ornl

OAK
RIDGE
NATIONAL
LABORATORY

UNION
CARBIDE

NUREG/CR-2204, Vol. 3
ORNL/TM-8162

Advanced Two-Phase Flow Instrumentation Program Quarterly Progress Report for July–September 1981

J. E. Hardy
G. N. Miller
S. C. Rogers
W. L. Zabriskie

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

H003
s
0/1

OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

Available from
GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG/CR-2204, Vol. 3
ORNL/TM-8162
Dist. Category R2

Contract No. W-7405-eng-26

Engineering Technology Division

ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JULY-SEPTEMBER 1981

J. E. Hardy   S. C. Rogers
G. N. Miller   W. L. Zabriskie

Manuscript Completed — December 17, 1981
Date Published — January 1982

NOTICE This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

NRC FIN No. B0401

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

CONTENTS

ADVANCED TWO-PHASE FLOW INSTRUMENTATION PROGRAM QUARTERLY PROGRESS REPORT FOR JULY—SEPTEMBER 1981

J. E. Hardy S. C. Rogers*
G. N. Miller* W. L. Zabriskie*

ABSTRACT

The performance of the Westinghouse Reactor Vessel Level Indicating System (RVLIS) in the S-UT-3 test (a communicative break in the cold leg of Semiscale) was analyzed. The Westinghouse RVLIS gave similar indications to Semiscale Test Facility instrumentation measuring the same phenomena [differential pressure (dP)] over equal spans. The Westinghouse measurement is apparently conservative when compared with the two-phase froth level. These dP measurements appear to be nonconservative estimates of level, however, when the measurement system spans the upper core support plate. Level measurement errors of up to 150 cm (60 in.) were observed during S-UT-3. Westinghouse claims that these differences are caused by differences between Semiscale and Westinghouse Reactors. A recommendation for resolving these differences is made.

---

1. INTRODUCTION

During the accident at the Three Mile Island (TMI) nuclear power plant, a condition of low water level in the reactor vessel and inadequate core cooling was not recognized for a long period. A review of the accident was conducted by the U.S. Nuclear Regulatory Commission (NRC) TMI-2 Lessons Learned Task Force.[1] Their report recommended that improved instrumentation systems, including reactor-vessel liquid level (coolant) sensors, be developed and implemented in all pressurized-water reactors (PWRs) in the United States.

For this purpose, as part of the NRC Action Plan[2] following the TMI accident, the Advanced Two-Phase Flow Instrumentation Program at Oak Ridge National Laboratory (ORNL) will evaluate instrumentation systems through funding from the NRC Division of Reactor Safety Research. The coolant sensors are intended to provide an unambiguous indication of the adequacy of core cooling. They must survive accident conditions and work under both natural- and forced-convection flow conditions. As part of this effort, two types of sensors were evaluated concurrently: (1) thermal-type sensors, such as heated junction thermocouples (HJTCs), and (2) ultrasonic torsional wave sensors. Test sensors were designed and fabricated at ORNL and procured from outside sources. A variety of tests were run to evaluate these devices for power reactor use.

---

*Instrumentation and Controls Division.

The experiments simulated thermal and hydraulic conditions typical of a postulated PWR loss-of-coolant accident (LOCA); both natural-convection (reactor coolant pumps off) and forced-convection (pumps on) two-phase flow tests have been run. The goals of these experiments were to evaluate the designs of the coolant sensors and to determine whether conditions exist under which ambiguous readings might occur. Generally, the test sequence for a particular device proceeded from static (covered and uncovered) tests at room temperature to static tests in saturated steam and water at elevated pressures to forced-convection tests at relatively severe pressure, temperature, and flow rate conditions.

Previous testing[3-5] has included experiments with several thermal-type level devices in a high-pressure and high-temperature natural-convection facility (a pressurizer). A low-pressure steam and water flow visualization test was performed with thermal sensors, and tests were run with an HJTC in the Advanced Instrumentation for Reflood Studies Test Stand, a cocurrent steam and water flow facility rated for moderate pressures. Additional studies of HJTC liquid level instrumentation were run in the Thermal-Hydraulic Test Facility (THTF) at ORNL in conjunction with THTF Test 3.09.10 (Small-Break LOCA Test Series).[6] An interim report was prepared for quasi-steady-state film-boiling THTF Test 3.07.9 (Ref. 7).

Subsequently, an HJTC was tested in an air and water facility, which is used to simulate flow phenomena in the core and upper-plenum interface region of a PWR. Results showed appreciable cooling of the HJTC at high void fractions.[8] An additional test measured the cooling effect of mist flows on a shielded HJTC. Detectable cooling was also observed for many mist-flow conditions.[8]

A multiple-position HJTC was designed, fabricated, and tested under various temperatures and pressures up to 10.8 MPa (~1550 psia) and 315°C (600°F). Excellent results were obtained from the four-level stations on the probe in steam and water natural circulation tests.[8]

A torsional ultrasonic level probe was developed and tested in a steam and water pressurizer.[5,6] Data analysis showed that the ultrasonic level measurements were within ±5% of the actual level over a temperature range from 28 to 230°C (82 to 450°F). A Hewlett-Packard HP85 calculator was interfaced with the data acquisition system to improve data quality. Conceptual designs for a pressure seal technique in an operating reactor for the ultrasonic level device were discussed with a reactor vendor.

A measurement system was developed by Westinghouse to monitor in-vessel coolant levels by means of differential pressure (dP) cells covering selected ranges. This system, Reactor Vessel Level Indicating System (RVLIS), and its basic requirements and components were described previously.[9] Evaluation of the performance of the RVLIS was completed for one of the tests in the upper-head injection series (S-UT-3) at the Semiscale Facility, EG&G Idaho, and is contained in this report.

2. EVALUATION OF THE WESTINGHOUSE RVLIS

The Westinghouse RVLIS monitors the in-vessel liquid level by means of dP measurements. This system was installed in the Semiscale Test Facility for Test S-UT-3, a 2-1/2% communicative break in the cold leg of the facility, to compare performance with installed test vessel instrumentation, such as densitometers and other dP sensors.

The NRC requirement for a liquid level measurement system is to provide reliable and unambiguous information as to the water level in a reactor vessel. The effectiveness of the Semiscale testing in meeting this criterion may be judged by comparison of the RVLIS data with test data taken from a combination of test vessel densitometers, flow measurements, fluid temperatures, and dP measurements covering selected ranges.

EG&G has reported the data from this test,[10] and the plots in this report are taken from the same source as the EG&G report (NRC Data Bank). Only the data that are considered pertinent are included here; to facilitate comparison with the other test data, the same nomenclature used in the EG&G report is used.

