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LIGHT WATER REACTOR SAFETY RESEARCH IN THE UNITED STATES

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I.0 INTRODUCTION

Light water reactor (LWR) safety research in the United States in conducted by both private and governmental organizations, the former represented principally by the four LWR manufacturers and the Electric Power Research Institute (EPRI) while the latter is represented principally by the U.S. Nuclear Regulatory Commission (NRC) with some LWR technology work being done by the U.S. Energy Research and Development Administration (ERDA). In general it can be stated that "the overall objective in the direction of reactor safety research programs is to provide the basis and means for reliable and credible analysis of the course of events in hypothetical accidents to nuclear reactors, and the estimated consequences of such accidents."[1]

1.1 NRC LWR Safety Research

The NRC LWR safety research program is directed at providing a capability for an independent confirmatory assessment of the safety of nuclear plants under postulated accidents. The research data and the analysis methods are iteratively applied to the assessment of hypothetical LWR plant accidents to gain confidence that the margins of safety identified in the licensing review are well defined and quantified.[1] The programs sponsored by EPRI and the reactor manufacturers are directed more at providing a basis for supporting the safety design of the plants. ERDA is pursuing a program to improve the operational performance of LWR plants.

Largely as a result of the U.S. AEC Hearings on Emergency Core Cooling Systems (ECCS) and the severity of the hypothetical loss-of-coolant accident (LOCA), which is one of some 47 different "initiating events" analyzed in safety analysis reports, the LOCA is receiving the major attention in the NRC LWR safety research program. However, a number of other safety issues are also being addressed and these will be discussed in succeeding sections.

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The NRC LWR safety research program is divided into the following five program elements:

- Metallurgy and Materials Research
- Systems Engineering
- Fuel Behavior Research
- Analysis Development
- Reactor Operational Safety

Figure 1 shows how the research programs are interrelated to confirm the LWR LOCA analysis methodology. Figure 2 shows how the data and models are interrelated to produce verified computer codes and correlations. Basically, the process consists of evaluated data banks which will be used to verify the models and codes during their developmental phase. New data, normally of an integral nature, will be used to conduct an independent verification of the codes and models.

Of course, throughout the various U.S. governmental and private nuclear-related programs are many research projects covering safeguards, fuel cycle, siting and environmental research. However, these are not LWR safety research projects per se.

1.1.1 Metallurgy and Materials Research

The NRC metallurgy and materials research is designed to generate a more confident basis for the criteria and analytical procedures for evaluation and operation of the primary system pressure boundary (pressure vessel, piping, pumps, heat exchangers, etc.) of LWRs. The metallurgy and materials program elements may be further divided into the following subelements:

- Fracture Mechanics - covers research on reactor pressure vessel performance, development of elastic-plastic fracture criteria, and development of crack arrest theory
- Operational Effects - covers crack growth studies, research on irradiation embrittlement, and studies on corrosion and sensitization of piping and steam generator tubes.
- Non-destructive examination - spans the efforts to develop on-line systems to monitor welding and plant operation.

1.1.2 Systems Engineering

The objective of the research in the Systems Engineering program element is "to provide sufficient experimental data for establishing and verifying analytical models to permit an assessment of the thermal-hydraulic response of the reactor primary coolant system, components and reactor containment to possible off-normal and accident conditions."[1]

The systems engineering element consists of two subelements:

- Separate Effects Tests - These are tests designed to study in detail some aspect of a postulated accident or some component.
- Integral System Tests - These tests are designed to study the interactions of the various components of a LWR during all phases of a postulated accident. These tests will help provide the verification of the complex systems code.

1.1.3 Fuel Behavior Research

The purpose of the fuel behavior research is to provide a more "detailed understanding of the response of nuclear fuel assemblies to hypothetical off-normal or accident conditions. This information is then used to develop physical models which are incorporated into fuel analysis codes. The fuel codes, which will eventually become part of the systems codes, are verified through integrated inpile tests."[1]

The fuel behavior research element consists of four principal sub-elements:

- Fuel Codes - covers computer codes designed to calculate the steady-state and transient behavior of fuel rods, including those codes which calculate fission product transport and fuel meltdown behavior.
- Basic Studies - treats the more fundamental safety research on the properties of the cladding and fuel.
- In-pile Tests - covers the actual nuclear testing of experimental fuel rod assemblies.
- Fuel Meltdown/Fission Product Release - treats those aspects of a spectrum of hypothetical accidents in which fuel melting and/ or fission product release may occur.

1.1.4 Analysis Development

The experimental programs lay the foundation for the development and verification of models and computer codes which can be used to compute the behavior of a full-size LWR under postulated accident conditions. The computer codes used by NRC include (1) "evaluation model" (EM) versions which incorporate NRC's ECCS Acceptance Criteria and by virtue of these conservative criteria they are used in the licensing process and (2) "best estimate" (BE) versions which incorporate more realistic (not necessarily conservative) mathematical descriptions of the system.

The Analysis Development program element consists of two principal subelements:

- Systems Codes - these are codes which treat the whole LWR system or a significant fraction thereof.
- Component Codes - these are codes which emphasize specific components (which could be the entire core) for a better understanding of the behavior of these components.

1.1.5 Reactor Operator Safety

NRC recognizes the safety margins of operating LWRs can be further improved through confirmatory research on fire protection, human engineering and the aging evaluation of existing reactor safety components. This is a relatively new element in the NRC LWR safety research program and it is one that is receiving more and more attention.

1.2 EPRI Nuclear Safety Research

As a part of its broad research program covering many phases of nuclear power, EPRI is sponsoring research in:

- LWR safety
- plutonium recycle
- primary system integrity
- reliability and diagnostics
- fuel performance and fuel cycle
- earthquakes and tornadoes
- inservice inspection
- plant chemistry

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An overview of the EPRI LWR safety program was given by W. B. Loewenstein in Nuclear Safety, Vol. 16, No. 6 (November - December, 1975).

NRC and EPRI exchange reports on a regular basis and meet periodically to exchange technical information. NRC and EPRI jointly participate with GE in the BWR blowdown heat transfer program and have agreed in principle to cooperate with Westinghouse in a future FLECHT experimental program.

1.3 NRC International Agreements on LWR Safety Research

The U.S. Nuclear Regulatory Commission is participating in a number of bilateral agreements on LWR safety research. Specifically, NRC has jointly signed bilateral agreements with Brazil, Denmark, France, Federal Republic of Germany, Italy, Japan, South Korea, and Sweden. These bilateral agreements call for an exchange of information in specified areas of LWR safety research. In addition, NRC has jointly signed bilateral agreements with Germany, Japan, and the Nordic countries (Denmark, Finland, Norway and Sweden) and Austria for participation in the LOFT program. A similar set of bilateral agreements has been signed with Austria and Japan and immediately with the Federal Republic of Germany and the Nordic countries on participation in the PBF program. Much of this cooperation was done under the auspices of the International Energy Agency of OECD.

In addition to these agreements, NRC is contributing to Marviken containment response test and Halden fuel irradiation programs.

2.0 CURRENT TECHNICAL STATUS

The following sections describe recent results achieved both in the NRC and in selected other U.S. LWR safety programs. The references to additional documents are not intended to be exhaustive, rather they indicate where additional information may be found. The reader is also referred to References (2) and (3) for additional source material.

2.1 Metallurgy and Materials Research

The recent results achieved under this program element are grouped under the headings of fracture mechanics, operational effects and non-destructive examination.

2.1.1 Fracture Mechanics

Results obtained by the Oak Ridge National Laboratory (ORNL) in the Heavy Section Steel Technology (HSST) Program[4-7] and listed in Table I have shown that for flaws less than half of the wall thickness in depth a pressure of nearly three times the design pressure

TABLE I

SUMMARY OF TEST RESULTS FROM EIGHT 6-IN.-THICK INTERMEDIATE TEST VESSELS

VESSEL NO.	TEST TEMPERATURE (°F)	FLAW DIMENSIONS DEPTH (in.)	LENGTH (in.)	FLAW LOCATION	FRACTURE PRESSURE (ksi)	FRACTURE STRAIN (%)[a]
V-1	130	2.56	8.25	BASE METAL (o)[b]	28.8	0.92
V-2	32	2.53	8.30	BASE METAL (o)	27.9	0.19
V-3	130	2.11	8.50	WELD METAL (o)	31.0	1.47
V-4[c]	75	3.00	8.25	WELD METAL (i)[d]	26.5	0.17
	75	3.10	8.10	BASE METAL (o)	26.5	0.17
V-6[e]	190	1.87	5.25	WELD METAL (o)[d]	31.9	2.0
	190	1.34	5.20	BASE METAL (i)	31.9	2.0
	190	1.94	5.30	WELD METAL (i)	31.9	2.0
V-5	190	1.20	3.75	BASE METAL (i)[f]	26.6[g]	0.25
V-7	196	5.30	18.0	BASE METAL (o)	21.4[g]	0.12
V-9	75	1.20	3.75	BASE METAL (i)[f]	26.9	1.05

[a] OUTSIDE CIRCUMFERENTIAL STRAIN ON CENTER LINE OF VESSEL REMOTE FROM FLAW.

[b] (o): OUTSIDE SURFACE, (i): INSIDE SURFACE.

[c] CONTAINED TWO FLAWS

[d] FLAW WHERE FRACTURE OCCURRED.

[e] CONTAINED THREE FLAWS.

[f] NOZZLE CORNER FLAW.

[g] LEAK-BEFORE-BREAK.

1255 066

must be applied to intermediate size pressure vessels to initiate rapid fracture. For one vessel having a flaw almost 90% of the wall thickness in depth, a pressure of 2.2 times the design pressure was required to drive the crack through the wall. It should be noted that the intermediate test vessels (ITVs) of the HSST program were designed with essentially full-scale thickness so that fracture initiation behavior could be demonstrated with minimal uncertainity associated with scale. Since the terminal fracture was rapid and extensive in most of the ITV tests, the stiffness of the hydraulic loading system relative to that of a real reactor system was of no consequence.

However, two of the intermediate test vessels (ITVs) resulted in a leak without a burst, leading to the suggestion that a study be made of an ITV under sustained load. The results of a pneumatic test of an ITV showed no distinguishable behavior from a comparable hydraulic test, which suggests that the demonstrations of "leak without burst" in the ITVs, both hydraulic and pneumatic, are applicable to the evaluation of the behavior of reactor pressure vessels with similar flaw geometries under sustained load. Overall, the results from all tests, which were predicted from both analytical methods and from small scale models, showed that the design basis safety margin against fracture of reactor pressure vessels is well founded.[5,6]

The fracture analysis methodology is also being verified through a series of experiments designed to test the response of a thick vessel to the quenching action, or thermal shock, resulting from the injection of relatively cold ECCS water into the reactor vessel during a postulated LOCA.[6] Test specimens consisting of 0.15-m-thick (6-in) steel cylinders were fabricated from PWR pressure vessel material. These cylidners were given a quench-only heat treatment to reduce the toughness and they were subjected to thermal shocks more severe than those that would be imposed on PWR vessels in hypothetical LOCAs. These two measures served to further reduce the toughness and to enhance the stress intensity factor (K_I) thereby more closely simulating the behavior of irradiated vessels. To date four thermal shock tests have been run and each cylinder had a different flaw placed in it. The cold simulated ECCS water (either 4 C (40 F) or -23 C (-10 F) for the simulant) flowing at a high rate, through the flawed cylinders (initially at 283 C (550 F)), caused crack extension and arrest in three tests (see Table II), but no crack extension in the fourth test, all as predicted.

NRC is sponsoring work at the Naval Research Laboratory (NRL) to determine whether warm prestressing can preclude crack extension

1255 067

TABLE II

THERMAL SHOCK TEST CONDITIONS

EXPERIMENT	TSE-1	TSE-2	TSE-3	TSE-4
TEST SPECIMEN	TSV-1	TSV-2	TSV-1	TSV-2
TEST SPECIMEN DIMENSIONS, m (in.)				
OD	0.53 (21)	0.53 (21)	0.53 (21)	0.52 (21)
ID	0.24 (9.5)	0.24 (9.5)	0.24 (9.5)	0.24 (9.5)
LENGTH	0.91 (36)	0.91 (36)	0.91 (36)	0.91 (36)
TEST SPECIMEN MATERIAL	A508 CLASS 2	A508 CLASS 2	A508 CLASS 2	A508 CLASS 2
HEAT TREATMENT	QUENCH ONLY FROM 871°C (1600°F)	QUENCH ONLY FROM 871° (1600°F)	QUENCH ONLY FROM 871°C (1600°F)	QUENCH ONLY FROM 871°C (1600°F)
FLAW	LONG AXIAL CRACK, a=11mm (0.42 in.)	SEMICIRCULAR, AXIAL, a=19mm (0.75 in.)	LONG AXIAL CRACK, a=11mm (0.42 in.)	LONG AXIAL CRACK, a=11mm (0.42 in.)
TEMPERATURES, °C (°F)				
WALL (INITIAL)	288 (550)	289 (552)	288 (550)	291 (555)
SINK (INITIAL)	4 (40)	-23 (-9.5)	-23 (-10)	-25 (-13)
SINK (FINAL)	7 (45)	-15 (-4.5)	-15 (4.5)	-19 (-2)
COOLANT	WATER	40 wt % METHYL ALCOHOL, 60 wt % WATER	40 wt % METHYL ALCOHOL, 60 wt % WATER	40 wt % METHYL ALCOHOL, 60 wt % WATER
COOLANT FLOW RATE, m^3/hr (gpm)	59 (260)	114 (500)	59 (260)	114 (500)
COOLANT PRESSURE IN TEST SECTION, kPa (psi)	1520 (220)	917 (133)	917 (133)	1020 (148)
BACK PRESSURE ORIFICE DIAM, mm (in.)	25.43 (1.001)	43.18 (1.700)	43.18 (1.700)	43.18 (1.700)
HEAT TRANSFER COEFFICIENT, W m^{-2} $°C^{-1}$ (Btu hr^{-1} ft^{-2} $°F^{-1}$)	~2800 (~500)	~5700 (~10^3)	~5700 (~10^3)	~5700 (~10^3)
$\gamma(K_I\ K_{Ic})_{max}$	0.74	1.33 (θ = 75°)	1.2	1.29
TIME OF OCCURRENCE OF $(K_I\ K_{Ic})_{max}$, min	~8	~4	2	~5
DURATION OF EXPERIMENT, min	30	30	30	30

when $K_I = K_{I_c}$ (critical fracture toughness for crack initiation) during thermal shock. While fractures have been observed at K levels below those of warm prestress, the data in Figure 3 suggests that, for warm prestress levels more than 120 MPa $\sqrt{m}$, brittle fracture will not occur until the K level exceeds at least 100% of the warm prestress level irrespective of the K_{I_c} of the material.[8]

Sufficient analytical, experimental and theoretical progress has been made in crack arrest in the past several years so that the two crack arrest theories which have been proposed, for which a great deal of confirmatory evidence is already in hand,[9,10] converge to a unified basis. NRC is sponsoring work at Battelle Columbus Laboratories (BCL)[11] and the University of Maryland[12] to study crack arrest both experimentally and theoretically. Two-dimensional, static and dynamic analyses are being developed to predict material behavior related to fast fracture and crack arrest from test specimens and related to analyses of components. A consistent body of experimental data on crack arrest toughness has also been developed, with the result that much confidence can be placed on definition of a crack arrest toughness parameter[9] and standard test specimens and testing methods are now being written. A reference curve for minimum toughness, called K_{IR}, has been developed from the data shown in Figure 4 and was incorporated in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Section III, Appendix C) in 1973. This curve is a representative lower bound of the lowest toughness values, dynamic and arrest, that have been measured.

For loads which impose elastic-plastic stresses in structures, the present analysis method of using linear elastic fracture mechanics (LEFM) is too conservative but useful engineering solutions are being obtained. (Reference 13 is a summary of the methods which are currently used for assessing the fracture toughness of materials under elastic and elastic-plastic conditions.) The goal is for a valid criterion for elastic-plastic fracture mechanics evaluations to be available by the late 1970s. Verification of the K_{IR} curve for unirradiated data[14] has been completed and the verification for irradiated steels is underway. An "irradiated K_{IR}" curve should be available for the late 1970s. Analytical methods for evaluating crack arrest characteristics are under active development and should be complete by 1978.

NRC has also sponsored an initial evaluation of repair welding performed according to "Procedure Number 4, Welding Low Alloy Steels," of Section XI, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The repair weld was made on ITV-7 in preparation for the previously described pneumatic test. The test results showed that the weld metal exhibits toughnesses

1255 069

similar to those of the base metal and the weld repair was tested successfully to 2.15 times the vessel design pressure.[15]

2.1.2 Operational Effects

Factors which can significantly reduce neutron irradiation embrittlement in pressure vessel steel and welds were determined in the late 1960's and early 1970's.[16] The limiting of residual elements, mainly Cu and P, from the composition can reduce embrittlement in reactor vessel steel in nuclear radiation service. The influence of Cu, in particular, is illustrated in Figure 5. The growing body of data on this subject (see Reference 17 for an excellent review) has even resulted in the ability to quantify the effects, as shown by the curves in NRC Regulatory Guide 1.99[18] published in 1976. Significant new information on radiation embrittlement became available by early 1975 on static and dynamic fracture toughness obtained from irradiated 4-inch thick specimens[19,20] which show, as seen in Figure 6[20] that irradiation-condition toughness rises rapidly with temperatures in the transition region and the absolute level of fracture toughness is quite high; although not shown here, the magnitude of the upward shift in transition temperature by these large specimens is quite well predicted by smaller surveillance type specimens.[20]

The reduction of radiation embrittlement by postirradiation heat treatment has been shown feasible in many experimental studies (see p. 197 of Reference 17). Irradiation programs and fracture toughness recovery are both underway to provide the necessary data for improving the confidence in embrittlement safety analyses.

The ultimate solution for elimination of intergranular stress corrosion cracking (IGSCC) in austenitic stainless steels has not yet been found, but some of the prime factors causing it are known.[21-23] Thus, it becomes important to define all the factors and their interrelationships and to extend these findings to field application. A primary reason for the susceptibility of stainless steel to intergranular stress corrosion cracking is sensitization - the depletion of the Cr from solid solution at the grain boundaries as it combines with C to form chromium carbides.

One way to preclude IGSCC in service is to assure that the material is not sensitized and therefore not susceptible to intergranular stress corrosion cracking. An electrochemical test is under development that will permit detection of sensitized material in the field. Already this development has produced such correlations as to permit use of the test for materials qualification tests.[24]

Although stress corrosion cracking cannot be fully controlled in service, it nevertheless is not considered to be an urgent safety consideration.[25] Because it creates such operational costs, however, much research is underway to effect positive solutions.

Corrosion of steam generator tubing can lead to leaking or failure of the tubes. The operational cost loss is so great that much industry and government research is underway to eliminate it or at least mitigate the effects. One NRC program has as its goal the verification of tube integrity in the presence of part-through cracks or tube wastage under realistic operating conditions and accident transients.

Crack growth caused primarily by cyclic stresses in the primary system is addressed in detail in the ASME Boiler and Pressure Vessel Code, Section XI. An effort is currently underway to define those specific loading parameters which cause greatest crack growth questions; thus, reactor safety is not compromised with respect to crack growth.

2.1.3 Non-Destructive Examination

The in-service inspections forming the basis for ASME code evaluations are performed primarily by ultrasonic testing (UT). Thus, UT techniques are being upgraded to yield far better data on flaw characterization. A digital synthetic array processing procedure for improved lateral and longitudinal resolution of ultrasonic images is in late stages of development with excellent progress being recorded.[26,27]

Steam generator tubing is inspected by eddy current (EC) techniques. A new appendix (Appendix IV) for eddy current examination was incorporated into Section XI of the ASME Boiler and Pressure Vessel Code on Inservice Inspection during 1976. Improvements in eddy current instrumentation and characterization of signals are being made primarily using multiparameter and multifrequency approaches.[28,29] The techniques are of high importance because EC inspection is rapid and is the currently accepted method for inservice inspection of steam generator tubing.

Laboratory studies are underway to use acoustic emission (AE) for monitoring of stress corrosion cracking, and initial favorable results have been attained.[30] AE is also being used to monitor for cracking during welding. In this procedure, begun in 1971, transducers are mounted on the component, and as welding proceeds, any cracking that develops as the weld bead cools is detected by

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the system so that the bad weld can be removed and the cause of the weld defect corrected. A weld monitor has been performing well in service in a non-nuclear stop since 1974, and extensive development is underway in nuclear welding shops; the results to date are encouraging.[31]

2.2 Systems Engineering

The Systems Engineering program elements will be discussed in terms of its two major subelements as follows:

- Separate Effects Tests
  - Hydrodynamics of Two-Phase Flow during the LOCA
  - Blowdown Heat Transfer
  - Emergency Core Cooling
- Integral Systems Tests
  - LOFT
  - Semiscale

While a LOCA can be postulated for either a PWR or a BWR, the major emphasis on U.S. LOCA-related research has been on the PWR LOCA, first because the PWR LOCA is more complex than the BWR LOCA and secondly, if the basic thermalhydraulic processes can be understood over a range of scales in the PWR case it should be possible to model them in the BWR case. Thus, for reference purposes Figure 7 is used to show schematically the postulated PWR LOCA. The right-hand side of Figure 7 shows the assumed broken loop while the left-hand side of the figure represents the unbroken loops (which would be three loops for a four-loop PWR).

If the pipe breaks, the high pressure (about 1.5 x 10^7 Pa (2200 psi)), high temperature (about 300°C (600 F)) primary coolant would be rapidly expelled from the primary system. This is the "blowdown" phase of the postulated LOCA and it is during this phase that steam/water or two-phase flows are created. In an attempt to escape through the break, some of the steam may move through the core opposite to the normal water flow.

When the vessel pressure decreases to about 4,000 kPa (600 psi) check valves open causing the injection of large volumes of emergency core cooling (ECC) water into the cold legs (or downcomer). Thus there is an immediate interaction between the ECC water and the steam which can lead to flow and pressure oscillations. The steam may also temporarily block the penetration of the ECC water into the lower plenum so that the water flows around the cylindrical downcomer and out the break. This situation is referred to as "ECC bypass." As the depressurization continues, there will be a decreased steam flow rate so that the ECC water refills the lower plenum (termed the "refill phase") and then refloods the core (termed the "reflood phase"). When refill occurs, the "end-of-bypass" state is said to be reached.

The system engineering research addresses the thermal hydraulic aspects of each of these phases which in turn allows a definition of the model and conditions to be used in assessing the safety margins for the prevention of fission product release.

2.2.1 Hydrodynamics of Two-Phase Flow During a LOCA

If a LOCA were to occur the resulting flow situation could become rather complex as a result of water flashing to steam leading to two-phase flow discharge at the break and steam/water interactions during the injection of ECCS water.

NRC is sponsoring research on steam-water mixing and system hydrodynamics at BCL.[32-33] The purpose of the program is to carry out experimental studies investigating ECCS water penetration and entrainment in small scale models (see Figure 8) of a four-loop PWR. BCL is also developing correlations to describe the ECCS water penetration and entrainment data. The parameters investigated have included system pressure, injection water flow rate, injection water subcooling, annulus gap size, and cold leg steam flow.

BCL has run tests in 1/15-scale steel and transparent models using a quasi-steady state operation of the facility. Two of the principal results of the BCL steel vessel experiments at system pressures up to 410 KPa (60 psig) are[33]

- Except at the upper range of pressures investigated, the fraction of ECCS water penetrating to the lower plenum is a strong function of system pressure, i.e., increasing the system pressure decreases the amount of penetration at a given steam flow (see Figure 9).

- Increasing the water subcooling increases the amount of penetration for a given steam flow (see Figure 10).

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In addition to the BCL work, NRC is funding research at Creare, Inc. to study the fluid and thermal processes governing the ECC injection and vessel refill period of a postulated PWR LOCA. The principal research tool currently being used is a 1/15-scale heated, elevated pressure [860 KPa (125 psia) system operating pressure; 4100 KPa pressure] cylindrical model (see Figure 11) capable of handling various downcomer and lower plenum geometries.[34]

Creare has shown the following:[35]

- the characteristic dimension is something other than the hydraulic diameter (see Figure 12).
- the inlet flow rate of the liquid influences the delivery rate.
- condensation effects are important. (Figure 13 indicates that increased subcooling reduces the ECC water delivery delay time).

Creare has also studies the effects of lower plenum voiding on entrainment. Figure 14 indicates that the critical water level at which no liquid entrainment is expected can be correlated using a Weber number approach.

EPRI has sponsored some steam/water mixing tests at 1/14-scale[36] and 1/3-scale[37] as well as studies in cold leg ECC flow oscillations.[38]

NRC envisions future ECC bypass research at larger scales (at least 2/15-scale).

2.2.2 Blowdown Heat Transfer

The depressurization or blowdown phase of a hypothetical LOCA is being studied at the following two separate effects test facilities:

- Thermal-Hydraulic Test Facility (THTF)[39]

  A large nonnuclear pressurized water loop that incorporates a full-length 49-rod electrically heated bundle and other scaled system components designed to simulate the blowdown phase of a postulated PWR LOCA (See Figure 14A).

  The THTF data are proving very useful in determining time-to-CHF for a full-length bundle. The pretest predictions made with the systems code RELAP4/MOD5[57] were in rather good agreement with the test results (as an example see Figure 15). Improved modeling of energy transport has been achieved as a result of these early THTF tests.[42] Semiscale has also been used to study blowdown behavior. A summary of this work may be found in Reference 41).

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. Two-Loop Test Facility (THTF)[40]

A nonnuclear BWR simulation that incorporates a single full-length electrically heated fuel bundle simulation with two jet pump loops and other system components scaled to produce, on a real time basis, test conditions that are representative of the environment expected in a postulated BWR LOCA (see Figure 16). (The TLTA research is funded by NRC, EPRI and GE).

In the case of the TLTA, tests were run under conditions of a postulated rupture of a steamline or recirculation line in a BWR and the results were compared with calculations based on current BWR LOCA evaluation methods. Figure 17 shows an example of such a comparison. In general the TLTA results have shown that:[42] wrong #

- the actual system depressurizes slower than predicted.
- there is substantially more fluid inventory remaining in the system than predicted.
- there is a noticeable margin (about 315 C (600 F)) in the calculation of peak cladding temperature (PCT) at the end of the blowdown period.

This work has also indicated areas of improvement in the modelling of BWR LOCA phenomena.

Phenomenological heat transfer correlations are being developed to predict the thermal behavior of the system during blowdown. Hsu and Beckner[43] have recently suggested a correlation to incorporate the transient departure from nucleate boiling or critical heat flux (CHF) into blowdown.

In addition to the NRC research, EPRI is sponsoring PWR blowdown heat transfer research.[44,45]

2.2.3 Emergency Core Cooling

To address the reflood phase of a hypothetical PWR LOCA, NRC and EPRI are sponsoring research in the Westinghouse Full Length Emergency Cooling Heat Transfer (FLECHT) facility[46] (see Figure 18). The FLECHT experiments are designed to improve the understanding of heat transfer and water entrainment during reflood. The recently completed controlled low flooding rate (~2.5 cm/sec) test program was based on a cosine axial power shape test series (1.66 peak-to-average) and a skew axial power shape test series (1.35 peak-to-average at the 3-m (10-ft) elevation). The cosine data have been reported in Reference 47 and analyzed in Reference 48. The more recent low flooding rate FLECHT data appear to be consistent with the older FLECHT data obtained at higher flooding rates.

The REFLUX computer program[49] has been developed at MIT under NRC sponsorship to describe all modes of flow boiling during the reflood phase. The REFLUX computer code couples a core hydraulics model with a rod heat transfer model to calculate the temperature history of a rod undergoing the reflooding process. One example of a REFLUX prediction is shown in Figure 19. In general, the REFLUX code has achieved moderate success in predicting FLECHT and Semiscale results but further refinement is needed and expected.

