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INTERIM REPORT

EG&G Idaho, Inc.

(LTR)

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# INTERNAL TECHNICAL REPORT

Title: EVALUATION OF INSTALLED INSTRUMENTATION FOR LIQUID LEVEL MEASUREMENTS

Organization: LOFT EXPERIMENTAL MEASUREMENTS BRANCH

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NRC Research and Technical Assistance Report

POOR ORIGINAL

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EG&G Idaho, Inc.

LOFT TECHNICAL REPORT
LOFT PROGRAM

FORM EG&G-229
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## ABSTRACT

This report presents an evaluation of the capability of several installed LOFT instruments to measure liquid level. This evaluation was accomplished by comparing thermocouples, neutron detectors, and differential pressure transducers with the LOFT liquid level transducers. A brief discussion of a heated differential thermocouple liquid level transducer is also presented.

## DISPOSITION OF RECOMMENDATIONS

The areas of interest will be initially explored to determine the feasibility of doing the studies recommended by LTR LO-87-79-128.

## SUMMARY

This report compares the data from several installed LOFT measurement transducers to the data from LOFT liquid level transducers (LLTs). The objective of this comparison was to determine if non liquid level transducers presently installed in LOFT could be used to measure liquid level. Transducers analyzed were: cladding thermocouples, fluid thermocouples, self powered neutron detectors (SPND), differential pressure transducers, power range ion chambers, and intermediate range ion chambers.

Discussion of a heated thermocouple liquid level device not presently installed in LOFT has been included to give a more comprehensive review.

Data from two LOFT large break nuclear tests (L2-2 and L2-3) were used for the primary evaluation. One differential pressure measurement from the first LOFT small break nuclear test is also presented.

The results indicate that liquid level may be obtained from those instruments reviewed. However, there are limitations such as: response time, reliability, and ease of presenting real time data. Voiding and refilling of the reactor vessel occurred in 50 seconds for the large break tests; therefore, it is not clear that the conclusions drawn are applicable beyond that time.

Of all the transducers analyzed, the heated thermocouple is the most adaptable to LPWRs. Of the transducers currently installed in LOFT, excluding the liquid level transducer, the SPND and nuclear channels show the most promise for the time frame investigated.

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## 1.0 INTRODUCTION

The Three Mile Island incident highlighted the fact that instrumentation was lacking to define the water inventory in the reactor vessel during a non-normal operating mode. The fact that no commercial LPWR reactors presently incorporate liquid level detectors and that the level of fluid in a reactor vessel is of paramount importance during certain accident modes indicates the importance of obtaining liquid level detectors for existing and proposed commercial reactors. It is also desirable that costs be kept to a minimum to obtain this information; therefore, a study was undertaken in an attempt to determine which measurement devices presently used in commercial reactors would define liquid level. This study was accomplished by using data from LOFT LOCEs.

Following the blowdown portion of a Loss-of-Coolant-Accident (LOCA), the Emergency Core Coolant (ECC) refloods the core to prevent fuel rod damage. The LOFT system was designed to run Loss-of-Coolant-Experiments (LOCEs) and sufficient instrumentation has been included in the core to record this behavior; including fuel clad thermocouples, coolant thermocouples, guide tube thermocouples, self powered neutron detectors, and liquid level transducers.

The question of whether liquid level can be measured by transducers other than liquid level detectors during a LOCE has been presented. To answer this question, a comparison of cladding thermocouples, fluid thermocouples, intermediate and power range neutron ion chambers, self powered neutron detectors, and differential pressure transducers were made with the conductivity liquid level detectors during L2-2 and L2-3. The center fuel module was the primary source of data as this module contained most of the transducers and detectors under examination.

A special differential measurement from the intact loop hot leg to the reactor vessel head was made during small break test L3-1. The data from this measurement was also evaluated.

A brief discussion of the PBF heated differential thermocouple liquid level indicator is included in this report for completeness. There is no cross correlation of this transducer to LOFT liquid level phenomenon during a LOCA; however, the results of tests using this device will be documented by PBF.

## 2.0 DISCUSSION

### 2.1 Comparison Methodology and Assumptions

To evaluate the potential of using thermocouples, SPNDs, ion chambers, and differential pressure transducers to measure liquid level in a reactor vessel, it was decided that those transducers would be compared against the liquid level detectors presently used in LOFT.

Due to the complexity of the environment, i.e. thermal-transients, neutron flux, gamma flux, and pressure changes during a LOCE, the following assumptions were required:

1. It was assumed that transducers in the same general geometric location would see the same hydrological and nuclear flux phenomenon.

2. Only liquid level electrode voltages exceeding 80% of maximum would be used to determine a dry (voided) condition.

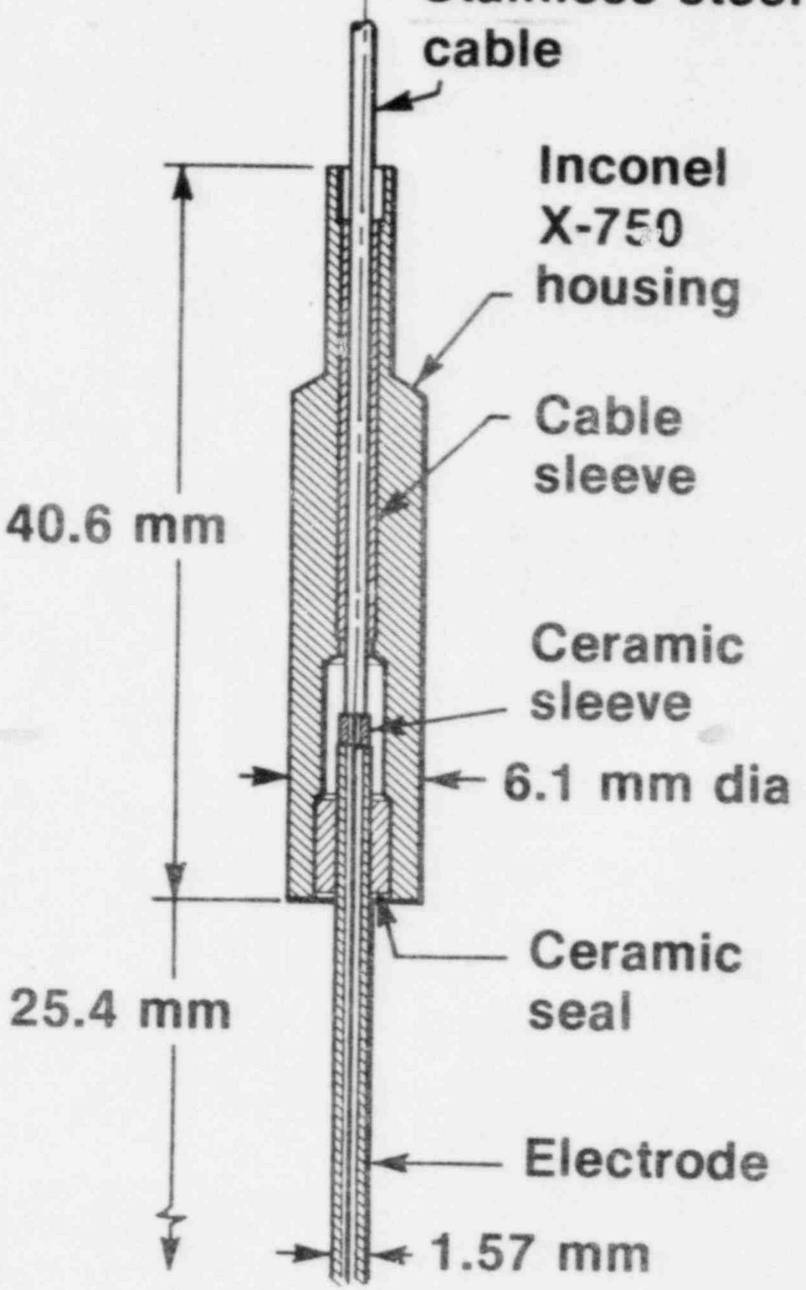
3. Thermocouple outputs that exceed 120% above the saturation temperature should be considered to indicate a change in the fluid state.

4. The shape of the neutron and gamma flux decay curves obtained during L3-1 would be representative of those curves for L2-2 and L2-3.

## 2.2 Liquid Level Transducer (LLT)

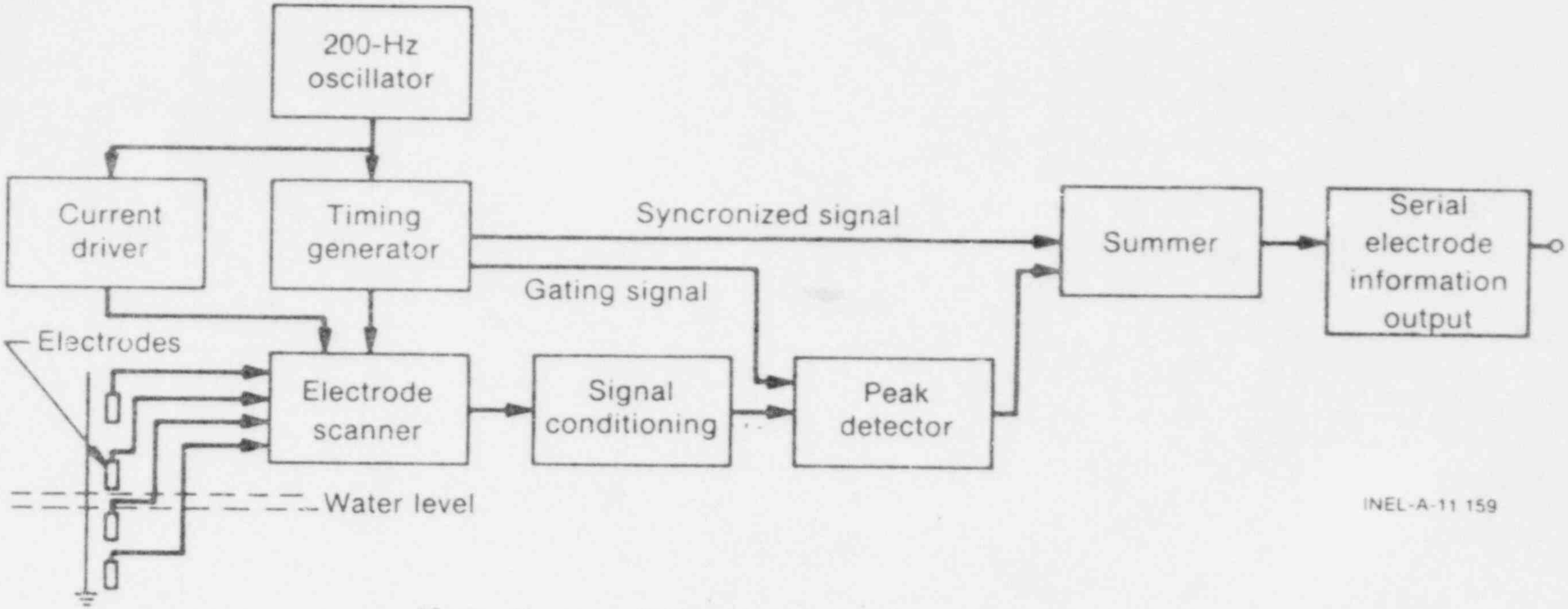
The LLTs used in LOFT are the conductivity electrode type. The electrode consists of a center conductor that extends beyond a conductive outer grounding sheath. The conductor and sheath are separated by a ceramic that forms a pressure seal (Figure 1). Figure 2 is a block diagram of the LLT electronics. The system effectively measures the resistance of the media between the electrode and the ground plane. This resistance is then interpreted as to the presence or absence of fluid. The operation and uncertainty analysis is covered in Reference 1.

# LLT Electrode



INEL-S-19 136

FIGURE 1



Block diagram of the liquid level transducer and instrumentation for four electrode probe output signals.

FIGURE 2

The LLT used for reference consisted of 19 discrete electrodes, 9.65 cm apart. This transducer was located in the center fuel assembly and was designated by LE-5E11. The electrode is designated by adding a number which represents its location from the top of the core. For instance, LE-5E11-3 would designate the third electrode from the top of the core.

The LLT has proven useful in the measurement of liquid level, and its behavior is adequately understood. Therefore, the interpretation of the other transducer data will be compared to the electrode data that is closest to the same geometric plane.