2.1 Analysis

Figure 1 is a schematic diagram of Semiscale reproduced here from EGG-SEMI-5494, June 1981.[10] The nomenclature used to describe the test measurements are included in Fig. 1. Note that all height measurements refer to the centerline of the cold leg; both negative and positive measurements refer to distances above and below the cold leg.

Figure 2 is a plot of Westinghouse upper-head measurement and Semiscale dP measurement from +421 to −13 cm and illustrates the dP measurements during the test. Both of these measurements span the upper core support plate. The level indicated by the densitometer is plotted on this same figure as a cross with the direction of the last liquid-to-vapor transition indicated by the arrowhead. These measurements indicate that the actual level was 50 to 125 cm lower than both dP measurements. This occurred from 250 to 390 s and from about 675 to 910 s. These time periods will be discussed in the next paragraph.

Figure 3 is a plot of Westinghouse vessel level and the Semiscale dP measurement from −578 to −13 cm. Figure 4 is an expanded plot of the Semiscale dP measurement from Fig. 3 and the difference between the Westinghouse and Semiscale dP measurements with the Semiscale measurement taken as reference. The region of concern is again from 250 to 390 s and from 675 to 910 s. During these times, the densitometers indicated that the froth level was below the upper tap of the Semiscale measurement, so the two dP readings should have been equivalent. However, the Semiscale measurement did not span the upper core support plate, and the Westinghouse level measurement read as much as 150 cm (60 in.) higher than the Semiscale measurement. Since the flow in the core is low after about 130 s, the frictional dP should be low, so the Semiscale reading should be correct. The discussion of Fig. 5 will verify the validity of the Semiscale reading.

Figure 5 is a plot of void fraction based on the densitometer readings in the Semiscale vessel at two different times, including the time

ORNL-DWG 81-17873

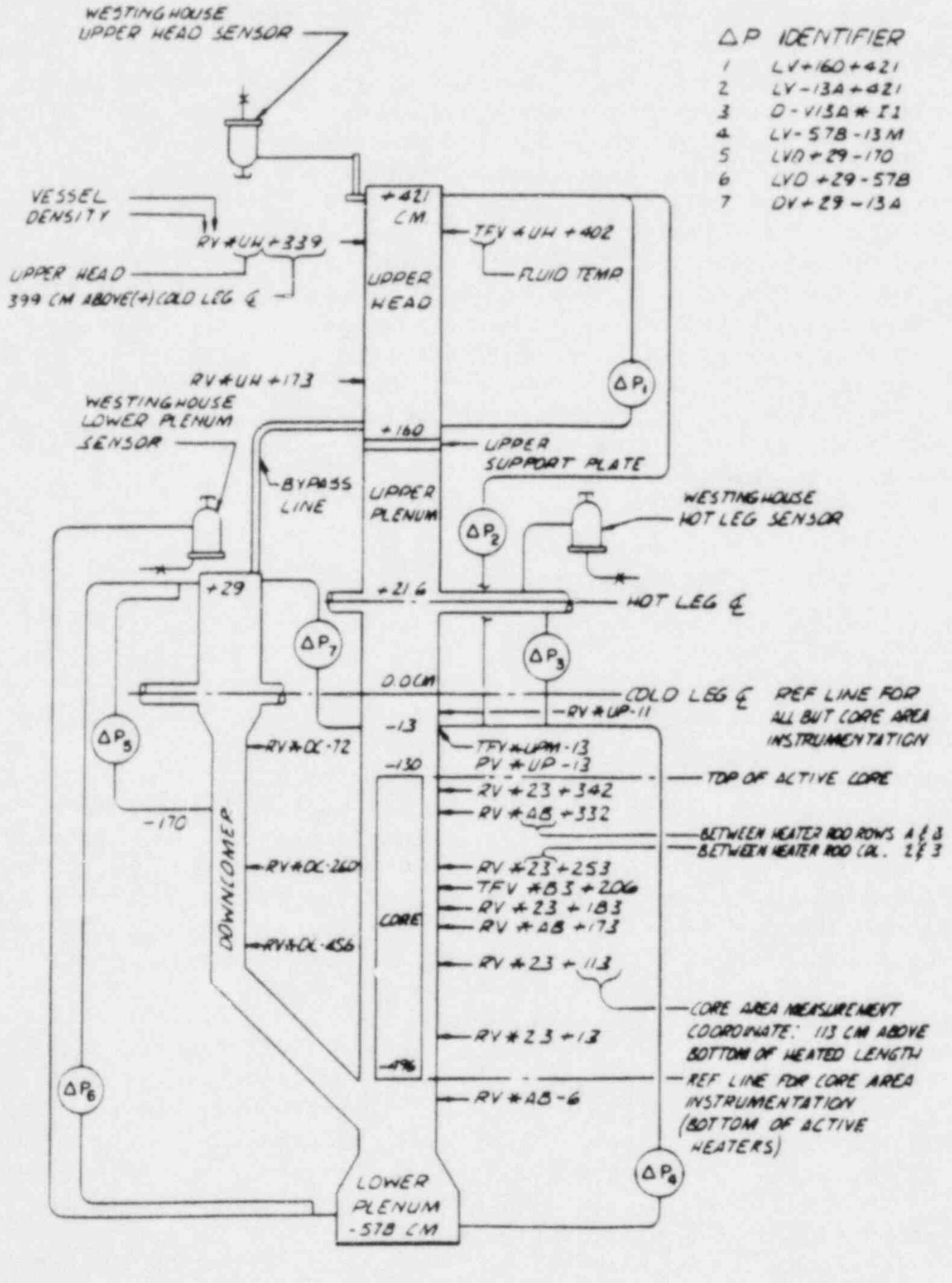


Fig. 1. Instrumentation of significance for the Westinghouse vessel liquid level indicating system test. Source: W. W. Tingle and R. W. Golden, *Installation and Initial Test Data Report: Westinghouse Reactor Vessel Level Indicating System Performance During Semiscale Test S-UT-3*, EGG-SEMI-5494 (June 1981).

