TELEX 492-7603

TO: Ashok Thadani, NRC

FM: C. E. [illegible], [illegible] Licensing

RE: Responses to Staff Questions on Moderator Coefficients as Used in ATWS Analysis 1-22-76

We intend to include the following questions and responses in the next formal ATWS submittal to NRC staff. Should you have any corrections to the questions as written herein, please inform me. We would like to treat these questions/responses with the same weight as given to telephone conference calls.

<u>QUESTION</u>: Give bases used to predict moderator coefficient as a function of Xenon, Boron, and rod position for first and subsequent fuel cycles.

<u>RESPONSE</u>:

The bases used to predict moderator coefficient for ATWS analyses is that a conservative value be determined which would be valid for at least 95%-of-life. The assumptions from these bases are that the calculated moderator coefficient be determined for full-power equilibrium Xenon, with all full-length rods withdrawn. These assumptions are applied to moderator coefficient determinations for all fuel cycles.

<u>QUESTION</u>: Provide information on moderator coefficient comparisons/measurements for first core and reloads. Discuss accuracies, bias, trends, and doppler subtractions. Discuss extrapolation to full-power values and associated accuracies. Compare calculated to measured values.

<u>RESPONSE</u>:

Moderator coefficient comparisons of predicted to measured

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values for first core and reload show negative bias. Past statistical summaries indicate a mean bias of $|.2 \times 10^{-4}\ \Delta k/k/°F|$ with a 2σ of $|.4 \times 10^{-4}\ \Delta k/k/°F|$. Measurements at zero power and full power show no appreciable difference in accuracy. Boron predictions of boron concentration, for example, show deviations of less than 30ppm between measured and calculated for both first cores and reloads.

Power Doppler measurements have tended to be more positive than calculated. Studies of measurement and calculation techniques have [led] to incorporation of a fuel temperature dependency and re-[illegible] calibration of available flux information. Most recent measurements [illegible] using these techniques — approximately 1/2 through a first cycle — indicates agreement to within 6%.

There is a doppler subtraction when ATWS moderator coefficients are quoted. A moderator temperature coefficient is determined, and a doppler coefficient subtracted. These coefficients then determine the appropriate time for detailed ATWS analysis of a given fuel cycle, using a reactivity vs. moderator density curve. Fuel temperature is fixed at nominal values during determination of these curves, and therefor doppler coefficient has no impact in determination of these curves. For ATWS analyses, the most appropriate calculated value of doppler coefficient is selected independent of moderator temperature coefficient.

S. Dawmont to A. Thadani

QUESTION : Discuss assumptions used, and bases, for determining moderator coefficient for future reloads. Are there any differences from reactor to reactor or plant type to plant type? If there is a need to stretch a fuel cycle, how is the moderator coefficient affected?

RESPONSE :

Currently, B&W does not offer much flexibility in nominal fuel cycle length. Therefore this confines all reloads to nominal conditions as analyzed in BAW-10099.

In BAW-10099 an effective moderator coefficient corresponding to each B&W plant type was used in determining the moderator density curves.

QUESTION : How are operating modes considered in moderator coefficient predictions?

RESPONSE :

The basis for ATWS calculations was to look at rated-power ATWS events. Therefore the only assumptions used are those given in the response to question (1).

QUESTION : Should there be Technical Specification limits imposed with respect to moderator coefficient in consideration of ATWS events?

RESPONSE :

should be in terms of probabilistic methods. Consistent with this approach, moderator coefficient values used in ATWS analyses should be those anticipated for classes of reactors. Specification of limiting moderator coefficient values on specific facilities would not be consistent with the "probabilistic" approach and wouldn't be consonant with the [illegible] [illegible] [illegible] expressed by WASH-1270 [illegible]

Enclosure 2 FROM: C. S. BAHWARTH
SUBJ: MATERIAL FOR
11/12 ATWS MEETING

EXAMPLE EVENT-TREE ANALYSIS

The "Technical Report on Anticipated Transients Without Scram For Water-Cooled Power Reactors", WASH-1270, embodies the concept that event sequences with probabilities less than 10^{-7} per year should not be considered. Specifically, in Section III, the following is stated:

> "The safety objective is that the likelihood of all accidents with significant consequences not included in the design basis envelope should not be greater than one chance in one million per year; i.e., should not occur with a failure rate greater than 10^{-6} per year. For the particular potential failure path of ATWS, the staff believes that a failure rate of the order of one tenth of the overall safety objective is an appropriate objective."

The 10^{-6} to 10^{-7} per year probability criterion is consistent with previous NRC and industry usage.

In the Babcock & Wilcox NSS, the limiting "Anticipated Transients" are Loss of Feedwater (LOFW) and Loss of Off-site Power (LOOP). Event trees, as used in the Reactor Safety Study (RSS), WASH-1400, for these transient sequences have been performed. It is the Babcock & Wilcox position that LOOP should be evaluated using operable relief valves and a most probable moderator coefficient. B&W has also determined the LOFW should be evaluated using operating electromechanical relief valves and a 75%-of-life moderator coefficient, and this analysis follows.

In the LOFW, the critical event sequence is the transient followed by failure to trip due to mechanical common mode failure (CMF) of the control rod drive mechanisms (CRDMs). If trip failure due to RPS electronics CMF occurs, the Integrated Control System (ICS) can runback rods to yield acceptable consequences. It will be shown that the sequence (transient, RPS failure, ICS failure) has probability much less than 10^{-7} and should not be considered.

It will also be shown that the probability of LOFW and CRDM failure coupled with failure of powered relief valves is much less than 10^{-7} per year using more conservative data than that used in the RSS. Therefore, it is appropriate to assume operability of these valves in the analysis of LOFW.

The critical sequence determining the criterion on the analysis moderator coefficient is then (transient, CRDM failure, relief valves open). The probability of this sequence multiplied by the probability that the actual moderator coefficient exceeds that assumed in analysis must be equal to or less than 10^{-7} per year. As will be shown, a calculational moderator coefficient which is not exceeded over at least 80% of plant life assures compliance with the criterion.

With this accident sequence of transient - no trip, operating relief valves, and a 80% moderator coefficient - the probability is less than 10^{-7} per year that an actual transient's severity would exceed that calculated.

Figure 1 shows the event tree for the sequences initiated by loss of feedwater. Nodal or decision points are labeled with letters. Discussions of specific events are referenced with superscript numerals. Probabilities of specific branches are indicated in parentheses.

The nodal points defining the structure of the event tree are discussed below as keyed to the event tree in Figure 1.

A. Following LOFW, the trip function performs or fails to do so. If trip occurs, plant shutdown with acceptable consequences results. If trip fails, other events must be considered.

B. If trip function fails, either the RPS electronics have malfunctioned in which case the CRDMs are not bound and are not de-energized, or the CRDMs have suffered a mechanical CMF. In the latter case, the RPS has de-energized CRDM stators and ICS runback is discounted.

C. No trip has occurred but the CRDMs are energized and operable. The ICS either runs back thre rods or fails to do so. If runback occurs, acceptable consequences result. As will be shown subsequently, the sequence culminating in no runback has probability less than 10^{-7} per year.

D. If trip failure is caused by CRDM failures, either the powered relief valves open or fail to do so. If they open, subsequent events are considered. The probability of the sequence ending with relief valve failure is subsequently shown to be less than 10^{-7} per year.

E. The moderator coefficient, α_m, is either less than or equal to or greater than the value assumed in mechanistic analysis.

The specific events or branches in the event tree are discussed below:

1. The WASH-1270 number of once per year for all "anticipated transients" is conservatively applied to LOFW alone. - 4 p. 15 of App D vol 1
upper bound of 1x10

2. Reactor Trip: the RSS used a value of 3.6 x 10^{-5} as the CMF - induced unavailability of reactor protection systems. However, in BAW-10016, a CMF analysis of the B&W RPS showed that system to possess much less CMF vulnerability than the average due to the large amount of diversity built into the system from Oconee I on. In that report a conservative estimate of CMF - unavailability of 10^{-5} was derived and is used here.

3. Epler in 1969 reported seventeen instances of actual or potential RPS (electronics) CMFs in research reactors. Since then, at least four actual or potential RPS (electronics) CMF events in power reactors have occurred. Although the potential and partial failures may not be used to estimate the probability of total system failure, they may be used to estimate the relative probabilities of RPS (electronics) failures and of mechanical CRDM failures. More than twenty actual and potential RPS (electronics) failures have happened; no CRDM CMFs have occurred. Conservatively assuming that the next CMF of trip function is

due to CRDMs, then the relative probability of CRDM failure to RPS (electronics) failure is less than one in twenty. However, a conservative upper-bound value of 0.05 for the conditional probability of CRDM CMF given trip function failure is used here. The conservatism of this value is established by the CMF evaluation of the B&W roller nut drive as reported in BAW-10101P.

4. Given that the CRDMs are unfailed and energized, the ICS is capable of running back the rods. B&W has demonstrated the diversity of RPS and ICS in BAW-10099. The two systems operate on diverse principles, using independent sensors, were designed by independent groups, and are manufactured of diverse components. They are not susceptible to common-mode failure.

   Although a detailed reliability analysis of the ICS has not been performed, the results of the RSS show that the unavailability of operational systems lies between 10^{-2} and 10^{-4}. We conservatively assume the 10^{-2} number applies to the ICS ability to perform runback.

   Safety Valves only P. 62 of vol V

5. Relief Valves: The RSS uses a value of 3×10^{-5} as the probability that any one of three powered relief valves will fail to function upon demand. A more conservative value of 10^{-3} is used here.

6. Moderator Coefficient: The probability of the actual moderator coefficient exceeding the value assumed in analyses is treated below.

The event sequences and the conclusions from evaluation of these sequences are described below.

I. (LOFW, Trip) results in safe shutdown.

II. (LOFW, No Trip, CRDM operable, ICS runback) results in safe shutdown.

III. (LOFW, No Trip, CRDM operable, No ICS runback) has probability: $1/\text{year} \times 10^{-5} \times 0.95 \times 10^{-2} = 9.5 \times 10^{-8}/\text{year}$, which is less than 10^{-7}/year and should not be treated further.

IV. (LOFW, CMF of CRDM, Relief valves failed) has probability: 1/year x 10^{-5} x 0.05 x 10^{-3} = 5 x 10^{-10}/year, and should not be considered further.

V. (LOFW, CMF of CRDM, Relief valves open, $\alpha_m < \alpha_{calc}$) has consequences less than the sequence in which $\alpha_m = \alpha_{calc}$ and is not considered further.

VI. (LOFW, CMF of CRDM, Relief valves open, $\alpha_m \geq \alpha_{calc}$) is the critical sequence which has probability:
1/year x 10^{-5} x 0.05 x 1 x Pr ($\alpha_m \geq \alpha_{calc}$).
The value of moderator coefficient to be used in calculations, α_{calc}, is that value which assures that the probability of this sequence is equal to or less than 10^{-7}/year. Solution of this equation results in using a moderator coefficient which is not exceeded over at least 80% of plant life.

Calculations using operable relief valves and an 80% moderator coefficient assure that the probability of a LOFW sequence with more severe consequences than that calculated is equal to or less than 10^{-7} per year. To assure conservatism, B&W has utilized a 95%-of-lifetime moderator coefficient value in their ATWS analyses (BAW-10099).

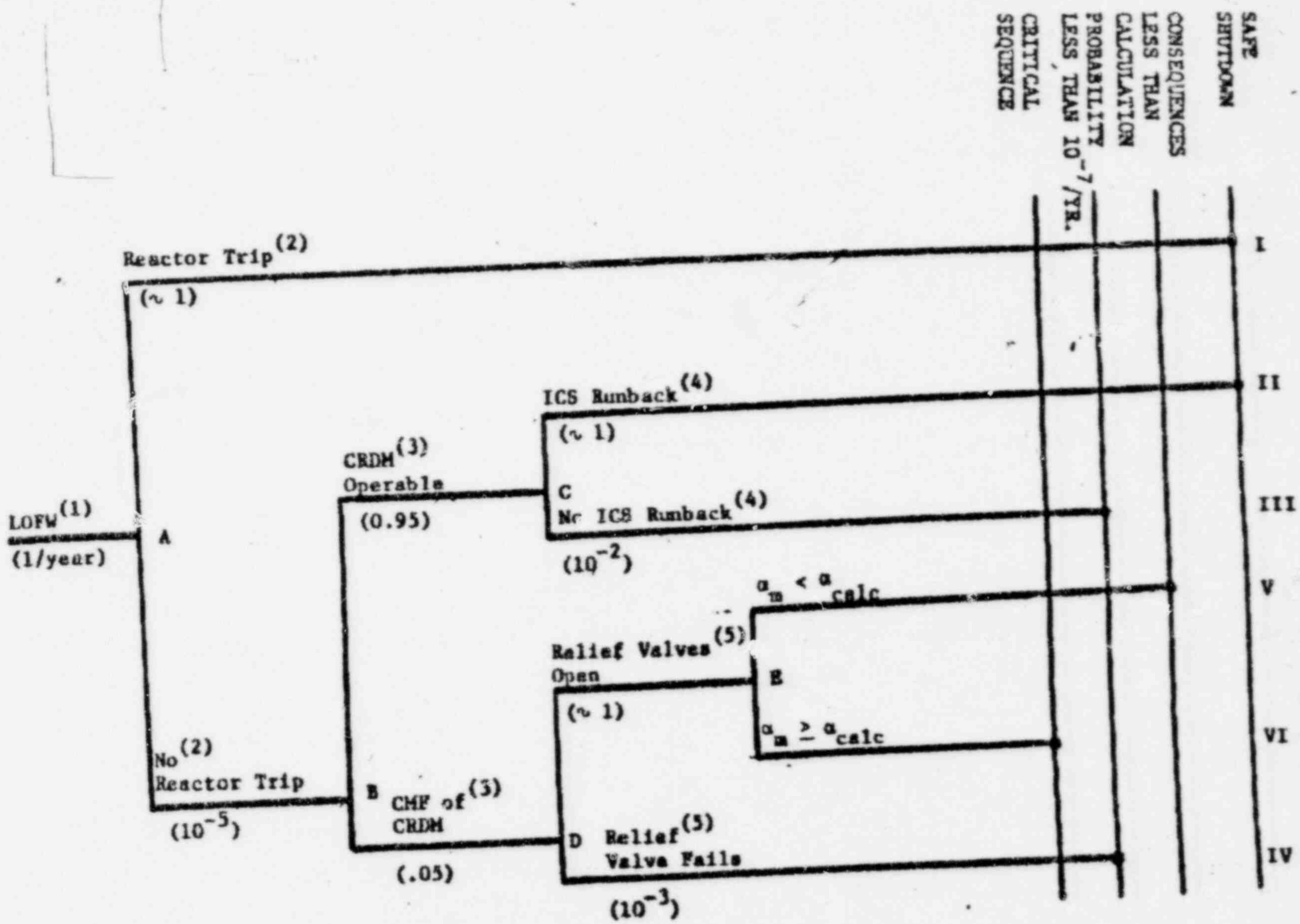


FIGURE 1 - LOFW EVENT TREE

Assume dilution in progress

RESPONSE: A dilution operation coincident with an ATWS event results in ~~the~~ an event sequence with a probability ~~outside~~ less than the 10^{-6} - 10^{-7} safety objective of WASH-1270.

B+W has researched the ~~operating~~ records of several operating B+W units and has determined the following as being typical schedules:

Assume one safety valve fails to reclose

PONSE: The consequences of ~~a~~ failure a safety valve to reclose following all ATWS events have already been established by the analysis of the pressurizer-stuck-open ATWS event. Thus, B&W does not consider that ~~any new information~~ additional analyses will provide any new information in this area.

We need to show staff the specifics of why it's bounded

Demonstrate operability of isolation valves.

PONSE:
i+W will demonstrate operability of the isolation
lves.

Assume one power-operated relief valve fails to open.

PONSE: The probability of an Anticipated Transient, shutdown system failure, ~~and~~ with α_m 99% probable, and an inoperable power-operated-relief valve is less than 10^{-7}, the safety objective of WASH-1270. Therefore, B&W does not consider such an event sequence to be an ATWS event.

Details of the B&W position have been provided on previous occasions.

assume on train of HPI fails to perform it function

RESPONSE: same as 5

assume only half the capacity of AFW is available

PONSE: same as 5

Assume no automatic initiation of emergency fan coolers, HPI, containment isolation unless diverse and reliable means for actuating is provided.

ONSE: This concern in essence questions the diversity of RPS and SFAS. B+W considers this concern to be irrelevant. If the shutdown-system failure is due to RPS failure, then ICS drives in the rods; or else they have dropped when de energized. In either case, SFAS may be assumed to have also failed. No matter, the rod insertion has turned the transient so that SFAS and the actuated systems are not needed.

