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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket Nos. 50-400 OL
and NORTH CAROLINA EASTERN	)	50-401 OL
MUNICIPAL POWER AGENCY	)	
	)	
(Shearon Harris Nuclear Power	)	
Plant, Units 1 and 2)	)	

APPLICANTS' MOTION FOR SUMMARY DISPOSITION OF CHANGE CONTENTION 44 AND EDDLEMAN CONTENTION 132

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December 7, 1983

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DS03

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Attachment 1: NUREG/CR-2628, Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement (March 1982)

Attachment 2: NRC Generic Letter No. 82-28, Inadequate Core Cooling Instrumentation System (Dec. 10, 1982)

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APPLICANTS' MOTION FOR SUMMARY
DISPOSITION OF CHANGE CONTENTION 44
AND EDDLEMAN CONTENTION 132

I. Introduction

Carolina Power & Light Company and North Carolina Eastern Municipal Power Agency ("Applicants") hereby move the Atomic Safety and Licensing Board, pursuant to 10 C.F.R. § 2.749, for summary disposition in Applicants' favor of Contention 132 by intervenor Wells Eddleman ("Eddleman 132") and Contention 44 by intervenors Chapel Hill Anti-Nuclear Group Effort and Environmental Law Project ("CHANGE 44"). As grounds for their motion, Applicants assert that there is no genuine issue of material fact to be heard with respect to CHANGE 44 and Eddleman 132, and that Applicants are entitled to a decision in their favor on these contentions as a matter of law.

This motion is supported by:

1. "Applicants' Memorandum of Law in Support of Motions for Summary Disposition on Intervenor Eddleman's Contentions 64(f), 75, 80 and 83/84," dated September 1, 1983;

2. "Applicants' Statement of Material Facts as to Which There is No Genuine Issue to be Heard (CHANGE Contention 44 and Eddleman Contention 132)";

3. "Affidavit of Clarence G. Draughon"; and

4. "Affidavit of Leonard I. Loflin."

II. Procedural Background

CHANGE 44 states as follows:

> A direct water level indicator for the reactor is essential to assure the public health and safety. Although it may be true that there are no absolutely certain indicators, a direct, environmentally-qualified level indicator is necessary to prevent the sort of confusion about reactor water level that contributed so significantly to the accident at Three Mile Island. (FSAR TMI-18).

Supplement to Petition for Leave to Intervene [CHANGE/ELP], at 16 (May 14, 1982). Eddleman 132 states as follows:

> Applicants have failed to provide the design for a direct water level indicator for the reactor vessel.

Board Memorandum and Order (Addressing Applicants' Motion for Codification), at 6 (Jan. 17, 1983). These contentions were admitted by the Board in its September 22, 1982 Memorandum and Order (Reflecting Decisions Made Following Prehearing

Conference), LBP-82-119A, 16 N.R.C. 2069, 2084, 2109 (1982), as modified by Memorandum and Order (Addressing Motions for Reconsideration and Clarification of the Board's Prehearing Conference Order), at 5-6 (Jan. 11, 1983).

Discovery has been open on CHANGE 44 and Eddleman 132 since September 22, 1982. See LBP-82-119A, supra, 16 N.R.C. at 2113 (1982). Discovery activity with respect to CHANGE 44 has included: Applicants' discovery requests to CHANGE of January 31, 1983, to which CHANGE responded on March 10, 1983; and CHANGE's discovery requests to Applicants of June 30, 1983, to which Applicants responded on August 12, 1983. Discovery activity with respect to Eddleman 132 has included: Applicants' discovery requests to Mr. Eddleman of January 31, 1983, and his March 21, 1983 responses; Mr. Eddleman's discovery requests to Applicants of March 21, 1983, to which Applicants responded on April 28, 1983; and Mr. Eddleman's discovery requests to the Staff of August 31, 1983, to which the Staff responded on October 12, 1983 (and supplemented on November 21, 1983).

III. Argument

A. Standards for Summary Disposition

The general standards by which motions for summary disposition are judged are set forth in "Applicants' Memorandum of Law in Support of Motions for Summary Disposition on Intervenor

Eddleman's Contentions 64(f), 75, 80 and 83/84," September 1, 1983, which is incorporated herein by reference.

B. Timeliness of the Motion

The instant motion is timely and the subject contentions are ripe for summary disposition. A motion for summary disposition may be filed at any time in the course of a proceeding.1/ Wisconsin Electric Power Company (Point Beach Nuclear Plant, Unit 1), ALAB-696, 16 N.R.C. 1245, 1263 (1982); see also 10 C.F.R. § 2.749(a). Here, the sponsors of the contentions have had ample opportunity -- more than 14 months -- in which to conduct discovery on the issues.

C. Relevant Regulatory Requirements

Following the accident at Three Mile Island, the NRC's TMI-2 Lessons Learned Task Force made a series of recommendations, some of which addressed the capability to detect the approach to and existence of inadequate core cooling ("ICC"). In particular, the Task Force recommended improvements in

1/ Thus the deadline established by the Board as the last day for filing motions for summary disposition of safety contentions (May 16, 1984), Memorandum and Order. . . at 7 (March 10, 1983), does not bar an earlier motion. In the case of CHANGE 44 and Eddleman 132, the intervenors were advised by Applicants in January, 1983, that summary disposition would be pursued in advance of the deadline, so that discovery should be pursued expeditiously.

operator training and procedures, the installation of a primary coolant saturation (or subcooling) meter, and consideration of additional supplementary instrumentation to provide an unambiguous, easy-to-interpret indication of ICC. NUREG-0578, TMI-2 Lessons Learned Status Report and Short-Term Recommendations (July 1979) at A-11, A-12; <u>Metropolitan Edison Company</u> (Three Mile Island Nuclear Station, Unit No. 1), LBP-81-59, 14 N.R.C. 1211, 1234 (1981), <u>aff'd as modified</u>, ALAB-729, 17 N.R.C. 814 (1983), <u>petition for review filed</u>.

Subsequently, the Commission approved for implementation by applicants for operating licenses and construction permit holders, as a part of its TMI Action Plan, certain Staff proposed requirements for additional instrumentation to detect ICC. <u>See</u> NUREG-0737, Clarification of TMI Action Plan Requirements (Nov. 1980) at item II.F.2 (reprinted in NUREG/CR-2628, <u>infra</u>, as Appendix A). The requirements included, among other things, the conduct of an evaluation of new reactor-water-level-indication instrumentation.

In NUREG/CR-2628, Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement (March 1982), which is Attachment 1 hereto, Oak Ridge National Laboratory documented a technical review, performed for the NRC Staff, of a reactor vessel level monitoring system using a differential pressure measurement system proposed by

Westinghouse for pressurized water reactors.2/ The conclusion of the review is that the Westinghouse proposed system meets the requirements of NUREG-0737.

In Generic Letter No. 82-28, Inadequate Core Cooling Instrumentation System (Dec. 10, 1982), the Staff states that on November 4, 1982, the Commission determined that an instrumentation system for detection of inadequate core cooling consisting of, among other things, a reactor coolant inventory tracking system, is required for the operation of pressurized water reactor facilities. See Attachment 2 (Generic Letter No. 82-28).

D. The Contentions are Satisfied

As the above history illustrates, the NRC's requirements for reactor vessel level indication instrumentation (which eventually became a requirement for an inventory tracking system) were still evolving during the period when contentions were being proposed for adjudication in this proceeding. Consequently, in 1982 Applicants had not yet determined what, if any, reactor coolant level or inventory indication system to install at the Harris plant. This situation was reflected at the time in the TMI Appendix of the Final Safety Analysis

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2/ NUREG/CR-2628 was produced to intervenors Eddleman and CHANGE during discovery.

Report and was the available information upon which CHANGE 44 and Eddleman 132 were based.

The situation has subsequently changed, however, in a determinative way. In an August 11, 1983 letter to the NRC Staff (Exhibit B to Loflin Affidavit), which was clarified and superseded by a letter of November 4, 1983 (Exhibit C to Loflin Affidavit), CP&L updated its response to Item II.F.2 of NUREG-0737 and indicated that the Westinghouse reactor coolant inventory tracking system is being installed as a part of the ICC instrumentation system at the Harris plant. This system fully satisfies the call, in CHANGE 44 and Eddleman 132, for fully qualified, direct water level indication in the reactor vessel at the Harris plant.

As the Staff has stated:

> The addition of a reactor coolant inventory system will improve the reliability of plant operators in diagnosing the approach of ICC and in assessing the adequacy of responses taken to restore core cooling. The benefit will be preventive in nature in that the instrumentation will assist the operator in avoidance of ICC when voids in the reactor coolant system and saturation conditions result from over cooling events, steam generator tube ruptures, and small break loss of coolant events. The addition of a reactor coolant inventory system, coupled with upgraded in-core thermocouple instruments and a subcooling margin monitor, provides an ICC instrumentation package which could significantly reduce the likelihood of human misdiagnosis and errors for events such as steam generator tube ruptures, loss of instrument bus or control system upsets, pump seal failures, or

> overcooling events originating from disturbances in the secondary coolant side of the plant. For less frequent events, involving coincidental multiple faults or more rapidly developing small break LOCA conditions, the ICC could also reduce the probability of human misdiagnosis and subsequent errors leading to ICC.

Attachment 2 at 1.

Further, the NRC has completed its review of the Westinghouse system and has accepted it for compliance with the requirement for a reactor coolant inventory tracking system. As stated in Generic Letter No. 82-28:

> The Nuclear Regulatory Commission has completed its review of several generic reactor level or inventory system instrumentation systems which have been proposed for the detection of ICC in PWRs. The Combustion Engineering Heated Junction Thermocouple (HJTC) system and the Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) are acceptable for tracking reactor coolant system inventory and provide an enhanced ICC instrument package when used in conjunction with core exit thermocouple systems and subcooling margin monitors designed in accordance with NUREG-0737 and operated within approved Emergency Operating Procedure Guidelines. The details of the NRC Staff review of these generic systems are reported in NUREG/CR-2627 and NUREG/CR-2628 for the Combustion Engineering and Westinghouse systems, respectively.

Id. at 2. See also NRC Staff Response to Interrogatories (2nd Set) dated August 31, 1983 Propounded by Wells Eddleman (Oct. 12, 1983), at 42-45 (Staff disagrees with Eddleman 132; Applicants have provided design for direct water level indicator for

the reactor vessel; Applicants' system is the Westinghouse system which has been generically approved).

During discovery, Mr. Eddleman was advised that the NRC Staff, in NUREG/CR-2628, had accepted the Westinghouse system and was asked whether Eddleman 132 would be satisfied if Applicants committed to install that system in the Harris plant. See Applicants' Interrogatories and Request for Production of Documents to Intervenor Wells Eddleman (First Set), January 31, 1983, at 43-44. Mr. Eddleman answered in the negative based upon his understanding of the system, although he had not reviewed the design of the Westinghouse system nor seen NUREG/CR-2628 at that time. See Wells Eddleman's Response to Applicants' First Set of Interrogatories and Request for Production of Documents, March 21, 1983, at 44.

The same inquiry was made of intervenor CHANGE. See Applicants' Interrogatories and Request for Production of Documents to Intervenor CHANGE (First Set), January 31, 1983, at 6-7. CHANGE likewise declined to consider CHANGE 44 satisfied if Applicants agreed to install the Westinghouse system. Citing NUREG/CR-2628 as a source of its information, CHANGE asserted that the Westinghouse system may not be sufficiently accurate, and that the Staff had not analyzed the effects of blockage or corrosion. The CHANGE responses were sworn to by Daniel F. Read, CHANGE President, whom Applicants

believe to have been a law student at the time. In response to specific inquiries made, CHANGE identified no other person as contributing to its responses. See Response of CHANGE to Applicants' First Set of Interrogatories, March 10, 1983, at 1-2.

Applicants submit that CHANGE 44 and Eddleman 132 have been satisfied as a result of the change in factual circumstances since those contentions were proposed and admitted for adjudication. The contentions amount to an allegation that the Harris plant will have no direct means for assessing the sufficiency of reactor coolant inventory. The facts now clearly demonstrate that such a system, acceptable to the NRC, will be provided. The intervenors' attempt to shift the focus of the litigation to alleged shortcomings of the Westinghouse system, for which the intervenors have advanced no supported technical basis, cannot overcome the demonstrated absence of an issue of material fact with respect to CHANGE 44 and Eddleman 132 as admitted by the Board.

While the shortcomings alleged in summary fashion by the intervenors are superfluous to the contentions, Applicants have responded to them in connection with this motion in an effort to assist the Board in its assessment of the weight to be given those arguments.

E. The Westinghouse System is Sufficiently Accurate

The very language of CHANGE 44 concedes that ". . . it may be true that there are no absolutely certain indicators . . ." of reactor water level. In fact, the RVLIS is sufficiently accurate to perform its intended functions. The ICC instrumentation system at Harris includes, in addition to RVLIS, a subcooling margin monitor and a core exit thermocouple system. See Loflin Affidavit.

During the early stages of a hypothetical ICC transient (i.e., before reactor coolant system (RCS) saturation is reached), the subcooling margin monitor informs the operator concerning core cooling and RCS inventory. If saturation is reached (i.e., the subcooling margin is zero), RVLIS comes into play by providing the operator with information regarding the trend in RCS inventory and an approach to core uncovery if depletion continues. If RCS inventory depletes to the point where the active fuel region of the core becomes uncovered, then superheated steam will be generated and the core exit thermocouple system will observe this condition. The RVLIS also provides a diverse indication of core uncovery along with the core exit thermocouples. Draughon Affidavit at ¶¶5-7.

In short, the unique contribution of RVLIS during an ICC transient is to indicate the trend of RCS inventory after subcooling is lost but before the core becomes uncovered.

Thereafter, it confirms the core exit thermocouple indications. It is not necessary that the capability be provided (assuming that it is technically feasible to do so) for an error-free measurement of the level of coolant in the reactor vessel.

The maximum allowable error for RVLIS wide range and narrow range indications is six percent of the indicated span. This maximum error is small enough that ICC transients can be detected and distinguished from other transients. It also does not impair the effectiveness of operator actions which, pursuant to emergency operating procedures, will be taken on the basis of indication changes much larger than the maximum error. Draughon Affidavit at ¶16.

Intervenors' unfounded desire for a more accurate inventory measurement appears to ignore the function of RVLIS in the ICC instrumentation system and the relationship between operator actions and the indications being provided by that system. The extensive testing of RVLIS at the Semiscale facility has demonstrated that it acceptably tracks RCS inventory. Draughon Affidavit at ¶¶17-21.

F. The Potential for Corrosion has been Addressed Adequately in the RVLIS Design

Materials of construction (304/316 stainless steel) of the capillary tubing, sensors, hydraulic isolators, and differential pressure transmitters and the fill fluid were selected to

minimize corrosion.3/ See Draughon Affidavit at ¶¶11-13, 22.

G. Potential Flow Blockage has been Evaluated

While blockage in the core resulting from a significant overheating will result in a higher RVLIS indication, prolonged core uncovery would first have to occur in order to reach this condition. The very purpose of the entire requirement for an enhanced ICC detection capability is to provide the necessary information, procedures and training to avoid such a condition. In any case, even with a large amount of flow blockage, the resulting RVLIS error is small, and the RVLIS will trend with the vessel inventory and provide useful information for monitoring the recovery from ICC. Draughon Affidavit at ¶¶23, 24.

IV. Conclusion

While a basis once existed for CHANGE 44 and Eddleman 132, that basis evaporated when Applicants made the commitment to install the Westinghouse RVLIS at the Harris plant. That system has been accepted generically by the NRC to meet its TMI Action Plan requirement for a reactor coolant inventory

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3/ Intervenor CHANGE's concern appears to have been based at least in part on a misapprehension that the capillary lines are directly attached to the reactor coolant system. In fact, the lines are isolated from the RCS through a bellows arrangement. See Applicants' Responses to Interrogatories to Applicants of Intervenor CHANGE/ELP (Contention 44), August 12, 1983, at 7; Draughon Affidavit at ¶12.

tracking system to be part of an enhanced instrumentation system to detect inadequate core cooling. The NRC's acceptance is based upon a thorough and well documented technical review of RVLIS, as well as extensive testing of the system. The intervenors' refusal to accept the fact that their contentions have been satisfied cannot change the situation. Irrelevant and baseless questions about accuracy, corrosion and blockage cannot save CHANGE 44 and Eddleman 132 from summary disposition. Applicants submit that there is no genuine issue of material fact for hearing on these contentions.

Respectfully submitted,

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Dated: December 7, 1983

Attachment 1

NUREG/CR-2628
ORNL/TM-8269

Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement

Prepared by G. N. Miller, J. L. Anderson, R. L. Anderson

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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EXECUTIVE SUMMARY

This document presents a technical review of the Inadequate Core Cooling Instrumentation with a Reactor Vessel Level Monitoring System using a differential pressure measurement system proposed by Westinghouse Inc., for pressurized water reactors. This system is Westinghouse's response to requirements of NUREG-0737 to evaluate the need for additional instrumentation to detect the approach to inadequate core cooling and in particular to evaluate means for measuring reactor vessel water level. Because the question of the need for reactor vessel measurement instrumentation has been a somewhat controversial issue, this report includes a great deal more material than is normally found in a technical review of this nature. It is the intention to provide in one document, coverage of all of the relevant material that has accumulated since the accident at TMI-2.

Emphasis was placed on evaluation of the generic Inadequate Core Cooling (ICC) Instrumentation system as a whole which includes, besides the differential pressure reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples and the display system (either the 7300, an analog display, or the microprocessor based system with a plasma panel display). Westinghouse refers to this complete system as the Reactor Vessel Level Instrumentation System (RVLIS). The system was evaluated on the basis of documentation supplied by Westinghouse Inc., tests run at ORNL and tests run at INEL (SEMISCALE).

Any generic description of an ICC instrumentation system is necessarily incomplete when applied to a specific plant because of differences in the individual plants. In the course of the evaluation of submittals by the individual licensees, it has become apparent that some utilities have chosen not to install the complete generic system offered by Westinghouse, particularly with respect to location of transmitter outside containment, use of isolators in impulse lines, and the display systems.

In Sect. I, a brief of the TMI-2 background is followed by a detailed discussion of the definition of Inadequate Core Cooling (ICC). In this discussion, measurement variables are defined to provide a basis for evaluation of the ICC instrumentation system. Following, is a discussion of the evaluation considerations used in the reviews.

In Sect. II, the existing plant instrumentation which can be used for ICC detection is reviewed and this review is followed by an analysis of the need for additional instrumentation. Westinghouse's proposed differential pressure coolant level measurement system is described in some detail.

In Sect. III, the testing of the Westinghouse system both at Westinghouse, Forrest Hills, and Idaho National Engineering Laboratory, SEMISCALE, are reviewed. Also Sect. III contains a review of anlayses of small break accidents performed by Westinghouse at the request of NRC and those are related to the RVLIS.

Section IV includes the evaluations of the differential pressure level measurement system with respect to the NUREG-0737 requirements and the instrumentation considerations. An extensive discussion of the ICC instrumentation system related to possible ambiguities incorporates responses to concerns about the system that resulted from the review process and expressed by others.

The conclusion of this evaluation is contained in Sect. V.

A brief description of the differential pressure level measurement system (RVLIS), a brief discussion of the concerns, and a summary of our conclusions are written below.

The RVLIS is a differential pressure measurement system. There are two trains of differential pressure measurement. On each train there are three differential pressure transmitters. Two transmitters are connected from the bottom of the vessel to the top of the vessel via tapping existing penetrations [on a Upper Head Injection (UHI) plant, the top connection is to the Hot Legs]. One of these two transmitters referred to as the narrow range unit is set up to measure the collapsed level in the vessel (0% to 100%) with the pumps off. With the pumps on, the narrow range unit is off scale high. The second transmitter, referred to as the wide range unit, is scaled to read 0% vessel empty and pumps off, and 100% with vessel full and all pumps on. With pumps off and vessel full, the wide range unit reads about 33% (15% on a UHI plant). The third transmitter of one train is connected between the hot legs and the top of the vessel and is referred to as the upper range unit. Westinghouse says this unit is not to be used except during head venting. When the vessel is full and pumps are off, the instrument indicates 100%, the unit indicates off scale in the 0% direction with vessel full and pumps on due to the frictional differential pressure.

These transmitters are connected to the vessel by armored capillary tubing. The transmitters are outside the containment wall, and there are two isolators in each capillary tube, one close to the vessel penetration tap point and one close to the containment wall. The capillary tubes are vacuum filled with demineralized deaireated water. The isolators have switch closures that indicate loss of capillary tubing water. Further, the isolators have valve stops that prohibit excessive fluid transfer. Problems in this area can be confirmed by the other train measurements and by the switch closures. Temperature measurements are made on any vertical run of these capillary tubes to compensate for density variations.

The generic display system for either plant could be an analog processor with panel meter readout (7300 system) or a microprocessor based system with a plasma panel readout (microprocessor system). Either system is supplied with a strip chart recorder for trending the analog outputs. Either system compensates for density variations between reference legs and the vessel. When the pumps are on, the 7300 system has a light telling the operator to disregard the narrow range and the upper range indications. The microprocessor unit has a status indication for these measurements to indicate when they are to be disregarded.

Analyses have been presented by the Westinghouse Owner's Group in WCAP-9753, of the system behavior with 1 and 4 in. diam breaks. Summary reports describing the generic analog and microprocessor based differential pressure level measurement system together with the Saturation Margin monitor and core exit thermocouples assert that these systems are adequate for detecting an approach to inadequate core cooling for breaks up to 4 in. diam. Tests of the differential pressure system were added to the regular testing program at SEMISCALE and the results reported in EGG-SEMI-5494 and EGG-SEMI-5552. Additional analysis of these results are forthcoming in ORNL-TMs. Some differences between indications of the Westinghouse system and the SEMISCALE differential pressure level system were noted in the upper head. Westinghouse claims that this difference is mainly a result of differences in the configurations betweeen the full-sized Westinghouse reactor and SEMISCALE upper head regions. Indications of other Westinghouse differential pressure level measurements were in good agreement with the SEMISCALE instruments in the same range. A repeat test was performed with the configuration of SEMISCALE modified to simulate a Westinghouse reactor. The dp level measurement in this test were in good agreement (less than 5% error) with SEMISCALE instrumentation. On August 8, 1981, the NRC requested additional information from the utilities proposing to use the Westinghouse differential pressure system. Most of these questions have been resolved to the staff's satisfaction, but a few outstanding questions remain to be answered. The generic description of the system along with the clarification supplied appear to be adequate for approval of the system for trial installation and use. Plant specific features, however, will still require review on a plant by plant basis.

In summary, the systems proposed by Westinghouse do provide an unambiguous indication of water level above the core when, in fact, such a level exists. During rapid transients, ambiguous indication may occur, but are expected to be of brief duration. For cases where the reactor vessel is filled with a two-phase mixture, experimental evidence indicates that the differential pressure systems will indicate collapsed liquid level or the trending of the reactor vessel coolant inventory. The conclusion of this evaluation is that the Inadequate Core Cooling Instrumentation system which includes the differential pressure reactor vessel level indicating system (RVLIS) proposed by Westinghouse will meet the requirements at NUREG-0737 to provide the plant operator with an unambiguous indication of the approach to adequate core cooling in small break LOCA transients. Furthermore, the system will provide the plant operator with a valuable indication of the effect of the recovery measures. Final approval is contingent on resolution of the open items listed below.

