitute an unacceptable hazard to the population.

"It is apparent that a major program will be required to develop, through both theoretical studies and experimental measurement, information adequate to design reactor safety features that will ensure protection against core-meltdown following a loss-of-coolant accident.

"We believe that a complete and adequate understanding of loss-of-coolant accidents can be gained.

Moreover, there appear to be no technical difficulties so acute that adequate protection from the consequences of a major accident cannot be assured. The United States will become increasingly dependent on nuclear power. Nuclear power can be both clean and safe but it will not be, in years to come, if the country is allowed to accumulate large numbers of aging reactors with flaws as critical and important as those we now see. It is past time that these flaws be identified and corrected so the nation will not be exposed to hazards from the peaceful atom.

"John G. Dow
"M. C."

CHAIRMAN JENSCH: Regulartory Staff Council, do I understand you have a statement in reference to the fuel loading? Is this the fuel loading authorization that was issued by the Atomic Safety and Licensing Board in July of 1971?

MR. KARMAN: Yes, Mr. Chairman.

CHAIRMAN JENSCH: Has the loading been undertaken?

MR. KARMAN: No, Mr. Chairman.

CHAIRMAN JENSCH: Excuse me. I understood in July you were ready to load and that is why we, I think we--

MR. KARMAN: If I may, Mr. Chairman.

CHAIRMAN JENSCH: We stopped the proceedings in order to get the order out promptly for the loading. You have not started the loading?

MR. KARMAN: If I may, Mr. Chairman, possibly this will help explain the situation.

CHAIRMAN JENSCH: Yes. Please.

MR. KARMAN: On July 20th the Atomic Safety and Licensing Board issued an order for fuel loading, issued an order authorizing the Director of Regulation to issue a license for the fuel loading for Indian Point 2 plant. At the time, Mr. Chairman, certain prerequisites have not been met with respect to the completion of the plant which would not allow the Director of Regulation to issue a fuel loading license.

On July 23rd the Court of Appeals in the District of Columbia announced its decision in the renowned Calvert's Cliffs case and on September 9th the Atomic Energy Commission promulgated Revised Appendix D to 10 CFR 50.

We have before us, Mr. Chairman, a unique situation. This is the only occasion of its kind where an order of a Board was issued which would authorize a license by the Director of Regulation. Because of circumstances this license was not issued. It is my understanding--

CHAIRMAN JENSCH: Excuse me. You say

because of the circumstances. What circumstances?

MR. KARMAN: The circumstances were as I indicated, Mr. Chairman, that certain prerequisites were not met with respect to the completion of the plant to allow the Director of Regulation to issue the fuel loading license.

CHAIRMAN JENSCH: You say that some of the hardware was not available?

MR. KARMAN: That is correct, sir.

CHAIRMAN JENSCH: What hardware was not included?

MR. KARMAN: If need be, Mr. Chairman, I will allow either the Applicant or the Regular tory Staff's compliance witness to indicate what the--

CHAIRMAN JENSCH: Well, if you could generally give us one or two or three or four fuel items. I mean that the reactor, the vessel is there. The ground is available. Do you have the material?

This gentleman has been sworn, has he not?

MR. KARMAN: Yes. This is Mr. Madsen, Mr. Chairman.

CHAIRMAN JENSCH: You can stand there, Mr. Madsen. Just tell us generally. If you can pick up a microphone and speak perhaps everyone can hear.

end

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MR. MADSEN: As of this date the plant has not met the requirements of the motion that was presented to this Board. Many of the items or most of the items, I should say, are complete. Things that I must still verify as having been completed include such things as pipe hangers and supports, completion of the pre-operational tests, results, review, and along with this resolution of any problems which come about.

CHAIRMAN JENSCH: Namely, what problems?

MR. MADSEN: Well, any time you run a test, for instance in the case of the RHR, residual heat removal system.

CHAIRMAN JENSCH: Thank you.

MR. MADSEN: There was an indication of a flow discrepancy from one loop to the next. It is believed that it is an instrumentation problem. This test is to be redone during the refill of the rapid cooling system just prior to core loading.

Since the last session there has been work going on in the repair of steam generators. This work is nearly complete and may be complete as of this time. This is in a general way what the situation is. It is conceivable from the best of my knowledge that the work could be completed by as early as, say, Monday, which is the 10th, I believe, of this month.

CHAIRMAN JENSCH: That's your present judgment of it now?

THE WITNESS: That's based on what is reported to me and based on what I have been told is the schedule of work towards the completion of the activities to fulfill the motion presented to the Board.

CHAIRMAN JENSCH: Are all departments of the Applicant coordinated on this information?

MR. TROSTEN: I am sorry, Mr. Chairman, I don't quite understand your question.

CHAIRMAN JENSCH: Do all departments of the Applicant know about this status of--

MR. TROSTEN: Yes, Mr. Chairman.

CHAIRMAN JENSCH: The reason I asked I wondered if all departments are kind of putting their shoulder to the wheel. As I understand, there is an endeavor to expedite the proceeding and get things underway, and the Board is anxious to provide every opportunity that the Board can provide so that the matter will go forward.

MR. TROSTEN: Yes.

CHAIRMAN JENSCH: And if there is anything further the Board can do--is there anything the Board can do about this right now?

MR. TROSTEN: Mr. Chairman, in terms of our completing all of the work that was a prerequisite for the Staff's issuing the fuel loading authorization, no, Mr. Chairman, we simply want to complete the work as expeditiously as we can. And let me assure the Chairman that we are

C2Bt3

certainly doing everything we can to have that work completed.

CHAIRMAN JENSCH: At least there is nothing that the Board can now do to expedite that phase of it.

MR. TROSTEN: With respect to that particular phase of it, no, Mr. Chairman, that is correct.

CHAIRMAN JENSCH: That will have to precede any other phase, will it not?

MR. TROSTEN: Mr. Chairman, it is necessary for us to complete that aspect of the work before the license can issue. However, as Mr. Karman is about to say, there are certain administrative--

(Discussion off the record.)

CHAIRMAN JENSCH: Will you proceed with what you are going to say. Who wants to say it the way it's going to be?

MR. TROSTEN: The point is, Mr. Chairman, there may be certain administrative steps that may have to be taken as well as the completion of this work.

CHAIRMAN JENSCH: Will you proceed, Mr. Karman.

MR. KARMAN: Yes. Mr. Chairman, as I indicated, the revised attended was promulgated on September 9 and as this is the only case of its kind we of the Regulatory Staff in light of our endeavor to comply with the mandates and the wishes of the Court of Appeals in the Calvert's Cliffs case feel that it is incumbent upon us, Mr. Chairman, to call to the attention of

C2Bt4

the Board and bring to the Board's attention the possibility that the Board may, if it desires, reaffirm its order to the Director of Regulation upon an analysis of the environmental impact or lack of environmental impact on the Indian Point 2 plant under the provisions of Section D-2 of Appendix D-2 CFR 50.

CHAIRMAN JENSCH: Based upon what evidence?

MR. KARMAN: Mr. Chairman, we will, hopefully within the next two or three days, or very shortly I should say, serve upon all parties and the Board, of course, determination on findings of any environmental impact on the Indian Point 2 plant, and we would suggest that as a matter of expedition that the Board take into consideration, assuming of course, that there are no contesting objections by any of the Intervenors, and make its determination on the submissions which we will serve upon the Board very, very shortly. And I assume that the Applicant will, if necessary and required by the Board, serve its findings on environmental impact.

CHAIRMAN JENSCH: I understand your statement about findings, but the findings have to be based on evidence.

Now is somebody going to adduce some evidence about this? You don't take them out of the air or in effect the air.

MR. KARMAN: This can be in the form of an affidavit or findings which the Board can take as evidence.

CHAIRMAN JENSCH: I have difficulty understanding

C2Bt5

your word findings. The findings must be based on evidence. What record of evidence is there about the environmental impact to which you would direct our attention?

MR. KARMAN: They will be incorporated in the findings themselves. This will be findings and determinations based upon evidence which the Director of Regulation of the Division of Reactor Licensing will include in its findings.

CHAIRMAN JENSCH: He will have some evidence, is that it?

MR. KARMAN: This is correct. This will be evidence submitted by the Regulatory Staff, Mr. Chairman.

CHAIRMAN JENSCH: Oh, the evidence will be submitted in this proceeding to the Board, that's correct.

MR. KARMAN: This is exactly what I indicated we would serve upon the Board.

CHAIRMAN JENSCH: Oh, I understood you were going to propose some findings.

MR. KARMAN: No, no, no. We were going to serve these findings and determinations on the Board.

CHAIRMAN JENSCH: Proceed. Thank you.

MR. KARMAN: This is all I have to say on this, Mr. Chairman. We feel that there can be no--if the Board so desires, no dragging of this and that the schedules could very well be met at the time that the prerequisite completion of the plant is done for the fuel loading.

C2Bt6

CHAIRMAN JENSCH: Well, we understood in July that this was a very urgent matter and it's my recollection that we suspended these proceedings to rush to Washington to give consideration to the motion for fuel loading and an order was issued in response to that urgent request. What happened during August? Was it still some of the matters to which Mr. Madsen referred, that they didn't have the components ready and that sort of thing?

MR. KARMAN: That is correct, Mr. Chairman. The only thing that's been holding up the issuance of this fuel loading license has been the position of these compliance matters.

MR. TROSTEN: Mr. Chairman, may I comment on what Mr. Karman has said?

First, let me say this, Mr. Chairman, that the Applicant has been bending every effort to complete the work which is a prerequisite to the Staff's issuing the fuel loading license. We have unfortunately not been able to complete this work as expeditiously as we had hoped. It has taken us more time than we had hoped and that we had expected in order to complete this work. But I can assure the Chairman that we are exceedingly conscious of the need to complete this work as quickly as we can, and we are bending every effort to get this work done, and we appreciate the fact that the Board acted expeditiously on our fuel loading motion, and we have been

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trying to act just as expeditiously as we can in this respect.

Now with regard to the matter of any environmental findings that have to be made under the Commissioner's new regulations, it was the Applicant's understanding until yesterday afternoon, Mr. Chairman, that in the event that any environmental finding had to be made prior to the time that a fuel loading authorization was issued that this finding would be made by the Regulatory Staff and that it would not be necessary for evidence to be introduced in this proceeding in order that such a finding be made.

In our judgment, Mr. Chairman, a finding by the Regulatory Staff concerning the environmental impact of fuel loading is entirely appropriate and completely consistent with the Commission's regulations.

Our fuel loading motion was made on June 18th and the Board issued its order on July 20th authorizing the Director of Regulation to make appropriate findings and to issue a fuel loading license. The Board's order was issued prior to the time that the Commission's revised Appendix D was promulgated on September 9, 1971. And in our view under the Commission's regulations there is simply no requirement that evidence be introduced in this proceeding and that the Board make any environmental finding. If an environmental finding has to be made in connection with the Staff's issuance of a fuel loading authorization, then it certainly appears

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clear to us that the Staff is empowered to make such a finding without referring this matter to the Board.

I have taken this position with the Staff, but the Staff counsel, as he has indicated, has now put this matter to the Board, if you will, Mr. Chairman, and sought the Board's view as to whether the Board requires evidence to be introduced in this proceeding, and the Board's view as to whether it must make some environmental finding confirming its authorization to the Director of Regulation to issue the fuel loading license.

Now I have stated our position on this, Mr. Chairman, and I hope the Chairman would agree with me.

On the other hand, if the Chairman feels in response to what the Staff counsel has said that somehow Appendix D does require the Board to make an environmental finding and to confirm its authorization to the Director of Regulation to issue the fuel loading license in view of the new revised Appendix D, then Applicant is prepared today, Mr. Chairman, to offer in evidence the same information which we have submitted to the Regulatory Staff by letter dated October 1, 1971, for its consideration in connection with the issuance of a fuel loading authorization.

We are prepared if the Chairman feels in response to the Staff counsel's inquiry that the Board must make some sort of a finding to distribute to the Board and the parties

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today, and I might add, Mr. Chairman, that we already distributed this information to counsel for the Intervenors, and that counsel for the Intervenors will say in response to this matter that they do not object to the issuance of this fuel loading authorization and to the issuance of the fuel loading authorization by the Regulatory Staff without referral to the Board; we are prepared, as I say, Mr. Chairman, to put on evidence today with a witness to support any findings that the Board feels that it has to make.

end

CHAIRMAN JENSCH: I think the endeavor of the parties here to confer among themselves respecting these simple matters is very helpful. The Board certainly encourages the parties to continue to confer on matters that will help expedite the proceeding in any way.

The Board would also suggest, however, that the fact that parties don't have objection doesn't mean that the Board may not call upon the parties to come forward with some comments and statements in this regard. This environmental matter is new, and I think that full consideration should be given to the revised Appendix D, part fifty of the regulations which the Commission has issued. The Board is anxious to conform to those regulations and the intention of those regulations to provide a consideration of the total environmental impact of any licensing action that may affect the environment.

In other words, I'm sure the Atomic Safety and Licensing Boards, generally, welcome and seek the assistance of the parties in reference to these simple matters.

Another Board with whom I'm familiar is not inclined to say, well, since there is no objection, therefore it can be done without evidence. Maybe at this time it may be well to hear from some of the parties respecting this matter at this time.

Hudson River Fishermen's Association.

MR. MAC BETH: Mr. Chairman, on behalf of the Hudson River Fishermen, I have had considerable conversations with Mr. Trosten in the last few weeks. The Hudson River Fishermen are not prepared to offer evidence on this question, and we will have no objection to the issuing of an authorization.

CHAIRMAN JENSCH: Is the reason that you are not objecting, is that you feel there is no environmental impact of the proposed licensing action to that extent?

MR. MAC BETH: There may be an environmental impact, Mr. Chairman, but we felt it is important to be able to work out some sort of a schedule that would allow us to present full evidence on this question before the final operating license is issued; that the Board make its determination on the issuance of the license. In order to do that, we have tried to work out a schedule with the Applicant. It would simply be impossible at this time to present full evidence on the possible effects of fuel loading.

CHAIRMAN JENSCH: Is your thought there will be no significant impact on the environment for this phase of licensing action at this time?

MR. MAC BETH: The Fishermen do not wish to take a position on this one way or another.

CHAIRMAN JENSCH: Why not?

MR. MAC BETH: We would like to be able to adhere to the scheduling arrangements we have tried to work out with the Applicant, and as part of that we are willing to simply not take a position on the question at this time. We would not object to the fuel-loading authorization.

CHAIRMAN JENSCH: The Board is seeking guidance on the environmental matters. We would like you folks to consider your position a little further and kindly say yes or no. Will there be any significant adverse impact on this fuel loading and testing, and request for fuel loading? Will you give consideration on that and let us know when you can?

MR. MAC BETH: Yes.

CHAIRMAN JENSCH: The New York State Atomic Energy Council.

MR. MARTIN: The Atomic Energy Council has no position on this.

CHAIRMAN JENSCH: Why not?

MR. MARTIN: We have not been consulted up to this point. However, we have no objection. The Council itself has not developed a position with respect to the environmental aspects. Our position at this point is with respect to the need. Our position is if the Board is satisfied, we have no objection.

CHAIRMAN JENSCH: I know in words I am going to use it is not the way you presented it. But let me suggest that there might be a possibility of this aspect being considered by others. We don't like to see the buck being passed to the Board without the parties standing up and being counted on these things. Will you consider further whether you can answer the question, will there really be any significant adverse impact on the environment by this fuel-loading request? Will you give consideration and advise the Board, if you can?

MR. MARTIN: Yes.

CHAIRMAN JENSCH: I suppose the counsel should address itself to the Calvert's Cliffs decision, and will he speak to the adverse impact on the environment?

MR. ROISMAN: As you know, I wear two hats, the Citizens' Committee for the Protection of the Environment and the Environmental Defense Fund. They have earlier in the proceedings taken the position that they will be concerned exclusively with matters of radiological safety, that is the Citizens' Committee for the Protection of the Environment, and with respect to that it was its position back in July that there was no radiological safety implications of the fuel loading. It simply is not technically qualified.

For purposes of this hearing, it will not submit any review. It is an area it is not choosing to operate in in this particular hearing.

With regard to the Environmental Defense Fund, the area is a bit more complex and perhaps I shall go into it a bit more.