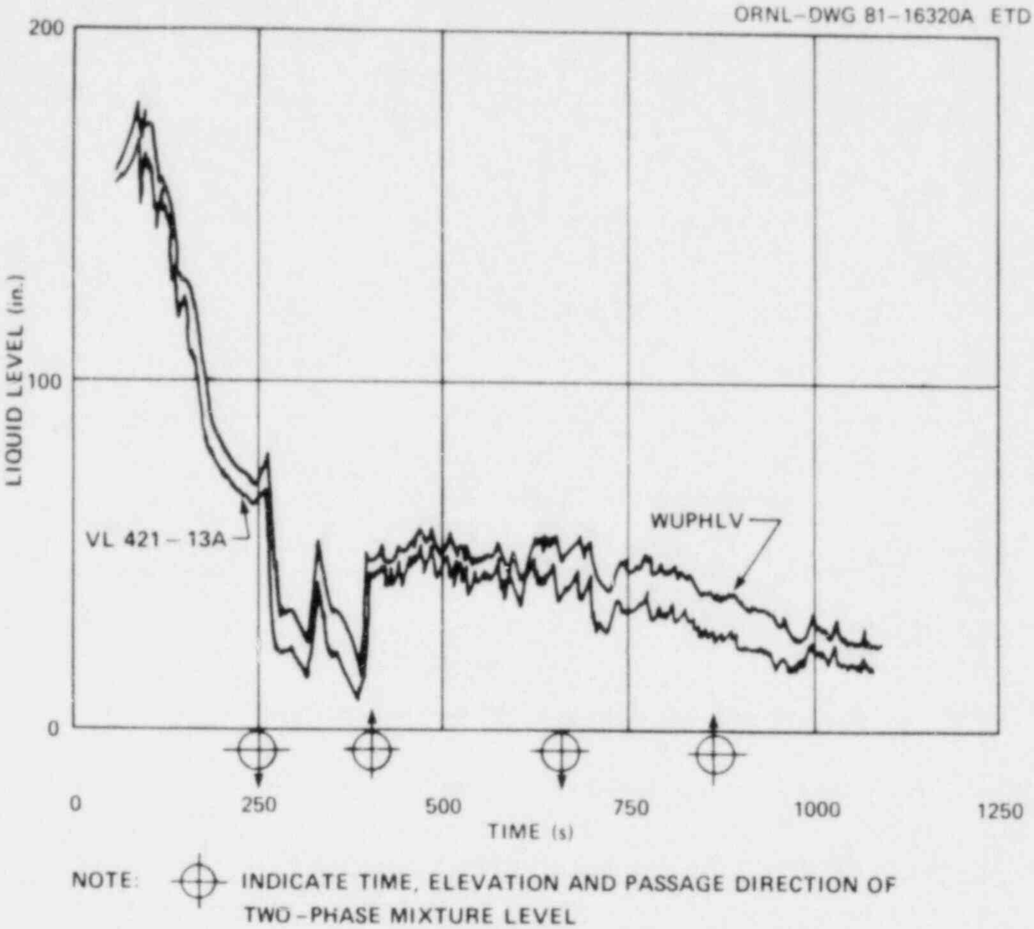


Fig. 2. Westinghouse and Semiscale upper-head and upper-plenum dP measurements. Source: W. W. Tingle and R. W. Golden, *Installation and Initial Test Data Report: Westinghouse Reactor Vessel Level Indicating System Performance During Semiscale Test S-UT-3*, EGG-SEMI-5494 (June 1981).

spans of concern, 250 to 390 s and 675 to 910 s. It was interesting that this curve turned out to be linear. From this curve the distributed void fraction can be calculated (assuming linearity between points) and a collapsed liquid level calculated. This collapsed liquid level was calculated to be ~200 cm (80 in.) below the cold leg during the region of concern, which confirms the level indicated in Fig. 3 by the Semiscale dP measurement.

The next two figures illustrate that the Westinghouse dP measurements are equivalent to Semiscale dP measurements when they cover the same span. Figures 6 and 7 are plots of differences between Westinghouse and Semiscale dP measurements. The Semiscale measurements are the sum of $\Delta P2$ and $\Delta P4$ illustrated in Fig. 1. This measurement spans the same levels as the Westinghouse measurements. Figure 6 is a plot of the differences between Westinghouse dynamic dP measurement and the Semiscale data, and Fig. 7 is a plot of the differences between the Westinghouse vessel level measurement and the Semiscale data. Neither plot illustrates an error or

ORNL-DWG 81-16321A ETD

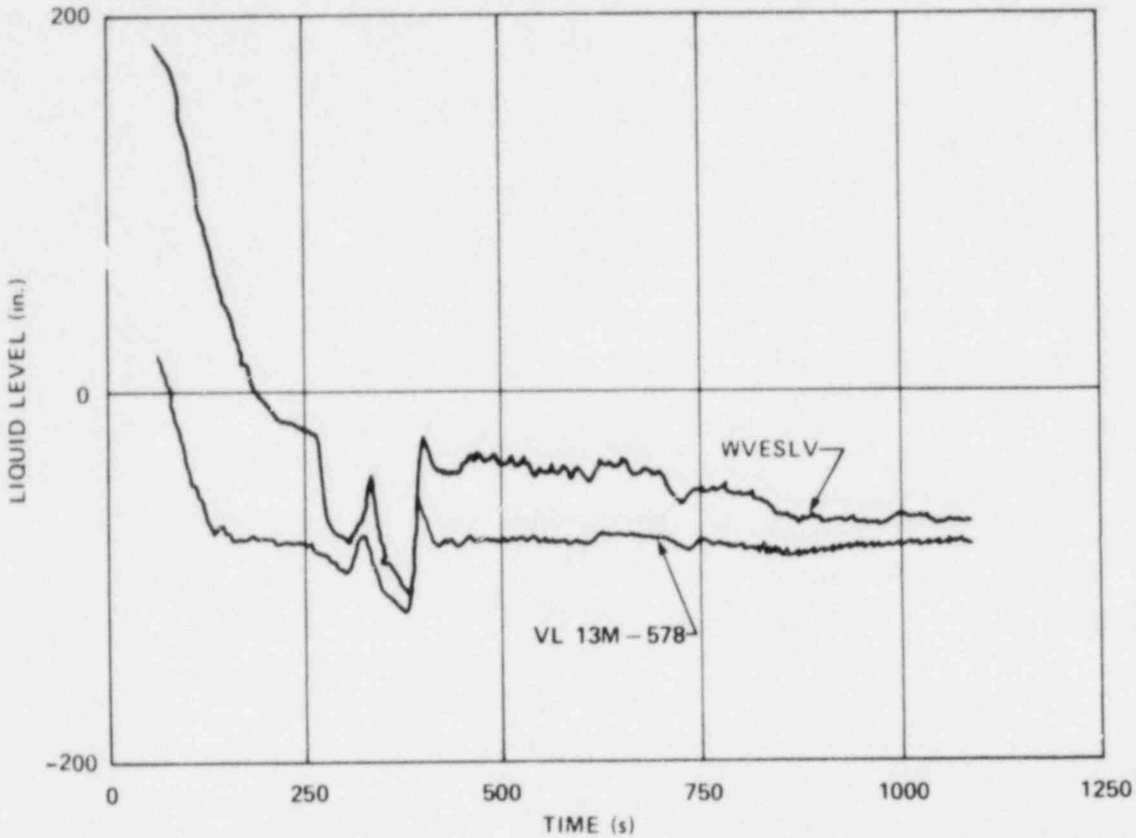


Fig. 3. Westinghouse collapsed liquid level (upper head to lower plenum) compared with Semiscale collapsed liquid level (-13 cm to lower plenum). Source: W. W. Tingle and R. W. Golden, *Installation and Initial Test Data Report: Westinghouse Reactor Vessel Level Indicating System Performance During Semiscale Test S-UT-3*, EGG-SEMI-5494 (June 1981).

difference greater than 25 cm (10 in.) after about 130 s. Some transient errors greater than 25 cm are caused by a difference in response time between the Westinghouse and Semiscale measurement systems. The Westinghouse measurement should be slower because of the long line lengths [45, 60, and 75 m (150, 200, and 250 ft)], whereas the Semiscale line lengths are only a few centimeters.