Generic emergency operating procedures have not been provided in the descriptions of the Westinghouse ICC systems. Detailed emergency operating procedures, however, are considered plant specific and must treat ambiguities, these will be reviewed separately for each plant. There are no other significant open issues to be resolved with respect to the generic Westinghouse ICC instrumentation system.

I. INTRODUCTION

I.A. Background - TMI

The accident at Three Mile Island, Unit 2 originating on March 28, 1979, has become one of the most intensely studied technological mishaps in history. The Report of the Kemeny Commission provides a careful reconstruction of the events during the week of the accident, from a more human than technical point of view.[1] The accident was initiated by malfunctions of a pressure release valve and its actuator indicator, and propagated by operator errors because of inadequate information about the true state of the reactor system (loss of coolant through the leaking or open pressure relief valve). Inadequate training on the part of the operators led them to shut off safety injection flow based on a high pressurizer level indication. By shutting off the high pressure injection into the primary system that had automatically been initiated by the control system to make up the loss of coolant from the pressure relief valve, the subsequent sequence of events led to the uncovery of and severe damage to the core.

At least one result of the post-TMI-2 studies was the mandate by the Kemeny Commission to the NRC to "consider the need for additional instrumentation to aid in understanding of plant status."[2] Through the efforts of the TMI-2 Lessons Learned Task Force, the NRC recommended the installation of additional instrumentation (if required) to "provide unambiguous, easy-to-interpret, indications of inadequate core cooling."[3] The Advisory Committee on Reactor Safeguards stated:

> "The Committee believes that it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. '...The Committee believes that as a minimum, the level indication should range from the bottom of the hot leg piping to the reactor vessel flange area."[4]

With the publication of NUREG-0737, all operating licensees and applicants for operating licensees were required to provide a description of any additional instrumentation to indicate inadequate core cooling and a time table for its installation.[5]

I.B. Definition of Inadequate Core Cooling

Before describing the role of the ICC instrumentation system, we must first discuss the definition of ICC. Simply speaking, inadequate core cooling must refer to a state of the reactor coolant system in which it is no longer capable of removing sufficient heat from the core to prevent core damage, i.e., rupture and/or melting of the fuel cladding occurs with release of radioactive materials into the primary coolant system. Avoidance of inadequate core cooling is necessary for safe reactor operation. One definition of inadequate core cooling that is given by the NRC staff is:[6]

"The staff considers the core to be in a state of inadequate core cooling whenever the two phase froth level falls below the top of the core and the core heatup is well in excess of conditions that have been predicted for calculated small break scenarios for which some uncovery with successful recovery from the accident have been predicted. Possible indicators of such a condition are core exit superheat temperature and/or the rate of coolant loss or level drop prior to core uncovery and the extent and duration of uncovery."

While this is a good phenomenological definition, a more detailed operational definition of inadequate core cooling is still required to guide the design and evaluation of ICC instrumentation. To limit the discussion to manageable proportions, we will be concerned only with situations for which the ICC instrumentation is intended to be used, i.e., small break LOCAs.

It is generally accepted by the NRC staff that the use of the ICC instrumentation is limited to those slow transient situations where significant response time is available for operator intervention, i.e., primarily "small break loss of coolant accidents" (LOCAs). The progress of events in a large break LOCA is so rapid that the operator would not have time to react and the protective measures are completed automatically. For large break LOCAs, the ICC instrumentation would be used to monitor recovery. Prior to the TMI accident, reactor safety research concerned with ICC had been limited, mainly, to the study of large break LOCA accidents. It is generally realized now, that the small break class of accidents has a much greater likelihood of occuring. The point which defines the difference between a "small" break and a "large" break is then partially defined by a sufficiently large interval of time that allows the operator to take action. Because of difference in plant designs, this period will be different for the plants of different vendors, however, a general scheme for the progression of events is shown in Fig. 1 for a 76-mm-diam (3 in.) cold leg break in a Westinghouse PWR plant. Core uncovery begins about 8 min after the break in this model. In these worst case calculations, with a 4 in. break, fuel cladding temperature is calculated to reach 1200°C (2200°F) after about 20 min.[7]

The definition adopted by Westinghouse is a "high temperature condition in the core such that operator action is required to cool the core before damage occurs."[7] From the results of their analyses, the Westinghouse staff concluded that "a core exit thermocouple reading of 650°C (1200°F) is a satisfactory criterion to alert the operator that action is required to cool the core before damage occurs."

We find that this definition is in agreement with the staff definition.

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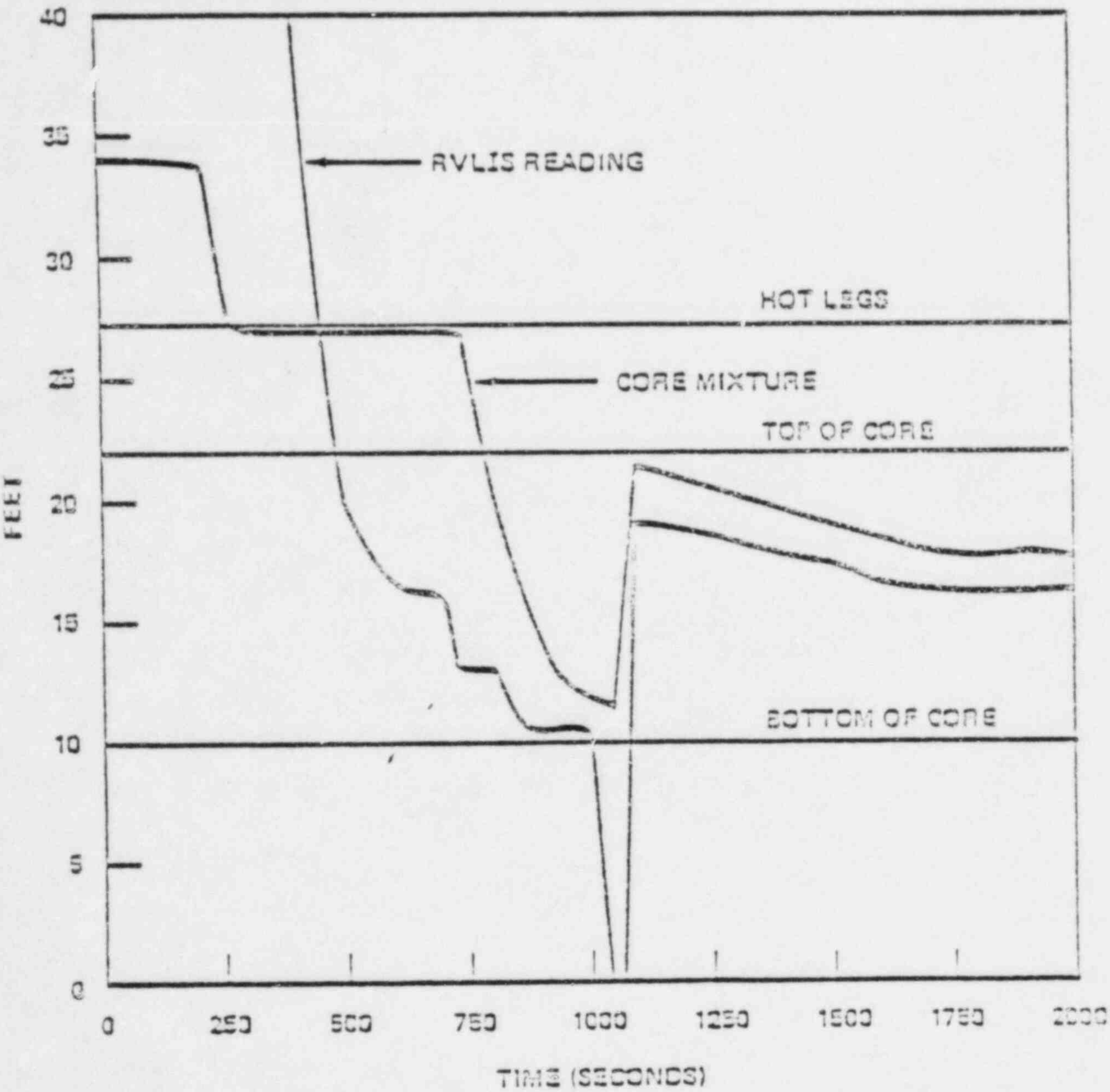

Fig. 1. A 3-loop plant, 3 in. cold leg break, pump trip at progress of events for 750 s, RVLIS reading and mixture level.

I.B.1. A Detailed Approach to Defining ICC Instrumentation

A detailed approach to defining ICC instrumentation has been described[8] as a three step process. First, an operational definition for the state of ICC was selected. Second, typical accident events that lead to ICC within the constraints of the definition, were analyzed. Third, instruments that would indicate the progression of one or more of these events were selected and evaluated.

II.B.1.1 Definition of ICC. A definition of ICC and the functional requirements for an ICC Instrumentation System can be established assuming the following reactor conditions:

1. The reactor is shut down, so only decay power needs to be considered.
2. The coolant level falls below the top of the core due to a loss of coolant mass from the Reactor Coolant System (RCS).
3. The event proceeds slowly enough so that the operator is allowed sufficient time to observe the instrument displays and take appropriate actions.

These conditions provide the boundaries for a regime of sizes of small break loss of coolant accident (LOCA) caused by either RCS rupture or primary coolant expansion due to loss of heat sink (dry out of the steam generators).

The ICC instrumentation is to be used to warn the operator of the approach to the state of inadequate core cooling. Before selecting such instrumentation, a definition for a measureable indication of the existence of ICC is required. Once this is established, then it is possible to define the instrumentation to indicate the approach to ICC. Several possible indications exist for the state of ICC. The list below is not exhaustive, but demonstrates a range of possibilities.

The following indications for the state of ICC were considered:

1. (First) occurence of saturation.
2. Core uncovery.
3. Fuel clad temperature increases and finally exceeds the maximum value predicted for normal recovery from an analyzed small break LOCA scenario.

It has been concluded from the small break LOCA analyses that some events progress slowly enough for practical instrumentation to reliably detect and display the approach of ICC.

Definition of Events For Approach To and Recovery From ICC

The system of instruments and sensor for detection of the approach to ICC provides to the operator a continuous indication of the thermal-hydraulic states within the reactor vessel during a progression of events towards or away from ICC. This progression can be divided into conditions based on the physical processes occuring within the vessel.

Conditions Associated with the Approach to ICC

Condition 1a Loss of fluid supercooling leading to the first occurrence of saturation conditions in the coolant.

Condition 2a Decreasing coolant inventory within the vessel head, (from the top of the vessel to the top of the active fuel).

Condition 3a Increasing core exit temperature due to the uncovery of the core resulting from the level of the mixture of vapor bubbles and liquid dropping below the top of the active fuel.

Conditions Associated with Recovery from ICC

Condition 1b Establishment of saturation conditions followed by an increase in fluid subcooling.

Condition 2b Vessel filled by the increase in liquid inventory so that the level of the two-phase mixture is above the fuel.

Condition 3b Decreasing core exit steam temperature resulting from an increasing coolant level within the core.

These conditions encompass all possible coolant situations associated with any ICC event progression. The conditions denoted with an "a" refer to fluid situations that occur during the approach to ICC. Conditions denoted by a "b" refer to fluid conditions which occur during the recovery from ICC. Thus, "a" conditions differ from "b" conditions in the trending (directional behavior) of the associated parameters.

To provide indicators during the entire progression of an event, an ICC instrument system consists of instruments that provide an unambiguous indication for the state of the RCS for at least one of each of the ranges of physical conditions described above.

Based on this description of the "approach to," and "recovery from" ICC, instruments were selected to:

1. Provide assurance that the entire progression of events could be followed by the selected ICC system.
2. Demonstrate the extent of instrument diversity or redundancy that is possible with the available instruments.

By defining the ICC progression in an operational fashion, the processes of "approach to" and "rocovery from" ICC can be associated with measureable physical quantities.

The instrument sensor package selected to monitor the ICC event progression consists of: (1) resistance temperature detectors (RTDs), (2) pressurizer pressure sensors, (3) reactor vessel level monitors employing differential pressure sensors, and (4) core exit thermocouples (CETs). The signals from the RTDs, CETs, and pressure sensors can be combined to indicate the loss of subcooling and occurrence of saturation (Condition 1a) and the achievement of a subcooled condition following core recovery (Condition 1b). The reactor vessel level monitors provide information to the operator on the decreasing liquid inventory in the reactor pressure vessel as well as the increasing liquid inventory following core recovery (Conditions 2a and 2b). The core exit thermocouples (CETs) monitor the increasing steam temperatures associated with the approach to ICC as the two-phase mixture level drops below the top of the core and the decreasing steam temperatures associated with recovery from ICC (Conditions 3a and 3b).

I.C. Evaluation Considerations for Reactor Vessel Coolant Level Instrumentation

The Nuclear Regulatory Commission[9] requires installation of reactor vessel coolant level instrumentation or equivalent to detect the approach to inadequate core cooling. Many owners and operators of nuclear power reactors submitted their plans for implementing this requirement to the Nuclear Regulatory Commission (NRC) for evaluation by January 1982. The design requirements for this instrumentation are provided in Item II.F.2 and Appendix B of NUREG-0737[10] (see also Appendix A, this document)., These criteria for the most part are functional in nature. There are additional characteristics that these systems should meet from the standpoint of the instrumentation and controls function these systems must fulfill. To assist in an objective evaluation of the methods proposed for reactor vessel level measurement, we present discussions of the instrumentation and controls characteristics with respect to the different operating requirements.

In response to TMI Task Force Action Plan II.F.2, the nuclear industry has proposed three different systems of additional information to monitor reactor vessel coolant level during the occurrence and recovery of a small break loss of coolant accident in a nuclear power reactor.

1. Differential pressure system proposed by Westinghouse and its plant owners' group.
2. Heated junction thermocouple system proposed by Combustion Engineering owners' group.
3. Excore Neutron Detector systems proposed by Alabama Power for their Farley nuclear plants (Westinghouse NSSS Design) and by National Nuclear Corporation.

The operational characteristics of nuclear power reactors may be divided into four broad classes: normal operation, abnormal operation-slow transient, abnormal operation-fast transient, and post accident operation. Abnormal operation resulting from a small break in the primary coolant system will produce a slow transient with loss of coolant inventory. An indication of the approach to core uncovery will facilitate an orderly handling of the problem. Post accident monitoring is needed to determine conditions in the vessel after a loss of coolant accident to insure that the reactor has been brought to a safe shut down and remains in a safe condition.

As seen in Sect. I.B above, a set of conditions associated with the approach to and recovery from ICC is falling or rising liquid (or two-phase mixture) level in the vessel head region (Conditions 2a and 2b). Loss of RCS inventory after the system depressurization to saturation cannot be detected either by the SMM or CET. Additional instrumentation is required to eliminate this blind spot for the operators so that they can observe the entire progression of events. Reactor vessel coolant level instrumentation is applicable primarily to the slow transient, small break loss of coolant accident and recovery. Reliable operation of reactor vessel coolant level instrumentation during normal reactor operation is necessary, so that the reactor operators will be confident in the indications of the instruments and that the instruments will be in good operating order if a problem arises.

Neither the ICC instrumentation as a whole nor water level instrumentation in particular is required to follow fast transients resulting from a large break loss-of-coolant accident, since Reactor Protection System instrumentation will identify this problem and initiate automatic coolant injection systems before the operator could respond to the ICC instrumentation. It is important, however, that the level instrumentation be capable of surviving such a transient so that it could be used in post accident monitoring and recovery.

The table below lists in various functional areas, but not necessarily in order of importance, the considerations used in this evaluation to assess the capability of proposed instrumentation to meet NUREG-0737 design requirements and application objectives.

General Considerations for Evaluation of
Reactor Vessel Coolant Level Instrumentation

A. Installation Specific

1. Requirements on operator
2. Calibration, procedures, in-situ procedures, frequency standards, etc.
3. Redundancy or diversity
4. Useful output during normal operation
5. Ease of retrofit or replacement
6. Interference with refueling

B. Sensor and Transducer Specific

1. Expected in-service life
2. Radiation resistance
3. Environmental resistance
4. Resistance to temperature damage or effect
5. Accuracy and resolution
6. Speed of response

C. Accident and Post-Accident Monitoring

1. Effects of core uncovery
2. Effects of reactor internals movements
3. Effects of pressure excursions
4. Effects of flow variations
5. Ability to measure water quality

General technical specifications have been derived based on these considerations, and these may be found in Appendix C of this document.

II. ICC INSTRUMENTATION

II.A Existing ICC Instrumentation

Adequate instrumentation is necessary to diagnose the approach to ICC and to determine the effectiveness of the mitigation actions taken. The following is a list of existing instrumentation and conclusions derived:

Parameters Critical to ICC

The analysis provided in ref. 7 and 11 delineates those parameters critical for the detection of and the necessary mitigation actions for the recovery from an ICC condition.

To briefly summarize those parameters, ICC is detected by either high core exit thermocouple temperatures or by a low reactor vessel level indication (core uncovery) in conjunction with core exit thermocouple indications. Mitigation actions consist of depressurizing the reactor coolant system (RCS) to permit injection of accumulator water and/or to establish low head safety injection flow. The RCS is itself depressurized by depressurizing the steam generator secondary side. Critical parameters at this point are steam generator pressures and wide range RCS loop temperatures. Once low head safety injection flow is established, transfer out of the ICC procedure can be made when core exit thermocouple temperatures are reduced and the reactor vessel level gauge indicates a level above the top of the core.

With the exception of reactor vessel level, all parameters are monitored by currently existing instrumentation.

1. Current Instrumentation
   a. Wide range reactor coolant pressure - present instrumentation is available for indicating general RCS pressure trends during the ICC event. Following ICC events the expected accuracy is such that this instrument cannot be used for precise determinations of the pressure required to assure onset of low head safety injection flow to the RCS.

   b. Pressurizer pressure and level - conditions in the pressurizer will generally lie outside the ranges of these instruments during an ICC event in a Westinghouse PWR. Pressurizer pressure and level are not used for determining mitigation actions to be taken during ICC.

   c. Auxiliary feedwater flow - present instrumentation is available for assuring a sufficient flow of makeup water to the steam generators during an ICC event.

d. Wide range resistance temperature detectors - present instrumentation is available for determining trends of recover actions but may not be available for determining the one ICC condition for all break sizes.

e. Core exit thermocouples (CETs) - present instrumentation is available for determining both the existence of ICC and the trends of recovery actions.

f. Core subcooling - saturation margin monitor (SMM) - does not provide useable information during an ICC condition. It will indicate superheat conditions in core coolant. It will help indicate the approach to ICC by showing saturation conditions. General specifications for the core subcooling margin monitors are given in Table 1 for information.

g. Steamline pressure - present instrumentation is available for determining heat sink availability and heat removal capability during ICC mitigation actions.

h. Steam generator level - present instrumentation is available for determining the availability of a heat sink for the RCS during an ICC condition.

In summary current plant instrumentation can determine heat sink availability, to detect the onset of ICC through the SMM and CETs, and to detect the effectiveness of mitigation actions following the onset of an ICC event. The RVLIS permits a continuous indication of all phases of the approach to ICC as a result of a small break LOCA.

II.B Need for Additional Instrumentation

The studies of the TMI-2 accident generally concluded that additional instrumentation to indicate the approach to ICC should be considered for installation in both operating plants and plants applying for operating licenses. It was also concluded that too great a burden was placed on the operators to diagnose the state of the RCS in the vessel by inference from the indications of existing instrumentation. Formerly, (before TMI) the subcooling margin had to be obtained by taking the indication of the hot leg temperature RTDs and the indication of the pressurizer pressure sensor and looking up the corresponding amount of subcooling in steam tables (a copy of which was not in the control room at TMI during the early hours of the accident). The procedures the operators were following instructed them to infer the RCS inventory from the pressurizer level indication. Because the system was losing coolant via the pressurizer through the stuck-open PORV, there was an indicated high level in the pressurizer while the reactor coolant inventory was low. The decay heat from the core caused the coolant in the primary system to expand, filling the pressurizer. The operators shut off the high pressure inject on to keep (so they thought and had been taught) the RCS from going "solid". With continued loss

Table 1. Information on the saturation margin monitor

Display	
Information displayed (T-Tsat, Tsat, press, etc.)	P-Psat subcooled T-Tsat superheat
Display type (analog panel meter, microprocessor CRT)	Analog and digital
Continuous or on demand	Analog - continuous Digital - on demand
Single or redundant display	Redundant
Alarms (include setpoints)	Alarm - 0°F subcooled for RTD and T/C (Caution - 25°F subcooled for RTD; 15°F subcooled for T/C)
Overall uncertainty (°F, psi)	Digital - 4°F for T/C; 3°F for RTD Analog - 5°F for T/C; 5°F for RTD
Range of display	Calibrated region - 1000 psi subcooled to 2000°F superheat; overall - never offscale
Calculator	
Type (process computer, dedicated digital or analog calc.)	Dedicated digital
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	Highest T for RTD or T/C (CETs) lowest P

Table 1. (Cont'd.)

Calculational technique (steam tables, functional fit, ranges)	Functional fit - ambient to critical point
<u>Input</u>	
Temperature (RTDs or T/Cs)	RTD, T/C and Tref
Temperature (number of sensors and locations)	RTD - 2 hot and 2 cold leg per channel T/C - 8 per channel
Range of temperature sensors	RTD - 0 - 700°F T/C - 0 - 1650°F (calibration unit range 0 - 2300°F)
Pressure (number of sensors and locations)	2 wide range - loop 1 narrow range - pressurizer
Range of pressure sensors	Wide range - 0 - 3000 psi Narrow range - 1700 - 2500 psi

of coolant through the stuck PORV, the core began to uncover because of the loss of subcooling leading to the formation of voids in the reactor vessel.

Two parameters that the operators did not have available which would have aided the operators to diagnose the true state of the RCS were an indication of the subcooling margin and a direct measurement of the reactor vessel level. This was also the conclusion of the ACRS in their initial report on the accident:

> "...Additional information regarding the status of the system will be needed in order for the plant operator to follow the course of an accident and thus be able to respond in an appropriate manner. As a minimum, and in the interim, it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel..."[12]

The final report of the "TMI-2 Lessons Learned Task Force" published in October of 1979 include the following recommendation:

"Each licensee should be required to define and adequately display in the control room a minimum set of plant parameters (in control terminology, a state vector) that defines the safety status of the nuclear power plant. The minimum set of plant parameters should be annotated for sensor limits, process limits, and sensor status. The annotated set of plant parameters should be presented to the operator in real time by a realiable, single-failure-proof system located in the control room. The annotated set of parameters should also be available in real time in the Onsite Technical Support Center.

"The objective of this recommendation is to require a concise set of information that is easily available and assessed by the operator and the shift technical supervisor to ascertain the safety status of the operating process. As a further guideline for the development of the safety state vector, the status of the plant process should be designed and instrumented as a function of the various barriers against the release of radioactivity. For example, the two primary barriers are the fuel cladding and the reactor coolant pressure boundary. Thus, parameters such as primary liquid inventory and coolant radioactivity levels would be principal components of the state vector for these levels of defense. Similarly, reactor coolant level, ...would be principal components of the state vector for the engineered safety feature levels of defense."[13]

The high cladding temperatures resulting from ICC can lead as a first step to rupture of the fuel pin cladding. This becomes more likely as the fuel is used. The fission reactions lead to a pressurization of the fuel pins due to formation of gaseous fission products. Secondly, at somewhat higher temperatures, the rate of the chemical reaction between the water and the Zircaloy fuel pin cladding becomes appreciable and results in the production of gaseous hydrogen inside the reactor vessel. This was undoubtedly the source of the hydrogen bubble inside the reactor vessel at TMI-2.