Hsu[50] has developed a best estimate heat transfer correlation for transition boiling based on data from the low void FLECHT forced reflood tests.[51] The results are shown in Figure 20. EPRI has published a summary of the earlier reflood models and experiments.[52]

2.2.4 Integral System Tests

The integral tests provide a means of experimentally determining the interrelationship of the various phases of the LOCA in the presence of the principal system components. NRC is sponsoring LOCA integral test research in the Semiscale and Loss of Fluid Test (LOFT) facilities[53] at the Idaho National Engineering Laboratory (INEL).

2.2.4.1 Semiscale

The Semiscale experimental program "consists of a continuing series of thermal-hydraulic experiments having as their primary purpose the generation of experimental data that can be applied to the development and verification of analytical models describing LOCA phenomena in water-cooled nuclear power plants. Emphasis is placed on acquiring system effects data that characterize the most significant thermal-hydraulic phenomena likely to occur in the primary coolant system of a nuclear plant during the depressurization (blowdown) and emergency core cooling (ECC) phases of a LOCA. The experiments are performed with a test system that simulates the principal physical features of a nuclear plant but which is much smaller in volume. Nuclear heating is simulated in the experiments by a core consisting of an array of electrically heated rods, each of which has dimensional and heat flux characteristics similar to those of nuclear fuel rods."[53]

In addition to providing research information in its own right, the Semiscale experiments are closely related to the LOFT program. The Semiscale experiments can be thought of as experimental precursors to the LOFT experiments.

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Figure 21 shows the Semiscale Mod-1 facility. The volume of the Semiscale Mod-1 primary system is about 1/50 of that of LOFT and about 1/1500 of that of a typical large commercial nuclear plant of current design. Both Semiscale and LOFT has a 1-1/2 loop configuration, i.e., they each have an intact loop with a volume three times that of the 1/2-loop portion or "broken" loop. This greater volume serves to represent three intact loops of a four-loop nuclear plant. The broken loop provides for system blowdown through two rupture assemblies that represent the upstream and downstream ends of a sheared pipe in the primary coolant system.

The Semiscale Mod-1 vessel can house either a reactor simulator or a core resistance simulator. The core resistance simulator is an orificed structure that simulates the resistance of the LOFT core simulator. The reactor simulator consists of 40 electric heater rods each 1.67-m (5.5-ft) long (same as LOFT length) and designed to operate at a total power of 1.6 MW. The dimensions of the principal heat transfer surfaces in the Semiscale Mod-1 system (core heater rods and steam generator tubes) are the same as the LOFT counterparts, but the quantity of these surfaces is reduced to maintain the Semiscale-to-LOFT system volume ratio.[53]

Semiscale is a virtual data factory in the U.S. thermal-hydraulic test program. During FY 1976 twenty-six tests were performed with the Mod-1 Semiscale system. Fourteen of these tests were blowdown experiments[41] initiated from full power, steady state conditions at 1.55×10^7 Pa (2,250 psig),[54,55] and 12 were reflood experiments initiated with the system at containment pressures. Of the 14 blowdown tests conducted, seven were to obtain specific information on time to CHF and post-CHF heat transfer, two were specified NRC Regulatory Standard Problems, four were baseline ECC tests (emergency core cooling parameters were varied to achieve more representative time to initiate core reflooding by compensating for the nontypical downcomer surface to volume ratio), and one provided results to evaluate the effect on overall system performance of the lower plenum depth. Of the 12 reflood experiments, eight were forced feed reflood tests during which the core reflood rate was controlled by forcing simulated emergency core coolant into the core inlet at a constant rate while the four remaining tests were gravity feed reflood tests during which the core reflood rate was determined by the interaction between the water in the vessel downcomer, the core hydraulics and heat transfer, and the dynamic response of the system. For the gravity feed reflood tests the ECC water was injected into the lower plenum.

The Semiscale isothermal blowdown tests (performed with no electrically heated core and completed in July 1975) have been summarized in Reference 56. These tests provided a data base for comparison of the isothermal blowdown characteristics with the LOFT nonnuclear system.

The blowdown heat transfer tests have been summarized in Reference 41. Analysis of the results of these tests showed that the measured system response was consistent with RELAP4/MOD 5[57] calculations. One such comparison based on blowdown heat transfer test S-02-5 is shown in Figure 22. In this case, the predicted maximum rod clad temerature was 980 C (1,800 F) and the measured volume was 970 C (1,780 F). For the high power rod hot spots, departure from nucleate boiling (DNB) occurred at 0.5 s into Test S-02-5 as predicted. On the low power rod hot spots, DNB occurred between 0.6 and 4.0 s after rupture, whereas RELAP4 predicted DNB at 0.5 s followed by immediate rewet and then another DNB at 1.5 s after rupture. The measured peak clad temperature on the low power rods varied between 650 C (1,200 F) and 900 C (1,650 F), whereas the predicted peak temperature was 810 C (1,500 F).

The forced feed reflood tests in Semiscale Mod 1 were compared with the calculated performance obtained with the REFLUX[49] and SUPERH computer codes.[55] These codes were developed for analysis of forced feed tests and are not applicable to gravity feed systems. The gravity feed tests were analyzed with the FLOOD4 code which couples the system hydraulics to the core region.[54]

Figure 23 shows a comparison of the Semiscale rod cladding temperature response and REFLUX calculations at several elevations. A summary of the overall comparisons has been given.[55]

- The REFLUX computer code calculated the correct trend of the peak cladding temperature as the system parameters were changed; however, in some tests the magnitude of the calculated peak temperature was significantly lower than that observed.

- The SUPERH computer code also calculated the correct trend of the cladding temperature response as the separate system parameters were changed although the peak temperatures were calculated to be somewhat lower than actually occurred.

- The FLOOD4 computer code calculated slightly lower peak rod temperatures during gravity feed reflood tests, resulting from the calculated core inlet flow being higher than measured.

From tests in Semiscale, researchers have discovered the importance of the local mixing effect of unheated rods (analogous to a control rod thimble) on the delay of transient CHF during blowdown. The presence of three unheated rods in a test section of 40 rods changes the time delay of CHF from 0.5 s to 3 s.

Testing is underway to evaluate alternate ECCS concepts. Currently the Semiscale program is under major redirection to reflect the configurational aspects of the Westinghouse Upper Head Injection (UHI) concept.

2.2.4.2 LOFT

LOFT is an integral nuclear reactor test facility which has been designed to simulate, as nearly as possible, all of the important effects that are anticipated to occur during a postulated LOCA in a PWR.[53] LOFT will provide data to help evaluate the adequacy of and to improve the analytical methods currently used to predict (1) the response of a PWR to a postulated LOCA, (2) the performance of engineered safety features (ESF) with particular emphasis on ECCS, and (3) the quantitative margins of safety inherent in the performance of the ESF.

A schematic of the LOFT primary coolant system blowdown loop is shown in Figure 24, and the reactor vessel and internals are shown in Figure 25. The LOFT coolant system has been designed to be representative of a four-by-four PWR primary loop. The operating parameters for the LOFT primary coolant system, such as system pressure, reactor outlet temperature, and core enthalpy rise, were selected to be similar to the parameters utilized in the current design for PWRs. The core length was made sufficiently long so that the maximum PWR enthalpy rise, could be achieved. On the basis of these and other considerations the nominal core power level is 55 MW(t) and the peak linear power density is 620 kW/m (19 kW/ft). The nuclear core is approximately 1.7 m (5.5 ft) long and .6-m in diameter, and contains 1,300 fuel rods and four control assemblies typical of PWRs.

Four isothermal (nonnuclear) blowdown experiments have been performed with the LOFT system and the results are summarized in Reference 58. The LOFT test data may be found in References 59-62. Of the four LOFT experiments, three were double-ended cold leg break simulations initiated from isothermal conditions of 15.5 MPa (2,250 psig) and one was a hot leg break simulation initiated from isothermal condition at 8.9 MPa (1,322 psig).

A preliminary evaluation of the LOFT results indicates that the effect of physical scale on the parameters controlling the system hydraulic response during a nonnuclear LOCA are understood. This is most apparent from the fact that LOFT data essentially overlay data from counterpart tests in the Semiscale system (see Figures 26-28). Relatively good agreement between the RELAP4 calculation and the measured test parameters provides additional confirmation of this understanding--thus there is evidence that the preservation of the power-to-volume ratio is a good scaling criterion for the blowdown of both Semiscale and LOFT.

The LOFT results were equally gratifying in terms of systems performance capability. Performance characteristics were repeatable, pressure suppression loads were considerably less than expected, the downcomer hot wall delay was negligible, and there was evidence of two-dimensional effects in the downcomer during the ECC delivery phase of a LOCA experiment.

By the end of 1978 it is planned that the remaining two nonnuclear LOCA/ECCS tests will be completed in the LOFT facility and preparations will be well underway for the first nuclear test to be conducted in 1979.

2.2.5 Additional LOCA-Related Research

NRC is sponsoring a BWR Mark I suppression pool test program at the Lawrence Livermore Laboratory (LLL) to better quantify the response of the suppression pool to the dynamics loads during a hypothetical LOCA. Figure 29 shows the 1/5-scale, 90° sector of the torus. The initial test program will involve using an air supply to pressurize the dry well and hence the wet wall test sections. EPRI is also sponsoring work in this area (cf., e.g., Reference 63).

EPRI is sponsoring two-phase pump performance program to obtain experimental information in which to base a more refined and analytical model for determining pump speed during LOCA conditions.[64] In this program, a one-fifth scale model reactor coolant pump will be tested at Combustion Engineering in both steady-state and transient two-phase mixtures of water and steam over a range of operating conditions representative of postulated LOCAs.

2.3 LOCA Computer Code Development

This research includes improvement of existing codes, development of advanced systems codes and development of component codes. The fuel modeling codes are discussed in Section 2.4, and the code verification effort is illustrated in Figure 2 (see also Section 2.2.4). The overall goal is to have a complete verified LOCA advanced systems code in 1981.

2.3.1 Improvement of Existing Codes

Top priority is being given to the improvement of the present intermediate level system code, RELAP4.[57] RELAP4 exists in two principal versions: a "best estimate" model which is used to describe realistically the accident phenomena and the "evaluation model" which incorporates the conservative assumptions required for licensing analyses. RELAP4/Mod 5,[57] which contains an improved description of the blowdown phase of a hypothetical PWR LOCA, was released in Summer 1976 and it predicts Semiscale and LOFT results with reasonably good accuracy as discussed in Section 2.2.4.

RELAP4/Mod 6, which extends the RELAP improvements to the reflood phase of a postulated PWR LOCA, will be released to the Argonne Code Center (ACC) by Summer 1977. The BWR reflood analysis capability will be covered by RELAP4/Mod 7 which is to be released to ACC during the Winter 1977/1978.

2.3.2 Advanced System Codes

Because of the complexity of the mathematical description of the two-phase (steam/water) flow and heat transfer and other physical processes occurring during a postulated accident, NRC is examining alternative approaches to arrive at a satisfactory description of the overall system behavior during an accident. These different approaches include using alternate model formulations or solution procedures or both.

Brookhaven National Laboratory is developing the THOR code[65] which is an advanced computer code for predicting the accident-induced, one-dimensional thermal hydraulic transients in LWRs including (1) the capability to account for thermodynamic non-equilbrium between phases, (2) unequal phase velocities, (3) axial variations in power generation, (4) initial steady state conditions, and (5) components. This code is intended to serve as the next generation licensing code. By Fall 1976, BNL completed the THOR system self-initalization, break flow models consistent with those employed for interior regions, and numerical coupling of the distributed and the lumped parameter regions. During 1976 a parallel path effort was chosen to insure that the next generation licensing code will be timely and will satisfy licensing requirements. Hence, an initial effort is being sponsored at INEL on the development of the RELAP-5 code to complement BNL's effort on THOR.

LASL is developing the TRAC advanced system code[66] to describe LOCA phenomena. TRAC will differ from THOR in two aspects: (1) TRAC will utilize the most appropriate techniques for solving the basic equations of non-equilibrium two-phase fluid dynamics, dependent on the degree of steam/water coupling in various system components, and (2) TRAC will employ multi-dimensional descriptions of those reactor components which require it (e.g., plenums, downcomer, and the core). In FY 1976 LASL completed the TRAC (1) calculational strategy (data transfer, and storage organized and coded in modular fashion), (2) implicit solution technique for multi-dimensional views of the whole reactor vessel, (3) drift flux non-equilibrium module with wall heat transfer, and (3) two-dimensional transient simulation of ECC penetration and bypass in a PWR downcomer, including condensation effects.

Work is also underway on improved containment analysis techniques. This work will be correlated with the BWR pressure suppression testing underway at LLL (see Section 2.2.5). The first version of the BEACON containment system code[55] will be released in 1978.

2.3.3 Component Codes

A number of component codes are under development through NRC programs. The purpose of the component code development is to model in greater detail the behavior of the various single components of a reactor. In this way, the simplifications made in developing a systems code can be assessed in terms of their degree of accuracy. The following component codes are under development:

SCORE[67] - this is a "core" component code which is used for the multi-dimensional description of core flow under the assumption of a homogeneous, thermal equilibrium two-phase mixture. The code will be documented during 1977 and released to the Argonne Code Center (ACC).

COBRA[68] - this is another "core" component code which can also model a significant assembly or even a single (hot) channel. The latest version is COBRA-DF which includes the drift flux and full non-equilibrium in axial and lateral flow components. During 1977 this code will be applied solely to modeling the multi-dimensional aspects of UHI behavior in the entire PWR vessel.

A comparison of the results from the COBRA-4 and SCORE codes is shown in Figure 30. Both codes give similar core flow patterns at $\sim$ 300 msec after blowdown. The importance of this agreement is two-fold:

1. An agreement between two completely independent codes indicates a sign of validity of the code predictions even without data verifications.

2. At early blowdown, the flow velocity in the void region at the middle of the core is almost lateral, because of the difference in subcoolings between neighboring channels. This lateral velocity strongly affects the delay of transient CHF and the post-CHF heat transfer. A good code can explain phenomena which occur so fast in a hypothetical LOCA that they can hardly be measured in an experiment.

K-TIF[69] - This advanced component code, which is a derivative of the KACHINA code,[70] is being applied to detailed, multi-dimensional modeling of PWR downcomer flow during the ECC injection period.

K-FIX[71] - This advanced component code, which is another derivative of the KACHINA code, is being used to explore the numerical simulation of the details of the transport processes across steam/water interfaces. The first version has been released to ACC. K-FIX will also figure in INEL's development of the BEACON advanced containment code.

Both K-FIX and K-TIF will be continually updated. These codes represent NRC's frontier of knowledge in the numerical simulation of multi-dimensional, transient two-phase flow and they use the most sophisticated models available to NRC. The advanced systems code, TRAC, profits from these detailed analyses since it retains modeling of only those effects which detailed component codes show to be important.

SOLA-FLX[66] - This is a hydro-elastic code which permits a more realistic description of the blowdown-induced loads on PWR vessel internals.

SOLA-DF, SOLA-PLOOP[72] - These two codes are experimental predecessors of TRAC. They will be released to ACC in 1977.

2.4 Fuel Rod Behavior

This research includes basic studies on the constituents of the fuel rod (fuel, gap, and cladding) studies on integral fuel rods, and definitions of fuel failure limits and consequences. All of this information will be incorporated into computer codes which can be used to assess the safety margins in fuel rods exposed to hypothetical reactor accident conditions.

2.4.1 Basic Studies

In response to the NRC ECCS Acceptance Criteria[73] the basic studies have focused on Zircaloy cladding oxidation, Zircaloy mechanical behavior (including deformation on bursting which could influence the ability to meet the coolable geometry criterion), UO_2 pellet properties, gap conductance and decay heat. The last three items have a strong influence on the fuel behavior for such postulated accidents as the LOCA. A useful compilation of fuel rod material properties has recently been made.[74]

2.4.1.2 Zircaloy Oxidation

Considerable attention has been focused on the rate of oxidation of Zircaloy, in the presence of steam in order to calculate the hydrogen generation, the extent of cladding oxidation, and the heat generation from oxidation of the Ziracloy by steam during a hypothetical LOCA.

Figure 31 shows the reported results obtained in the three years of Zircaloy-steam reaction kinetics research. For comparison, the oxygen constant obtained from the Baker-Just equation[75-77] is also shown.

From Figure 31 it is evident that the oxidation rate at 1200 C (2192 F) is only one-half to two-thirds the rate obtained in the Baker-Just equation. This in turn implies less heating and less hydrogen generation than predicted by the Baker-Just equation for Zircaloy steam reactions.

Since oxygen-contaminated Zircaloy may be brittle, it may not withstand either the forces calculated to occur during a hypothetical LOCA or the thermal shock caused by the quenching action of the cooler ECC water. The ability of a partially oxidized fuel rod cladding to withstand these stresses depends on the oxygen content of the unoxidized portion of the cladding.[83] The oxygen penetration is usually calculated from the diffusion coefficients obtained by Mallett et al[84] some years ago. New work obtained by Cathcart and Perkins[85] at the Oak Ridge National Laboratory is shown in Figure 32 along with the results of Mallett et al[84] Debuigne,[86] Schmidt et al,[87] and Dechamps et al.[88] The diffusion coefficient of oxygen in beta Zircaloy is shown to be only half of the previous value. The activation energies are comparable. By comparison, the diffusion data for unalloyed zirconium[86] are still lower.

2.4.1.3 Zircaloy Mechanical Properties

Information on the mechanical behavior of Zircaloy cladding is of importance in determining the ability of the cladding both to contain the fission products and to avoid appreciable flow blockage. Some recent results[89,90] show that when tested under more realistic conditions, Zircaloy exhibits less deformation before failure than previous work[91] indicated. Axial restraint, testing in steam and reduction in the internal gas volume will each reduce the clad ballooning.

The results of Chapman et al[90] at ORNL are shown in Figure 33 in comparison with other existing data.[92-94] The ORNL work, based on more realistic modeling of the internal gas volume, internal pressure, and external environment, indicates significantly less circumferential strain at failure than previous tests conducted in inert atmosphere.

Two of the tests (marked A), which were run in argon instead of steam, indicate that a steam atmosphere greatly reduces strain in the high strain regions. Examination of the cladding after the tests revealed that deformation before the onset of plastic instability was comparable for rods heated in steam or argon but that the steam limited the amount of strain to failure after onset of plastic instability, suggesting that a surface effect may be governing.

The burst temperature vs. pressure of Chapman et al[89] agrees with the data of Hobson et al[91] shown in Figure 34. The tests showed that the large deformation is very localized extending only a few rod diameters at most. The location of the burst is very sensitive to the highest local temperature. These observations appear to have an important bearing on the ballooning and subsequent flow blockage experienced by nuclear fuel rods; i.e., axial and circumferential temperature distributions will tend to concentrate significant ballooning in highly localized hot spots on one side of the rod.

Axial restraint during the deformation also reduces the circumferential strain as shown in Figure 35 developed from recent experiments conducted by Kassner et al.[90] These biaxal burst tests were conducted in argon on Zircaloy as-received and thus show a higher strain than would be expected from tests run with steam. Kassner et al have identified superplasticity as a source of the high circumferential strain causing excessive ballooning.

The results discussed above are all obtained on unirradiated Zircaloy cladding. An NRC sponsored program at BCL[95] is conducting similar experiments on irradiated rods from commercial reactors to determine the influence of prior irradiation on the cladding response. The results to date obtained on PWR rods indicate that ductility, as determined by axial elongation, is the slowest of the measured properties to recover during a rapid temperature transient. The low ductility gives less ballooning.

2.4.1.4 Pellet Properties

Programs are being conducted at ANL,[90] INEL and ORNL to investigate and model the transient fission product release from irradiated UO_2 pellets. The programs have the dual purpose of understanding the fission gas contribution to the total pressure inside the fuel rod during an accident and determining the amount and chemical and physical nature of the fission products transported from a failed fuel rod.

Much of the current work on pellet properties (e.g., densification, cracking, restructuring, etc.) is sponsored by EPRI and the nuclear industry.[96] A useful compilation of UO_2 physical and mechanical properties used in fuel code development is given in reference 74.

2.4.1.5 Fuel Rod Thermal Performance

The stored energy in the fuel rod at the onset of an accident has a large influence on the magnitude of thermal transient. Prediction of the stored energy necessitates knowledge of the pellet/cladding gap conductance. As fission gas is released, the composition of the gas in the gap changes from helium which has a high thermal conductivity, to a mixture containing helium, xenon and krypton, which has a lower thermal conductivity. Comparisons of FRAP-S2[97] predictions with

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experimental fuel centerline temperature data[98-100] obtained with fuel assemblies tested in the Norwegian Halden reactor at various burnups to 15,000 MWD/MtU are shown in Figure 36.

The code contains an empirical gap conductance correlation assuming pellet cracking and relocation resulting from burnup.

The effects of burnup do not become noticeable in the experimental data until about 1,000 MWd/MtU of burnup and become constant after 10,000 MWd/MtU.

Experiments to measure the gap conductance of LWR design test rods have been performed in the Power Burst Facility using a thermal oscillator technique.[53,101,102]

2.4.1.6 Decay Heat

The NRC ECCS Acceptance Criteria[73] require the decay heat to be calculated on the basis of an American Nuclear Society standard[103] with a 20% uncertainty factor added. NRC-sponsored research is aimed at providing more accurate data[104,105] and analysis[106] on the decay heat. Figure 37 compares the latest decay heat evaluation[106,107] with the ANS standard. It is evident that the more recent (with averaged standard deviation of 5%) work shows there will be less heat available to increase the fuel temperature during a hypothetical LOCA.

2.4.2 Fuel Rod Performance under Transients

The preceding sections have described programs in which only portions of a fuel rod or of a possible accident sequence have been studied. The results of these programs are being utilized in the development of the fuel modeling codes FRAP-T[108] and FRAP-S.[97] The verification of the codes is accomplished through comparison with the integral tests conducted in-reactor on single or multiple fuel rods. The Power Burst Facility (PBF)[53] at INEL is an important source of this information. The Power Burst Facility is illustrated in Figures 38 and 39. PBF was designed to provide experimental data which will aid in defining the behavior of nuclear fuels in off-normal operating conditions. The testing program includes tests which will extend to the point of fuel cladding failure including fuel-coolant interactions under postulated accident conditions. These LWR safety tests will be performed in PBF by testing single fuel rods and clusters of test fuel rods in a central test space in the core. The PBF reactor can be operated in three modes: (1) a steady state mode with power levels up to 40 MW, (2) a natural power burst mode which yields reactor periods as short as 1.3 ms and peak powers as large as

240 GW, and (3) a shaped burst mode (approximating a square wave power shape) resulting in energy generations up to 1500 MW-sec. Because of this versatility, PBF can provide the power and energy densities in test fuel rod clusters that are analytically derived for a broad spectrum of postulated reactor accidents. The facility contains a , essurized water flow loop that permits the control of the coolant conditions of flow rate, temperature, and pressure in the test fuel rod environment.[53]

As of January 28, 1977 twelve experiments of the power-cooling mismatch type have been conducted utilizing a total of 38 highly instrumented fuel rods. Sixteen of the rods were previously irradiated to burnups of approximately 16,000 MWD/MtU. Three types of experiments have been run in PBF: gap conductance (see Section 2.4.1.5), flow coastdown and power ramp. Additional tests are planned to include reactivity-initiated accidents (RIA) and LOCA simulations. Approximately one-third of the basic test series has been completed.

The flow coastdown or power ramp experiments are intended to show what might happen to a fuel rod in which the heat flux from the rod exceeds the ability of the coolant to remove the heat. This results in the rod passing thermal hydraulically into the film boiling region from nucleate boiling. The time-dependent fuel rod power, measured cladding surface temperature, axial length and fuel centerline temperature in response to such an experiment are shown in Figure 40. A comparison of the predicted and measured fuel rod parameters is given in Table III for several thermal hydraulic correlations. The code predictions using the combination of the W-3[110] and Groeneveld 5.7[111] correlations were found in this first test to be closest to measured values and have been used for subsequent pretest predicitions. Additional test results may be found in references.[54-57, 113-115]

The results of these tests have been summarized in a recent paper by Quapp and McCardell of INEL.[116]

> "The tests to date have been conducted at powers of 500 to 800 W/cm and have resulted in film boiling periods ranging from a few seconds up to more than ten minutes. Cladding surface temperatures experienced have been as high as 1400°C. Fuel temperatures have exceeded the UO_2 melting temperature. Extensive oxidation of the zircaloy surface has resulted from metal water reaction. Additionally, following cladding collapse onto the fuel pellets in the areas of high temperature, an internal reaction between the UO_2 and the zircaloy has contributed to additional cladding embrittlement. The combined effects of internal and external cladding

TABLE III

COMPARISON OF MEASURED, ESTIMATED, AND CALCULATED FUEL ROD BEHAVIOR

Parameter	Maximum Measured During Test	Estimated From Fuel Rod Posttest Condition	FRAP-T1 Predictions Using These CHF and Post-CHF Correlations*					
			W-3 and Tong-Young	W-3 and Groeneveld 5.7	W-3 and Groeneveld 5.9	B&W-2 and Tong-Young	B&W-2 and Groeneveld 5.7	B&W-2 and Groeneveld 5.9
Cladding surface temperature at 25 inches (°F)	1,530	2,200	1,875	2,150	2,525	1,850	2,175	2,520
Maximum cladding surface temperature	—	2,560	2,300	2,450	2,900	2,100	2,350	2,800
Fuel centerline temperature at 29 inches (°F)	4,155	Less than UO_2 melting (5,144)	3,800	4,250	4,550	3,800	4,275	4,525
Maximum fuel centerline temperature (°F)		Less than UO_2 melting	5,600	5,950	6,300	5,200	5,800	6,100
Fuel rod internal pressure (psig)	1,770	Less than 2,080	1,630	1,630	1,650	1,630	1,630	1,630
Axial length change after CHF (mils)	90	Not possible	220	315	510	110	230	285

*The following references provide more information on the indicated correlations:
B&W - 2 - Ref. 109
W - 3 - Ref. 110
Groeneveld 5.7 and 5.9 - Ref. 111
Tong-Young - Ref. 112

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> attack have resulted in failure of the fuel rods in several of the tests following reaction shutdown. The highly embrittled cladding fails a few minutes following shutdown apparently as a result of the small forces induced by the flowing coolant."

Quapp and McCardell[116] point out the significant PBF result that "none of the tests have failed at power in spite of the presence of molten fuel and severe mechanical interaction. The nature of the failures in these tests is such that failure propagation or severe fuel coolant interactions are considered unlikely during an abnormal event in a power reactor resulting in conditions similar to those of the current test."

2.4.3 Fuel Meltdown Research

In addition to the PBF fuel damage studies, NRC is also sponsoring research on phenomena associated with hypothetical fuel meltdown accidents. A good background review report has been prepared by NRC by Sandia Laboratories,[117] and an overall summary of NRC work has been prepared by DiSalvo.[118] The basis for these studies may be found in the Reactor Safety Study.[119] "The only way that potentially large amounts of radioactivity could be released is by melting the fuel in the reactor core....To melt the fuel requires a failure of the cooling system (such as a failure of all the ECCS after a LOCA) or the occurrence of a heat imbalance that would allow the fuel to heat up to its melting point, about 5000°F." Using probabilistic techniques it is estimated that the total probability of melting the core is about one in 20,000 per reactor per year. The Reactor Safety Study observes: "It is significant that in some 200 reactor years of commercial operation of reactors of the type considered in the report there have been no fuel melting accidents."

NRC sponsors several programs to address aspects of the hypothetical core meltdown accident and the release of fission products.