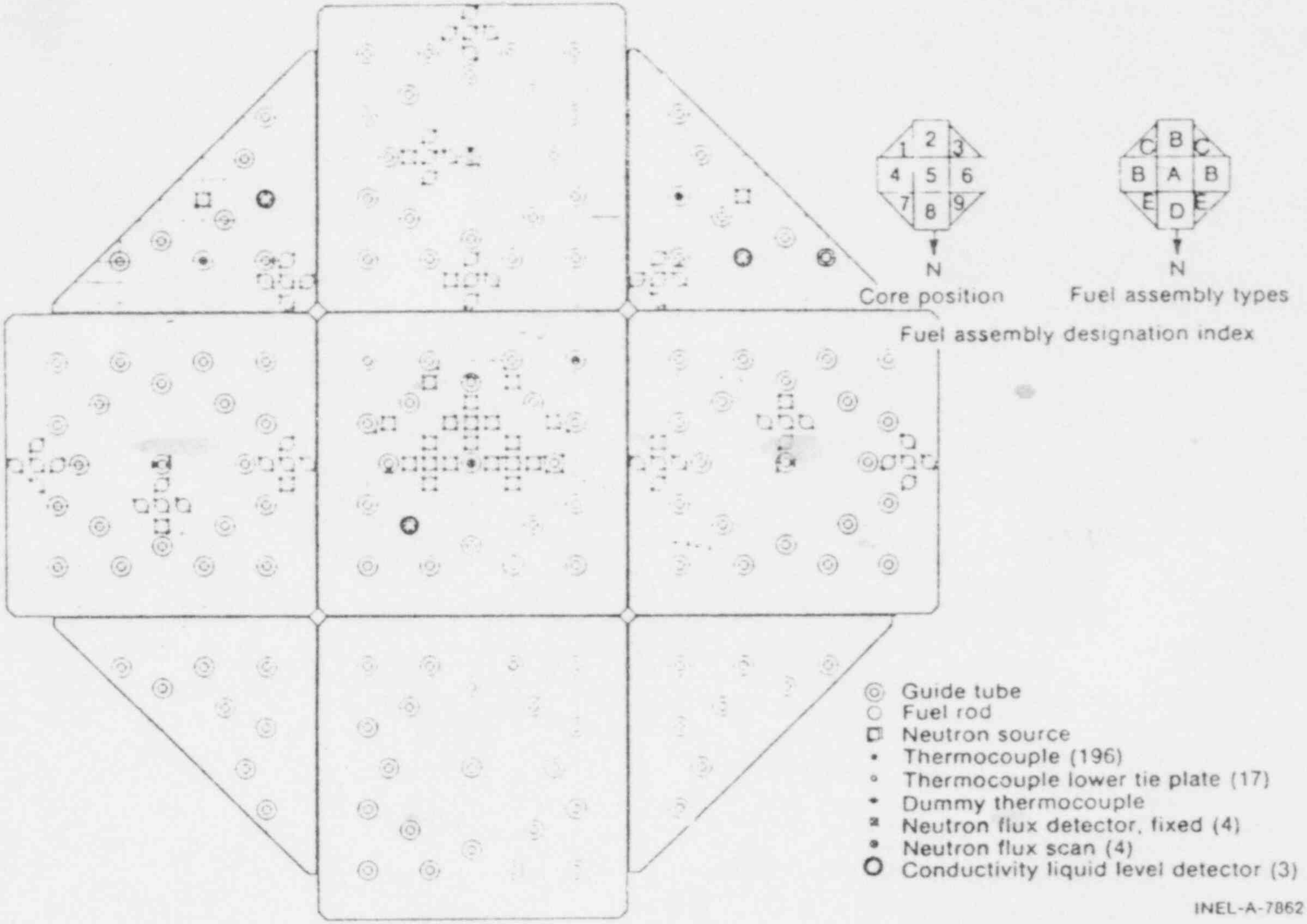
It should be noted that application of the LLT to a commercial reactor would require resolution of the following problems:

1. reliability ( ~3,000 to 4,000 EFPH)
2. real time data interpretation

The extent of the development work that would be required to make this adaptation is currently being investigated.

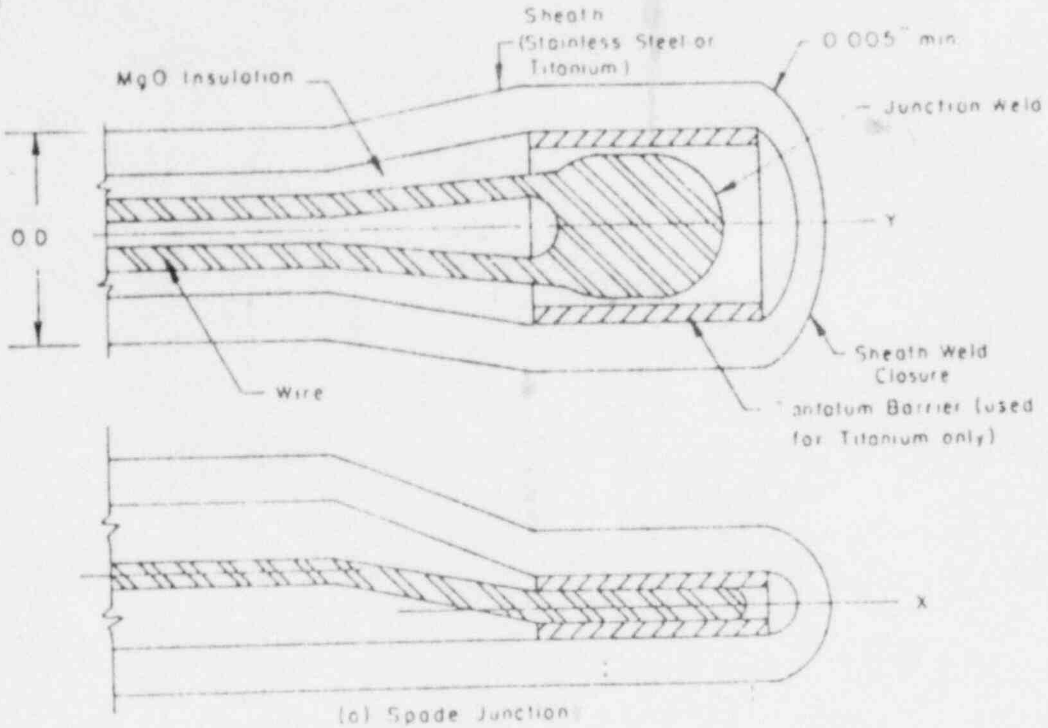
### 2.3 Clad Thermocouples

Clad temperatures in LOFT are measured at various axial and radial locations in the LOFT core (Figure 3). The clad thermocouples are Type K thermocouples with a spade junction that is laser welded to the cladding of the fuel rods (Figure 4). Cladding thermocouples are designated by a measurement identification as follows:

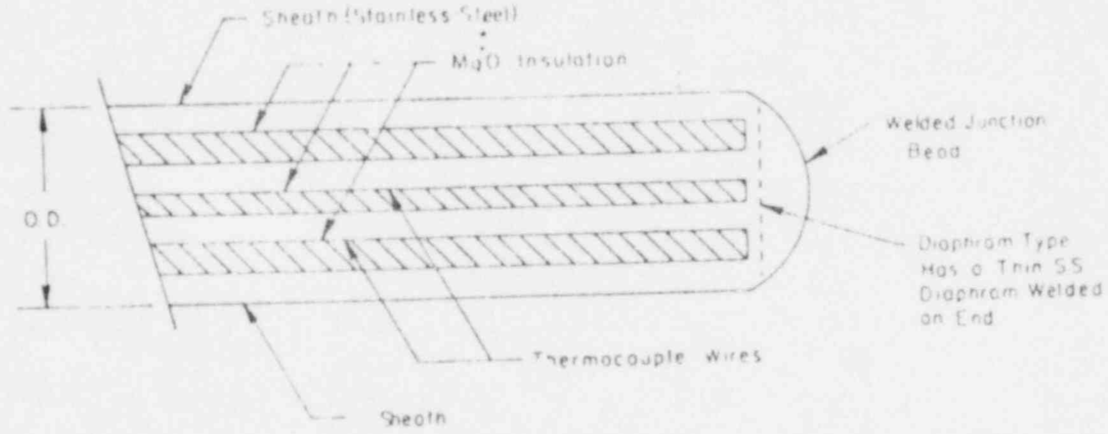


LOFT core map showing instrument locations.

FIGURE 3



(a) Spade Junction



(b) Grounded Weld Junction

Thermocouple weld details.

FIGURE 4

TE-5J8-32

Axial distance above bottom of core in inches

Location within fuel assembly

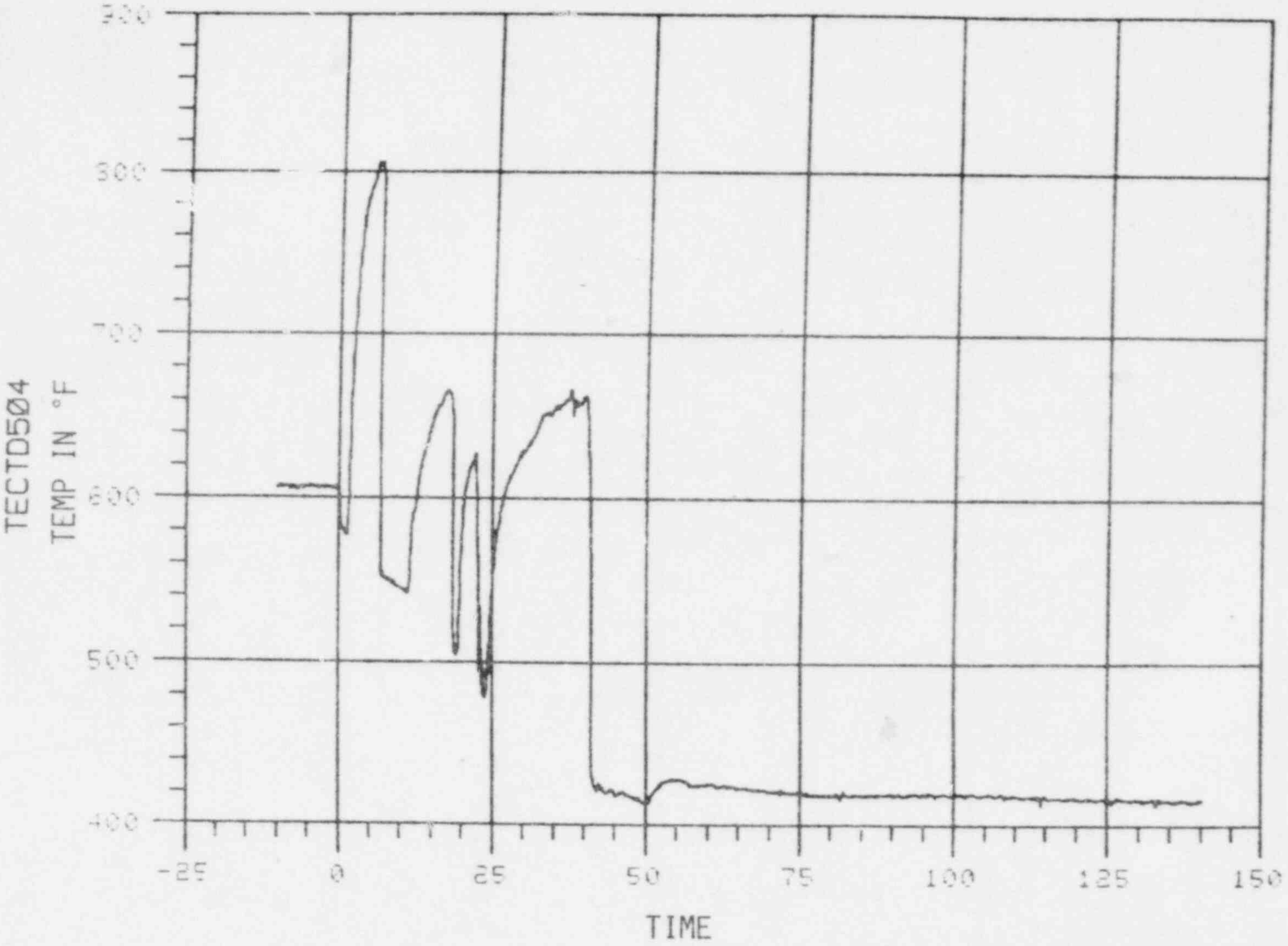
Fuel assembly number

Identifies an experimental temperature measurement

This identification philosophy applies to all in-core measurements, with the first two numbers changing according to the transducer type.

For this evaluation, the general criteria used for determining whether liquid or vapor exists around a thermocouple is whether the measured temperature is lower than, higher than, or equal to the saturation temperature. If the temperature is lower than the saturation temperature obtained from an absolute pressure measurement and steam tables, the surrounding media is subcooled water. If the measured temperature equals the saturation temperature, the surrounding media may be either saturated liquid or steam. If the measured temperature is greater than the saturation temperature, the surrounding media is superheated steam. It should be noted that these criteria are compromised on clad thermocouples because of the continual heat source. If the heat transfer from the fuel rod to the surrounding media causes a vapor boundary to form, the above criteria would give an erroneous liquid level measurement.

A typical data trace for a clad thermocouple is shown in Figure 5. Using the criteria established above, the following comparisons were made:



DATA FOR CLAD THERMOCOUPLE TE-5E8-034.5 LOCATED ON THE CENTER FUEL BUNDLE FOR L2-3

FIGURE 5

1. The cladding thermocouples, approximately 40 cm above the bottom of the fuel rods, indicate wetting and drying (voiding) conditions that agree very well with the liquid level electrodes in that vicinity (Figure 6).

2. The cladding thermocouples, approximately 86 cm above the bottom of the fuel rods agree quite well with the liquid level electrodes during the initial voiding and rewet, but do not agree during the final voiding and rewet. This may be due to cooling of the rods and thermocouples by saturated steam. The final rewet of the core, in this vicinity, varies from 24 (LE-5E11-9) to 54 seconds (TE-5J8-32), with the liquid levels LE-5E11-9 and LE-5E11-011 indicating rewet at 24.0 and 42.0 seconds respectively. Due to the differences in liquid level readings, it is not clear as to the conditions existing in this area of the core at that time.

3. The cladding thermocouples appear to lag the liquid level transducer by approximately 1 to 2 seconds.

## 2.4 Fluid Thermocouples

Another method of evaluating liquid conditions in the core is to analyze the fluid thermocouples in the upper plenum. If liquid exists in the vessel, the thermocouples will indicate saturation or subcooled conditions; and if no or low quantities of water exist, superheated steam temperatures or saturation temperatures will be measured.