"The NRC Action Plan Developed as a Result of the TMI-2 Accident,"[14] identified water level in the core as one of the new instrumentation needs. More specific requirements were issued in NUREG-0737, "Clarification of TMI Action Plan Requirements," Sect. II.F.2, "Instrumentation for the Detection of Inadequate Core Cooling."[15]

"Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling."

The requirements set forth in NUREG-0737 did not explicitly require the installation of water level instrumentation in the reactor

vessel, but it did require that such reactor-water-level instrumentation be evaluated. Instrumentation to detect inadequate core cooling was required to: 1) give advance warning of the approach to ICC, 2) to cover the full range from normal operation to complete core uncovery. In practice, ICC systems that have not included reactor vessel level measurement have been found unacceptable by the NRC staff. All three of the PWR reactor vendors have described ICC instrumentation systems that include the saturation margin monitor (unequivocally required by all plants) and indications of the core-exit thermocouples. Two vendors have included reactor vessel level measurement, and one offers an integrated computer driven ICC display system as a part of an accident monitoring system. The small break analyses performed by the vendors after TMI-2 have shown that a period of decreasing reactor vessel inventory may occur, particularly with smaller break sizes where the water level or two-phase mixture level above is located above the core in the vessel-head. Indication of level in the reactor head fills in a blind spot in the available plant instrumentation. Water level measurement allows the operator to follow the course of the accident and determine if action or actions are required. The reactor vessel level measurement component of the ICC instrumentation is the only direct measurement of coolant inventory during significant portions of a small break accident (See summary of analyses in Sect. III.B below).

Similarly, during recovery operations, without a measurement of reactor vessel level, the operator would have to infer the effectiveness of the actions from other plant instrumentation. But, between the recovery of the core and reestablishment of subcooling, there may be a substantial period (hours) where the only effective indicator of the progress of recovery is a measure of the water or mixture level above the core.

Finally, if a "bleed and feed" mode of cooling were used, it would be essential for the operator to monitor the reactor vessel level - particularly during the "bleed" operation, to avoid uncovering the core.

II.C Reactor Vessel Level Instrumentation System - System Description

II.C.1 General Description

The reactor vessel level instrumentation system (RVLIS) uses differential pressure (d/p) measuring devices to measure vessel level or relative void content of the circulating primary coolant. The system is redundant and includes automatic compensation for potential temperature variations of the impulse lines. There are three versions of this system which will be discussed below.

The three systems are Upper Head Injection (UHI), the 7300 (an analog processor and panel meter display) and the microprocessor based system (CRT or Plasma Panel display). The last two systems could be

used with either a UHI plant or a non-UHI plant since they differ only in the processor and display areas. Essential information is displayed in the main control room in a form directly usable by the operator.

The functions performed by the RVLIS are:

1. Assist in detecting the presence of a gas bubble or void in the reactor vessel
2. Assist in detecting the approach to ICC
3. Indicate the formation of a void in the RCS during forced flow conditions

II.C.2 Detailed System Description

II.C.2.a Hardware description - normal plant (non-UHI)

II.C.2.a.1 Differential pressure measurements. The RVLIS (Fig. 2) utilizes two sets of three d/p cells. The UHI version of the RVLIS is illustrated in Fig. 3. These cells measure the pressure drop from the bottom of the reactor vessel to the top of the vessel, and from the hot legs to the top of the vessel. This d/p measuring system utilizes cells of differing ranges to cover different flow behaviors with and without pump operation as discussed below:

1. Reactor vessel - upper range (ΔP_a)

   The d/p cell ΔP_a shown in Figs. 2 and 3 provides a measurement of reactor vessel level above the hot leg pipe when the reactor coolant pump (RCP) in the loop with the hot leg connection is not operating.

2. Reactor vessel - narrow range (ΔP_b)

   This measurement provides an indication of reactor vessel level from the bottom of the reactor vessel to the top of the reactor during natural circulation conditions. This measurement is from the vessel bottom to the hot leg for a UHI plant because the injection system flow would result in an ambiguous RVLIS output if the connections spanned the upper head.

3. Reactor vessel - wide range (ΔP_c)

   This instrument provides an indication of reactor core and internals pressure drop for any combination of operating RCPs. Comparison of the measured pressure drop with the normal, single-phase pressure drop will provide an approximate indication of the relative void content or density of the circulating fluid. This

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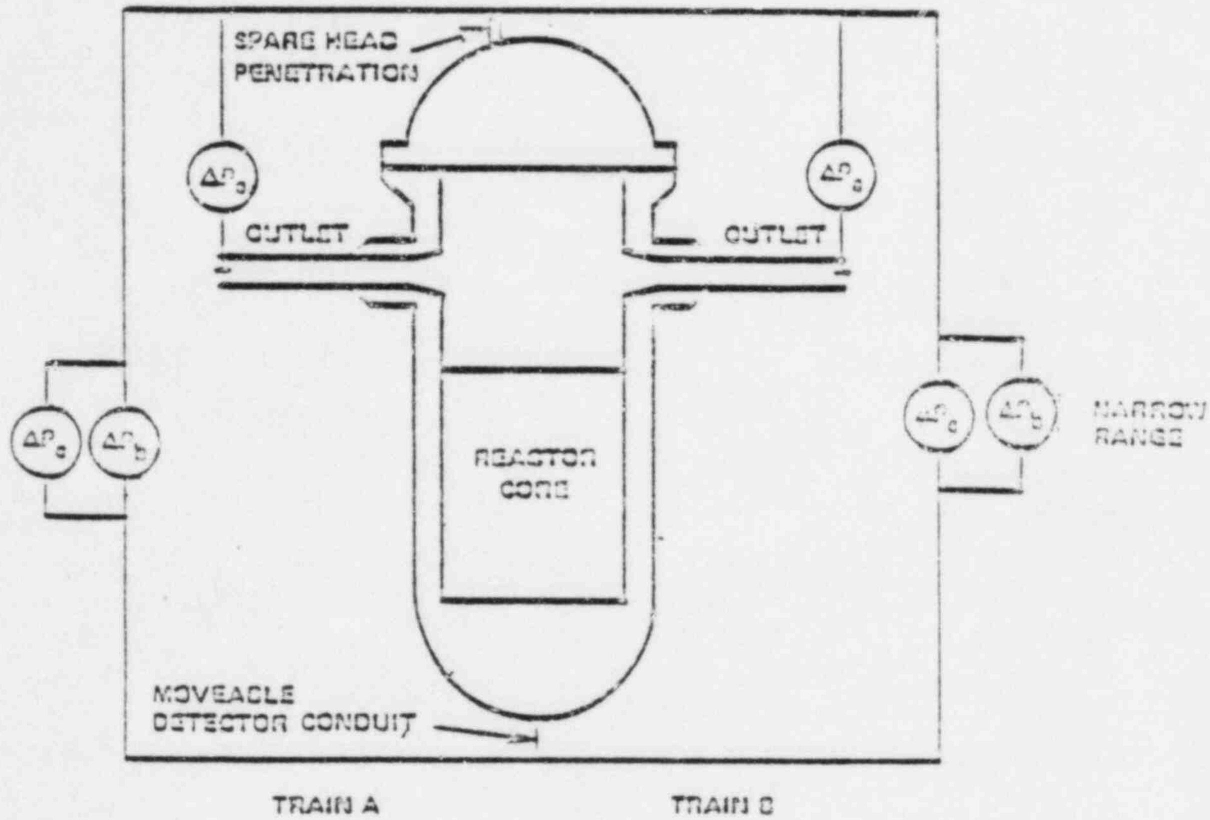

Fig. 2. Reactor Vessel level instrument system for non-UHI plant.

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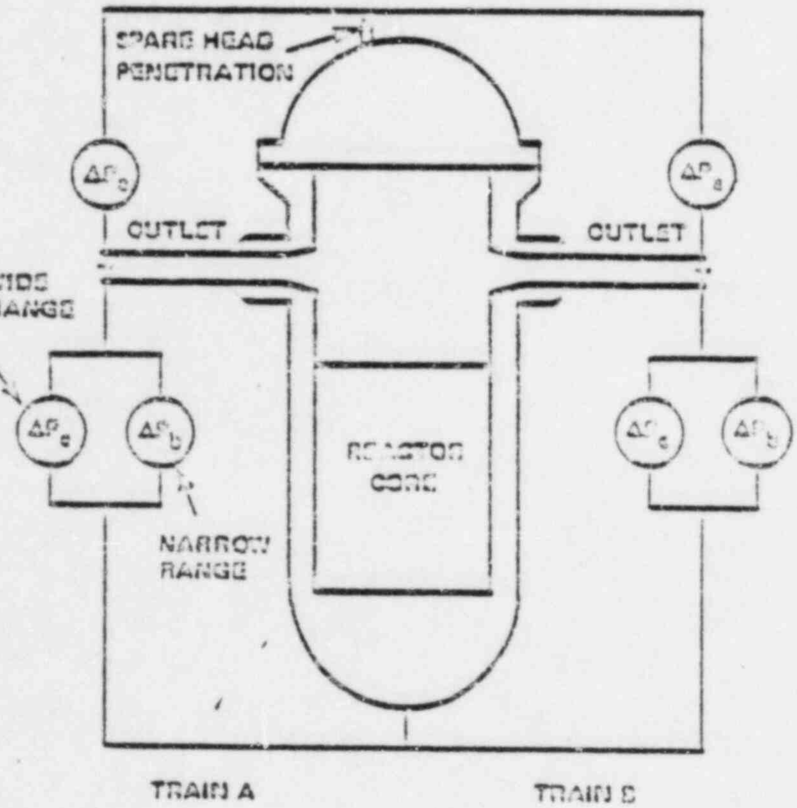

Fig. 3. Reactor vessel level instrument system for UHI plant.

instrument will monitor coolant conditions on a continuing basis during forced flow conditions.

The connections of these transmitters to the vessel is illustrated in Fig. 2 for a non-UHI plant and Fig. 3 for a UHI plant.

To provide the required accuracy for level measurement, temperature measurements of the impulse lines are provided. These measurements, together with the existing reactor coolant temperature measurements and wide range RCS pressure, are employed to compensate the d/p transmitter outputs for differences in system density and reference leg density, particularly during the change in the environment inside the containment structure following an accident.

The d/p cells are located outside of the containment to eliminate the large reduction (approximately 15% estimated by Westinghouse) of measurement accuracy associated with the change in the containment environment (temperature, pressure, radiation) during an accident. The cells are also located outside of containment so that system operation including calibration, cell replacement, reference leg checks, and filling is made easier.

II.C.2.a.2 System layout. A schematic of the system layout for the RVLIS is shown in Fig. 4. There are four RCS penetrations for the cell reference lines; one reactor head connection at a spare penetration near the center of the head or the reactor vessel head vent pipe, one connection to an incore instrument conduit at the seal table, and connections into the side of two RCS hot leg pipes.

The pressure sensing lines extending from the RCS penetrations will include manual isolation valves. These lines connect to sealed capillary impulse lines (at the reactor head, at the seal table and at each hot leg) which transmit the pressure measurements to the d/p transmitters located outside the containment building. The capillary impulse lines are sealed at the RCS end with a sensor bellows which serves as a hydraulic coupling for the pressure measurement. The impulse lines extend from the sensor bellows through the containment wall to hydraulic isolators, which also provide hydraulic coupling as well as a seal and isolation of the lines. The capillary tubing extends from the hydraulic isolators to the d/p transmitters, where instrument valves are provided for isolation and bypass.

Figure 5 is an elevation plan of a typical plant showing the routing of the impulse lines. The impulse lines from the vessel head connection must be routed upward out of the refueling canal to the operating deck, then to the containment penetration. The connection to the bottom of the reactor vessel is routed axially and radially to join with the head connection line in routing to the penetrations. Similarly, the hot leg connection impulse lines are routed toward the penetration routing of the other two connections.

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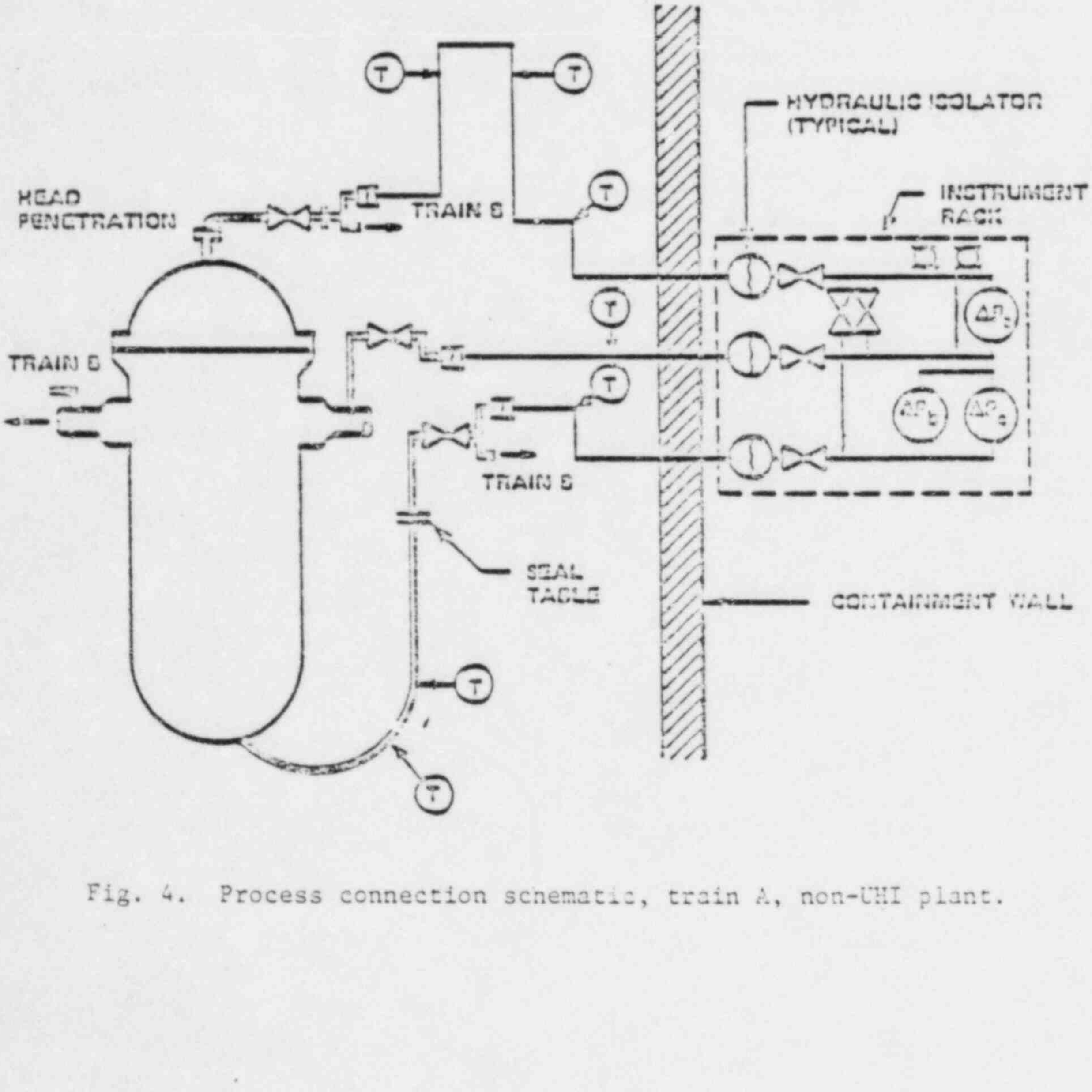

Fig. 4. Process connection schematic, train A, non-UHI plant.

ORNL-DWG 82-10184

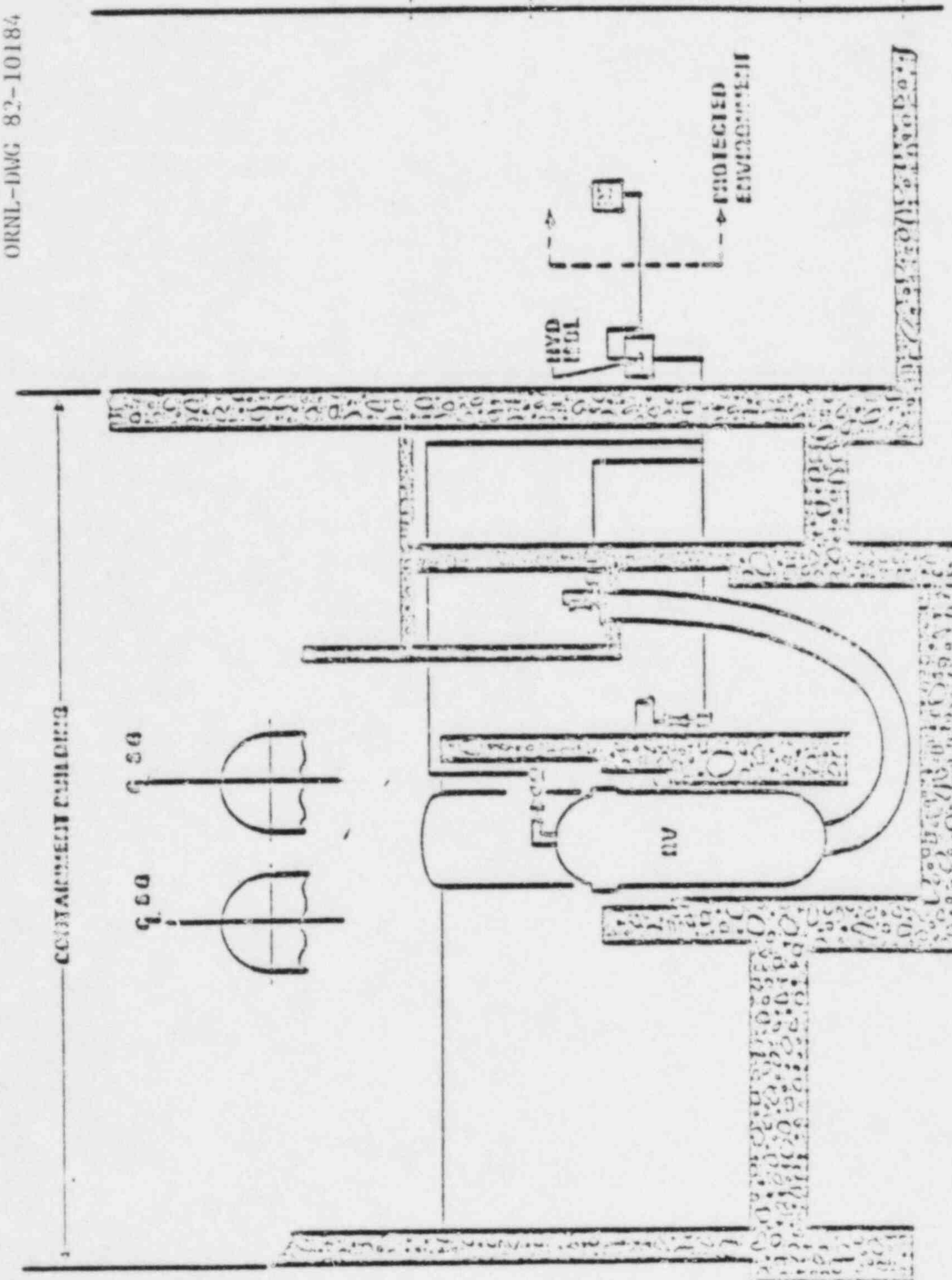

Fig. 5. Typical plant arrangement for RVLIS.

The impulse lines located inside the containment building will be exposed to the containment temperature increase during a LOCA or HELB. Since the vertical runs of impulse lines form the reference leg for the d/p measurement, the change is density due to the accident temperature change must be taken into account in the vessel level determination. Therefore, a strap-on RTD is located on each vertical run of separately routed impulse lines to determine the impulse line temperature and correct the reference leg density contribution to the d/p measurement. Temperature measurements are not required where all three impulse lines of an instrument train are routed together. Based on the studies of a number of representative plant arrangements, a maximum of 7 independent vertical runs must be measured to adequately compensate for density changes.

II.C.2.a.3 Microprocessor RVLIS. The microprocessor RVLIS includes equivalent reactor vessel level indications on redundant flat panels with alphanumeric displays for control room installation in addition to having this information available for display at the microprocessor chassis. RVLIS is configured as two protection sets, in certain installations in separated sections of a single instrument rack and in other installations in two separated instrument racks. In addition to the reactor vessel level (d/p) transmitter input to the microprocessor, there are also temperature compensating signals, reactor pump running status inputs, and RCS parameter inputs to each chassis of the two redundant sets. The output of each set will be to displays and to a recorder, as well as an output for a serial data link. A general microprocessor display arrangement is shown in Fig. 6. A typical display for the other two systems is illustrated in Fig. 7.

II.C.2.a.3.a RVLIS inputs. The system inputs are as follows. If existing unqualified inputs are used, isolation as required is to be provided by the owner.

Differential Pressure Transmitters

The three d/p transmitters per set are used to measure the d/p s between the three pressure tap points on the primary system, as discussed below:

1. ΔP_a is connected between a tap on the head of the vessel and tap on the hot leg of one of the coolant loops, which is typically about 4 ft or more above the top of the core.

   The direction of this transmitter's output is full scale (20 ma) with the vessel full and zero scale (4 ma) with the vessel emptied to the hot leg tap. These endpoints are nominal and are for low coolant temperatures. If no pumps are operating, ΔP_a gives an indication of level in the region above the hot leg.