Program	Responsible Laboratory
Fission Product Release from LWR Fuel	Oak Ridge National Laboratory
Molten Fuel Interactions	Sandia Laboratories
Natural Convection in Molten Pools	Ohio State University
Steam Explosion Phenomena	Sandia Laboratories
Transient Fuel Response and Fission Product Release	Argonne National Laboratory

Vapor Explo... Tiggering	Argonne National Laboratory
Fission Prod... ...ansport Analysis	Battelle Columbus Laboratories
Analysis of the Physical Events Associated with Degraded Reactor Accidents	Battelle Columbus Laboratories

Exemplary is the work[113-122] proceeding at Sandia Laboratories to identify the chemical and physical processes which occur when molten core materials contact concrete, specifically the gas production rate and the penetration rate of a melt. To examine the thermal aspect of the decomposition of concrete, Sandia has subjected specimens to heat fluxes on one surface of 28-122 W/cm^2 in the Radiant Heat Flux Facility and 123-180 W/cm^2 in the 2MW Plasmajet Facility.[122] Both basalt and limestone aggregates are used yielding concrete with 5 weight percent (4 w/o H_2O, 1 w/o CO_2) and 20 weight percent (4 w/o H_2O, 16 w/o CO_2) volatiles, respectively. Preliminary observations and conclusions include:

- The principal thermal erosion mechanism is quiescent melting of the concrete matrix (primarily silica) with little spallation. Thermal shock effects do not appear to be important.
- The rates of surface erosion (typically 1 cm/min) appear to be directly dependent on surface heat flux for a given type of concrete (Figure 41).
- In tests employing similar heat flux, no significant or unexplainable differences in erosion rates were observed between radiant and plasmajet heating.

Differential thermal analysis and thermogravimetric analysis have been used to identify up to four distinct decomposition reactions in the 100-1200°C region involving dehydration or decarboxylation.

At Ohio State,[123,124] an experimental study of the transient response of a horizontal fluid layer subjected to a step change in internal energy generation has been conducted to determine the times scales for the development and decay of natural convection. For both cases, the time required for the development of the final steady state is determined by measuring the temperature response of the fluid with a thermocouple probe. The time required for the development of the maximum temperature difference in a horizontal layer with internal generation is correlated with the Rayleigh number by (Figure 42).

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$$Fo = 11.577\ Ra^{-0.213}.$$

The time required for the complete decay of the maximum temperature difference of steady convection at a given Rayleigh number when internal energy penetration is suddenly stopped is given by

$$Fo = 11.956\ Ra^{-0.215}.$$

In both of these equations, Fo is the Fourier number for the layer, and Ra is the Rayleigh number. The equations will find general application in analyzing the post-accident heat removal (PAHR) situation in nuclear power reactors.

2.5 Reactor Operational Safety

NRC is expanding its research into reactor operational safety matters, specifically fire protection, aging evaluation and human engineering. This research is an integral part of the overall NRC reactor safety research program.

2.5.1 Fire Protection and Aging Evaluation

At present the fire protection program is focused on the evaluation of the effectiveness of cable tray separation in preventing the spread of a cable fire to redundant trays. Screening tests were conducted to determine which of the currently utilized generic cable types are most susceptible to the spread of a cable tray fire, and two full-scale cable tray tests completed. The results of these tests are being evaluated and a report will be issued shortly. Cable insulation, cable jacket and coating materials are continually being developed by manufacturers and new products introduced. The performance of these materials when used in Class 1 equipment and systems as well as the change in the properties of those materials with age need to be evaluated. Performance evaluation of the materials in use today is primarily by separate effects testing oriented towards single component evaluation and where aging is considered, only accelerated aging methods are being utilized.

The future NRC effort will be aimed at verifying performance of safety class materials and equipment in systems typically found in power plants and under the conditions that they will encounter during their respective design basis events. Tests are being designed now for the verification of all aspects of NRC Regulatory Guide 1.75[125] including cable conduits, fire breaks, fire barriers and penetration fire stops. Future plans include program in the following areas:

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1. Evaluation of the effectiveness of coating materials.

2. Evaluation of aged materials.

3. Development of small-scale cable fire tests to predict total system performance.

4. Evaluation of the vulnerability of safety class equipment to non-electrically initiated fires.

5. Evaluation of fire and smoke detection systems.

6. Evaluation of the vulnerability to fire of Class 1F equipment other than cable.

The qualification test program is to provide technical information for improved aging qualification testing of safety class equipment. Areas to be addressed include improvement of aging models, nuclear source definition, synergisms, the performance indicators to be monitored during qualification testing, failure definition, allowable thermal and nuclear radiation flux gradients, test sample preparation and quality control, mounting and corrections to test samples, chemical and steam flow rates, impingement effects and vibration. Research programs are currently underway at Sandia Laboratories.

2.5.2 Human Engineering

The risk assessment analysis as presented in Report WASH-1400[119] identified the role that human interaction and intervention play in the unavailability of safety systems and components in nuclear power plants. This conclusion was based in part on a preliminary human factors analysis of a typical PWR nuclear power plant control room performed earlier by Swain[113] of Sandia. He observed three general categories of human factors problems, (1) human engineering design in deficiencies in control rooms (2) shortcomings in training, and (3) poor format for written operating instructions.

Shortcomings in human engineering were also listed in the major recommendations in the report[126] to the American Physical Society by the study group on light water reactor safety as follows:

> "Human engineering of reactor controls, which might significantly reduce the chance of operator errors, should be improved. We also encourage the automation of more control functions and increased operator training with simulators, especially in the accident-simulation mode."

1255 092

Safety research programs by the Nuclear Regulatory Commission and EPRI directed at providing answers for these concerns are in progress. The NRC safety research program goals for Human Engineering are to (1) reduce the potential for human error by identifying human factors improvements in operator training, the design of control consoles and optimum use of automation of controls for safety system; (2) develop an actuarial human factors data base to enable more accurate risk assessment; and (3) provide the technology base for developing guides and standards for controls and control room designs.

Efforts to identify human errors and develop a statistical data base for human reliability have been initiated by NRC through the Licensee Event Reports (LER), whose format has been modified to categorize human errors. A contract has also been written with Sandia Laboratories for the preparation of a handbook of human error rates in nuclear power plants. Improved operator training in order to reduce human errors is being studied by both NRC and EPRI. The first phase of an EPRI contract with the General Physics Corporation for a "Performance Measurement System for Training Simulatory" is scheduled to be finished in May 1977. A "Human Factors Review of Nuclear Power Plant Control Room Design" by Lockheed Missile & Space Systems Division for EPRI was completed in December, 1976. "An Analysis of Control Room Displays and Operator Performance" by Aerospace Corporation under a NRC contract was finished in February 1977. Industrial guides and standards are being drafted for safety related operator actions (ANSI No. 660), for the design of display and control facilities (IEEE P. 566) and for the design of control rooms (IEEE P. 567).

3.0 CONCLUSIONS

The water reactor safety research efforts in the primary system integrity area have (1) validated the conservatism of the reactor pressure vessel design by testing on nine intermediate vessels; (2) developed the remedy for curing radiation damage by using thermal annealing techniques; and (3) improved the nondestructive inspection techniques used for flaw detection by digitizing the signals emanating from acoustic emission and ultrasonic testing. Future research in this area is directed toward providing additional verification of NRC analytical techniques and further improvements in NRC inspection techniques.

NRC is expanding its reactor operational safety research, specifically concentrating on fire protection and human engineering research. The objective of this research is to further reduce the already low risk associated with the operational aspects of commercial reactors.

The experimental and analytical research programs which address the fuel behavior and thermalhydraulic behavior of reactor plants have confirmed, either quantitatively or qualitatively, a number of conservatisms used in current licensing assumptions. As examples, the decay heat, the Zircaloy oxidation rate and the ECC bypass rate have been found to be lower than assumed while the post-CHF heat transfer rate appears to be higher than assumed. Based on the experimental data from this research we have developed more realistic correlations to predict these phenomena. In addition, this research has also greatly improved the understanding of the hypothetical LOCA through comparisons of RELAP4 computer code predictions with the test data from such facilities as Semiscale and LOFT. To further improve both the physical understanding of LOCA behavior and predictive capabilities, NRC is developing more sophisticated analytical formulations of the computer codes with the goal of having operational versions in December 1977.

In general, the water reactor safety research program has greatly expanded the data base in such areas as fracture mechanics, transient boiling heat transfer and two-phase flow analyses. Special credit is due the individual investigators for their scientific and engineering achievements in these areas.

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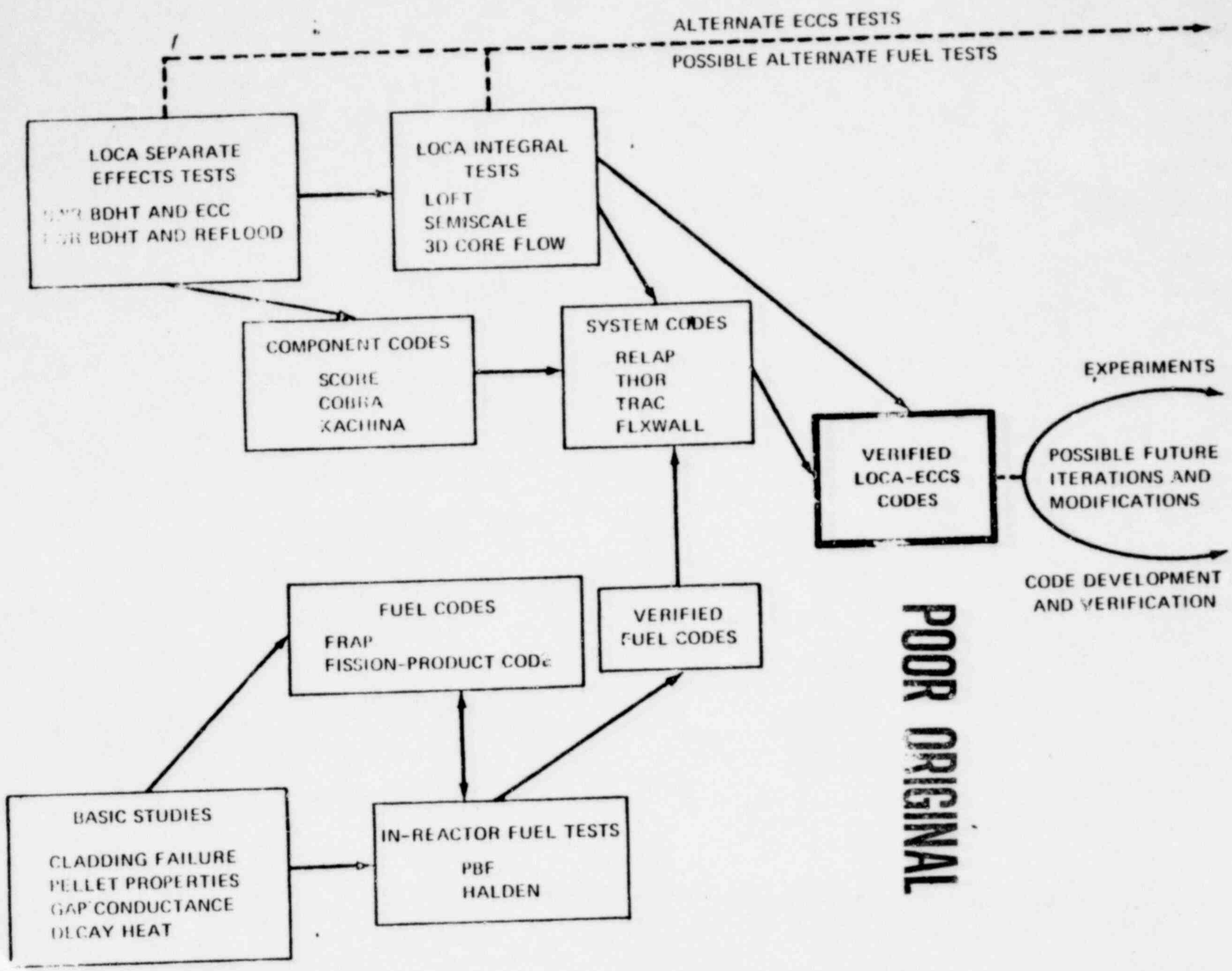
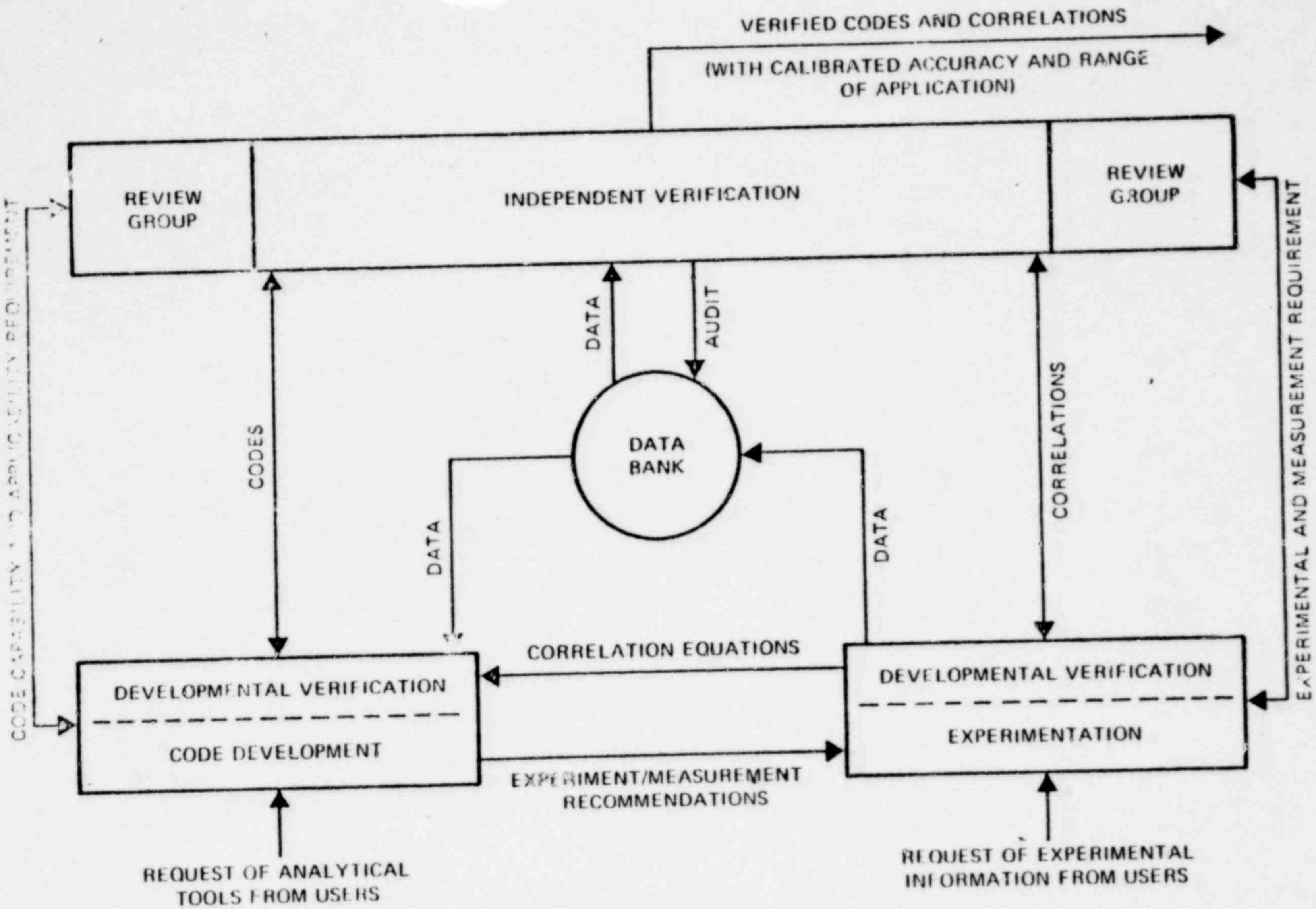

ALTERNATE ECCS TESTS
POSSIBLE ALTERNATE FUEL TESTS
LOCA SEPARATE EFFECTS TESTS
BDHT AND ECC
BDHT AND REFLOOD
LOCA INTEGRAL TESTS
LOFT
SEMISCALE
3D CORE FLOW
COMPONENT CODES
SCORE
COBRA
XACHINA
SYSTEM CODES
RELAP
THOR
TRAC
FLXWALL
VERIFIED LOCA-ECCS CODES
EXPERIMENTS
POSSIBLE FUTURE ITERATIONS AND MODIFICATIONS
CODE DEVELOPMENT AND VERIFICATION
FUEL CODES
FRAP
FISSION-PRODUCT CODE
VERIFIED FUEL CODES
BASIC STUDIES
CLADDING FAILURE
PELLET PROPERTIES
GAP CONDUCTANCE
DECAY HEAT
IN-REACTOR FUEL TESTS
PBF
HALDEN

FIGURE 1



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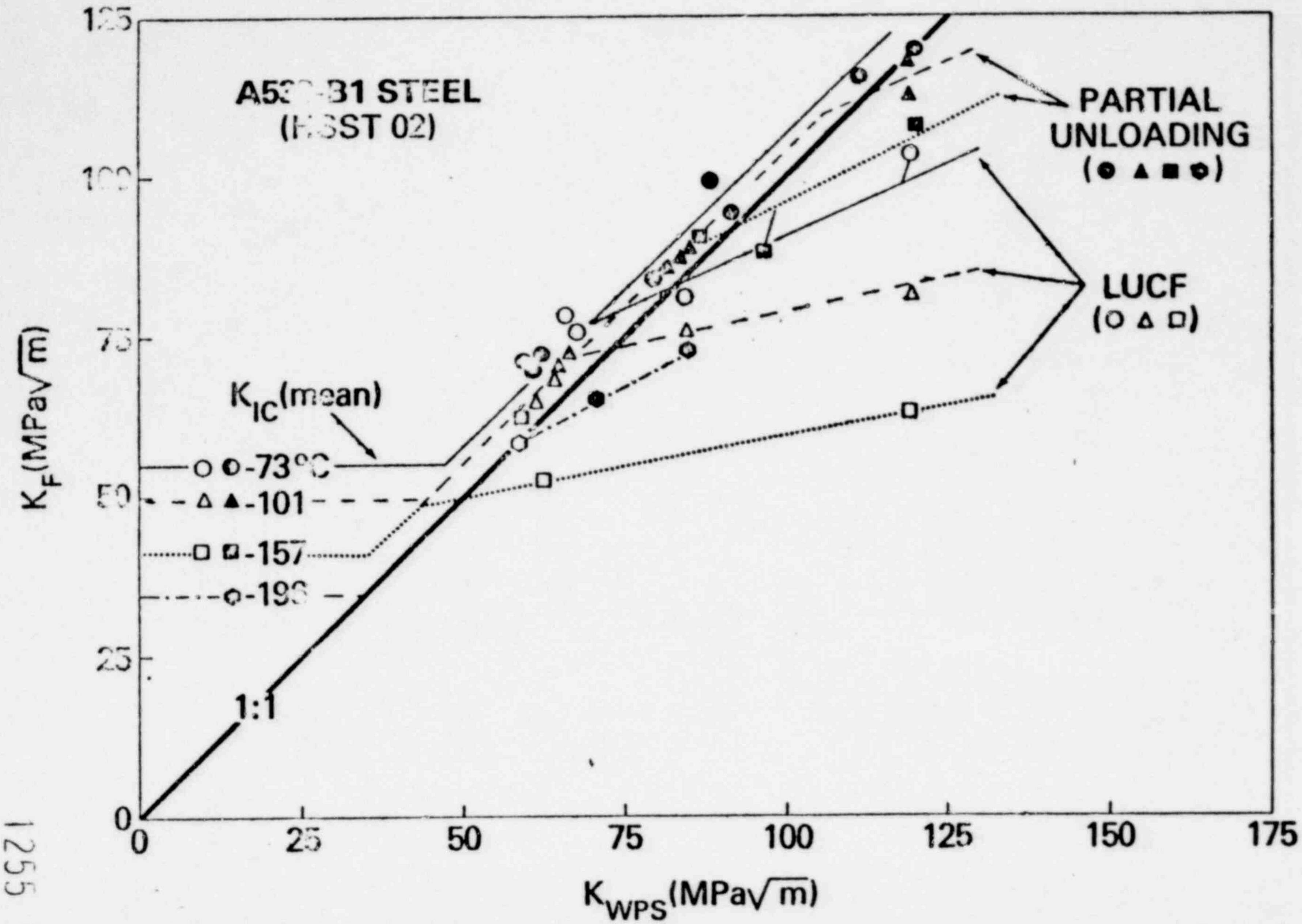
1255 106

FIGURE 2



Comparison of the K level at fracture (K_F) with the warm prestress level (K_{WPS}). The warm prestress was accomplished at room temperature. The specimens were unloaded in accordance with a prescribed path and then isothermally loaded to fracture at one of the indicated temperatures.

FIGURE 3

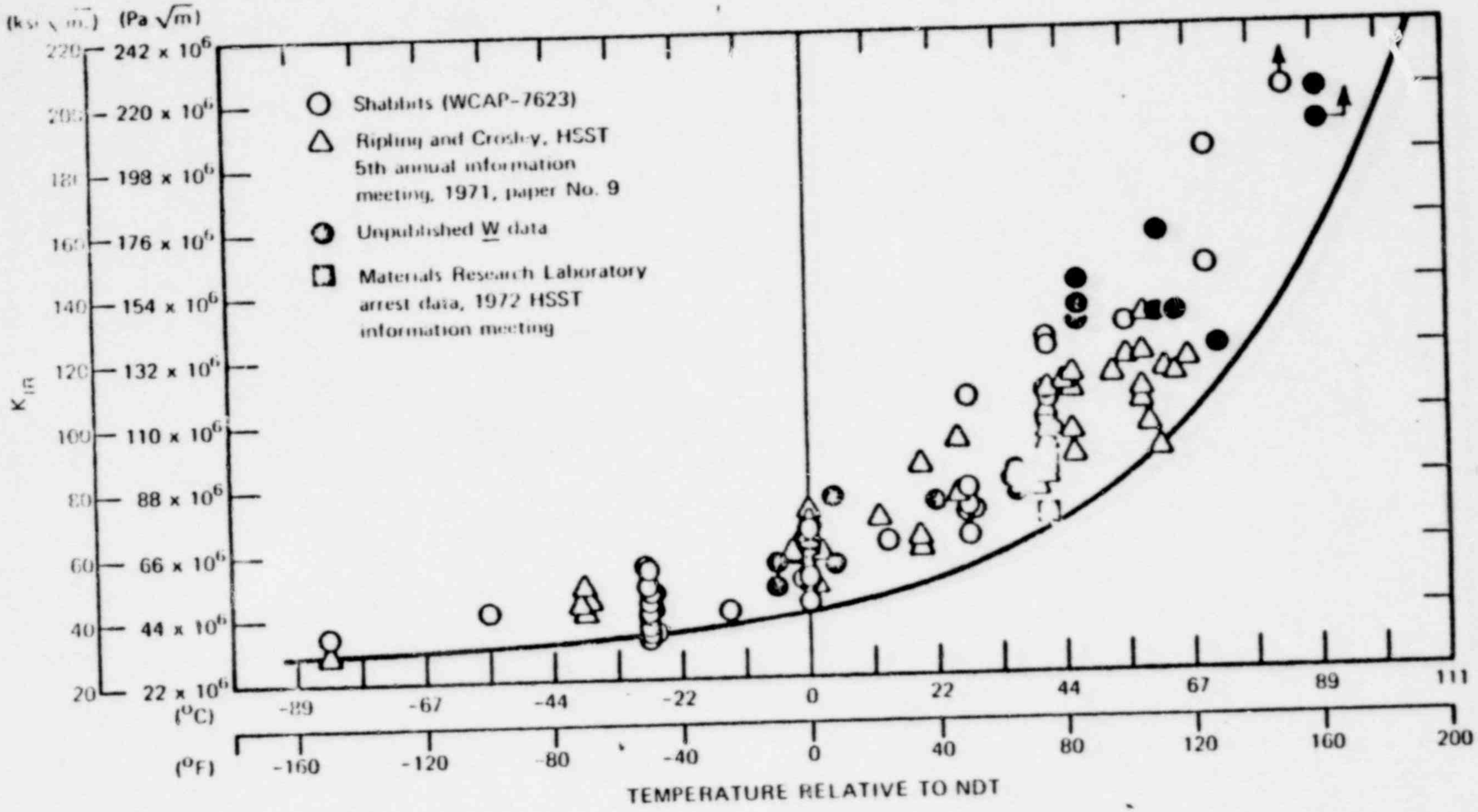


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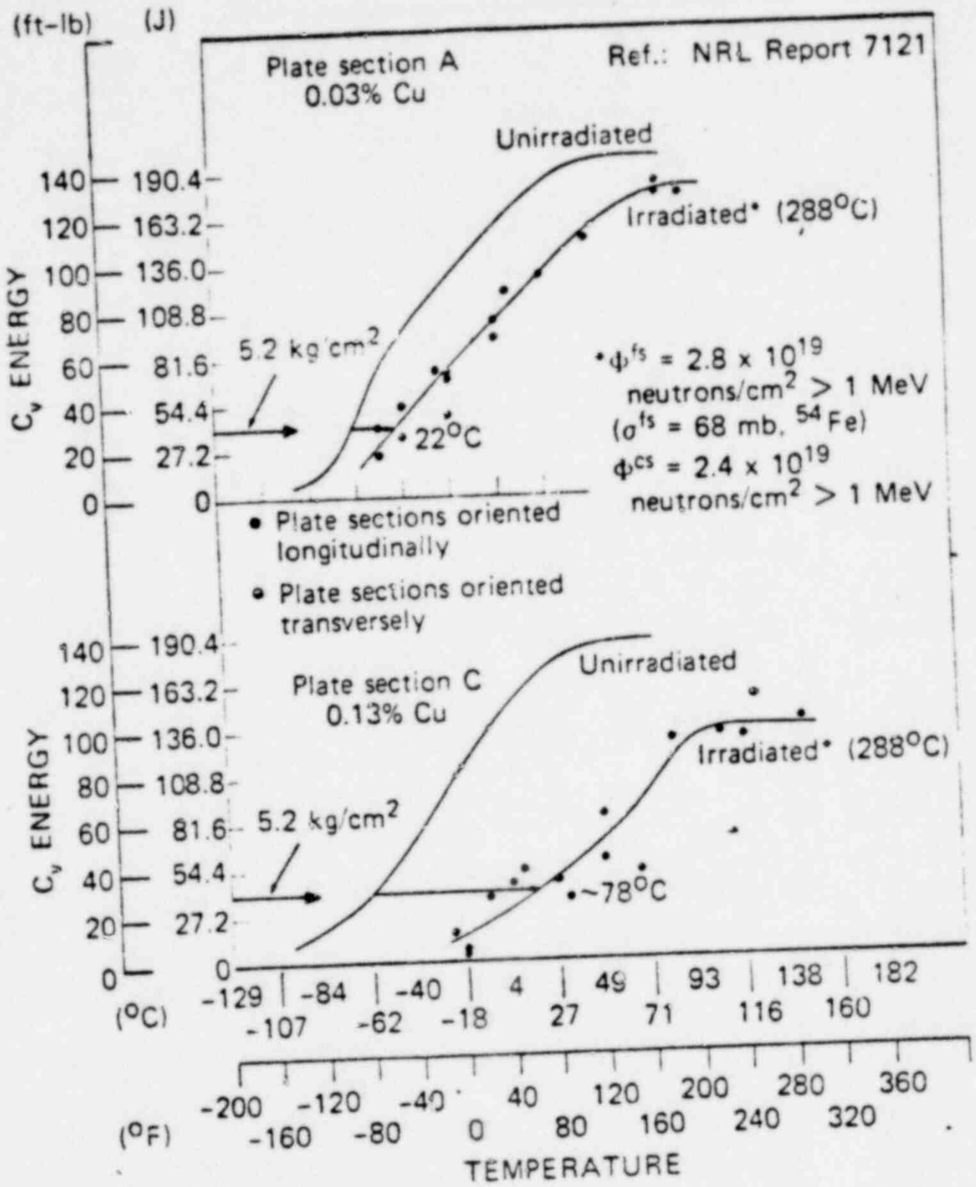
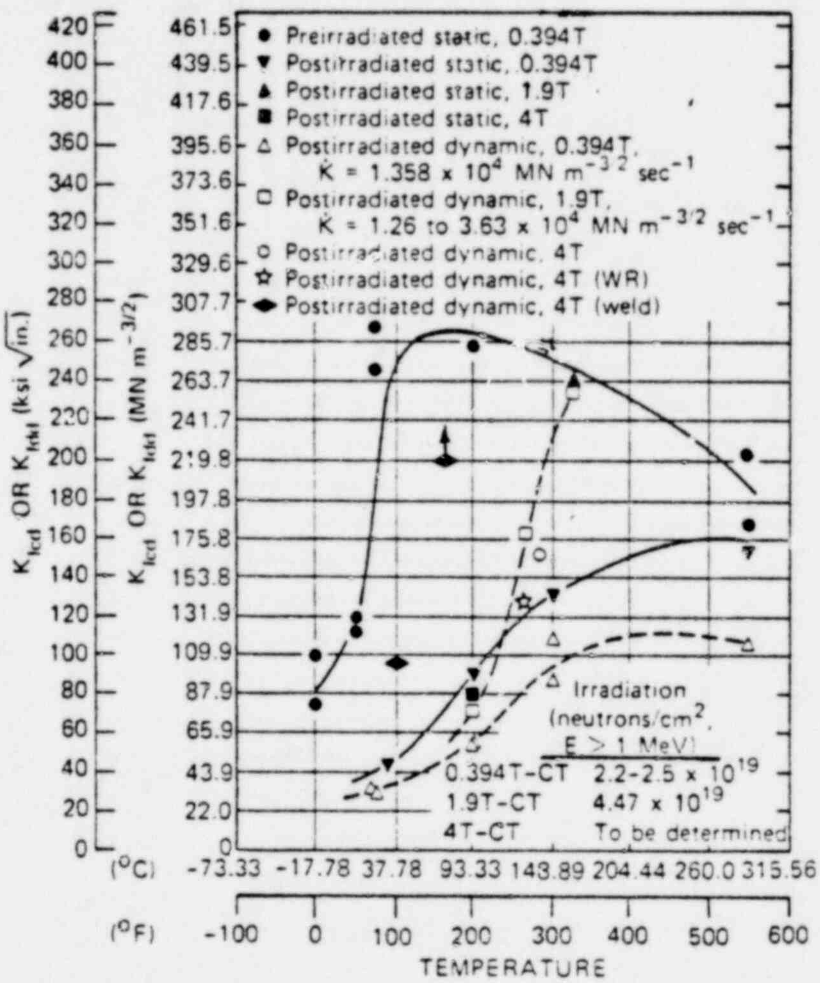
FIGURE 4