We don't think it necessarily is a question of what is the meaning of the Calvert's Cliffs decision. I suspect that a literal meaning of that decision might support either side of this dispute between the Staff soul-searching and the Applicant's position. We do think, however, that in the Calvert's Cliffs decision there is mood which the Environmental Defense Fund and the Hudson River Fishermen's Association and the Applicant have attempted to comply with, the spirit, if you will, of the decision and not merely the letter of the law. That is that these environmental matters, when they come up as they did in this case, in the midst of things, as opposed to coming up as they will for new nuclear plants when they are beginning construction, do raise problems, conflicts between the need for the plant and the need to have environmental reviews.

We believe that if those problems can be worked out -- In other words, if some accommodation can be made to have a maximum ability to achieve both objectives, get a decision that will help the Applicant either find out that the particular plant isn't going to be licensed so it can pursue an alternative, or get a decision that it is going to be licensed so that they can proceed to utilize it, without infringing upon the important environmental considerations, that that should be done.

With that thought in mind, we have been having, over the last several weeks, and the Hudson River Fishermen's Association as well, extensive discussions with the Applicant, not in terms of what are the legal requirements of Appendix D or the Calvert's Cliffs decision, but what are the realities of the situation in which we find this nuclear power plant. We think that Appendix D certainly encompasses that idea.

For instance, the license that will be issued here. If it is being issued under 5057, it is being issued with the proviso in Section D-2 of Appendix D that says that its issuance will be without prejudice to subsequent consideration. I think a clear indication there in the regulation is that the issuance now may be made even though the parties may want to change their mind, may want to say later, we didn't really want to say that there was no adverse environmental effect of that particular thing you were doing. It was just that on the basis of the information that we had at that time, we thought that you ought to go ahead and do that when we went ahead and continued the study of the environmental matters. That is the position in which the Environmental Defense Fund finds itself today. We simply do not have enough information with the kind of position-taking that we would want to do on an environmental matter. The Applicant is aware of this matter and they are making every effort to provide us with documents, information, and access to the plant-testing records, and other things so we can conduct a thorough environmental review.

In the meantime we think it would be unreasonable of us with respect to the Applicant to lie down before its request for fuel loading and say that this particular fuel-loading request should not be granted until this environmental review is completed. In a way that is a question of settlement between the parties. We can't tell you a yes or no to the question that you asked. I suspect the Hudson River Fishermen's Association, with whom we are working very closely in the environmental review, are in very much the same position. I'm not sure the problem requires that kind of response from us. We want to adjust the time schedules here to permit the Applicant to do as much as possible without ultimately committing itself to this plant and operating this plant in exchange for the environmental groups having an opportunity to conduct a proper environmental review. If we don't do that, if something interferes with our opportunity to do that, then the result will be that instead of having both sides working towards a common goal -- And we do have a common goal.

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-- we want to resolve the question at a reasonably early time based upon a reasonably adequate record. That's a common goal we have with the Applicant and with the Staff.

But if our attempt to work out scheduling arrangements and our attempt to give a little even though if we were in a really hotly contested proceeding we wouldn't give it, if those break down then the Applicant is left with taking hard positions and we are left with taking hard positions. The net result is that we are going to have a very contemptuous proceeding where a contemptuous proceeding wouldn't necessarily have to follow. I understand the Board's problem, and that is that it has a special responsibility with regard to environmental matters. As the Court said, not really as an umpire merely calling balls and strikes. I think, though, the Board is entitled in performing that responsibility, to take into account the same factors that we are taking into account. That is the fact that the Applicant's fuel loading is in the context of the total operation of the nuclear power plant, a small factor, that if the parties are prepared to permit it to go ahead and if it is all done with the understanding that it is without prejudice to any subsequent action, that we are in much the same condition as we were in with regard to the original question of fuel loading on radiological safety, because it was an analogy that the Court in Calvert's

Cliffs was making to the radiological safety review, that it imposed upon the environmental review, no more than that radiological safety review.

In that context that the Board could, just on the basis of the submission which we have seen from the Applicant, refer this matter to the Director of Regulation, and say to the Director of Regulation, we think that this is just like when we referred to the radiological safety matter. In short, we find there is enough information here for you to make a judgment rather than the Board itself having to make a judgment and then refer it to the Director of Regulations to make, if you will, a new judgment on the matter.

As we will discuss later with regard to the whole scope of the environmental review and the way that we are trying to reach accommodations in this proceeding, the essence of what we are attempting to do is to permit the Applicant to take gradual steps leading toward operation of the plant while we are given an opportunity to conduct a full environmental review at the end of which time we may oppose the issuance of a license, propose that if the issues with certain modifications or conceivably support the issuance of a license. What we have attempted to/is not do let the proceedings degenerate into a technicalities kind of argument in which we all lose sight of what the principles are here. We would be as disappointed and as disturbed as the Applicant would be if after completing an environmental review it were our position that the plant should be licensed, only to find out that that review was completed at a time when the licensing of the plant would not meet the need for which the plant was originally designed.

What we are trying to do is to see to it that we can do our review and if the ultimate decision is a yes on the plant, that the plant can be ready to operate at the time that the Applicant wants it.

So just to summarize that and go back to the specific question, we really are not prepared to answer yes or no with regard to this question which you have asked, that is, will there be a significant adverse environmental effect. What we are prepared to say is, regardless of what that effect may be, we are prepared to support the issuance of the fuel loading and the no-power testing license application which the Applicant has put in on the ground, if you will, that there are superventing and overriding considerations which justify the issuance of the license, considerations which we think are implicit if not extrinsic in Appendix D, as written, and for that matter in the way that we would read the Calvert's Cliffs decision.

CHAIRMAN JENSCH: If I don't use the correct terms, I hope you will keep a caution in mind. For a fuel loading and no-power testing, there is a different environmental consideration than there would be in any power testing at several different levels. So in a sense, as I understand your statement, there may well be, from the evidence here, no significant adverse effect upon the environment for this type of fuel loading and no-power testing; is that correct?

MR. ROISMAN: No, not exactly, Mr. Chairman. Perhaps the plant situation is somewhat unique. The biggest environmental problem with the Indian Point Plant -- And this is no secret or anything. -- is the problem of killing fish in the river. That seems to be a problem associated only with the operation of the pump in the plant. Apparently fish have been killed whether the plant itself has been turned on or not. If the pumps are operating and the intake is working, there have been problems with fish. Those problems can occur and do occur with three, four, five, six pumps operating. We are not prepared to say at this time that the quantity of fish that might be killed, the type of fish that might be killed, the age of the fish that might be killed and the time of year in which the kill might occur is necessarily not a significant adverse effect.

The Applicant has indicated, frankly, in its submission with regard to this question of environmental impact of fuel loading, that there might be some fish killed associated with the operation of the pumps during a two-week period. We would not want to take a position that said we think it is not a significant impact on the environment. Particularly of later thorough analysis of the problem, we are in a position to say the amount of fish that might be killed was a significant number, 500 or 1000 or 10,000 a day of a certain type of species at a certain time of year. All we are prepared to say is that we would like to, if you will, waive on the subject, subject to the fact that Appendix D, paragraph D-2 indicates, we do so without prejudice to the issuance of the subsequent license. We think that that, without prejudice language, does not require that we say affirmatively now there is no significant adverse environmental effects. But in fact, we will do exactly what we are doing, and that is saying that we choose to withhold our opinion on that subject until we have had an opportunity to study the matter.

The matter is complicated, and in a way answering the question about whether or not there is a significant adverse environmental effect of the running of the pumps associated with fuel loading is not really any different than answering the question of whether there is a significant environmental effect of running the pumps generally when the plant is operating. The only difference is that in one case it is a two-week period and in the other case it is a forty-year period. It might very well be that if our people had an opportunity to study the matter, that maybe they will conclude that what happens in those two weeks really is critical. Maybe those two weeks involve something truly critical. We don't think so. I think it is extremely unlikely that that is so. But the Environmental Defense Fund and the Hudson River Fishermen's Association feels an obligation not to speak on the basis of a rather quick review of the matter and then later have to recant and say we were really wrong about that. I say again, I think Appendix D contemplates that that can be done. It is just that --

CHAIRMAN JENSCH: It may be done but it comes back to the question, is everybody saying you do it, but we are not going to tell you whether to do it or not? Is there anything significantly adverse? I think the Board is entitled to have the comments. You say it is complicated and you don't have the information. We don't want the Board to be sitting duck on this thing. Sooner or later the Board shouldn't have done it.

MR. KARMAN: Mr. Chairman --

MR. ROISMAN: Permit me to say it is not our position that this is something that the Board has to resolve. I think that in the face of a no opposition from the parties, that the Board is entitled to refer the matter to the Director of Regulation not with a stamp of approval but merely by saying to the Director of Regulation, when we have an uncontested proceeding such as we have on this one issue in this proceeding, then we believe that the Director of Regulation can adequately protect the public interest by conducting its own review since, in effect, that is exactly what is going to happen.

The Board is, for the most part, the arbitor of the contesting issues. Once the issue is no longer contested as would appear to be the case on the question of the fuel loading, then the Director of Regulation is entitled and has imposed upon him -- Because he too may not be an umpire in this. He too must ask for the public interest to review the matter much the same as if this entire licensing proceeding for the issuance of an operating license itself had been uncontested and no hearing had been requested.

Radiological safety and environment would have been reviewed by the Director of Regulation and the matter -- In fact, there wouldn't have been a Board and it certainly would not have come to the attention of the Board. We think that the Director of Regulation, as Staff counsel has indicated today, is perhaps the one that is passing the buck. We don't think that the Board has to be the one to make the decision in this case. No party is requesting the Board to do so. That is none of the parties except the Staff is requesting the Board to do so.

MR. KARMAN: Mr. Chairman --

CHAIRMAN JENSCH: I know you want to speak. There will come a time.

I am having difficulty reconciling your statements with the Calvert's Cliffs decision. This new loading situation regulation was drawn up in radiological matters and are paramount by the Atomic Energy Commission.

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To that extent, that that regulation has been in effect modified by the Calvert's Cliffs decision, may be open to some discussion. The Calvert's Cliffs decision indicated, as I recall, that the decisional group in a proceeding involving environmental matters at least must rely upon evidence. Any proceeding that purports to limit the review of a board in this proceeding would be in a sense making a mockery of the decisional process. If the parties are as ready to express their views, then the Board will take some kind of evidence into the record before a decision can be made. I don't know whether this fuel loading regulation that provides that the parties decide not to contest it, and it will be turned over to a Director of Regulation, who hasn't appeared in the proceeding, to state how he will formulate his judgment in that regard. I don't know that. I don't know whether the Commission has reviewed this particular regulation in the light of, in a sense, the mandate of the Court of Appeals on environmental matters. I would want to recess in a few minutes to give consideration to the several statements. I think since the Commission has made very substantial revisions in the regulation pertaining to environmental matters and has spoken about the total environmental impact, that if that be the contention of the Applicant for fuel loading, that some evidence that the Applicant has to present perhaps briefly on the record here, so the Board can give consideration to the matter and may well arrive at the same conclusion. It seems to me, as I read the Calvert's Cliffs decision, they expect a decisional group to rely on evidence presented in the public regard.

MR. KARMAN: Mr. Chairman, there has been several passing references to passing the buck. The Director of Regulation has certainly been acting upon this matter of fuel loading and the environmental impact, adverse environmental impact, and as I indicated before, I will within the next few days, hopefully, have my findings on this matter. We, as the Regulartory Staff, felt that under the tenor and the import and the regulation itself, that we owed a duty to the Board. We are not passing the buck to the Board under any circumstances. We are doing our work.

CHAIRMAN JENSCH: There is no suggestion that you are not, sir.

MR. KARMAN: Not from the Board, but we have had it from some passing references which might be interpreted in that respect. As I indicated before, we feel that when we submit this to the Board, that the Board can, after looking at our evidence and looking at the Applicant's evidence, return it, if it so desires, to the Director of Regulations to issue the license. We felt that before any such license was issued, we should call this to the attention of the Board.

MR. ROISMAN: Mr. Chairman, let me, if I may, just say that my reference to passing the buck has to do with the decision, not the work on it, the Staff's work. I have no problem with that. I also don't think that there is any objection to the Board taking the evidence which gets eventually referred to, to the Director of Regulation. I know the Applicant is prepared to present its case today, and I will hope, if not today, tomorrow. We have two days of hearing set aside here. That the Staff could come forward in orally, if not in writing, and present its evidence . We don't think the Board should be inhibited in waiting for evidence to come from the intervening parties in referring this matter to the Director of Regulation for his decision on the question.

If you will, to analogize it to a criminal situation, no, we are either pleading guilty or innocent with regard to the matter. We will have a great deal to say on the environmental impact of the plant. We are not prepared to do that now nor are we prepared to delay the issuance of the fuel loading license because we are not prepared to issue it. That is not to say that we will want to have the evidence that the Board has said that they would like to have from the Applicant and the Staff.

MR. TROSTEN: Mr. Chairman--

CHAIRMAN JENSCH: Yes, proceed.

MR. TROSTEN: May I just summarize Applicant's position on this matter?

CHAIRMAN JENSCH: If you want to do it after you present your evidence, the Board will be prepared to receive your evidence at this time.

MR. TROSTEN: We are prepared to present our evidence.

CHAIRMAN JENSCH: Call your witness.

We will understand the summary better when we hear what the evidence is. We may miss the evidence and get the summary.

MR. TROSTEN: Mr. Chairman, if the Board wishes, we are prepared to introduce evidence at this point in support of any finding which the Board considers that it must make, that there is no significant adverse impact on the quality of the environment from fuel loading. As I mentioned, Mr. Chairman, we feel the Staff is taking a super-conservative attitude toward the language of Appendix D, but we will produce the evidence nonetheless.

CHAIRMAN JENSCH: We want to commend the Staff for its presentation in this regard. It is entirely within the spirit of the Appendix D regulations. Will you proceed.

MR. TROSTEN: Thank you, Mr. Chairman.

Mr. Chairman, I have a document before me, copies of which will be distributed to the Board at this point, and copies made available to the reporter as well as to the other parties to the proceeding. It is entitled, "Testimony of William J. Cahill, Jr., Concerning Environmental effects of Fuel Loading and Subcritical Testing at Indian Point Unit No. 2, dated October 5, 1971."

This document, Mr. Chairman, was prepared by the Applicant and delivered to the Regulatory Staff on October the 1st, 1971, in support of any environmental finding which the Staff considered that it had to make prior to issuing the fuel loading license. The thrust of his testimony, Mr. Chairman, is that considering the basic potential environmental impact of fuel loading and subcritical testing, the conclusion that may be drawn from the facts presented herein is that there is no significant impact on the quality of the environment as the result of the activities in question.

Just for summary purposes, I will point out that his testimony discusses the radiological effects of fuel loading and subcritical testing in Section 3. Beginning on Page 4 there is a discussion of Thermal Discharges. In Section 4 there is a discussion of Chemical Discharges. Finally, in Section 5, there is a discussion of the Effects on Fish and Entrained Organisms of the activities in question.

I would now like to show this document to Mr. Cahill who has previously been sworn.

Mr. Cahill, was the document to which I have just referred prepared under your supervision and direction, and are the contents of this document true and correct to your own knowledge?

MR. CAHILL: Yes.

MR. TROSTEN: Do you desire to have this document received in evidence in this proceeding as your testimony?

MR. CAHILL: Yes.

MR. TROSTEN: Mr. Chairman, I hereby offer the testimony to which I have just referred in evidence in this proceeding in support of a request which I hereby make to the Board as an oral motion, that it confirm its authorization to the Director of Regulation to issue the fuel loading license. Such authorization having previously been given by the Board's order of July 20, 1971, on the basis of a showing herein that the activities in question will not have a significant adverse impact on the quality of the environment.

CHAIRMAN JENSCH: Is there any objection on behalf of the Regulatory Staff?

MR. KARMAN: No objection, Mr. Chairman.

CHAIRMAN JENSCH: Hudson River Fishermen's Association.

MR. MACBETH: No objection.

CHAIRMAN JENSCH: New York State Atomic Energy Council.

MR. MARTIN: No objection.

CHAIRMAN JENSCH: Citizens' Committee for the Protection of the Environment and the Environmental Defense Fund.

MR. TROSTEN: No objection.

MR. JENSCH: Very well. The request is granted and the previously-prepared statement of William J. Cahill, Jr., may be accepted as his testimony and received in evidence in this proceeding. If you have copies sufficient for the reporter, the reporter is directed to physically incorporate in the transcript the previously-referred statement to which Applicant's counsel has just referred. The Board will give consideration to the second request after review of the matter.

Does that complete the presentation of evidence on behalf of the Applicant?

MR. TROSTEN: Yes, Mr. Chairman, we have no further evidence at this point.