Modification needed to take credit for rod run-in of ICS

PONSE: B&W considers a mechanical CMF of all CRDM to be of such low probability that a postulated "ATWS" due to such is outside the scope of WASH-1270. Therefore, no power-assurance system is needed.

Position Statement

Long-Term Transient ATWS Consequences

In topical report BAW-10099 and subsequent responses to NRC Questions 210.2-210.5 and 310.1-310.4 on BAW-10099, B&W has discussed in detail the termination of transient ATWS consequences, equipment availability, and anticipated long-term effects such as radiological consequences. The principal conclusions of this work are as follows:

ATWS Transient Termination

An ATWS event is effectively terminated when either the reactor is shutdown (zero power), the core is in a geometry amenable to cooling and is being cooled, and all other plant operating conditions such as containment and RCS pressure are within normal design limits or the radiological consequences during the entire transient are no worse than the guidelines of 10CFR100. In section 3.5 of BAW-10099, B&W has described how these goals can be accomplished through design features inherent in the design of the B&W PWR.

Reactor shutdown can be accomplished for all ATWS events through control rod insertion or reactor coolant boration. Unique features of the B&W NSS design are the capability to supply system makeup and boration through high pressure injection at elevated system pressures and ICS initiated control rod runback for this purpose.

Reactor shutdown would in turn limit RCS pressure and containment pressure within normal design limit. Containment pressure is additionally limited within normal design limit through operation of other engineered safety features such as the containment air coolers and reactor building sprays. Alternate sources of RCS heat removal are also summarized in Section 3.5.2.3 of BAW-10099 such that continuous long-term core cooling can be maintained.

B&W has thus established in BAW-10099 inherent design features in its NSS and overall plant designs which are available to mitigate transient ATWS consequences during long-term cooling and reactor shutdown. This conclusion is further supplemented by the response to NRC Question 210.2 on safe shutdown conditions which also includes a qualitative assessment of long-term sequence of events and system conditions for the ATWS events analyzed.

Equipment Availability

The equipment assumed to operate in mitigating ATWS consequences was based on criteria stated in section 3.3.1.1 of BAW-10099. Specific equipment is delineated in sections 3.3.1.2, 3.3.1.3 and 3.3.2. Additional information on equipment availability and operation is presented in response to NRC Questions 210.4 and 210.5 and in BAW-10101P, "Anticipated Transients Without Scram Program-Common Mode Failure Analysis of Control Rod Drive Mechanism".

Radiological Consequences

The radiological consequences of ATWS events were examined in accordance with the guideline for Class B plants set forth in WASH-1270 which states that the radiological consequences of all ATWS events shall be within the guideline values established by 10CFR100. In making this assessment, consideration was given to the barriers to fission product release established by the fuel cladding, the reactor coolant boundary, and the containment.

The integrity of the fuel cladding was quantitatively assessed in section 4.1 of BAW-10099 throughout the duration of the pressurizer safety valve stuck open ATWS event to assure the adequacy of core cooling as a result of RCS depressurization. For all other ATWS events where the RCS remains pressurized only the short-term quantitative evaluation presented in Section 4.2-4.5 of BAW-10099 was considered necessary since these analyses establish the most severe fuel operating conditions and the fuel damage does not occur. That fuel damage does not occur during long-term heat removal is further supported by the B&W NSS capability to provide system makeup and boration for continued core cooling reactor shutdown while maintaining the RCS in an elevated pressurized state.

For the loss of offsite power ATWS event, B&W also examined the long-term effect of loss of forced RC flow via the reactor coolant pumps. As discussed in response to NRC Question 210.2(A.1) on BAW-10099, the effective natural circulation characteristics of a pressurized B&W NSS have been verified during normal hot shutdown by experimental data obtained from an operating plant. Furthermore, the introduction of cold auxiliary feedwater into the steam generators and greater miswatch in core power and coolant flow during the loss of offsite power ATWS would additionally promote good natural circulation conditions. Thus, an adequate natural circulation would be anticipated during this ATWS event which would provide sufficient long term core cooling and assure that fuel damage does not occur.

B&W has thus established in the course of its evaluation of ATWS events that fuel damage will not occur. This has been shown through a detailed analytical evaluation of transient fuel conditions during the most critical period of concerns and supported in the long term by the existence of unique design features in the B&W NSS design for long term core cooling and reactor shutdown.

The integrity of the reactor coolant boundary was assessed in accordance with the emergency pressure limit and shown in BAW-10099 to be maintained during peak system pressure conditions for the most severe ATWS events. For this reason, only coolant release through the pressurizer safety and relief valves was considered as a result of an ATWS occurrence. Additionally, normal operating experience at six B&W plants using a total of twelve once-through steam generators, i.e. approximately 180,000 steam generator tubes, indicates no primary to secondary tube leakage has occurred. Thus, no tube leakage prior to or as the result of an ATWS occurrence is an appropriate assumption consistent with the philosophy that the ATWS analysis be based on nominal or expected valves for initial conditions.

Containment pressure conditions have been shown for the most limiting ATWS event in section 4.1 of BAW-10099 to be well within normal containment design conditions. As a result, the reactor building leak rate is well within the design leak rate and will provide more time for fission product decay prior to release to the environment in the event of coolant release to containment.

Thus the available barriers to fission product release, as further discussed in responses to NRC Questions 310.1-310.4, limit the fission product release from the fuel and additionally limit the transport path of fission products from the coolant to the containment through the pressurizer safety and relief valves and then via containment leakage to the site boundary. Since the applicant's SAR must demonstrate that release of all fuel gap activity to the containment following the loss of coolant accident results in site boundary doses within 10CFR100, it is concluded that the radiological consequences of ATWS events will have a less severe environmental impact.

' Additional information on components required

'SPONSE: this concern addresses the pressure limit of 3750 psig. BG&E will continue to work with the staff on this matter.

2, #13, #14

Applicant must ....

3PONSE – B-tw makes no response to these statements

5 B&W must calculate failed fuel rods

a) Assumption of fuel failure for all rods with DNBR < 1.3, for dose calculations

b) Assumption of fuel failure for rods with DNBR > 1.3 which experience clad collapse for dose calculation

c) Evaluation of cladding oxidation

d) Assessment of rods which fail by pellet/clad mechanical interaction

e) Fuel swelling effect on coolable core geometry for pressurizer ~~relief~~ safety valve stuck open ATWS.

RESPONSE: B&W has previously remarked on radiological dose consequences, and does not see that the staff fuel failure ~~criteria~~ will alter our conclusions on site doses.

With respect to the clad integrity questions B&W is working with the staff on this matter; we do not foresee any particular problem, but do not anticipate a complete answer before the ACRS hearings on 8, 9 Jan.

6 Extended analyses are required.

SPONSE: The staff requires additional mass and energy data for all ATWS events. B&W discussed this very issue with the staff early in the ATWS program and prior to beginning work on BAW-10099, so that only the limiting case would be required in the topical report. Until the status report, B&W was lead to believe that the pressurizer safety-valve stuck-open event represented satisfies the staff as representing the limiting situation for containment pressure.

In BAW-10099, B&W doubled the mass and energy release of an event continually relieving to the reactor building, and was only able to generate a peak pressure of 25 psi, well within design limits. Therefore, B&W considers that further analyses in this area is not needed.

The staff also requests additional information on

16 continued

long term cooling. See response to #10. B&W further concludes that in going from hot shutdown to cold shutdown, the operator will be in control. Therefore, an orderly shutdown would be achieved and no W-1270 limits would be exceeded.

=17 Additional information for dose calculations is required.

RESPONSE: The staff requires that containment purging be considered. B&W has researched several operating units and finds that, typically:

a) Purging operations are performed infrequently

b) Purging can be and is controlled by a radiation-level detection system separate, independent, and diverse from RPS

Therefore, B&W considers that ~~offsite~~ dose calculations previously presented, assuming no purging, are satisfactory.

In addition, the staff has requested that vessel conditions, containment pressure, and primary and secondary mass releases be provided. B&W has provided its position previously (item 16) and considers further analyses in this area unnecessary.

~~FAILURE~~

FUEL DAMAGE LIMITS ⟹ FOR AN ATWS EVENT

RL

[illegible] limits are characterized by the following

possible [illegible] to reactor [illegible]

sudden increase in coolant system coolant pressure

[illegible]

increase in reactor [illegible]

fuel rods exceeding departure from nucleate boiling with sudden, large increases in cladding temperature

It is the purpose of this memo to propose fuel damage limits which [illegible] the above effects ~~[illegible]~~

and which may be conservatively used for calculating dose levels in the event an ATWS occurs

NUREG 1270 ( ) quotes the following general criteria for fuel behavior under ATWS conditions.

1. ~~(ii)~~ Fuel Pressure Limit.

   The calculated reactor coolant system transient pressure should not exceed a value for which tests and analyses demonstrate that there is no significant safety problem wi h the fuel.

2. ~~b~~. Fuel Thermal and Hydraulic Performance.

   (i) The calculated average enthalpy of the hottest fuel pellet should not result in significant cladding degradation or significant fuel melting.

   (ii) A calculated critical heat flux event should not occur unless the calculated peak cladding temperature can be shown not to result in significant cladding degradation.

To further define and to elaborate on the limits set forth in Wash 1270, the following critical criteria for fuel behavior during an ATWS must be set forward. The reasons for the selection of each of the criteria are given.

- No fuel melting

Fuel melting increases significantly the size of the source term used to evaluate the [illegible] limits. Moreover, it is difficult to [illegible] the amount of fuel melting which can occur before cladding will fail.

- the critical heat flux event mentioned in item 2 (ii) shall be defined to be a DNBR = 1.3 with film boiling assumed for any DNBR < 1.3 for the W-3 correlation, DNBR = 1.32 with film boiling assumed for any DNBR < 1.32 with the B&W-2 correlation and MCHFR = 1.0 (or MCPR) with film boiling assumed for any MCHFR (or MCPR) = 1.0

This criteria will give conservative peak cladding temperatures since it causes film boiling at an earlier time than if the nominal value of DNBR=1.0 is used.

- A rod will be considered to fail if a critical heat flux event occurs on that rod unless the peak cladding temperature is less than 15[illegible] °F and a mechanical analysis of the rod shows that the rod does not (1) swell and rupture or (2) collapse into an axial gap formed by fuel densification. Also, if the rod hot power increases, analysis must show that fuel cladding mechanical interaction will not cause failure of the cladding.

It [illegible] – [illegible] this will be taken

[illegible] thermal-hydraulic analysis

The 1500°F temperature was selected for the peak cladding temperature because this temperature is approximately 80°F below the phase change temperature in zircaloy at which zircaloy loses a significant amount of its strength. This temperature is below that for zircaloy oxidation so that no limit on the amount of zircaloy which may oxidize is required.

The temperature of the cladding only defines part of its condition. The mechanical state of the cladding is equally important. If the cladding [illegible] (fuel) melting is possible. This [illegible] be examined.

Since collapsed cladding is considered to have failed in licensing analyses for the purposes of calculating dose limits, the same requirement should exist for ATWS. This analysis can be done assuming that the cladding is perfectly round and that elastic instability is the only mode of failure leading to collapse. In order to estimate the number of fuel rods subject to collapse, the distribution of [illegible] pressures, cladding thicknesses, cladding temperatures and axial gap sizes in the core should be considered.

If the power increases fuel cladding mechanical interaction is possible. This should be considered as a possible mode of failure by an appropriate method of analysis

These criteria are formulated with the assumption that the reactor will not be restarted and that the fuel assemblies will never again be used. If it is desired to reuse any of the fuel, careful consideration should be given to the use of fuel which has experienced cladding temperatures above prior normal operating temperatures.

Using the above criteria there is no need for a criterion involving the average enthalpy of the hottest fuel pellet.

[illegible] criteria listed above should result in a conservative estimate of the number of failed [illegible] rods for use in dose calculations.

5.2.2 (c) What is the allowable back pressure for the safety valves. Provide the method used, including experimental verification, in determining ~~a~~ back pressure limit. If this limit is exceeded, what would be the effect on safety valve relief capability.

5.2.2.3 (BAW-10043)

~~① Since BAW-10043 is referenced as a basis of over~~

① Show that all the assumptions and initial conditions used ~~in the analysis~~ are identical for both the Three Mile Island Analysis and the BAW-10043 analysis

② BAW-10043 does not provide the basic plant parameters such as plant geometry and power level. Further BAW-10043 does not provide the set points for both the primary and the secondary safety valves. ~~set points~~. Until this information is provided overpressure protection can not be evaluated

③ BAW-10043 does not address the severity of a complete loss of feedwater and feedwater line rupture on the overpressure protection capacity provided.

Provide analyses to substantiate adequacy of safety valve discharge capacities for [illegible] a complete loss of feedwater transient and feedwater line rupture accident. Show that the single failures considered in the analyses are the most limiting ones.

(4) In BAW-10043 analysis no credit was taken for pressurizer spray although high pressurizer pressure signal was used to scram the reactor. Provide analysis where no credit is taken for sprays and therefore scram is delayed.

(5) Show the pressurizer does not go solid for any overpressure transients. Otherwise provide bases for water discharge rates through the safety valves.

210.1 The analyses presented in BAW-10099 are unacceptable because they are based on operation of the power operated relief valve and a value of the moderator coefficient which is less negative than the calculated full-power value for 95% of plant life. An acceptable analysis can be performed assuming a) No relief valve action (in compliance [illegible]), and b) a moderator coefficient which is less negative than the calculated full-power value 99% of the time the reactor is critical and at operating pressure.

210.2

210.1 The analyses presented in WCAP-8330 are unacceptable because they are based on actuation of all power-operated relief valves and a moderator coefficient which is more negative than the calculated full-power value only 95 percent of the time the reactor is critical. Therefore, the analyses do not demonstrate that the limits specified in WASH-1270 are not exceeded with a probability of [illegible] per year or less. An acceptable analysis would be performed assuming a) only half of the power-operated relief valves open and b) a moderator coefficient which is less negative than the calculated full-power value 99 percent of the time the reactor is critical and at operating pressure.

210.2 Section 3.5 of BAW-10099 does not ~~[illegible]~~ adequately demonstrate the capability to bring the plant to a cold shutdown condition. Use the following criteria for determining the most severe transient for shutdown capability:

a) Ability to reduce primary pressure
b) Ability to maintain core level
c) Ability to maintain long-term heat removal capability via steam generators
d) Ability to maintain containment pressure within containment design pressure

The analyses must analyze the following cases:

a) An average case
b) The relief valve fails to open ~~[illegible]~~
c) One safety valve fails to reclose.

For these analyses provide the following parameters as a function of time until the plant is at a cold shutdown condition (200°F and atmospheric pressure) or until the time when the analysis becomes plant specific limited:

a) Reactor power
b) Reactor pressure, temperature, and inventory
c) Auxiliary feedwater flow rate
d) Safety/relief valve flow rate
e) ECCS and makeup flow
f) [illegible] flow rate
g) Flow rate and [illegible] of any other source of water used [illegible]
h) All operator actions

i) [illegible]

j) [illegible]

210.3 Provide ATWS analyses results for each event until the system responses have stabilized.

210.4 Provide the signals, their set points and delay times used to actuate systems which mitigate the consequences of ATWS. Supplement this information with diagrams illustrating the implementation of instrumentation. Discuss the reliability of these signals and their diversity to the Reactor Protection system. For example demonstrate the diversity to RPS of ESFAS used to initiate turbine trip, aux feedwater and etc.

210.5 The analysis takes credit for the full capacity of the auxiliary feedwater system at 40 seconds. In order for this to be acceptable provide the following

a) Justification for assuming any auxiliary flow prior to 40 seconds in the transient. Provide total feedwater flow as a function of time

b) Justification for availability of steam driven auxiliary feed pump. Provide steam flow and steam pressure as a function of time.

c) assurances that [illegible] [illegible] auxiliary feed [illegible] [illegible] to allow [illegible]

d) Discuss the type of auxiliary feed

pumps used in Clifford plants and the effect on their availability for a loss of offsite power ATWS event.

Stopped

210.6 Provide sequence of events similar to table 4-2 for each ATWS event.

210.7 Your analyses show that the loss of two reactor coolant pumps results in higher peak clad temperature (PCT) than for a loss of offsite power ATWS event. Explain these differences " " " ... Provide PCT for these transients if film boiling is assumed to start at DNBR of 1.3.