1255 109

FIGURE 5



1255 110

FIGURE 6

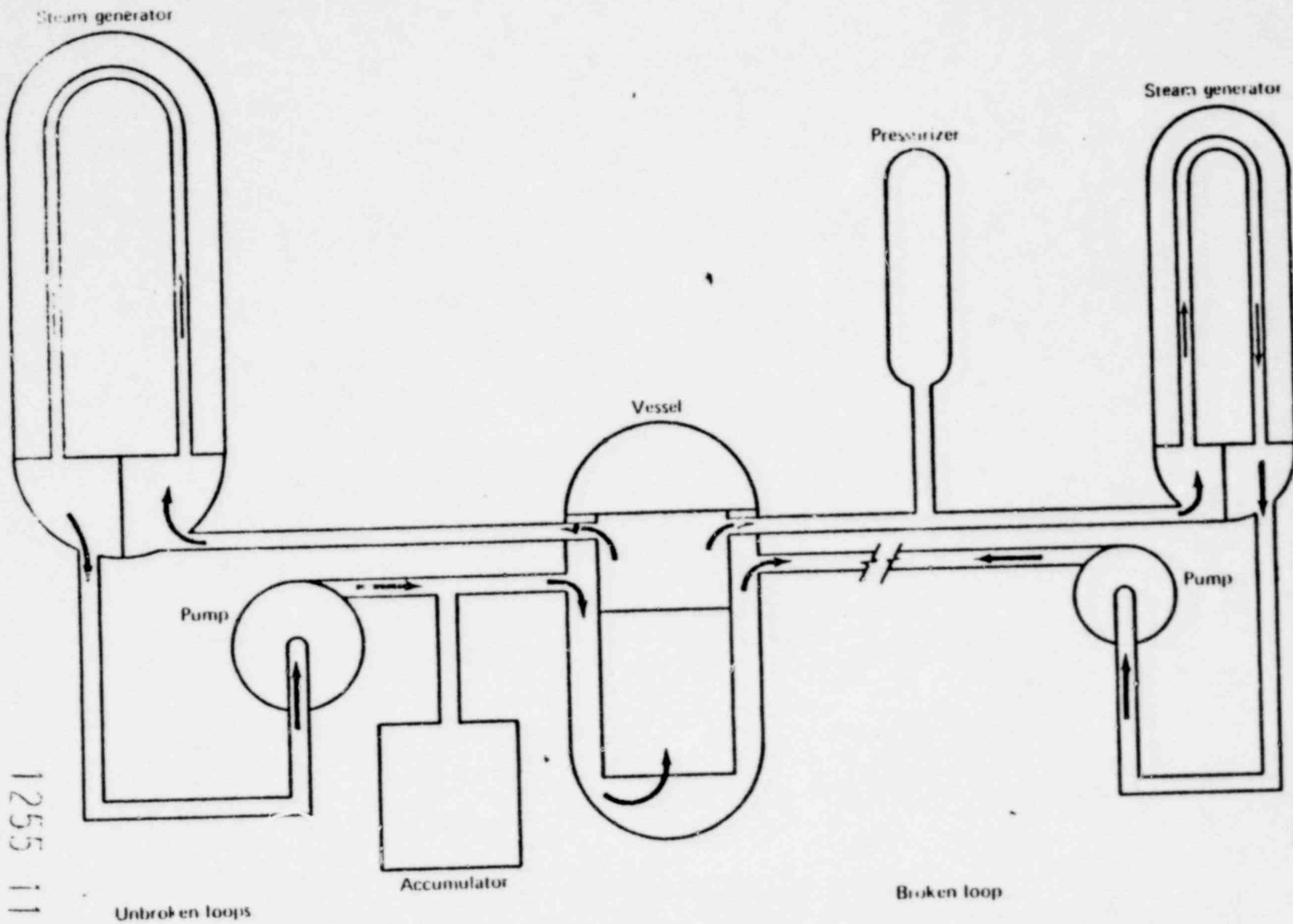
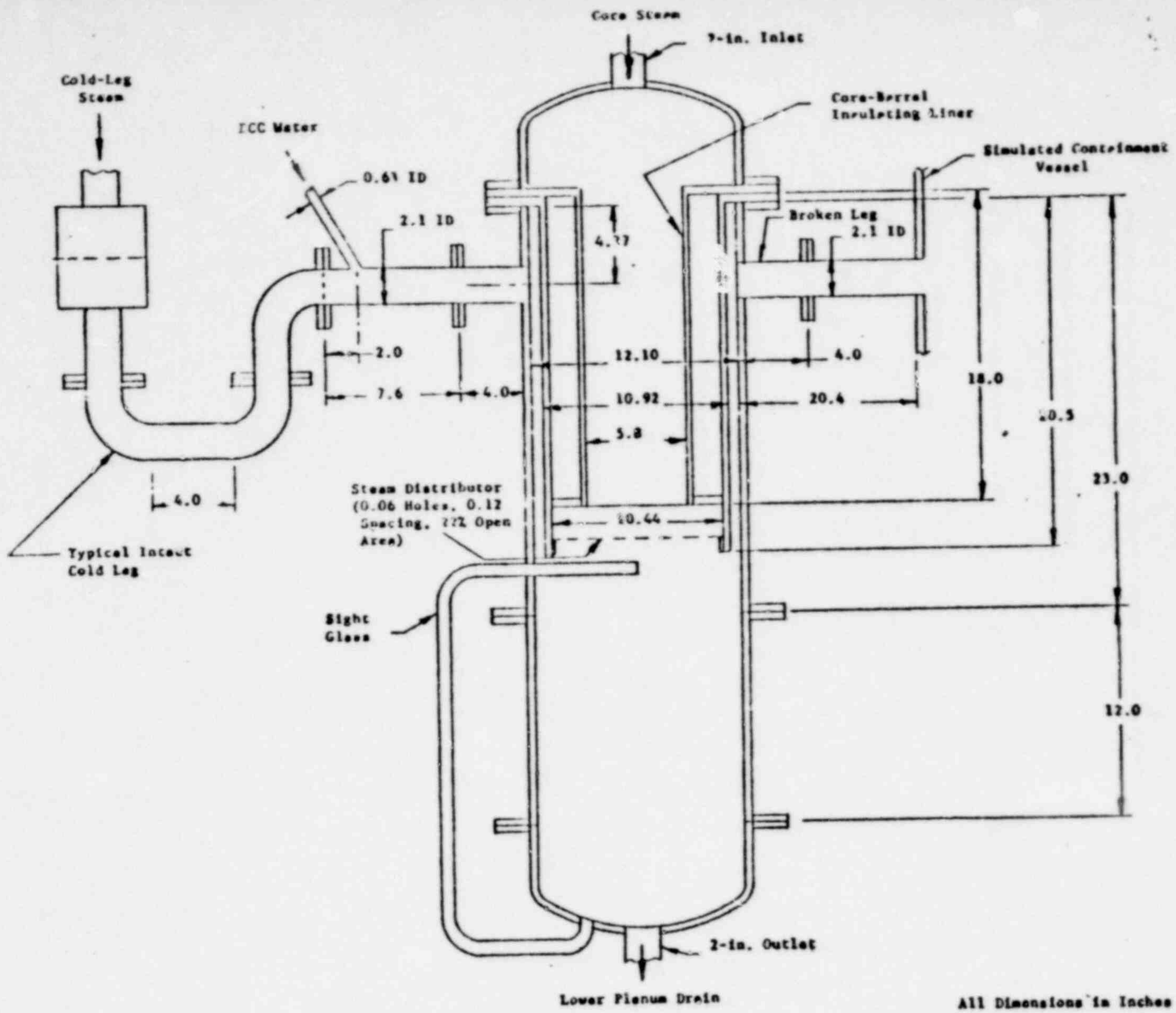

Steam generator
Steam generator
Pressurizer
Vessel
Pump
Pump
Accumulator
Unbroken loops
Broken loop

FIGURE 7

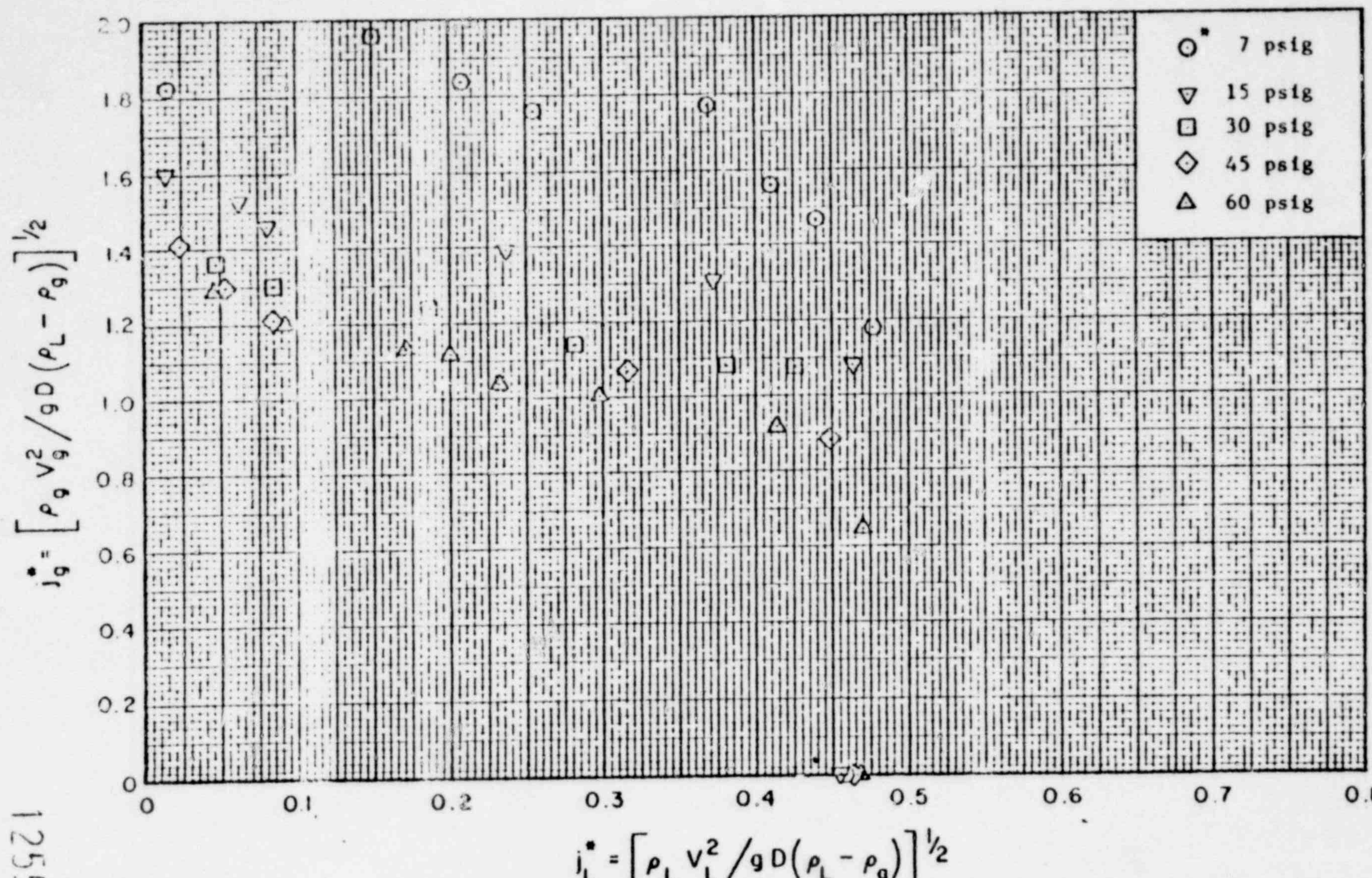


SCHEMATIC OF BCL 1/15-SCALE 60-PSI VESSEL MODEL

POOR ORIGINAL

1255 112

FIGURE 8



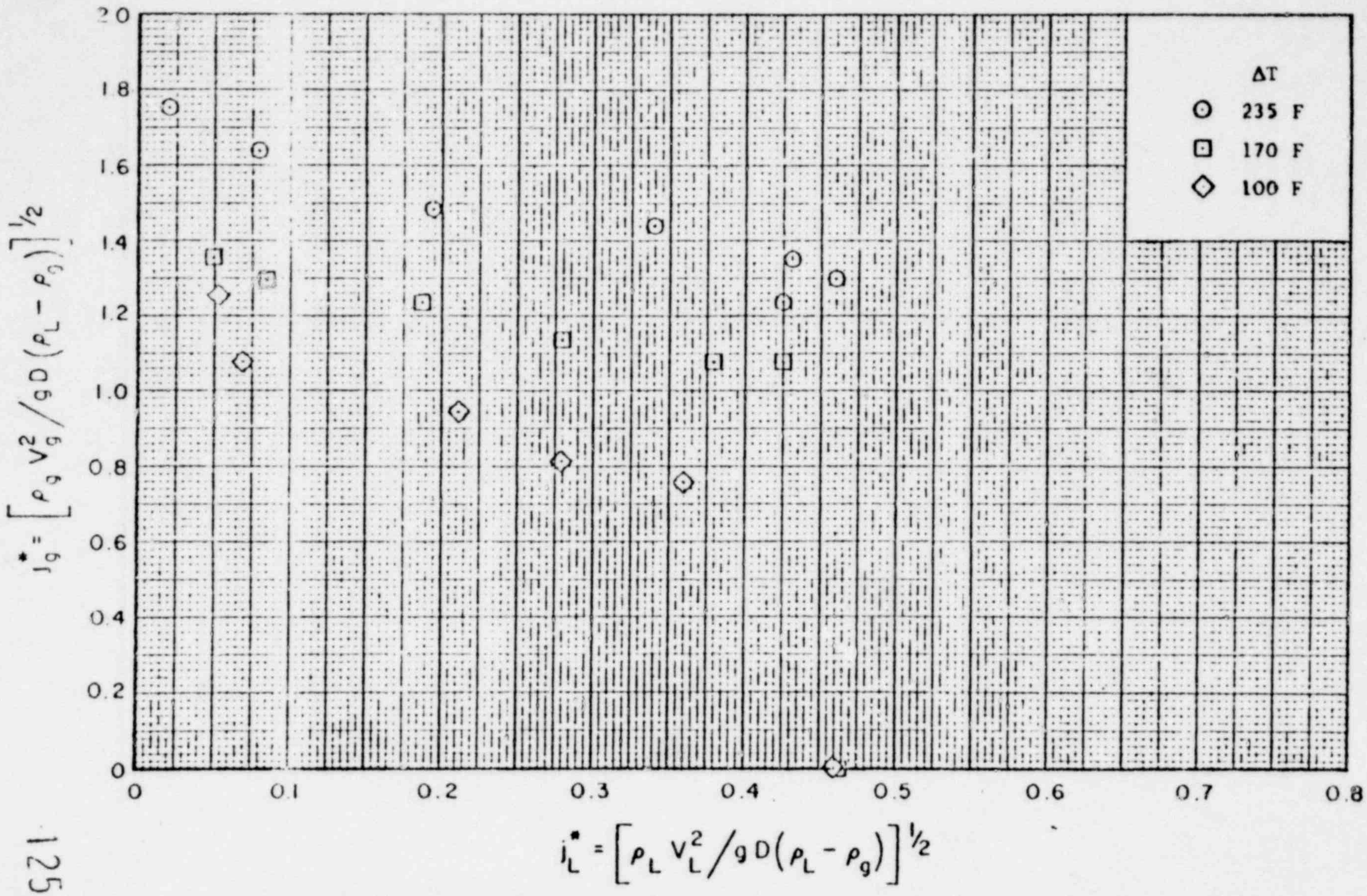
WATER PENETRATION WITH NO COLD LEG STEAM AT A $(j_L^*)_{in}$ OF 0.49 WITH ECC WATER SUBCOOLING OF 170 F