## L2-3

## LIQUID LEVEL COMPARISONS FOR CENTER FUEL BUNDLE

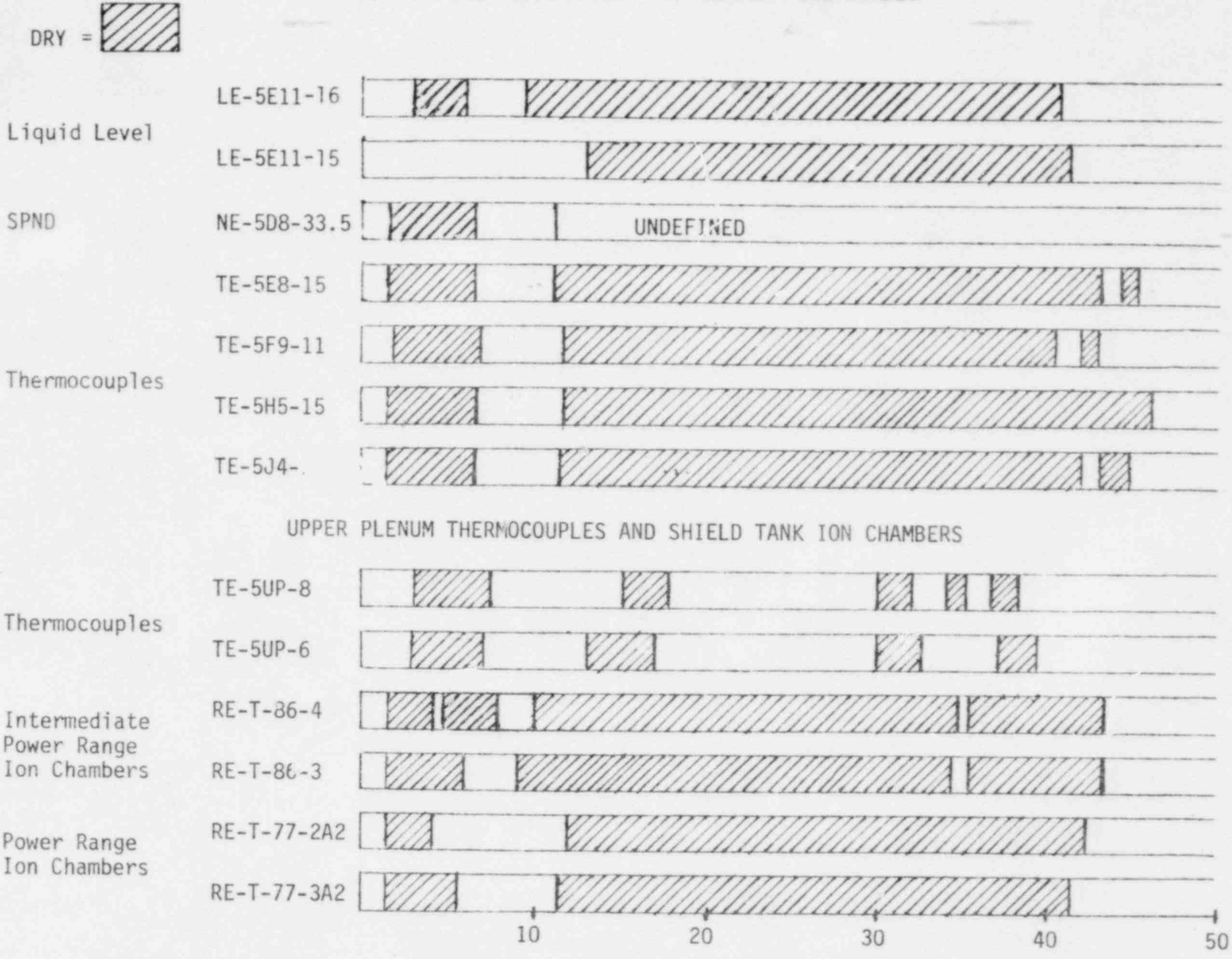


FIGURE 6

The fluid thermocouples evaluated (TE-5UP-6 and TE-5UP-8) are Type K, diaphragm grounded junction (Figure 4) located in the upper plenum area. The small mass of this thermocouple gives it a fast response ( ∿70 msec) to temperature changes.

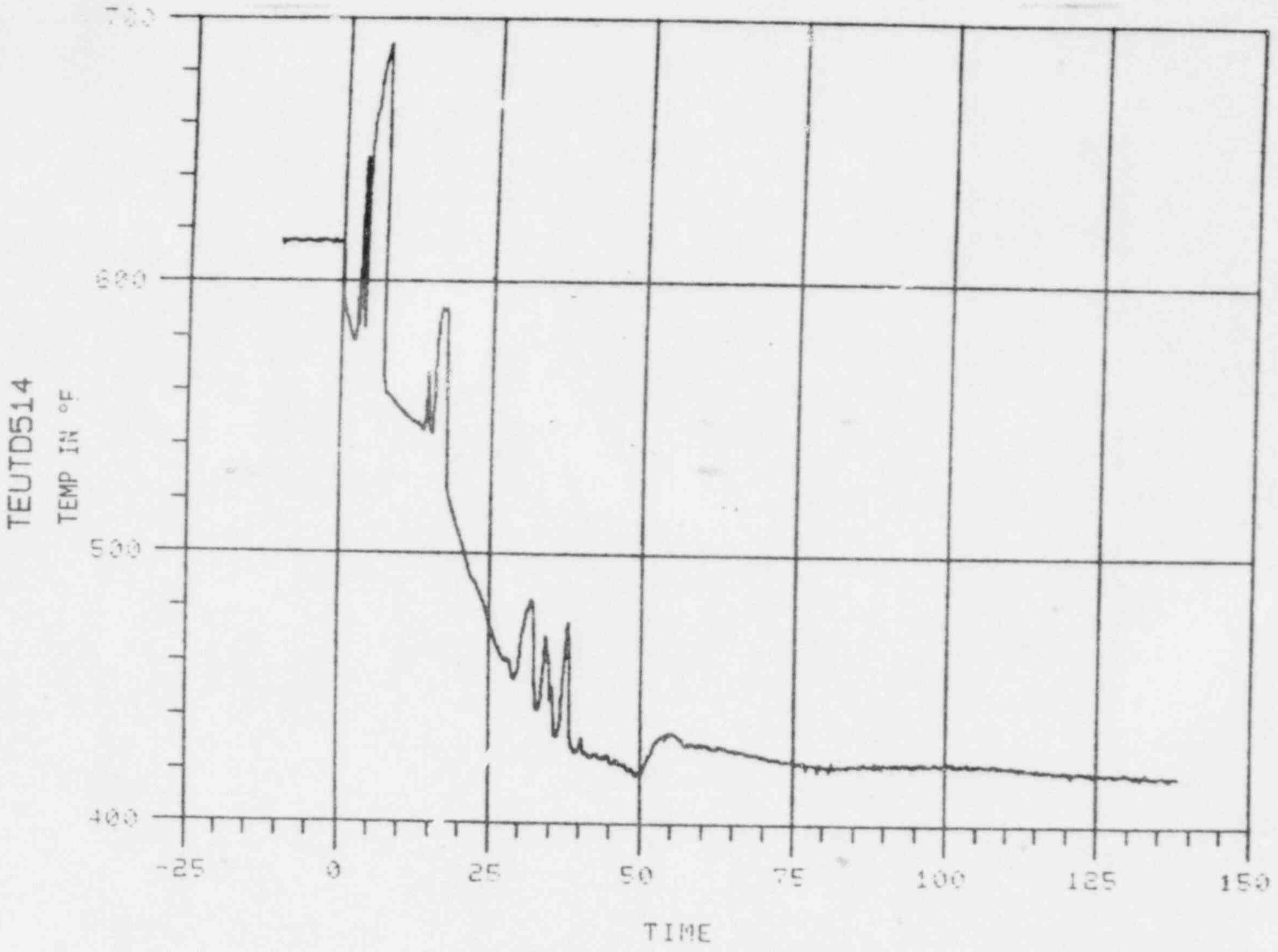
The criteria used to evaluate the fluid thermocouples is the same as that given in Section 2.3. The problems in evaluating the fluid thermocouples are:

1. Saturated steam cannot be distinguished from saturated water.

2. The fast response may make the thermocouples susceptible to temperature changes due to entrainment which would cause an early indication of quenching.

As shown in Figure 6, the initial voiding in the upper plenum, as determined by thermocouples TE-5UP-6 and TE-5UP-8, is approximately what one would expect given the liquid level time sequence of liquid level electrode LE-5E11-16. Data from a typical fluid thermocouple are presented in Figure 7. The initial rewet is also what one would expect. However, from that point on, it appears that the liquid level data has been influenced by saturated steam or by extrained liquid (Figure 6).

The fluid thermocouples may be used in a commercial reactor through the use of the subcooled meter. This device obtains the saturation temperature by using an absolute pressure transducer. The pressure is measured, and then the temperature is obtained from



DATA FOR FLUID THERMOCOUPLE (TE-5UP-008) LOCATED IN THE UPPER PLENUM OF CENTER FUEL BUNDLE FOR L2-3

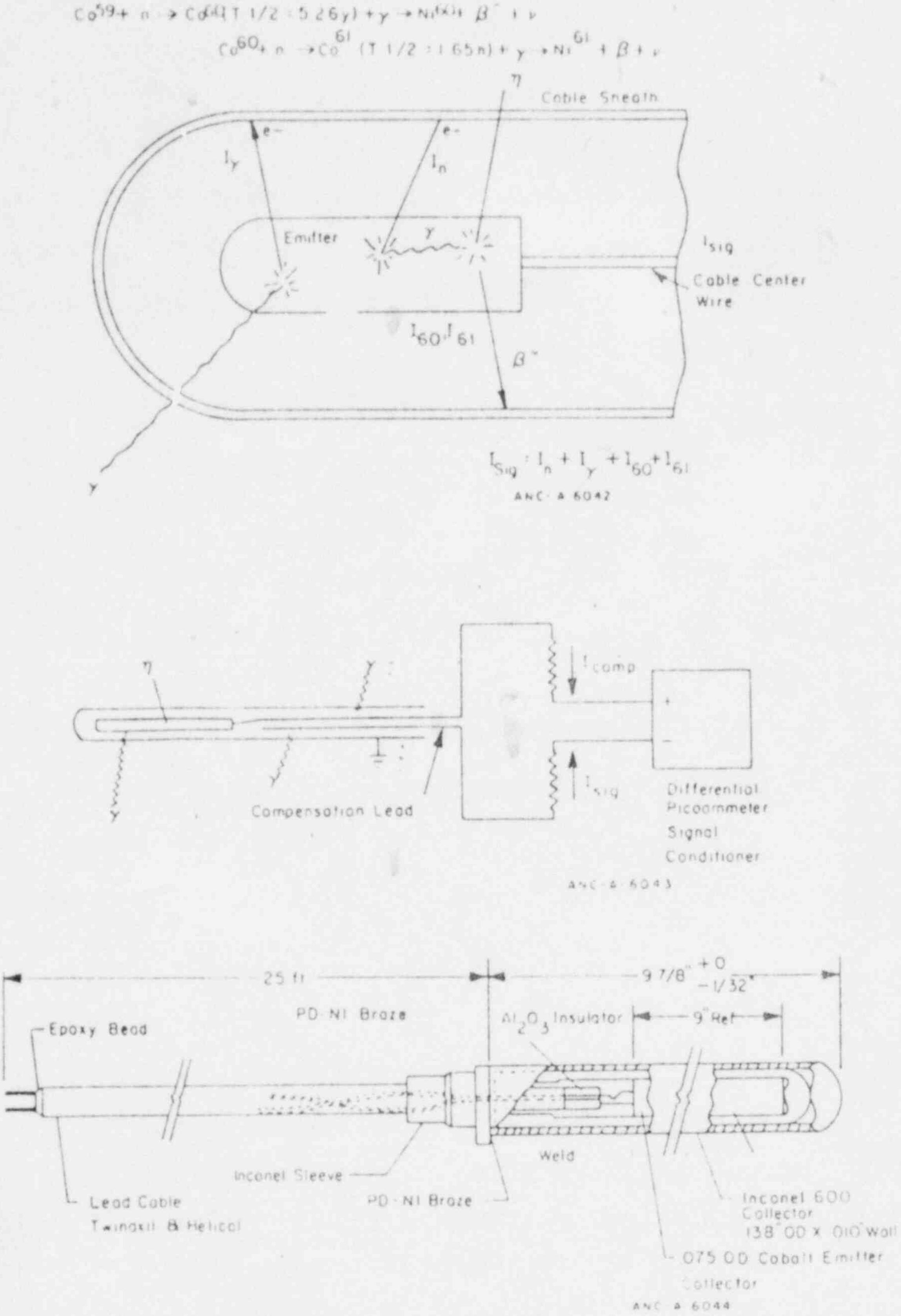
FIGURE 7

steam tables. The difference between the steam table temperature and the measured temperature indicates subcooled, saturation, or a superheated condition. This could also be used as a qualitative measure of the fluid state, i.e. subcooled would indicate water, saturation would indicate water or steam, and superheated would indicate steam or a voided condition.