210.8 BAW 10016 shows the loss of all coolant pumps to be a more frequent occurrence than the loss of two coolant pumps. Therefore provide analyses for loss of all coolant pumps unless it can be demonstrated that this transient is less severe than another for which analyses have been provided. In the analysis assume film boiling begins at DNBR of 1.3

210.9 For loss of offsite power provide the pressure response of the pressurizer relief tank and the piping between the pressurizer and the pressurizer relief tank. If design pressures are exceeded, discuss the possibility of missile generation and the effect the missiles may have on plant safety.

210.10 Provide results of your analysis of Standard Problem #1 of Appendix C of ANSI-N661, draft 3, "Standard for Evaluation of ATWS on PWR Plants", October 1974. Provide a schedule for submittal of the analyses of standard Problems 2, 3 and 4 of Appendix C of ANSI-N661, draft 3.

ENCLOSURE

223.1 The analysis presented in Section 4.3, on loss of off-site power, assumes full coastdown flow in the reactors coolant and main feedwater systems. However, in order to achieve full coastdown capability, a rapid isolation of the reactor coolant and feedwater pumps from the decaying power system is required.

Provide a detailed description of the instrumentation available to isolate the pumps in event of a rapidly decaying power system frequency. Supplement the information with sufficient number of diagrams that illustrate the implementation of such instrumentation.

223.2 From the information provided in Section 4.0, for various anticipated transients, we are unable to determine which specific systems are assumed to function to mitigate the consequences of a particular transient.

We require information similar to that provided in table 4-2 for the stuck-open Pressurizer Safety Valve Transient. In addition, we require a description of the initiating instrumentation for the above systems, and the functional and component diversity provided for that instrumentation. Also, supplement the above information with sufficient number of diagrams that illustrate the implementation of such instrumentation.

Table 210.10-1

Case 1-A, Infinite Pressurizer

Time (sec)	PZR Pressure (psia)	System Pressure* (psia)	Reactor Inlet Temp. (F)	Steam Gen. Inlet Temp. (F)	Surge to PZR (lbm/sec)	PZR Liq. Vol. (ft^3)	PZR Liq. Level (ft^3)
0.0	2250.0	2288.4	550.1	616.4	0.0	6.6 +22	18.91
5.0			552.3	616.4	650.8		
10.0			557.3	620.4	386.7		
15.0			554.9	622.0	-815.6		
20.0			549.6	617.3	-334.2		
25.0			548.2	616.3	-510.4		
30.0			542.7	612.9	-396.1		
35.0			541.3	610.5	-424.0		
40.0			536.1	608.0	-422.9		
45.0			534.3	604.9	-384.9		
50.0			529.7	602.6	-433.0		
55.0			527.3	599.2	-363.4		
60.0			523.1	597.1	-425.9		
65.0			526.2	593.6	1409.4		
70.0			539.1	603.3	685.1		
75.0			540.1	606.4	941.9		
80.0			552.7	612.4	855.7		
85.0			554.4	618.0	867.7		
90.0			565.2	622.7	975.7		
95.0			568.3	628.6	898.2		
100.0			577.6	633.0	1036.9		
105.0			581.8	638.6	972.7		
110.0			589.8	642.9	1101.4		
115.0			594.9	648.1	1075.5		
120.0**			602.0	652.4	1203.8		

*System pressure is pressure at core midplane.

**Run terminated when coolant reached saturation.

PRELIMINARY

Table 210.10-2

Case 1-B, No Pressure Control

Time (sec)	PZR Pressure (psia)	System Pressure* (psia)	Reactor Inlet Temp. (F)	Steam Gen. Inlet Temp. (F)	Surge to PRZ (lbm/sec)	PZR Liq. Vol. (ft^3)	PZR Liq. Level (ft^3)
0.0	2250.0	2288.8	550.1	616.4	0.0	750.0	20.7
5.0	2353.[illegible]	2401.3	552.7	617.2	433.8	773.1	21.3
10.0	2534.5	2576.9	558.6	622.4	262.5	815.4	22.4
15.0	2413.8	2441.5	555.5	623.2	-495.3	788.5	21.7
20.0	2219.9	2255.6	549.4	617.1	-260.8	742.3	20.5
25.0	2172.0	2201.0	547.8	615.7	-436.4	700.0	19.4
30.0	2118.9	2149.8	542.1	612.0	-373.7	619.2	17.3
35.0	2084.3	2115.0	540.6	609.3	-374.7	569.1	16.0
40.0	2037.1	2066.3	535.2	606.5	-399.0	503.7	14.3
45.0	2000.4	2031.3	533.3	603.2	-347.1	453.7	13.0
50.0	1956.6	1984.8	528.5	600.8	-406.7	396.0	11.5
55.0	1918.0	1948.6	526.0	597.4	-336.0	346.0	[illegible]
60.0	1876.3	1904.4	521.8	594.8	-393.7	295.9	8.9
70.0	2236.7	2290.4	539.1	603.2	572.7	449.9	12.9
80.0	2692.4	2750.6	554.6	615.3	631.9	588.4	16.5
90.0	3266.1	3323.3	569.7	630.1	606.4	723.1	20.0
100.0	3973.3	4026.4	585.6	646.1	528.4	838.5	23.0
110.0	4826.8	4875.1	602.4	663.3	413.8	938.6	25.6
120.0	5813.6	5860.5	620.2	681.5	379.9	1019.4	27.7
130.0	6976.5	7021.0	639.3	702.7	289.1	1084.8	29.4
140.0	8156.4	8199.4	658.8	722.5	234.6	1131.0	30.6
150.0	9492.9	9535.2	677.9	743.4	190.0	1169.4	31.6
160.0	10829.1	10872.6	700.1	764.4	217.3	1204.1	32.5

*System pressure is pressure at core midplane.

PRELIMINARY

Table 210.10-3

Case 1-C, Relief Valve, No Spray

Time (sec)	PZR Pressure (psia)	System Pressure* (psia)	Reactor Inlet Temp. (F)	Steam Generator Inlet Temp. (F)	Surge to PZR (lbm/sec)	Relief Valve Flow (lbm/sec)	PZR Liquid Vol. (ft^3)	PZR Liquid Level (ft^3)
0.0	2250.0	2288.8	550.1	616.4	0.0	0.0	750.0	20.7
5.0	2353.3	2401.3	552.7	617.2	433.8	0.0	773.1	21.3
10.0	2473.2	2520.1	558.4	622.0	379.9	56.2	823.1	22.6
15.0	2329.2	2356.5	555.1	622.6	-504.1	0.0	800.0	22.0
20.0	2176.4	2210.2	549.2	616.8	-317.4	0.0	742.3	20.5
25.0	2144.3	2173.2	547.7	615.5	-438.5	0.0	696.1	19.3
30.0	2092.6	2123.6	542.0	611.8	-373.8	0.0	615.3	17.2
35.0	2058.6	2089.2	540.5	609.2	-377.7	0.0	569.1	16.0
40.0	2012.5	2041.7	535.1	606.4	-399.4	0.0	503.7	14.3
45.0	1976.4	2007.3	533.2	603.1	-349.0	0.0	453.7	13.0
50.0	1933.2	1961.3	528.4	600.6	-407.2	0.0	396.0	11.5
55.0	1895.5	1926.1	525.2	597.2	-338.2	0.0	346.0	10.2
60.0	1853.9	1882.0	521.7	594.7	-393.7	0.0	292.1	8.8
70.0	2212.4	2266.2	539.0	603.1	574.0	0.0	446.0	12.8
80.0	2539.2	2615.9	554.0	614.4	858.4	111.1	603.8	16.9
90.0	2578.3	2668.0	566.8	625.3	977.0	146.0	803.9	22.1
100.0	2589.6	2685.1	579.3	636.2	1018.3	156.0	1027.1	27.9
110.0	2607.3	2708.0	591.9	646.7	1051.7	166.2	1273.3	34.3
120.0	2824.4	2878.1	605.4	658.5	448.1	389.0	1500.0	40.8
130.0	3709.8	3763.7	623.5	676.6	451.4	404.2	1500.0	40.8
140.0	4634.8	4688.8	641.8	696.8	451.7	414.9	1500.0	40.8
150.0	5476.6	5530.5	659.8	715.4	444.7	423.7	1500.0	40.8
156.0	5976.5	6030.6	670.1	726.4	449.3	428.9	1500.0	40.8

*System pressure is pressure at core midplane.

PRELIMINARY

Table 210.10-4

Case 1-D, Relief Valve and Spray

Time (sec)	PZR Pressure (psia)	System Pressure* (psia)	Reactor Inlet Temp. (F)	Steam Generator Inlet Temp. (F)	Surge to PZR (lbm/sec)	Relief Valve Flow (lbm/sec)	PZR Liquid Vol. (ft^3)	PZR Liquid Level (ft^3)
0.0	2250.0	2288.8	550.1	616.4	0.0	0.0	750.0	20.7
5.0	2351.2	2397.3	552.7	617.2	388.7	0.0	773.1	21.3
10.0	2468.7	2513.8	558.3	622.0	330.5	52.9	823.1	22.6
15.0	2319.2	2344.0	555.1	622.5	-550.5	0.0	800.0	22.0
20.0	2164.0	2197.7	549.2	616.7	-318.8	0.0	742.3	20.5
25.0	2132.4	2161.2	547.7	615.4	-440.6	0.0	696.1	19.3
30.0	2081.2	2112.1	542.0	611.8	-374.1	0.0	619.2	17.3
35.0	2047.9	2078.4	540.5	609.1	-378.1	0.0	569.1	16.0
40.0	2001.9	2031.1	535.0	606.3	-399.7	0.0	503.7	14.3
45.0	1966.3	1997.2	533.2	603.0	-349.4	0.0	453.7	13.0
50.0	1923.2	1951.4	528.4	600.6	-407.6	0.0	396.0	11.5
55.0	1886.1	1916.6	525.9	597.2	-338.9	0.0.0	346.0	10.2
60.0	1844.9	1873.0	521.6	594.6	-395.3	0.0	295.9	8.9
70.0	2202.7	2256.5	539.0	603.0	573.3	0.0	449.8	12.9
80.0	2533.2	2605.2	554.0	614.3	803.8	105.8	603.8	16.9
90.0	2575.4	2660.0	566.8	625.3	926.1	143.4	803.9	22.1
100.0	2587.6	2677.9	579.3	636.1	967.8	154.2	1023.2	27.8
110.0	2606.4	2701.8	591.9	646.6	1001.6	166.1	1265.6	34.1
120.0	2809.3	2861.2	605.3	658.4	405.1	388.7	1500.0	40.8
130.0	3690.8	3742.7	623.3	676.5	404.0	404.0	1500.0	40.8
140.0	4612.0	4663.9	641.7	696.6	403.0	414.6	1500.0	40.8
150.0	5453.6	5505.5	659.7	715.2	396.4	423.5	1500.0	40.8
156.5	5997.8	6049.8	670.9	727.1	402.4	429.2	1500.0	40.8

*System pressure is pressure at core midplane.

PRELIMINARY

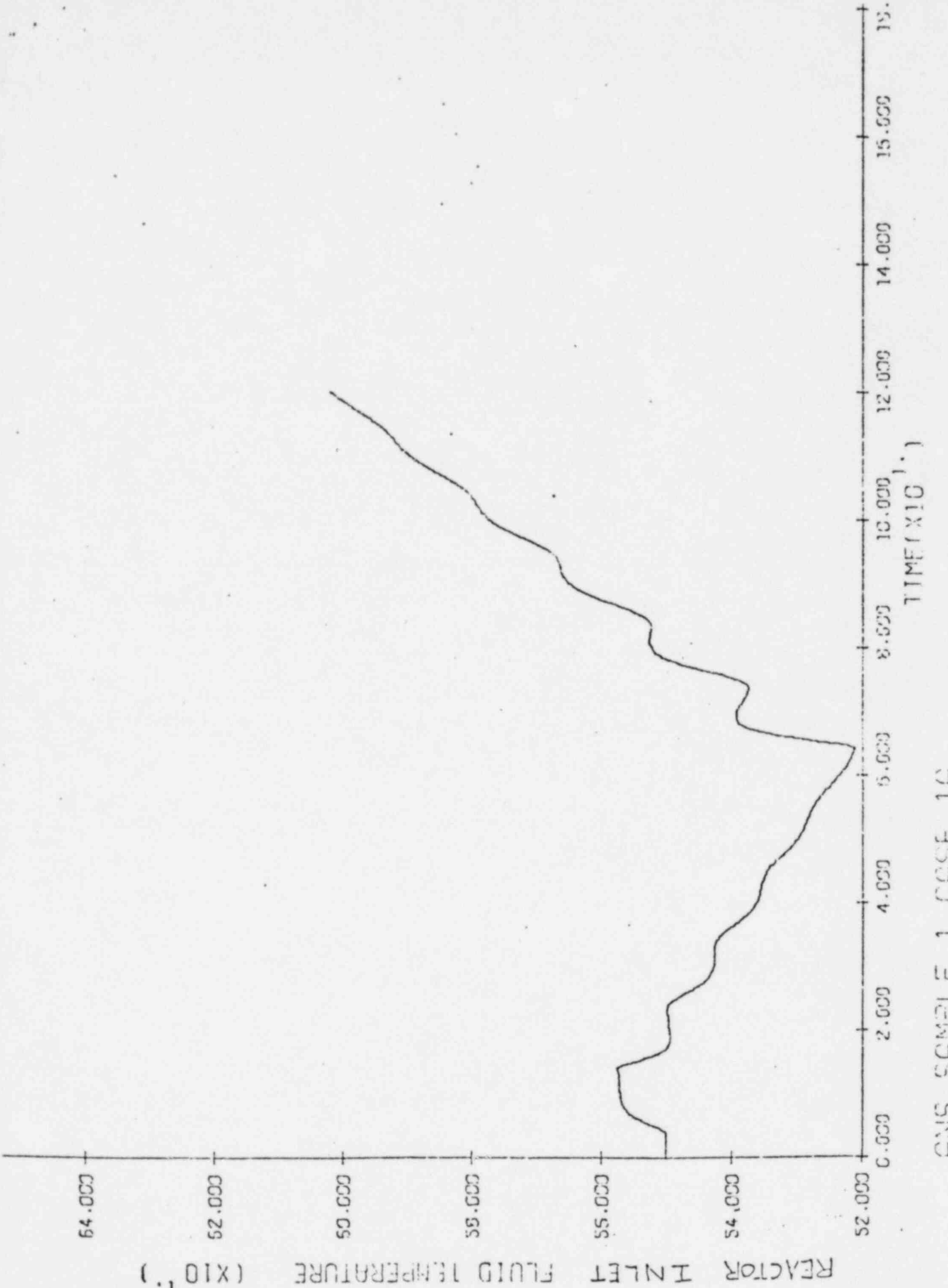
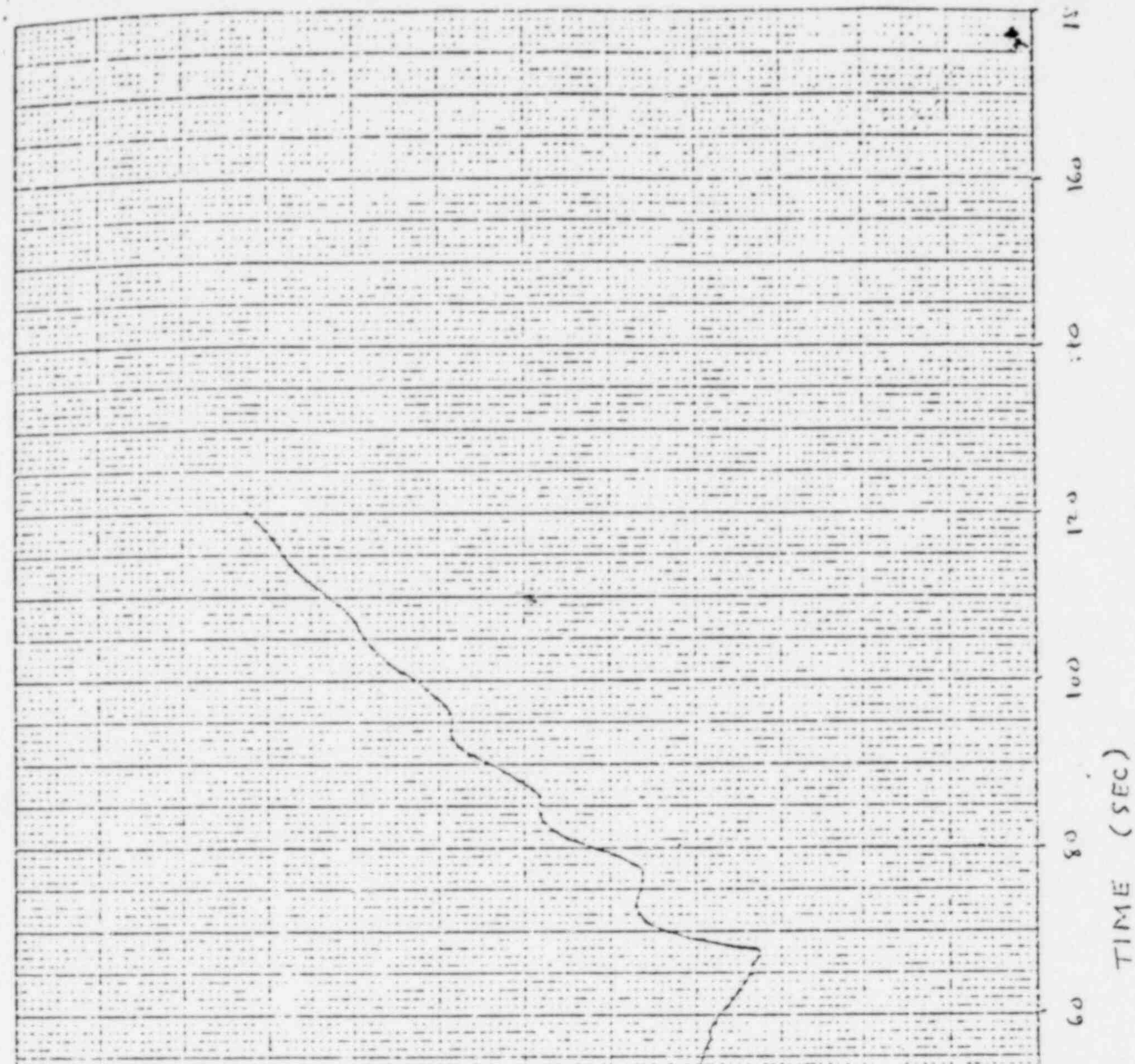