POOR ORIGINAL

1255 113

* For ○ Subcooling 190 F

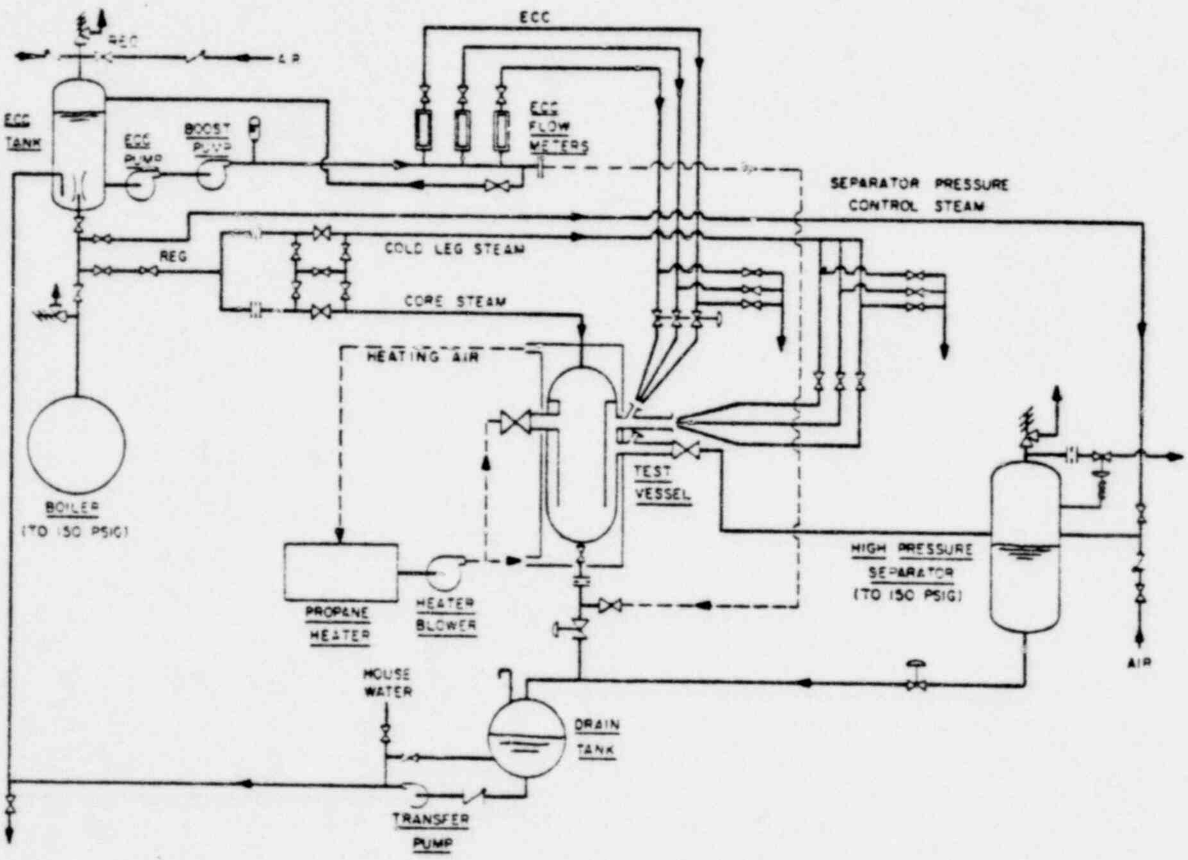
FIGURE 9



WATER PENETRATION WITH NO COLD LEG STEAM AT A LOWER PLENUM PRESSURE OF 30 PSIG AND $(j_L^*)_{IN}$ OF 0.49

FIGURE 10

POOR ORIGINAL



FLOW SCHEMATIC FOR CREARE 1/15-SCALE HIGH PRESSURE CYLINDRICAL VESSEL AND FACILITY

1255 115

FIGURE 11

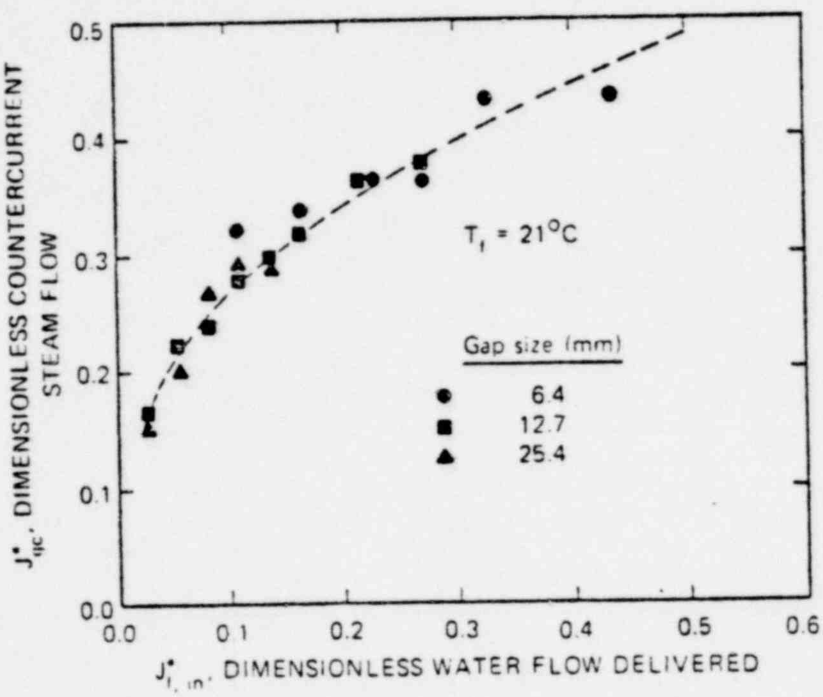


FIGURE 12

1255 116

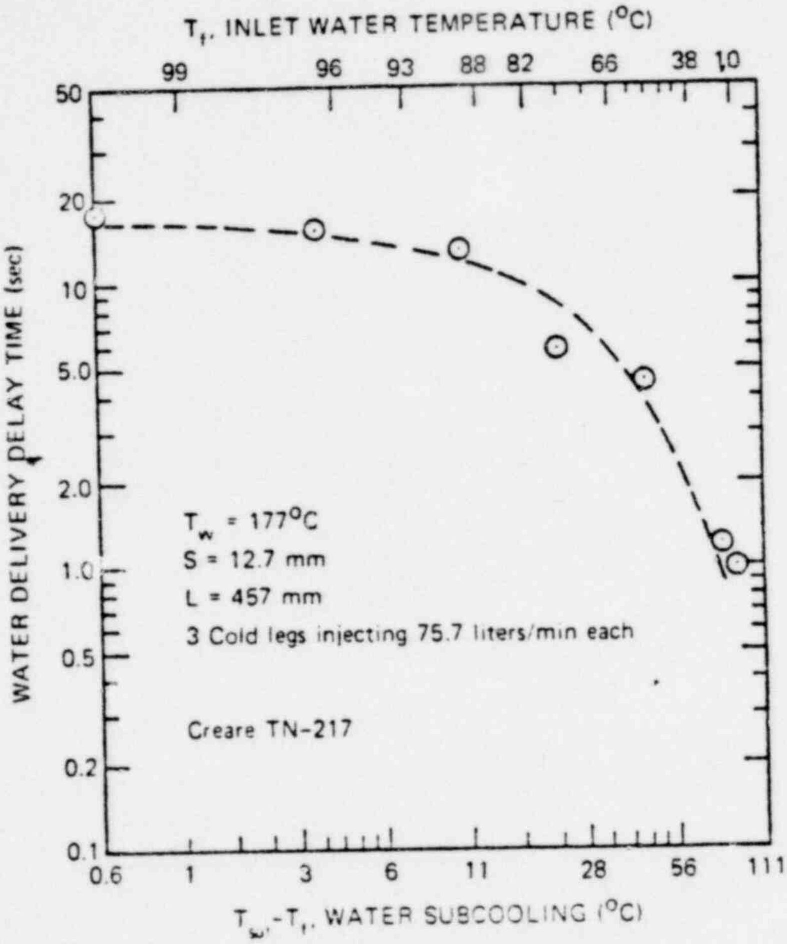
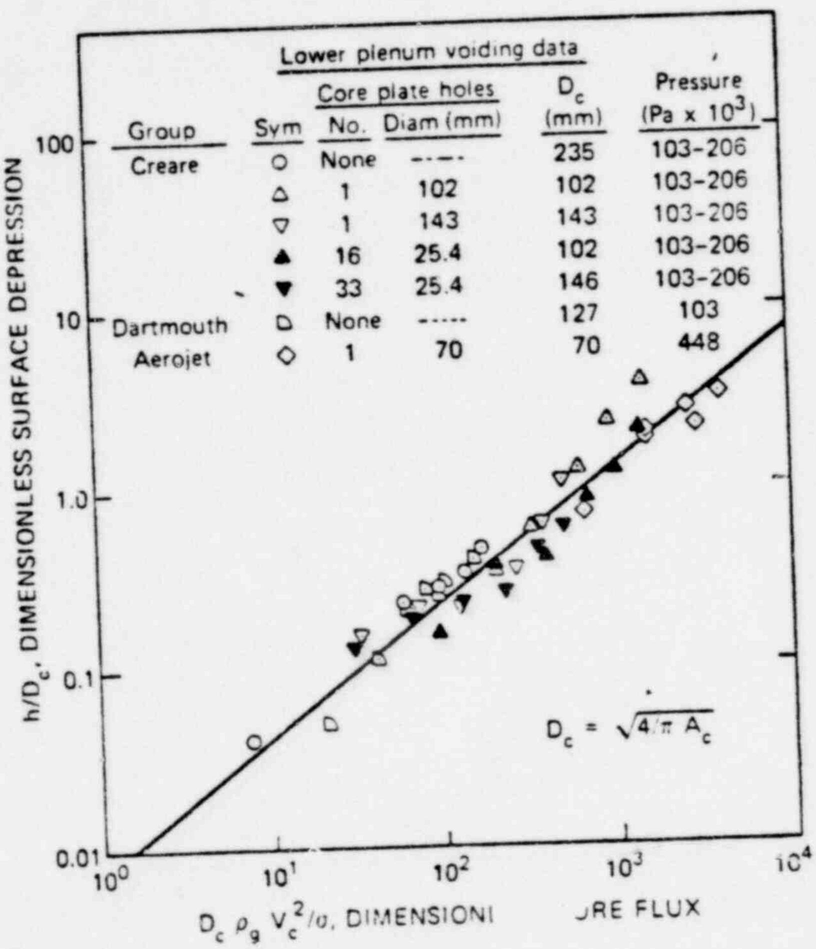
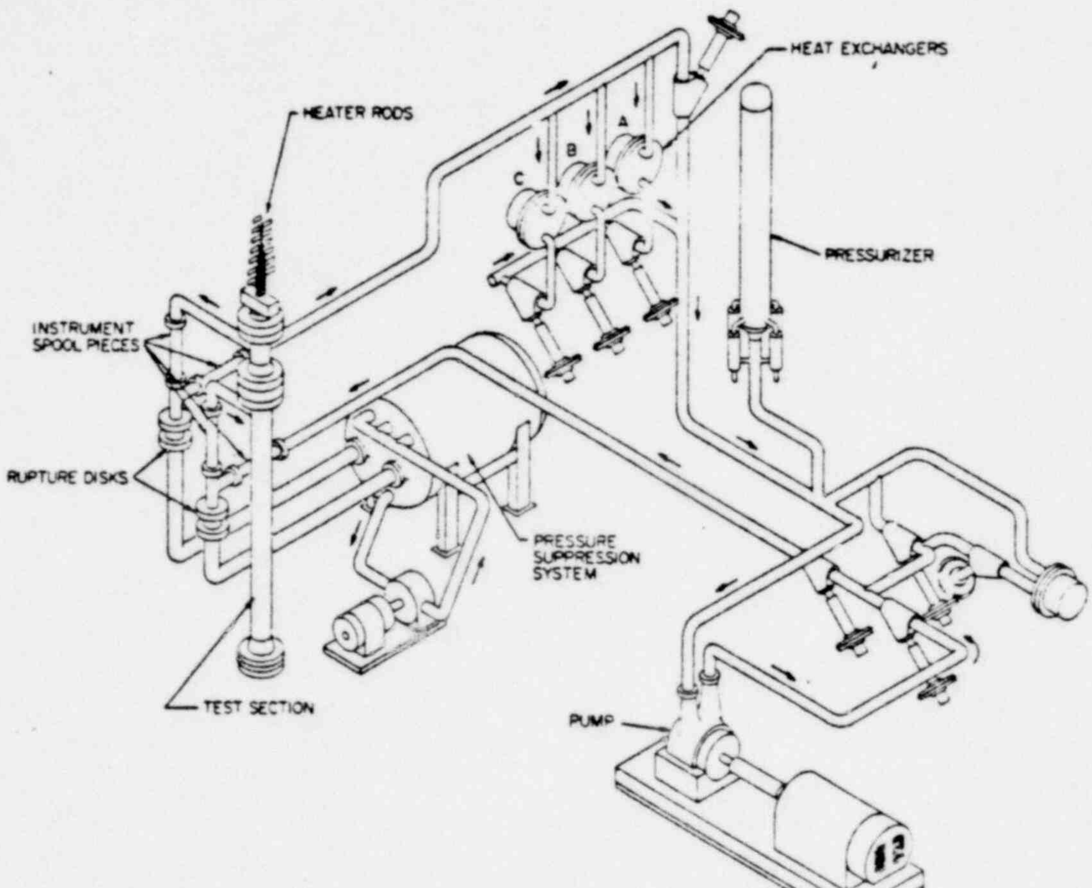

T_f, INLET WATER TEMPERATURE (^oC)
99
96
93
88
82
66
38
10
50
20
10
5.0
2.0
1.0
0.5
0.2
0.1
WATER DELIVERY DELAY TIME (sec)
T_w = 177^oC
S = 12.7 mm
L = 457 mm
3 Cold legs injecting 75.7 liters/min each
Creare TN-217
0.6
1
3
6
11
28
56
111
T_{sw}-T_f, WATER SUBCOOLING (^oC)

FIGURE 13



1255 118

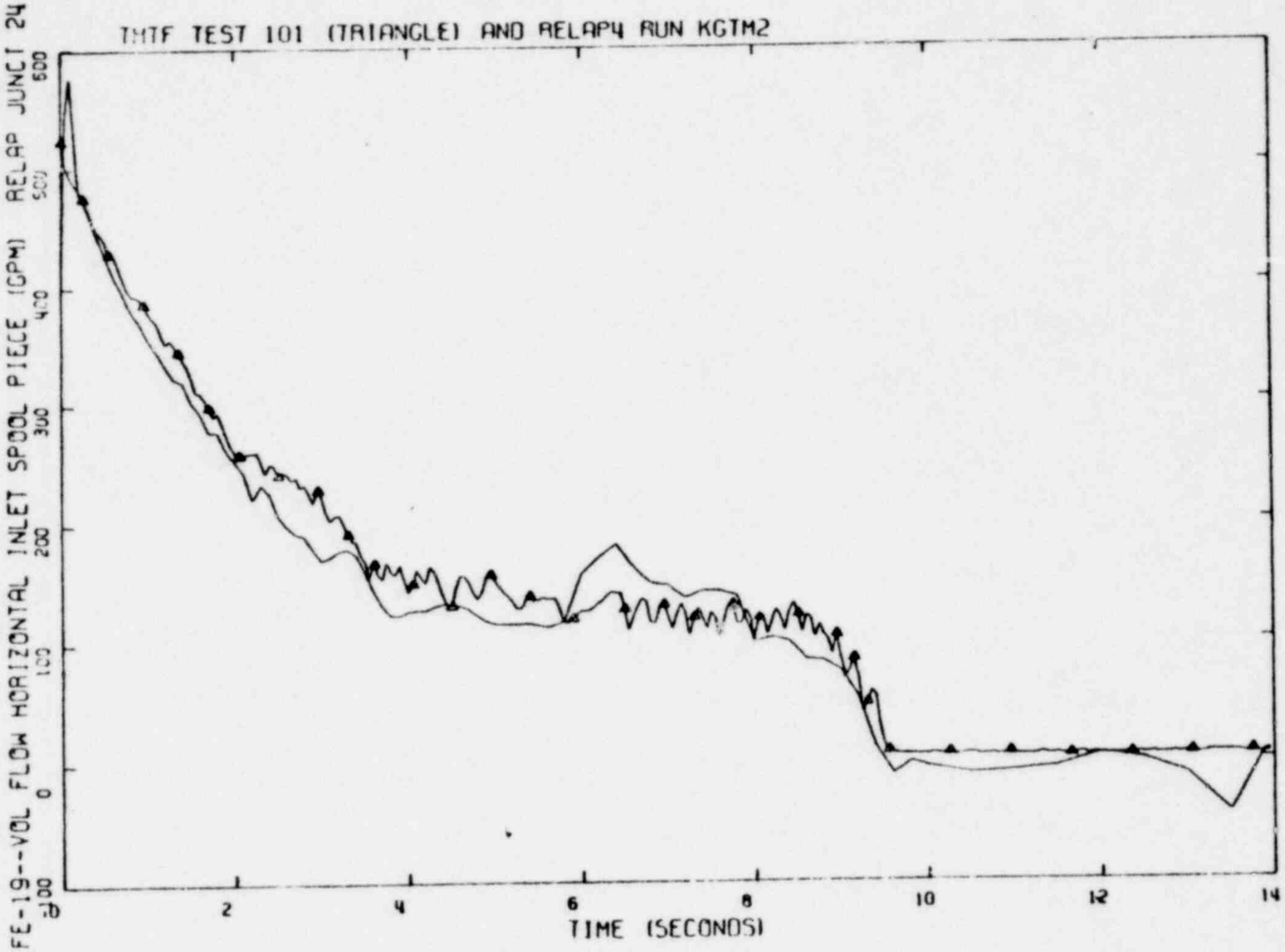
FIGURE 14



THERMAL-HYDRAULIC TEST FACILITY-ORNL-PWR-BDHT SEPARATE EFFECTS PROGRAM

1255 119

FIGURE 14A

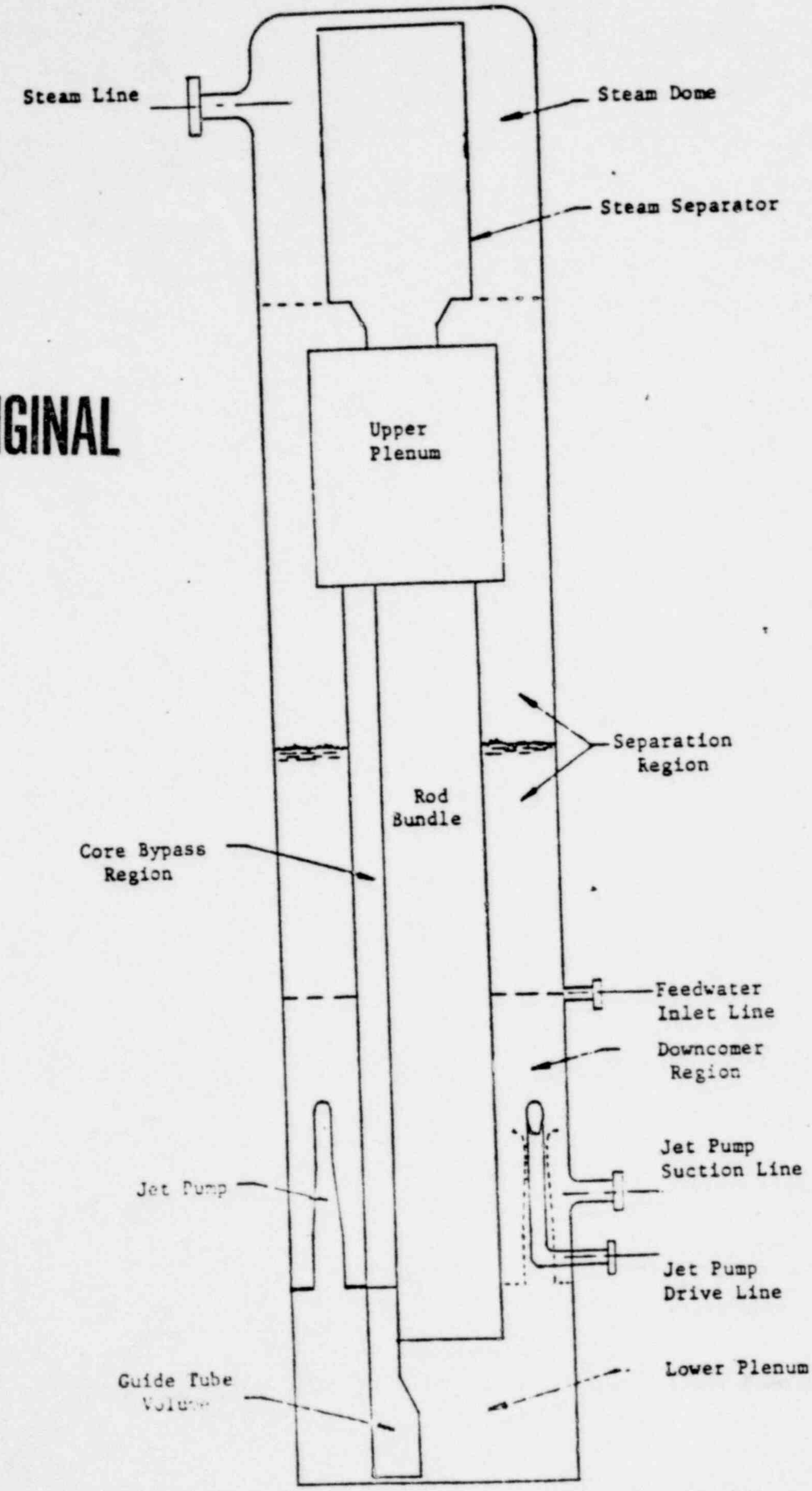


FE-19 volumetric flow horizontal spool piece — with slip.

1255 120

FIGURE 15

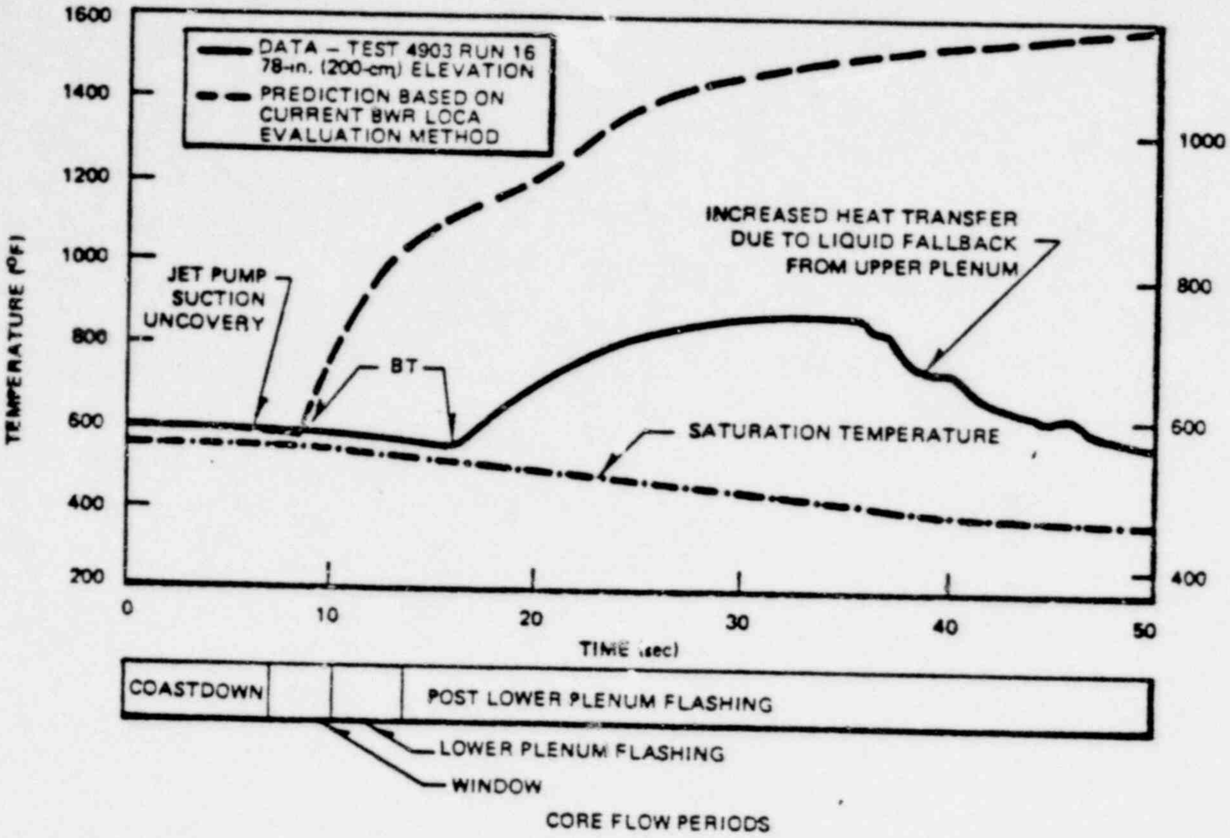
POOR ORIGINAL



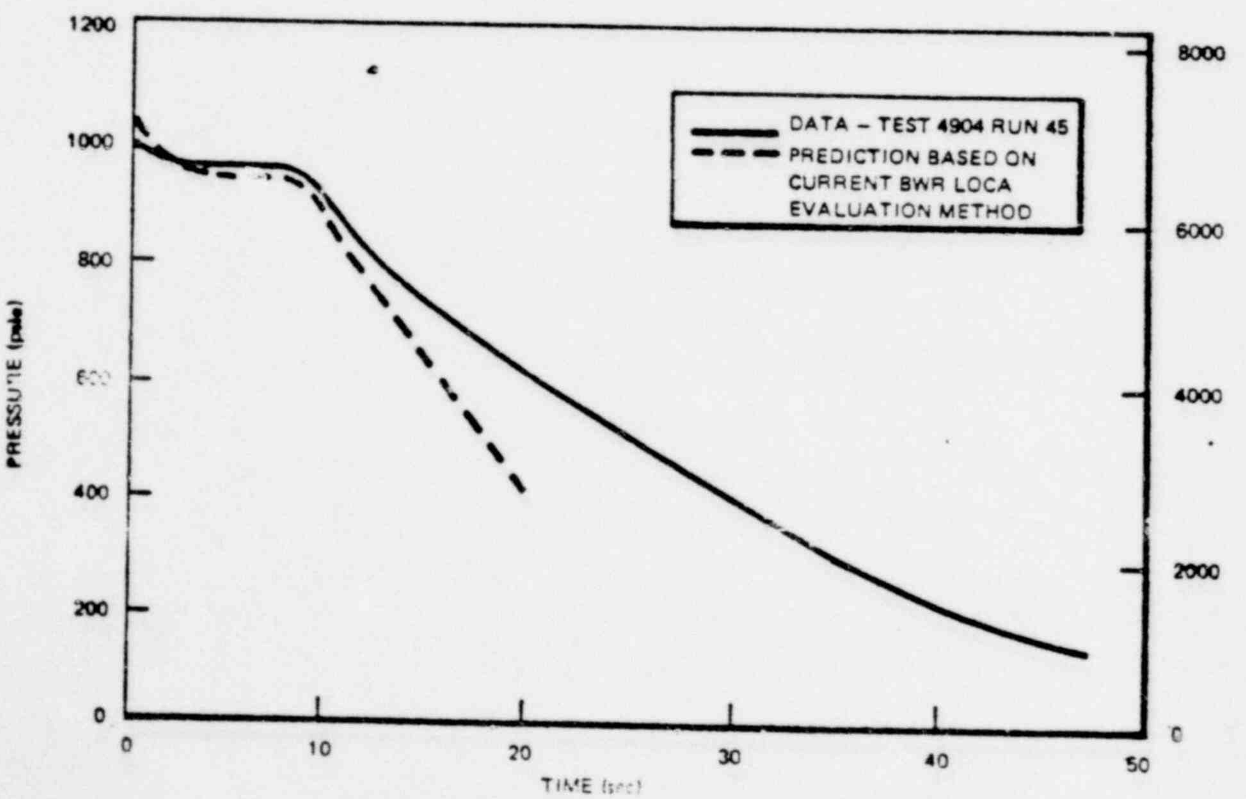
SCHEMATIC LAYOUT OF TEST APPARATUS GEOMETRY.

1255 121

FIGURE 1



High Power Rod PCT for Nominal Power Test



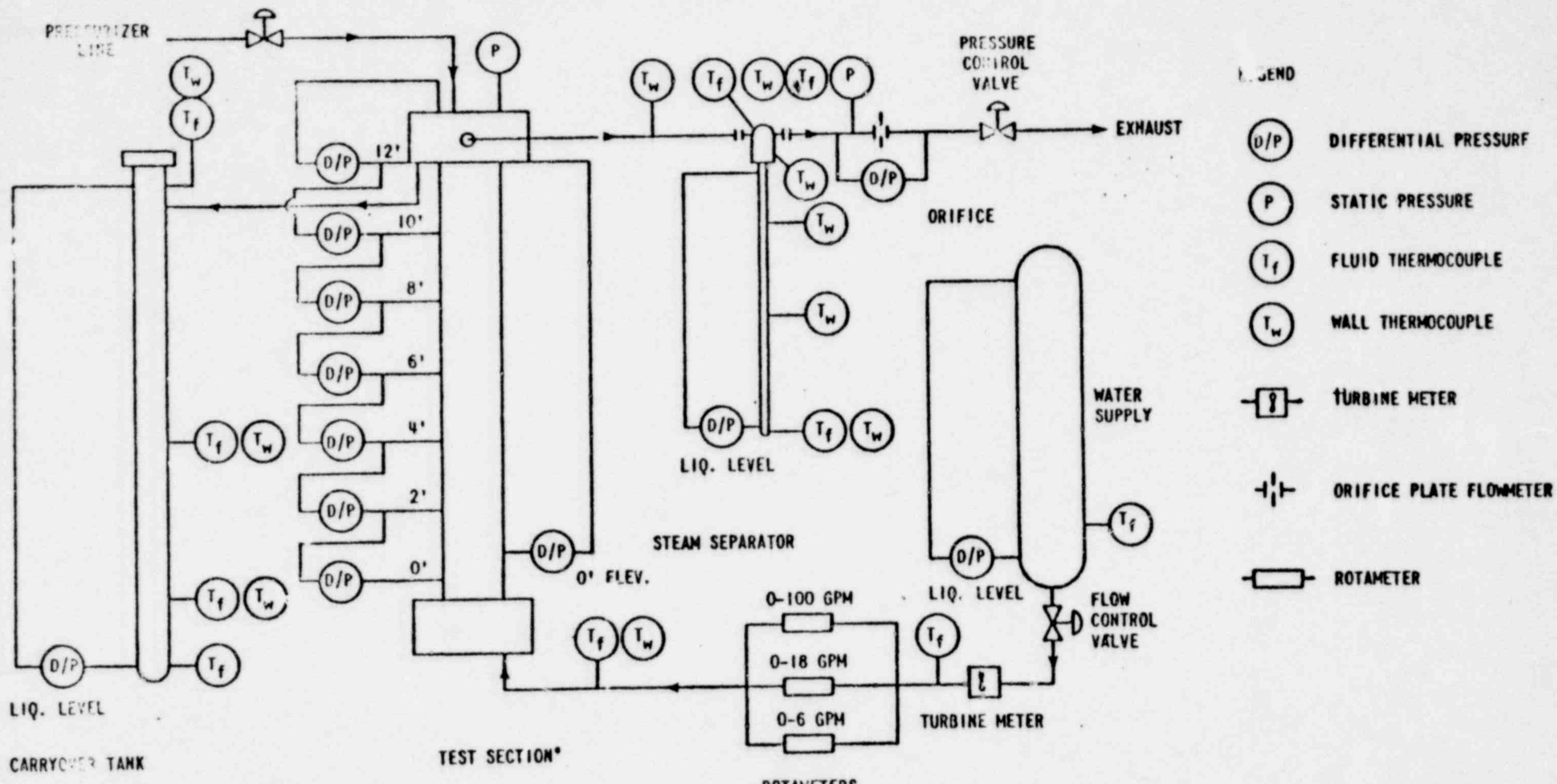
Pressure for Peak Power Test

1255 122

FIGURE 11

POOR ORIGINAL

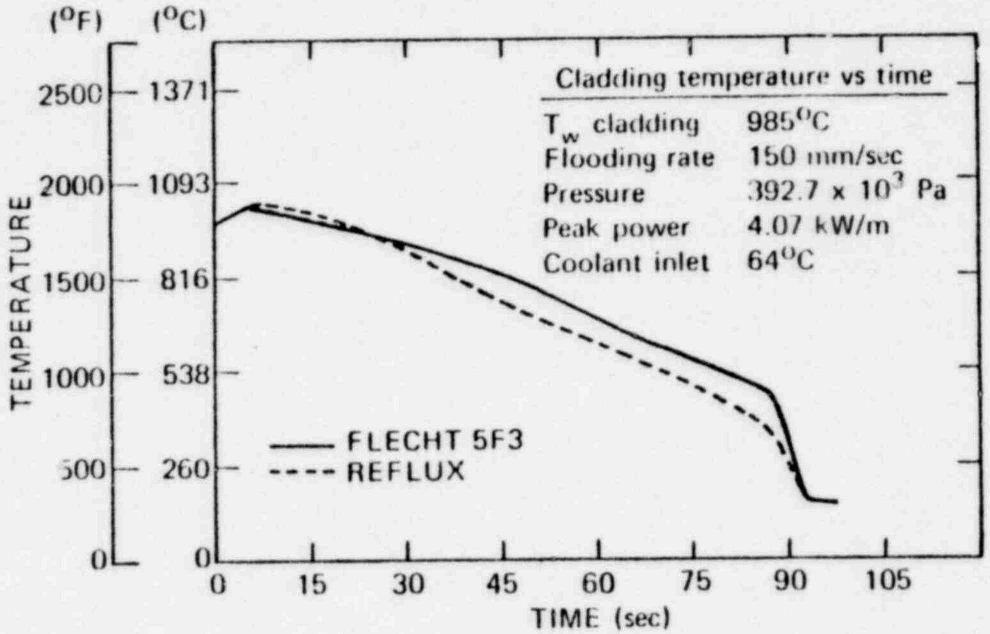
FLECHT LOW FLOODING RATE TEST CONFIGURATION