## 2.5 Self Powered Neutron Detector (SPND)

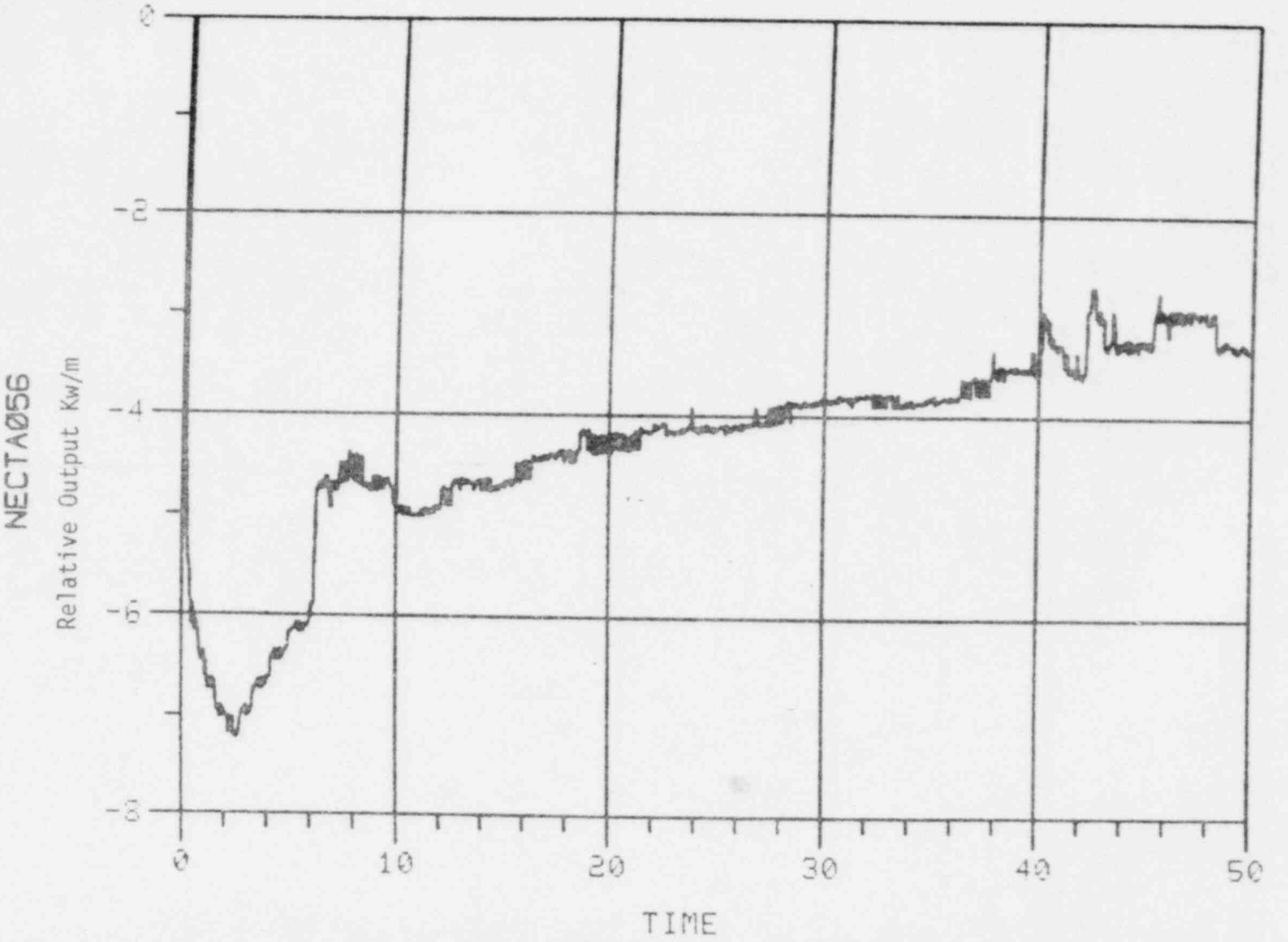
There are four cobalt-emitter SPNDs in the LOFT core. They are located in guide tubes in the following core locations: 2H8, 4H8, 5D8, and 6H8. They are centered about an axial location 26 inches above the bottom of the fuel pins which corresponds approximately to the elevation of the highest neutron flux (or core power) or hot plane. These detectors have an active length of nine inches consisting of a cobalt-emitter, inconel collector (which forms the outer sheath), and aluminum oxide insulator (Figure 8). There is an inconel signal wire leading from the emitter to the cable and connector and a parallel inconel wire for cable gamma compensation. Further details on the construction of the SPNDs are found in References 2 and 3.

The LOFT SPND output during the L2-3 experiment is depicted in Figure 9. As seen from this figure, the SPND response is initially positive prior to initiation of the LOCE (i.e. for time < 0 sec) and goes negative immediately after initiation of the LOCE. The negative response of these detectors is believed to be due to the domination of SPND gamma current (i.e. SPND current due to the



Self-powered neutron detector.

FIGURE 8



DATA FOR SELF POWERED NEUTRON DETECTOR (NE-5D8-033.5) LOCATED

IN CENTER FUEL BUNDLE FOR L2-3

FIGURE 9

interaction of the detector with the external gamma flux of the decaying fission products and activation products) over the neutron current. The negative response of cobalt SPNDs to a gamma flux is well documented (e.g. References 4, 5, and 6).

In a voided core, the neutron current of a cobalt SPND is depressed due to two complimentary effects: 1) the total neutron flux is very low due to the large amount of negative reactivity in the core which, in turn, is caused by the negative void coefficient of the core, and 2) since there is so little moderation of the neutron flux, the thermal neutron to fast neutron flux ratio is very small (the cobalt SPNDs detect neutrons by the $^{59}$Co (N,$\gamma$) $^{60}$Co interaction in the emitter which peaks at thermal neutron energies). Superimposed on this is the effect of the natural decay of the delayed neutron flux.

For the LOFT core, the effective decay kernel at 6.4 seconds (defined as:

$$I\ (t{=}o) \equiv e^{-\lambda}_{eff} \equiv \Sigma\ \frac{\beta_i}{\beta} e^{-\lambda it}$$

where $\beta_i$, $\lambda i$, and $\beta$ are as listed in Reference 7) is 0.2155. That is, in the absence of additional delayed neutron precursor production, at t - 6.4 seconds after shutdown, approximately 78% of the delayed neutron flux has decayed away.

The gamma current of the cobalt SPNDs is caused by a number of competing effects (Reference 8) including interaction of the gamma ray with the SPND collector, insulator, or emitter as well as interaction of the gamma ray with core structural materials to produce a Compton- or photo-electron and subsequent interaction of the electron with the SPND. Voiding in the core will enhance either of these effects due to the decrease of moderator attenuation of these fluxes (external gamma and electron fluxes).

Figure 10 depicts the expanded data to more clearly show the "step" change in SPND output shortly after LOCA initiation. As can be seen, the output increases (i.e. becomes less negative) in all cases.

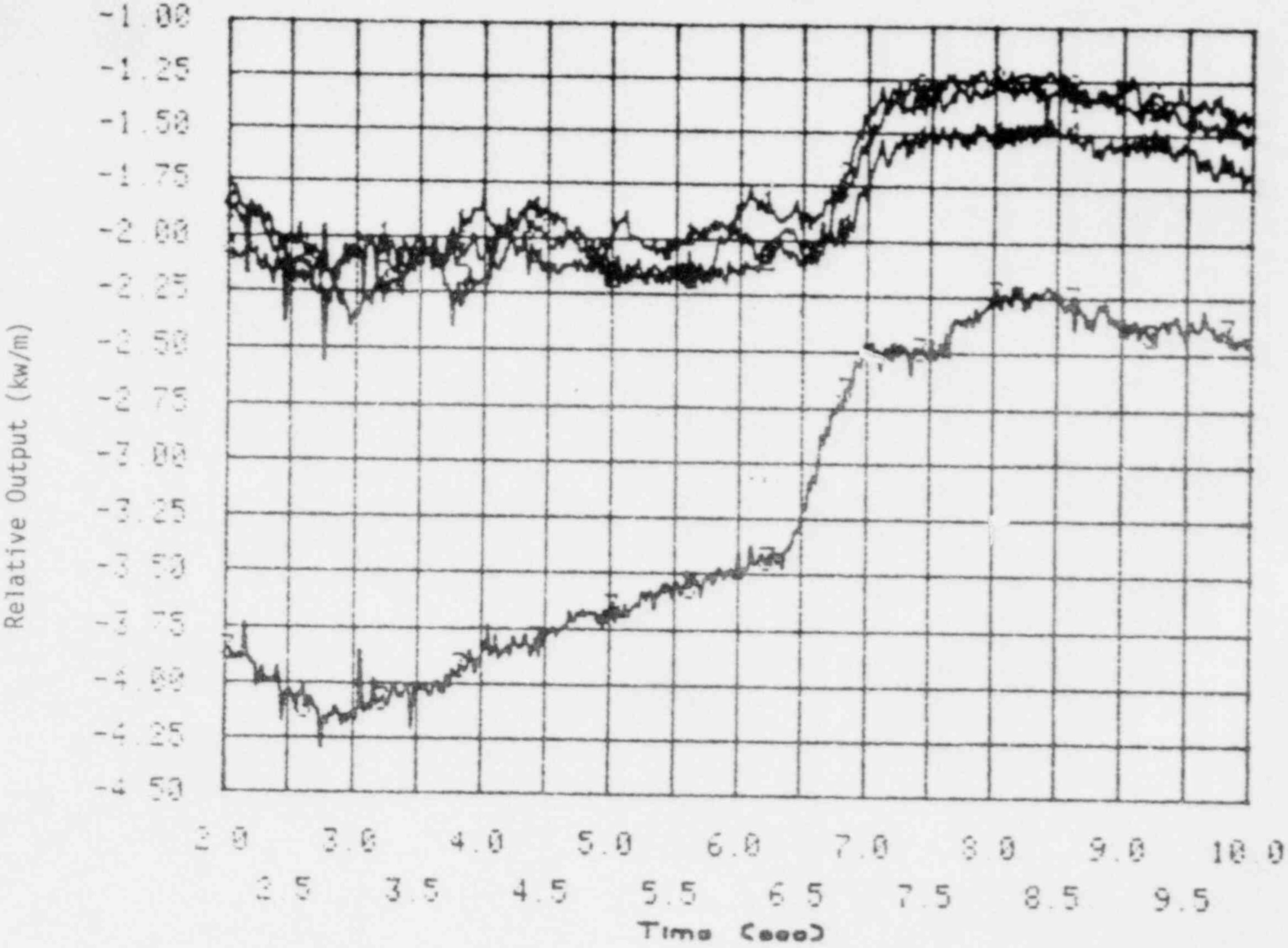
In the initial analysis, it was assumed that the entire change in SPND output was caused by attenuation of the external gamma flux (i.e. the SPND neutron and external electron currents were neglected). A single gamma energy group, infinite (i.e. no leakage) homgeneous reactor model was used:

$$\Phi_\gamma \mu_a = \text{Const.}$$

where

$\Phi_\gamma$ = external gamma flux
$\mu_a$ = total linear gamma absorption coefficient.

The results from this calculation showed that given the experimentally determined saturation pressure and temperature, it would require a physically non-realistic moderator density change to account for the change in SPND output solely on the basis of attenuation of gamma flux. Otherwise stated, there must be an appreciable current due to neutron and/or external electron fluxes.



SPND OUTPUTS DURING REWET PORTION OF L2-2

FIGURE 10

Subsequently, a calculation was made to attempt to quantify the effect of neutron flux on the SPND output. The following assumptions on the SPND current contributions were made in the calculation, which is shown in detail in Appendix A:

1. $I_{si} = I_{\gamma} + I_{ni}$, where $I_{si}$ = the total SPND current prior to rewet; $I_{\gamma i}$ = SPND gamma current prior to rewet; $I_{ni}$ = SPND neutron current prior to rewet $\sim 0.0$.

2. $I_{sf} = I_{\gamma f} + I_{nf}$ , where f → current subsequent to rewet.

3. The initial moderator density is equal to that corresponding to saturated vapor.

4. The final moderator density is equal to that corresponding to saturated liquid.

Assumption 4 was necessary since an analytical expression for $I_n$ as a function of moderator density is not available. The purpose of the calculation is to determine the neutron multiplication factor defined as:

$$n = \frac{I_{snf}}{[S_n][\phi_n\ (t < 0)][\rho_{eff}][e^{-\lambda t}_{eff}]}$$

where $\frac{I_{snf}}{S_n}$ = resultant neutron flux, and

$[\phi_n\ (t < 0)]\ [\rho_{eff}]\ [e^{-\lambda t}_{eff}]$ = source flux.

The results of this calculation are summarized below in Table I.

Using the data from Reference 7 for rod worths and calculating the neutron multiplication factor from:

$$n = \frac{\rho - 1}{\rho},$$

results in the calculated values for n ranging from n = 10 to n = 25. Thus, the values for n calculated using the method of Appendix A are reasonable in magnitude.

TABLE 1

| Experiment | SPND Location | n** |
|---|---|---|
| L2-2 | 2, 4, 6* | 16.6 |
| L2-2 | 5 | 14.0 |
| L2-3 | 2, 4, 6 | 19.0 |
| L2-3 | 5 | 21.0 |

---

* Based on core symmetry, SPNDs 2H8, 4H8 and 6H8 are in equivalent locations.

** n = neutron multiplication number

Results of the above analysis indicate that the LOFT SPNDs are sensitive to core voiding and subsequent rewet. This is based on:

1. The output from all SPNDs shows a similar "step" increase in output (i.e. decrease in negative output) which can be time-correlated with adjacent thermocouples.
2. The sign of the step increase is consistent with the hypothesis of an increase in moderator density.
3. The magnitude of the step increase is consistent, within experimental uncertainty, with a combination of gamma current decrease (due to attenuation of the gamma flux) and neutron current increase (due to increased neutron multiplication and moderation).