CHAIRMAN JENSCH: Perhaps this will be a convenient time for a recess. We will reconvene in this room at 10:30.

(A short recess is taken.)

end

BEFORE THE UNITED STATES
ATOMIC ENERGY COMMISSION

In the Matter of	)	
Consolidated Edison Company of New York, Inc.	)	Docket No. 50-247
(Indian Point Station, Unit No. 2)	)	

Testimony of William J. Cahill, Jr.
Concerning

ENVIRONMENTAL EFFECTS OF FUEL LOADING AND SUBCRITICAL TESTING AT INDIAN POINT UNIT NO. 2

October 5, 1971

ENVIRONMENTAL EFFECTS OF FUEL LOADING AND SUBCRITICAL TESTING AT INDIAN POINT UNIT NO. 2

1.0 GENERAL

The proposed fuel loading and subcritical testing consists basically of the following activities:

a) Loading fuel into the reactor vessel.

b) Closing the reactor vessel and establishing temperature and pressure in the reactor coolant system by operating the reactor coolant pumps. This involves only heat from the pumps in order to allow testing of the control rods and instrumentation under realistic conditions. No heat will be produced in the core at any time.

c) Testing and calibration of some of the instrumentation in the core and reactor coolant system.

d) Performing various tests on the control rods and control rod drives.

It is estimated that the activities discussed above will take from four to eight weeks to complete. Of that time period, from one to two weeks are expected to be spent in testing activities with the reactor coolant system at temperature and pressure.

The environmental effects of these activities are discussed in subsequent sections.

2.0 RADIOLOGICAL EFFECTS

2.1 Normal Operating Releases

Throughout the activities planned, the reactor will be maintained subcritical by a large margin and never allowed to achieve a self-sustaining chain reaction. Since the reactor cannot produce any power in the subcritical condition, fission products or other radioactivity will not be produced. There will, therefore, be no radioactive releases for the activities planned.

2.2 Accidents and Hypothetical Releases

As stated in Section 2.1, there will be no radioactivity produced during the proposed activities. Also, there will be no radioactivity present from prior activities since the fuel is new. As a result, all of the accidents postulated for this plant and previously analyzed (including the LOCA), are not of concern, and will not cause any radioactivity to be released, since none is available to be released. Containment and engineered safety features operation are not required to achieve this result.

In order to maintain the reactor in a subcritical condition and prevent the production of fission products in the core, the following two neutron absorbers will be used:

a) Boric acid dissolved in the water in the reactor coolant system.

b) Control rods inserted in the reactor core.

Since the reactor remains subcritical, even with all control rods removed, no accident involving the control rods can cause criticality. In any case, the control rods will not be moved from the core except during testing. During testing, the core will be maintained subcritical due to the boric acid.

Dilution of boric acid in the core is prevented by isolating the reactor from any potential sources of unborated water. This will be accomplished by either disconnecting pipes or closing valves and locking them closed. As a backup, the following precautions will also be taken:

a) A system to inject more boric acid into the reactor will always be available, if required. Boric acid concentration will be continually monitored.

b) The neutron flux in the reactor core, which is a measure of the amount by which the reactor is subcritical, will be monitored at all times.

c) Special Plant security measures will be established for these activities and an AEC licensed reactor operator will be on watch at all times.

In summary, the proposed core loading and subcritical testing activities present no radiological hazard to the environment.

3.0 THERMAL DISCHARGES

Due to the fact that the reactor itself will not be in operation the heat load to the circulating water will be minimal and due almost solely to the mechanical heat generated by the reactor coolant pumps. Plans currently call for operating either three or six of the circulating water pumps during the various phases of the fuel loading operations. However, other combinations may be operated during this period for test or checkout purposes. The pumps will be run intermittently, and it is estimated that the time they will be run in connection with the authorized activities will total about one to two weeks. The following table indicates the thermal contribution of pump operation:

Table 1 - Discharge Canal Temperature Increase(Δ T) From Indian Point Unit No. 2 Subcritical Testing

Number of Pumps In Operation*	Assumed Heat Load	ΔT (^{o}F)
6	20 Megawatts Thermal	0.16
3	20 Megawatts Thermal	0.32

* each pump capable of circulating 140,000 gpm

As indicated by the above Table, ΔT never exceeds 0.4^{o}F; hence, the impact of the thermal discharge to the river due to the fuel loading operation is insignificant. It should be noted, moreover, that the circulating water is discharged via the common discharge canal with Indian Point Unit No. 1, which itself discharges 280,000 gpm of heated water. The addition of 420,000 to 340,000 gpm of virtually unheated water from Indian Point Unit No. 2 fuel loading operations will substantially reduce the Δ T in the discharge canal resulting from heat being discharged from Indian Point Unit No. 1.

4.0 CHEMICAL DISCHARGES

During the proposed activities, water will be circulated throughout the reactor coolant system even though the reactor itself will not be critical. The usual water treatment procedures for the primary and secondary cooling water systems as well as the circulating water will be maintained for the duration of the fuel loading operation. Hence, chemical releases will be made to the Hudson River during this period which will correspond to those outlined in Section 2.3.4 of the supplemental environmental report for Unit No. 2. Under all circumstances and modes of operation (long-term operation or short-term testing) the concentration of chemicals at confluence with the Hudson River will be maintained below the concentrations as given in Table 2.3.3 of the supplemental environmental report. Hence, as in the case of full power operation, chemical releases to the environment during fuel loading operations should have negligible environmental impact.

3.0 EFFECTS ON FISH AND ENTRAINED ORGANISMS

A. Fish Diversion

As described in Section 2.3.6.2 of Con Edison's Supplemental Environmental Report filed on September 9, 1971, Indian Point Unit No. 1 has experienced problems of fish impingement with its cooling water intake. Because of a number of changes which have been or are being made in the Unit No. 2 intake structure and which are described in that report, there is reason to expect substantial improvement over Unit No. 1 experience. Nevertheless, if one assumes that there will be no such improvement, it could be predicted that as many as 500 fish per day would be collected at the intake screens of Unit No. 2 with three pumps operating at full flow, or 1000 fish per day with six pumps operating at full flow. This prediction is based upon a one to two week period of pump operation in the fall, with flow and velocity differences from Unit No. 1 taken into account. The numbers of fish collected on the screens at Unit No. 1 on a daily basis have been highly variable, making prediction difficult. The numbers for Unit No. 2 could on a given day be significantly higher than indicated above, perhaps as much as by a factor of 10.

Based on Unit No. 1 experience at the same time of year, the fish impinged would be predicted to consist primarily of white perch (estimated 30%), herrings (blueback herring, primarily, estimated 30%), anchovy (estimated 20%), striped bass (estimated 4%), tomcod (estimated 5%), and others (estimated 11%). The blueback herring and white perch will be primarily young-of-the-year fish 1 1/2 inch to 2 3/4 inches long. The anchovy will be young fish and adults 1 1/8 inches to 4 inches long. Few fish of any species will exceed 5 inches long. The weight of these fish would range from .2 ounces to .5 ounces.

It is Con Edison's judgment that operation of the pumps during the fuel loading and subcritical testing operations will not have a significant adverse effect on the fish population of the Hudson River, on the basis of studies referred to in Section 2.3.6 of the supplemental environmental report, and particularly taking into account the length of time the pumps will be operating. As noted above, only one to two weeks of pump operation is anticipated in connection with the authorized activities. It should be noted that intermittent testing of the circulating water pumps is being and will continue to be performed as a normal part of the construction and testing schedule prior to licensing, and operation of the pumps during subcritical testing will be little different.

All these pump operations will be of value in evaluating any fish impingement problem at Unit No. 2 and in developing designs to minimize it.

B. <u>Effects on entrained organisms at Indian Point Unit No. 2.</u>

The expected temperature rise during passage through the condensers is given in Section 3.0 above. The expected ΔT is less than 0.4°F for any of the potential operating modes. No detrimental thermal shock effect is expected for the temperature rises predicted. Preliminary results of a current study of entrainment effects at much higher ΔT's indicates no mortality to zooplankton due to condenser passage, but mortality (not yet quantified) to some fish larva. Very few fish larvae are present in the river in the fall of the year and therefore, no significant effect will occur. Phytoplankton are not expected to be affected by the predicted ΔT.

After passing through the condensers at Indian Point Unit No. 2, the water will mix with the heated effluent from Indian Point Unit No. 1 in the discharge canal. The organisms carried through the condensers of Indian Point No. 2 will be exposed to a range of temperature increases depending on the flow at Indian Point Unit No. 2. The range will be from approximately 3.2°F to 9°F. The preliminary studies mentioned above indicate that no mortality for zooplankton or phytoplankton is expected from exposure to heat in the discharge canal. The exposure in the canal will not be instantaneous as in the condenser, but more gradual as the water from the two units mix.

Based on the studies mentioned, no significant adverse effect on the river biota is expected from entrainment effect.

E1Bt1

THE CHAIRMAN: Please come to order. Will everybody take their seats, please, and will you cease your talking so we may proceed with the interrogation of the witnesses and the discussion with the attorneys.

Does any party desire to ask any clarifying questions of witness Cahill, who is here, and having been previously sworn is available for an interrogation?

Hudson River Fishermen's Association?

MR. MAC BETH: No, Mr. Chairman.

CHAIRMAN JENSCH: New York State Atomic Energy Council?

MR. MARDIN: No, Mr. Chairman.

CHAIRMAN JENSCH: Citizens' Fund for the Protection of the Environment or the Environmental Defense Fund.

MR. RCISMAN: No, Mr. Chairman.

CHAIRMAN JENSCH: Let me inquire of the Regulatory Staff. You expect, as I understand your statement, to have a statment on the environmental impact, if any, in reference to this proposed or sought fuel loading and no power testing, is that correct?

MR. KARMAN: That is correct, Mr. Chairman.

CHAIRMAN JENSCH: Will your statement comment also upon the statement submitted by witness Cahill?

MR. KARMAN: Yes, it will take it into consideration, yes, Mr. Chairman. That statement had previously been

E1Bt2

furnished to the Regulatory Staff and is being used as part of the materials with which the Regulatory Staff is making its examination.

CHAIRMAN JENSCH: Is there any objection to receiving a statement from the staff of the character just indicated by Regulatory Staff counsel and being accepted upon the basis that if a witness were sworn he would testify as the statement will reflect? Is there any objection to having that?

MR. TROSTEN: No, Mr. Chairman.

CHAIRMAN JENSCH: The Hudson River Fishermen's Association?

MR. MACBETH: No objection.

CHAIRMAN JENSCH: New York State Atomic Energy Council?

MR. MARTIN: No objection.

CHAIRMAN JENSCH: The Citizens' Committee for the Protection of the Environment or the Environmental Defense Fund?

MR. ROISMAN: No objection.

CHAIRMAN JENSCH: The Board will accept this statement from the Staff upon the basis indicated by the parties here, will consider this statement as of an evidentiary character. The physical incorporation of the statement can be arranged into the record at a later time when the Board will proceed to consider it as evidentiary in this respect.

ElBt3

MR. KARMAN: Thank you, Mr. Chairman.

MR. TROSTEN: Mr. Chairman, do I understand following the receipt by the Board of this statement of the Regulatory Staff, which will be received in evidence at this proceeding, that the Board will then consider and act upon this as on the papers but without further hearing sessions?

CHAIRMAN JENSCH: That is the present intention of the Board, unless there is something startlingly different than anticipated or as indicated by the Regulatory Staff.

MR. TROSTEN: Thank you, Mr. Chairman.

CHAIRMAN JENSCH: And the Board will issue a further order in this regard.

Before we leave Mr. Cahill, however, though, I would like to ask a question or two about his statement, if I may. Perhaps the information is in your statement, Mr. Cahill, but in my hurried reading I may have missed it and therefore if you duplicate in part I hope you will excuse the interrogation because I have not had your statement.

MR. CAHILL: Yes, sir.

CHAIRMAN JENSCH: Until this morning.

During this fuel loading and no-power testing how many pumps will you be operating?

MR. CAHILL: Sir, there are six pumps. We will operate generally between three and six, but we might operate any combination of pumps. There is virtually no heat to be

E1Bt4

disposed of other than the heat generated by the primary coolant pumps, and this heat is expressed as steam, which is condensed in the condensors and there is in effect a drop in temperature. We will use the circulating water pumps to dispose of that heat to the river and at the same time whether or not we are involved in this testing we from time to time, for reasons of completion of construction or for testing purposes to determine the characteristics of the intakes with regard to fish entrainment would be operating these pumps. We have operated them, we will continue to operate them, and in that context we might be running one pump or two or all the way up to six.

CHAIRMAN JENSCH: Well, you won't be operating six all the time every day?

MR. CAHILL: No, sir.

CHAIRMAN JENSCH: And whatever you do will depend upon the necessities of your testing, is that correct?

MR. CAHILL: Yes.

MR. BRIGGS: Mr. Cahill, you point out that you may be operating three pumps or six pumps. Is it really necessary to operate three pumps in order to dispose of the heat during this zero power testing?

MR. CAHILL: No, it is not necessary simply to dispose of the heat.

MR. BRIGGS: You indicate that you will operate the

ElBt5

pumps for maybe two weeks. This would be continuous operation for two weeks, is that right?

MR. CAHILL: Well, it would be intermittent operation with some periods of continuous operation. As we are running the flow characteristic test whenever the primary coolant system is in continuous operation, then the circulating water system would be in continuous operation. But this may be several days within a day or two of no operation for this two-week period.

MR. BRIGGS: Well, let me ask you why don't you operate just one pump in the cooling water system during this period? Why pick three or six?

MR. CAHILL: Well, simply that coincident with fuel loading, but if we were not fuel loading, we may also be running these pumps. I didn't want to imply that we would only be running one pump. We may be running any combination, and this is the intention here, is to have full exposure of and information submitted as to the conditions we expect to have.

MR. BRIGGS: But the operation of these pumps is for the purpose of testing the pumps and the circulating water system rather than for the purposes of providing primarily heat removal during the loading operation or testing operation following loading.

MR. CAHILL: For this small amount of heat we wouldn't need three pumps, or certainly not six pumps.

E1Bt6

DR. GEYER: Yes, Mr. Chairman, I'd like to ask Mr. Cahill about the fish that are collected. Are they all killed or are they flushed back into the river? What becomes of them?

MR. CAHILL: Well, fish that are caught on the screens are, some of them are alive, but many of them and perhaps most of them are either dead or injured so that they would probably not survive. These fish are counted and recorded and returned to the river.

DR. GEYER: Do they present a problem of floating dead fish in the river?

MR. CAHILL: Not that I know of.

There have been periods when there were a lot of dead fish in the river. That was a general condition around the river. But--

DR. GEYER: Not attributable specifically to this operation?

MR. CAHILL: Not attributable to us and certainly not in the amounts and the quantities of these very small fish which are sardine size fish.

DR. GEYER: Perhaps something feeds on them, I guess?

MR. CAHILL: That may very well be.

DR. GEYER: Thank you, sir.

MR. BRIGGS: Mr. Cahill, have you been operating these pumps during the past month or so?

MR. CAHILL: I'm not sure. I'd have to check that.

E1Bt7

I know that there is some mechanical modifications being made to the pumps and I don't believe there has been any operation in the past month.

MR. BRIGGS: What I was wondering about was whether you have any counts on the number of fish that had been killed during that operation during the past month. How does it correspond to this number that you indicate like you can expect five hundred per day?

E2

MR. CAHILL: I would have to check and see if we have any counts and I could consult with someone there. The five hundred to a thousand is based on some experience with these pumps in past test runs.

CHAIRMAN JENSCH: Let me inquire, Mr. Cahill, as to your judgment about the operation proposed for low-power testing and fuel loading, that there will be no significant adverse effect. This is related to the operations proposed for Indian Point 2, and I presume Indian Point No. 1, is that correct? You mentioned--

MR. CAHILL: Indian Point 1 is presently in operation, yes.

CHAIRMAN JENSCH: And are you having this same fish problem there?

MR. TROSTEN: Excuse me, Mr. Chairman. I am not certain whether the witness understood your question. Were you asking whether his judgment related to the fuel loading for Indian Point 2 and the operation of Indian Point 1? Because this testimony of Mr. Cahill's relates only to fuel loading for Indian Point 2. It does not pertain to the Indian Point 1 plant.

CHAIRMAN JENSCH: Except that he had mentioned number 1, and that is on page 6, and that is why I asked him about it. And I assumed his judgment of no significant adverse effect was based upon the fact that likewise there had been nothing in his experience to indicate that there would be any significant adverse effect. Is that correct?

MR. CAHILL: That's correct.