The response time can be calculated using the equation[11]

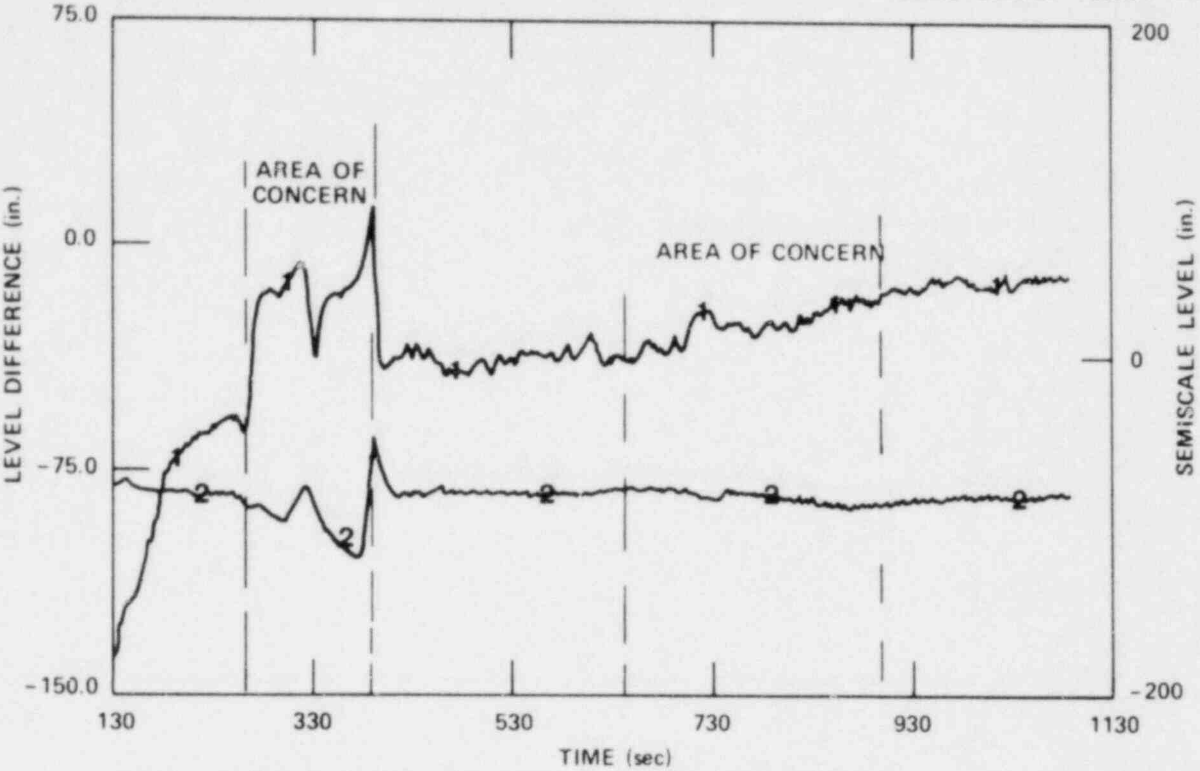
$$f = 4.22 \left(\frac{D^2}{L} \frac{dP}{dv}\right)^{1/2} ,$$

where

f = 3-db frequency response in hertz,

4.22 = constant in the equation to take care of unit conversion,

ORNL–DWG 81–16322A ETD



NOTE: CURVE 1 – [(VL 13 M-578) – WVESLV] = DIFFERENCE
CURVE 2 – VL 13 M-578

Fig. 4. Collapsed liquid levels in Semiscale as measured by dP4 and Westinghouse vessel level system. (Expanded plot of Fig. 3 with difference between two traces plotted as trace 1).

D = inside diameter of the tube in inches,
L = length of the tube in feet,
dp/dv = compliance of the dP transmitter ITT Barton Model 752 supplied by vendor to be $\frac{1000 \text{ in. of water}}{0.02 \text{ in.}^3}$; units are in $\frac{\text{lb/in.}^2}{\text{in.}^3}$.

Calculating the frequency response of the Westinghouse level monitoring system yields

$$f = 4.22\left(\frac{8.1 \times 10^{-3} \times 36.13}{400 \times 0.02}\right)^{1/2} = 0.81 \text{ Hz} .$$