* NOT ALL INSTRUMENTATION IS SHOWN

FIGURE 18

1255 123



1255 124

FIGURE 19

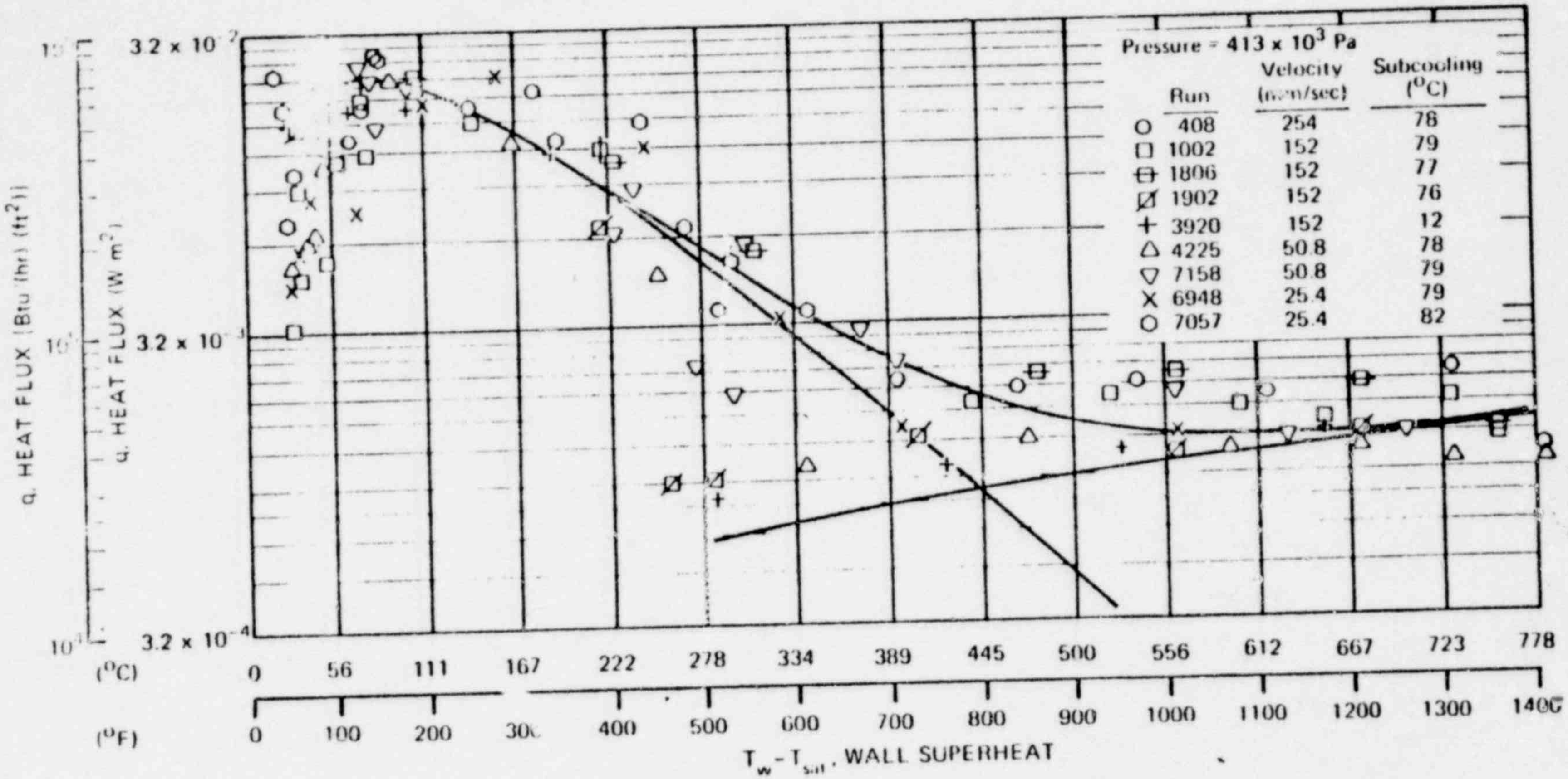
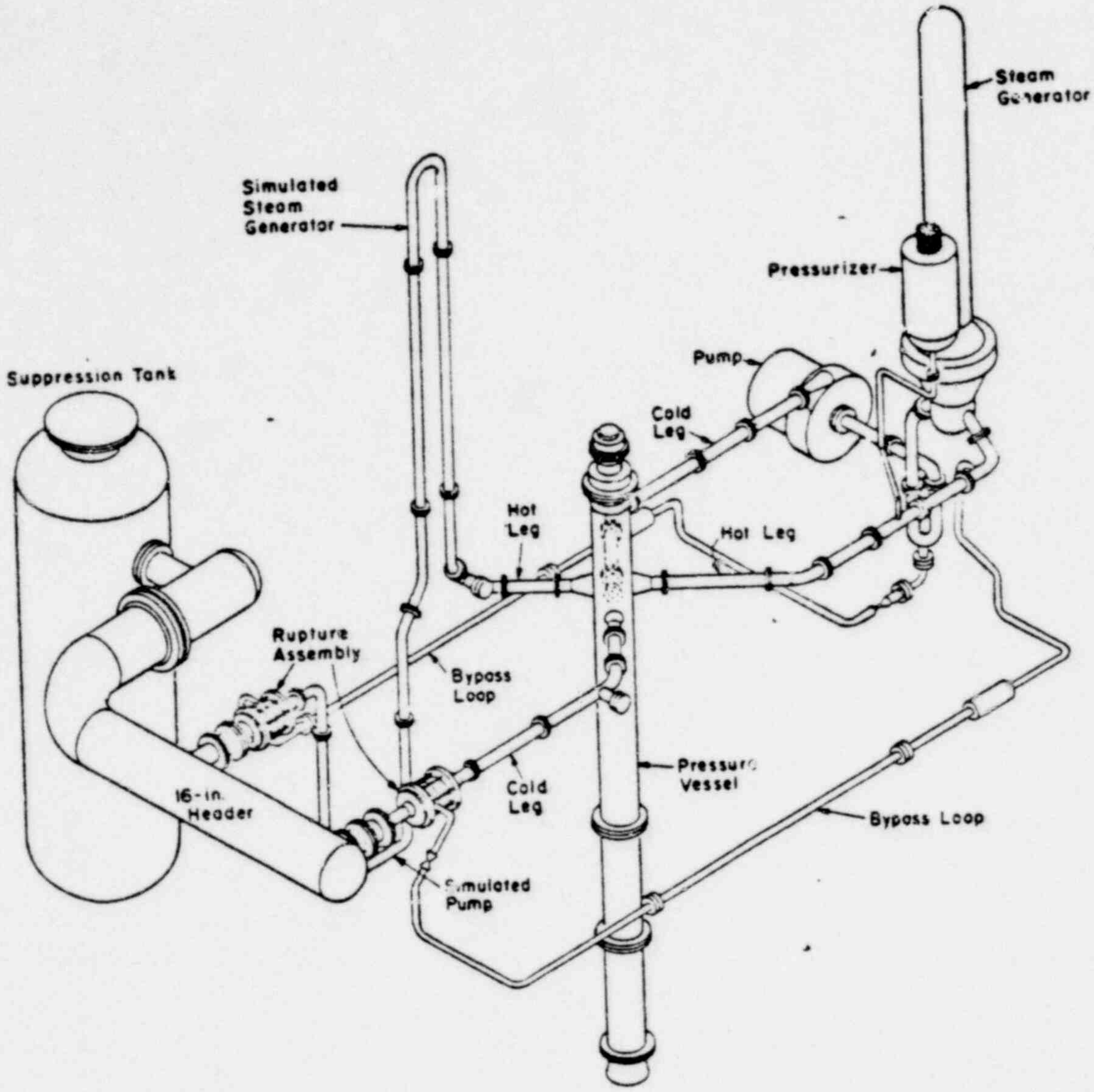

Pressure = 413 x 10^3 Pa
Run
Velocity (mm/sec)
Subcooling (°C)
408 254 78
1002 152 79
1806 152 77
1902 152 76
3920 152 12
4225 50.8 78
7158 50.8 79
6948 25.4 79
7057 25.4 82
q, HEAT FLUX (Btu/hr) (ft²)
q, HEAT FLUX (W m²)
3.2 x 10^-2
3.2 x 10^-3
3.2 x 10^-4
(°C)
0 56 111 167 222 278 334 389 445 500 556 612 667 723 778
(°F)
0 100 200 300 400 500 600 700 800 900 1000 1100 1200 1300 1400
T_w - T_sat, WALL SUPERHEAT

FIGURE 20

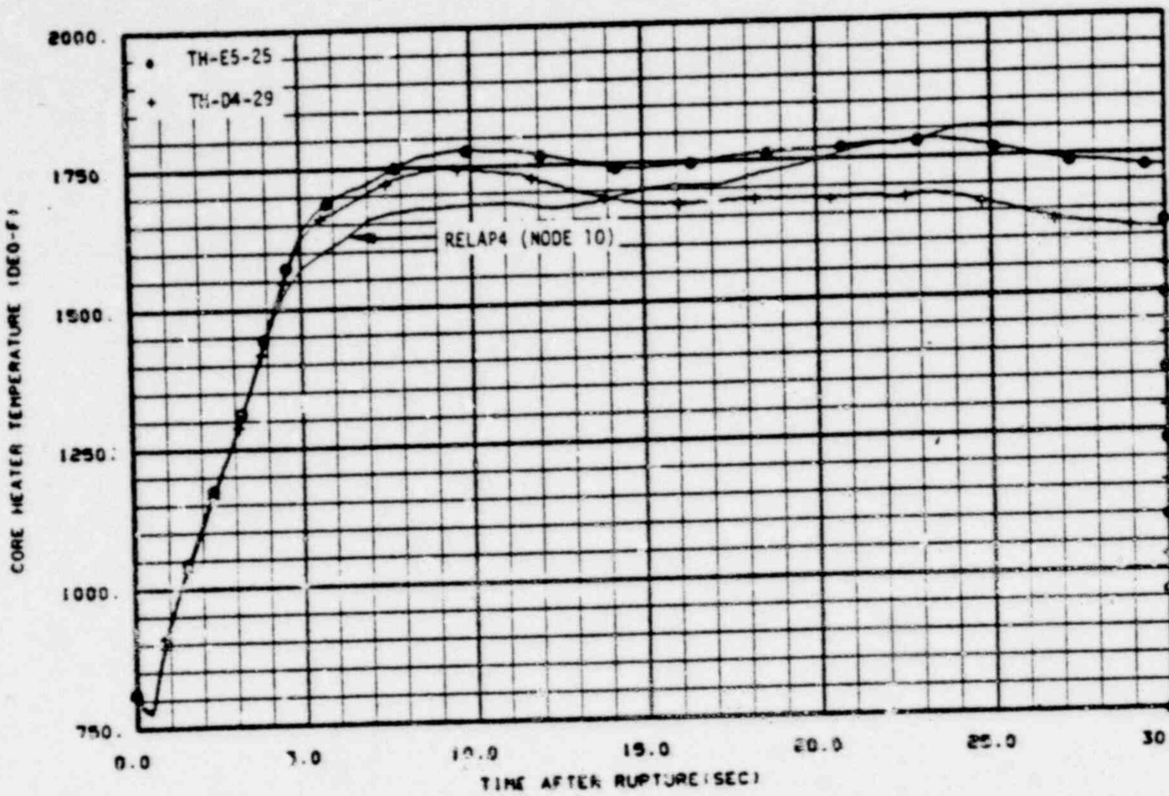


SEMISCALE MOD-1 SYSTEM COLD LEG BREAK CONFIGURATION - ISOMETRIC

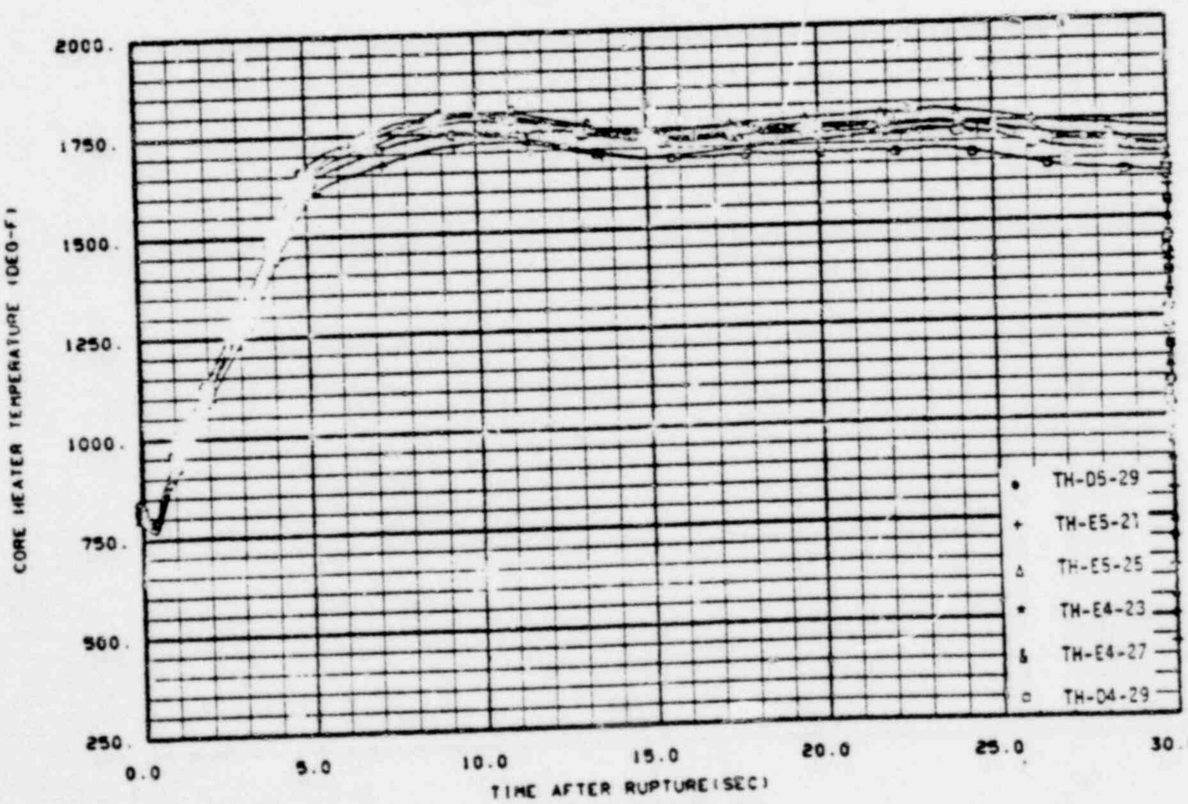
1255 126

FIGURE 21

POOR ORIGINAL



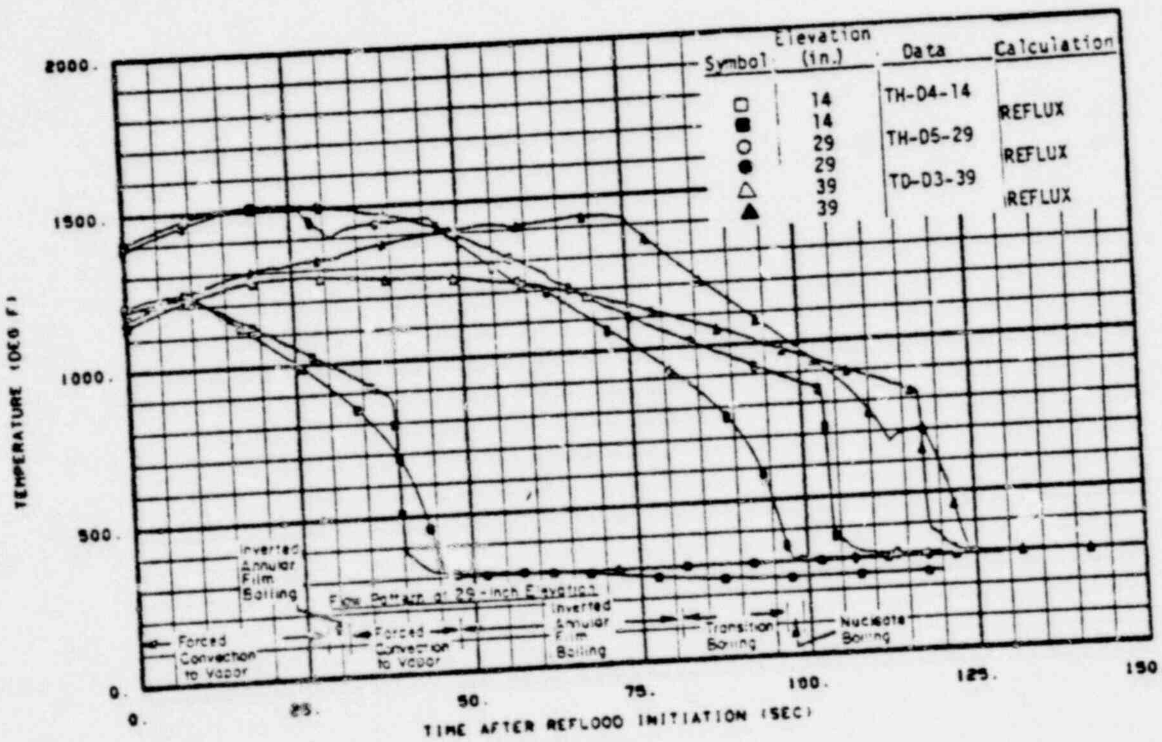
HIGH POWER ROD HOT SPOT CLAD TEMPERATURE, TEST S-02-5



HIGH POWER ROD HOT SPOT CLAD TEMPERATURE, TEST S-02-5

1255 127 FIGURE 22

POOR ORIGINAL

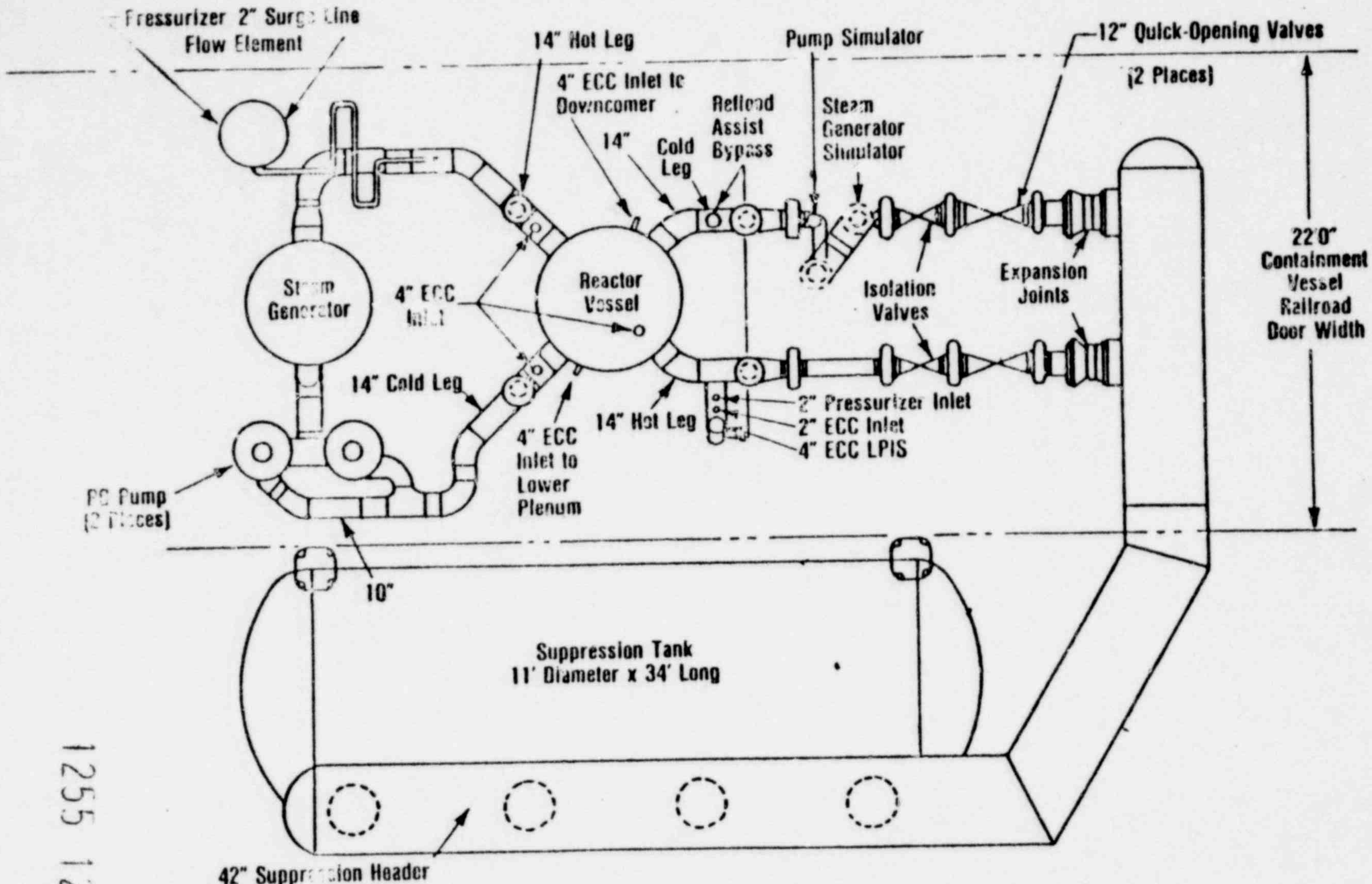


COMPARISON OF THE ROD CLADDING TEMPERATURE RESPONSE - REFLUX CALCULATIONS AND DATA FROM 14-, 29-, AND 39-INCH ELEVATIONS (TEST S-03-2)

1255 128

FIGURE 23

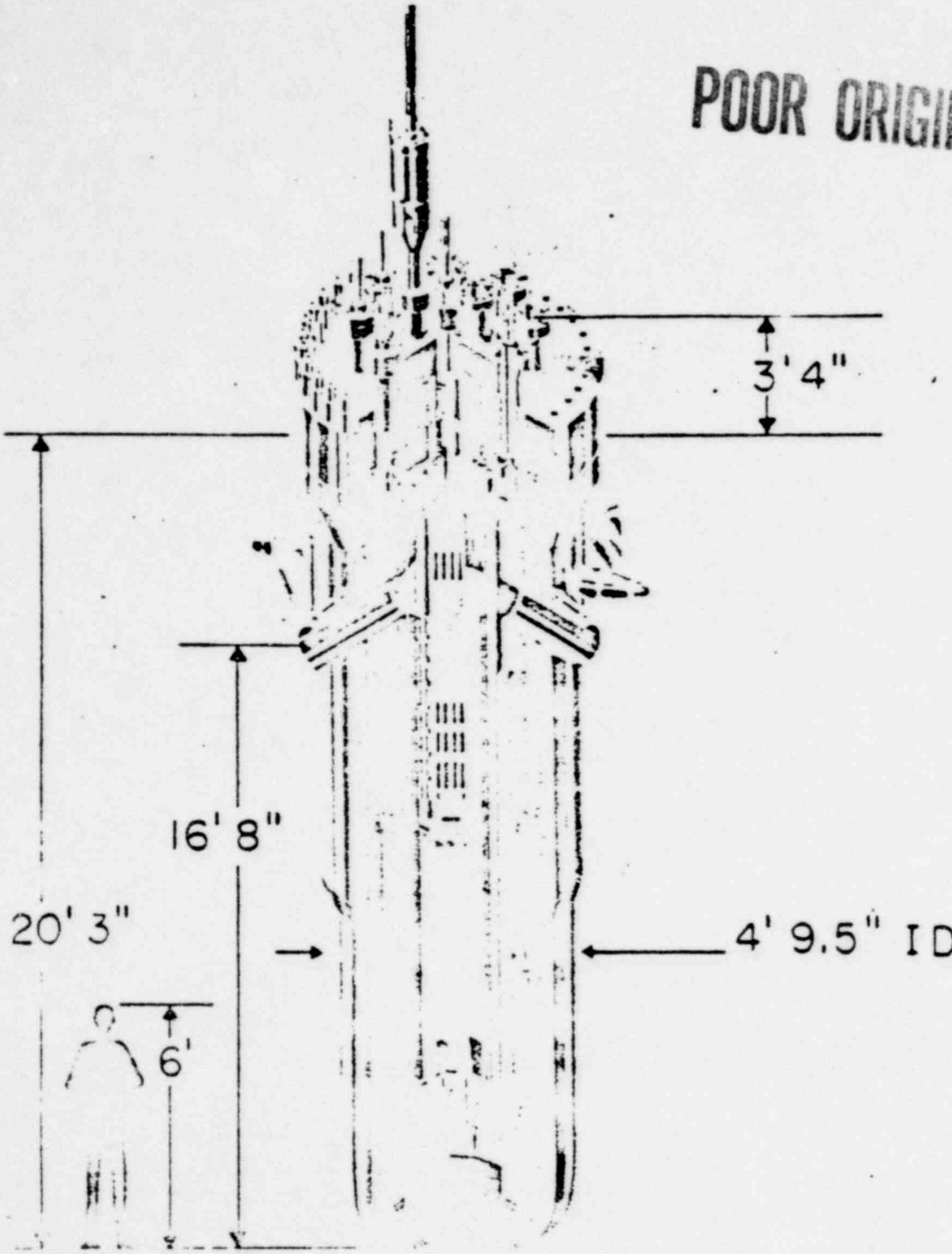
PLAN VIEW TEST ASSEMBLY
(Hot Leg Break Configuration)