CHAIRMAN JENSCH: And the heat situation, thermal-wise, can you put it in degrees? It's only going to come from the pumps as I understand it.

MR. CAHILL: It's only going to come from the pumps and the temperature rise at most is less than four-tenths of a degree Fahrenheit.

CHAIRMAN JENSCH: Are there any other figures of that kind that you can give, for instance, on the chemical releases that you mentioned?

MR. CAHILL: Well, the chemical releases are all made on the basis of measurement of the concentrated chemicals and then having dilution with the circulating water system these concentrations are kept below the standards established by the State of New York, well below that, and also below

E2Bt9

concentrations which through bioassay tests on selected species show no adverse effect on the species.

There is rarely for this very short period of operation of the primary and secondary situation any significant amount of chemical being discharged, and any that are will be discharged under careful monitoring.

CHAIRMAN JENSCH: Let me inquire. As I understand it this time proposed for this fule loading and no-power testing is something on the order of approximately two weeks, is that all?

MR. CAHILL: Well, the total fuel loading and testing program we estimate from four to eight weeks, of which some two weeks would involve generation of the heat from the primary coolant pumps.

CHAIRMAN JENSCH: That's the four-tenths of one per cent Fahrenheit?

MR. CAHILL: Yes.

CHAIRMAN JENSCH: To which you referred?

MR. CAHILL: Yes.

CHAIRMAN JENSCH: Well, outside of chemicals, thermal and fish problems, are there any other concerns from an environmental point of view reflected in your statement?

MR. CAHILL: Sir, in response to the scope of the environmental review, we have also in the new regulations, we have also gone back through the radiological discharge question, B2Bt10

and of course we are not making any radioactivity, and there is none to be discharged.

CHAIRMAN JENSCH: So that there is no concern at all from the radioactive point?

MR. CAHILL: There is no concern.

CHAIRMAN JENSCH: In other words, you are saying there is no concern, and I just repeat it because your voice was a little low.

MR. CAHILL: Excuse me.

CHAIRMAN JENSCH: Under this proposed operation of fuel loading and no-power testing there will be no release of any radioactivity, is that correct?

MR. CAHILL: That's correct.

CHAIRMAN JENSCH: Thank you for this purpose at least.

Will you proceed, Applicant's counsel.

MR. TROSTEN: Mr. Chairman, we have no further evidence to offer at this time and at this point our witnesses are panel witnesses.

CHAIRMAN JENSCH: It may be that we would like to propound a question or two to your panel a little later, but let us go forward on some other matters. Outside of that you have no further evidence?

MR. TROSTEN: Outside of that we have no further evidence and we are prepared if the Board wishes to discuss

E2Bt11

the procedural aspects of the hearing, if you wish to do that.

CHAIRMAN JENSCH: Let me go further.

Did you receive a letter, Mr. Martin? Did it arrive in time to suggest to you that if you could work out an agreement among counsel that your witnesses would testify?

MR. MARTIN: Yes, Mr. Chairman.

CHAIRMAN JENSCH: Do you desire to make an offer of evidence at this time?

MR. MARTIN: Yes, I do.

CHAIRMAN JENSCH: Will you proceed, please.

MR. MARTIN: On behalf of the Atomic Energy Council I would like at this time to submit two documents to become part of the transcript in this proceeding, the first entitled Supplementary Testimony of Sherwood Davies, Director, Bureau of Radiological Health, New York State Department of Health, dated September 15, 1971, and the second entitled Supplementary Testimony of Edward H. L. Smith, Assistant Director of Civil Defense Planning, Office of Natural Disaster and Civil Defense, New York State Department of Transportation, also dated September 15, 1971.

It has been stipulated by all counsel that if Mr. Davies and Mr. Smith were to be present upon this proceeding that they would testify as shown in these prepared statements offered at this time.

CHAIRMAN JENSCH: Very well. I understand that your

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stipulation also, therefore, necessarily includes no objection to the offer.

The proposed testimony as identified by The Atomic Energy Counsel may be received in evidence as supplementary evidence from witness Sherwood Davies and supplementary evidence from the witness Edward Smith, and if you have copies sufficient for the reporter, the reporter is directed to incorporate within the transcript at this place the statements now received into evidence from the witnesses Davies and Smith.

MR. MARTIN: I will supply the reporter with sufficient copies.

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SUPPLEMENTARY TESTIMONY OF SHERWOOD DAVIES,

DIRECTOR, BUREAU OF RADIOLOGICAL HEALTH,

NEW YORK STATE DEPARTMENT OF HEALTH

Dated: September 15, 1971

Q1. Is evacuation necessary to protect the residents of the low population zone from receiving doses in excess of 30 rem in base case situations?

A1. No. In the base case situation, the 2-hour, 8-hour, and 30-day doses at the house nearest to the facility are 21, 32 and 43 rem, respectively. Moving people indoors for the initial three to four hour period would effect an estimated dose savings of 50% during that time period. The projected thyroid exposures at the closest house would be reduced to approximately 11, 20 and 31 rem for the three time periods, and reduced to 6, 13 and 19 rem at the outer limit of the low population zone.

Q2. Would the residents of the low population zone be evacuated in any base case situation?

A2. Yes, but only if a substantial dose savings could be effected by evacuation, and evacuation could be carried out without resulting in greater risk than if the residents simply remained indoors. Moving indoors to reduce inhalation dose would be effective only for the first few hours after the accident. This time period is when the greatest dose savings are accomplished. For example, it is conceivable that evacuation might be ordered within the first few hours, to prevent exposure under favorable conditions, or after four hours to minimize further exposure, depending upon various factors including wind direction, time of day, weather, traffic and other conditions.

Q3. What instructions would you give the police in the event of evacuation of the residents of the low population zone and for how long a time would they be prevented from returning?

A3. The State Police would be instructed to evacuate the residents along an easterly route on Bleakley Avenue or a southerly route on Broadway, depending upon wind direction. Routes would be selected to minimize exposure. The period of time before residents would be permitted to return depends upon the duration of releases and the projected man-rem savings by their continued absence.

Q4. To what extent would the information supplied by the nuclear facility operator immediately following an accident be used in determining what responsive actions should be taken?

A4. The information to be supplied by the operator includes the location, type and time of the accident, weather conditions, wind direction and speed, the status of the facility and safeguards, and an estimate of the radioactivity that has been or may be released. Based upon this information, the Bureau of Radiological Health will determine the need for immediate protective actions. If the estimates of offsite doses are within the base case, the protective actions to be taken are pre-planned. Should the estimates exceed the base case, the general procedures of the emergency plan will be followed.

After the initial actions are taken, calculated dose estimates including the results of actual field measurements will be used as the basis for any necessary modification of response actions, and for all subsequent decisions.

Q5. In response to the request of Mr. Roisman on July 21, 1971, (transcript page 1764), have you located any records made of time for responses to radiation incidents?

A5. Yes. I have located the following reports and have forwarded copies to Mr. Roisman:

1. Radiation incident - Consolidated Freightways Trailer Truck Terminal, Colonie, New York

2. New York Central Train Derailment, Irving, New York, Erie County

3. Railroad Accident Involving Radioactive Material, Township of Grand Island, Erie County, June 3, 1965

4. Reported Incident - Railroad Car, Buffalo, Erie County, September 22, 1966

5. Lost Radium at Columbia Memorial Hospital, Hudson, New York

6. Radiation Incident - Patchogue, New York

Q6. In response to question "h. P. 8a" on page 6 of Mr. Roisman's proposed cross-examination with respect to supplemtal direct testimony of the New York State Atomic Energy Council, is there a document that embodies the State's large scale general emergency response capacity?

A6. Yes. It is entitled "The Emergency Plan for the Civil Defense of the State of New York". A copy has been sent to Mr. Roisman.

SUPPLEMENTARY TESTIMONY OF EDWARD H.L. SMITH

ASSISTANT DIRECTOR OF CIVIL DEFENSE PLANNING,

OFFICE OF NATURAL DISASTER & CIVIL DEFENSE,

NEW YORK STATE DEPARTMENT OF TRANSPORTATION

DATED: September 15, 1971

The Health Department then calls back to the nuclear facility operator to determine the immediate impact of the accident. Experience indicates that this can be accomplished within 15 to 30 minutes.

The next stage is to relay Health Department instructions to police and local officials via the various emergency communication systems. These instructions will be based upon estimates of offsite doses provided by the nuclear facility operator and will be directed to the Division of State Police at Albany and the local Civil Defense Director and Disaster Coordinator. If warranted, the Division of State Police will notify its barracks in the area of the accident, which in turn will relay instructions to the local police. In a parallel sequence of communications, the local Civil Defense Directors and Disaster Coordinators will instruct the county executive, the chief executive of any municipality involved, the county highway department, other appropriate local officials, and also the local police. Fifteen minutes from the time the Health Department calls back the nuclear facility operator is ample time for it to give its instructions to the Emergency Operating Center and to commence the primary protective actions set forth in alert A.

After assessment of the accident by the Department of Health, instructions for subsequent protective actions will be communicated to the local police, local disaster coordinators and local government chief executives by means of the emergency disaster communications systems. These actions can be commenced within one to two hours from the verification of the accident. This estimated time span is well within the time required in past emergencies that occurred without warning. In the 1965 blackout, for example, assessments of the situation and emergency response actions commenced in less than one hour. In the 1964 ice storm in Albany, Schenectady and Saratoga Counties, the State Emergency Operating Center was manned and emergency response action commenced within 30 minutes after notification, notwithstanding that the duty officer's telephone was a casualty of the storm.

Q4. Describe the mechanism for notifying the local police, other local officials, and the affected public with respect to the pre-planned emergency actions.

A4. There is 24-hour direct communication between the State Emergency Operating Center and the warning centers at the Westchester County Parkway Police Department and at the Peekskill Police Department via the Civil Defense warning system known as the National Alert Warning System (NAWAS). This is a dedicated telecommunications line. In addition, the State Police Headquarters has direct land line communications links with its regional headquarters, which can in turn communicate by radio with mobile units and by telephone or messenger with police. The State Police Headquarters is located in the same building as the Emergency Operating Center. Communication of instructions to the State and local police can be made by either or both of these means. The local warning point at either the Westchester Parkway Police Department or the Peekskill Police Department can also notify the local Civil Defense Director and Disaster Coordinator by telephone or mobile radio or messenger. By the same means, the local Civil Defense Director and Disaster Coordinator will advise other county and local officials whose names and numbers are listed in the Westchester County emergency directory.

Members of the public present in the low population zone will receive instructions from the State Police, with the assistance of local police, including the Westchester Parkway Police, the Civil Defense Auxiliary Police and the Peekskill Police if warranted. Communication will be by either bullhorn announcement or by house-to-house messengers. These means of notice have proved effective in other types of emergencies, such as gas leaks and chemical explosions, in which the danger was more immediate than in a radiation accident. If the hazard

Q1. Are you familiar with the New York State Emergency Plan for Major Radiation Accidents Involving Nuclear Facilities?

A1. Yes.

Q2. What is your connection with the New York State Emergency Plan for Major Radiation Accidents Involving Nuclear Facilities?

A2. I participated in the development and preparation of the plan and am responsible for supervising its maintenance. I am also a senior staff member of the unit responsible for coordination of the actual operation of the plan in the event of an accident upon receipt of instructions from the Department of Health.

Q3. What is the basis for the estimate of time needed to accomplish emergency notifications and the other pre-planned emergency actions?

A3. Experience in responding to other local emergencies indicates that one to two hours from verification of the accident is a reasonable estimate of the time needed to carry out essential notifications and commence preliminary response procedures for an accident of the type covered by the plan.

The Office of Natural Disaster and Civil Defense maintains a 24-hour warning point operated by the Division of State Police. The office has designated "on-call" senior personnel, who in the past have made initial response to storms, explosions and similar disasters within 10 to 15 minutes. The initial response is to determine by telephone the immediate impact of the disaster. Under the State's plan for major radiation accidents, in this initial stage the warning point receives the call from the nuclear facility operator, then notifies the Health Department representative, and the senior Civil Defense personnel on call.

potentially were to affect a broader area than the low population zone, then instructions could be broadcast via radio and television as well.

Q5. Describe the extent to which the emergency procedures to be undertaken in the event of a major radiation accident have been communicated to and discussed with the various State and local officials with potential responsibility under the plan.

A5. State and local agencies potentially responsible for action under the plan have received copies of the State's Emergency Plan for Major Radiation Accidents. In the State government this includes the Departments of Health, Education, Transportation, Environmental Conservation, Labor, Commerce and Agriculture and Markets, the Divisions of State Police and Military and Naval Affairs, the State Office for Local Government, and the Public Service Commission. In local government, this includes the Westchester Parkway and City of Peekskill Police Departments, the City of Peekskill Fire Department, the Westchester County Health Department, the Westchester County and City of Peekskill Civil Defense Directors and Disaster Coordinators and the New York City Department of Health, Office of Radiation Control. Most of these agencies participated in the developmental stages of the plans by indicating their response capabilities and the procedures they would follow if called upon to participate in its execution. The plan, therefore, is a joint statement by agencies with primary emergency responsibility and portrays anticipated actual responses based on what each agency has indicated is its response capability and the procedures it will follow in responding.

Q6. Is it necessary or desirable to provide the public with advance instructions or information about the plan for it to be effective?

A6. No, it is neither necessary nor desirable. The many variable factors involved in determining what instructions should be given to the public make it both impractical and undesirable to instruct in advance. Among the factors to be considered are the amount of radioactivity released, wind direction, traffic conditions, the time of the year and weather conditions. Any advance instructions would have to be stated as several alternative courses of action to be taken, depending upon which combination of factors existed at the time of the accident. Instructions of this nature are necessarily complicated, difficult to follow and confusing to the public. Also, because of the many variables involved, the actual emergency actions determined to be appropriate at the time of the accident might vary substantially from, or even contradict, the set of advance instructions, creating a high probability that some persons would take the wrong action, exposing them to an unnecessary risk.

Q7. If it were decided that evacuation of the low population zone was desirable to effect significant dose savings, how would residents be evacuated?

A7. The State Police would direct the evacuation, with assistance available from the Westchester County Police, Buchanan Police, and Civil Defense Auxiliary Police, if required. With so few houses involved, the residents would be instructed either by bullhorn or receive direct verbal instructions from a police officer. Evacuation would be routed over Bleakley Avenue or Broadway, depending upon the pattern of the plume. Because the zone is small in area, there would be no need for remedial movement within it.

Residents would either walk or use personal autos to leave the zone. Aged and infirm persons would be transported by police car. The instructions given would depend upon the circumstances existing at the time of evacuation, but basically would be to evacuate in a given direction over a specified route, and if on foot, to walk to a designated area to await the arrival of public transportation. Residents with autos would be told of available shelters set up to receive them.

Evacuation would be accomplished by six to eight State Police officers from the 30-man force at the local substation, with the assistance of seven to nine local or Civil Defense Auxiliary Police officers. Because fewer than 100 persons would be moved, evacuation could be completed within 45 to 60 minutes.

Q8. If it were decided to evacuate areas beyond the low population zone because of an accident exceeding the base case, how would such an evacuation be undertaken?

A8. The State has no pre-planned action for an accident beyond the base case. Evacuation of any larger area, however, would follow the same general procedure as evacuation of the low population zone, depending upon the size of the area and the number of persons to be moved. If a large number of people were involved, such as the population of the City of Peekskill, instructions also would be given by radio and television broadcast. More police personnel might be needed, the number depending upon the scope of the evacuation. In addition to the State Police, there are over 2,300 local police officers in the various units of government in Westchester County and an additional 1,300 local Civil Defense Auxiliary Police. The direction of evacuation would depend upon where the danger area is located, but any great population movement would be directed toward Route 9 and the Taconic Parkway. Arrangements would be made in neighboring communities or in New York City for shelter for evacuees. Food would be provided from the New York City area. Supply and transport would be through the normal means of food distribution by food processors and transporters.

Q9. Describe the State's large-scale emergency response capabilities and the time needed to commence responses under that plan.

A9. The State maintains an Emergency Plan for the Civil Defense of the State of New York to be placed in operation should the State suffer a nuclear attack. In the event of a major radiation accident endangering many persons over a large area, all applicable portions of the nuclear attack response plan can be implemented. In the event of such a catastrophe, Section 10 of the New York Executive Law empowers the Governor to call upon all State agencies to use their resources to provide aid to areas affected.