Fig. 210.10-1

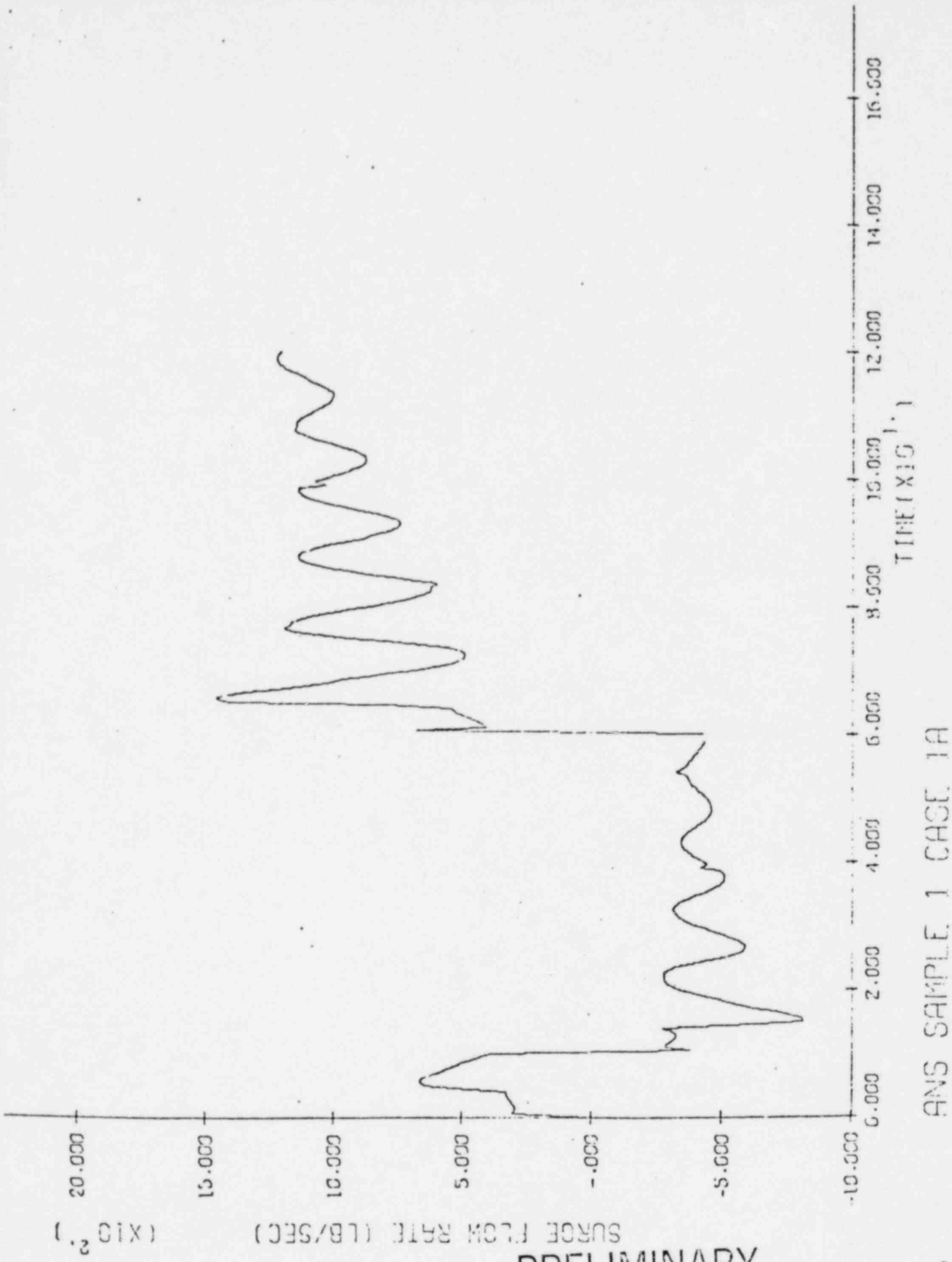
PRELIMINARY



ANS SAMPLE 1 CASE 1A

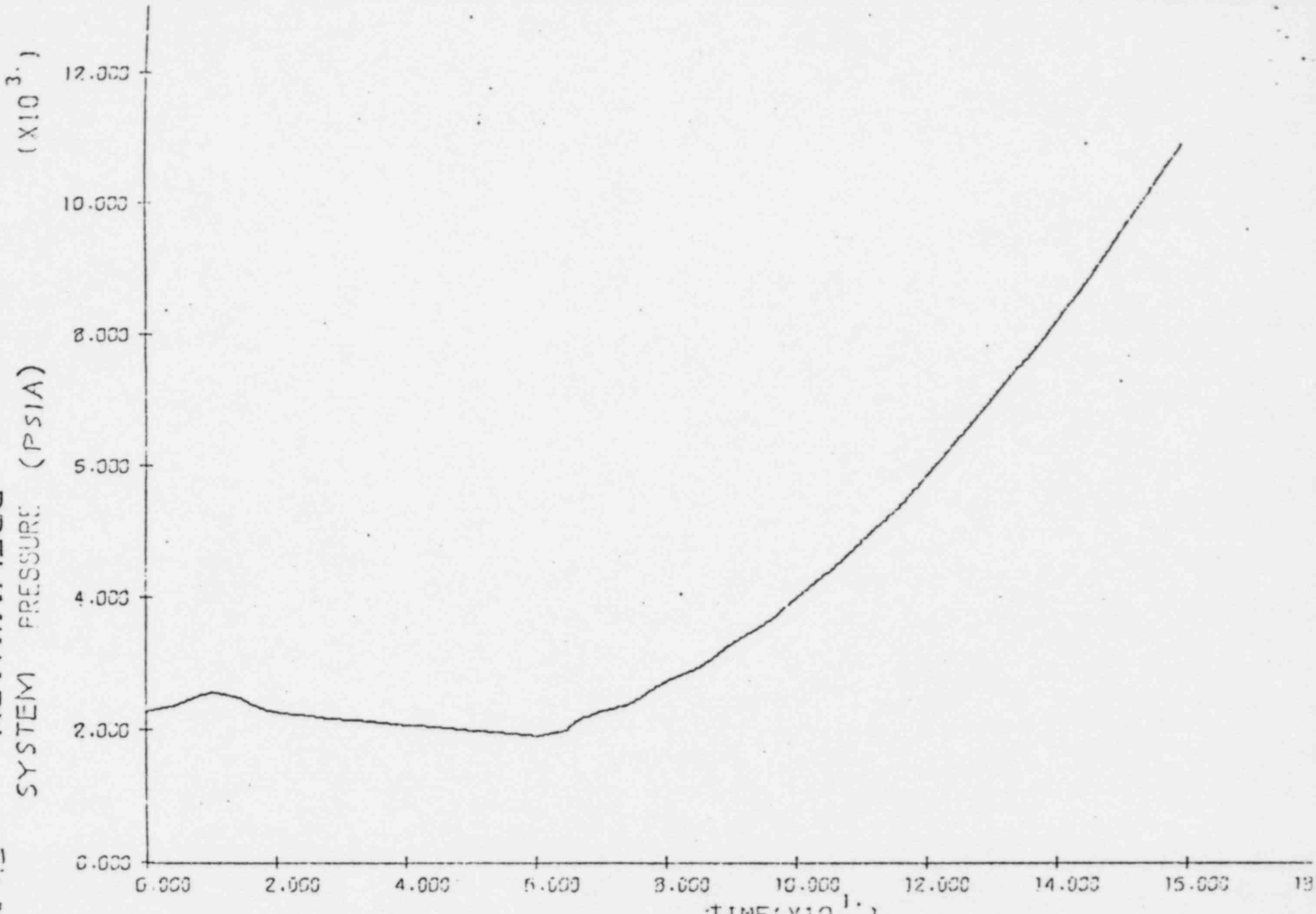
PRELIMINARY

Fig. 210.10-2



PRELIMINARY

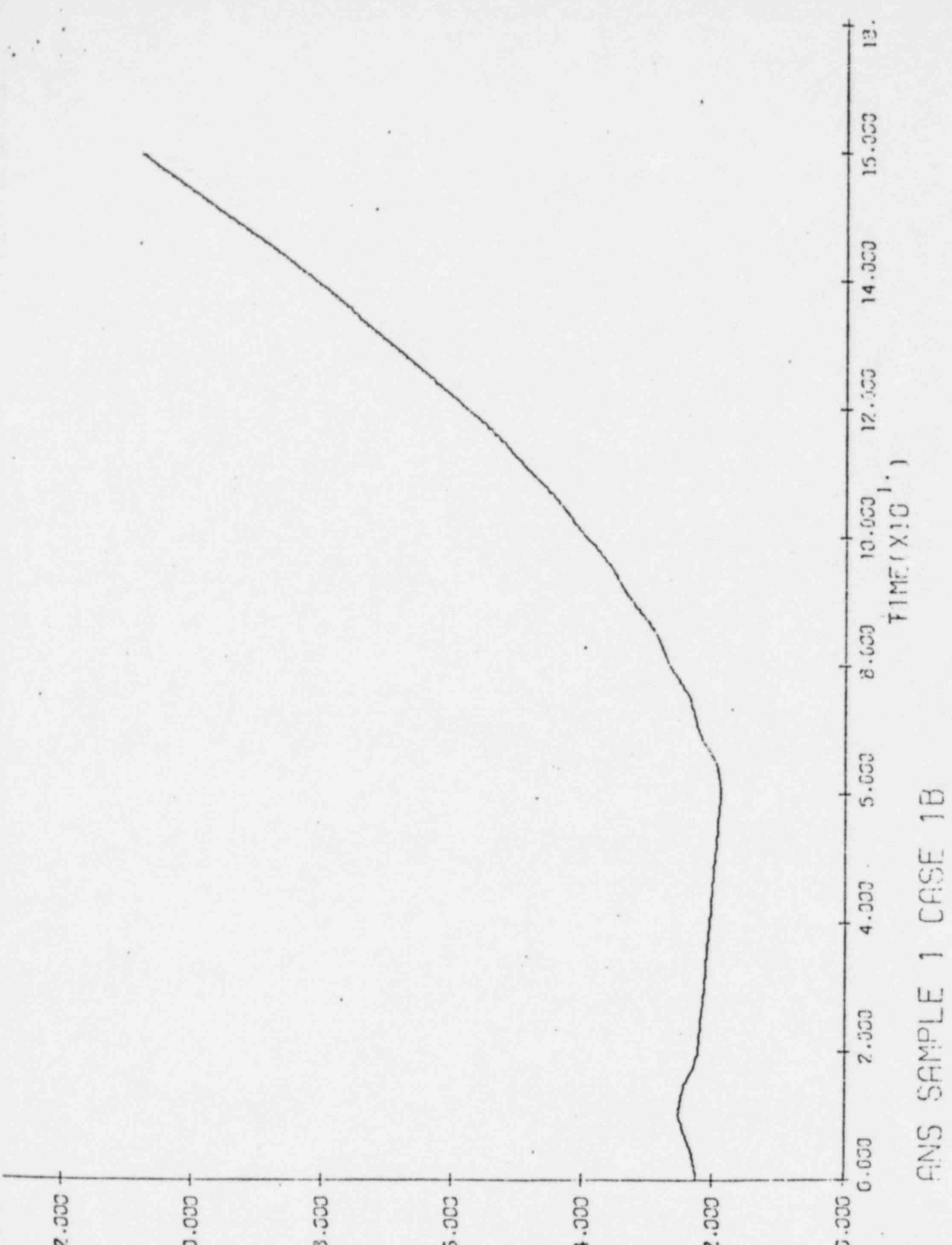
Fig. 210.10-3



ANS SAMPLE 1 CASE 1B

PRELIMINARY

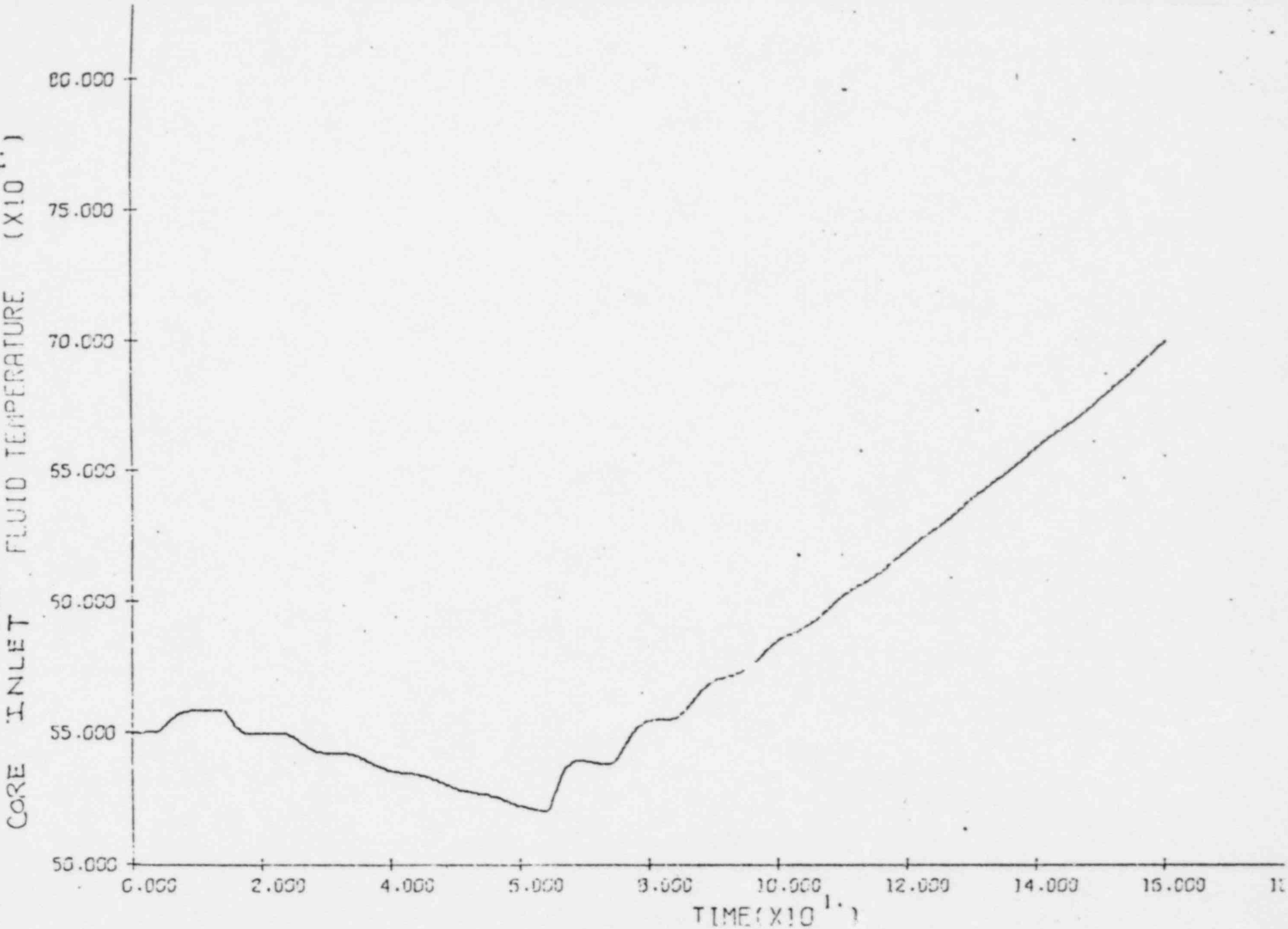
Fig. 210.10-4