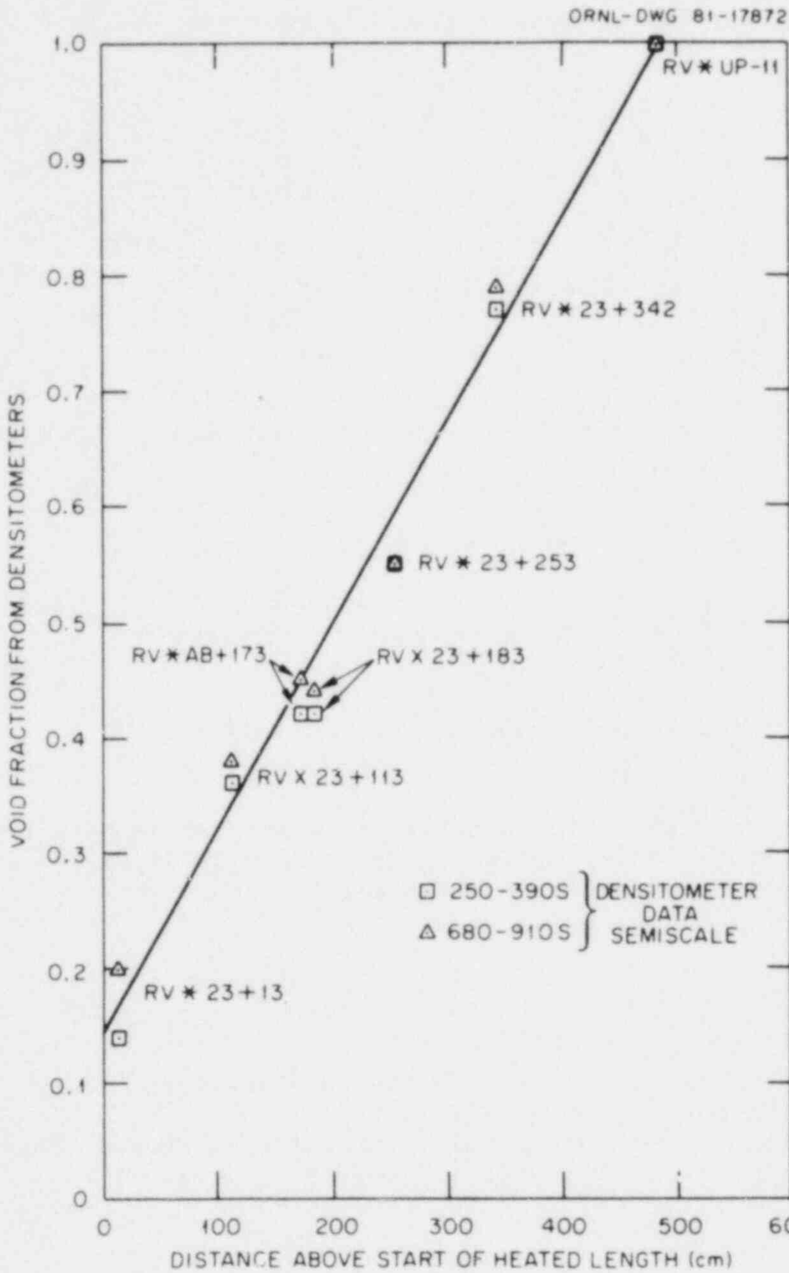


Fig. 5. Plot of void fraction based on densitometers.

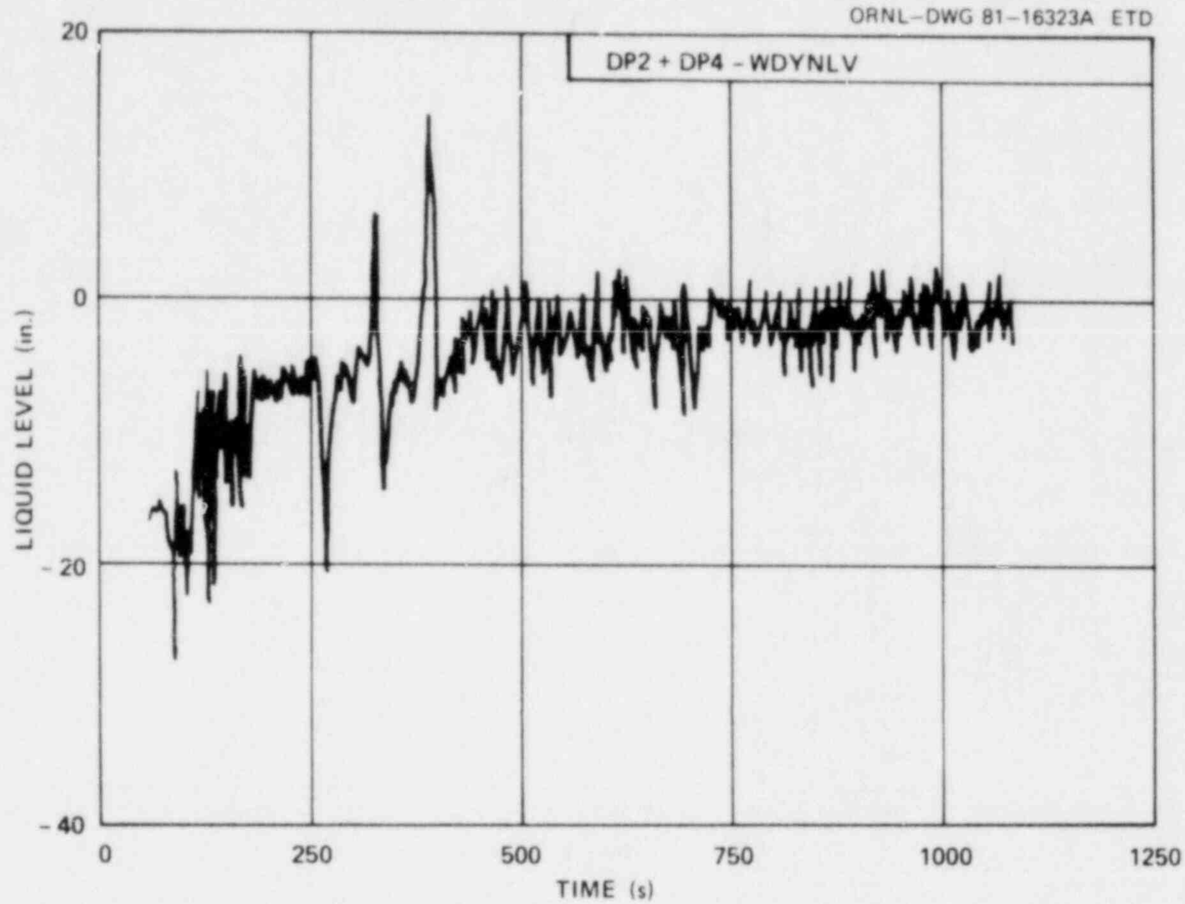


Fig. 6. Plot of difference between Westinghouse dynamic measurements and Semiscale measurements. (For entire vessel — note good agreement.)

Rise time is 2.2 (1/f) = 2.2 x 1.0/0.81 = 2.73 s. This time constant corresponds to the transient times observed in Figs. 6 and 7.

A comparison of the Westinghouse RVLIS with two-phase froth level is shown in Fig. 8. Considering the time response of the Westinghouse system, the level given by that system is a conservative indication of froth height and possible core cooling. Approximate void values are labeled where possible. These void fractions are from Semiscale densitometers.

The Westinghouse algorithm for calculating liquid level is:

$$L = \frac{H(\rho_r - \rho_v) - (10^5 \times dP_m/9.80665)}{(\rho_\ell - \rho_v)} + D ,$$