In such an emergency, the Division of State Police would maintain communications with the various governmental and police agencies in the area affected, in addition to its basic functions of controlling traffic and the movement of people. The Department of Social Services, together with the Red Cross and the Division of Military and Naval Affairs would provide housing in hotels and armories and would distribute food. Food would be obtained by the Department of Agriculture and Markets from the United States Department of Agriculture stores of surplus food, as well as through normal food distribution channels. The Health Department would provide medical assistance, and, together with the Department of Environmental Conservation, would furnish technical personnel and instrumentation for radiation measurement. The State Office for Local Government would coordinate fire protection activities. The Division of Military and Naval Affairs would provide troops. The Department of Transportation would provide heavy equipment and would organize and supervise the clearance of highways, in addition to coordinating all State emergency activities through its Office of Natural Disaster and Civil Defense. The latter office would also serve as the channel for obtaining aid and assistance from municipalities throughout the State and from the Federal government.

By utilizing the response capabilities of communities in the area affected, medical assistance, traffic control and population movement could be commenced within one to two hours from the request therefor. Immediate action would be taken by the State Police, local police, local Civil Defense Directors and Disaster Coordinators, and the local government chief executives. Subsequent action such as clearing transportation routes or furnishing food and lodging would be commenced within 24 hours. The State has not experienced a large-scale emergency requiring the nuclear attack response plan to be put into operation. Periodically, however, the plan has been put in effect on a test basis. In each case, the State Emergency Operating Center was manned and operational in less than 20 minutes. On April 1, 1966, moreover, a surprise test was conducted in which the State Emergency Operating Center was manned and operational within 20 minutes, and manned with sufficient personnel to deal with the consequences of a nuclear attack in less than three hours.

UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

In the Matter of)
)
CONSOLIDATED EDISON COMPANY)
OF NEW YORK (Indian Point)
Unit No. 2))

Docket No. 50-247

CERTIFICATE OF SERVICE

I hereby certify that copies of (1) "Supplementary Testimony of Edward H.L. Smith" dated September 15, 1971; (2) "Supplementary Testimony of Sherwood Davies" dated September 15, 1971; and (3) a letter of transmittal to the Board dated September 20, 1971, have been served upon the following by deposit in the United States mail, first class or airmail, on September 20, 1971:

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E2Bt13

CHAIRMAN JENSCH: Very well. Somewhere in accordance with the correspondence which has passed among the parties, copies of which we have received, I inferred that no party was ready or desirous at this time to interrogate in reference to this evidence, so that matter would be deferred to our November session, is that correct?

MR. ROISMAN: That's correct, Mr. Chairman.

MR. KARMAN: That's correct.

CHAIRMAN JENSCH: Applicant?

MR. TROSTEN: That's correct, Mr. Chairman.

CHAIRMAN JENSCH: Very well. That concludes your presentation of evidence, is that correct?

MR. MARTIN: Yes, Mr. Chairman.

CHAIRMAN JENSCH: Does the Regulatory Staff have any evidence to offer?

MR. KARMAN: Mr. Chairman, possibly to expedite the hearings in November I would like to indicate that on September 3rd, 1971, I forwarded to the Board and to all parties to this proceeding a document entitled <u>Supplement Number 3 to Safety Evaluation by the Division of Reactor Licensing, U.S. Atomic Energy Commission, in the matter of Consolidated Edison Company of New York, Inc., Indian Point Nuclear Generating Unit Number 2, Docket Number 50247</u>.

Mr. Chairman, this relates to the emergency core cooling system and will be taken up at our November session.

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I have obtained a stipulation from all parties to this proceeding which would agree to allowing this document to become Staff Exhibit Number 4 with the stipulation that had our witnesses been present today they would testify as indicated in this safety evaluation.

CHAIRMAN JENSCH: It is your desire that that be marked as an exhibit or rather be incorporated in the transcript as evidence? My recollection was that your original safety evaluation was incorporated within the transcript in order to be more generally available.

MR. KARMAN: That would be preferred, Mr. Chairman, if the original was, and I believe you are correct. The problem is I will have to furnish sufficient copies to be included.

CHAIRMAN JENSCH: You may.

MR. KARMAN: Thank you, Mr. Chairman.

CHAIRMAN JENSCH: With that stipulation from all parties these documents which the Regulatory Staff counsel has referred to, September 3 Staff safety evaluation may be accepted as evidence from the Regulatory Staff and the reporter is directed to physically incorporate that supplement within the transcript at this place as the evidence from the Regulatory Staff, and likewise it's my understanding that the stipulation encompasses the schedule that cross-examination reference will be undertaken at the November session.

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Is there anything further from the staff?

MR. KARMAN: That's all at this time, Mr. Chairman.

CHAIRMAN JENSCH: That I take it then constitutes all the submittal of evidence that we can receive and give consideration to at this session, is that correct?

MR. TROSTEN: That's correct.

CHAIRMAN JENSCH: Does that then leave matters of procedure for consideration? I believe, Applicant, you were going to speak to that matter.

MR. TROSTEN: Yes, Mr. Chairman.

But first I would like to address myself to the matter of the continuation of the radiological safety aspects of the proceeding. We have discussed this with Mr. Roisman and it is our understanding that Mr. Roisman will furnish to the Applicant, Board and the other parties all cross-examination by October 20th. It is further my understanding that Mr. Roisman will give consideration to the need for any direct testimony and will advise applicant and the parties by that date as to whether any direct testimony, as well as cross-examination, will be offered at the November 1st session.

CHAIRMAN JENSCH: Excuse me. May I understand your last statment?

MR. TROSTEN: Yes.

CHAIRMAN JENSCH: In other words, he will furnish a request to you to adduce further direct testimony, is that correct?

MR. TROSTEN: No. I am referring to direct testimony, if any, from the Intervenors.

CHAIRMAN JENSCH: From the Intervenors, very well.

MR. TROSTEN: Correct, Mr. Roisman?

MR. ROISMAN: Yes, sir.

MR. TROSTEN: Further, I understand that Mr. Roisman will have available by the 25th of October a trial memorandum similar to the one that he produced with respect to other issues in this proceeding dealing with the emergency core cooling system matter and indeed all other matters to be encompassed in the November 1st session of the hearing.

I would like to say also that we are in the process of providing information informally to Mr. Roisman in response to his questions concerning emergency core cooling system matters. We have received one set of questions from Mr. Roisman and I understand we will be receiving more. We are going to answer these questions as rapidly as we can, those questions that are addressed to us. There are some of these questions that are addressed to the Staff. I agree with Mr. Roisman that providing responses informally to these questions will tend to expedite the hearing commencing on November 1st, and in that spirit we are going to try to answer as many of these as we can. However, it is my understanding that in the event we are unable to complete the answers to any of these questions we nevertheless will simply go forward on November 1st although this may involve a somewhat more lengthy session that would otherwise be the case.

CHAIRMAN JENSCH: How long is more lengthy? More lengthy than what?

MR. TROSTEN: Well, Mr. Chairman, I think Mr. Roisman would probably prefer to speak to that.

MR. ROISMAN: Mr. Chairman, to answer the question I don't know if the Board has had an opportunity to look at the questions which we have asked. We will be asking others, of equal detail. The amount of time that will be required if they are not answered in advance will be that as we get answers if those answers themselves upon further study indicate that we want more information that isn't readily available at the hearing we may then ask that information be presented, and we don't know how long the Applicant or the Staff will take to get that information. We think that the questions that have been put are, if you will, sort of first-line questions. They themselves will produce information which may result in a second line of questioning, hopefully the cross-examination. But that information obviously has to be studied.

It's my plan that as we go through the hearings and we are dealing with the ECCS questions that we will be asking questions during the day, studying that testimony in the evening and in many cases recalling the same witnesses the following day to further interrogate them with regard, in effect, to their previous day's testimony. I am not suggesting and I know Mr. Trosten is aware of this, that I am going to be able to conjure up all of the new lines of questioning that information that I am merely hearing for the first time might warrant right on the spot during cross-examination. I hope that it isn't going to mean even as much as several days of delay, but that the witnesses will be -- They know what we are concerned about. We have told them that in a statement of issues which went even before that letter that listed the detailed questions. They know the problems that we have with core geometry, the problems that we have with the tests and analyses that have been conducted, and I suspect that if they come prepared to open their hearts to us and the Board on the question of the inadequacies of present testing measures and their basis for believing that they made conservative assumptions to compensate for them that we won't have more than maybe an extra two days of hearing as a result of getting testimony at the last minute. But I should say, and I wish to stress this, that in our previous experience we found that the Staff witnesses have been more helpful for us, more frank and open and willing to discuss problems that they see than have the witnesses from Westinghouse, which perhaps is not illogical, giving the variant responsibilities of the two organizations.

CHAIRMAN JENSCH: Can I interrupt for a moment? I think the Commission at one time indicated that in one sense the hearings should not be scheduled until all discovery proceedings have been completed and the Board set this proceeding for November 1 with the hope that all discovery will have been completed by that time and that we will sit continuously to dispose of it.

I hope this does not mean any further sessions in regard to these matters, leaving open the environmental matters related to duration.

MR. ROISMAN: No, Mr. Chairman. Let me say that in effect the principle that we the Citizens' Committee will be operating under is that the cross-examination is always, of course, in the nature of a form of discovery and that we will simply be burning the midnight oil, provided the Applicant doesn't expect us to have hearings at night, to analyze and digest the information that we are getting on cross-examination and to just proceed further the next day. We do not anticipate, as far as our part of the case is concerned, any request that the hearings be adjourned to a later date other than adjourning on a Friday and coming back on a Tuesday morning. It is not in any way our intent or our belief that we cannot dispose of all radiological safety matters with continuous hearings beginning on November 1 and ending, at this time I would expect, within two weeks; in other words, eight hearing days, Tuesday through Friday of two weeks, should do it unless something really incredible comes up.

But what I was going to say is that the Staff's answers to these questions and their contribution in it is very important to us, and the earlier that we have that the better it will be for to make our presentation.

CHAIRMAN JENSCH: Is it going to be of any help if we put this off for one week? I almost got a shudder from the Applicant's counsel. Because I think he has been hopeful that there will be no further postponements. But the Board is not inclined to take hearing days for matters of discovery or discussion with the Staff that might eliminate many of the matters that would be propounded at a hearing.

MR. TROSTEN: Mr. Chairman, may I just offer a clarification here. I think Mr. Roisman will certainly agree that his questions are not in the nature of discovery in the sense of the Chairman's questions. These are in the nature of preliminary cross-examination and what he is doing as I understand it is simply asking those questions which he would ask if the hearing were actually going on, and we are trying to answer those questions and we expect that we will be discussing this with Mr. Roisman between now and then in some detail in an effort to eliminate those questions which simply would have to asked at the hearing if the hearing were to be held today, so the discovery process really in the sense in which the Chairman has asked the question is really over, Mr. Chairman. That's the reason why I feel that we are going to be able to proceed smoothly into this hearing session on December 1st.

MR. ROISMAN: Mr. Chairman, I think I can basically agree with that. The Applicant and the Staff have been most helpful in supplying documents and we have been given this list of documents the emergency core cooling system task force used and have received within days of request either on the telephone or by writing a copy of the document that we have asked for, and our people are able to study them and I think Mr. Trosten does characterize the questions as being closer to cross-examination questions than they are to what you would call discovery questions. We, as you know our philosophy has been that this isn't a Perry Mason trial where we will surprise the witness with a piercing question that will cause him to break down on the witness stand. We want the Applicant to know and the Staff to know well in advance what our concerns are and if they have got adequate explanations to alleviate those concerns we are most anxious to hear them, and if they don't we want them to know what we think our problems are.

One aspect of all of this is that they may decide to recant on their support of the plant after realizing that we have probated an error that they haven't originally considered.

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CHAIRMAN JENSCH: You don't care when they break down, I understand, or otherwise.

MR. ROISMAN: We are prepared to accept a breakdown at any time, Mr. Chairman.

CHAIRMAN JENSCH: I think the observation from the many reactor proceedings that were before the Commission, I thinkthat this reflects the greatest of cooperation among the parties that I have witnessed before the Commission. I think the Applicant has well undertaken a lot of the so-called extra effort in order to provide information. From the review I have had from these documents, I think it contributes substantially to expedite this proceeding.

Proceed, counsel. Do you think two weeks will do it?

MR. ROISMAN: Yes, Mr. Chairman, I would think it would do it.

MR. JENSCH: The Regulatory Staff, do you have any suggestions for procedure or anticipations for time involved?

MR. KARMAN: No, Mr. Chairman. We feel as does the Applicant and the Intervenor, that the case probably can be completed, the radiological aspect of it, within that time frame. We have received questions from Mr. Roisman somewhat in the nature of cross-examination which we are studying and evaluating, and we will get the answers to him as quickly as possible.

CHAIRMAN JENSCH: The Board will adhere to its schedule of meeting here on November 1. The Board has had some advance time to make these preparations and is making itself available to proceed continuously. We hope it will be possible to adhere to that. Is there any other procedural matter to consider?

MR. MARTIN: Mr. Chairman, in behalf of New York State, I have discussed with counsel at a previous meeting, and I would like to ask the Board's permission, that the witnesses be produced by the State for cross-examination and be given permission to be put on at the end of the November session rather than at the beginning.

CHAIRMAN JENSCH: Very well. The Board will adjust to that schedule. I think the matter of timing and the call of a witness can better be left to arrangements among counsel. We will not specify a date for their appearance. I trust that you, Mr. Martin, will arrange with other counsel as to a time convenient for their call. The Board will be available to continue discussions when the time can be arranged in that regard.

Hudson River Fishermen's Association, do you have any information to give us about this subject? Do you plan to introduce direct evidence? Have you completed your discovery proceedings?

MR. MACBETH: We would have no evidence in the radiological hearings in November, Mr. Chairman.

CHAIRMAN JENSCH: Your presentation will be entirely on the environmental matter?

MR. MACBETH: Yes.

CHAIRMAN JENSCH: Very well.

Does that complete all the procedural consideration?

MR. TROSTEN: Yes, Mr. Chairman, excepting, of course, for the matter that I mentioned at the outset, that Mr. Roisman and Mr. Macbeth have been conducting, and I have been conducting several discussions concerning the conduct of the consideration of the limited operation motion which we filed on September 24, and also the conduct of the environmental hearing called for under Appendix D. We agreed that Mr. Roisman would present the general understanding to the Board.

CHAIRMAN JENSCH: Proceed.

MR. ROISMAN: Yes, Mr. Chairman.

As the Board mentioned only a couple of moments ago, this has been a proceeding which is at least unusual in the sense that the parties have been able to work together, I think, more than has been true in other contested proceedings, and yet the proceeding has remained clearly very contested.

The environmental review question has [illegible] raised several problems in this proceeding. As I understand it, from the Applicant's standpoint, they now have pending both before the Commission and before this Board motions requesting highly expedited proceeding for either the issuance of licenses to operate the plant pending the environmental review, or a highly expedited environmental review that would produce the issuance of a license in very short order.

The position of both the Hudson River Fishermen's Association and the Environmental Defense Fund is that the amount of time that would be required to adequately review the issuance of a license or to adequately review the environmental matters is too short as proposed by the Applicant in those motions. We toy with the idea simply of opposing them flat out and making this into a very acrimonious argument over scheduling, an acrimonious argument over the merits of the case. But at the Applicant's urging, we sat down and have discussed with them and have been in discussions with them for several weeks, to see if there wasn't some middle ground that would accommodate both sets of interests.

The principle that we hammered out--and I will discuss the details of that--was that we would go through a process of gradual compliance with the requirement of the environmental review; that the Applicant would permit a period of time which we felt was reasonable to conduct an adequate environmental review, and we would permit a certain amount--not opposed to a certain amount of no-power testing which we felt the environmental effects of were sufficiently small or were not going to be large enough that we would want to take a stand on them without being able to fully investigate it, as part of the same thing as we talked about on fuel loading.

The matter would then come to a much quicker hearing because we wouldn't be arguing, spending hearing days arguing with the procedures for the proceeding, but we would limit our argument to the things that we are both interested in, which are the merits. Consistent with that sort of a gradual compliance approach, we have agreed with the Applicant to support the application for a license to operate the plant for testing purposes only, up to fifty per cent of power. We would, by December 1 of this year, complete our analysis of all environmental matters in terms of a presentation of our case. We would be prepared to identify our direct testimony, our cross-examination, issues which we are concerned with and the like. The Applicant, for its part, will provide us with all of the information as they have on environmental matters. There will be no procedural argument over discovery. Our technical people will be given access to the plant at appropriate time to examine things. We will be able to discuss with the Applicant's technical people on an informal basis question as we have. In short, it will be a very open process in which we will be given every opportunity, in the next two months, to get the information that we need in order to be able to state our position by December 1.