PRELIMINARY

Fig. 210.10-5

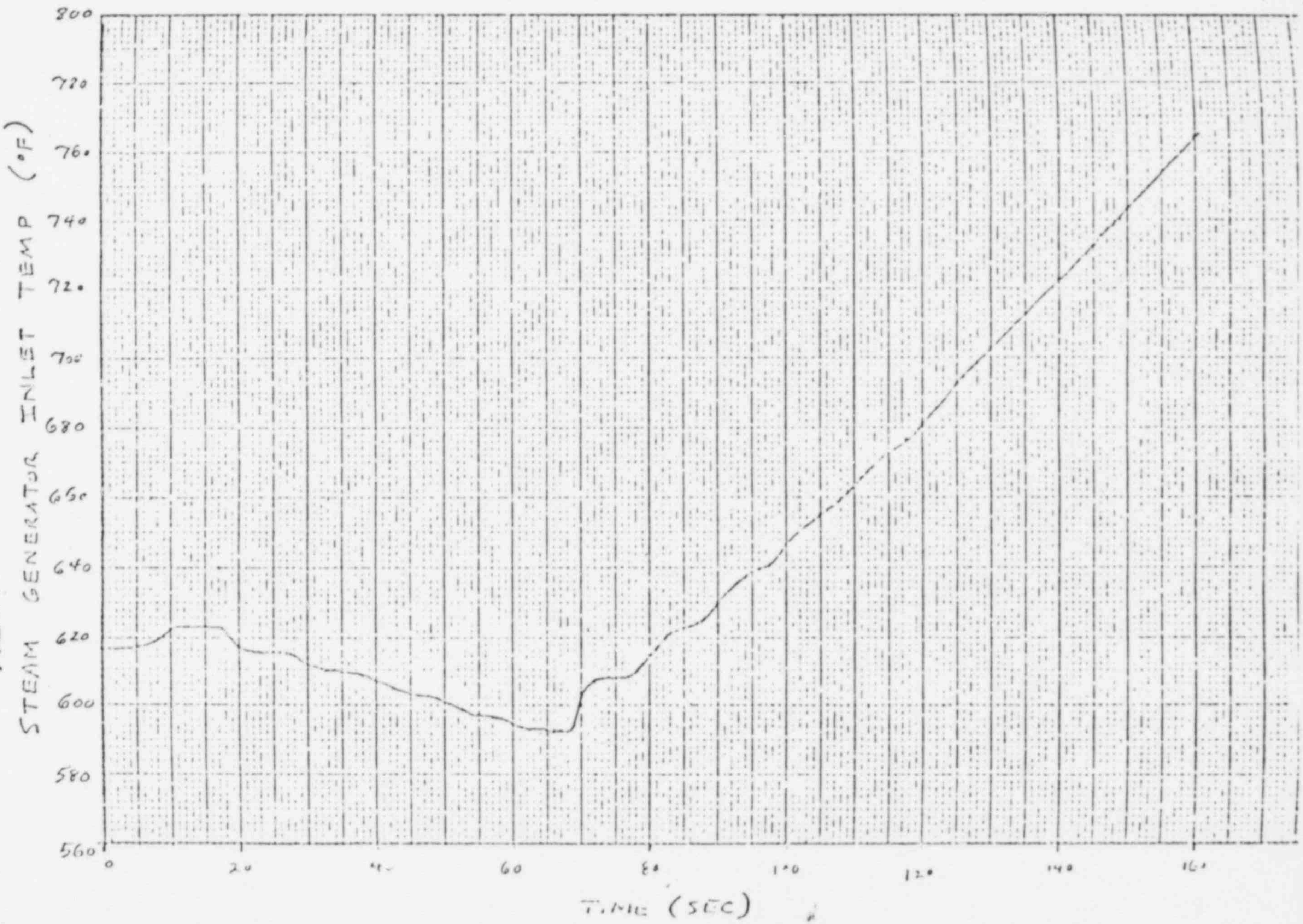
PRELIMINARY



ANS SAMPLE 1 CASE 1B

NODE 1 - PIN 1

Fig. 210.10-6



ANS SAMPLE 1 CASE 1B

PRELIMINARY

FIG. 210.10-7

PRELIMINARY

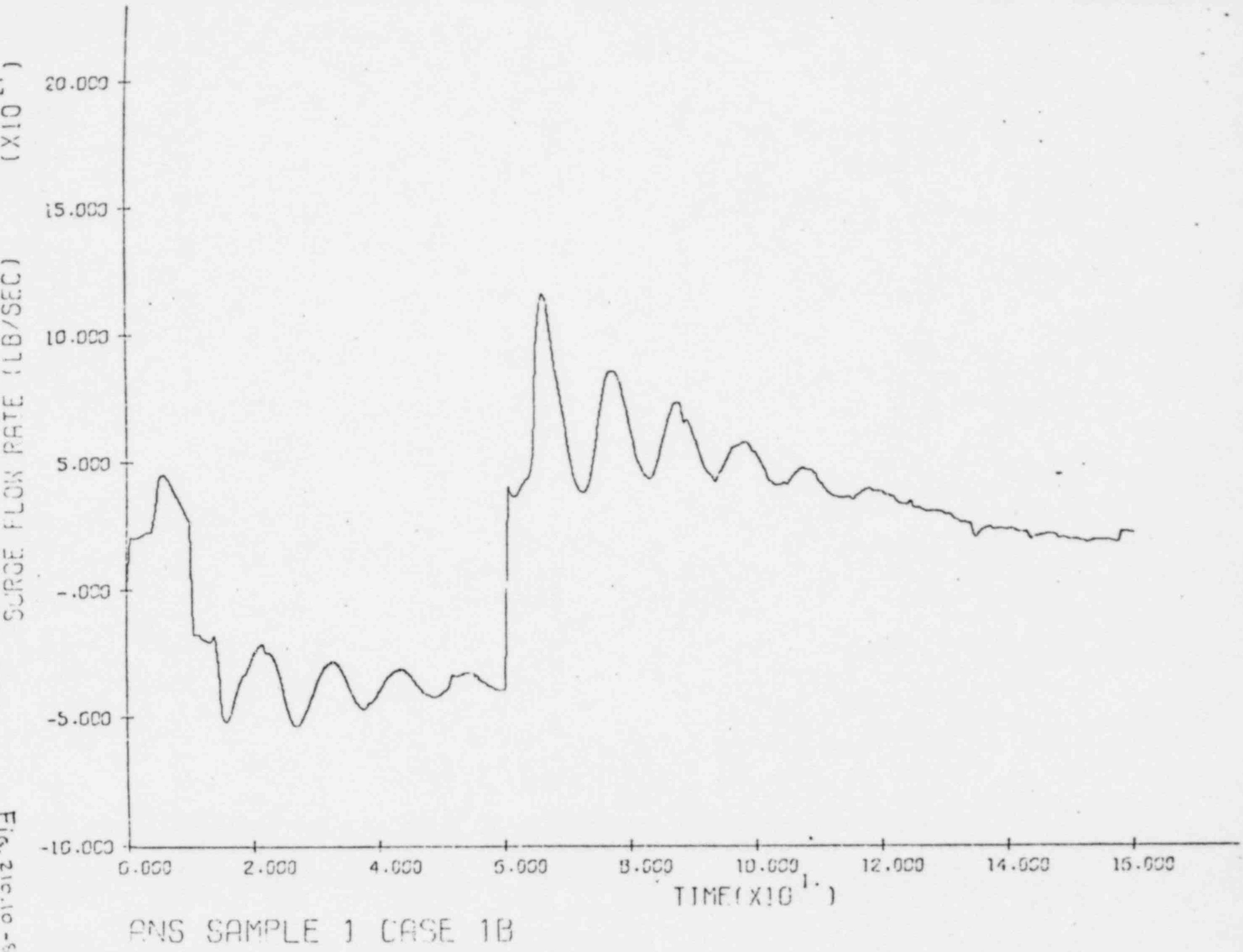
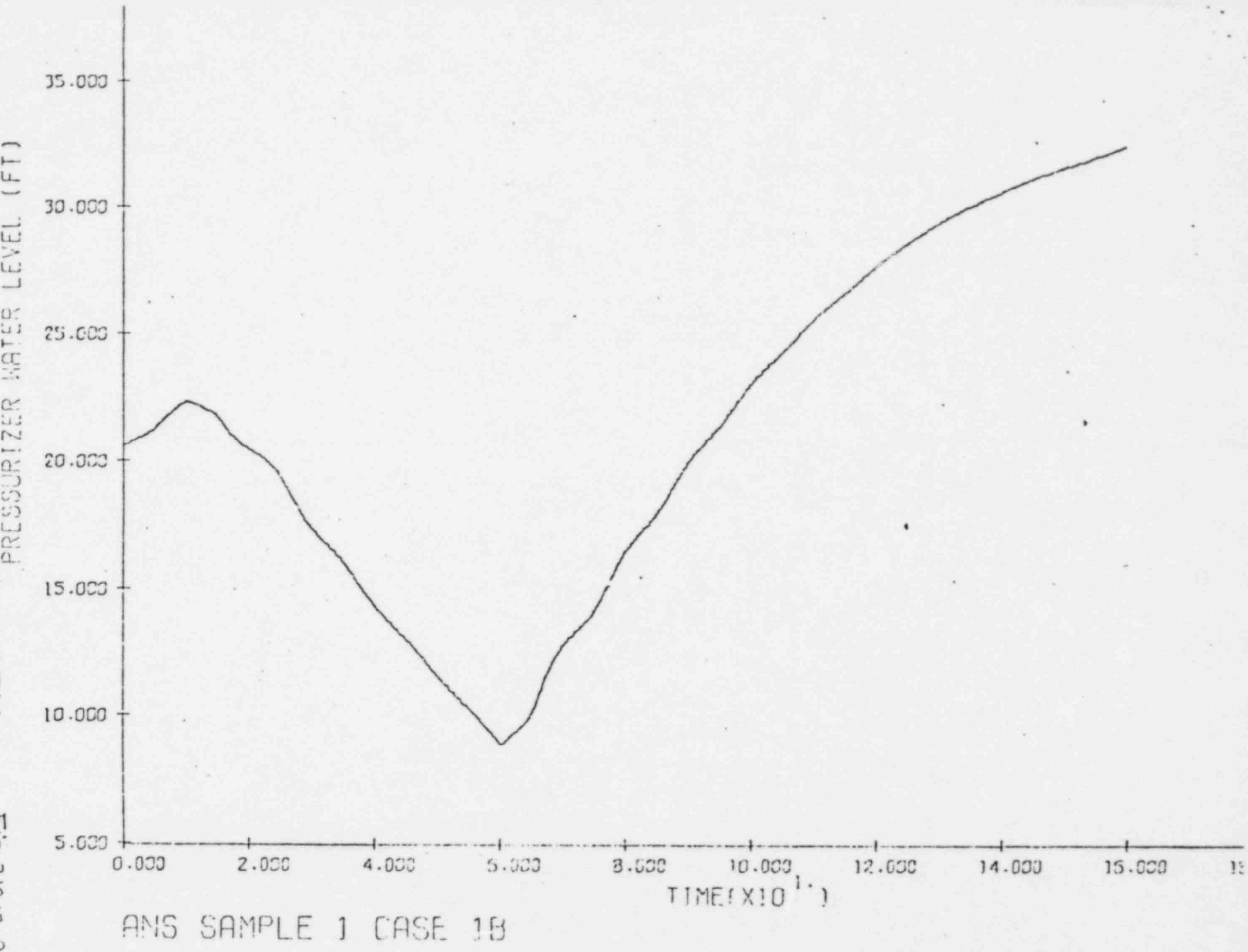
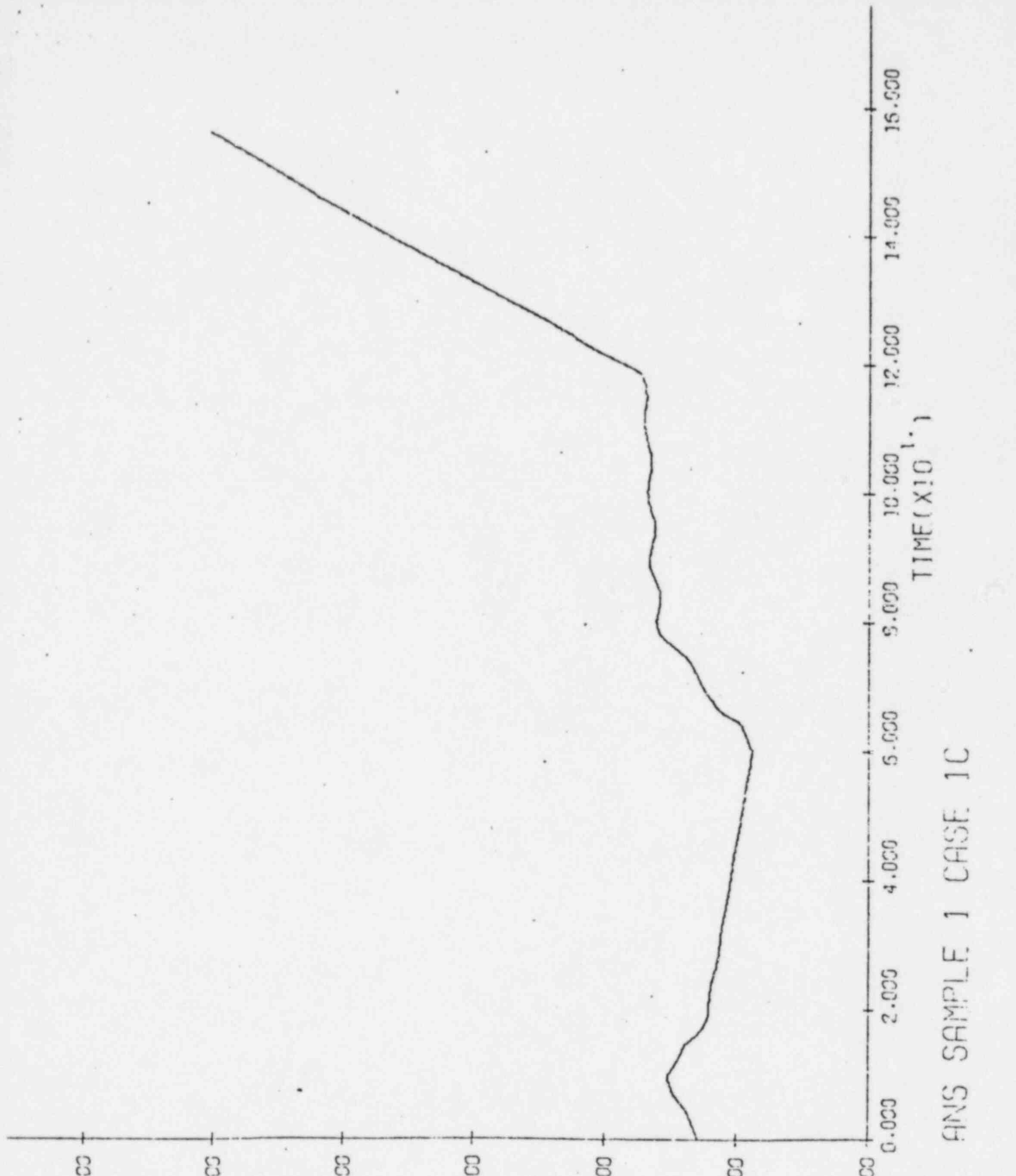