## 2.6 Differential Pressure Transducer

The basic principle for obtaining liquid level from a differential pressure measurement is to have a reference leg that stays full of water for one side of the differential pressure and the variable side for the other. The basic equation is:

$$\Delta p = \rho g h$$

where: $\Delta p$ = differential pressure

$\rho$ = density

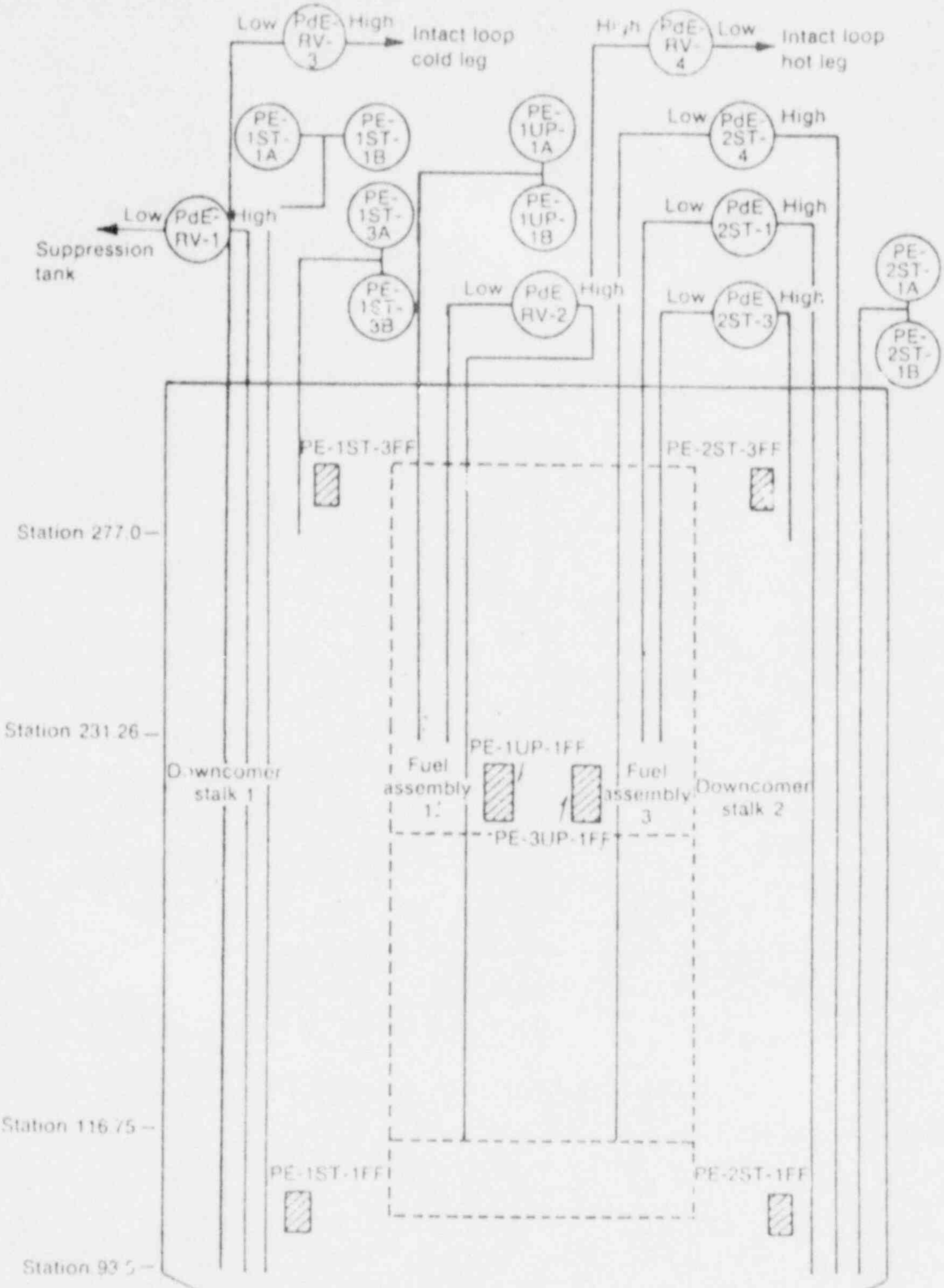
$g$ = gravitational constant

$h$ = height of variable liquid level leg

The differential pressure transducer that was used for the analysis measured the pressure difference between the upper end box and the lower end box (PdE-RV-2, essentially across the core) of the reactor (Figure 11). The differential pressure transducer was of the strain gage balanced bridge configuration located external to the reactor vessel.

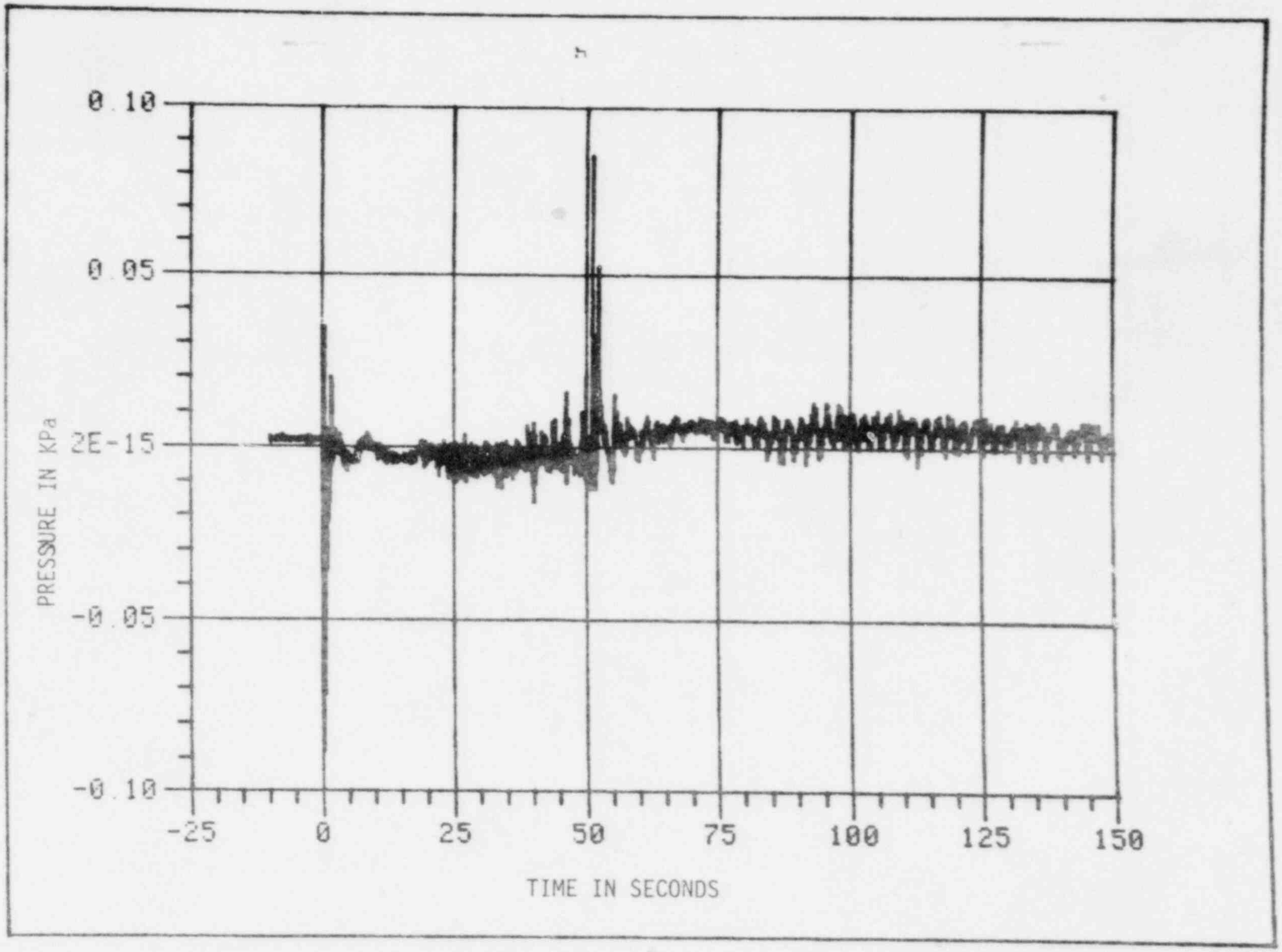
A brief review of the data (Figures 12 and 13) displays some points of inflection that appear to correlate with liquid level. However, at this time it is not apparent how to reduce the data to obtain quantitative liquid level information. The differential pressure is also sensitive to flow which means the flow part of the differential pressure must be removed before the liquid level can be evaluated. To further complicate this, the upper pressure tap is a dynamic tap and the reference leg blows down because it is located inside the reactor vessel.

A differential pressure transducer (PdE-RV-005) was added to the reactor vessel prior to L3-1 to obtain data on the upper plenum liquid level (Figure 14). The transducer measures the differential pressure from the top of the reactor vessel to the outlet pipe. The transducer is connected in such a manner that the upper connection is to the high side of the transducer and the low side is to the intact loop hot leg. This gives a positive reading in pressure equal to 2.02 meters in height when the upper plenum is void. The reference leg is external to the reactor vessel and remains full throughout the blowdown.



Reactor vessel pressure measurement locations.

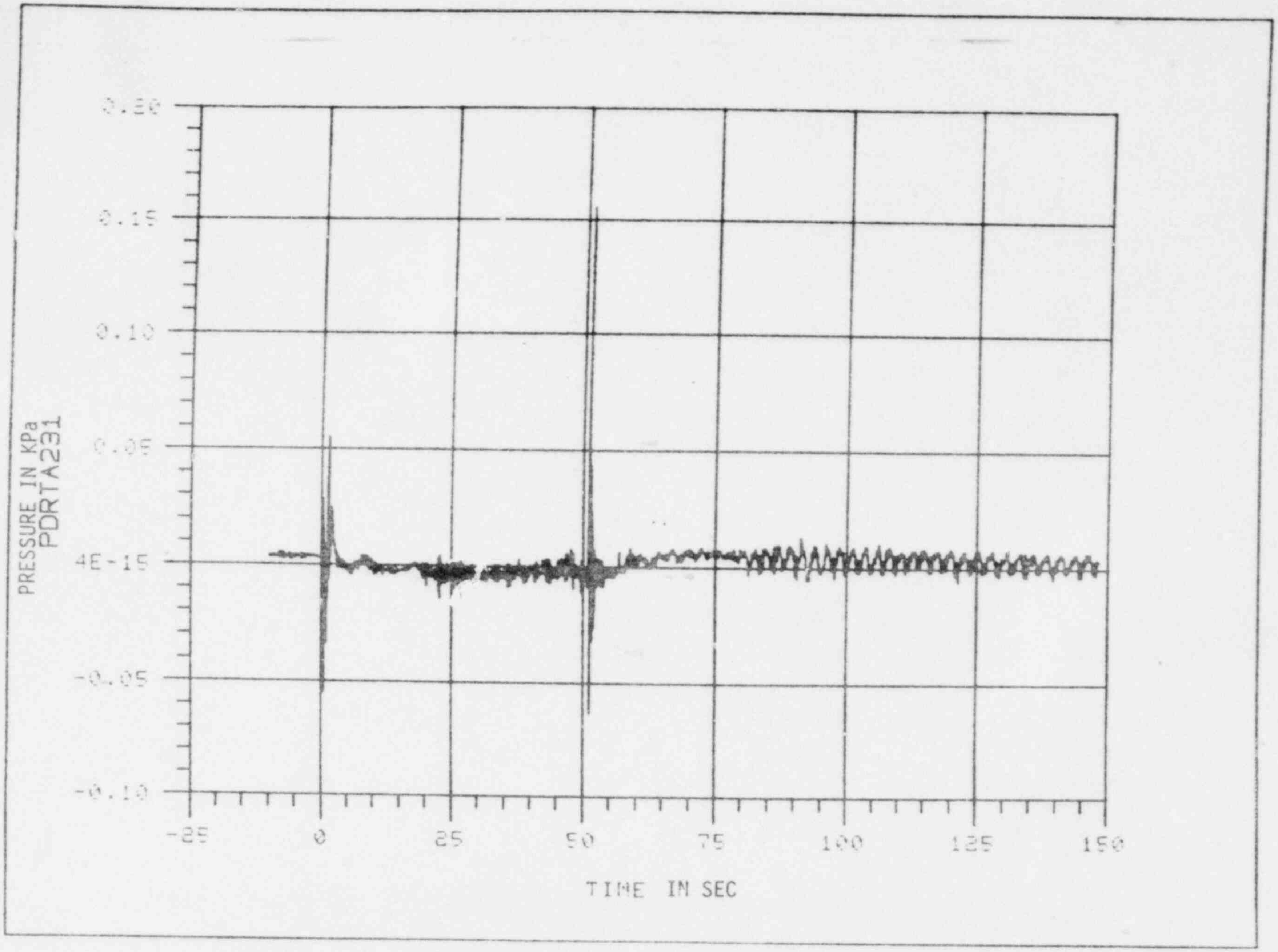
FIGURE 11



DATA FOR DIFFERENTIAL PRESSURE TRANSDUCER (PDE-RV-002)