We hope that perhaps on December 1 we will have a sufficiently strong case that if we oppose the issuance of the license or support it with modification, that we will be able to persuade the Applicant of those modifications and will be able to reach some sort of a settlement between us regarding our concerns with the environmental matters. If not, we would be prepared and are going to request the Board today to schedule, beginning on the 14th of December, the full environmental hearing, to have four hearing days on environmental matters in December, and if necessary, an additional eight hearing days on environmental matters beginning on the 11th of January. Our thought is at this time, given the environmental concern that we are focusing upon, our environmental concerns that are relatively narrow in scope.

For instance, the radiological releases from the plant are not matters with which we are going to be focusing in the environmental review. The issue I mentioned earlier is the problem with the fish and protections for the fish in terms of the operation of the plant. We think that twelve hearing days on the environmental issues will be adequate.

At the same time the Applicant will be presenting its motion for issuance of a license to test up to fifty per cent of power with our support. We would urge that the Board set that hearing for the conclusion of the radiological safety hearing in November, whatever day it is that we finish on radiological safety. We simply add on to that the number of days that the Board feels are required to conduct its environmental review of the question of the environmental impact of low power testing. That will be under D-2 of Appendix D.

We would suggest to the Board that setting three hearing days should be more than adequate for the Board to consider the environmental effects associated with that no power testing.

CHAIRMAN JENSCH: When you say no power, you mean up to fifty per cent?

MR. ROISMAN: Yes. That is a testing as opposed to an operational license.

If I may, let me explain. There are some other dates which we can discuss as well. We believe the essence of the agreement is to try and bite off a little bit as we go along the way. The question of fifty per cent testing is something that we think we can live with. That is not to say that we are delighted with it. Settlements also involve something where people give up something in order to get something else in return. That fifty per cent testing involves the Applicant providing us with information on this.

Approximately seventy-five testing days under an ideal testing schedule, that is. In short, if no problems arise in the testing schedule, the no power testing takes seventy-five days to reach and conclude the testing at the fifty per cent level. That seventy-five days, in effect, is as much time as we are going to need to conduct our environmental review. It makes good sense to us to not put the Applicant in a position of having to delay the plant to a point where it couldn't meet--that is, it would be physically impossible to meet its customers' electric needs that summer, assuming it gets an operating license for the plant. If there was a way to give them the opportunity to meet that need in the summer, they get a license for the plant and still conduct an environmental review, that is.

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Obviously the essence of the entire matter has to do with the ability of the parties and the Board's agreement with the kind of scheduling that we are suggesting. It is short. Our people are working day and night on nothing but Indian Point, as I think the Applicant is, also. That schedule depends on a number of factors. Let me go through some more aspects of the details of that to indicate what those factors are.

Radiological hearings that begin on November 1 -- And we will assume will be required some time around the 10th or 12th of November. At that time the Applicant will submit proposed findings of fact on radiological safety factors within twenty days. The Citizens' Committee for the Protection of the Environment will submit its proposed findings within twenty days or January 11th, whichever is later. It is to get the radiological safety issues disposed of in the sense of getting before the Board all of our various feelings on those before the environmental review is completed, so that the Board can have the time to meditate on that aspect of the case even though there will be the whole cost analysis and all issues to be presented.

Secondly, the hearing on the no-power testing will take place, we will say, if the hearings on radiological safety end on November 12th, that they will be on November 13th, 14th and 15th, three hearing days, during which time the

Board will have an opportunity to interrogate the witnesses for the Applicant and the Staff with regard to the fifty percent testing license.

CHAIRMAN JENSCH: Including environmental information?

MR. ROISMAN: Oh, yes, absolutely.

The radiological safety aspects of that will be from the standpoint of the Citizens' Committee on Protection of the Environment, and will be its entire case. In other words, the Board will have at that point the full record on radiological safety, and it will be the position of the Citizens' Committee on Protection of the Environment, that that record is the record for all purposes of the 5057A considerations of the fifty percent testing license. Within seven days after the conclusion of that fifty percent testing review, the Applicant will submit its proposed findings of fact. Seven days thereafter the proposed findings of fact of the Citizens' Committee for the Protection of the Environment and any other Intervenors, and four days after that the Applicant would submit any rebuttal matter.

It would be our hope -- And this is very important to all of the parties. -- that the Board, within thirty days, be able to act on the motion for fifty percent testing, which of course would be to approve immediately twenty percent and to refer to the consideration of the testing above twenty percent. That, also, in line with the regulations under Appendix D.

During this period of time the technical people will be working on the full environmental review, and on December 1 a presentation of that will be made. First to just sit down with the Applicant and the Staff and let them know what our feelings are and see what if any agreement can be reached, and of course, shortly thereafter, a day or so, file that matter with the Board. Everybody will at that point be pointing toward the beginning of hearings on environmental matters on December 14th. We have all checked our schedules and hope this is adequate notice that the Board will be able to set aside four hearing days in December. The split of the environmental hearing between December and January has two purposes: First, we realize the detailed environmental statement must be in the hands of the party and the Board before we conclude hearings. We hope that if that detailed environmental statement is not ready by the commencement of hearings on the 14th of December, that it would definitely be ready in advance of the commencement of hearings on the 11th of January, and the natural recess that the Board and the parties might take from public proceedings during the Christmas recess, New Year's recess, will still be available time to get that matter completed by the Staff if it had not been previously completed.

Secondly, the Applicant, as the Board knows, now has pending a license request for a ninety percent testing and operating license under Part D-2 of Appendix D. That pending motion will be in effect delayed by the Applicant in lieu of its fifty percent testing license. But it will be renewed with the understanding that the hearings in December for the first couple of days will focus on that aspect of the environmental review which would enable the Board to resolve the ninety percent testing and operation licensing request, and the Board will then have before it the necessary information to make a decision on the ninety percent licensing and operating.

That is, if you will, a fallback position by the Applicant, which we don't have any disagreement. If the environmental review and the Board's consideration of environmental matters following that review prevents the issuance of an initial decision with regard to the licensing of the plant, say, by the 16th of April, the Board would still be able to make the more limited findings associated with a ninety percent testing and operating license that are envisioned under Section D-2, and would be able to, if it is a green light, to give the Applicant the green light, or if it is a stop, be able to warn the Applicant that it best be looking for other alternatives for power in the summer of

1972 at Indian Point, at an early stage, so that some action can be taken by the Applicant to cope with that problem.

For our part, we will, and the Applicant, be submitting, following the hearings in January -- And that will be no later than the 21st of January when we will conclude hearings. -- the Applicant would submit within fourteen days its proposed findings of fact on environmental matters. We will submit within fourteen days our proposed findings of fact and the Applicant would have ten days for rebuttal, and the Board --

CHAIRMAN JENSCH: Let's not hurry quite so fast. I haven't heard the Staff in here yet. We certainly aren't going to act without the Staff.

MR. ROISMAN: When I said the second set of fourteen days, I mean us. You see, I read Calvert's Cliffs literally. They, like us, are supporting public interest here. I include them as one of the parties.

MR. KARMAN: We would be included in the same category as the Intervenors with respect to findings and responses.

CHAIRMAN JENSCH: Very well.

MR. ROISMAN: I didn't mean to exclude them by that.

Following the submission of the proposed findings, I should point out that by that time all of the proposed findings and submissions on radiological safety will be already in. In other words, the Board will have all the proposed findings on that aspect of the case. The Board, consistent with the guidelines, within forty-five days, which comes very close to being around the 15th of April, can issue its initial decision in this proceeding. This scheduling is geared, we think, to having a full environmental review at a time when we are prepared, the Intervenors, Hudson River Fishermen's Association and the Environmental Defense Fund, to make a full presentation of our case, a time in which we feel confident to have our information necessary to state our views on these matters and support these views with additional evidence and cross-examination, as the case may be.

I think this represents the Applicant as well. We are very pleased that we can try to work out this problem. Both of us concentrating not upon the narrow objective of either getting a license or not getting a license, but on a broader objective of getting the facts before the Board and having reasonable opportunity for a hearing and getting a decision within a reasonable period of time. We have tried to work this out early because we hope that the Board will set its schedule to accommodate this hearing in October to deal with hearings in December and January. Twelve days of hearings we think are not unreasonably long. But we are not asking the Board to block out an unreasonable period of time compared with its other commitments. We hope that the procedures will meet with the Board's approval and we can, at this hearing, schedule those dates and proceed with the business of getting, if you will, to the merit, which is where we all want to get very quickly.

Knowing that the procedural matters are covered, and at the minimum, we will be having three days of hearings or so in November on the fifty percent testing. That will be an uncontested proceeding. Also, hearings beginning on the full environmental review on the 14th of December.

CHAIRMAN JENSCH: We certainly appreciate the fact that the parties have conferred and propose a specific schedule. The Board will not give any commitment that the schedule will necessarily be our guide. We are going to take the time with this case that we think is necessary, whether it is contested in in certain parts or not. There have been suggestions as to how much time should be allowed for decisions. I think the Commission's early guidelines in those regards are somewhat unrealistic on many occasions, and the Board has had to take more time. This Board will expedite this proceeding in every way it can. We will necessarily be guided by the content of this record in the amount of time that has to be taken. We will try to schedule these dates, and perhaps after the noon recess we can indicate how well we can set our schedules for the hearing time. A time for review of these matters is something that is not easily gauged in advance, and we don't intend to indicate to the parties that in any way because they have a schedule that they suggest for us, it will necessarily be our guide. The parties are to be commended for the fact that they are spending a lot of time in conference about both the procedural and the substantive matters. I want to hear from the Staff, though, in this regard. I don't know anything about the Staff work at all. All I can take a look at is the calendar I see posted from time to time. I know they have a heavy schedule. The Board is not inclined to say to the Staff that you have thirty more seconds to submit some document or forever hold your peace. We will be inclined to give some latitude toward any scheduling which affects them. We are looking to the Staff -- And I say this without any reference in any adverse way to any of the parties. But for the balance of the presentation, after review of the evidence from the Intervenors and the Applicant, we are inclined to receive from the Staff only their deliberate determinations and not any hurried judgments submitted in this proceeding.

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MR. KARMAN: Mr. Chairman, I wish that the Regulatory Staff can afford the luxury of indicating that we are concentrating the entire Regulatory Staff's intention on the Indian Point 2 plants. I must and I can reassure the Board and the Chairman that this case does have a high priority on our schedule, and we are doing everything we can to move it along as quickly as we can. I would have to look and check all those dates that were thrown at me rather quickly this morning. But from the sound of those dates and from the scheduling as indicated to me before I left Bethesda yesterday, it would appear that we can, barring any unforseen difficulties, meet that schedule. We will certainly do everything we can to expedite it.

The main problem, as far as I can see, will be an environmental problem. We are continually and constantly kept in touch of all radiological problems, but we are concentrating very heavily now under the new Appendix D and various other regulations in the environmental field related to this particular plant.

I, too, would like to take a look at the schedules. Unless there is something in here that disturbs me at the moment, I feel the Regulatory Staff can go along basically with the scheduling as outlined by Mr. Roisman.

CHAIRMAN JENSCH: I'm sure your protective safety with that "basically going along," we expect you would

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basically go along. We suggest to the Staff, that we are not looking for any hurried judgments in this matter. We expect a deliberate determination to be submitted to the Board.

Is there anything further that we can consider from a procedural point of view at this time, or in any other respect?

MR. TROSTEN: Might I comment with respect to Mr. Roisman's presentation?

CHAIRMAN JENSCH: Yes.

MR. TROSTEN: We are in general agreement with what Mr. Rosiman has said, and I think he has presented our situation very clearly. I would like to add a few points with respect to what he has told the Board. It would be our earnest hope, Mr. Chairman, that the Board would be able to give consideration to that portion of our motion which relates to testing above fifty per cent of power, and to operation at any level of power as opposed to testing. Expeditiously at the commencement of the December hearing as suggested by Mr. Roisman, and that the Board would be in a position following due consideration of that aspect of our motion, to be able to issue an order granting or denying the motion within a period of seventy-five days after the commencement of the testing period.

I mention that, Mr. Chairman, simply because that is the estimated time that we feel that it would take us to complete

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that portion of the testing program which would take us up to and including the fifty per cent of power level.

In order to make this a meaningful progressive program, as Mr. Roisman had outlined, it would be highly desirable and we would ask that the Board take into consideration this factor and attempt to conclude its deliberations on that aspect of the motion within the seventy-five-day period.

CHAIRMAN JENSCH: May I interrupt?

MR. TROSTEN: Yes, Mr. Chairman.

CHAIRMAN JENSCH: I hope in your presentation is a clear demarcation as to the effect on the environmental situation, if that is your intention to request authority to do. Maybe this is an experimental case in that regard. If you raise the lever another ten degrees or another ten kilowatts or other MW, you get another first on the environmental, and you give it another ten shots and you get something else. I don't know how fine a line can be drawn for each megawatt of increase, or whether the consideration, after we get twenty per cent of power, which is the extent that the Commission has authorized the Board to act on, the balance power increase is included in one summation so that the entire matter can be closed.

From that point of view, I don't know. If you are going to ask for step-by-step increases in power, I hope you are going to be able to tell us that one more leaf will fall

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from the tree or another dozen fish will get it in the neck or something like that. I don't know how well you can clearly define the power increases that you have apparently been discussing among yourselves. I hope you can tell us that either there won't be any effect or--we want the significant increase, that there will be no significant change from ten megawatts to twenty megawatts, and from twenty to thirty, and we get no significance all the way through. It seems to me that you may necessarily be in a consideration of once you get beyond the twenty per cent, preparing the record for the Commission, it would feel adequate for its consideration, that whatever the power level is.

I just raise that as a problem in the preparation of a record for the Commission.

MR. TROSTEN: We appreciate the Chairman's suggestion and we are indeed very conscious of his point. To the extent that one can draw some meaningful significant distinctions at these very points, it would be our intention to have the record clear in this respect, Mr. Chairman. So that is our intention.

CHAIRMAN JENSCH: Proceed.

MR. TROSTEN: The second point I would like to add as a supplement to Mr. Roisman's presentation, is that in view of the progressive nature of the presentation, we would hope that the Board would give preference to its consideration of

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the fifty per cent testing motion in preference to matters pertaining to the hearing as a whole. It is possible, in view of the schedule, that we have discussed with the Board this morning, that the Board's deliberation on the motion for authority to test up to fifty per cent of power would overlap to some degree the commencement of the hearing. It is our hope, as the Applicant, that the Board would in general give preference to the consideration of the testing motion as opposed to the full hearing.

Finally, with respect to the general matter of schedules, time has not permitted up to this point, Mr. Chairman, the additional step that the Applicant wishes to take, which is to conduct the most detailed kind of discussions with the Regulatory Staff to assist the Staff in its performing its function. So the Staff would be able to have its detailed statement, for example, completed by the 1st of December. It would certainly be our hope that we would be able, by working with the Staff, to assist the Staff in carrying out its regulatory functions so that the draft detailed statement could be prepared and circulated, and then the final detailed statement could be available significantly in advance of the December hearing. We certainly will do our very best to work with the Staff in this respect, to help them, and quite obviously, it is a very important aspect of this whole matter that the Staff be prepared to move forward on

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the schedule as well as the Board and the parties.

The only other point I wanted to make, Mr. Chairman, concerns the affidavits which the Applicant intends to submit in support of our motion. We are very hard at work on those affidavits right now, Mr. Chairman. It is our expectation to have these to the Board and the parties in the very near future, certainly well in advance of the starting of the November 1st hearing.

CHAIRMAN JENSCH: I know you have drawn up schedules that you feel will accommodate your interests and the interests of the Intervenors and the Staff. I just wonder if you could limit your so carefully defined interests that you have imposed the burden for the decisional process, and above twenty per cent of the power, I think you are talking about the Commission's activity; that you may be defeating your endeavor to have the full consideration of this matter as to whether the Commission will give consideration to piecemeal considerations about this, or whether the case shouldn't be resolved upon the basis of the final safety analysis report. You are asking for the power level that your plant is capable of operating and letting it stand or fall on that basis.

I don't know whether your schedule is self-defeating your purposes. I don't know. These are problems that we probably will be exploring in the course of the hearing. It may well be that the Commission itself, which has reserved

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to itself the determinations above twenty per cent of power, may decide that they want to hear it once and that's all. Whatever power level this record will justify, if any, is a matter that they will consider and none else. We haven't had guidelines from the Commission in that regard. I just mention that as a consideration that may be a problem in our hearing. After all, when we go beyond the twenty per cent of power, I think we will be developing a record that we think the Commission will want for its considerations.