Fig. 210.10-6



PRELIMINARY

Fig. 210.10-9



PRELIMINARY

Fig. 210.10-10

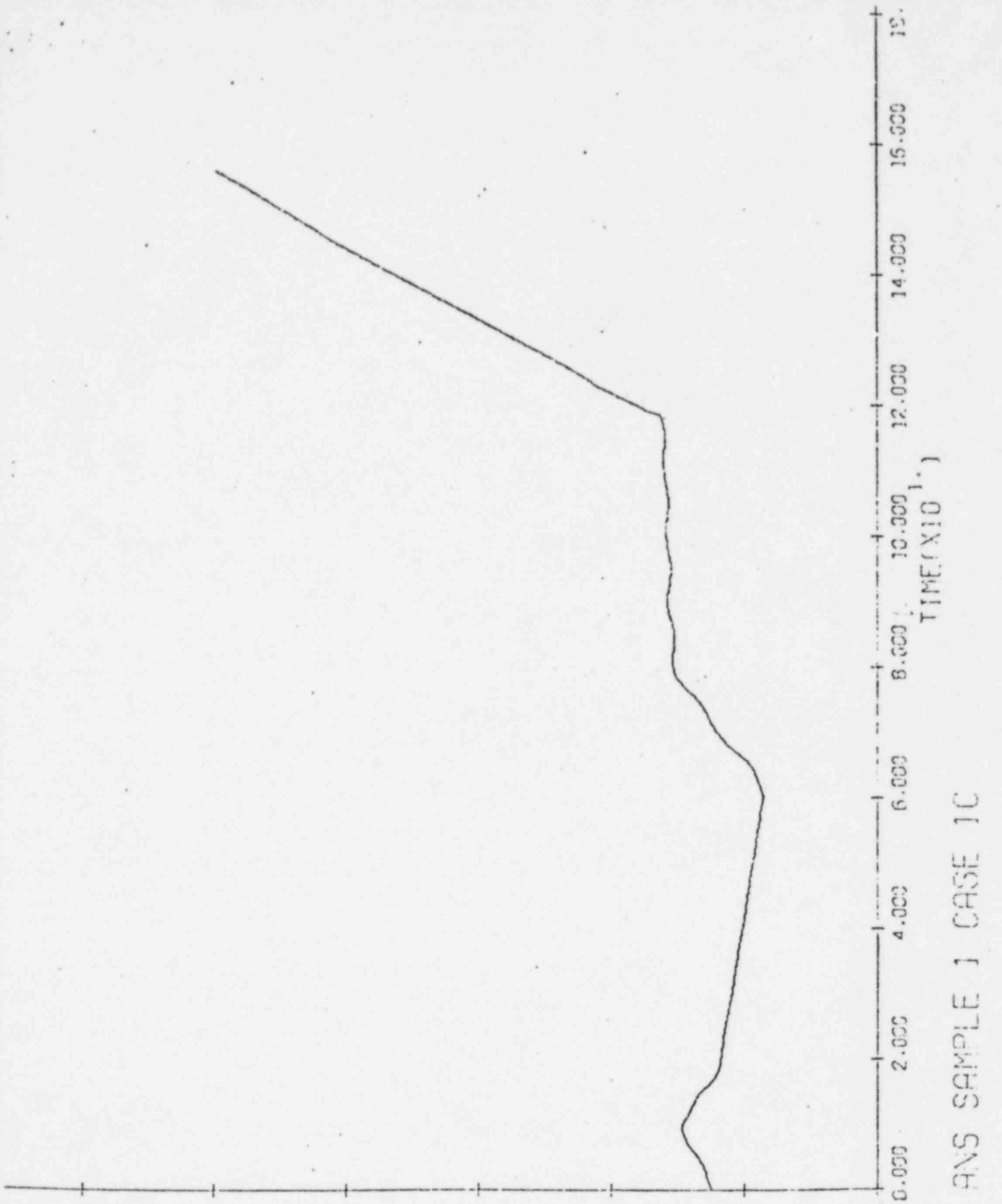
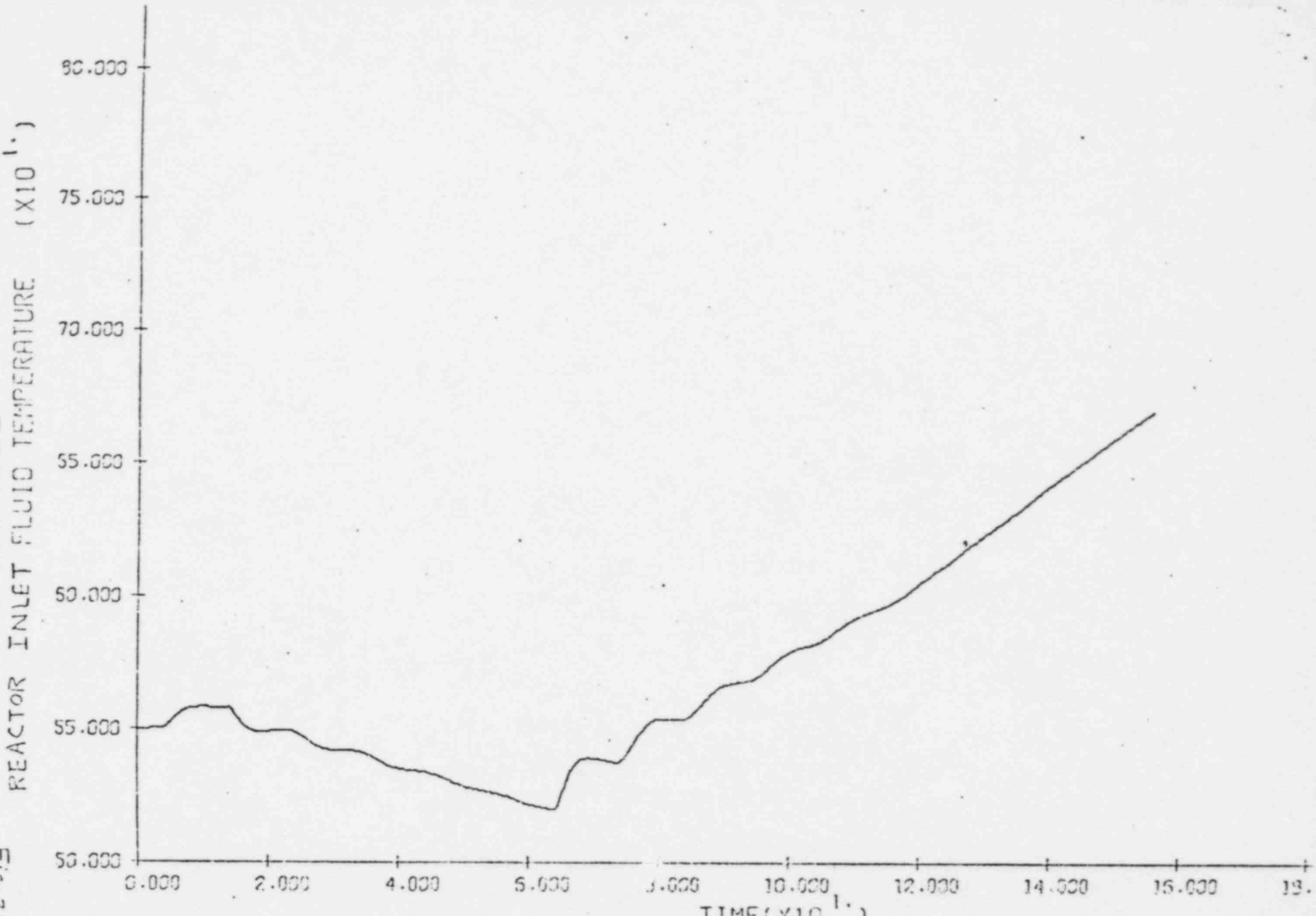