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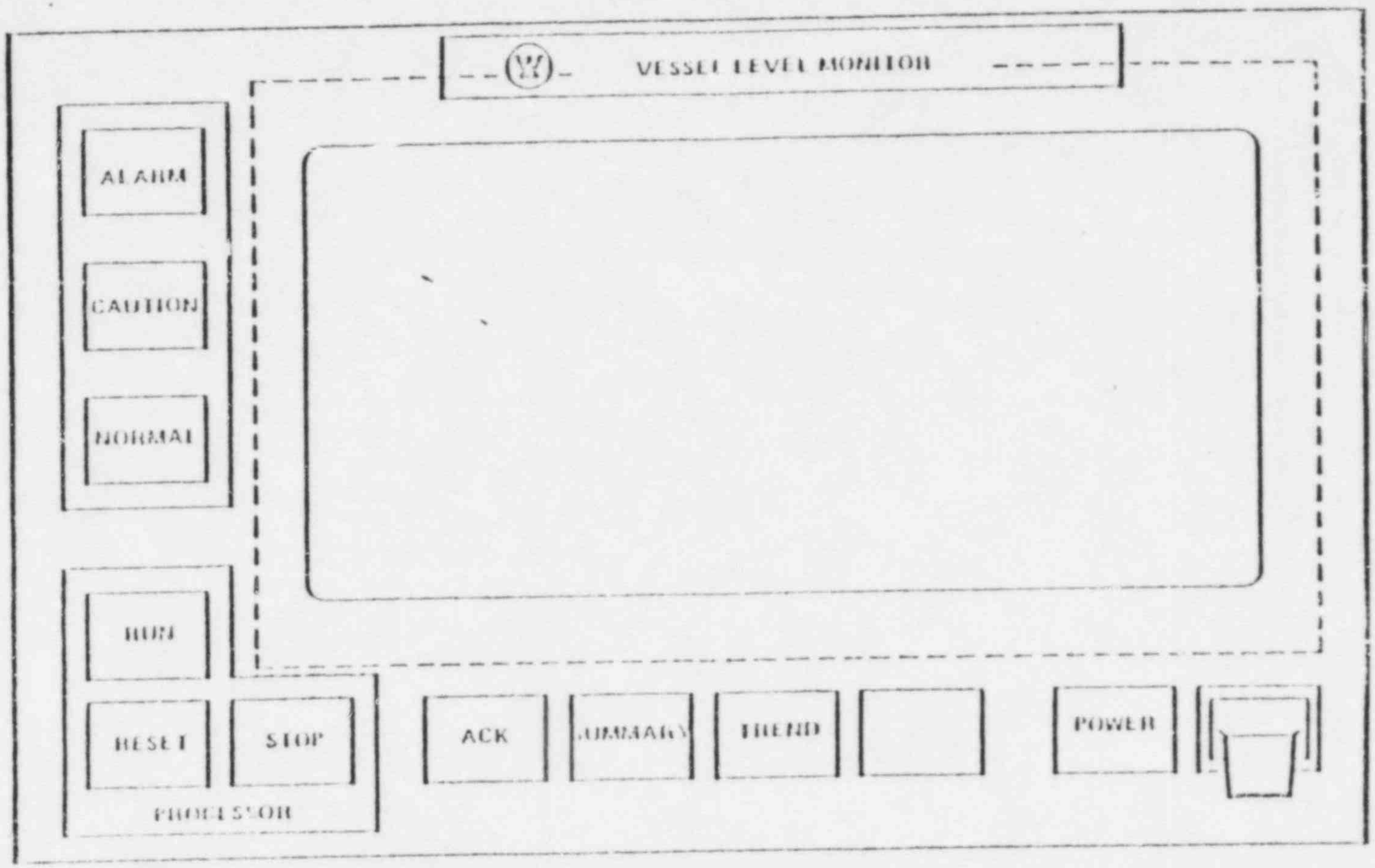

Fig. 6. Remote display module (control board).

ORNL-DWG 82-10203

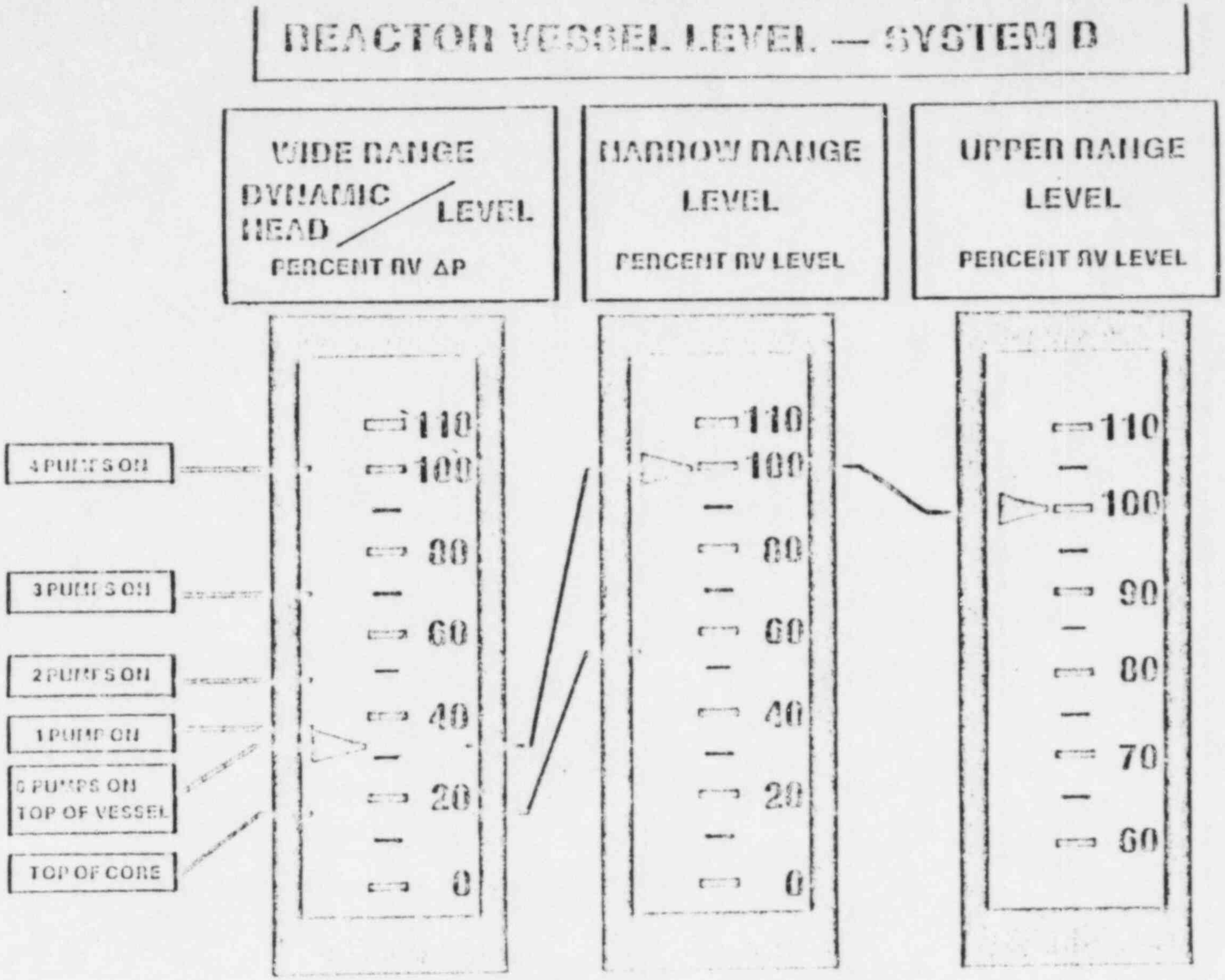

Fig. 7. RVLIS display one set-other set same.

If the pump is running in the loop with the hot leg connection, this indication will be inval.d and most likely off-scale in the zero direction. The reading would be flagged as "invalid" under these conditions. The effect on the indication from the pump not running in this loop, but running in other loops, is less than 10% of the range.

2. ΔP_b is connected between the top of the vessel and the bottom of the vessel. For a UHI plant the top tap is connected to the hot leg.

   ΔP_b gives an indication of reactor vessel level when no pumps are running. If one or more pumps are running, ΔP_b will be off-scale (greater than 100%) and the reading invalid.

   The sense of the ΔP_b output is such that a 20 ma signal is a nominally full vessel and a 4 ma signal is for a nominally empty vessel.

3. ΔP_c is a higher range d/p cell connected between the same two pressure taps as ΔP_b. ΔP_c covers the entire span of all pumps running t vessel empty. The sense of the ΔP_c output is that 20 ma rep.esents all pumps running and 4 ma is empty vessel. With all pumps running and no void fraction, the ΔP_c should read 100% at zero power. The reading at full power is slightly higher.

Reference Leg Temperature RTD

The reference leg temperature RTDs are 100 ohm platinum four wire RTD and are used to measure the temperature of the coolant in the capillary tube reference legs. This is used to compute the density of the reference leg fluid.

The conversion of RTD resistance to temperature shall cover the temperature range of 32° to 450°F.

Hot Leg Temperature

Either existing or new wide range hot leg temperature sensors are used to measure the coolant temperature. This temperature is used to calculate coolant density.

Wide Range Reactor Coolant Pressure

Either existing wide range pressure sensors or new pressure sensors will be used to measure reactor coolant pressure. The pressure is used to calculate reactor coolant density.

Digital Inputs

The reactor coolant pump status signals indicate whether or not pumps are running. The hydraulic isolators provided on each impulse line for containment isolation purposes have limit switches to indicate if they have reached the limit of travel.

II.C.a.3.b Density compensation system. To provide the required accuracy for vessel level measurement, temperature measurements of the impulse lines are provided. These measurements, together with the existing reactor coolant temperature measurements and wide range RCS pressure, are employed to compensate the d/p transducer outputs for differences in system density and reference leg density, particularly during the change in the environment inside the containment structure following an accident. The d/p cells are located outside the containment.

The reference leg fluid density calculation must cover a range of 32° to 450°F. The fluid is assumed to be compressed liquid water at 1200 psia.

Each of the three d/p measurements will have density corrections from certain temperature measurements. Some of these will have a positive correction and some negative depending on the orientation of the impulse line where the temperature is being measured.

Vessel Liquid Density Calculation

Three inputs are used to calculate the density of the liquid phase in the reactor vessel. The two hot leg reactor coolant temperatures are auctioneered with the highest used to calculate the density of saturated liquid, $\rho f(T)$. The wide range pressure signal is used to calculate the density of saturated liquid $\rho f(P)$. The highest of these two densities is used as the liquid phase density.

Vessel Vapor Phase Density Calculation

The same basic algorithm is used as for the liquid phase. The auctioneered high hot leg coolant temperature is used to calculate the density of saturated steam, $\rho g(T)$. The pressure is also used to calculate the density of saturated steam, $\rho g(P)$. The lowest of these two densities is used as the vapor phase density.

Vessel Level Calculation

This calculation applies only to ΔP_a and ΔP_b.

Pump Flow d/p Calculation

This algorithm applies only to ΔP_c. The measured ΔP_c is first corrected for impulse line water density. An expected ΔP_c is then calculated for the condition of all pumps running and a full reactor vessel (no circulating voids) using a technique essentially identical to that used to calculate the liquid and vapor phase densities. The auctioneered high temperature of the two hot leg RTD signals is auctioneered with the pressure measurements and the result is used to calculate a d/p correction as a function of temperature. This is a calculated curve which can be corrected using data obtained during plant startup. It will have a shape very similar to that of the liquid density curve.

Reactor Vessel Water Level Signal Compensation for SEMISCALE Tests

Reactor Vessel Fluid Density

Figure 8 illustrates the differential pressures acting on the transmitter. The net differential pressure across the transmitter is defined by the equation:

$$\Delta P = \rho_o h - [\rho_f h_f + \rho g\ (h - h_f)]$$

where ΔP = differential pressure across the transmitter

ρ_o = density of reference leg fluid (water) at ambient temperature

h = height of measurement span (reference leg)

ρ_f = density of water in reactor vessel

h_f = actual height of water in reactor vessel (collapsed water level when voids are present)

ρ_g = density of steam in reactor vessel

When the reactor vessel is full at ambient temperature, ΔP across the transmitter would be zero. The electronics of the transmitter are set up so that the electrical output is low (4 ma) when ΔP is at a maximum (vessel empty), and the electrical output is high (20 ma) when ΔP is zero (vessel full). The indicated ΔP is therefore defined by the following equation:

$$\Delta P_i = \rho_o h - \Delta P$$

where ΔP_i = ΔP equivalent to the electrical output, or equivalent to indicated water level.

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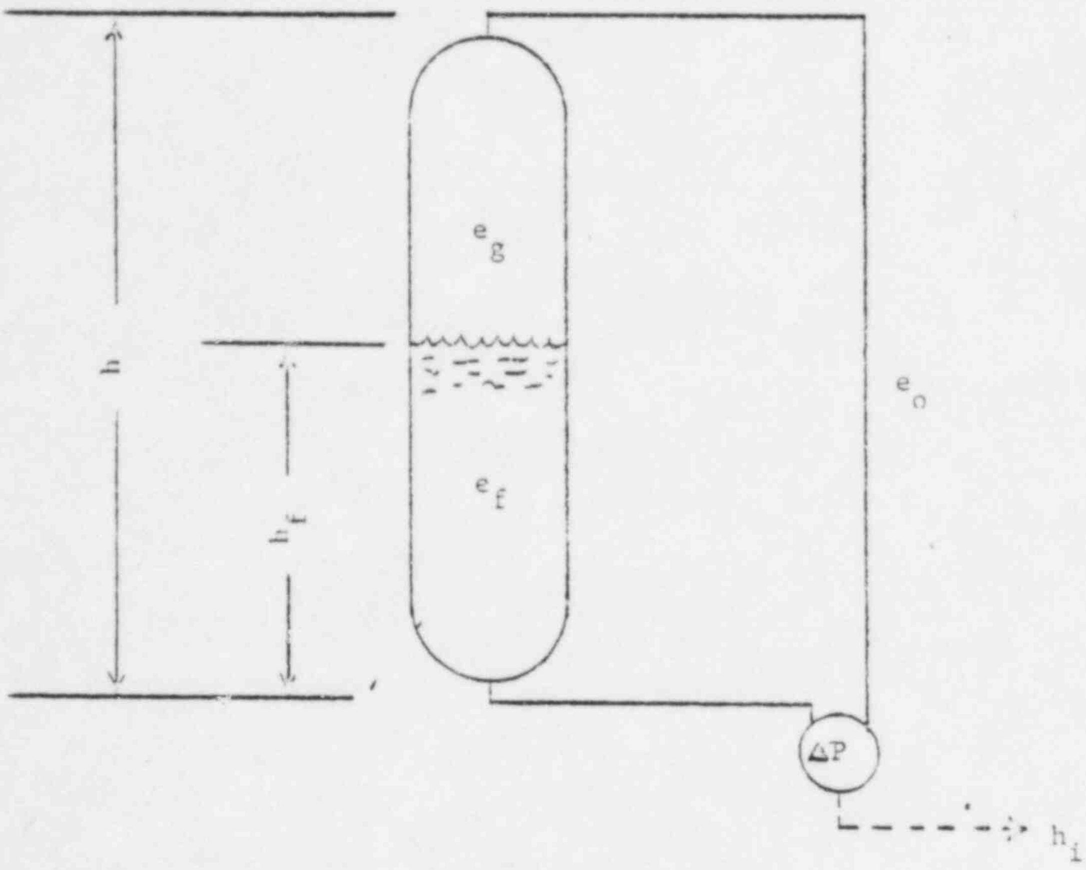

Fig. 8. Illustration of reactor vessel differential pressures.

In terms of indicated water level, the above equation becomes:

$$\rho_o h_i = \rho_o h - \Delta P$$

$$= \rho_o h - \rho_o h + \rho_f h_f + \rho g\ (h - h_f)$$

where h_i = indicated water level.

Since actual water level is the desired information, the equation[16] rewritten to:

$$h_f = \frac{\rho_o h_i - \rho_g h}{\rho_f - \rho_g}$$

This equation is, therefore, the compensation function to convert indicated water level to actual water level. Figure 8a is a graphical solution of this expression.

During accident conditions when saturated fluid conditions exist, it is assumed that fluid temperatures within the measured span are essentially the same, and fluid properties are based on a system pressure measurement. If subcooled conditions exist, fluid properties are based on a hot leg temperature measurement.

The transmitter measuring the dynamic pressure drop has a span of minus 32.75 ft (vessel empty) to zero (vessel full) to plus 50 ft (all pumps operating, plus some margin). The electrical output of the transmitter at ambient conditions would be interpreted as follows: 4 ma = 32.75 ft, 10.33 ma = 0 ft, and 20 ma = 50 ft. The calculated output of the transmitter for the standard sytem arrangement installed in the SEMISCALE facility from ambient to operating temperature is illustrated on Fig. 9. Since there are several variables involved in calculating this relationship, the relationship was determined by measurement during heatup of the facility. This data would become the reference for tests when pumps are operating. A reduction in pressure drop compared to the reference would be an indication of the presence of a circulating void condition.

The primary coolant pressure is used to calculate an expected d/p correction as a function of pressure. This function is also similar to the liquid density as a function of pressure curve and is obtained from the d/p correction curve versus temperature by assuming saturated conditions.

The lower of the two calculated d/p corrections is divided into the measured d/p. The result is the percent of expected d/p and should read 100% with all pumps operating and no circulating voids.

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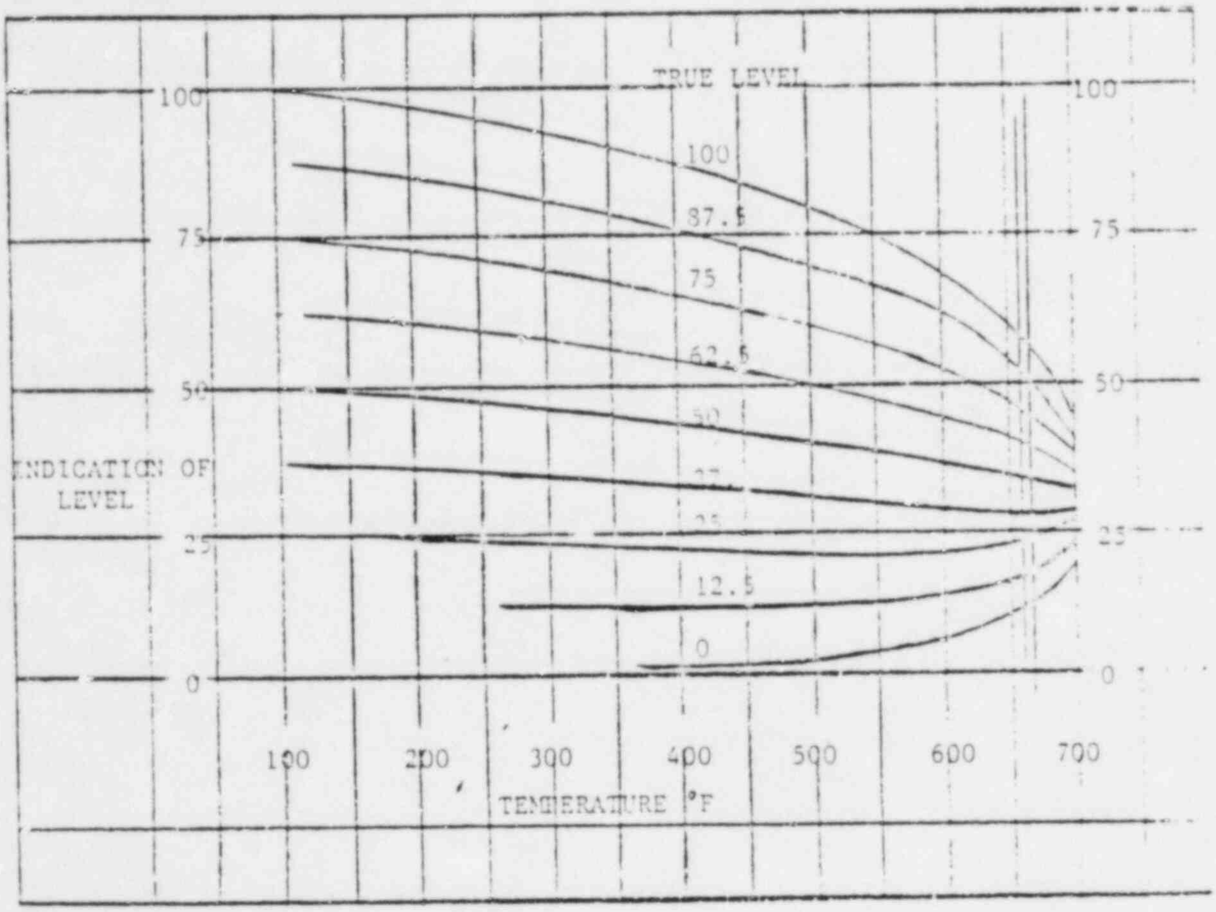

Fig. 8A. Graphical solution of density variation as a function of temperature.

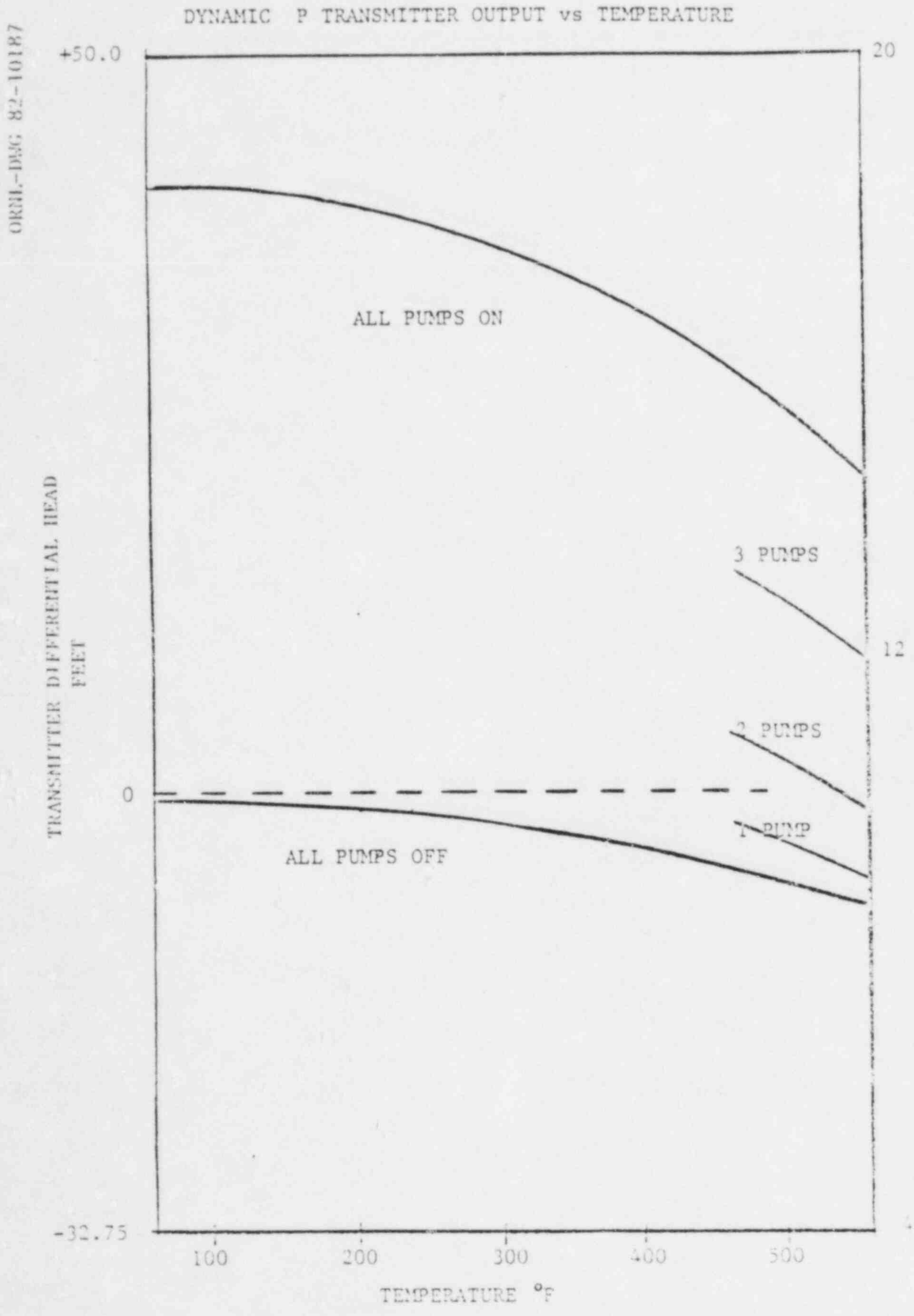

Fig. 9. Dynamic P transmitter output vs temperature.

Scaling of Displayed Values

Each of the three d/p measurements after the preceding calculations shall be scaled to read in percent. With the vessel full of water and no pumps running, the outputs of ΔP_a and ΔP_b should read 100%.