MR. TROSTEN: I understand the Chairman. We have certainly given the most serious kind of considerations to the point you are raising. It is our view, Mr. Chairman, that both from the standpoint of the Applicant's schedule which necessarily contemplates a series of actions leading up to full power operation, and also necessarily contemplates that from the standpoint of uncovering any problems that might be present in the testing system, that we commence this testing program and carry out as much of it as we can as soon as we can. So that we would be in a position to resolve any difficulties that might come up in time to enable the plant to be in full operation by the summer of 1972. Not only from those standpoints, but also from the standpooint of how Appendix D is set up, Mr. Chairman.

We feel a progressive approach to a full power operation is fully within the contemplation of both the

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Applicant's schedule and the regulatory process. We also feel that the nature of the factors which the Commission has directed the Board to give consideration to in Section D indicates that a progressive approach, first considering what can be considered, and then ultimately considering the full power full term license is what the Commission had in mind of what would be a fruitful approach to this problem.

CHAIRMAN JENSCH: Have you completed?

MR. TROSTEN: Yes, I have.

CHAIRMAN JENSCH: Is there any other matter that we can consider at this session?

The Board does have some concern or inquiry by way of clarification. We thought, however, we would not take a formal recess for lunch, at least, at this time, but we will take a fifteen-minute recess now and then come back and continue until we have completed our present inquiries on clarification. At that time perhaps we can adjourn. Is there any objection to that procedure? Hearing no subj objection, at this time we will reconvene in this room at twelve o'clock.

(Recess.)

THE CHAIRMAN: Please come to order. At this time I would like to discuss the effect of the situation, the pressure vessel situation.

MR. BRIGGS: As I read the additional testimony of Applicant concerning reactor vessel integrity, the design of the pressure vessel is based largely on the calculation of the stresses and stress distribution in the reactor vessel, is that right?

MR. VON OSINSKI: Yes, sir, that's correct.

CHAIRMAN JENSCH: Excuse me. Just a moment, if I may. At the time each speaker speaks will he give his name so that the reporter will have it.

MR. VON OSINSKI: My name is Roland J. Von Osinski.

MR.BRIGGS: And the vessel was designed by engineers at Combustion Engineering?

MR. VON OSINSKI: Yes, sir. The design was done by Combustion Engineers.

MR. BRIGGS: To what extent has this stress analysis for the Indian Point 2 vessel been reviewed by an independent group?

MR. VON OSINSKI: The stress analysis of the Indian Point vessel has been reviewed by Westinghouse Engineering organization.

MR. BRIGGS: Were calculations, some of the calculations repeated by Westinghouse people? To what extent, how thoroughly did they go through the stress report?

MR. VONOSINSKI: The independent analysis which Westinghouse performed consisted of a detailed rigorous thermal distribution analysis, and in addition we formed our own, based on the results that were obtained from that thermal analysis that Westinghouse performed we performed our own fatigue analysis and our own stress analysis calculations and compared the results. We did not in essence go through and just compare numbers. We compared results.

MR. BRIGGS: And this comparison was favorable?

MR. VON OSINSKI: Yes, sir. It was.

MR. BRIGGS: To what extent has the stress analysis for the reactor vessel been confirmed by experimental determination of stress in that vessel or in a similar vessel? In other words, were strain gauges put on the vessel during hydrostatic tests or anything like that to confirm analysis?

MR. VON OSINSKI: To my knowledge the Indian Point reactor vessel was not strain gauged.

MR. BRIGGS: To your knowledge have other vessels of that size and wall thickness been thoroughly tested by use of strain gauges?

MR. VON OSINSKI: May I confer with some of the fellows here, please?

MR. BRIGGS: Yes.

MR. VON OSINSKI: Excuse me for conferring with my associates here.

I am aware of the detailed strain test program performed by Combustion Engineering on a subsequent 4-loop reactor vessel, a Westinghouse vessel. It was done on their own. I do not have the details. I cannot confirm the results of their analysis, but I do know that they did strain guage test a pressure vessel that was very similar in construction and design details to the Indian Point Reactor vessel.

MR. BRIGGS: Could you possibly relate or provide a reference if there is a report that describes that testing?

MR. VON OSINSKI: Combustion was intending to write a paper, I believe, around the particular test program that they did on one of our vessel's closure heads. This was done only on a closure head in the ligament penetration area. But I'm sorry, I can't speak for Combustion Engineering because it was done by them. I can attempt to contact them and see if this result could be made available, but that's all. I can't commit them, I'm sorry.

MR. BRIGGS: It would be helpful if you could contact them and ask whether it's been reported and what the name and number of that report might be.

MR. VON OSINSKI: I will be most happy to.

MR. BRIGGS: On the materials that are used in the vessel I noticed that the Atomic Energy Commission has recently published a, well, published for comment a proposed regulation on fracture toughness of materials. Are you acquainted with that proposed regulation?

MR. HAZELTON: Warren Hazelton. Yes.

MR. BRIGGS: Does the material in the Indian Point 2 vessel satisfy the requirements of that proposed regulation?

MR. HAZELTON: Insofar as tests were performed on it, yes. We firmly believe the materials do comply with that proposed regulation.

MR. BRIGGS: You say insofar as the tests were performed on it. Do you mean according to the regulation there should be additional tests or--

MR. HAZELTON: That is correct. According to the regulation essentially, the proposed regulation, essentially two types of fracture toughness tests are to be performed on all the materials. These are the Charpy tests and the dropweight NDTT tests. At the time the Indian Point No. 2 vessel was being manufactured it was not the practice nor was it required to run the drop-weight tests on all materials. But we of course have the

Charpy test results and we can infer from these that the material will meet the requirements that were intended by the proposed regulation.

The materials in the Indian Point No. 2 vessel have good properties. There are none that appear marginal, so we have no concern in this respect.

MR. BRIGGS: And when you referred to materials in the Indian Point 2 vessel you are referring to welds, as well as the base material, is that right?

MR. HAZELTON: Yes.

MR. BRIGGS: Is there a substantial difference in the fracture toughness of the weld material as compared with the base material in the Indian Point 2 vessel?

MR. HAZELTON: I don't have the exact numbers in my head, but as I remember it there was not any substantial difference. That is the weld results that we had were good. Usually the fracture toughness of the welds is better than that of the base material and I don't remember anything that was different than that in the Indian Point vessel.

MR. BRIGGS: The Codes permit the grinding out of flawed areas in the plate and of course in the welds also and repairing those flawed areas by use of welding, that is repairing areas in the plate by use of welding. Were you close enough to the construction of the vessel so to what extent grinding out of the defects in the plate material was a required, grinding and repair of defects in the plate material?

MR. HAZELTON: I don't remember that from my own personal recollection. I just don't remember any of the details as to that.

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MR. BRIGGS: I believe it's the position of the Applicant's witnesses that if the pressure vessel as it's now fabricated, if it contain no flaws, that there are I shouldn't say no circumstances, but that if the temperatures and pressures in the vessel are kept within the hydrostatic test pressures, let's say, that there is no possibility of the vessel failing. Is that a reasonable way of putting the position that you people take?

MR. WEISEMANN: Mr. Weisemann. I believe that that is an over-simplification. I think that we have looked at different modes of failure. If you consider the vessel under its operating conditions after it has been brought up to temperature I believe your statement would be a correct statement, except there are certain modes of operation which involve transition from pressure to say room temperature and atmospheric pressure to the operating conditions. Where there are additional restrictions placed on operation to avoid applying significant stresses in the reactor vessel under conditions when the vessel would not behave in a ductile manner I believe it's the combination of these things that, together with the surveillance programs, assure ourselves that there are no surprises as far as the/other words keeping track of the special materials as they actually--as their properties change during operation, as opposed to relying on predictions that were made.

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MR. BRIGGS: If we consider the vessel operating at temperature and operating at pressure, it's the Applicant's position that it's necessary for there to be flaws in the vessel in order for the vessel to fail, is that right?

MR. WEISEMANN: Well, it's our position that if the vessel was once brought up to that operating condition and which is the case for hydrostatic tests that the vessel will continue to be able to operate throughout its lifetime when one considers that there is a surveillance program to take care of the problems that are associated with systems which are required to keep the vessel within its intended operating range, as well as to take into account the possible changes in vessel conditions.

Actually, I believe that the position is that the vessel as it is constructed will not fail.

MR. BRIGGS: How large a flaw in the vessel wall would be required for the vessel to fail?

MR. WEISEMANN: I'd like to have Mr. Hazelton answer that question.

MR. HAZELTON: That's a very complex question as you probably realize. That of course depends on exactly how the vessel is being operated, but as long as the vessel was being operated under conditions that are called for in the technical specification we feel according to our best judgment and our best ways of calculating this that we can that it would take a very, very large flaw to cause failure of the vessel.

Now when we talk about a very large flaw, we can talk about a flaw say perhaps halfway through the wall and many feet long or a flaw all the way through the vessel wall and several feet long. Obviously the particular geometry of the flaw, where it's located, so forth, has an effect on the size of the flaw that would be required to cause failure of the vessel. It's a very complex question, but I think the size of the flaw that we are talking about that would be necessary to cause failure of the vessel would be very large indeed.

MR. BRIGGS: How large is the largest flaw in the vessel as the vessel was fabricated?

MR. HAZELTON: Again, I don't see how we can answer the question how large is the largest flaw in the vessel. Now we can say, however, that because of the way the vessel was constructed and inspected we feel that it's highly unlikely that there will be flaws any larger than the order of half an inch in the vessel. We obviously don't know the size of the largest one now, but we feel very, very sure that it would not be larger than on the order of half an inch.

MR. BRIGGS: When you say on the order of half an inch, you mean something that's a half inch long, a crack a half an inch long and a half inch deep, for instance?

MR. HAZELTON: Well, I'd mean a flaw something on the order of say a half an inch deep by maybe several inches in the other dimension or something comparable to that. Again, the geometry of the flaw has a great deal to do with its detectability. So if the flaw is several inches long it need not be as deep in order to be positively detected.

MR. BRIGGS: What kind of standards are used in the ultrasonic testing of the material to assure that one can detect a flaw of this size?

MR. HAZELTON: There are several types of standards used. I think Mr. Noel Dressel could answer that better than I could.

MR. DRESSEL: Noel Dressel.

The acceptance standards are based on both longitudinal and shear mode, and normally in the case of a plate of three per cent notch by one inch long, the plate thickness.

MR. BRIGGS: Could you explain how the testing is accomplished and what these modes are, what you are talking about in layman's language?

MR. DRESSEL: Well, in the case of the longitudinal mode, sir, it's applying a sound into a plate to see whether a lamina or parallel of condition is evolved or is in that plate.

In the case of the shear mode we apply the ultra-sound in a shear direction to locate any indication that may be vertical to the plate, and this is done in two directions perpendicular to each other to provide for all location of all indication. So that the hundred per cent volumetric inspection of the material is performed.

MR. BRIGGS: As I remember, the vessel wall is of the order of eight inches thick.

Now what size--what would be the smallest flaw then one could expect to detect?

MR. DRESSEL: Well, shear modes, 3 per cent.

MR. BRIGGS: Three per cent then of eight inches? This would be three per cent in depth?

MR. DRESSEL: The three per cent notch, one inch long.

MR. BRIGGS: One inch by three per cent of the depth?

When you saythat there was one hundred per cent volumetric inspection of the vessel, this was done by moving the detector along the surface of the vessel?

MR. DRESSEL: Yes.

MR. BRIGGS: How is it accomplished?

MR. DRESSEL: Yes. By moving the crystal along the surface of the plate so that--it's difficult to explain--so that you get a shear mode in one direction of the plate and a shear mode in the other direction of the plate so that you are seeing any indication that might be oriented this way, any indication that may be oriented perpendicular to that.

MR. BRIGGS:Is this sensitivity equally good examining welds and examining base plate material? Are there differences?

MR. DRESSEL: Well, the welds are examined by radiography to two percent normal, two percent sensitivity.

MR. BRIGGS Are they examined ultrasonically also?

MR. DRESSEL: They are examined for inflammation after the hydrostatic test in the shell areas with both the same methods, shear and longitudinal mode, in the shell areas.

MR. BRIGGS: Let's see. I may have misunderstood you. You indicated that the welds were examined by X-ray methods and then that the welds were examined ultrasonically after the hydrostatic test.

MR. DRESSEL: In the area of the shell regions, yes, sir.

MR. BRIGGS: How about around the nozzles?

MR. DRESSEL: And around the nozzles, yes.

MR. BRIGGS: Well, the original question was one that was concerned with the sensitivity. If the sensitivity for detecting flaws in welds by ultrasonic methods as good as the sensitivity in detecting flaws in the base plate?

MR. DRESSEL: Well, I will have to -- I will answer that two ways. The technique is sensitive, will give the same sensitivity. The actual acceptance standards, that

I'd have to confer on and familiarize myself with them.

But the technique is capable of the same sensitivity, yes.

MR. BRIGGS: Does the presence of the cladding in the region between the clad and the base material affect the results that one gets on ultrasonic tests? Does it affect the sensitivity with which you can detect flaws in the material?

MR. DRESSEL: Not to my knowledge, no, sir. We haven't in my -- From what I know there has been no experience of interference with the cladding for getting into the weld with the fuel areas. It may require some adjustment to reduce background, but this does not reduce the sensitivity technique.

MR. BRIGGS: Are you an expert in ultrasonic testing? I am asking this to get some idea. Is this one of your major fields?

MR. DRESSEL: Well, I have been associated with ultrasonic testing for a great many years and, yes, it's my opinion, yes.

MR. BRIGGS: I understand that in some places there has been an indication of I will call it a failure of the bond or of cracking or what have you between the clad and the base material on clad vessels, not necessarily Indian Point vessel, not necessarily reactor vessel, but does this have any effect on the sensitivity of detection of flaws in the base material beyond the clad area?

MR. DRESSEL: Well, I don't know that I quite understand your question. If we are now talking about ultrasonic inspection on the pressure, on the weld section itself --

MR. BRIGGS: On welds and base material also.

MR. DRESSEL: On welds and the base material, if there were indications within that clad region area that you are speaking of, this would be -- If they were within the sensitivity range the technique would use them. In other words, we'd see them.

MR. BRIGGS: Let me ask you this. Suppose you have an area in which the bonding between the clad and the base metal is essentially disintegrated. Can you detect flaws in the base material beyond this --

MR. DRESSEL: No, no. No, no, because you would see the defect in the bond before you would get into the base material.

MR. BRIGGS: And you'd not detect anything in the base material?

MR. DRESSEL: That's correct.

MR. BRIGGS: If you were measuring through the clad.

MR. DRESSEL: That's correct.

MR. VON OSINSKI: Excuse me. My name is Von Osinski. I'd like to clarify a point if I may.

We require that this particular Indian Point reactor vessel cladding be ultrasonically examined for bond 100 percent so that we can assure you that the cladding was bonded. There were no lack of bond areas in the cladding.

MR. BRIGGS: But if now -- I don't know how to term this. -- but if flaws developed between the clad and the base metal, a question was one of would this obscure flaws in the base metal itself?

MR. DRESSEL: It would obscure them, but it would show that also you were not getting into the base material and you'd have a problem.

MR. BRIGGS: Well, we have indicated then that you gentlemen can conclude that the flaws in the reactor vessel, if there are any, any of any significance, it would be highly unlikely that they would be more than half an inch deep by several inches long, and you also conclude, I believe, that a flaw this size would not be sufficient to cause failure of the reactor vessel under the conditions that are imposed by the technical specifications.

There was a calculation requested, I believe, by the ACRS concerning the situation if there were a transient and the rods failed to drop, in which case, if I recall, the pressures in the reactor vessel would go as high as about

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4000 psi. Is this number right? Do you recall the calculation?

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MR. WEISEMANN: I don't recall the precise number, but I believe that's perhaps in the right order of magnitude.

MR. BRIGGS: Would you, before the next session that we have, look at that situation and indicate whether the conditions imposed on the reactor vessel by that transient would change any of the statements that you have made in the report here? I don't ask you to do it now. Unless someone is completely familiar with those numbers, that is.

MR. WEISEMANN: Excuse me. Are you speaking about whether after having experienced such a transient, you would change the opinions, or whether consideration of that transient would change what is said here to the vessel in general?

MR. BRIGGS: This additional testimony doesn't conclude an examination of that transient. There was consideration given when the transient was examined in your other reports of what these vessels would be and whether the reactor vessel would fail. I'd like for you to look at the case of that transient again, the conclusions reached then and what is stated in this report, and see if there is anything different that you put in this report as a result of the conditions that exist in that transient.