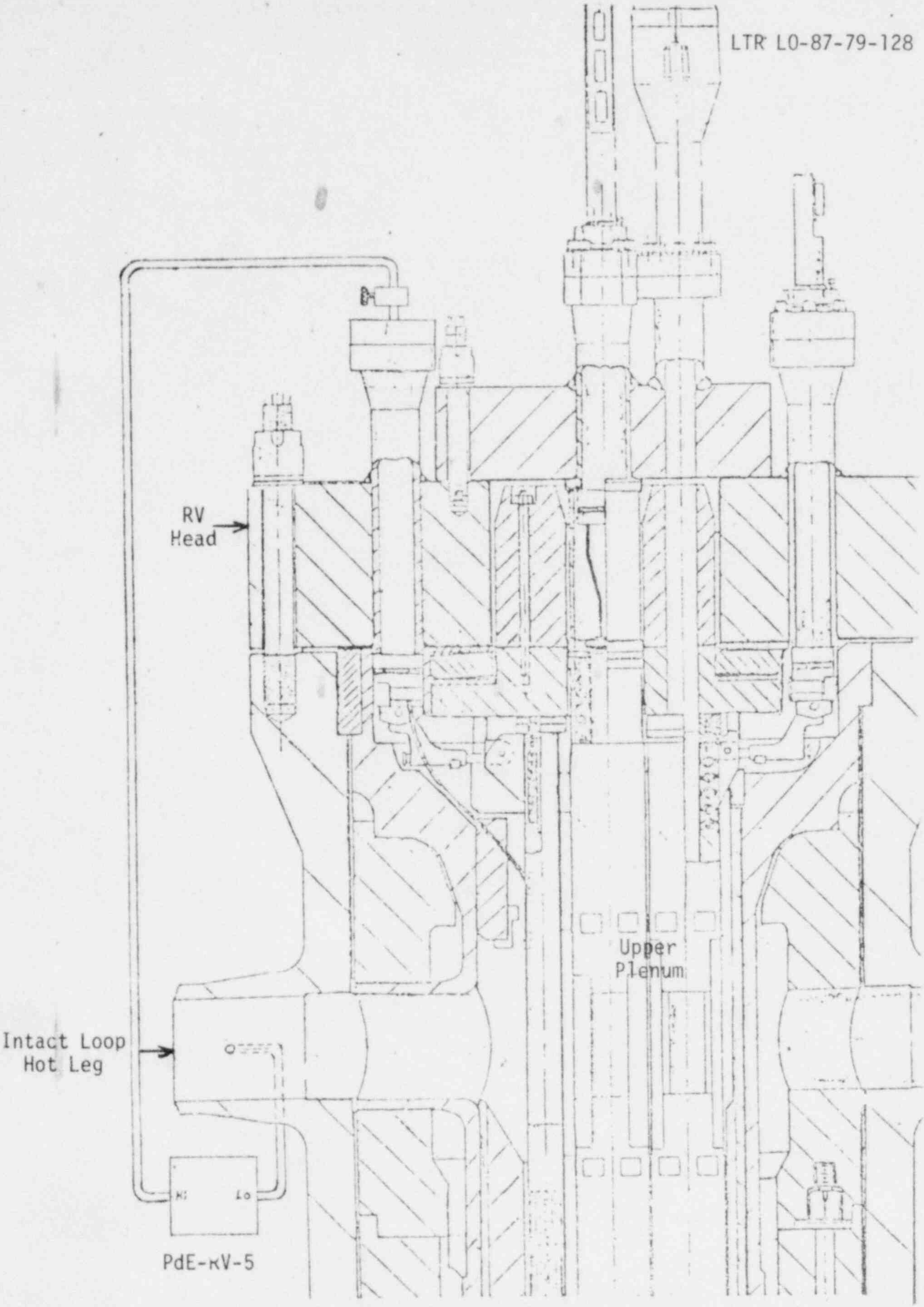
OUTPUT FOR L2-2

FIGURE 12



DATA FOR DIFFERENTIAL PRESSURE TRANSDUCER (PDE-RV-002) OUTPUT MEASURED BETWEEN THE UPPER END BOX AND THE LOWER END BOX

FIGURE 13



UPPER REACTOR AREA SHOWING CONNECTIONS OF DIFFERENTIAL PRESSURE TRANSDUCER (PDE-RV-005)

FIGURE 14

A qualitative review of the data from L3-1 indicates that the differential pressure transducer may be used as a liquid level indicator (Figure 15).

The initial reading is due to flow prior to the initiation of the LOCA. The drop in pressure is due to the flow decreasing to approximately zero within 50 seconds. After this time, the differential pressure is due only to the water level in the upper plenum. The increase in output is due to drainage of the upper entrance port. This could be alleviated by a different tap configuration on the head. The slow increase in pressure corresponds to a decrease in fluid level in the upper plenum.

The difficulties in using a differential pressure for level are:

1. Maintaining the reference leg full of water.
2. Compensating for changes in density during the blowdown.

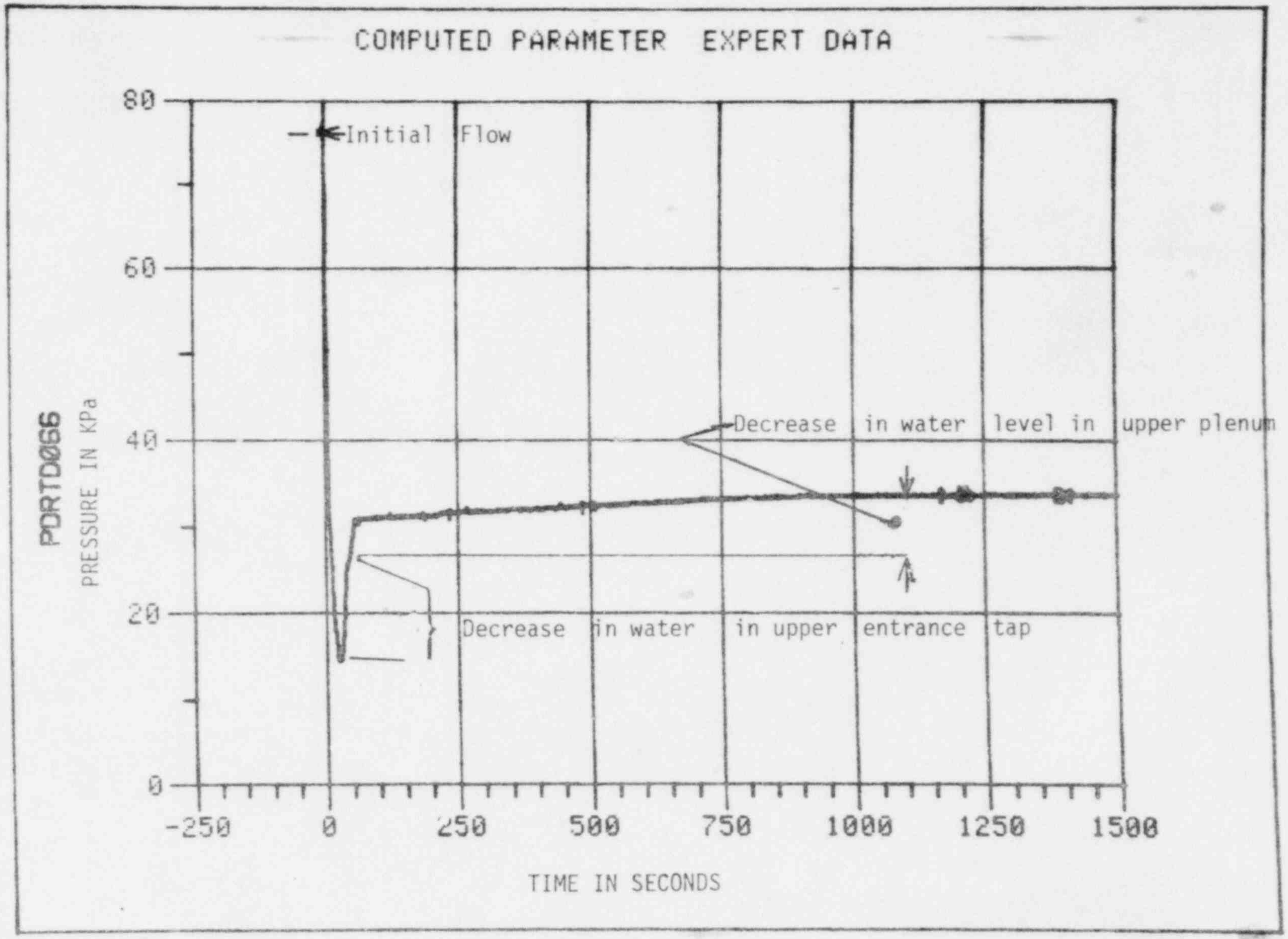
These problems are easily solved for an LPWR.

A liquid level measurement using a differential pressure is readily adaptable to LPWR's for measuring upper plenum liquid inventory. It is not adaptable to the core region unless a means of bringing the low tap (reference leg) out the side of the vessel can be obtained.

## 2.7 External Neutron Detector

### 2.7.1 Power Range Ion Chambers

The power range ion chambers (RE-T-77-2A2 and 3A2) are uncompensated ion chambers with sensitivity length of 23.5 cm and are



DIFFERENTIAL PRESSURE TRANSDUCER (PdE-RV-005)
OUTPUT MEASURED FROM TOP OF REACTOR TO OUTLET
VESSEL PIPE
FIGURE 15

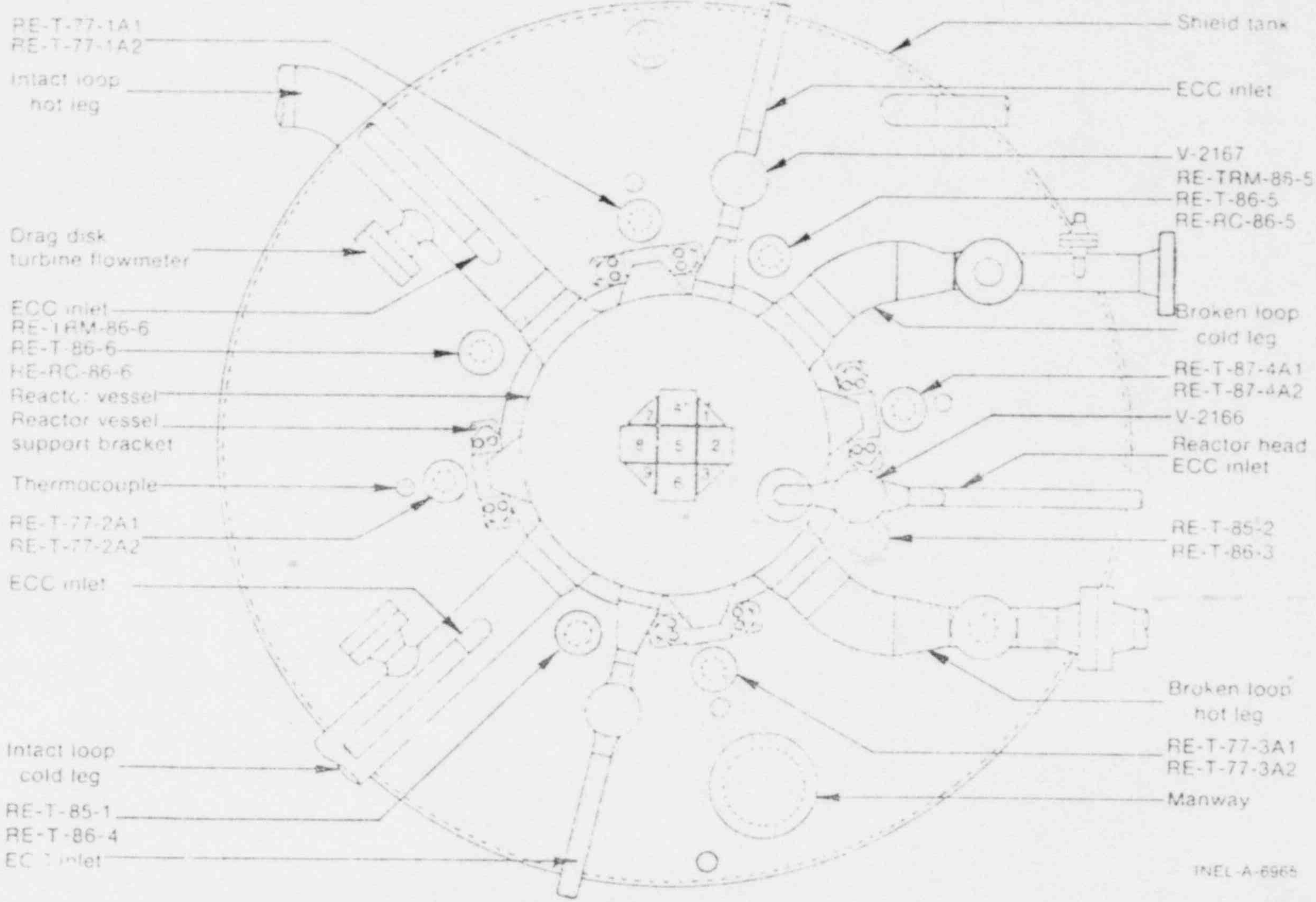
located in the shield tank (Figure 16) at a depth measured from the midplane of the fuel elements of 61 cm. This detector measures both gamma and thermal neutrons.

The basic principle used to evaluate these detectors is that the neutrons are over moderated; hence, a decrease in water density would result in an increase in the detector output due to both an increase in gamma and thermal neutron fluxes.

The ion chamber data appeared to have inflections that corresponded to events recorded by the liquid level electrodes. Until the L3-1 test, there were no baseline data, i.e. data that follow the reactivity decay without voiding the reactor. These baseline data obtained from L3-1 are assumed to represent the basic shape of the data that would have been obtained in L2-2 and L2-3 if voiding had not occurred. Therefore, a reading on L2-2 or L2-3 that lies above the L3-1 curve would represent some voiding in the reactor vessel. Figures 17 and 18 show these overlays. The magnitude of the reading should represent the amount of voiding present in the reactor vessel. This technique does not delineate between core voiding or downcomer voiding, but gives an average density of the fluid in the reactor vessel.