1255 129

FIGURE 24

POOR ORIGINAL

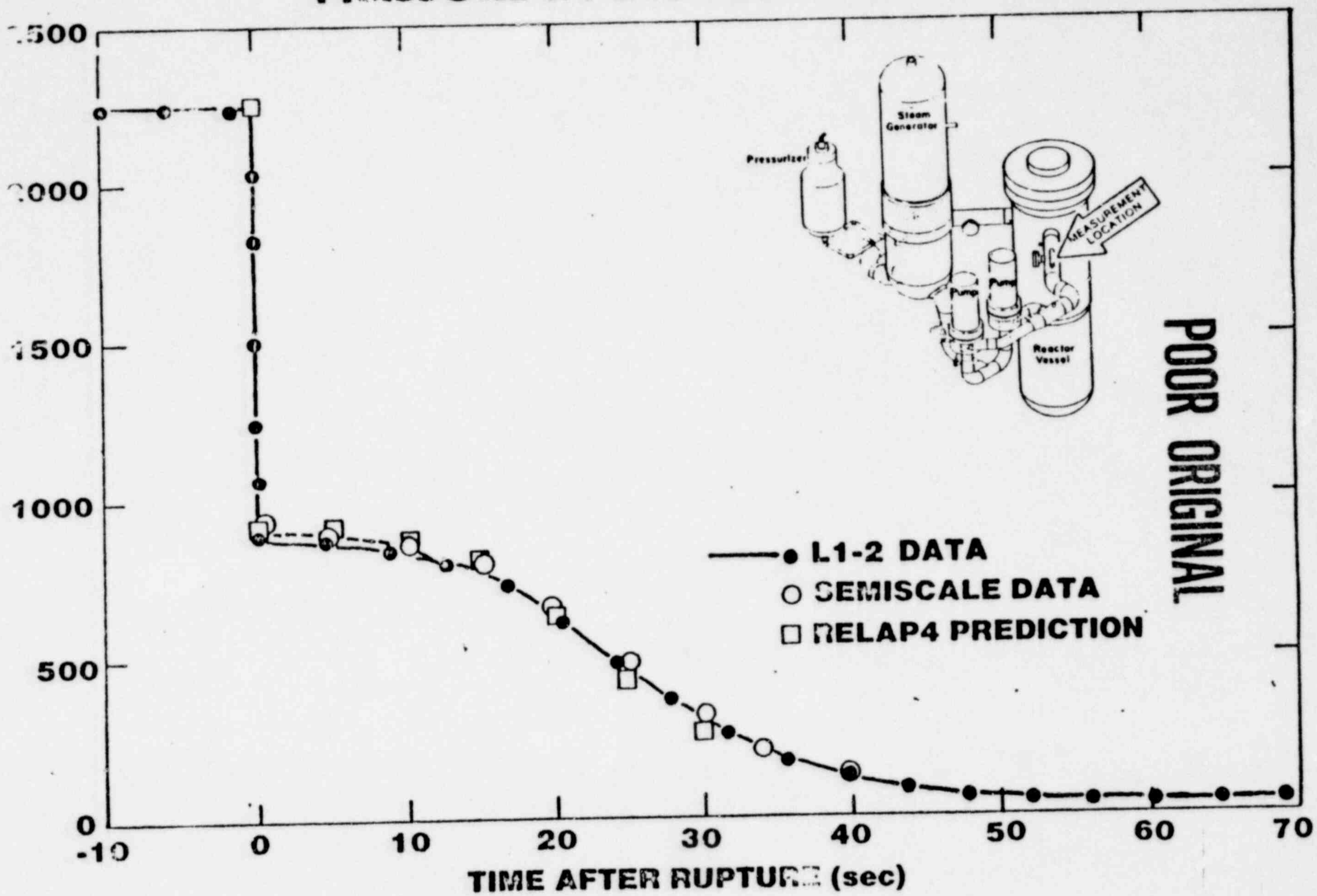


LOFT REACTOR VESSEL AND INTERNALS

1255 130

FIGURE 25

LOFT TEST L1-2
PRESSURE IN INTACT LOOP COLD LEG

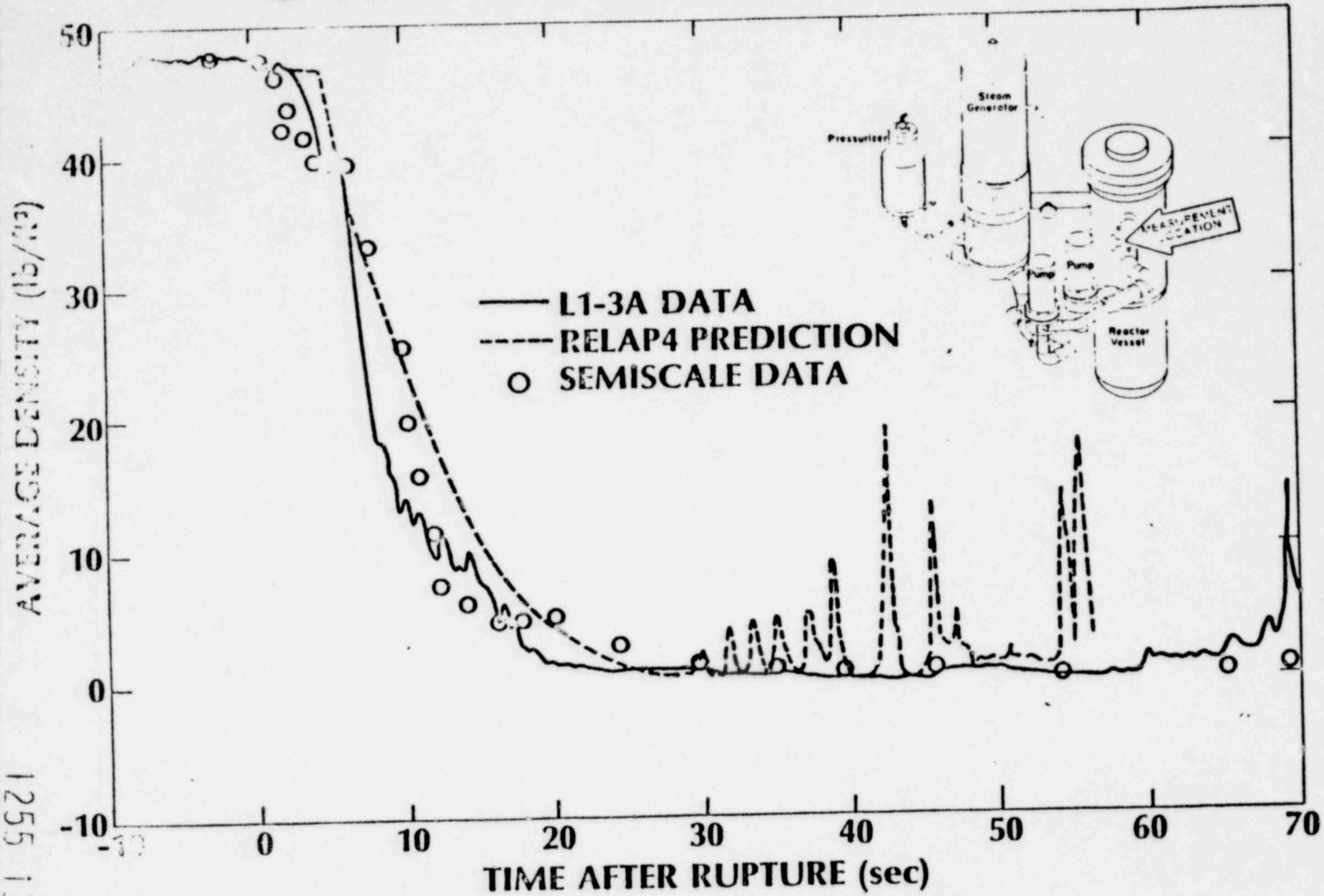


POOR ORIGINAL

1255 131

FIGURE 26

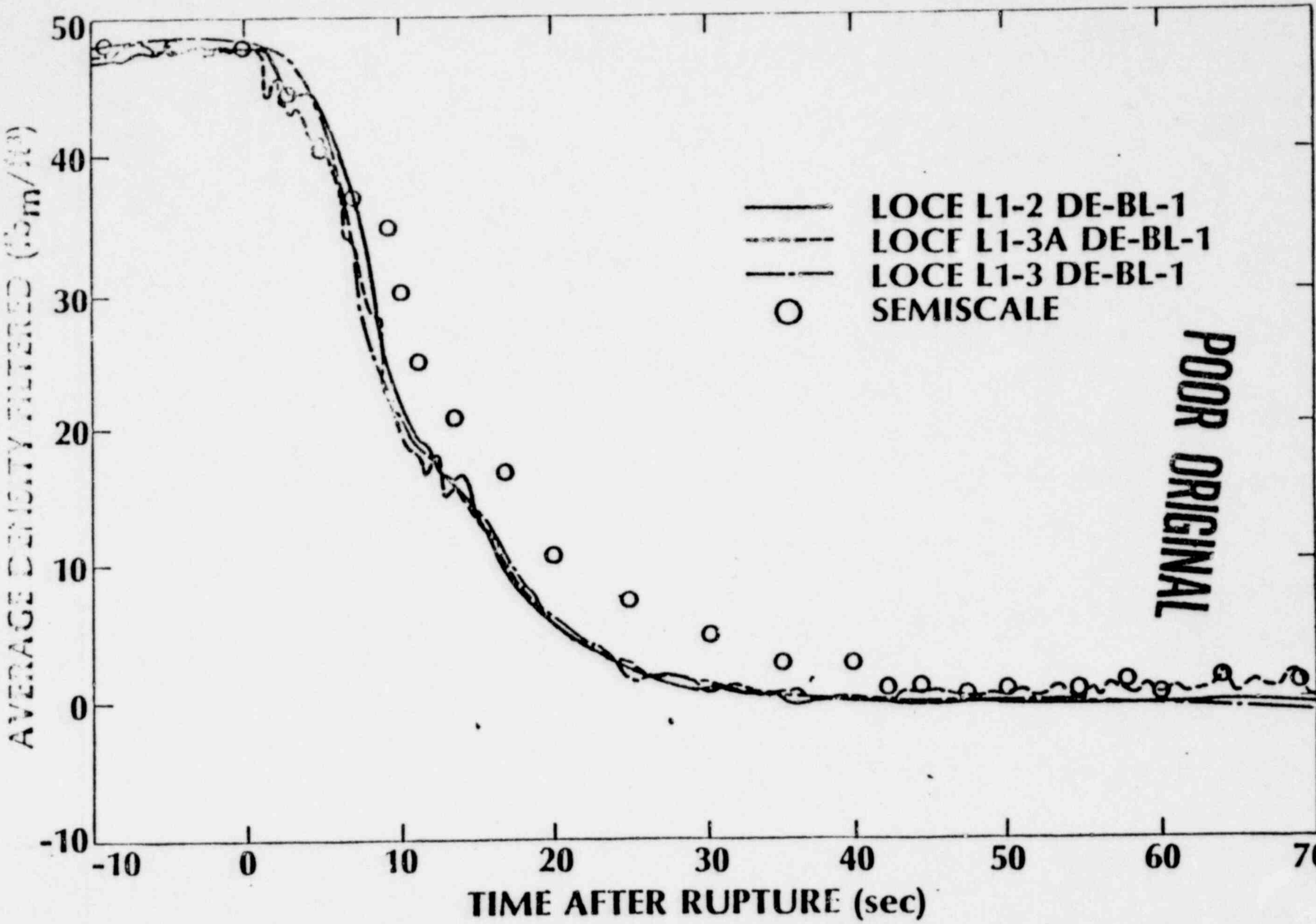
LOFT LOSS-OF-COOLANT EXPERIMENT L1-3A
DENSITY IN INTACT LOOP COLD LEG



1255 132

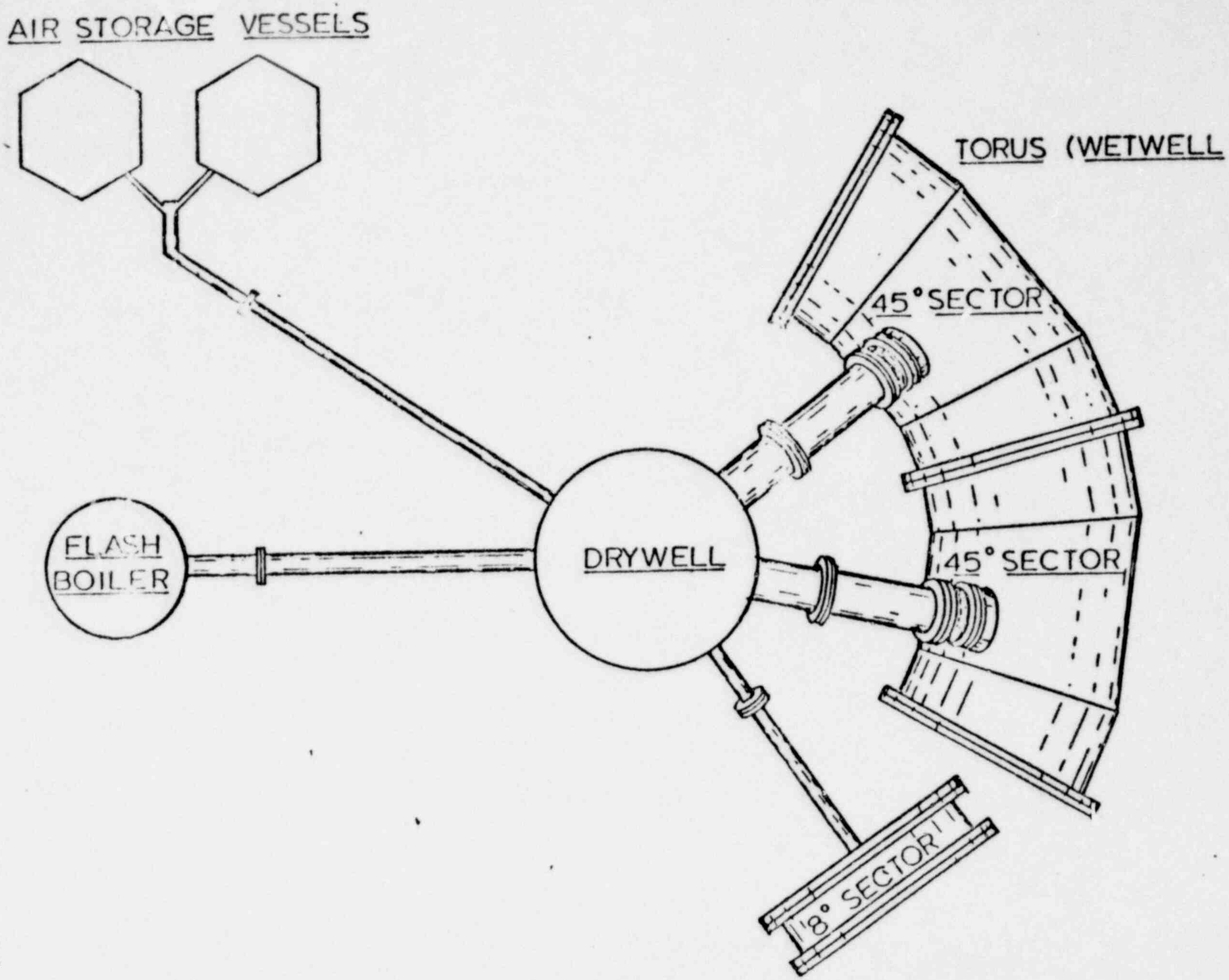
FIGURE 27

DE-BL-1 DENSITY OF FLUID IN BROKEN LOOP COLD LEG



1255 133

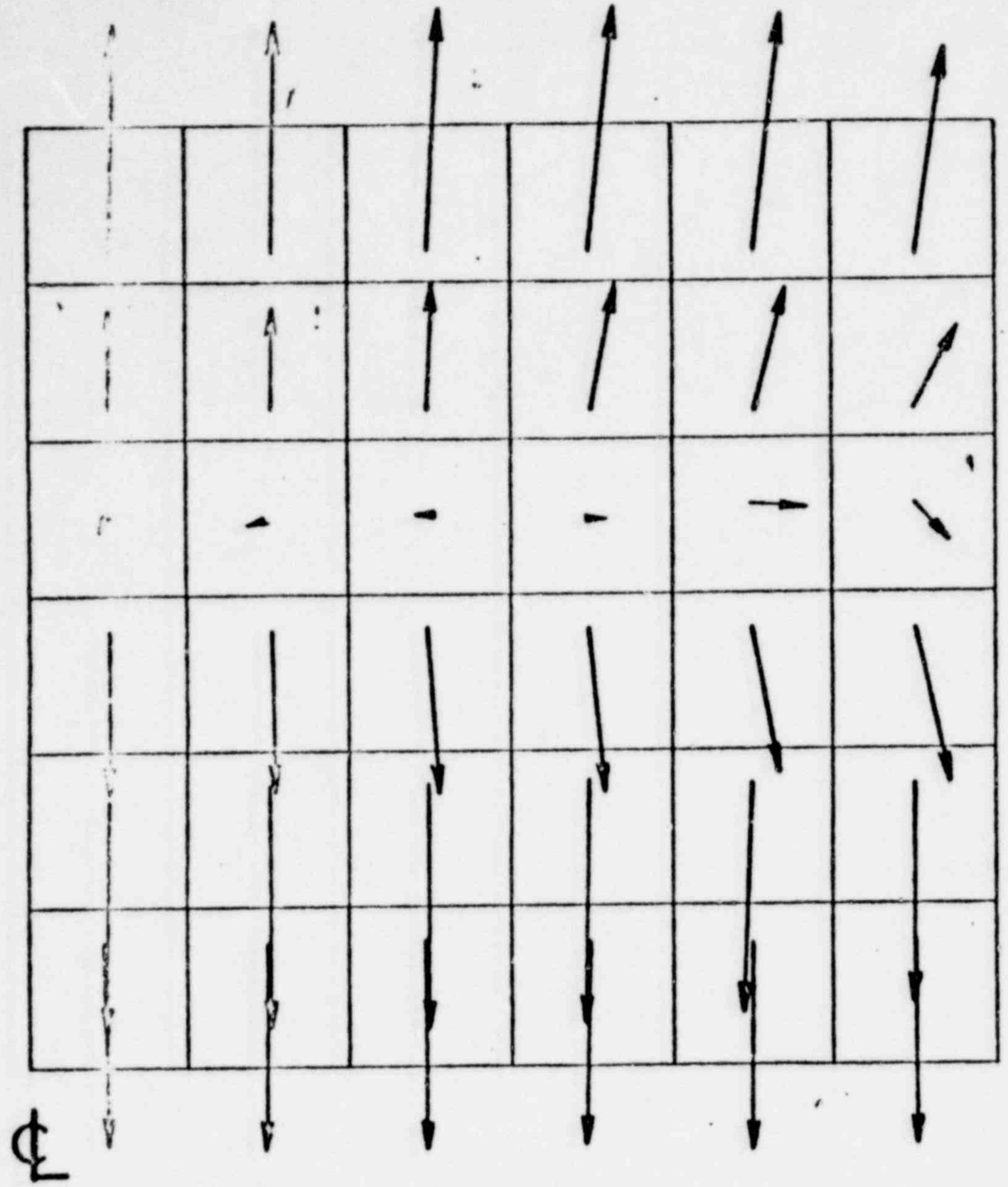
FIGURE 28



1255 134

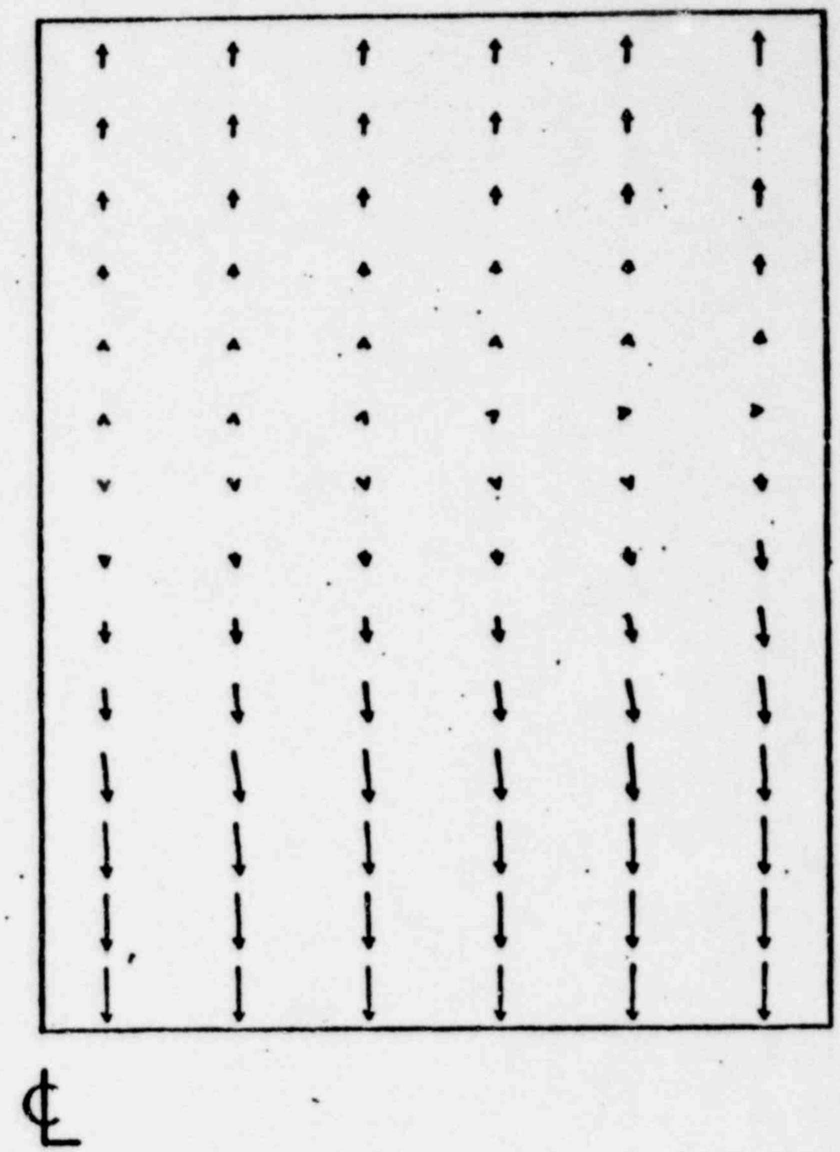
SCHEMATIC LAYOUT OF 1/5 SCALE PRESSURE SUPPRESSION EXPERIMENT

FIGURE 29



TIME - 0.300 second

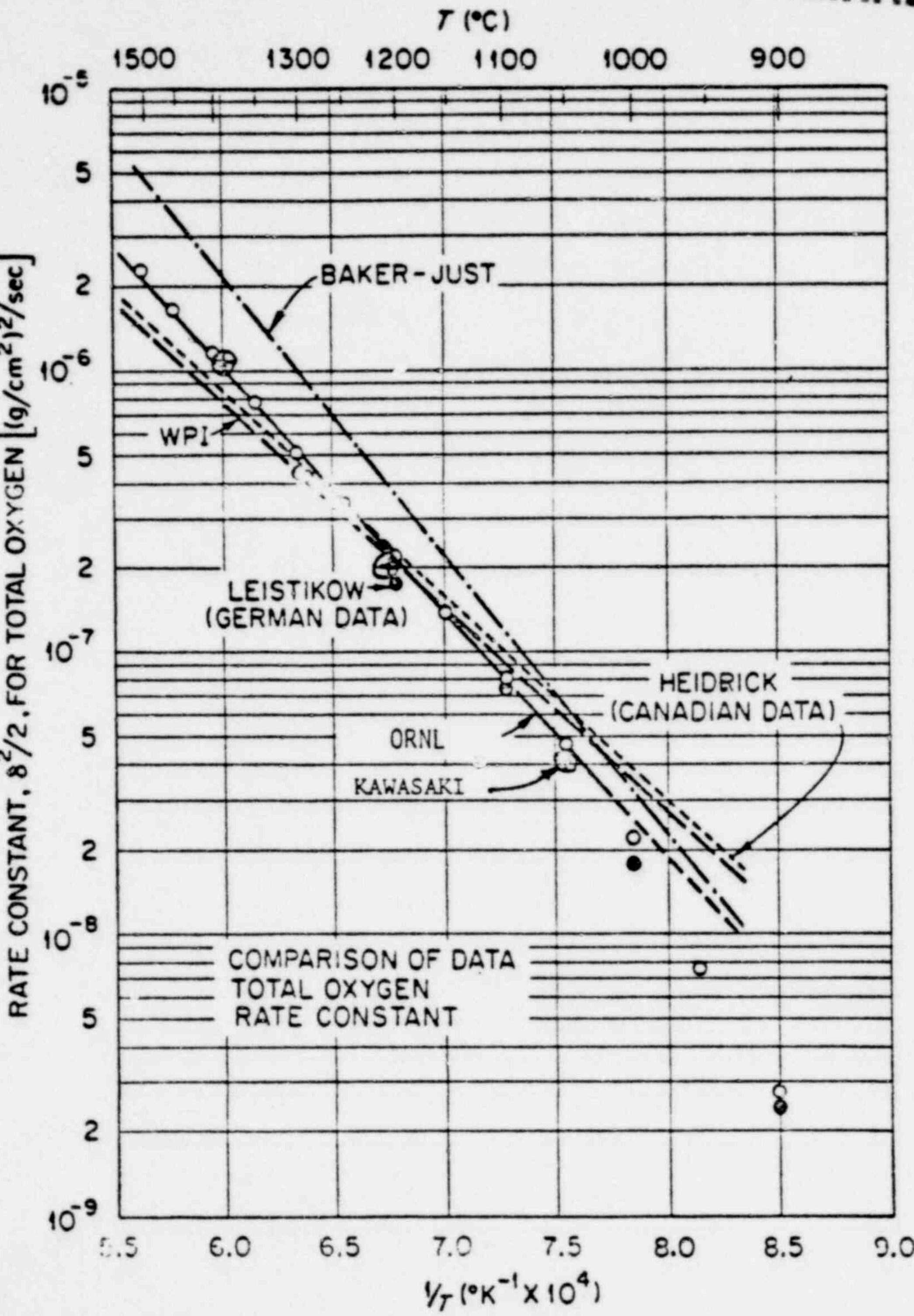
SCORE



TIME - 0.253 second

COBRA-4

FIGURE 30



Baker-Just (Ref. 75)
WPI (Ref. 79)
Leistikow (Ref. 82)
ORNL (Ref. 78)
[illegible]ki ([illegible] 80)
[illegible]

Arrhenius Plot of Oxidation Rate Data for Isothermal Oxidation of Zircaloy-4 in Steam, Total Oxygen Consumed.

FIGURE 31

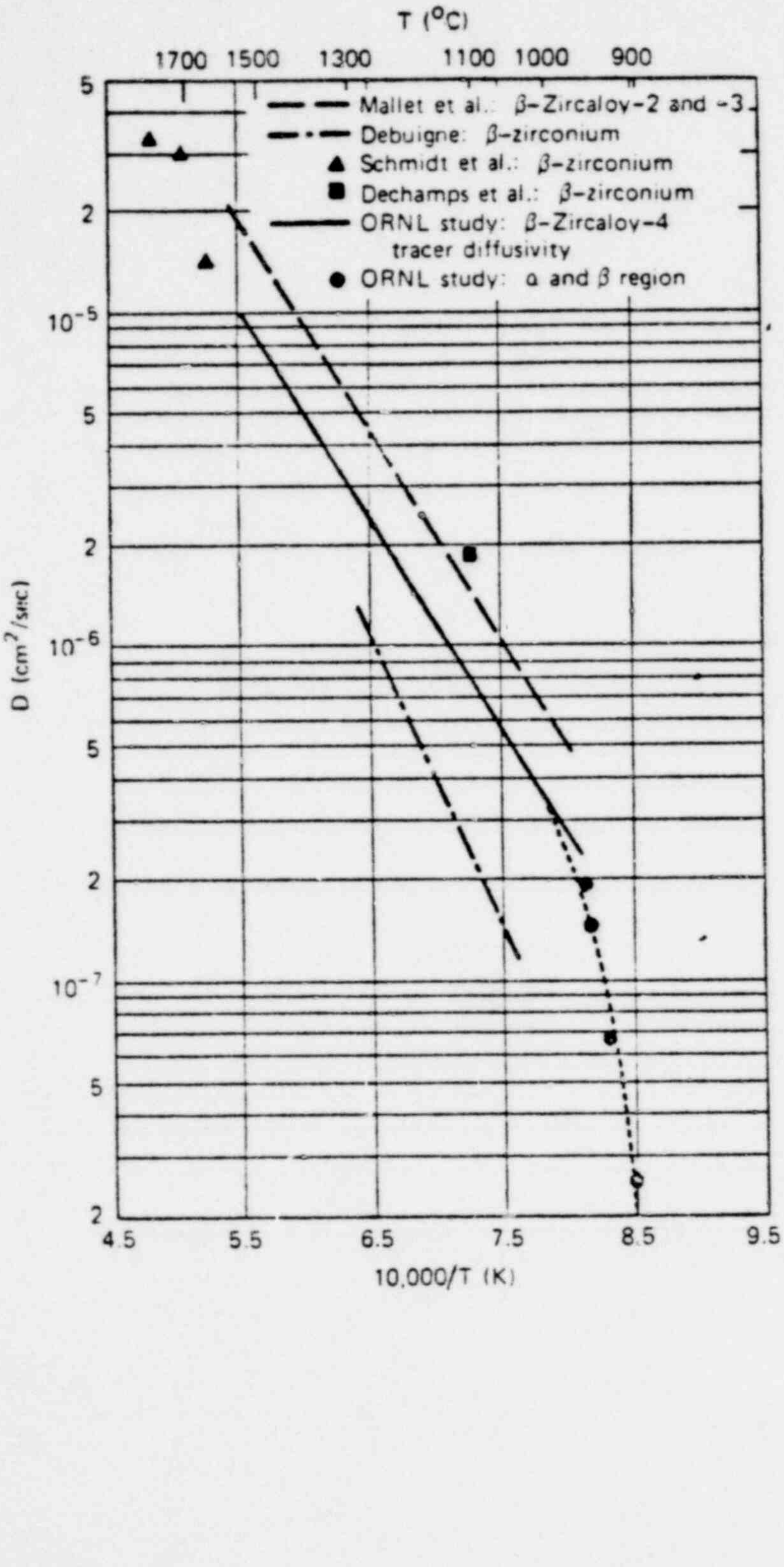


FIGURE 32

1255 137

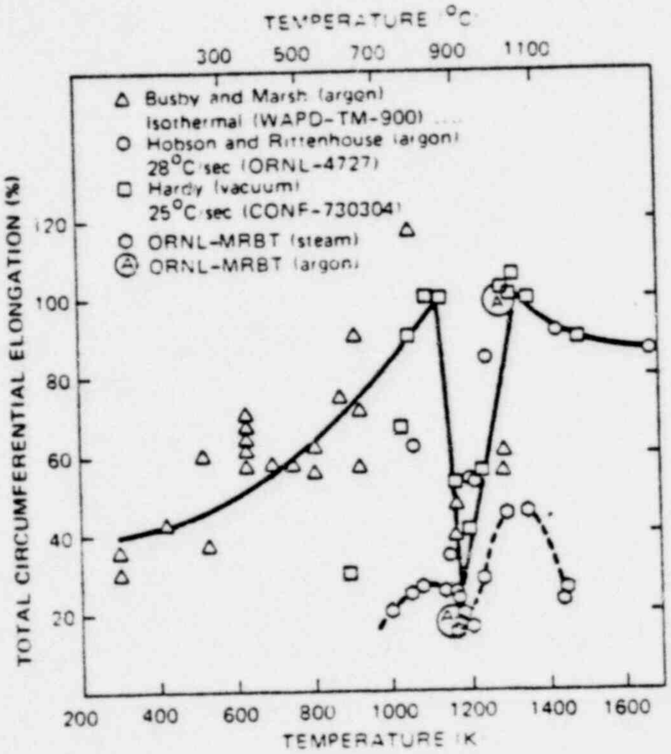
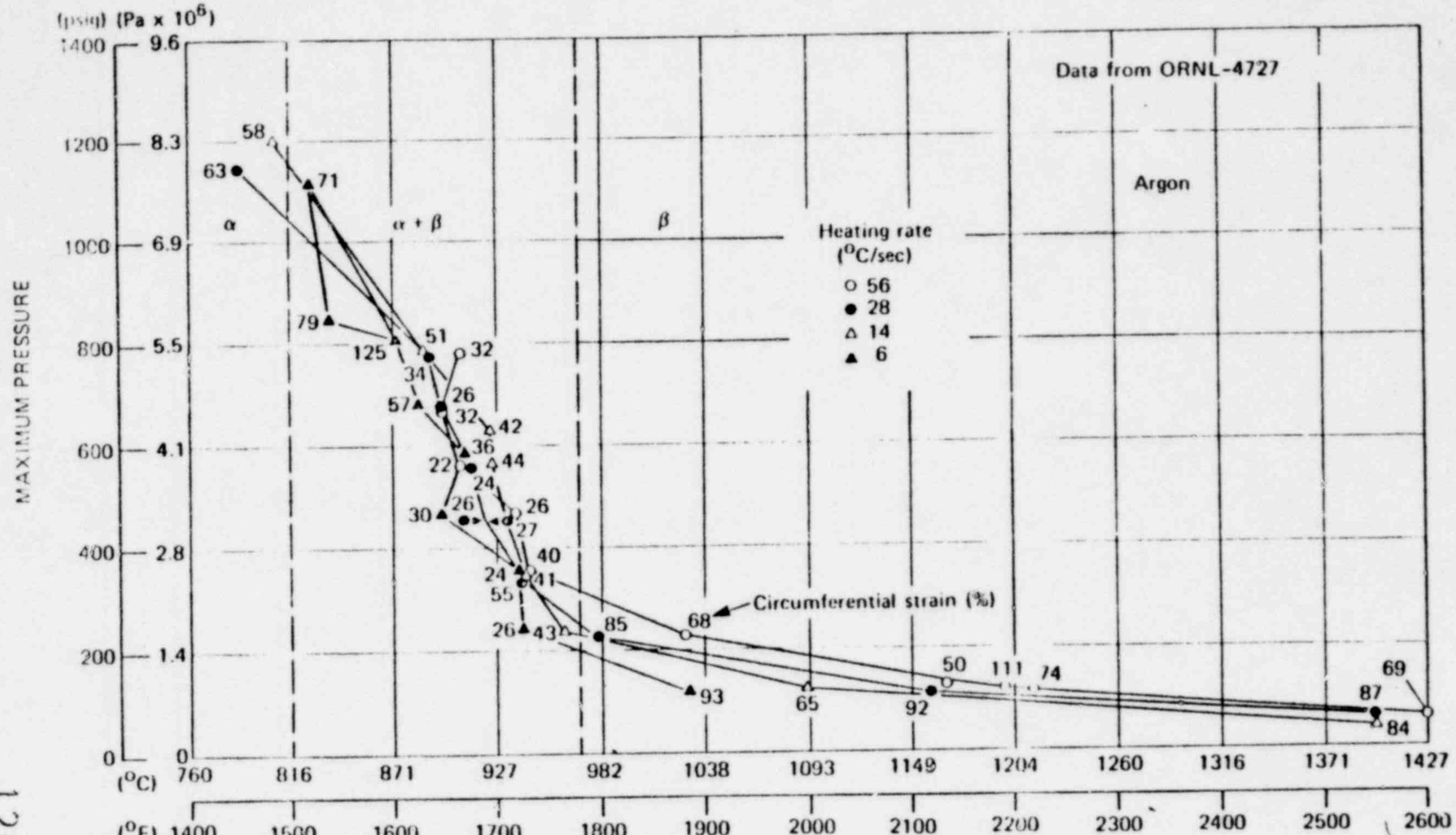