where

L = liquid level in centimeters,
H = elevation between taps in centimeters,

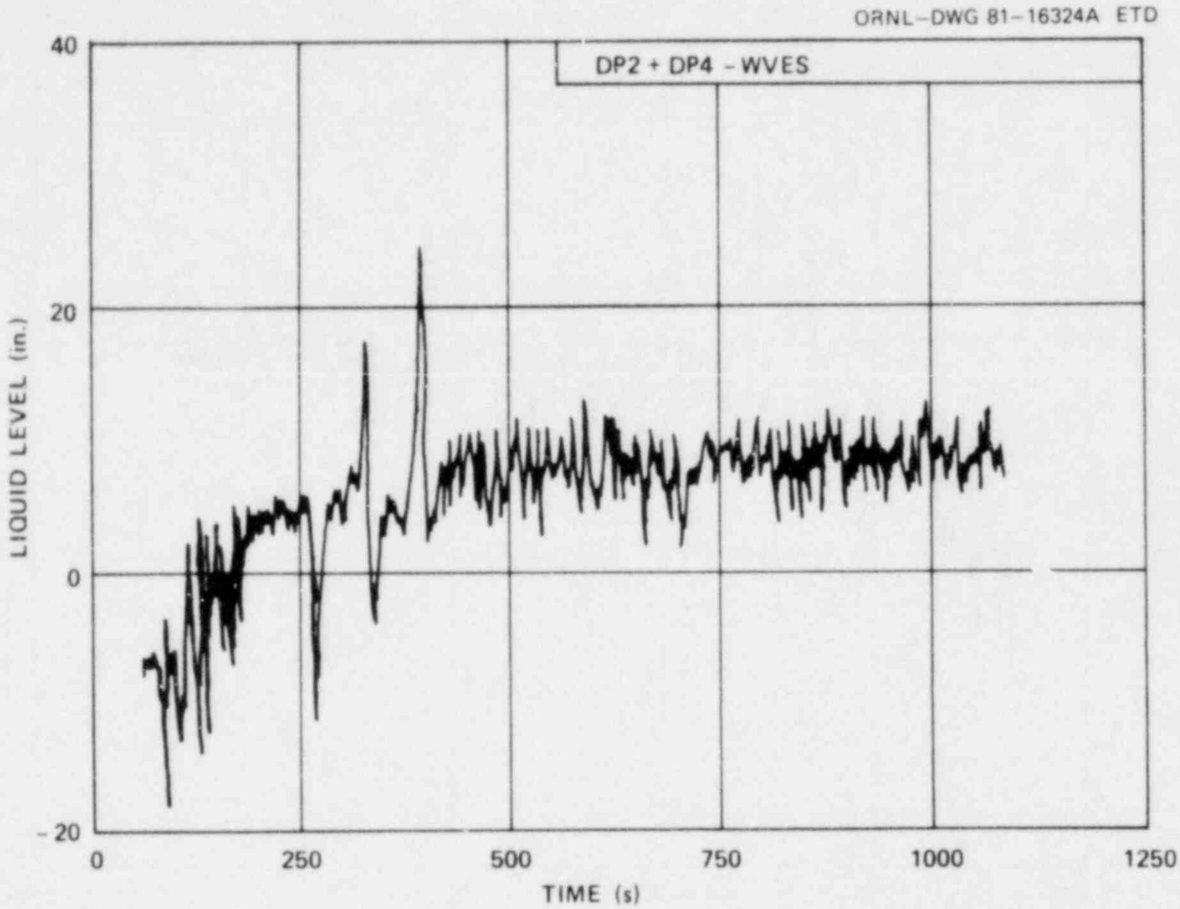


Fig. 7. Plot of difference between Westinghouse level measurements and Semiscale level measurements. (For entire vessel – note good agreement.)

ρ = density; the subscript r is for reference leg, v is for steam or vapor, and ℓ is for liquid - all in kilograms per cubic meter,

dP_m = measured dP in kilopascals,

D = distance of the lower tap above or below the reference elevation (cold leg centerline) in centimeters. The addition of D in this equation converts the Westinghouse equation for application in Semiscale. This equation can be simplified to:

$$L = \frac{H(\rho_r - \rho_v) - \Delta P_m}{(\rho_\ell - \rho_v)} .$$

This is the same equation derived and experimentally verified in the Blowdown Heat Transfer Program at ORNL.

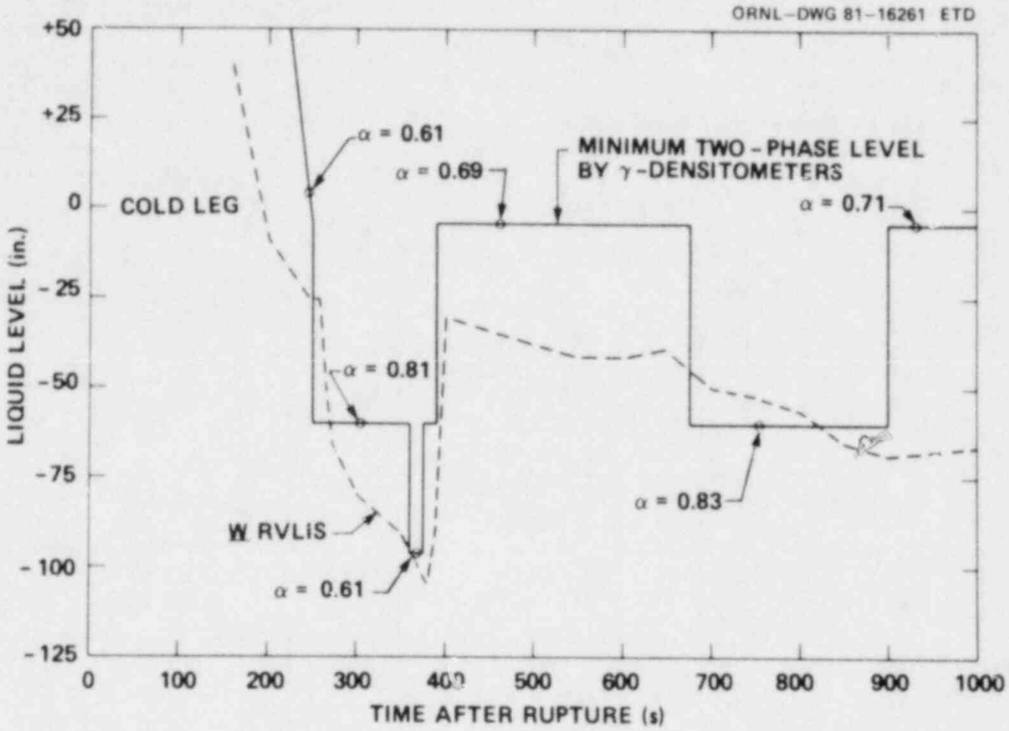


Fig. 8. Coolant levels in Semiscale for Westinghouse vessel dP and densitometers.

In the event that a break occurred and steam was blowing at the reference leg (capillary tube), an error could occur of as much as 33 cm (13 in.) in the reading. However, this is an unlikely occurrence and represents moreover a relatively small error.

2.2 Westinghouse Interpretation of Measurement Differences

According to Westinghouse, the structural differences between Semiscale and a Westinghouse reactor probably caused the large differences illustrated in Fig. 4. Their schematic of the Semiscale vessel, illustrated in Fig. 9, shows how the solid guide tubes block flow communication between the upper plenum and the upper head, resulting in the trapping of a two-phase mixture between the water below and water in the upper head. A typical design of a Westinghouse PWR is illustrated in Fig. 10. These reactors have perforated guide tubes that allow good flow communication between the upper plenum and the upper head. According to Westinghouse, the guide tubes in Semiscale would require modification (perforation like a Westinghouse reactor) before the Semiscale tests would accurately simulate the behavior of a Westinghouse PWR.