This technique could be applied to an LPWR by:

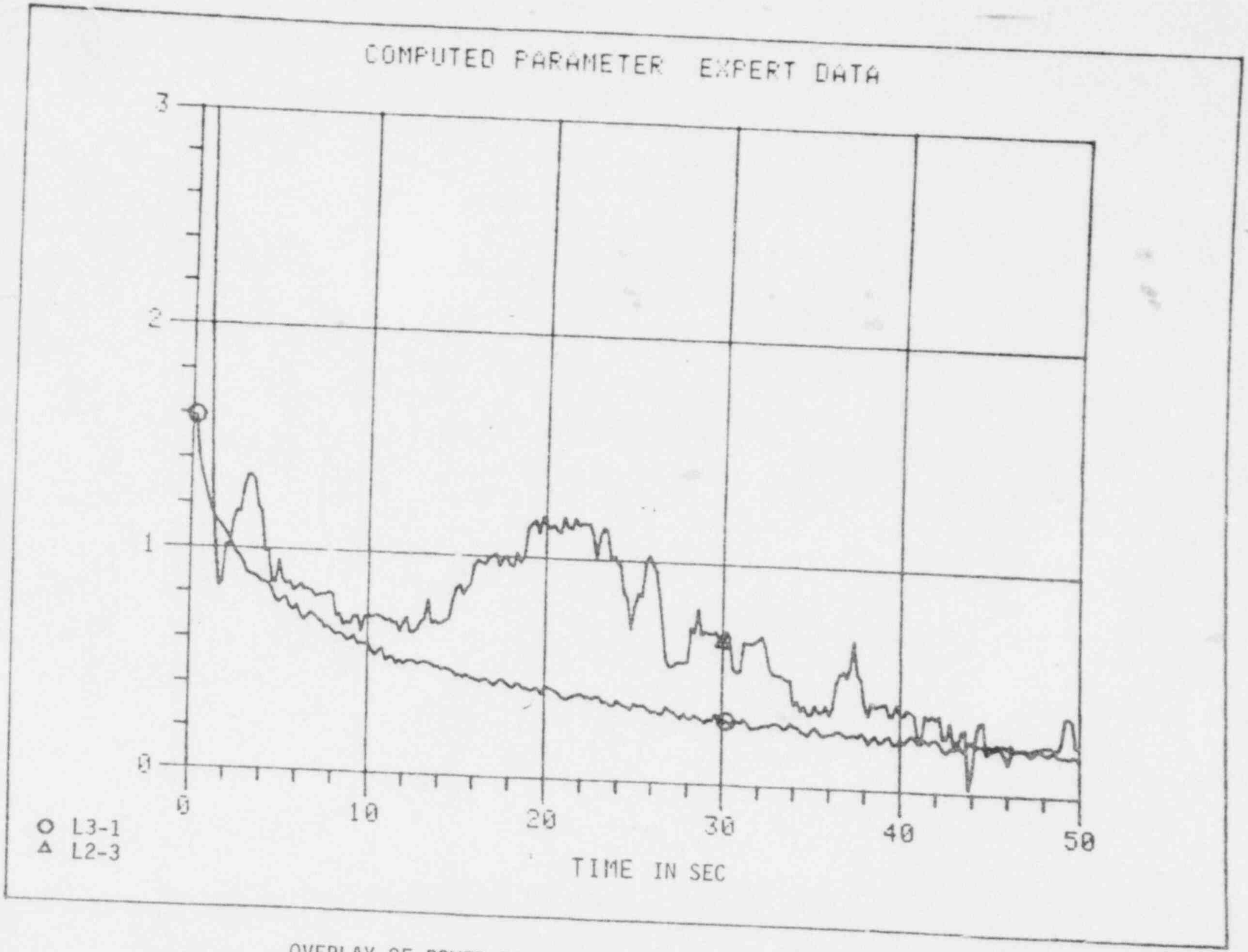
1. Obtaining a typical decay curve under all water conditions.
2. Normalizing the background curve to some point on the decay with suspected voiding.



TOP VIEW OF THE REACTOR AND SHIELD TANK

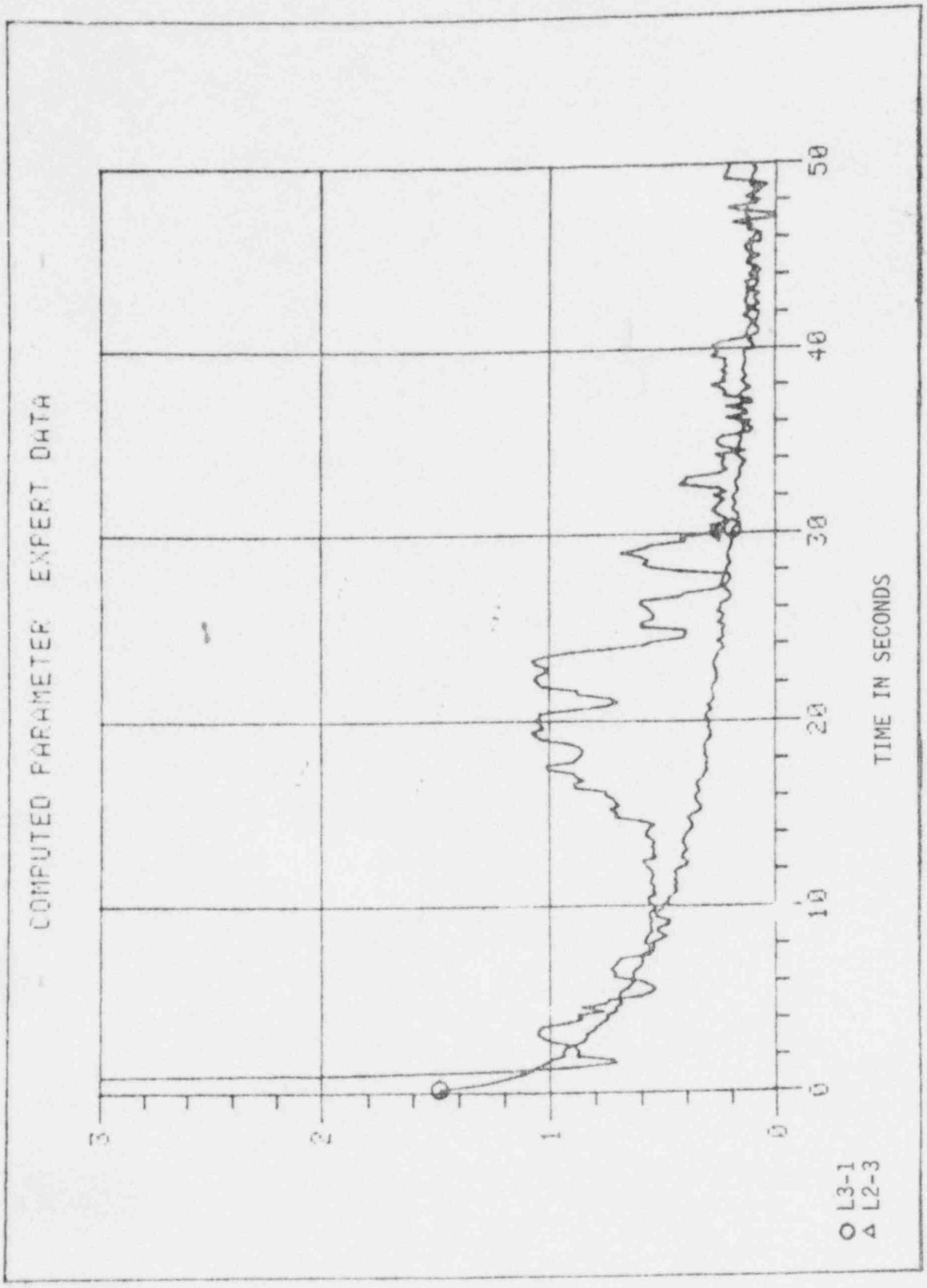
SHOWING LOCATIONS OF NUCLEAR PROCESS INSTRUMENTS

FIGURE 16



OVERLAY OF POWER RANGE ION CHAMBERS (RE-T-77-2A2)
OUTPUT FOR L2-3 and L3-1

FIGURE 17



OVERLAY OF POWER RANGE ION CHAMBER (RE-T-77-3A2)

OUTPUT FOR L2-3 and L3-1

FIGURE 18

3. Taking the difference between the two curves with a positive reading indicating voiding.

The above could be accomplished in a microprocessor and displayed to an operator.

The difficulties in using this measurement are:

1. It is not clear if the technique used in the evaluation would be applicable after long shutdown periods. A source of gammas or neutrons must be available.
2. How do you distinguish between an increase in output due to voiding and criticality?

If these questions can be successfully answered, this technique could be used to evaluate reactor vessel voiding.

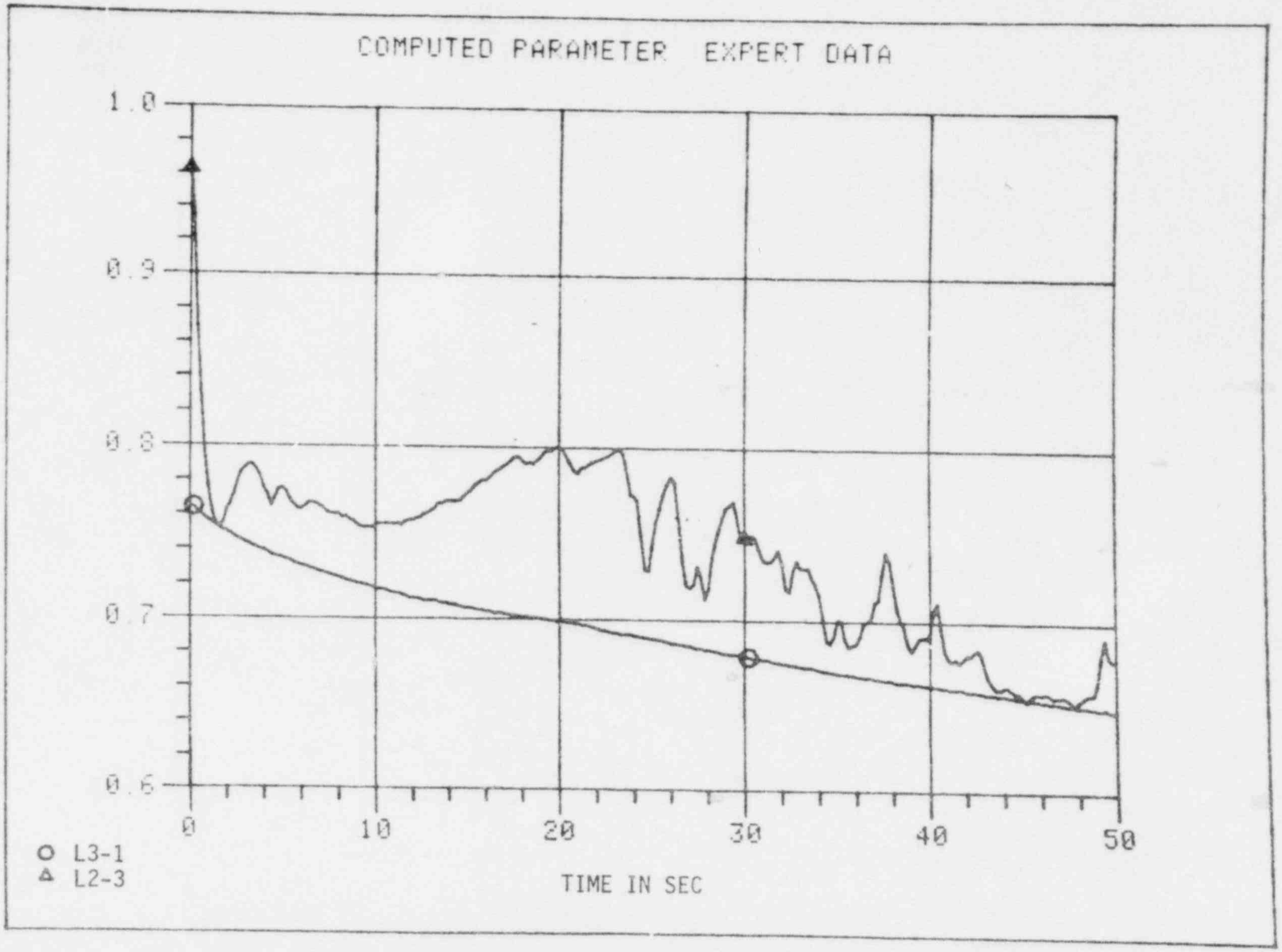
### 2.7.2 Intermediate Neutron Detectors

The intermediate range ion chambers (RE-T-86-3 and 4) are compensated ion chambers with a sensitivity length of 35.6 cm and are located per Figure 16. They are axially located at the middle of the core. The intermediate range ion chambers are compensated for gamma and measure only thermal neutrons.

The data obtained from the small break test L3-1 establishes the neutron and gamma decay baseline required to evaluate the data obtained during L2-2 and L2-3.

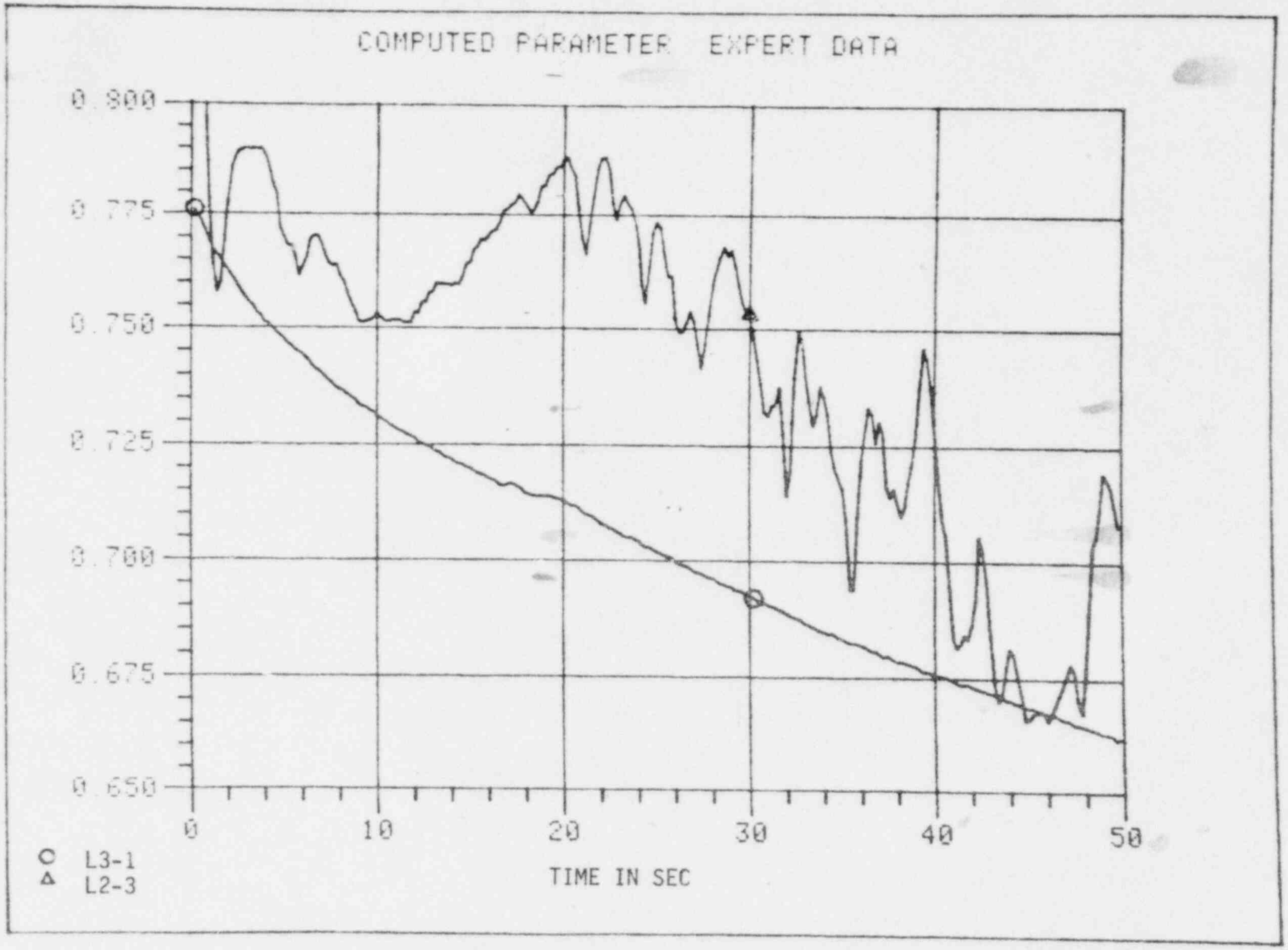
By overlaying the L2-3 data with that from L3-1, it becomes apparent that information about the fluid inventory in the reactor vessel may be obtained. This is a composite of the fluid in the core area and the downcomer area. It is believed that ion chambers in the shield tank are shielded and that voiding in the reactor will cause an increase in the ion chamber current. With this assumption, the plots for L2-3 were evaluated and are shown in Figures 19 and 20. The initial voiding indicated by the intermediate range power detectors (RE-T-86-3 and 4) agrees very well with the initial voiding of the core. The initial reflood does not agree well with the clad thermocouples or liquid level electrode data. This occurs at 6.8 seconds for clad thermocouples and 5.8 seconds for the intermediate power range detectors (RE-T-86-3 and 4). The second voiding correlates between clad thermocouples, SPNDs and the intermediate power range detectors. The second voiding, according to the liquid levels, takes place at 15 seconds (LE-5E11-8), 17 seconds (LE-5E11-9), 21 seconds (LE-5E11-11), 13 seconds (LE-5E11-15) and 9 seconds (LE-5E11-16) while RE-T-86-3 and RE-T-86-4 indicate voiding at 11 seconds.