II.C.2.a.4.a Plant operator interface and displays. Information displayed to the operator for the RVLIS is intended to be unambiguous and reliable to minimize the potential for operator error or misinterpretation. The redundant control board displays provide the following information:

1. An indication of reactor vessel level (narrow range) for each instrumented set displaying vessel level in percent from 0 to 100% after compensation for the effects of the reactor coolant and capillary line temperature and density, when reactor coolant pumps are not operating.

2. An indication of reactor d/p (wide range) from each instrumented set displaying d/p in percent from 0 to 100%, after compensation for the effects of the reactor coolant and capillary line temperature and density effects, when reactor coolant pumps are operating.

3. An indication of upper range vessel level on each of the two instrumented sets displaying vessel level in percent from 60% to 100% after compensation for any reactor coolant and capillary line density effects, when reactor coolant pumps are not operating.

All signals are input to the (microprocessor or analog) data analysis system. The control room display format utilizes an alphanumeric display located remotely from the computational system. The analog display system utilizes panel meters.

Redundant displays are provided for the two sets. Level information based on all three d/p measurements is presented. Correction for reference leg densities is automatic. Any error conditions such as out-of-range sensors or hydraulic isolators are automatically displayed on the affected measurements.

There are two display sheets for reactor vessel level the first is a summary sheet, and the second is a trending of the three vessel level indications.

The system provides three analog signals for a single three-pen strip chart recorder.

II.C.2.a.4.b Display functions for remote control board. The display unit for the microprocessor based vessel level monitor is the 8 line, 32 character per line alphanumeric display which is located in the control board remote from the main processing unit. The analog system display is three panel meters for each of the two trains.

Vessel Level Monitor Summary Display

Figures 10 and 11 give example displays. The vessel level summary display is shown on Fig. 10. The following is a description of the display.

1. The first line gives the title of the display as shown. The use of the underbar feature delineates this line from the rest of the display.

2. The second line gives column headings as shown. Again, the use of the underbar clarifies the display.

3. The third line gives the measured and normally expected value from the ΔP_a measurement. The first field gives the title, the second gives the measured level, the third gives the normal value for the current status, and the last field gives the validity status and is blank under normal conditions.

4. The fourth line gives the ΔP_b measurement results using the same format as in line 3.

5. The fifth line gives the ΔP_c measurement results using the same format as in line 3. The use of underbar in line 5 delineates this line from the next.

6. The sixth line gives the status of the pumps as seen by the unit. The running pumps are identified.

7-8. The seventh line and eighth line are normally left blank and are reserved for hydraulic isolator limit switch indicators, out of range sensors and operator disabled sensors.

Trend Display

The trend display for the vessel level monitor shall use the format shown in Fig. 11.

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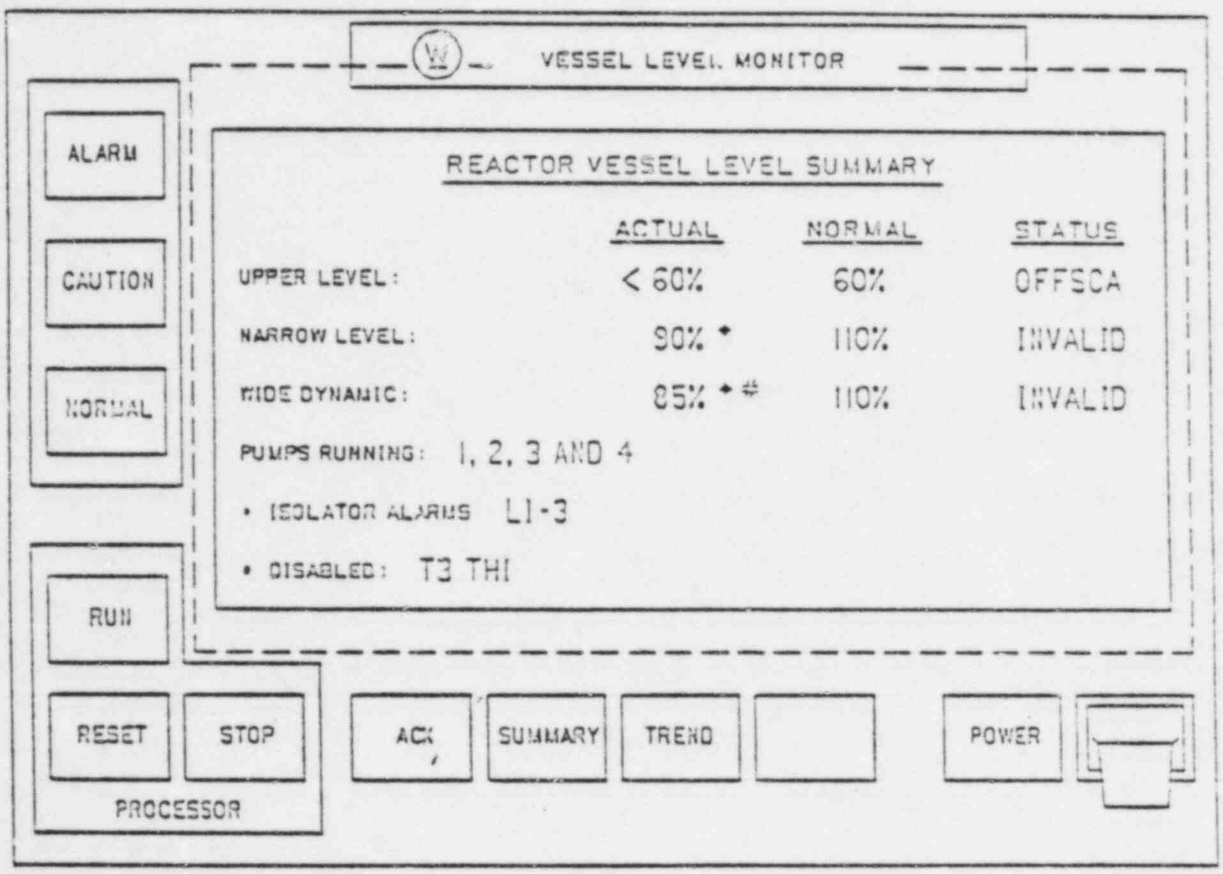

Fig. 10. Vessel level summary display for full power, all pumps on.

ORNL-DWG 82-10188

REACTOR VESSEL LEVEL TREND

TIME MIN	PLENUM LEVEL	VESSEL LEVEL	FLOW HEAD
00	73%	47% I	>110% OS
-15	78%	49% I	98%
-30	79%	62% I	97%
-45	82%	56% I	98%
-60	87%	99% I	99%

Fig. 11. Vessel level trend display.

Displays on Main Processing Unit

The one-line, forty-character, alphanumeric display on the front panel of the main processing unit is used to display individual sensor inputs. The sensor is selected with a two-digit thumbwheel switch.

The following information is to be given for each sensor:

1. Sensor identification
2. Input signal level
3. Input signal converted to engineering units
4. Status of sensor input.

Disabled Inputs

Any inputs can be disabled by the operator. This action is under the control of a keyswitch on the front panel of the main computational unit and causes the processor to disregard the analog input for that variable.

II.C.2.b.1 Resistance temperature detectors (RTD). The Resistance Temperature Detectors (RTD) associated with the RVLIS are utilized to obtain a temerature signal for fluid filled instrument lines inside containment during normal and post accident operation. The temperature measurement for all vertical instrument lines is used to correct the vessel level indication for density changes associated with the environmental temperature change.

The RTD assembly is a totally enclosed and hermetically sealed strap-on device consisting of a thermal element, extension cable and termination cable. The sensitive portion of the device is mounted in a removable adapter assembly which is designed to conform to the surface of the tubing or piping being monitored. The materials are all selected to be compatible with the normal and post accident environment. Randomly selected samples from the controlled (material, manufacturing, etc.) production lot will be qualified by type testing. Qualification testing will consist of thermal aging, irradiation, seismic testing and testing under simulation high energy line break environmental conditions. The specific qualification requirements for the RTDs are as follows:

1. Aging. The thermal aging test will consist of operating the detectors in a high temperature environment: either 400°F for 528 h or per other similar Arrhenius temperature/time relationship.

2. Radiation. The detectors shall be irradiated to a total integrated does (TID) of 1.2×10^8 rads gamma radiation using a CO^{60} source at a minimum rate of 2.0×10^6 rads/h and maximum rate of 2.5×10^6 rads/h. Any externally exposed organic materials shall be evaluated or tested to 9×10^8 rads TID beta radiation.

The energy of the beta particle shall be 6 MeV for the first 10 MRAD, 3 MeV for 340 MRad and 1 MeV for 150 MRad.

3. Seismic. The detectors will be tested using a biaxial seismic simulation. The detectors shall be mounted to simulate a plant installation and will be energized throughout the test.

4. High Energy Line Break Simulation. The detectors shall be tested in a saturated steam environment.

Caustic spray, consisting of 2500 ppm boric acid dissolved in water and adjusted to a pH 10.7 at 25°C by sodium hydroxide, shall be applied during the first 24 h. The test units will be energized

The RTD device is designed to operate over a temperature range of -58 to 500°F (the normal temperature range is 50 to 130°F).

II.C.2.b.2 Reactor vessel level instrumentation system valves. Two types of valves are supplied for the RVLIS. The root valve (3/4 T78) is an ASME Class 2, stainless steel, globe valve. The basic function of the valve is to isolate the instrumentation from the RCS. The other valve (1/4 × 28 ID), is an instrumentation-type valve. It is a manually actuated ball valve used to provide isolation in the fully closed position. The valve is hermetically sealed and utilizes a packless design to eliminate the possibility of fluid leakage past the steam to the atmosphere.

II.C.2.b.3 Transmitters, hydraulic isolators, and sensors differential pressure transmitters. The d/p transmitters are a seismically qualified design as used in numerous other plant applications. In the RVLIS application, accuracy considerations dictate a protected environment, consequently transmitters are rated for 40 to 130°F and 10^4 rad TID.

Several special requirements for these transmitters are as follows:

1. Must withstand long term overloads of up to 300% with minimal effect on calibration.
2. High range and bi-directional units required for pump head measurements.
3. Must displace minimal volumes of fluid in normal and overrange operating modes.

The first two requirements are related to the vernier characteristic of the pumps off level measurements and the wide range measurements, respectively. The third is related to the limited driving displacement of the hydraulic isolator when preserving margins for pressure and thermal expansion effects in the coupling fluids.

The d/p transmitters are rated 3000 psig working pressure and all units are tested to 4500 psig. Internal valving also provides over-range ratings to full working pressure.

Hydraulic Isolator

The hydraulic isolator is a high displacement d/p switch employed as a floating check valve which conveys pressures from the RCS ports to the remote transmitters. In normal operation only small displacements (less than 10% of range) will produce full scale changes in the connected transmitters. It serves as the second valve on the RCS and also serves as air isolation valve on the containment building as applied in the RVLIS application.

Two opposing bellows, liquid filled, are displaced when conveying pressures (or volumes) between the high pressure (HP) and low pressure (LP) ports. Displacement forces are as low as possible being only the elasticity of the bellows material. The bellows displacements become limited when valves on the coupled shaft close, stopping exchange flow of bellows fill fluids. This valves off the higher pressure of the two process connections and stops communication with the lower pressure port. Switch contacts are actuated on approach to these limits to advise operators of the undesirable operating conditions.

The isolators are rated at 3000 psig working pressure and are factory tested at 4500 psig. Principal gaskets are metallic for preservation of pressure boundaries through most severe postulated accident conditions. They are qualified for containment environment applications.

High Volume Sensor

The sensor bellows unit employed in the RVLIS is a new design to accomodate the thermal expansion of the capillary fluids in the long lines.

A second major feature of the RVLIS sensors is the inclusion of a check valve. This valve does not interfere with conveyance of RCS pressures when capillary lines are full and intact. If, however, a capillary line should fail or leak, the valve will be closed by RCS pressure preventing further fluid loss.

The sensor housings are rated for 3000 psig working pressure. All units are hydro tested at 4500 psig. The principal gasketing is metallic for ensured integrity in the event of accident exposures to either radiation or high temperatures.

II.C.2.c Accuracy. An uncertainty of 6% was established as the design target for the differential pressure level measurement system.

This is obtained by a statistical combination of all uncertainties including environmental effects (if any) on the instrumentation. For the upper range instrument, this corresponds to an allowable deviation of ±1 ft. This will give the operator a good estimate of the steam or gas volume in the upper head during a situation in which the head vent would be employed. For the narrow range instrument, this corresponds to an allowable deviation of about ±2.5 ft. This is required to provide adequate margin against inadvertant use of the ICC operating guideline and to assure that the vessel level reading can be reasonable used to the detection of the onset of ICC conditions. It is also adequate to indicate useful information regarding vessel level behavior during the vessel refill period following a LOCA transient.

The staff requested that Westinghouse describe in detail how the system uncertainties were derived. In particular, they were asked to detail how the individual uncertainties from the system components were combined, to detail the random and systematic errors. Their response[17] is included here.

Response

The system accuracy of ±6% water level was a target value established during the conceptual design and was related to the dimensions of the reactor vessel (12% from nozzles to top of core) and core (30%), and the usefulness of the measurement during an accident. Subsequent analyes have established a system accuracy based on the uncertainties introduced by each component in the instrument system. The individual uncertainties, resulting from random effects, were combined statistically to obtain the overall instrument system accuracy. Some of the individual uncertainties vary with conditions such as system pressure. The following table identifies the individual uncertainties for the narrow range measurement while at a system pressure of 1200 psia.

	Component and uncertainty definition	Uncertainty % level
a.	Differential pressure tansmitter calibration and drift allowance, (±1.5% of span) multiplied by the ratio of ambient to operating water density.	±2.1
b.	Differential pressure transmitter allowance for change in calibration due to ambient temperature change (±0.5% of span for ±50°F) multiplied by the density ratio.	±0.7

Component and uncertainty definition		Uncertainty % level
c.	Differential pressure transmitter allowance for change in calibration due to change in system pressure (±0.2% of span per 1000 psi change) multiplied by the density ratio.	±0.34
d.	Differential pressure transmitter allowance for change in calibration due to exposure to long-term overrange (±0.5% of span) multiplied by the density ratio.	±0.7
e.	Reference leg temperature instrument (RTD) uncertainty of ±5°F and or allowance of ±5°F for the difference between the measurement and the true average temperature of the reference leg, applied to each vertical section of the reference leg where a measurement is made. Stated uncertainty is based on a maximum containment temperature of 420°F, and a typical reference leg installation.	±0.64
f.	Reactor coolant density based on auctioneering for highest water density obtained from hot leg temperature (±6°F) or system pressure (±60 psi). Magnitude of uncertainty varies with system pressure and water level, with largest uncertainty occurring when the reactor vessel is full.	±2.3
g.	Sensor and hydraulic isolator bellows displacements due to system pressure changes or reference leg temperature changes will introduce minor errors in the level measurement due to the small volumes and small bellows spring constants. The changes, such as pressure or temperature, tend to cancel, i.e., the bellows associated with each measurement move in the same direction. Maximum expected error due to differences in capillary line volume and local temperatures is equivalent to a level change of about 5 in., multiplied by the density ratio.	±1.46

	Component and uncertainty definition	Uncertainty % level
h.	Density function generator output mismatch with ASME Steam Tables limited to a maximum of:	±0.50
i.	Electronics system calibration, overall uncertainty limited to less than:	±1.0
j.	Control board indicator resolution.	±0.5

The statistical combination (square root of the sum of the squares) of the individual uncertainties described above results in an overall system instrumentation uncertainty of ±3.9% of the level span for the narrow range indication of approximately 40 ft, or ±1.5 ft, at a system pressure of 1200 psia. Examples of the uncertainty at other system pressures are:

uncertainty = ±3.6% at 400 psia
uncertainty = ±4.2% at 2000 psia
uncertainty = ±4.6% at 2250 psia

Range

The wide range instrument will cover the full range of expected differential pressures with all reactor coolant pumps running. The maximum span of the wide range instrument will change with the number of pumps operating. The operator must be aware of the maximum span for a given number of operating pumps. Both the narrow range and the upper range instrument indications should be set to indicate that the vessel is full with the pumps tripped.

Time Response

The d/p instrument response time shall not exceed 10 s. This time delay is defined as the time required for the display instrument to reach the midpoint of a 50% step input d/p change.

Plant Startup Calibration

During the plant startup, subsequent to installing the RVLIS, a test program will be carried out to confirm the system calibration. The program will cover normal operating conditions and will provide a reference for comparison with a potential accident condition. The elements of the program are described below:

1. During refilling and venting of the reactor vessel, measurements of all 6 d/p transmitters would be compared to confirm identical level indications.

2. During plant heatup with all reactor coolant pumps running, measurements would be obtained from the wide range d/p transmitters to confirm or correct the temperature compensation provided in the system electronics. The temperature compensation, based on a best estimate of the flow and pressure drop variation during startup, corrects the transmitter output so that the control board indication is maintained at 100% over the entire operating temperature range.

3. At hot standby, measurements would be obtained from all transmitters with different combinations of reactor coolant pumps operating, to provide the reference data for comparison with accident conditions. For any pump operating condition, the reference data, represents the normal condition, i.e., with a water-solid system. A reduced d/p during an accident would be an indication of voids in the reactor vessel.

4. At hot standby, measurements would be obtained from the reference leg RTDs, to confirm or correct reference leg temperature compensation provided in the system electronics.

II.C.2.d Operating procedures

Purpose

The objectives of these instructions are to establish the requirements for the use of the Reactor Vessel Level Instrumentation System (RVLIS) for various plant conditions and to specify the maintainability requirements of the system equipment.

Prerequisites

- The capillary lines have been vacuum filled, per the installation instructions.
- Ensure that the hydraulic isolators are zeroed (within ± 0.1 in.^{3}).
- Calibrate the d/p cells per instructions of ITT Barton Manual for Model 752, Level B, transmitters.
- The process equipment must be scaled using the appropriate scaling document.
- Determine the height of the upper top piping above the inside top of the vessel.

Initialization

With the plant less than 200°F and less than 430 psig, obtain the following data for trains A and B:

(1) With an automatic data logger, record the following:
- T_{hot}
- RCS pressure
- d/p transmitter output
- signal to the remote display

(2) Manually record:
- level indication readings
- hydraulic isolator dial readings
- reference leg RTD output

(3) Record the above data for the following reactor coolant pump operations:

NOTE

The various configurations should be obtained through the normal startup if possible

NOTE

Upper plenum will read offscale if pump is running in the instrumented loop; narrow range will read offscale with one or more pumps running.

- No pumps running

NOTE

An indication of 100% reading represents a level to the inside top of the vessel. The height of the upper top piping above the inside top of the vessel will result in a reading greater than 100%. This added height is plant specific and must be determined prior to adjusting the process equipment (upper plenum and narrow range) for full scale in ication.

- one noninstrumented loop pump running
- two noninstrumented loop pumps running

- two noninstrumented loop pumps and one instrumented loop pump running
- all pumps running -- adjust process equipment so that wide range indication reads 100%.

(4) With all pumps running, increase RCS pressure - temperature to T_{avg} no-load and record data refer to step (1) every 50°F increment. Data of step (2) should be recorded at 350°F and at T_{avg} no-load. Adjust process electronics for density compensation at T_{avg} no-load. Verify that wide range indication reads 100%.

(5) Trip all pumps and record data per steps (1) and (2). Verify that upper plenum and narrow range indication is in agreement with the reading of step (3) "no pumps running".

(6) Restart pumps in sequence and record wide range readings for both n trains for each pump combination.

(7) Enter into the equipment programming the expected percent level for the various pump combinations per the microprocessor instruction manual.

Normal Plant Operation

With the plant at power, the level readings should be as follows:

Wide range	~110% (wide range reading will increase from 100% to approximately 110% with all pumps running, as reactor power is increased from zero to 100%).
Narrow range	Off scale - high.
Upper plenum	Off scale - low (RCP status light on main control board is off)

Any reduction in wide range expected readings (with all pumps running) can only be caused by the presence of voids in the circulating water. Voids will not exist without reduced pressure which could trip the reactor, so all accident conditions will proceed from a condition of zero power (100% reading on the wide range). Check that the pressure has decreased or that subcooling meter confirms saturation conditions exist; then readings below 100% are an indication of voids in the coolant.

If the actual readings differ from the expected readings by 3% for a single train, refer to Troubleshooting below.

If the indication for both trains differs from the expected readings, refer to the emergency operating instructions for immediate and subsequent action.

Refueling

After depressurization and prior to lifting the reactor vessel head, perform the following steps to prepare the RVLIS:

(1) Close reactor vessel level head connection isolation valve.
(2) Disconnect piping between the isolation valve and the sensors.

NOTE
Contaminated water residue may be in the pipe.

(3) Provide temporary plugs for the pipe ends of the removable section and stationary sections.

Restore the RVLIS after reactor vessel head installation as follows:

(1) Remove pipe end plugs and reconnect piping section.
(2) With the isolation valve open, backfill the piping from sensors by attaching a water source to the sensor vent.
(3) Disconnect waterfill apparatus.
(4) At startup (450 psig. 200°F), visually inspect piping/coupling of the reinstalled piping for leakage.
(5) At full system pressure, repeat inspection.

Periodic Testing

Plant at Power

Perform monthly calibration checks of the process electronics in accordance with the process equipment instruction manual.

Refueling Outages

(1) For the d/p transmitters, perform zero check of each d/p transmitter by closing the respective isolation valves and

opening the bypass valve. If zero reading differs from the last recorded reading by (*) percent, then recalibrate d/p transmitter using instructions of Barton Instruction Manual (Model 752) and the instruction contained in RVLIS system manual and the appropriate equipment instruction manuals.

(2) Record the appropriate hydraulic isolator dial readings and compare results with previous cold shutdown readings. Readings should be within plus or minus 0.1 cu. in.

(3) Perform the calibration check of the process electronics in accordance with the equipment technical manual.

(4) Verify the operability of the RVLIS during the startup/heatup of plant following a refueling or major plant outage by tracking the displays of the two trains. Readings should be within (*) percent of the previous recorded readings.

Every other Refueling Outage

In addition to the steps of above, perform the following every other refueling outage:

(1) At the process equipment cabinets, read the impulse line RTD resistances.

NOTE

Take the ambient temperature reading near the RTD and adjust the measured resistance accordingly. Compare the adjusted resistance to the original results or the previous recorded data.

(2) Employing a pneumatics calibration, per instructions for normal plant operation at the sensor vent ports, check the calibration of the transmitters and perform a time response check of the system. The calibration results should be within plus or minut (*) percent of instrument span of the previous recorded data. The time response of the system should be within 10 s. This is the time required for the display instrument to reach the midpoint of a 50% step input variable change.

(*)Number to be provided by Westinghouse.

II.C.2.e <u>Troubleshooting, plant at power</u>. If a single indication varies from the expected value, check the following:

(1) Compare hydraulic isolator dial reading with reading taken from diverse train and those taken at T_{avg} no-load conditions. Dial readings deviating by more than ± 0.1 cu. in. may be indicative of potential capillary line leakage: however, it may not be the reason for the deviation in the display reading until the isolator reached the valve-off point.

(2) Perform a calibration check of the process equipment, per the appropriate instruction manual.

(3) Perform a zero check of the appropriate d/p transmitter. If more than one indicator/display deviates from the diverse train or from T_{avg} no-load readings, check the following:

- common isolator dial readings versus previous reading
- d/p transmitter valve lineup
- process equipment power supplies

If repairs are required to the capillary lines, the system must be vacuum-filled and calibrated per the instructions contained in the RVLIS System Manual and the appropriate equipment instruction manuals.