In other words, it is a case that wasn't considered here. I would like to have it looked at again. I believe it is concluded here, then, in the report in the testimony that in order for the vessel to fail, the flaws that are present

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would have to be substantially enlarged as a result of the cycling that occurrs, the growth of floors in the vessel, and the rates of growth are so slow that one wouldn't expect a substantial increase inside of the floor during the life of a vessel.

To what extent are the cracked growth calculations based upon experimental information for, let's say, vessels of the size and wall thickness of the Indian Point 2 vessel? Has anyone anywhere followed the growth of floors in a large vessel like this under test conditions?

MR. HAZELTON: Basically the crack growth data given in the report, based on that approach to crack growth, of course, these data come from experimental tests on specimens. I think that a maximum size vessel that we run tests on is in the order of four inches, which is approximately half the thickness of the vessel. But the data also indicate, as the result of specific programs to determine this, that there is no side effect associated with that data as used this way.

There were some other portions of your question there that we might refer to another time.

MR. MANJOINE: We have programs in which we have been using fraction mechanics which really study the stress distribution at the base of the crack and really consider the local conditions that are the things that propagate the crack. Using different sizes of specimens and specimens of different geometries, of cracks, that are center crack plates, edge crack plates or the standard WOL, wedged open loading plates. In all of these cases the crack growth rates were nearly identical. So this shows that the mathematics involved in studying these stress feel at the end of the crack was indeed correct in giving us the right crack growth rate. So the size of the vessel in which you calculated the stress field would not be an important factor.

MR. BRIGGS: Do you think the stress feel can be calculated accurately enough so there is little concern about this?

MR. MANJOINE: The fact that we can use cracks of different geometries and come out with predicted crack growth rates does give you the confidence to know that you can use some other geometry, although larger in size, and predict a crack growth rate.

MR. BRIGGS: What we said then, if the vessel is properly designed and if the materials are properly qualified, and if the vessel is operated under the conditions that are specified in the technical specifications, that we wouldn't expect any flaws that might be present to grow sufficiently to cause the vessel to fracture. To what extent is inspection during operation important to confirm these expectations?

MR. WEISEMANN: The integrity of the reactor vessel is extremely important. Therefore, it has been considered to

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be prudent by experts in the field to institute surveillance programs to confirm the condition of the vessel material and also to check on the actual behavior of the possible indications of defects during the operation of the plant. Actually we do not expect that we will find anything that is detrimental in any way.

However, by having those programs, it would be possible to at any time ascertain the condition of the reactor vessel and also to review the operating procedures to make sure that those procedures are appropriate.

I might add that there is a high likelihood that this experience will show that there is more flexibility in the operation of these plants by relaxing some of the requirements for operation of the reactor vessel as a result of this experience. So there is really more than just simply concern of the safety. There is the possibility that because of the wide safety margin that you see from the calculations, there is a strong indication that experience will show that we can relax the requirements for operation which would allow greater flexibility in the operation of nuclear plants without encroaching upon the kinds of safety measures that are necessary to assure the safety of the public.

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MR. BRIGGS: To what extent is this inspection program concerned with initial flaws in the vessel? In other words, on page A-25, I believe, it indicates that only selected pressure vessel areas were tested ultrasonically after the hydrostatic test. Did these areas involve looking at places where flaws had been found by the initial ultrasonic test? In your test program are you planning to follow the behavior of flaws that were found by the initial test? What is the basis for the program?

MR. GROB: The post hydro test mapping of this vessel concentrated on some of the highly stressed regions of the vessel around the nozzles, and also on the base plate and welds in the cylinder section, and then around the knuckle portions of the head, as also a stressed area. So probably the most important ones for consideration are the highly stressed areas. We don't expect or believe that the vessel flaw changes or characteristics as occurs during the lifetime of the plant might warrant a need for a periodic routine inspection of flaws to look at that day by day growth, or something like that. Over the long period of time to produce an additional level of confidence that the flaws have not grown in some unexpected way or there have been any surprises, it is our intention to perform certain inspections in these accessible areas, and as equipment might develop as appropriate, look into some of the less accessible areas. When I say less accessible, I mean the areas can be made there for inspection from the inside, using underwater techniques which requires use of equipment that is being improved in developments all along.

MR. BRIGGS: Making the question a little bit simpler, maybe, certainly there must have been some indications from your ultrasonic testing in these areas.

MR. GROB: Yes, and these are what we would look at again. These provide a signature to look at again to see if the signature had changed. If it has, you would evaluate what changes have occurred.

MR. BRIGGS: So then at least a part of the test program is based on following the behavior of indications that you have gotten from your testing as the vessel was fabricated and as followed in the hydrostatic test?

MR. GROB: Yes.

MR. BRIGGS: One final question, I guess. In your report you mention that the critical flaw size of a junction of a nozzle shell can be much smaller than the critical flaw size in the shell itself. Suppose now that one had a critical flaw size at the junction of nozzle and shell. Would this be expected to propagate through this shell? What would happen to it in the event of a -- If it is a critical flaw size, it must mean it will propagate somewhere.

What happens where you have one at the junction between the nozzle and the shell?

MR. HAZELTON: I think perhaps we might clarify this a little more. I think when you were referring to critical flaw size, were you referring to Appendix C?

MR. BRIGGS: Yes, Appendix C, page ten.

MR. HAZELTON: This was a general discussion of where more critical areas are in the vessel. It also considers the situation where there is no radiation effects on the shell. Under these conditions it would take a smaller crack to cause a failure at that nozzle location than at the shell simply because the stress levels are higher there.

That situation, of course, would change during life. As the vessel shell is irradiated, you will find that for any given circumstance of temperature and pressure, the limited area would be the shell even with the flaw size in the same order as in the nozzle. What one has to do is operate the vessel in such a manner that you have no problems in any of the areas. I don't know whether I have answered your question or not.

MR. BRIGGS: Let's take the question at the beginning of life, if we may. When you say a flaw is of critical size, what do you mean?

MR. HAZELTON: That's the kind of terminology used by a fracture, when experts are saying under those particular circumstances of temperature and stress conditions and material properties, a certain size crack will cause failure, and is called the critical size. But the actual size is, of course, a function of the material properties under those conditions and stresses.

MR. BRIGGS: When you say it will cause failure, you mean it will begin to grow rapidly; is that correct?

MR. HAZELTON: Yes, that is correct.

MR. BRIGGS: The question was related to one, would you expect this crack then to grow down into the main vessel and to continue to grow in the main vessel, or would it be arrested in the main vessel?

MR. HAZELTON: This would depend on the exact circumstances, the exact transient that you are considering and the differences in the stress levels, the differences in the material properties that cannot be answered generally.

To answer your question, I think under certain circumstances --

MR. BRIGGS: So you couldn't count on the stopping at the main vessel necessarily; is that correct?

MR. HAZELTON: It depends on the particular circumstances that postulate. You would have these critical sizes and signs.

CHAIRMAN JENSCH: I wonder if you could be a little more specific in your answer. I think the question was, so, in other words, you can't expect it to stop when you get to the main vessel; is that correct?

MR. HAZELTON: I am trying to say that under some circumstances under certain situations I would expect it to stop. Under certain circumstances, I would not. I might say that in our approach here to positively prevent any fracture of a vessel, we do not rely on it stopping. I will try to answer your question. There may be cases where it would, but we don't rely on that to prevent fracture of a vessel.

DR. GEYER: I'd like to extend the questioning with regard to tests on a pressure vessel to tests on piping and other features of the primary system. What tests are made on the piping?

MR. WIESEMANN: What types of tests are you speaking of?

DR. GEYER: You just described radiography. What is the situation in the piping as compared to that?

MR. DRESSEL: Ultrasonic examination of the piping and --

DR. GEYER: Of all the piping?

MR. DRESSEL: Of the primary piping, and radiographic examination of the fittings.

DR. GEYER: The hydrostatic tests made after all the piping is assembled?

MR. DRESSEL: Well, in the loop, yes.

DR. GEYER: In this primary loop?

MR. DRESSEL: Yes.

DR. GEYER: What is done with the safety valves at the time your test pipe is --

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MR. DRESSEL: I can't answer that.

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MR. WEISEMANN: Generally the safety valves are kept closed in that type of test by gagging them.

DR. GEYER: The primary typing is made in what way? Is it drawn tubing or welded plate, bent and welded plate?

MR. GROB: In our hydrostatic tests, the safety valves were not installed. They were blind.

MR. HAZELTON: The primary coolant piping in Indian Point Number 2 plant is seamless forge pipe, austenitic stainless steel.

DR. GEYER: And you are confident that it is as free from flaws as the primary vessel itself?

MR. HAZELTON: Well, there are slightly different cirteria applied. The piping itself certainly is free from flaws. The welds that are welded together, in fabrication, of course, is another area of concern. But they are inspected radiographically. I probably should refer to Mr. Dressel to see if he has anything to state on whether he thinks there is any difference in the quality.

MR. DRESSEL: I didn't have an opportunity to go into any great detail on the acceptance standards to the main coolant piping. I wasn't completely prepared for that. I am aware of the fact that the piping is forged material, ultrasonically inspected and the fittings are cast material and radiographically inspected, and the welds radiographically inspected. Specific standards I can't call right out.

DR. GEYER: Thank you.

MR. BRIGGS: That were a few questions that I'd like to just mention concerning the Staff Supplement to their safety evaluation. It is not necessary that they be answered at the present time, but will come up for question early in November. You may choose to answer some of them now, if you wish, or just keep them until that time until the Intervenors are asking numerous questions. These might duplicate some of these questions.

MR. KARMAN: We will prefer to answer them at a subsequent time.

MR. BRIGGS: The Applicant calculates, as I see from his evidence and from the report here, that in the event of a pipe break, 8.24 square feet, double-ended cold-leg break, that the peak clad temperature will reach 2,300 degrees Fahrenheit. The Staff concludes that the maximum calculated fuel cladding temperature does not exceed the Commission's regulation of 2,300 degrees Fahrenheit. It comes very, very close but it doesn't exceed it. So there are some questions related to that.

One of these questions being, to what extent has the Staff checked the input numbers and the calculations at Westinghouse made to be sure there are no errors in the calculation which would lead to temperatures higher than 2,300 degrees?

My second question was whether the loss of cooling accident double-ended rupture of the largest pipe for the Indian Point 2 plant has been calculated, whether the temperatures have been calculated by an independent group by use of a program different from the one that Westinghouse uses. I guess that's straightforward.

The third question is somewhat similar. If the incident were calculated by use of the relap code and by Staff personnel, are all the plant characteristics known so accurately, and are the bases for the program so similar that the calculated maximum temperatures would be identical? If not, would it be expected that the relap code might calculate higher temperatures or lower temperatures?

And finally, a question that I suppose the Westinghouse people will have to answer. It may be in one of the reports. This has to do with the behavior of the accumulator water during the reflooding or the refilling. It says here in the Staff supplement, under 3.3 Analysis of the Reflood and Refill, under Part 4, that when the amount of steam generated becomes appreciable, the amount of steam in the core generated becomes appreciable during the refilling and reflooding, that flow is allowed through only the broken loop until the accumulator discharge in the intact loops is complete. That is steam flow. This gives a maximum pressure drop through the system. The accumulator water from the

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other loops is permitted to come into the reactor vessel.

It says finally that, "The steam flow resistance limits the rate of liquid rise in the core, but the annulus continues to fill with water until the liquid level reaches the inlet nozzle. After this it flows to the containment by way of the broken inlet pipe path."

What I am interested in here is what fraction of the water, from the three unbroken looks, actually flows out the broken loop into the containment during this period. I don't recall seeing curves of the amount. The curves that I do recall seeing seem to suggest that the level in the reactor vessel never did get high enough for water coming in from the accumulators, from the three unbroken loops to spill out through the opening in the fourth loop.

I am interested in knowing what fraction of the accumulator water does go out of the vessel in that way.

MR. TROSTEN: Mr. Moore is present here and is prepared to respond to your question.

MR. BRIGGS: If he can tell me which curve to look at, it will be very helpful.

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MR. MOORE: I don't believe there are any curves in the reports submitted to show it. The accumulator height is a function of time in the transient.

CHAIRMAN JENSCH: Would you take a microphone, please.

MR. MOORE: My name is James Moore from Westinghouse.

As a matter of fact, for the Indian Point calculation, we just filled the downcomer with the remained accumulator water. So there was very little spilled, if any, during the course of the accident, during the reflooding. We lost about twenty-five per cent of the accumulator water during available blowdown as a result of the assumption of bypassing during loading.

MR. BRIGGS: So then during the refill and reflood period, essentially none of the accumulator water, except for the one loop, spilled out?

MR. MOORE: That's correct.

MR. BRIGGS: RThank you.

There is a related question, then, and that is what would the situation be if one changed his assumption so if steam flowed through all four lines and the accumulator water that was entrained in the thing be carried out through the broken loop? In other words, one can make two kinds of assumptions: He can make the assumption that is made in the calculations that there is no steam flow through the three unbroken lines, that the accumulator water that comes into those lines goes into the space around the core barrel and is available at the core; he can make the other assumption that there is steam flow through those lines and that some of the accumulator water is entrained in the steam and is carried out the break. I'd like some discussion about which of these assumptions is the more conservative, and whether there is reason, really good reason for the assumption that is used here rather than the assumption of entrainment of accumulator water in the steam and this water being carried through.

I don't have any other questions.

CHAIRMAN JENSCH: Let we just have a brief inquiry on a matter that is wholly unrelated to that which has previously been discussed by my colleagues. This has to do with the environmental matter that I was reviewing very hurriedly on the entire environmental matter by the Applicant. I refer to Section 2.3.3-6. It says, "The effect of the expected river temperature rise--river dissolved oxygen concentration was evaluated and it was not expected to cause any significant changes in the devolved oxygen content of the water as passes through the plant."

On Pages 32 to 34 of the Applicant's submittal, it is stated, "Even if supersaturated conditions did develop, there is not sufficient time for a significant oxygen loss by gassing to the atmosphere. The time of passage of water through the plant from intake to outlet is 2.5 minutes."

The calculations show increased oxygen loss is insignificant. I am wondering if this isn't too much of an inconvenience, if this could be a written response and may be submitted in the next couple of weeks. Although we are not going to take up the environmental matters until December, I would like to have an opportunity to review it a little. If you could give references to studies that support those statements, that is. As I understand it, the indication is given that water will go through the condensor so fast that the oxygen cannot be lost. I wonder if you could describe to us the physical phenomenon of what occurs. As I understand it, on some occasions in still water you could get a supersaturation. I wondered whether moving water under pressure, that that is possible, and what experimental data support the position set forth by the Applicant. Does an air bubble cool the oxygen out of the water and does it stay there and go out of the outlet before there is any gassing? If it goes out into the water, what kind of a plume is there in the moving system, like the Hudson River? I wondered ifyou had experimental data to support that and also if you could describe the physical phenomenon with some more particularity. Give us a physical description of how this does happen.

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MR. TROSTEN: We will respond to this in writing, Mr. Chairman.

CHAIRMAN JENSCH: Thank you. Is there any other matter that can be considered at this session?

MR. ROISMAN: Mr. Chairman, you mentioned earlier that you might after the lunch break, which was just a regular fifteen minute break, give some further consideration to the scheduling of the environmental hearings, the taking up of the fifty percent testing at the conclusion of the November 1 hearings and the taking up of the full environmental to begin on the 14th of December. If the Board does have its views on that it would help us for just purposes of our own scheduling.

CHAIRMAN JENSCH: Yes. The Board will endeavor to conform, and our present indications are that we will reconvene the hearings as indicated by the parties, and I might say that, having gone to the Register early, I think it's advisable to get these dates established now, because other cases may conflict. We will accept the schedule that you propose for this hearing.

MR. TROSTEN: Thank you, Mr. Chairman.

CHAIRMAN JENSCH: Is there any other matter? If not this evidentiary hearing --

MR. TROSTEN: Excuse me. Just one point. Do I understand that subject to the presentation of the answer to the question that Mr. Briggs raised that the Board does not have any further questions for our panel of witnesses?

CHAIRMAN JENSCH: We cannot say at this time, but we would suggest that your witnesses need not be here at the first day nor the second. We will try to give you at least twenty-four or thirty hour indication if we do desire them.

MR. TROSTEN: Thank you, Mr. Chairman.

CHAIRMAN JENSCH: Is there any other matter? If not this evidentiary session is now concluded.

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