FIGURE 33

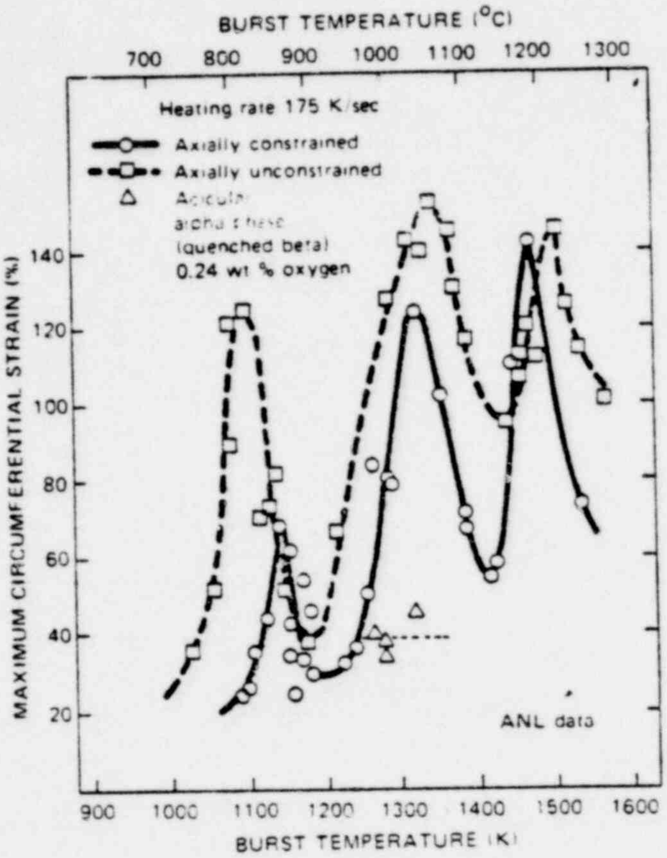
1255 138



1255 139

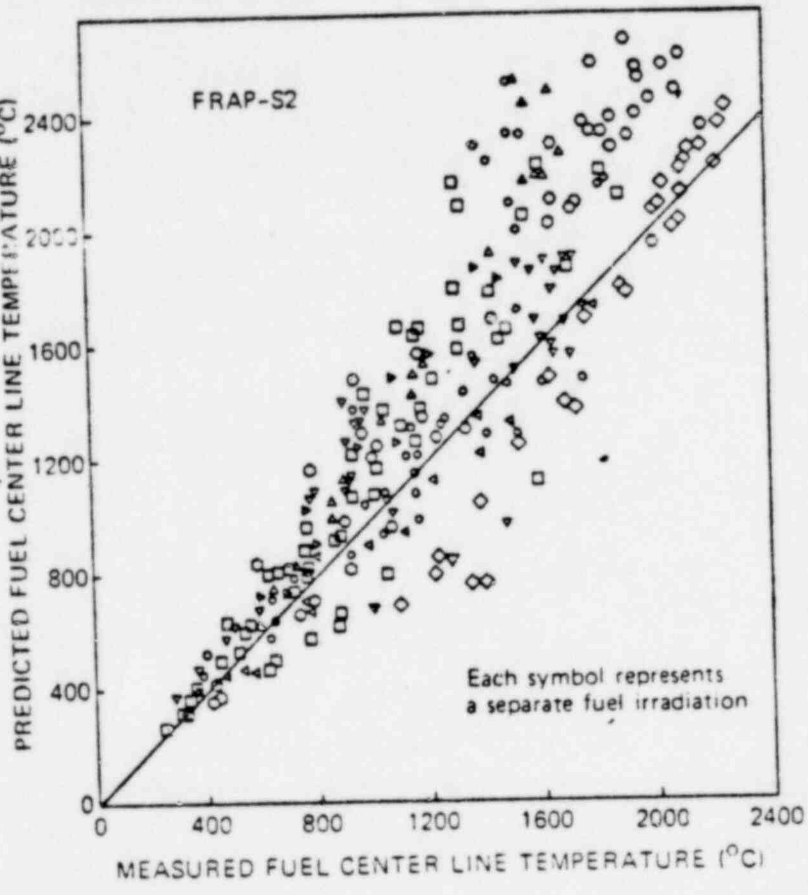
FIGURE 34

POOR ORIGINAL



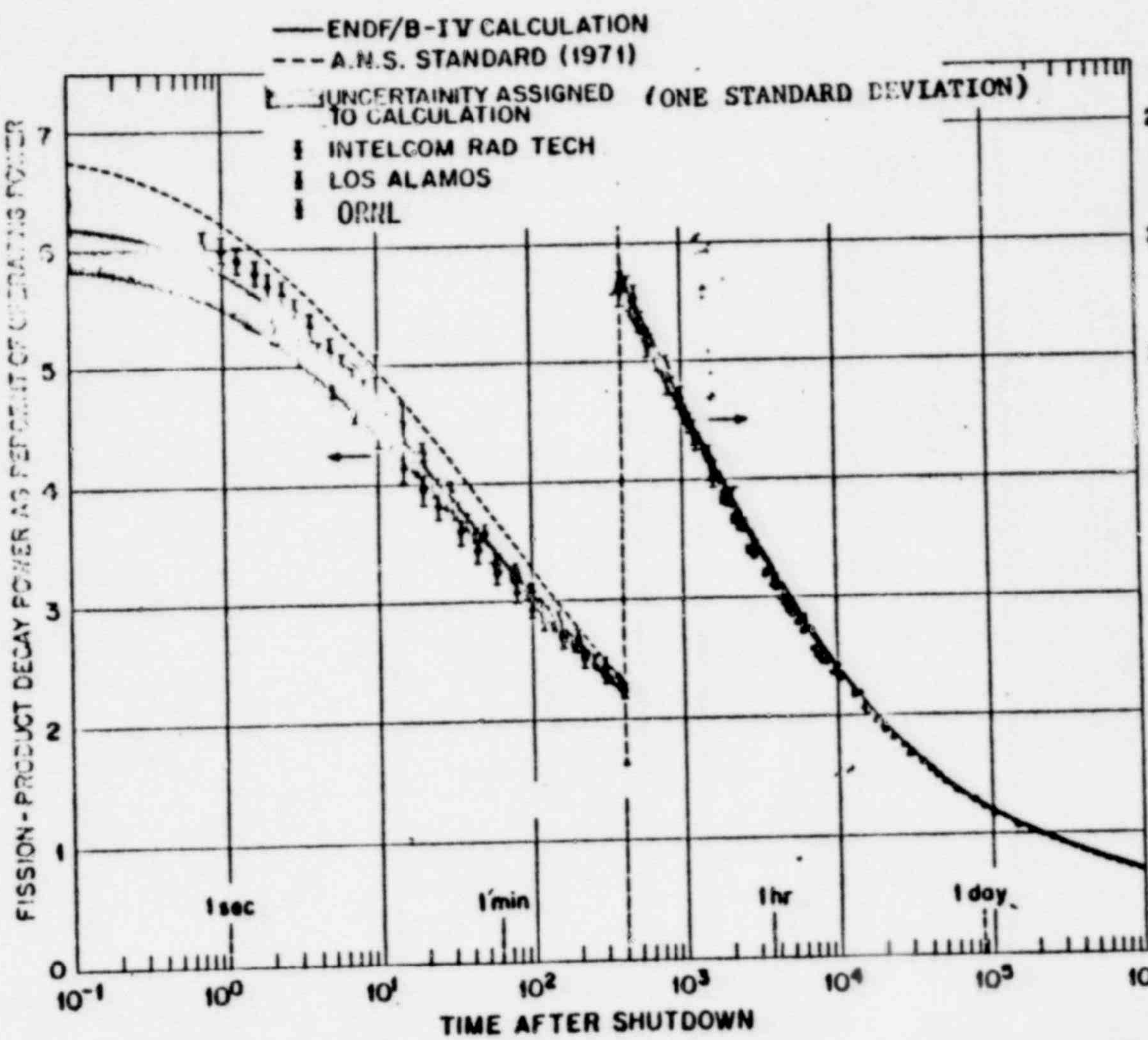
1255 140

FIGURE 35



1255 141

FIGURE 36

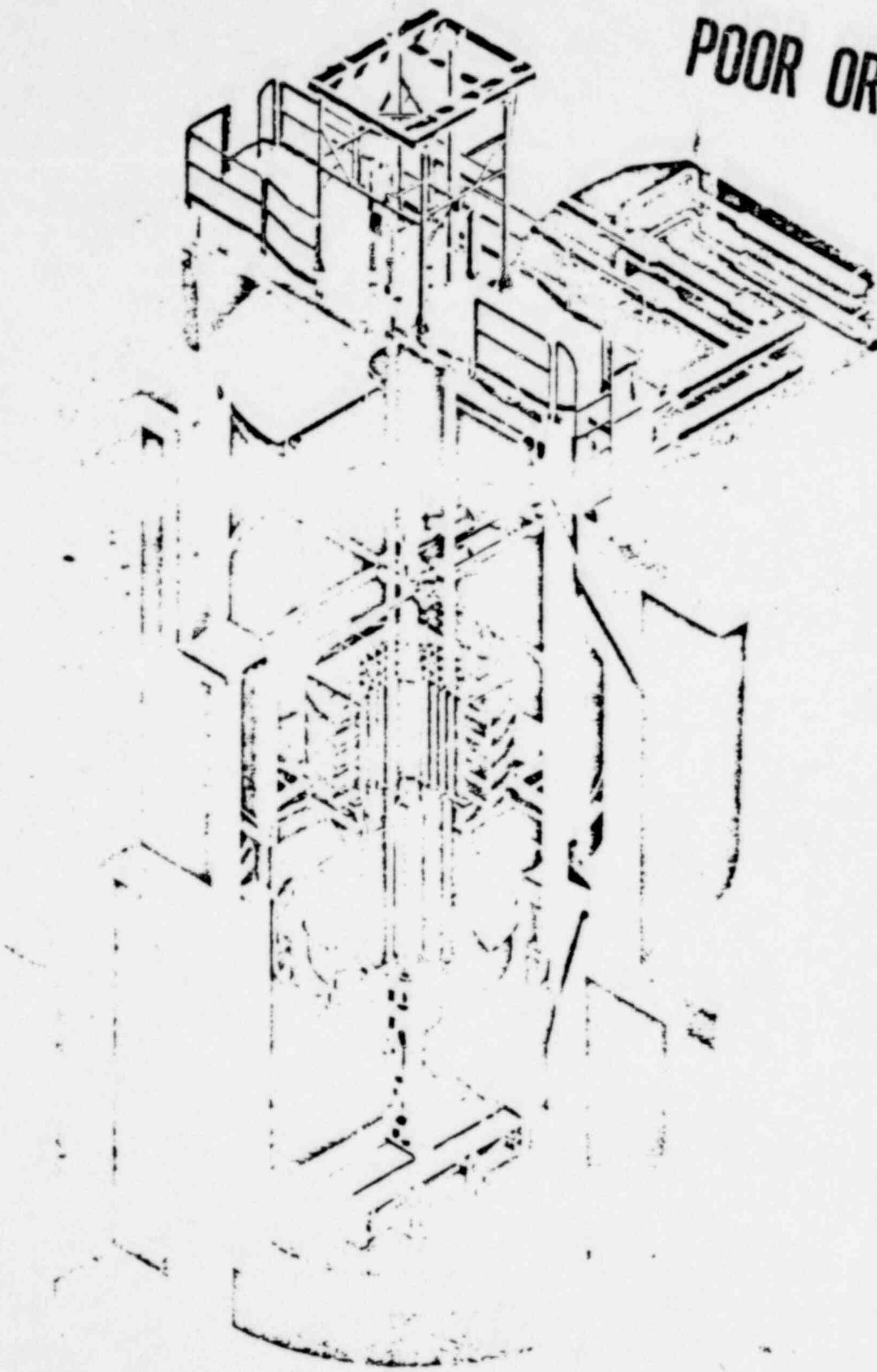


POOR ORIGINAL

FIGURE 37

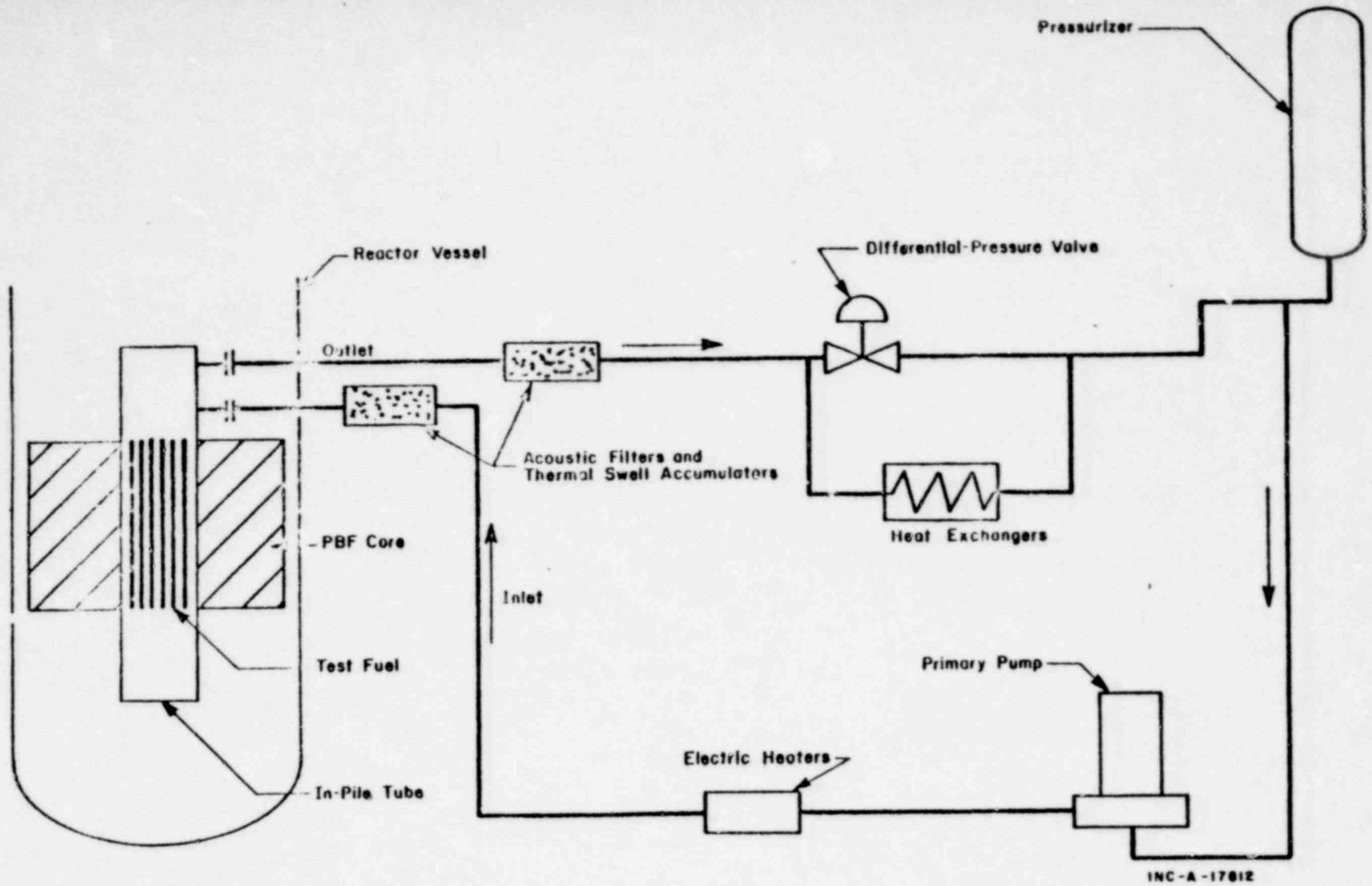
1255 142

POOR ORIGINAL



POWER BURST FACILITY

FIGURE 38



MAJOR COMPONENTS OF PBF IN-PILE AND LOOP COOLANT SYSTEMS

FIGURE 39

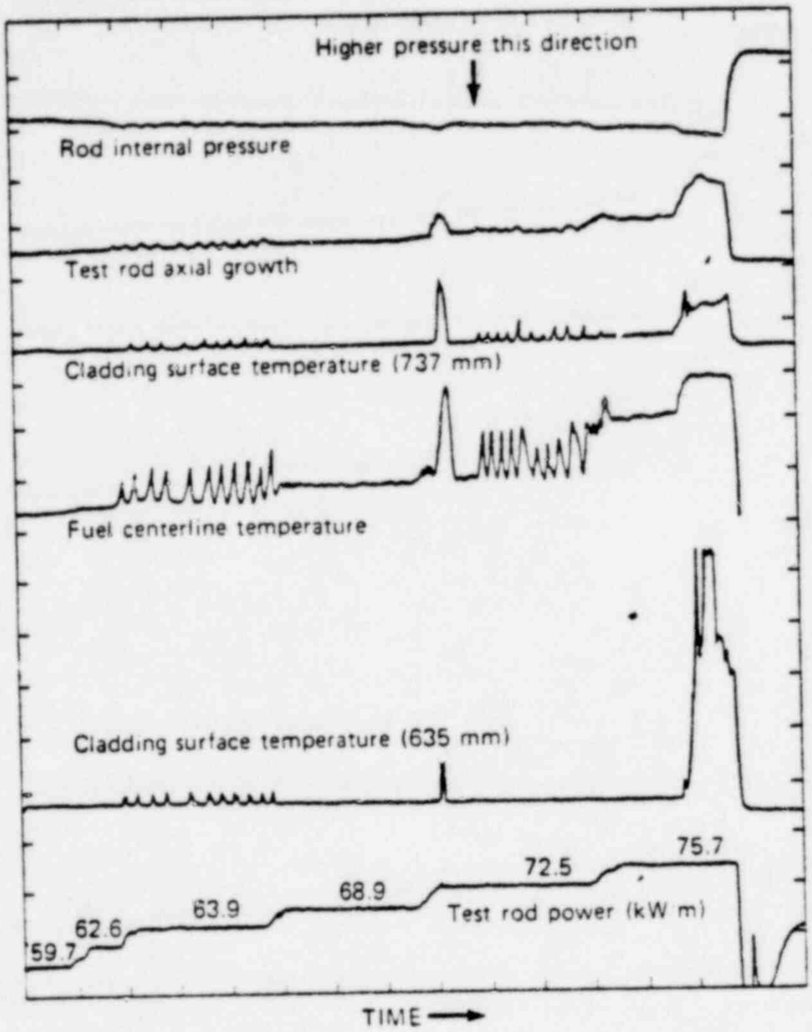


FIGURE 40

1255 145

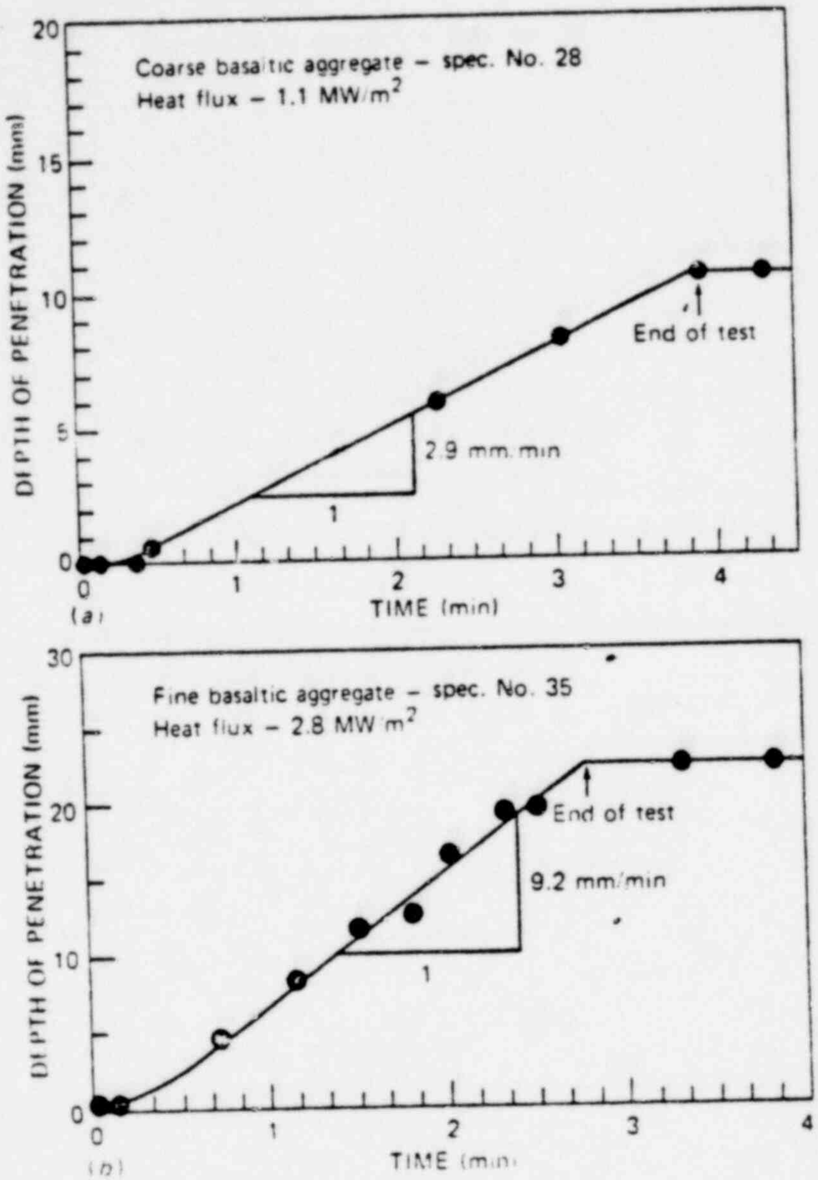
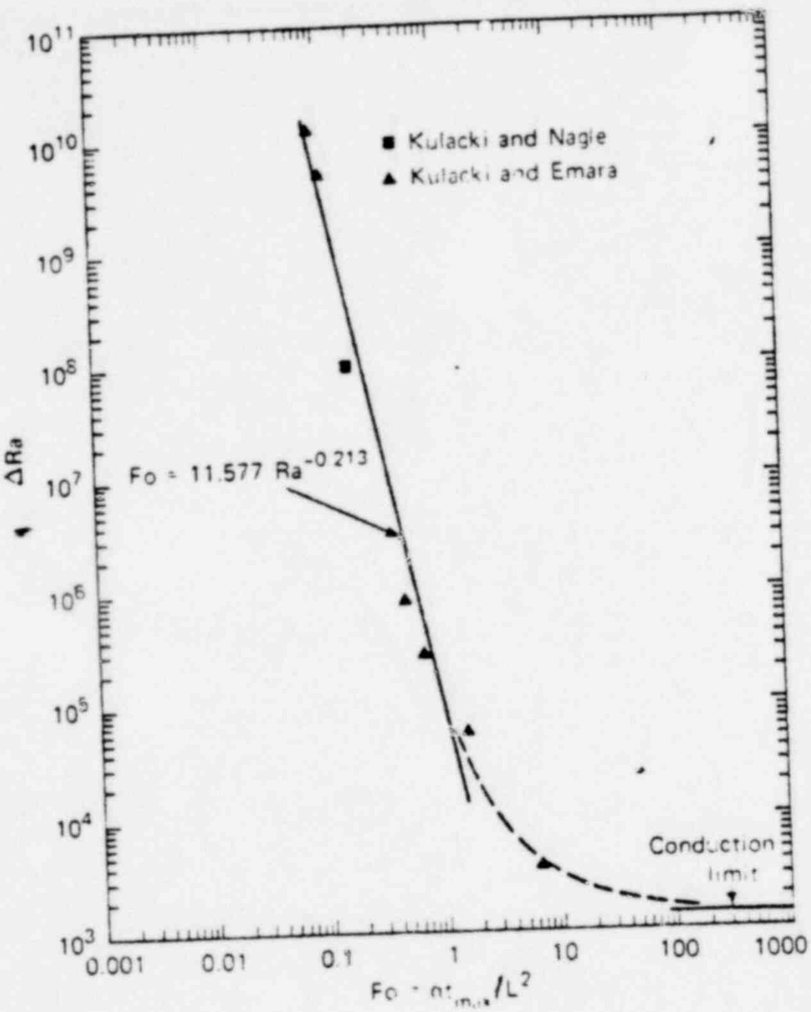


FIGURE 41

1255 146



1255 147

FIGURE 42