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III. DESCRIPTION OF LEVEL SYSTEMS TESTS AND ANALYSES

III.A Performance Tests of Westinghouse d/p System

A variety of test programs have been carried out to study the static and dynamic performance of the RVLIS at two test facilities, and to calibrate the system over a range of normal operating conditions at each reactor plant where the system is installed. These programs, which supplement the vendors' tests of hydraulic and electrical components, will provide the appropriate verification of the system response to accident conditions as well as the appropriate procedures for proper operation, maintenance and calibration of the equipment. A description of these programs is presented in the following section.

III.A.1. Westinghouse Tests

Forest Hills

A breadboard installation consisting of one train of a RVLIS was installed and tested at the Westinghouse Forest Hills Test Facility. The system consisted of a full single train of RVLIS hydraulic components (sensor assemblies, hydraulic isolators, isolation and bypass valves and d/p transmitters) connected to a simulated reactor vessel. Process connections were made to simulate the reactor head, hot leg and seal table connections. Capillary tubing which in one sensing line simulated the maximum expected length (400 ft) was used to connect the sensor assemblies to the hydraulic isolators and all joints were welded. Connections between the hydraulic isolators, valves and transmitters utilized compression fittings in most cases. Resistance temperature detectors, special large volume sensor bellows and volume displacers inside the hydraulic isolator assemblies which are normally part of a RVLIS installation were not included in the installation since elevated temperature testing was not included in the program.

The hydraulic isolator assemblies and transmitters were mounted at an elevation slightly below the simulated seal table elevation.

The objectives of the test were as follows:

1. Obtain installation, filling and maintenance experience.
2. Prove and establish filling procedures for initial filling and system maintenance.
3. Establish calibration and fluid inventory maintenance procedures for shutdown and normal operation conditions.

4. Prove long term integrity of hydraulic components.

5. Verify and quantify fluid transfer and makeup requirements associated with instrument valve operation.

6. Verify leak test procedures for field use.

Reactor Vessel Simulator

The reactor vessel simulator consisted of a 40 ft long, 2 in. diam stainless steel pipe with taps at the top, side and bottom to simulate the reactor head, hot leg and incore detector thimble conduit penetration at the bottom of the vessel. Tubing (0.375 in. diam) was used to connect this lower tap to the sensor at the simulated seal table elevation and the hot leg sensor to the head connection was simulated by 1 in. tubing which connected the sensor to the vessel.

The reactor vessel simulator was designed for a pressure rating of 1400 psig to comply with local stored energy and safety code considerations.

Installation

The system was installed in the high bay test area of the Westinghouse Forest Hills Test Facility by Westinghouse personnel under the supervison of Forest Hills Test Engineering. All local safety codes were considered in the construction. No details of these test results have been published.

Test Results

During test at Forest Hills, the details of the system design initial set-up and calibration were worked out. These procedures were utilized during the following tests at SEMISCALE.

III.A.2 SEMISCALE Tests

A Westinghouse Reactor Vessel Level Measurement System (RVLIS) was installed on the SEMISCALE out-of-reactor facility at the Idaho National Engineering Laboratory (INEL) so that indications of water level could be compared with well characterized differential pressure level sensors and gamma densitometers that are a part of the SEMISCALE instrumentation.

SEMISCALE is a scaled, highly instrumented non-nuclear model of 1 1/2 loops of a pressurized water reactor. The vertical dimensions are similar to a full sized reactor, but the diameter is essentially that of one fuel bundle. Several tests were scheduled at this facility during

1981 and the Westinghouse RVLIS system was installed for testing during several of these tests.

Table 2. SEMISCALE Tests with Westinghouse RVLIS

Test No.	Test Type
S-UT-3	2 1/2% cold-leg break
S-UT-6	5% cold-leg break
S-UT-7	5% cold-leg break with UHI
S-NC-2B	natural circulation, single-, two-phase, and reflux
S-NC-3	natural circulation two-phase,
S-NC-8	natural circulation, reflux
S-UT-8	repeat of S-UT-6 with modified guide tube

III.A.2.a Test S-UT-3. An analysis of the performance of the Westinghouse RVLIS showed that the RVLIS gave comparable readings to the SEMISCALE level sensors over similar spans. The Westinghouse system appears in most cases to give a conservative estimate (lower than) of the coolant inventory with a two-phase mixture. The Westinghouse level readings were compared to collapsed water levels measured by SEMISCALE instruments, and anomalous readings were obtained, however, when the system spaned the upper core support plate. Differences up to 150 cm (60 in.) were observed between the RVLIS and the SEMISCALE level sensors in this case. The apparent cause of the differences was an atypicality in the SEMISCALE construction that resulted in a flow restriction in the guide tubes structure between the upper head and upper plenum that caused an additional pressure drop.

III.A.2.a.1 Response Time of the RVLIS. Differences in the dynamic response between the SEMISCALE level instrumentation and the Westinghouse RVLIS were found. This was assumed to be due to the much longer lines used to connect the RVLIS to the test loop (45, 60, and 70 m or 150, 200, and 250 ft as compared with a few cm for the SEMISCALE instruments). The expected response times for the RVLIS were calculated from.[17]

When the above calculation was carried out, a time constant of 2.7 s was found. This agrees with that observed in a special test performed at SEMISCALE (Westinghouse Special Pressure Test Procedure SC-WSPT-20) to confirm the response.

III.A.2.b Tests S-UT-6 and S-UT-7. The results of the two tests, S-UT-6 and S-UT-7, were similar to those found in test S-UT-3. The

Westinghouse RVLIS provided a conservative estimate of the coolant level. In S-UT-6, comparing water levels rather than coolant levels, large non-conservative difference were observed between the SEMISCALE instrumentation and the RVLIS in the upper head and upper plenum level. This is shown in Fig. 12. In S-UT-7, the RVLIS was used only between the hot leg and bottom of the vessel, eliminating the region of suspected friction losses. The agreement with the SEMISCALE instrumentation in this span was quite good (see Fig. 13).

III.A.2.c Natural circulation tests S-NC-2B, S-NC-3, and S-NC-8. The preliminary data from the natural circulation tests S-NC-2B, S-NC-3, and S-NC-8, showed a maximum difference of 3.7% (1.7 ft) for the conditions encountered in these tests.

III.A.2.d Test S-UT-8. Test S-UT-8 was a rerun of test S-UT-6 after modification of the SEMISCALE guide tube to provide a drainage path similar, between the upper head and upper plenum, to that of a Westinghouse PWR. An analysis of the data from this test supplied by SEMISCALE shows that with the modified guide tube, the difference in the RVLIS decreased from about 5 in. of water to 2 in. after about 100 s, when compared to SEMISCALE dP measurement. This difference is equivalent to 0.5% of the level measurement. Figure 14 shows a comparison of the RVLIS narrow range and a SEMISCALE dP measurement. Additional results from this test are shown in Fig. 15, where the liquid level indication is compared with SEMISCALE densitometer readings. In this test, the RVLIS gave a conservative estimate (lower than) of both liquid level and coolant level in the vessel.

During most of the SEMISCALE tests the RVLIS performed as expected and provided a conservative estimate of the amount of coolant in the test vessel under two-phase conditions. Discrepancies found in the comparison of water level in the first series of tests appear to have been due to the atypical construction of the SEMISCALE guide tube between the upper head and the upper plenum. When the guide tube was modified to more closely resemble the upper internals of a Westinghouse reactor (for S-UT-8), the RVLIS and SEMISCALE level measurements of the liquid level agreed closely.

III.A.3 Plant Startup Calibration

During the plant startup, subsequent to installing the RVLIS, a test program will be carried out to confirm the system calibration. The program will cover normal operating conditions and will provide a reference for comparison with a potential accident condition. The elements of the program are described below:

1. During refilling and venting of the reactor vessel, measurements of all 6 d/p transmitters would be compared to confirm identical level indications.

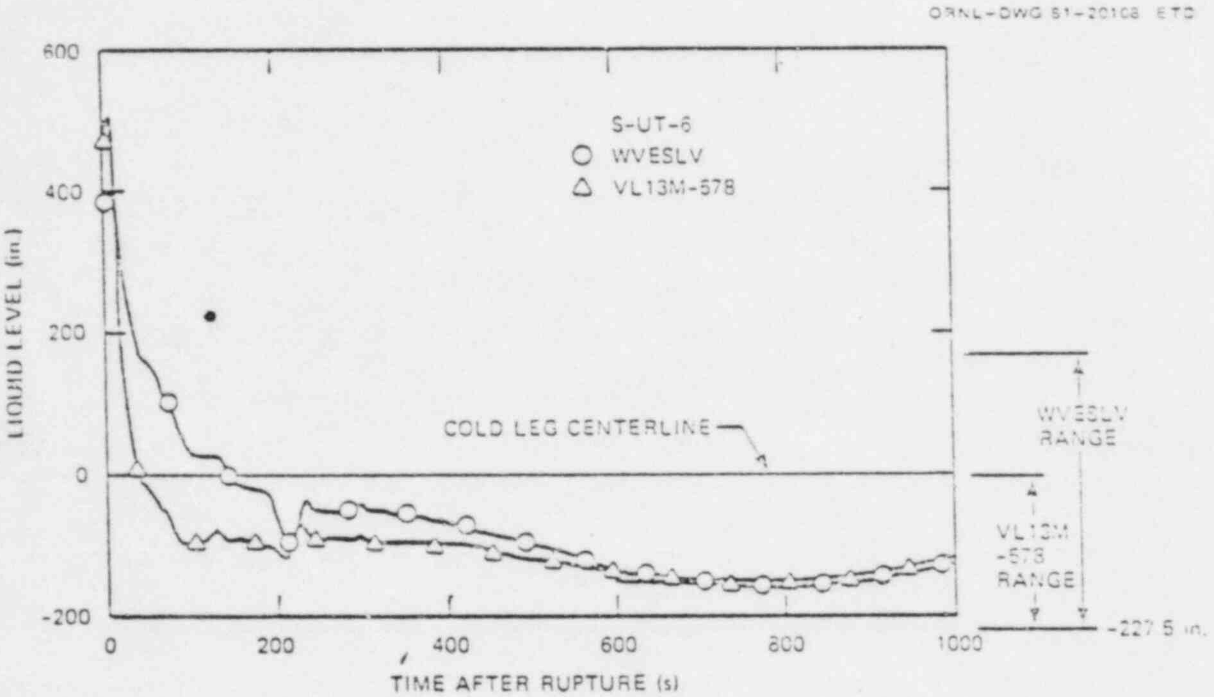

Fig. 12. Westinghouse vessel level measurement compared to SEMISCALE d/p 4 (VLM-578) for S-UT-6 (ref. 18).

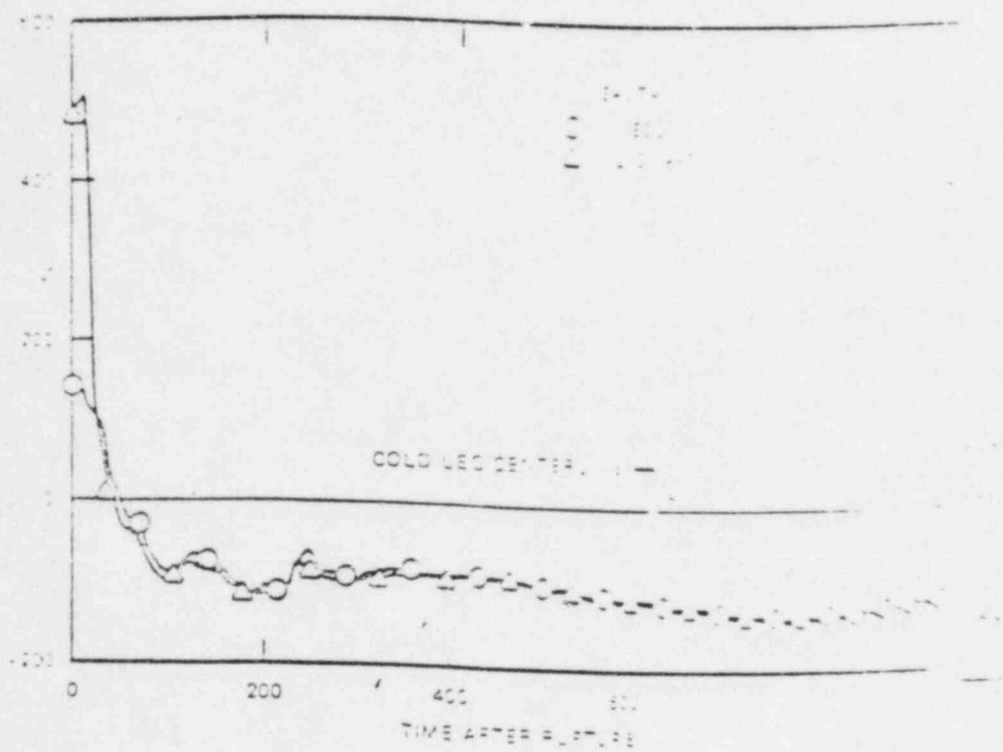

Fig. 13. Comparison of Westinghouse [illegible]
[illegible]MISCALE d/p 4 (VL13M-578) for S-07-[illegible]

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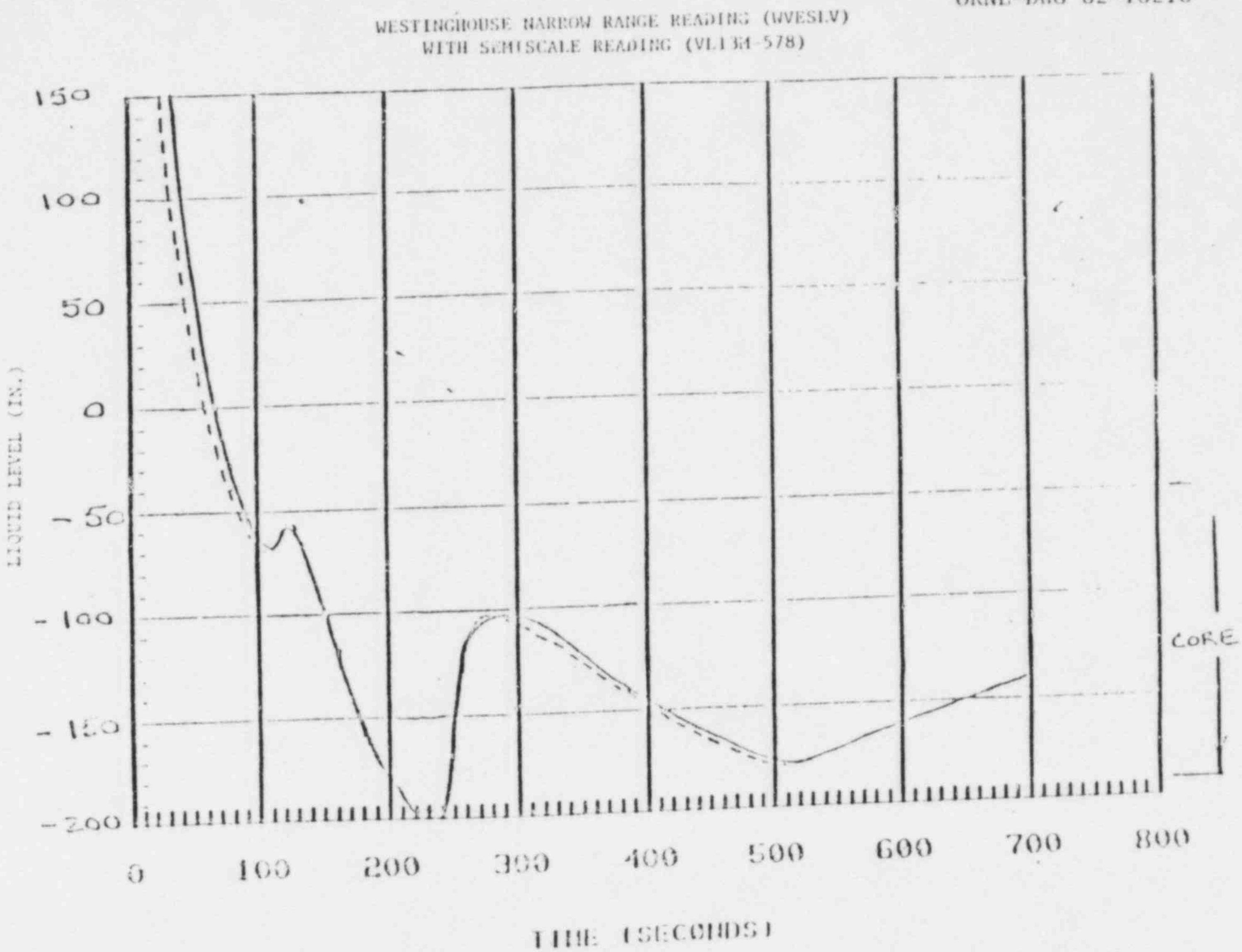

Fig. 14. Comparison of RVLIS narrow range indication with SEMISCALE differential pressure measured from the hot let to the bottom of the vessel for test S-UT-8.

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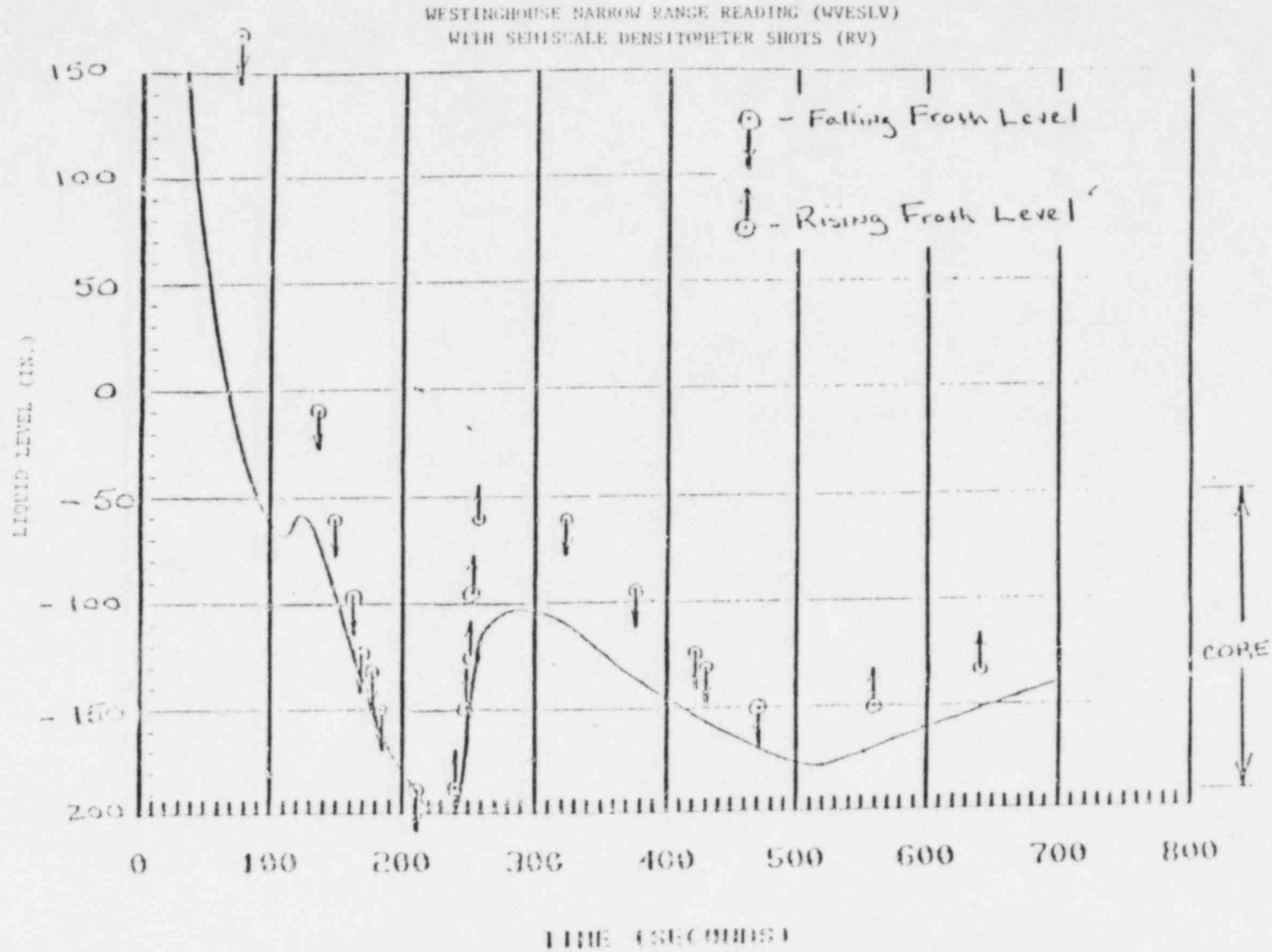

Fig. 15. Comparison of RVLIS narrow range with two-phase mixture level determined from densitometer indications.

2. During plant heatup with all reactor coolant pumps running, measurements would be obtained from the wide range d/p transmitters to confirm or correct the temperature compensation provided in the system electronics. The temperature compensation, based on a best estimate of the flow and pressure drop variation during startup, corrects the transmitter output so that the control board indication is maintained at 100% over the entire operating temperature range.

3. At hot standby, measurements would be obtained from all transmitters with different combinations of reactor coolant pumps operating, to provide the reference data for comparison with accident conditions. For any pump operating condition, the reference data represents the normal condition, i.e., with a water-solid system. A reduced d/p during an accident would be an indication of voids in the reactor vessel.

4. At hot standby, measurements would be obtained from the reference leg RTDs to confirm or correct reference leg temperature compensation provided in the system electronics.

III. B. Analyses of Performance

Each train of the RVLIS is capable of monitoring coolant mass in the vessel from normal operation to a condition of complete uncovery of the reactor core. This capability is provided by the three d/p transmitters, each transmitter covering a specific range of operating conditions. The three instrument ranges provide overlap so that the measurement can be obtained from more than one display under most accident conditions. Capabilities of each of the measurements are described below:

1. Reactor Vessel - Upper Range

The transmitter span covers the distance from the hot leg piping connection to the top of the reactor vessel. With the reactor coolant pump shutdown in the loop with the hot leg connection, the transmitter output is an indication of the level in the upper plenum or upper head of the reactor vessel. The measurement will provide an accurate indication for guidance when operating the reactor vessel head vent.

When the pump in the loop with the hot leg connection is operating, the d/p would be greater than the transmitter span, and the transmitter output would be deleted from the digital panel. An invalid status statement would be indicated.

2. Reactor Vessel - Narrow Range

Non UHI-Plant

The transmitter span covers the total height of the reactor vessel. With pumps shutdown, the transmitter output is an indication of the

collapsed water level, i.e., as if the steam bubbles had been separated from the water volume. The two-phase level is slightly higher than the indicated water level since there will be some quantity of steam bubbles in the water volume. Therefore, the RVLIS provides a conservative indication of the level effective for adequate core cooling.

When reactor coolant pumps are operating, the d/p would be greater than the transmitter span, and the transmitter output would be deleted from the digital display panel. An invalid status statement would be indicated.