Fig. 210.10-11

PRELIMINARY

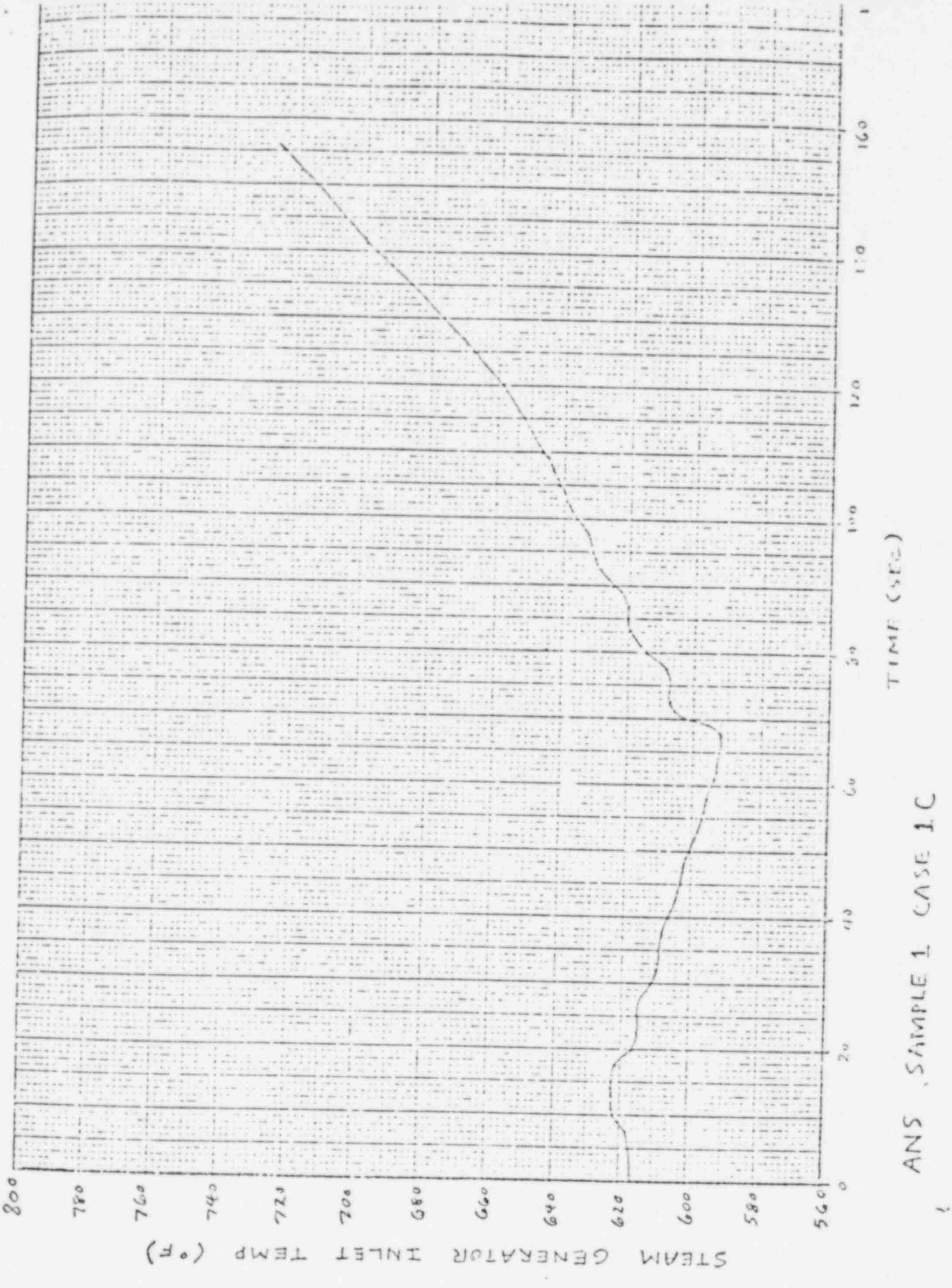


ANS SAMPLE 1 CASE 1C

NODE 1 - PIN 1

PRELIMINARY

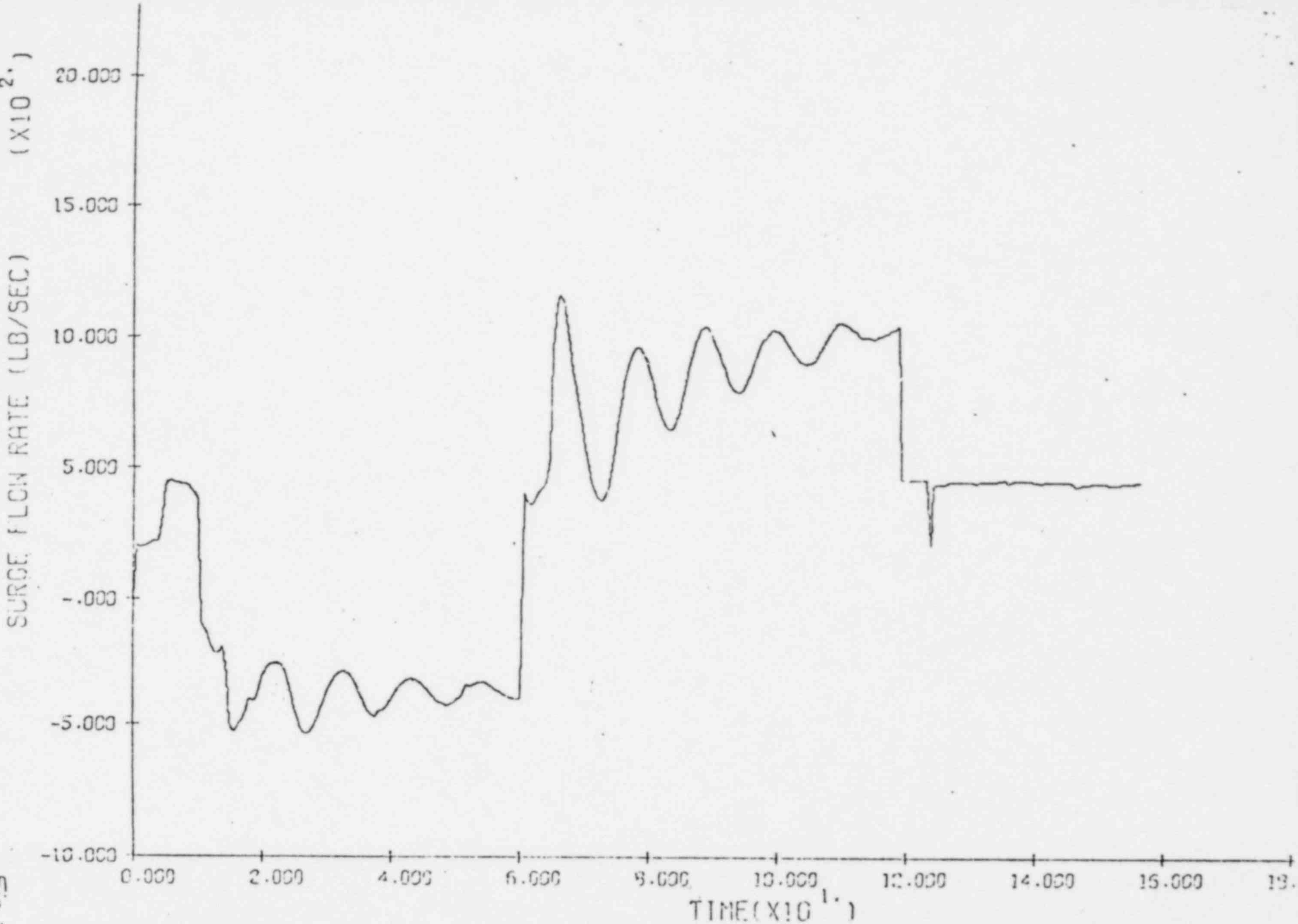
Fig. 210.10-12



PRELIMINARY

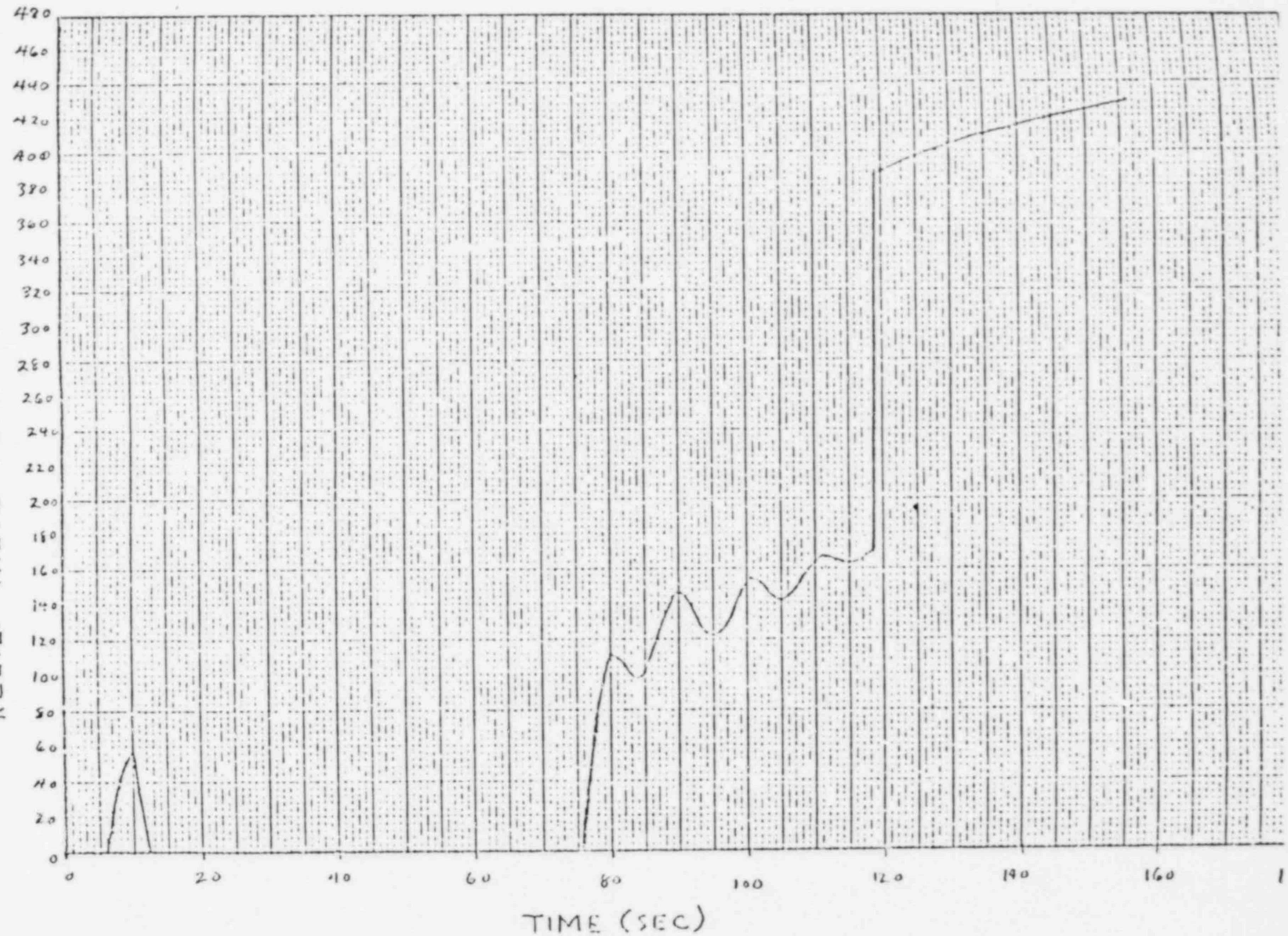
FIG. 210.10-13

PRELIMINARY



ANS SAMPLE 1 CASE 1C

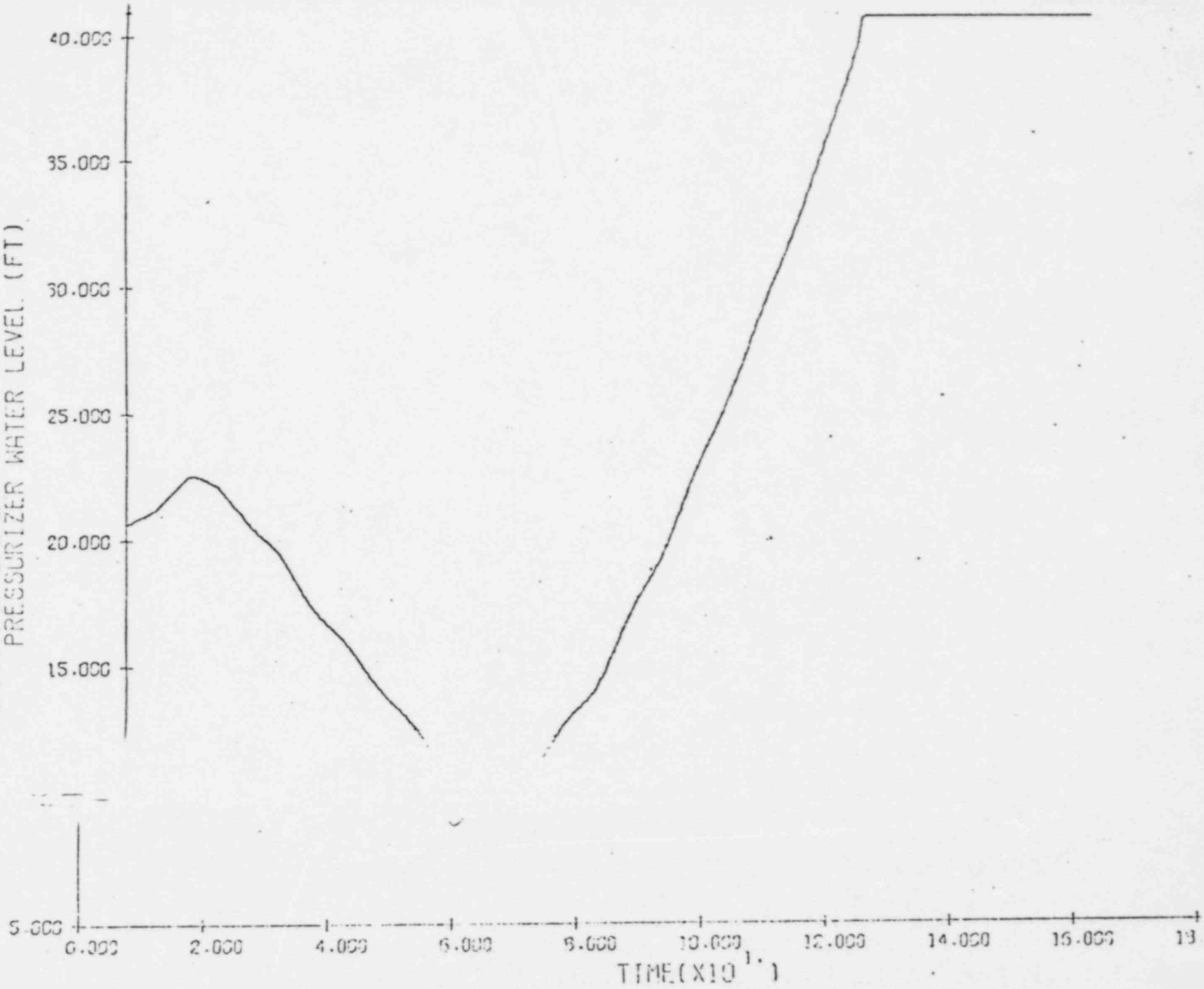
Fig. 210.10-14



ANS SAMPLE 1 CASE 1C

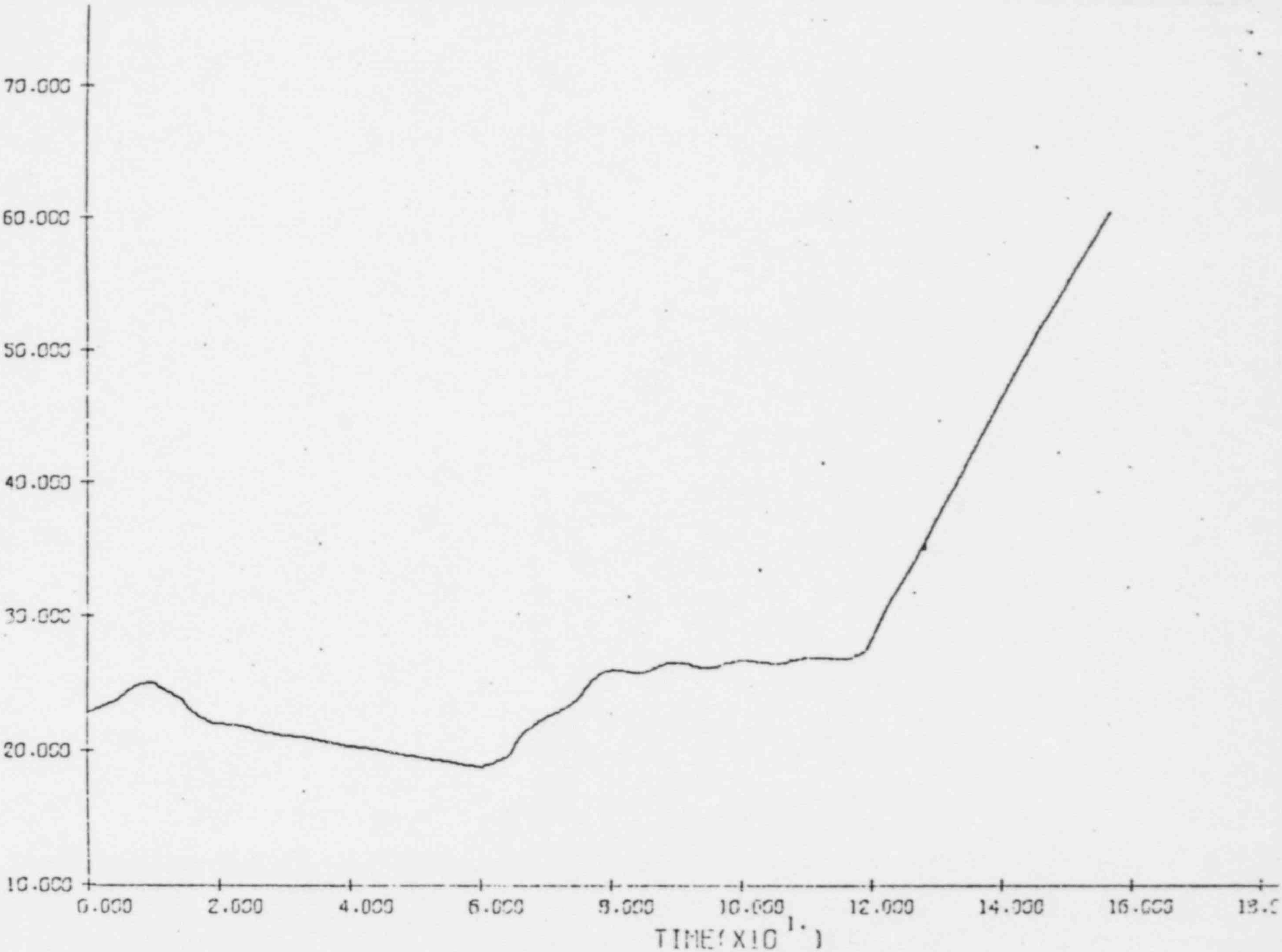
PRELIMINARY

Fig 210.10-15



PRELIMINARY

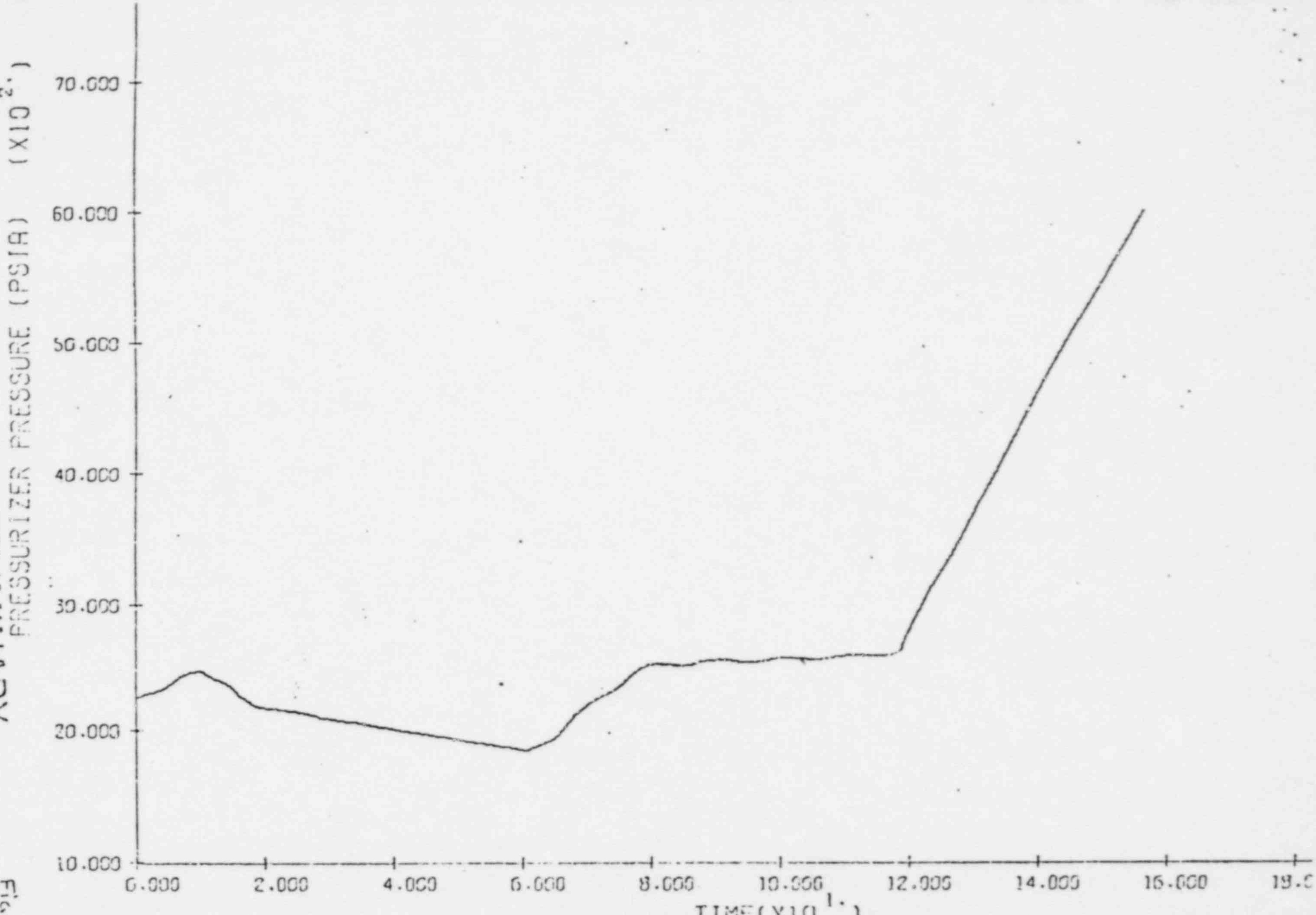
Fig. 210.10-16