The guide tube in the upper plenum at Semiscale contains four slots near the bottom of the tube. These slots permit flow communication between the upper plenum and the upper head. Since the slots are relatively

ORNL-DWG 81-17871

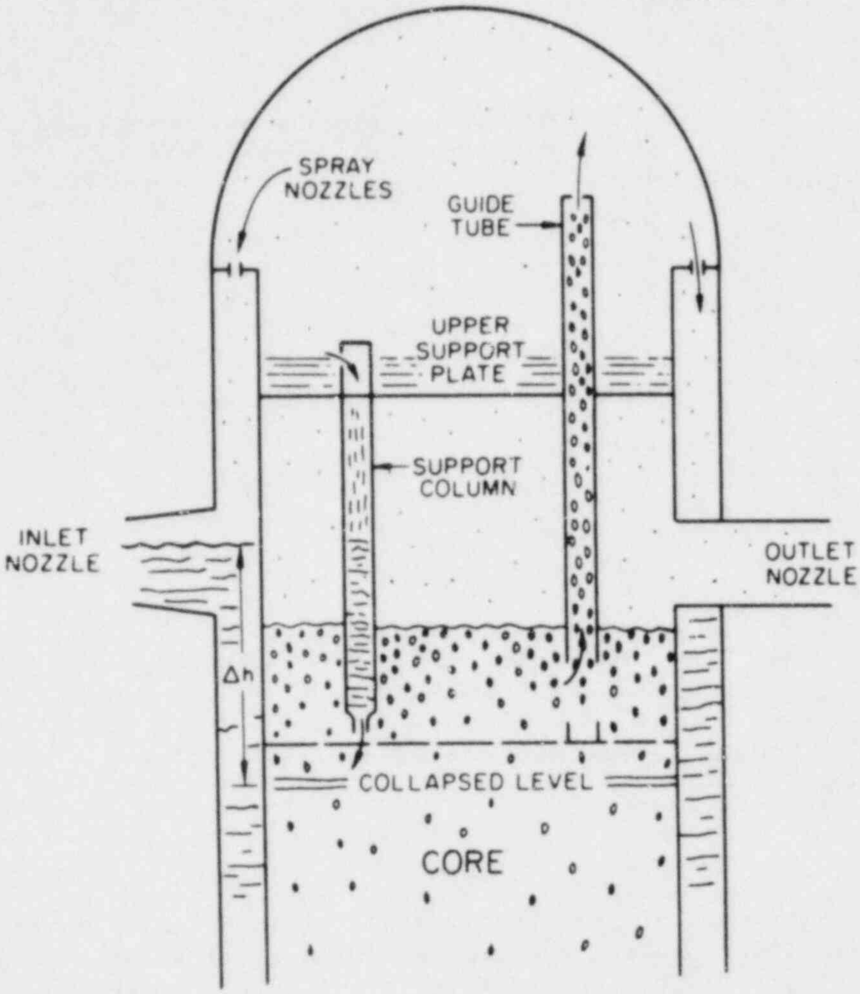


Fig. 9. Semiscale simulation of a PWR.

low in the upper plenum, Westinghouse suggests that a two-phase mixture may be trapped in the guide tube when there is a froth around the lower end of the tube. This trapped mixture adds to the level measured by the Westinghouse vessel system, but not to the Semiscale dP measurement from -13 to 578 cm. To evaluate this explanation for the difference between these two dP measurements, the following was done.

The top of the first slot is -36 cm below the cold leg, and the bottom of the fourth slot is at -91 cm. The guide tube is ~437 cm long. The "added" level resulting from the two-phase mixture in the guide tube can be calculated by

$$\text{Added Level} = (1 - \alpha)H \ ,$$

where α is the void fraction in the guide tube and H is the guide tube

ORNL–DWG 81–17870

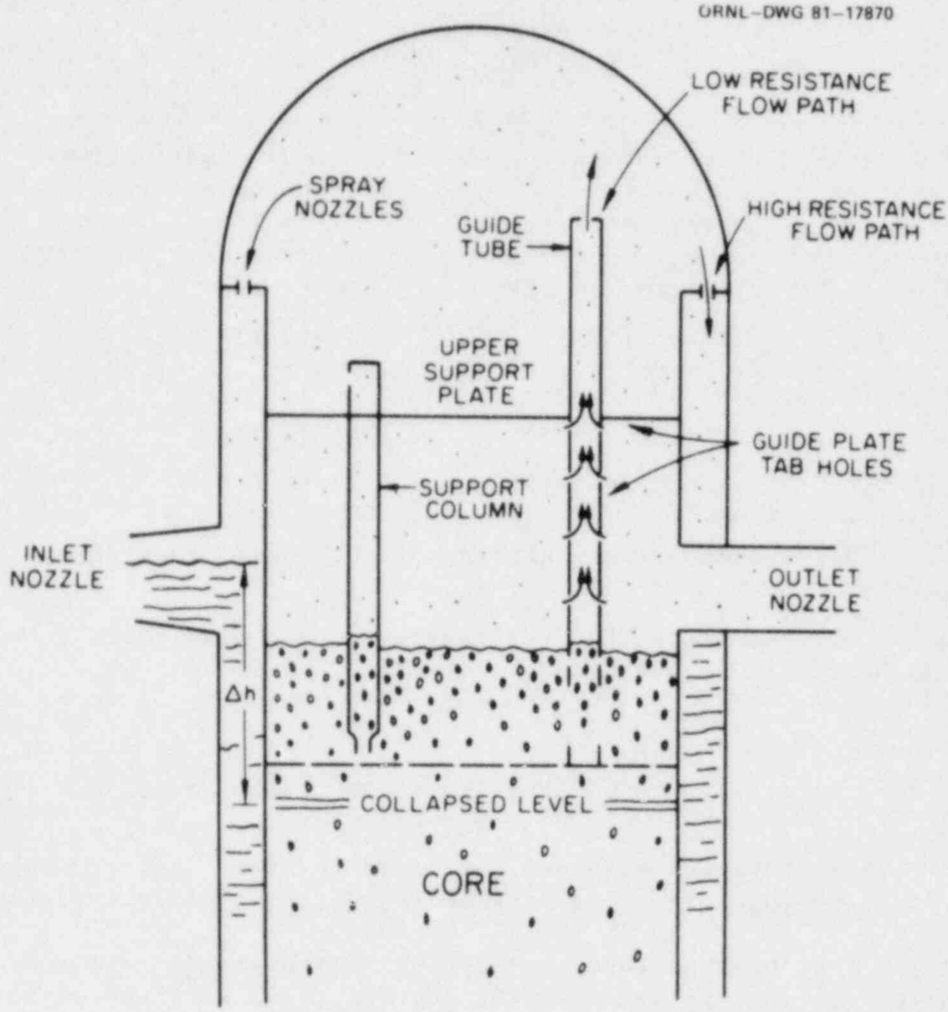


Fig. 10. Westinghouse reactor vessel configuration (PWR).

height. Using Fig. 5, a void fraction can be found at −36 and −91 cm to give a range of possible densities in the guide tube.