There does appear to be a good relationship between the liquid level or fluid inventory and the data obtained from the intermediate power range detectors. The comments made in Section 2.7 on methods of applying and difficulties with power range instrumentation also apply to the intermediate range ion chamber.



OVERLAY OF INTERMEDIATE RANGE ION CHAMBER (RE-T-86-3)
OUTPUT FOR L2-3 and L3-1

FIGURE 19



OVERLAY OF INTERMEDIATE RANGE ION CHAMBER (RE-T-86-4)

OUTPUTS FOR L2-3 and L3-1

FIGURE 20

## 2.8 Heated Differential Thermocouples Liquid Level Detector (LLD)

A heated differential thermocouples liquid level detector is currently not installed in LOFT. One has been tested at PBF and shows great promise. For completeness, a brief description of this detector is given.

Each detector of the LLD consists of two thermocouples---one heated with a resistive heater and one unheated. The LLT electronics attempt to maintain a constant differential output between the two thermocouples by varying the heater power. By measuring changes in this differential output, an electronic comparator determines whether the detector is in an air (steam) or water environment. This determination then operates latching relay closures that are connected to the Data Acquisition System. The response time for this measurement system is approximately 0.25 s from a gas-to-liquid change and 0.5 s for a liquid-to-gas change.

In addition to the wet/dry detector and heater control functions, the electronics automatically detect thermocouple open heater failures. This is done by measuring the power level to the heater. If the current drops below a certain preset value, but the voltage is not reduced correspondingly, a relay contact is closed indicating an open (broken) heater.

This method has been used at PBF (Power Burst Facility) and indications to date suggest that the data obtained in this manner are quite reliable. A report in greater detail is presently being prepared by TFBP. This detector may be installed in existing wells in the core, thereby making it attractive for both retrofit and new installation.

## 3.0 CONCLUSIONS AND RECOMMENDATIONS

The results of this analysis indicate that qualitative liquid level information may be obtained from instruments other than the liquid level transducers. It is not clear whether this conclusion applies for time frames greater than the 50 seconds that were examined. Conclusions and recommendations for the measurements examined follow.

The clad thermocouples indicate the initial dryout correctly, but may not indicate the presence of water correctly during rewet. This is due to the fact that the continual heat source from the fuel rod could cause a vapor barrier to form; hence, the presence of water would not be indicated at the correct time.

Fluid thermocouples in the upper plenum indicate similar results as the clad thermocouples, but they cannot differentiate between saturated water and steam.

It is apparent that the LOFT SPNDs are sensitive to core voiding and subsequent rewet. It may not be possible to determine voiding from criticality for any of the neutron instruments.

Differential pressure transducers can be used for liquid level, if core installation is taken to prevent transmission lines (reference leg) from blowing down. An easy installation is the outlet nozzle to upper plenum currently in use at LOFT. This measurement was successfully used during L3-1. A means of tracking instrument drift, e.g. zero offset, must be devised for any differential pressure installation.

The external neutron detectors indicate voiding correctly. It should be noted that the voiding seen applies both to the core and downcomer regions. If one knows the basic shape of the decay curves for the reactor, then information can be obtained. Real time readout should be possible. The ability of both the external neutron detectors and SPNDs to successfully detect voiding after periods >50 seconds was not proven and needs to be substantiated.

The heated differential thermocouple that was used in PBF appears to have promise. It may be the most logical transducer to use for either new installation or retrofit for LPWRs.

The LOFT conductivity liquid level transducer performs satisfactorily, but problems with reliability and real time data interpretation make it a poor choice for LPWRs.

The following recommendations are made:

1. Investigate the performance of neutron measurements (both internal and external) after the reactor is shut down for more than one minute. (The only data that may be available is from TMI.)

2. Methods for real time display of thermocouple and neutron detector liquid level information needs to be investigated.

3. Evaluate possible methods of determining the difference between voiding and criticality when neutron measurements are used for liquid level. Reactor noise analysis might give the answer.

4. More experience should be gained with the heated differential thermocouple.

## REFERENCES

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# APPENDIX A

Assume a one energy group, infinite, homogeneous reactor model:

$$\phi_\gamma \mu_a = S(t)$$

where $\phi_\gamma$ is the local gamma flux resulting from a source term S (t) which is a function of time. The total linear absorption coefficient is $\mu_a$.

## I. FIRST DERIVE AN EXPRESSION FOR $\mu_a$

LOFT Core Characteristics (References 1 and 2)

Volume of fuel pins = 6.945 ft$^3$

Volume of guide tubes = 0.148 ft$^3$

Volume of grids = 0.098 ft$^3$

Volume of water spaces = 10.300 ft$^3$

Since the fuel pins represent 97% of the non-water volume of the core, assume the core is comprised entirely of fuel pins and water.

$$\mu_{ai} = \sum_i \rho_i \sigma_{ai}$$

Using the data from Evans (Reference 3) and assuming the average energy of the gamma flux $E_\gamma$ is 1.6 Mev (References 4 and 5), the resulting $\mu_a$ for the homogenized LOFT core (using volume averaged absorption coefficients) is:

$$\mu_a = \frac{1.150 + 6.56\ \mu_{H_2O}}{11.140} \quad cm^{-1}$$

where $\mu_{H_2O} = (\rho_{H_2O})\ (0.028\ cm^2/gm)$

at T = 550°F (saturation temperature at 6.4 sec for L2-3 obtained from the output of TE-5L8-039).

$\rho_{H_2O}$ (liquid) = 0.736 $gm/cm^3$

$\rho_{H_2O}$ (vapor) = 0.0379 $gm/cm^3$

therefore

$\mu_a$ (liquid) = 0.1154 $cm^{-1}$

$\mu_a$ (vapor) = 0.1039 $cm^{-1}$

## II. ASSUME S(t) ≅ CONST. OVER TIME FRAME OF REWET (t < 1 SEC)

$$\Phi_{\gamma i}\ \mu_{ai} = \Phi_{\gamma f}\ \mu_{af}$$

where i denotes conditions just prior to rewet and f denotes conditions just subsequent to rewet.

Assume a saturated vapor moderator prior to rewet. Assume a saturated liquid moderator subsequent to rewet

$$\frac{\Phi_{\gamma i}}{\Phi_{\gamma f}} = \frac{\mu_{af}}{\mu_{ai}} = 1.111 \qquad \Phi_{\gamma f} = \frac{\Phi_{\gamma i}}{1.111}$$

where $I_{S\gamma}$ is the SPND gamma current and $S_\gamma$ is the SPND gamma sensitivity.

$I_S = I_{S\gamma} + I_{Sn}$ ($I_{Sn}$ is the SPND neutron current)

$I_{Sn} = I_{S\gamma i} + I_{Sni}$ (Assume $I_{Sni} \approx 0.0$)

$I_{Si} = I_{S\gamma i} = -47.8\ \eta$ Amp. (for L2-3 5D8 SPND)

$I_{Sf} = I_{S\gamma f} + I_{Snf}$

but $\dfrac{I_{S\gamma f}}{I_{S\gamma i}} = \dfrac{\Phi_{\gamma f}}{\Phi_{\gamma i}}$

or $I_{S\gamma f} = I_{S\gamma i} \left(\dfrac{\Phi_{\gamma f}}{\Phi_{\gamma i}}\right) = \dfrac{-47.8}{1.111} = -43.02\ \eta$ Amp

$I_{Snf} = I_{Sf} - I_{S\gamma f}$

$I_{Sf} = -38.2\ \eta$ Amp (for L2-3 SPND 5D8)

$= -38.2\ \eta$ Amp $+ 43.02\ \eta$ Amp

$= 4.82$ of $\eta$ Amp

$I_{Snf} = S_n \Phi_{nf}$

Assume $\Phi_{nf} = \left[\Phi_n\ (t < 0)\right] \left[\beta_{eff}\right] \left[e^{-\lambda t}_{eff}\right] [n]$

where n = neutron multiplication factor

$$n = \frac{I_{Snf}}{S_n\ \Phi_n\ (t < 0)\ \beta_{eff}\ e^{-\lambda t}}$$

$S_n = 3.49 \times 10^{-21}$ A -Cm$^2$-s/n (see Appendix B)

$\Phi_n$ (t   0) $= 6.299 \times 10^{13}$ n/cm$^2$ S (see Reference 7)

$\beta_{eff} = 7.259 \times 10^{-3}$ (see Reference 1)

and, using the data from Reference 1,

$e^{-\lambda t}_{eff} = 0.2155$ for $t = 6.4$ sec

Using the above data, $n = 14.0$

Repeating the above calculation for L2-2 and for the other core location yields the results of Table I.

## APPENDIX B

### Calculation of Revised Value for $S_n$

From Reference 6.

$S_n = 6.6 \times 10^{-21}$  A - Cm$^2$ sec/n

$s_\gamma = 1 \times 10^{-16}$  A - $\mu$/R

$$K = \frac{\phi_{\gamma p}}{\phi_n} = 5.37 \times 10^{-6} \text{ R-Cm}^2\text{-s/n}^2$$

= ratio of prompt $\gamma$ flux to prompt neutron flux

$$K1 = \frac{\phi_{\gamma d}}{\phi_{\gamma p}} = 0.86 \text{ for 2000 hr}$$

= ratio of delayed $\gamma$ flux to prompt $\gamma$ flux

Assuming an exponential buildup of delayed gamma flux as a function of irradiation time

i.e., $\phi_{\gamma d} = \phi_{\gamma p} (1 - e^{-Xt})$

where x is determined to be $9.83 \times 10^{-4}$ hr

from K1 = 0.86 for t = 2000 hr. (Ref. 6)

for L2-3  t = 67 hr. (50 EFPH at 75% power)

$$\left.\begin{aligned} K1 &= 6.37 \times 10^{-2} \\ \phi_n &= 6.30 \times 10^{13} \text{ n/cm}^2 \text{ - sec} \\ \phi_{\gamma p} &= K\phi_n = 3.38 \times 10^{8} \text{ R/h} \\ \phi_{\gamma d} &= K1\phi_{\gamma p} = 2.16 \times 10^{7} \text{ R/h} \end{aligned}\right\} \quad \text{at 5D8 SPND location (Ref. 7)}$$

$$I_n = S_n \phi_n = 4.16 \times 10^{-7} \text{ Amp}$$

$$I_p = S_\gamma \phi_{\gamma p} = 3.38 \times 10^{-8} \text{ Amp}$$

$$I_d = S_\gamma \phi_{\gamma d} = 2.16 \times 10^{-9} \text{ Amp}$$

Calculated $I_{TC} = 4.52 \times 10^{-7}$ Amp

but measured $I_{Tm} = 2.24 \times 10^{-7}$ Amp

$$\frac{I_{TC}}{I_{Tu}} = 2.01$$

Repeating the calculation for SPND's 2, 4, 6 location and both locations for L2-6 yields

| Test No. | SPND No. | $I_{TC}/I_{Tm}$ |
|---|---|---|
| L2-2 | 2, 4, 6 | 1.85 |
| L2-2 | 5 | 1.79 |
| L2-3 | 2, 4, 6 | 1.90 |
| L2-3 | 5 | 2.01 |

The mean value is 1.89 ± 5% (1σ)

Scale $S_n$ by

$$S_n' = \frac{S_n}{\frac{I_{Tc}}{I_{Tm}}} = \frac{6.6 \times 10^{-21}\ \text{A-Cm}^2\text{-sa/n}}{1.89}$$

$S_n' = 3.49 \times 10^{-21}$ A-Cm$^2$-s/n ± 11%

$S_\gamma' = 5.29 \times 10^{-16}$ A-h/R ± 8%.

## APPENDIX REFERENCES

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