ANS SAMPLE 1 CASE 1D

PRELIMINARY

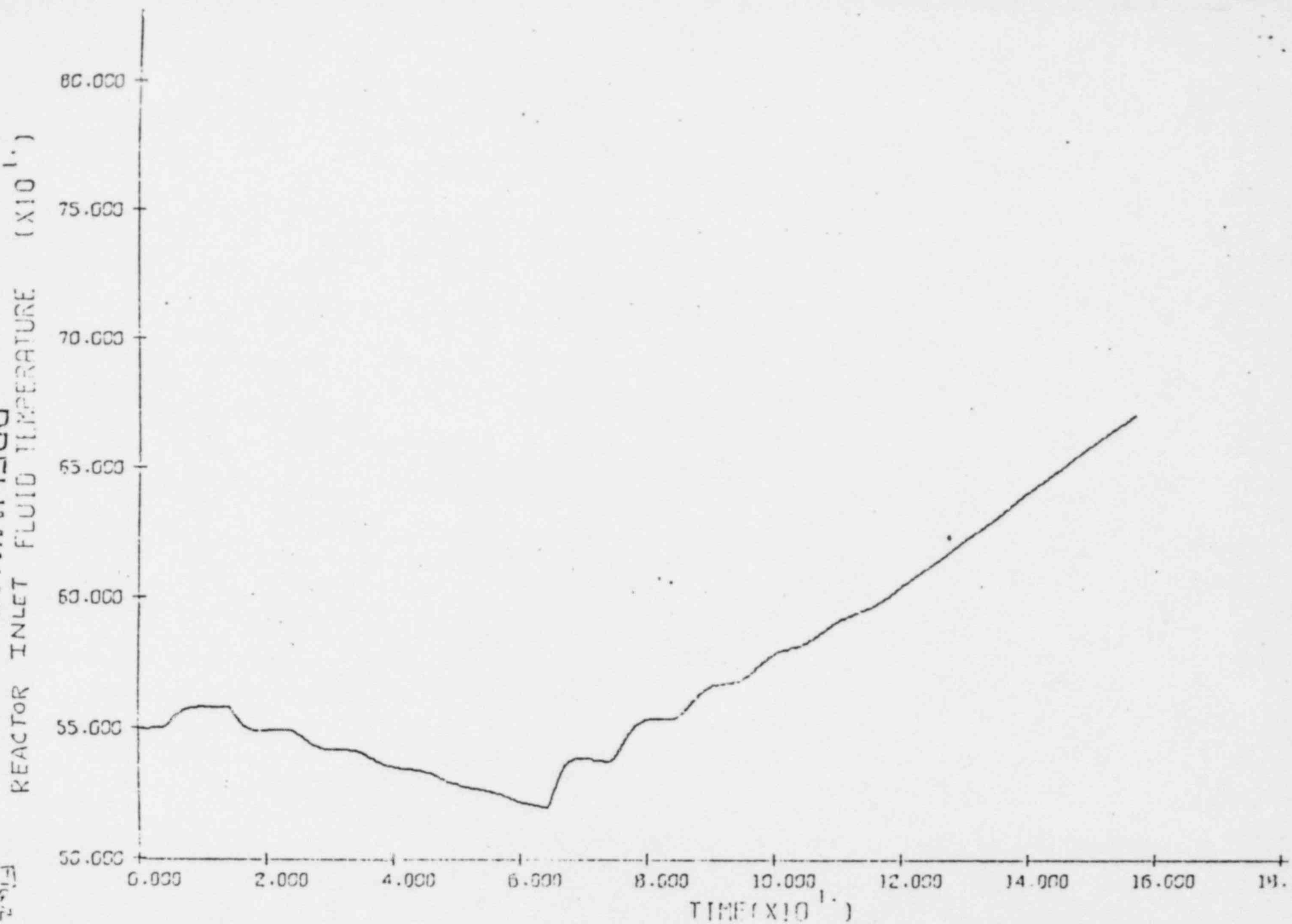
Fig. 210.10-17



ANS SAMPLE 1 CASE 10

PRELIMINARY

Fig. 210.10-14

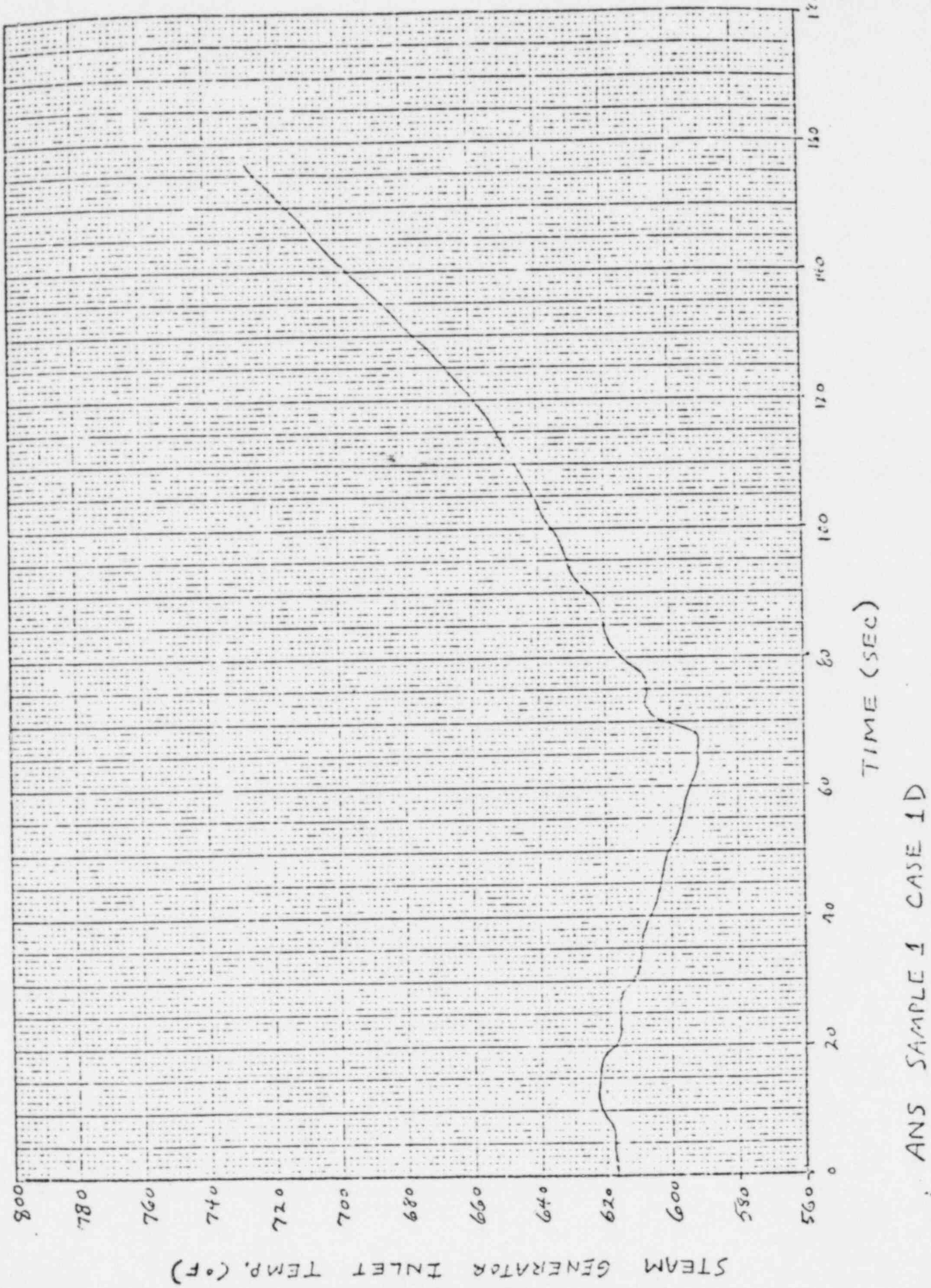


ANS SAMPLE 1 CASE 10

NODE 1 - PIN 1

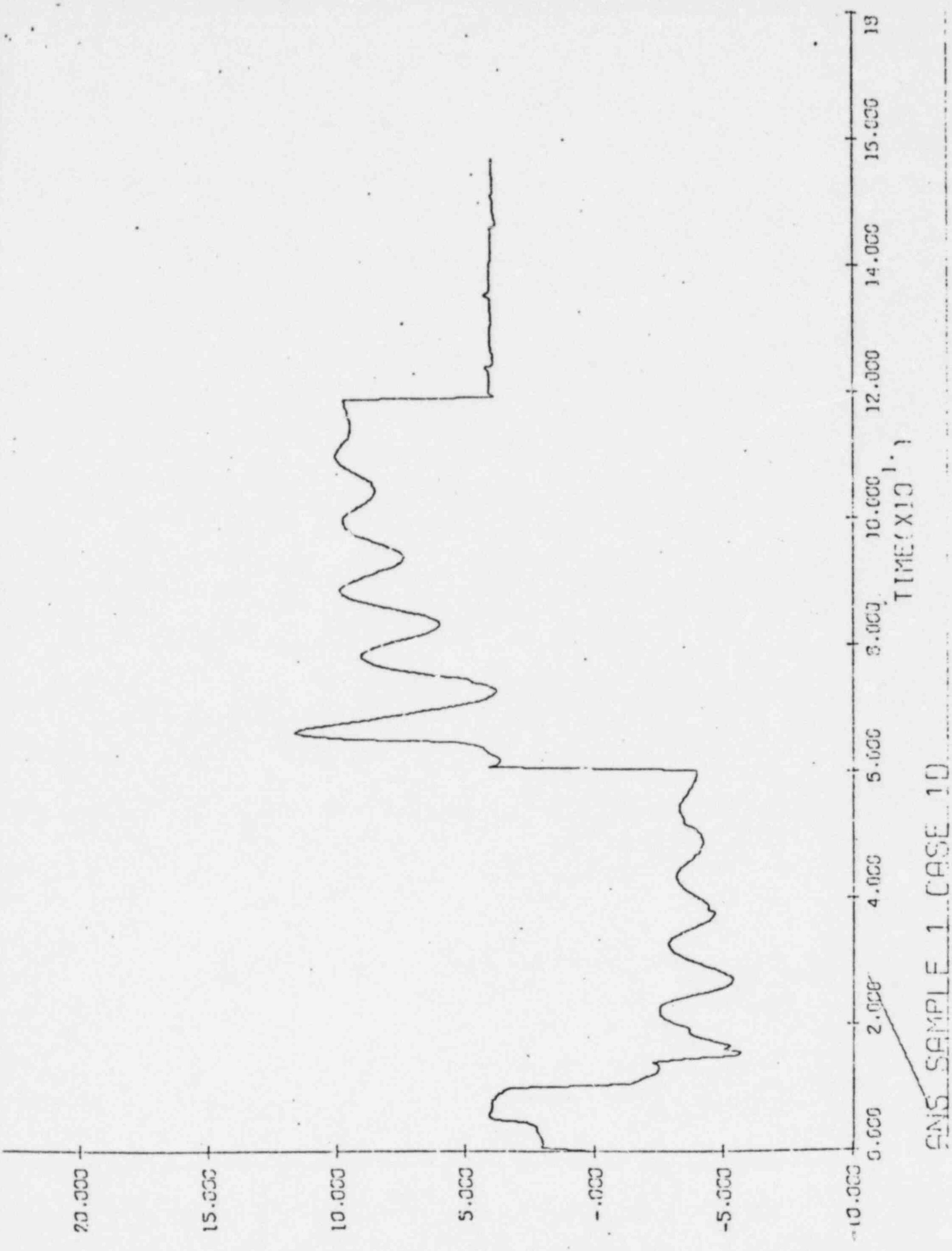
PRELIMINARY

Fig.210.10-19



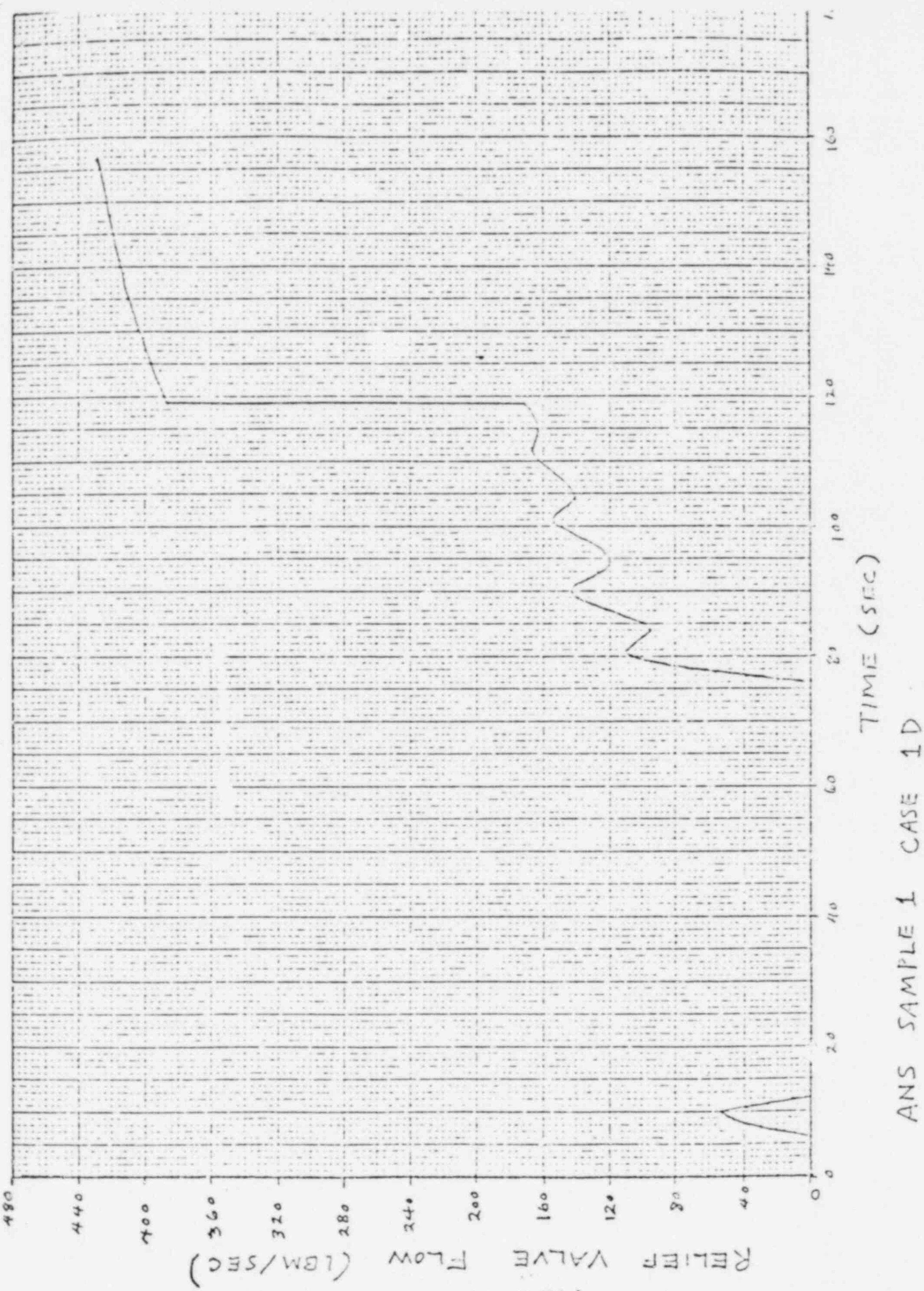
PRELIMINARY

Fig. 210.10 -2



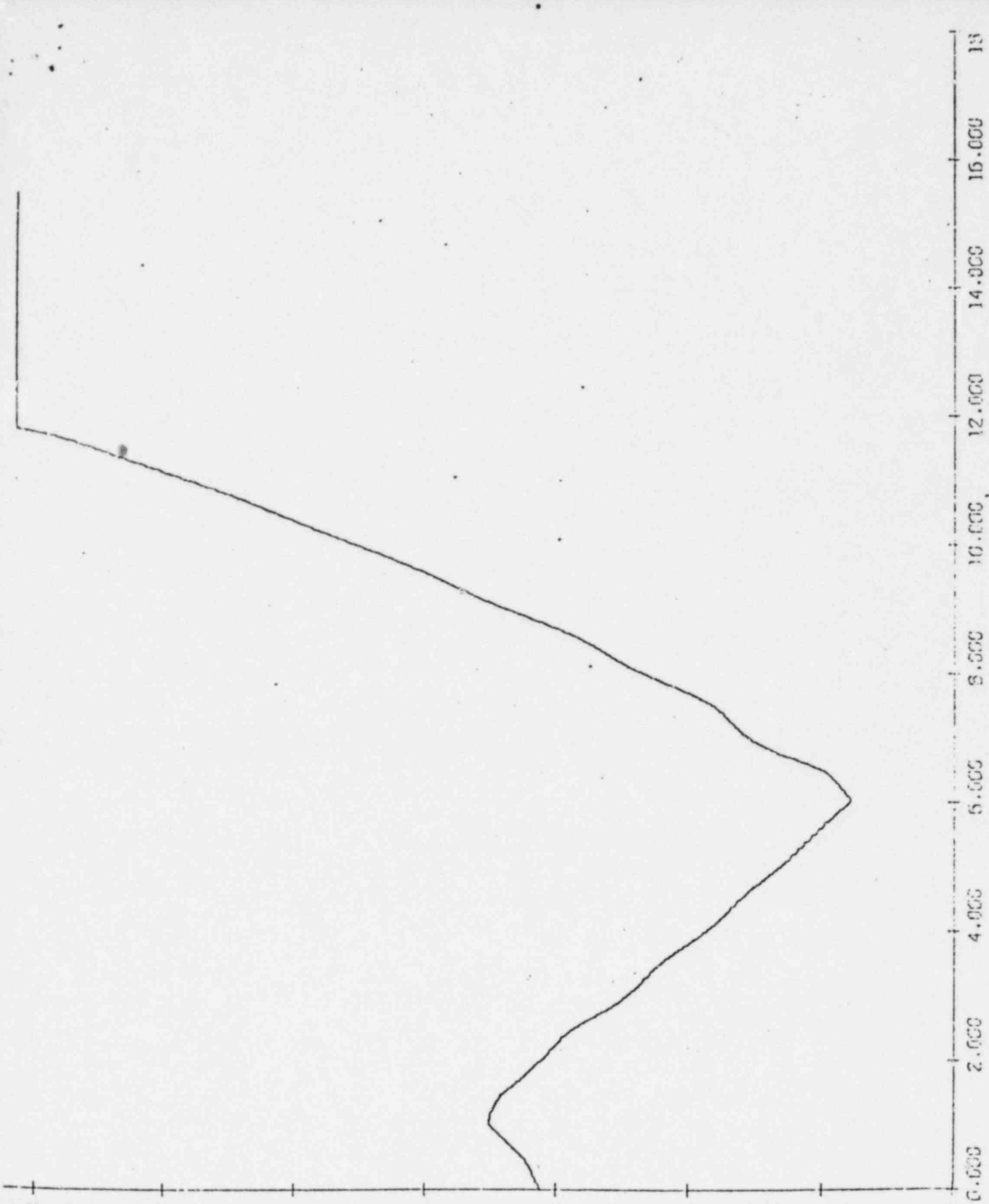
PRELIMINARY

Fig. 210.10-21



PRELIMINARY

Fig. 210.10-22



PRELIMINARY

Fig. 210.10 - 23