The possible added effect of the guide tube's trapping a two-phase mixture during the test is shown in Table 1. Also tabulated is the difference between Westinghouse and Semiscale dP measurements along with the difference between the Westinghouse upper-head value and Semiscale densitometers for the time range of 250–390 s. The discrepancies between the Westinghouse and Semiscale values are within the range of apparent guide tube levels. The only possible exception is that the upper-head measurement difference is slightly larger than the added guide tube level. This table illustrates the point that the guide tube effect is certainly a plausible explanation for the discrepancies between Westinghouse and Semiscale measurements.

Table 1. Level differences in Semiscale for test S-UT-3

Time after rupture (s)	Apparent guide tube level [cm (in.)]	Difference in Westinghouse and Semiscale levels [cm (in.)]
220	100–165 (40–65)	150 (60)[a]
250–390	70–60 (5–25)	40–50 (15–20)[a]
250–390	17–60 (5–25)	25–115 (10–45)[b]
500	85–130 (35–50)	100 (40)[a]
675–910	10–50 (5–20)	25–50 (10–20)[a]

[a] Difference between Westinghouse and Semiscale dP values.

[b] Difference between Westinghouse upper-head value and Semiscale densitometer.

2.3 Conclusions and Recommendations

Many conclusions and observations could be made, however, only those dealing with the possible ambiguous reading of the RVLIS are listed here:

1. The Westinghouse level and dynamic measurements read the same as Semiscale dP instrumentation when the tap locations were approximately the same.

2. Level measurements recorded during S-UT-3 that spanned the upper core plate were in error by as much as 150 cm (60 in.); this included all three Westinghouse measurements as well as Semiscale's Δp2.

3. The regions of concern discussed previously are times when data from other Semiscale measurements indicate that both Westinghouse and Semiscale measurements in Fig. 2 are incorrect and that the Semiscale measurement in Fig. 3 is correct. The Westinghouse dP level indication is in error during these times. Level estimates outside these regions of concern may also be in question. Our understanding is that Westinghouse does not intend to use the upper-head measurements in Fig. 2 for level measurement; they will only use this measurement for head venting.

4. Note that the Westinghouse level measurement was conservative when compared with the two-phase froth level (possible coolant level).

5. A trapped two-phase mixture in the guide tube can cause an added level value from 13–153 cm (5–60 in.). This effect is certainly a plausible reason for the differences between the Westinghouse and Semiscale level measurements. However, as seen in the dP measurements in the upper head, additional effects may be present.

Based on the analysis of this data, the recommendation is made that the Westinghouse explanation of level differences should be experimentally verified at Semiscale if possible.

REFERENCES

1. *TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendation*, NUREG-0578, pp. A-11-12 (July 1979).

2. *NRC Action Plan Developed as a Result of the TMI-2 Accident*, NUREG-0660, Sect. 1.D.5 and 2.F.2 (May 1980).

3. K. G. Turnage et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for January–March 1980*, NUREG/CR-1647 (ORNL/NUREG/TM-403).

4. K. G. Turnage et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for April–June 1980*, NUREG/CR-1768 (ORNL/NUREG/TM-414).

5. K. G. Turnage et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for July–September 1980*, NUREG/CR-1903 (ORNL/NUREG/TM-430).

6. K. G. Turnage et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for October–December 1980*, NUREG/CR-2007 (ORNL/NUREG/TM-443).

7. K. G. Turnage et al., *Preliminary Report on Heated Thermocouple Response During Thermal-Hydraulic Test Facility Test 3.07.9 Quasi-Steady-State Film Boiling* (December 1980).

8. J. E. Hardy et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for January–March 1981*, NUREG/CR-2204 (ORNL/NUREG/TM-7877).

9. J. E. Hardy et al., *Advanced Two-Phase Flow Instrumentation Program Quart. Prog. Rep. for April–June 1981*, NUREG/CR-2204/V2 (ORNL/NUREG/TM-8010).

10. W. W. Tingle and R. W. Golden, *Installation and Initial Test Data Report: Westinghouse Reactor Vessel Level Indicating System Performance During Semiscale Test S-UT-3*, EGG-SEMI-5494 (June 1981).

11. G. N. Miller, W. L. Zabriskie, and K. G. Turnage, "Frequency Response and Heat Transfer in a dP Measurement System with Long Sensing Lines," *Proceedings of the 26th International Instrumentation Symposium – Seattle, Washington, May 5–8, 1980*, Instrument Society of America, pp. 451–464.

NUREG/CR-2204, Vol. 3
ORNL/TM-8162
Dist. Category R2

Internal Distribution

1. J. L. Anderson
2. R. L. Anderson
3. S. K. Combs
4. W. G. Craddick
5. J. E. Hardy
6. A. F. Johnson
7. A. L. Lotts
8. G. N. Miller

9-10. T. W. Robinson, Jr.

11. S. C. Rogers
12. D. G. Thomas
13. H. E. Trammell
14. W. L. Zabriskie
15. ORNL Patent Office
16. Central Research Library
17. Document Reference Section

18-19. Laboratory Records Department

20. Laboratory Records (RC)

External Distribution

21. C. W. Connell, Jr., Babcock & Wilcox, Nuclear Power Generation Division, Lynchburg, VA 24505
22. R. E. Bryan, Combustion Engineering, 1000 Prospect Hill Road, Windsor, CT 06095
23. W. G. Lyman, Westinghouse Electric Company, Monroeville Nuclear Center, P.O. Box 355, Pittsburgh, PA 15230

24-31. Director, Division of Reactor Safety Research, Nuclear Regulatory Commission, Washington, D.C. 20555

32. Director, Reactor Division DOE, ORO, Oak Ridge, TN 37830
33. Office of Assistant Manager for Energy Research and Development, DOE, ORO, Oak Ridge, TN 37830

34-38. Director, Reactor Safety Research Coordination Office, DOE, Washington, D.C. 20555

39. H. D. Wills, General Electric, 175 Curtner St., MC 214, San Jose, CA 94303
40. P. G. Bailey, Electric Power Research Institute, P.O. Box 10412, Palo Alto, CA 94303

41-42. Technical Information Center, DOE, Oak Ridge, TN 37830

43-367. Given distribution as shown in category R2 (10-NTIS)

368-405. Special NRC External Distribution

120555064215 2 ANR2
US NRC
ADM DOCUMENT CONTROL DESK
PDR
016
WASHINGTON DC 20555