UHI Plant

The transmitter span covers a partial height of the reactor vessel, from the bottom of the vessel to hot leg nozzle elevation. With pumps off, the transmitter output is an indication of the collapsed water level, i.e., as if the steam bubbles had been separated from the water volume. The two-phase level is slightly higher than the indicated water level since there will be some quantity of steam bubbles in the water volume. Therefore, the RVLIS provides a conservative indication (lower than) of the level effective for adaquate core cooling. When reactor coolant pumps are operating, the dP would be greater than the transmitter span, and the transmitter output would be disregarded.

3. Reactor Vessel - Wide Range

In a non-UHI plant the transmitter spans from the bottom of the vessel to the top of the vessel; in a UHI plant the top connection is to the hot leg.

The transmitter span covers the entire range of interest, from all pumps operating with a water-solid system to a completely empty reactor vessel and, therefore, covers the measurement spans of the other two instruments. (In the UHI plant, the wide range d/p covers the narrow range span.) Any reduction in d/p compared to the normal operating condition is an indication of voids in the vessel. The reactor coolant pumps will circulate the water and steam as an essentially homogeneous mixture, so there would be no distinct water level in the vessel. When pumps are not operating, this transmitter output is an additional indication of the level in the vessel, supplementing the indications from the other instruments.

The output of each transmitter is compensated for the density difference between the fluid in the reactor vessel and the fluid in the reference leg at the initial ambient temperature. The compensation is based on a wide range hot leg temperature measurement or a wide range system pressure measurement, whichever results in the highest value of water density, and, therefore, the lowest value of indicated level.

Compensation based on temperature is applied when the system is subcooled, and compensation based on pressure (saturated conditions) is applied if superheat exists at the hot leg temperature measurement point.

The output of each transmitter is also compensated for the density difference between the fluid in the reference leg during an accident with elevated temperature in the containment and the fluid in the reference leg at the initial ambient temperature. The compensation is based on temperature measurements on the vertical sections of the reference leg.

For those systems not employing the digital display, the corrected transmitter outputs are displayed on panel meters - one for each transmitter. Also there is a warning light under the upper range to indicate if the pump in the loop with the hot leg connection is operating, to indicate that the meter should be disregarded.

The corrected transmitter outputs are shown on a digital display installed on the control board, one statement for each measurement in each train. Two three-pen recorders are also provided on the control board to record the level or relative d/p and to display trends in the measurements from each train. The display would also indicate which reactor coolant pumps are operating, and which level measurements are invalid due to pump operation.

During normal plant heatup or hot stand by operation with all reactor coolant pumps operating, the wide range d/p display would indicate 100% on the display, an indication that the system is water-solid. If less than all pumps are operating, the display would indicate a lower d/p (determined during the plant startup test program) that would be an indication of a water-solid system. With pumps operating, the narrow range and upper range displays would indicate off-scale.

If all pumps are shutdown, at any temperature, the narrow range and upper range displays would indicate 100%, an indication that the vessel is full. The wide range d/p display would indicate about 33% (15% for UHI plants) of the span of the display, which would be the value (determined during the test program) corresponding to a full vessel with pumps shutdown.

In the event of a LOCA where coolant pressure has decreased to a predetermined setpoint, existing emergency procedures would require shutdown of all reactor coolant pumps. In these cases, a level will eventually be established in the reactor vessel and indicated on all of the displays. The plant operator would monitor the displays and the recorder to determine the trend in fluid mass or level in the vessel, and confirm that the ECCS is adequately compensating for the accident conditions to prevent ICC.

Future procedures may require operation of one or more pumps for recovery from certain types of accidents. When pumps are operating while voids are developing in the system, the pumps will circulate the water and steam as an essentially homogeneous mixture. In these cases, there

will be no discernible level in the reactor vessel. A decrease in the measured d/p compared to the normal operating value will be an indication of voids in the system, and a continuously decreasing d/p will indicate that the void content is increasing, that mass is being lost from the system. An increasing d/p will indicate that the mass content is increasing, that the ECCS is effectively restoring the system mass content.

III.B.1 Discussion of Analysis Using NOTRUMP Code

The analyses performed[7,11] for the 1 in. and 4 in. diam cold leg breaks demonstrated the need for the assumption of multiple failures in order to achieve inadequate core cooling conditions, i.e., the loss of all high head safety injection and additionally the loss of the accumulators (locked out) in the latter case. The results show that core exit thermocouples may be used as a reliable indication that ICC conditions are occurring.

For the 1 in. break the results indicate that most of the recovery techniques, initiated when core exit thermocouples reach 1200°F, are effective and provide long term cooling to the core:

Restoration of high pressure safety injection results in beginning of core recovery in less than 2 min after operator action, and complete core recovery at 10 min.

Opening of the secondary system steam dump valves leads to depressurization of the RCS and subsequent delivery by the low head safety injection system, and complete core recovery in less than 3 min after operator action.

With inventory remaining in the system, reactivation of a reactor coolant pump leads to complete core recovery in 20 s after operator action; as discussed above, it is expected that the system will slowly depressurize and LHSI will be activated before core exit temperatures of 1200°F are achieved. Note: The case studied is conservative in that reactivation of the pump in the broken loop maximizes inventory lost to the break; the qualitative conclusions are applicable to cases in which pumps are restarted in any loop.

III.B.2 RVLIS Analytical Analysis

In order to evaluate the usefulness of the RVLIS during the approach to ICC, it was decided to determine the response of the RVLIS under a variety of fluid conditions. The RVLIS response was analytically determined for a number of small break transients. The response was determined by calculating the pressure difference between the upper head and lower plenum and converting this to an equivalent vessel head in feet. (Note that RVLIS indications will actually be represented by percent of span.) Saturation density at the fluid temperature in the upper plenum was used for this conversion. This approximates the calibration that will be used for the RVLIS.

This indication corresponds to the RVLIS configuration used for non-UHI plants. The conclusions of the study are expected to be the same for the UHI configuration. The indication of the upper span (hot leg to upper head) is not included in this analysis. The upper span indication will be used for head venting operations and will not be used to indicate the approach to ICC.

When the reactor coolant pumps are not operating, the RVLIS reading will be indicated on the narrow range scale ranging from zero to the height of the vessel. A full scale reading (100% of span) is indicated when the vessel is full of water. This reading represents the equivalent collapsed liquid level in the vessel which is a conservative indication of the approach to ICC. The RVLIS indication can alert the operator that a condition of ICC is being approached and the existence of ICC can be verified by checking the core exit thermocouples. When the reactor coolant pumps are operating, the narrow range RVLIS meter will be pegged at full scale.

When the reactor coolant pumps are operating, the RVLIS reading will be indicated on the wide range scale which reads from 0 to 100%. The 100% reading corresponds to a full vessel with all of the pumps in operation.

With the pumps runnings the RVLIS reading is an indication of the void fraction of the vessel mixture. As the void content of the vessel mixture increases, the density decreases and the RVLIS reading will decrease due to the reduction in static head and frictional pressure drop. The latter effect will be enhanced by degradation in the reactor coolant pump performance. When this reading drops to approximately 33%, there will also be an indication on the narrow range scale. This fraction approximately corresponds to a vessel mass at which would just cover the core if the pumps were tripped.

Four small-break transients under a variety of conditions are discussed in the next section. Three of these cases were obtained from WFLASH analyses and the other was obtained from the ICC analysis using NOTRUMP. A description of these codes can be found in refs. 7 and 11.

The transients included in this report are listed in Table 3 which gives a brief description of the transient, the plant type, and the model used for the analysis. A discussion of each transient is provided in the next section. Figures 16 through 27 provide plots of vessel two-phase mixture level, RVLIS narrow range reading, mixture and vessel void fraction, and for Case B with pumps running, RVLIS wide range reading and cold leg mass flowrate.

The two-phase mixture level plotted is that which was predicted by the codes for the mixture height below the upper support plate. The RVLIS reading that would be seen is plotted on the same figure for ease of comparison.

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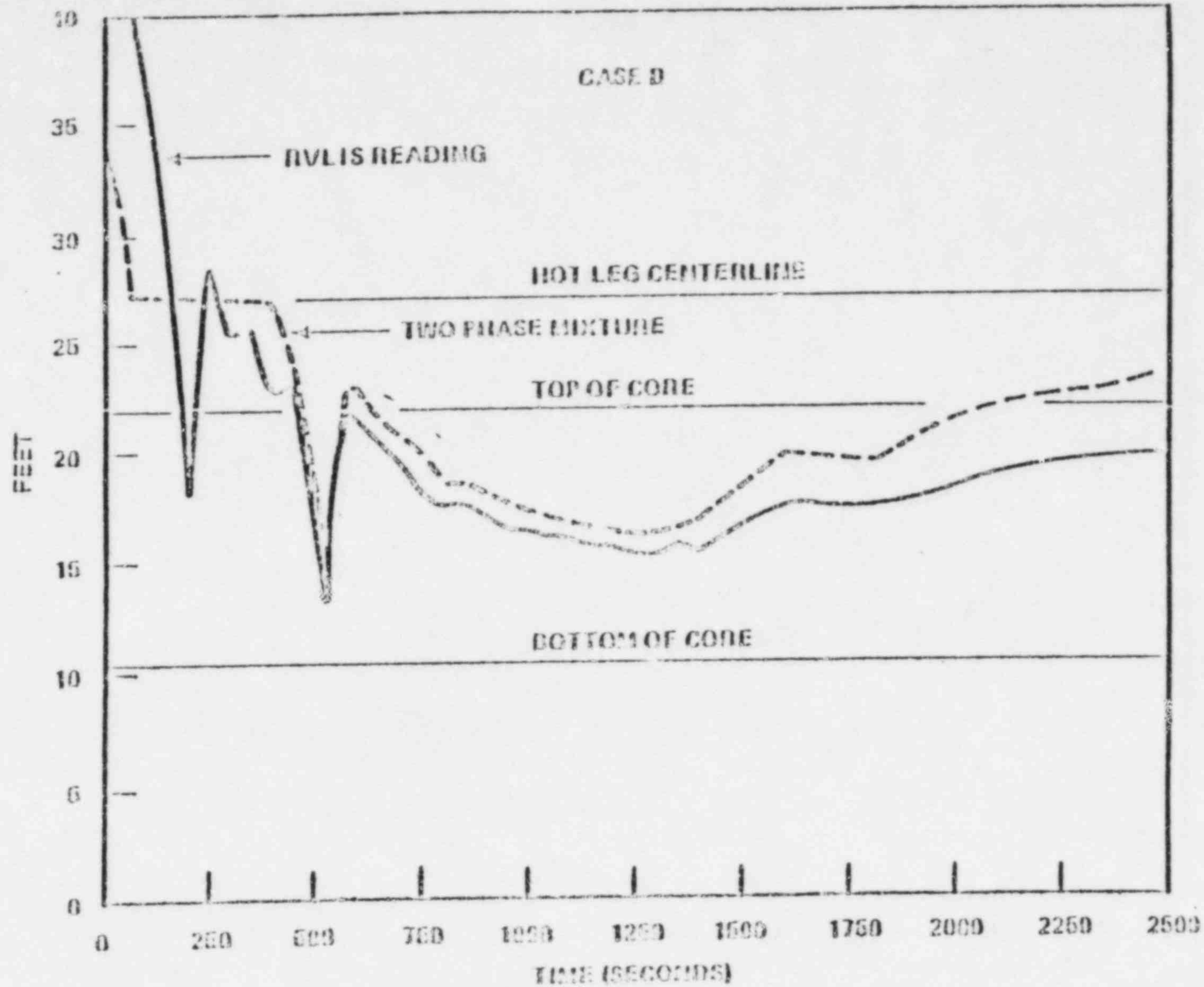

Fig. 16. Case A 3-loop plant, 3 in. cold leg break, pump trip with reactor trip, RVLIS reading and vessel mixture level.

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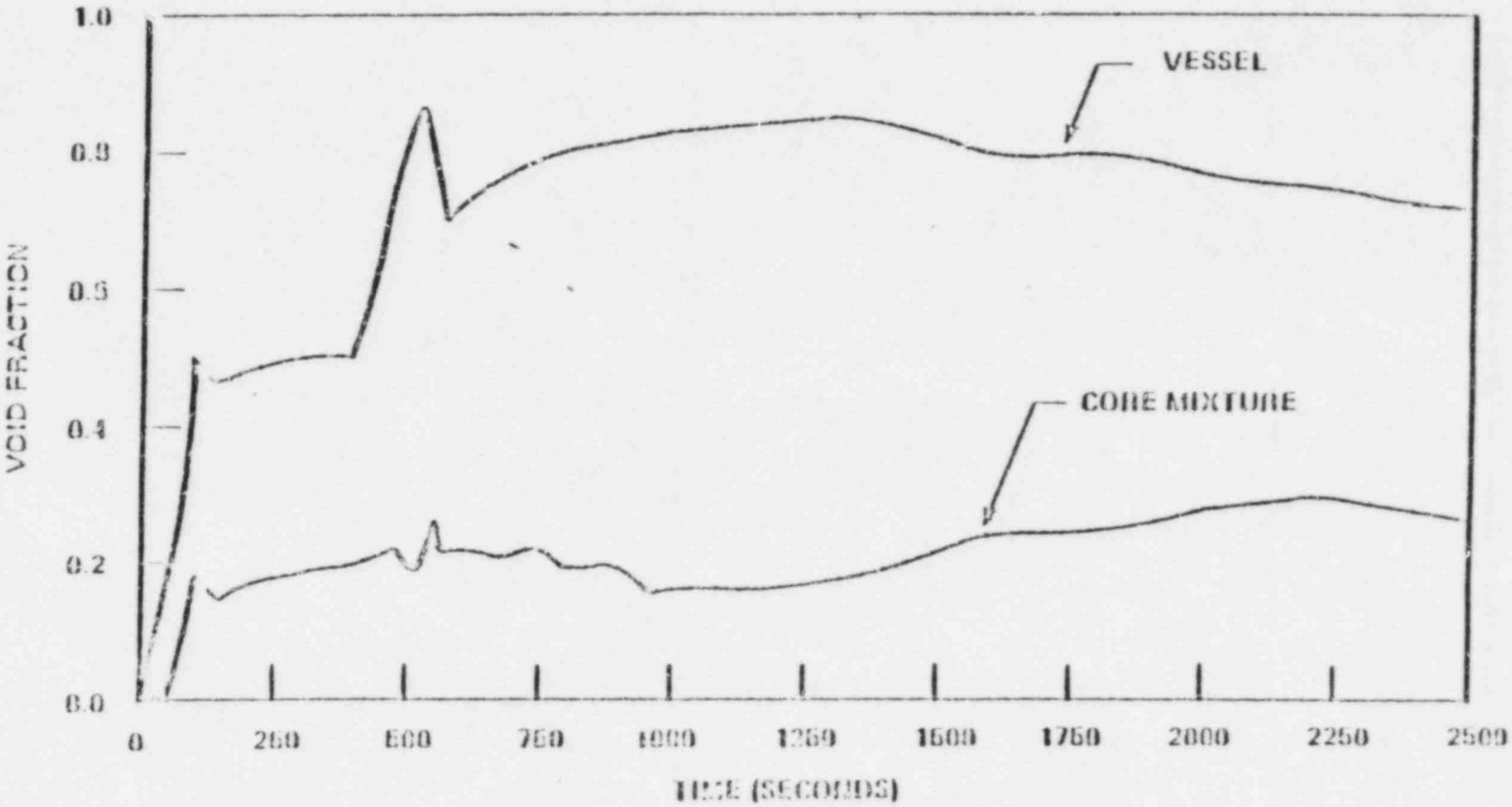

Fig. 17. Case A 3-loop plant, 3 in. cold leg break, pump trip with reactor trip, void fraction.

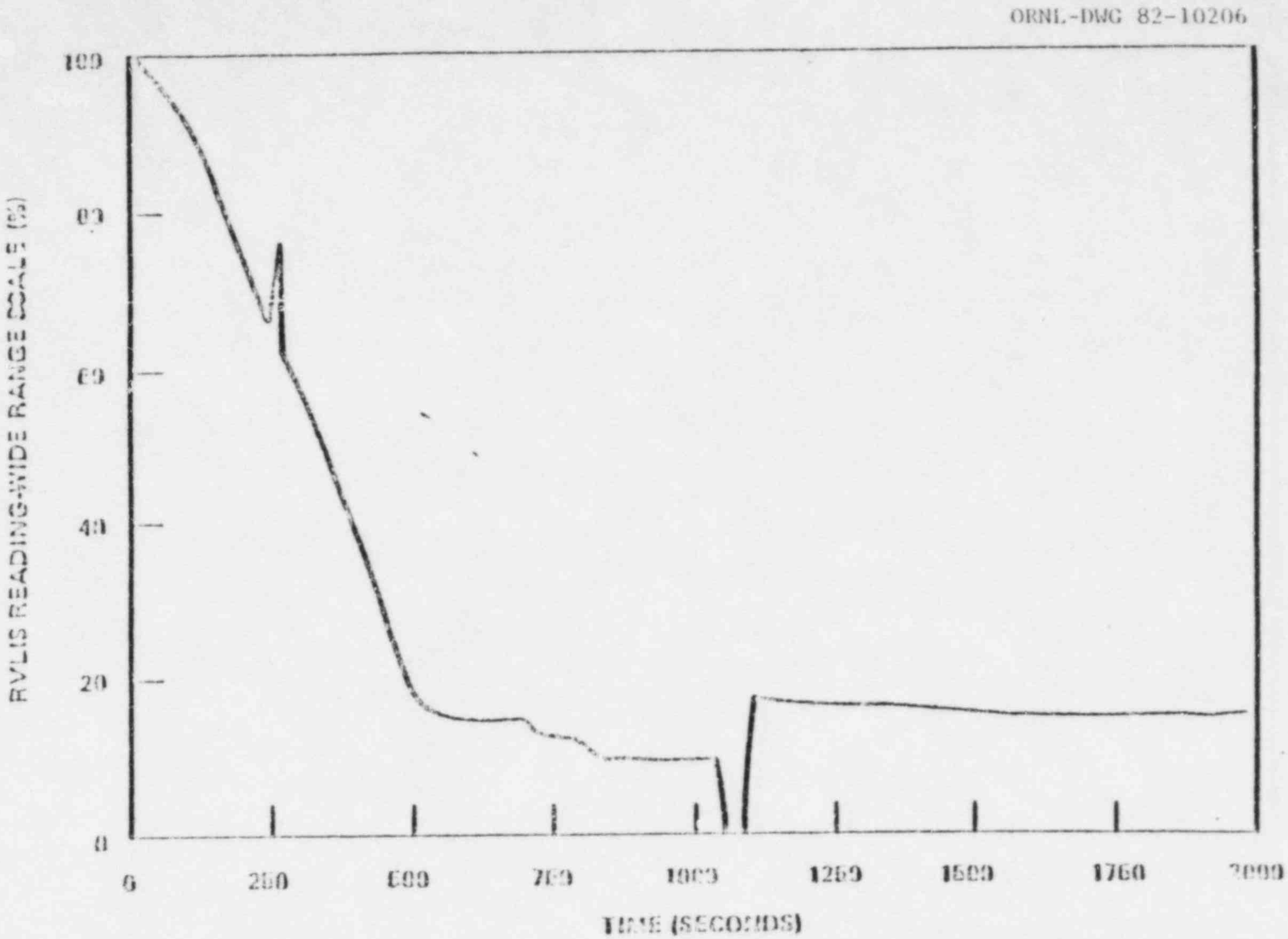

Fig. 18. Case B 3-loop plant, 3 in. cold leg break, pump trip at 750 s, wide range reading.

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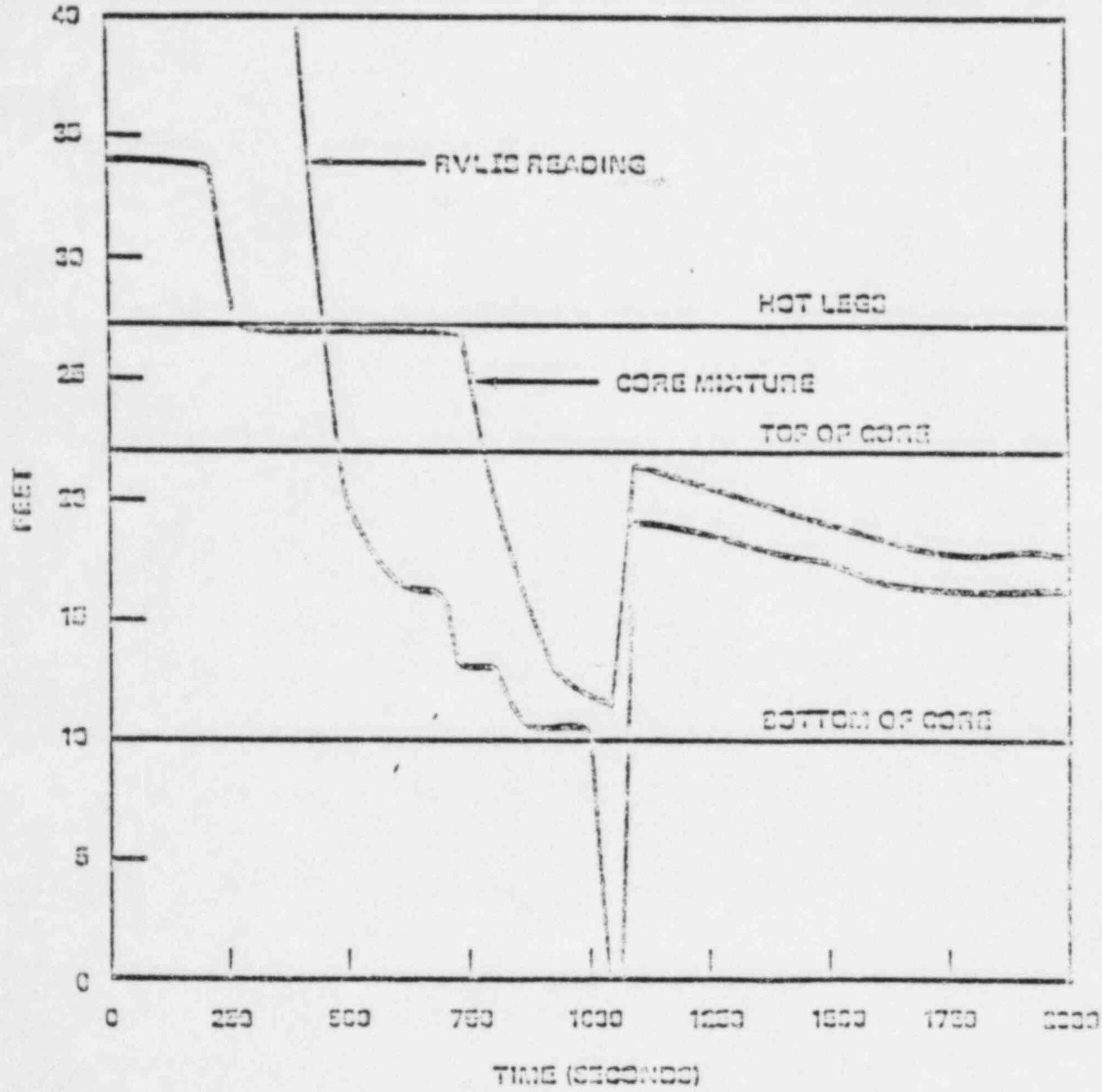

Fig. 19. Case B 3-loop plant, 3 in. cold leg break, pump trip at 750 s, RVLIS reading and mixture level.

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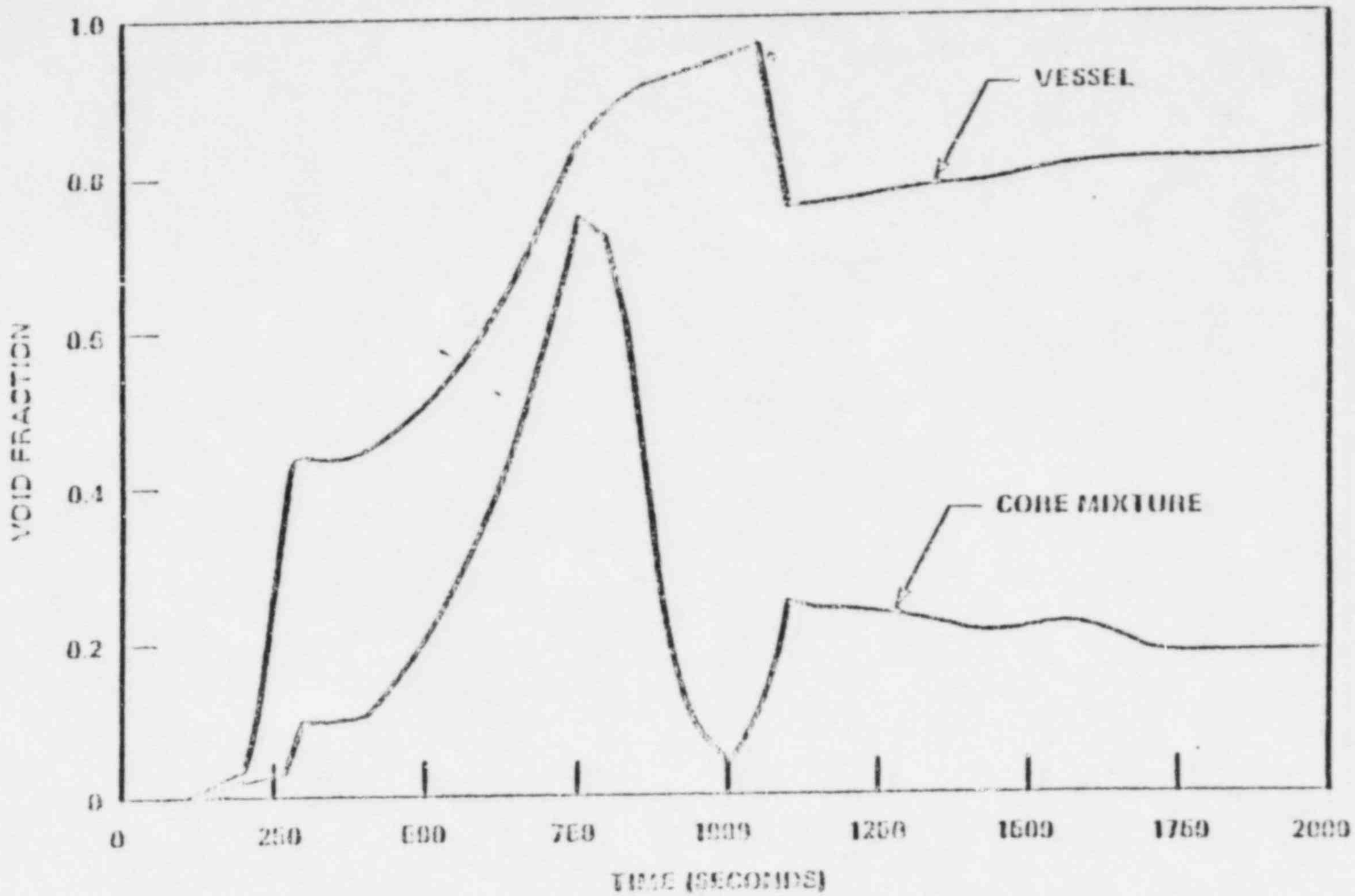

Fig. 20. Case B 3-loop plant, 3 in. cold leg break, pump trip at 750 s, void fraction.

ORNL-DWG 82-10194

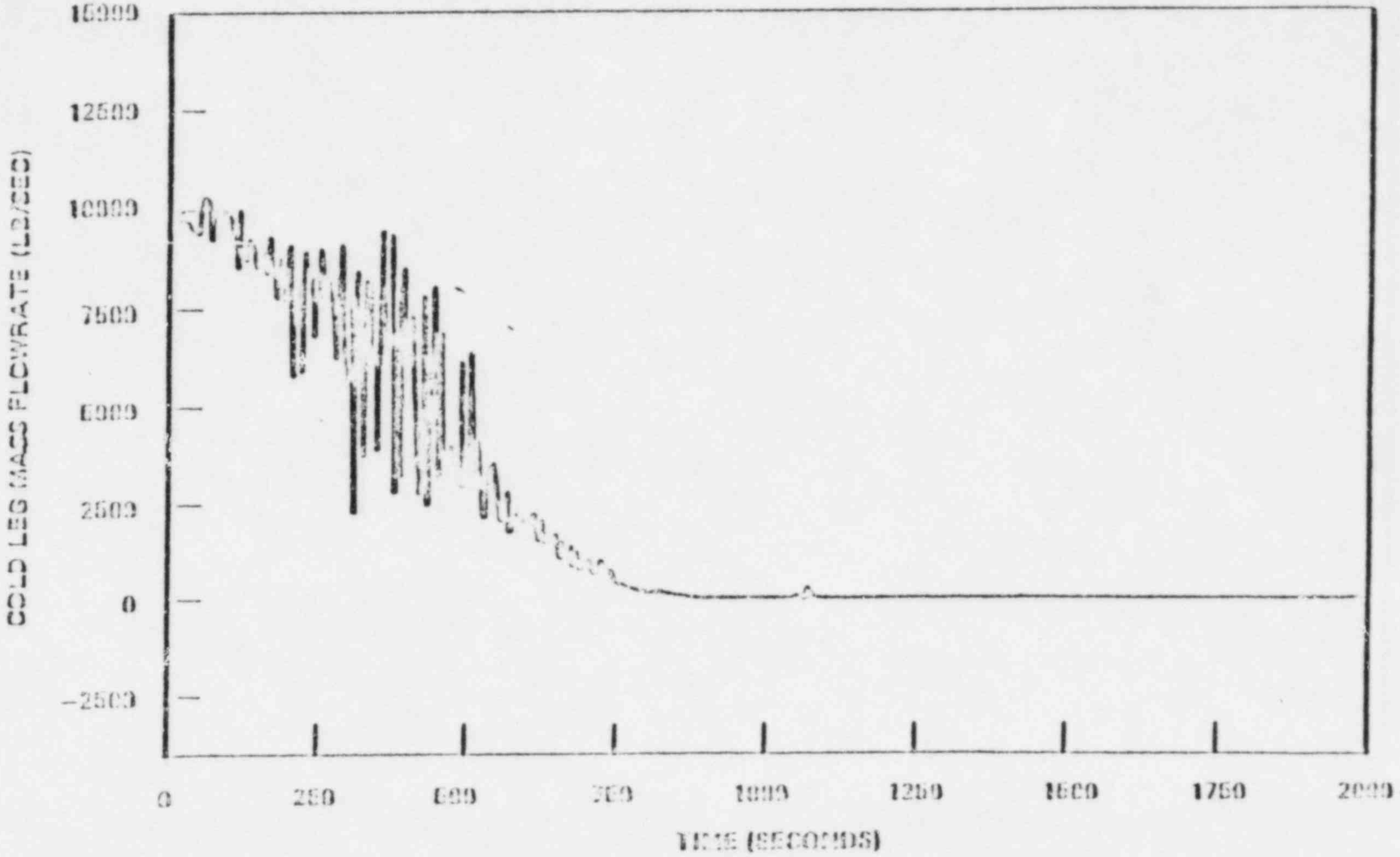

Fig. 21. Case B 3-loop plant, 3 in. cold leg break, pump trip at 750 s, cold leg mass flowrate (LB/s).

ORNL-DWG 82-10193

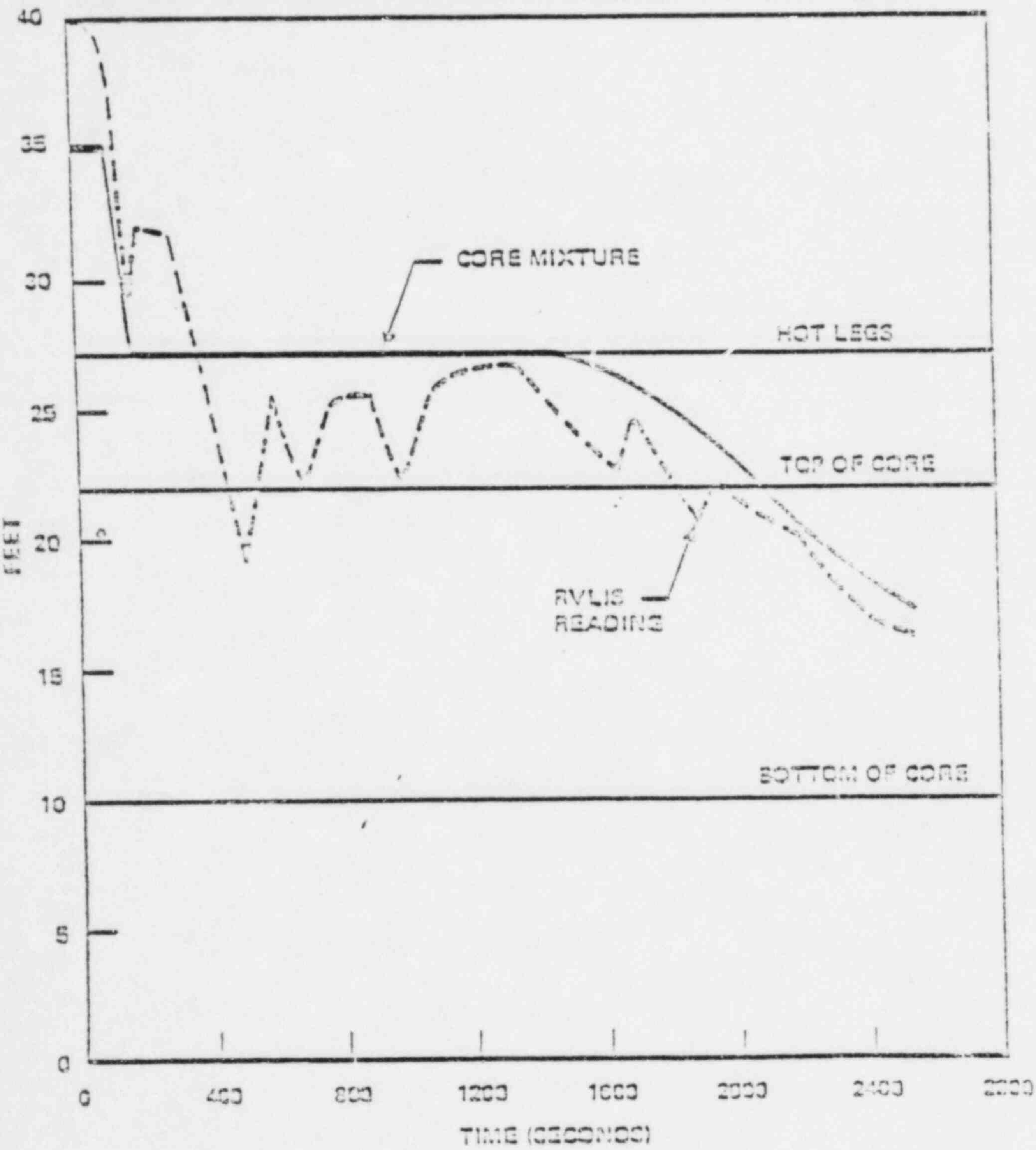

Fig. 22. Case C 2.5 in. pressurizer break, no. SI, RVLIS reading and mixture level.

ORNL-DWG 82-16192

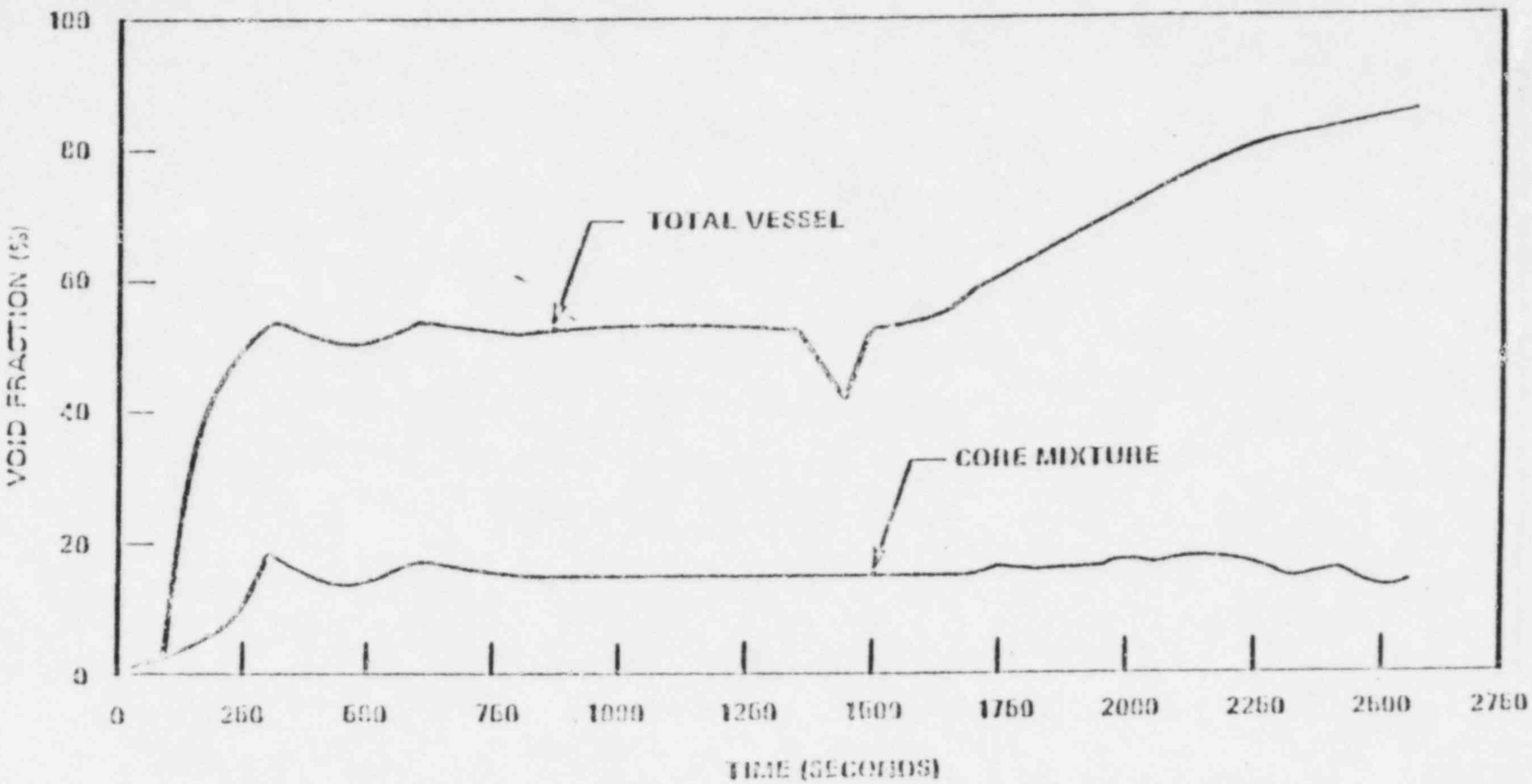

Fig. 23. Case C 2.5 in. pressurized break, no. SI void fraction.

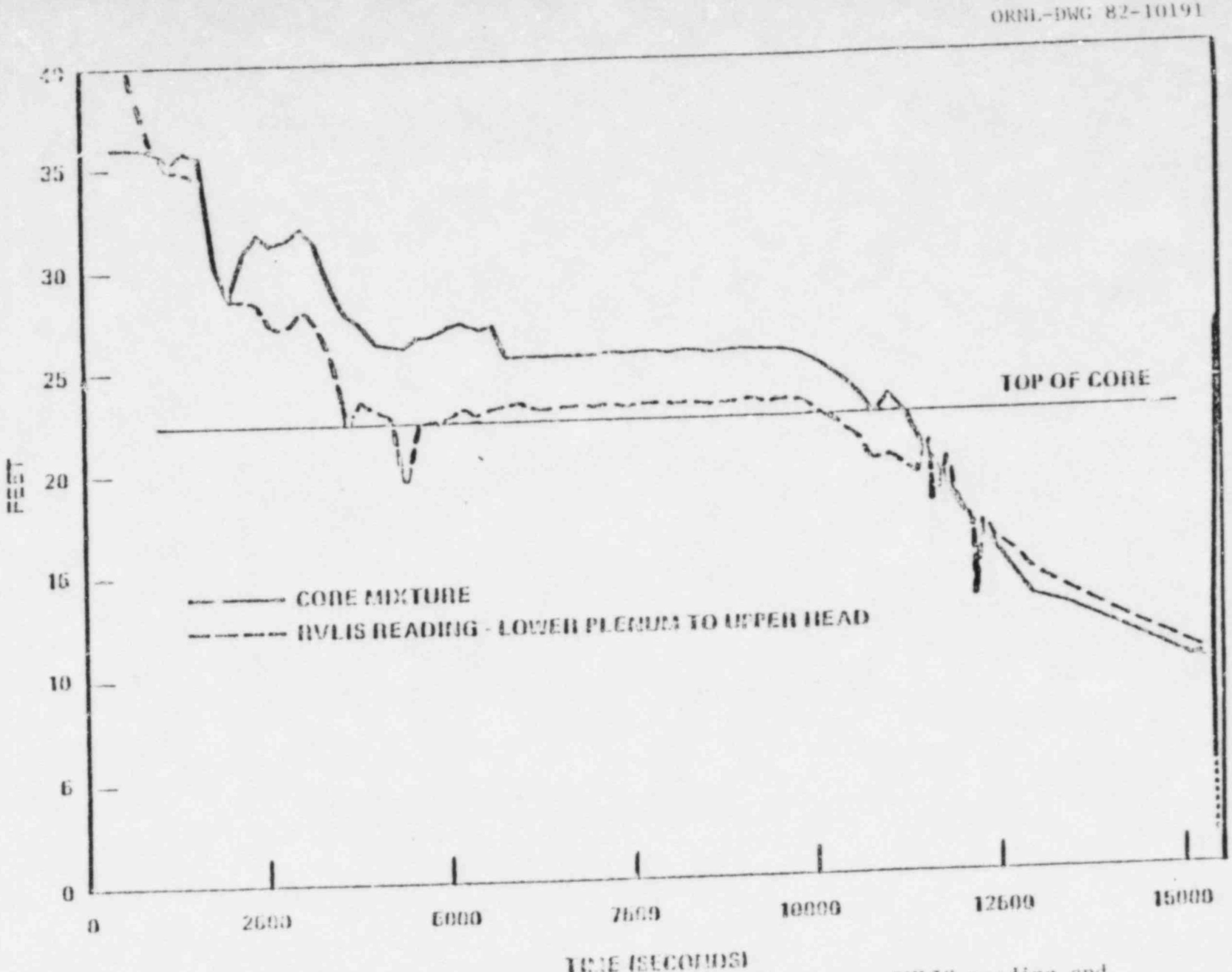

Fig. 24. Case D 1 in. cold leg break, ICC case, RVLIS reading and mixture level.

ORNL-DWG 82-10190

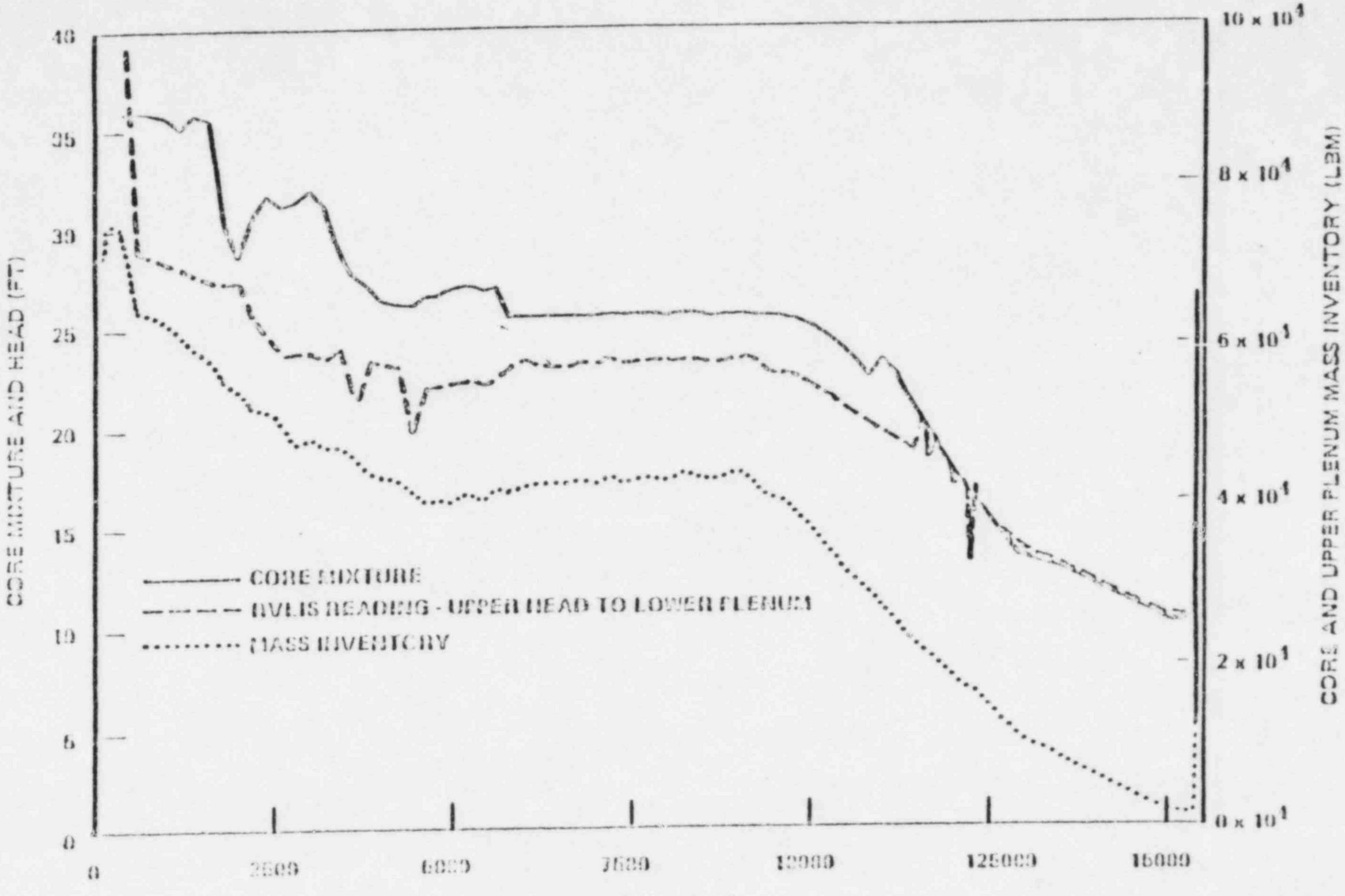

Fig. 25. Case b 1 in. cold leg break, ICC case, mixture level, RVLIS reading and measured inventory.

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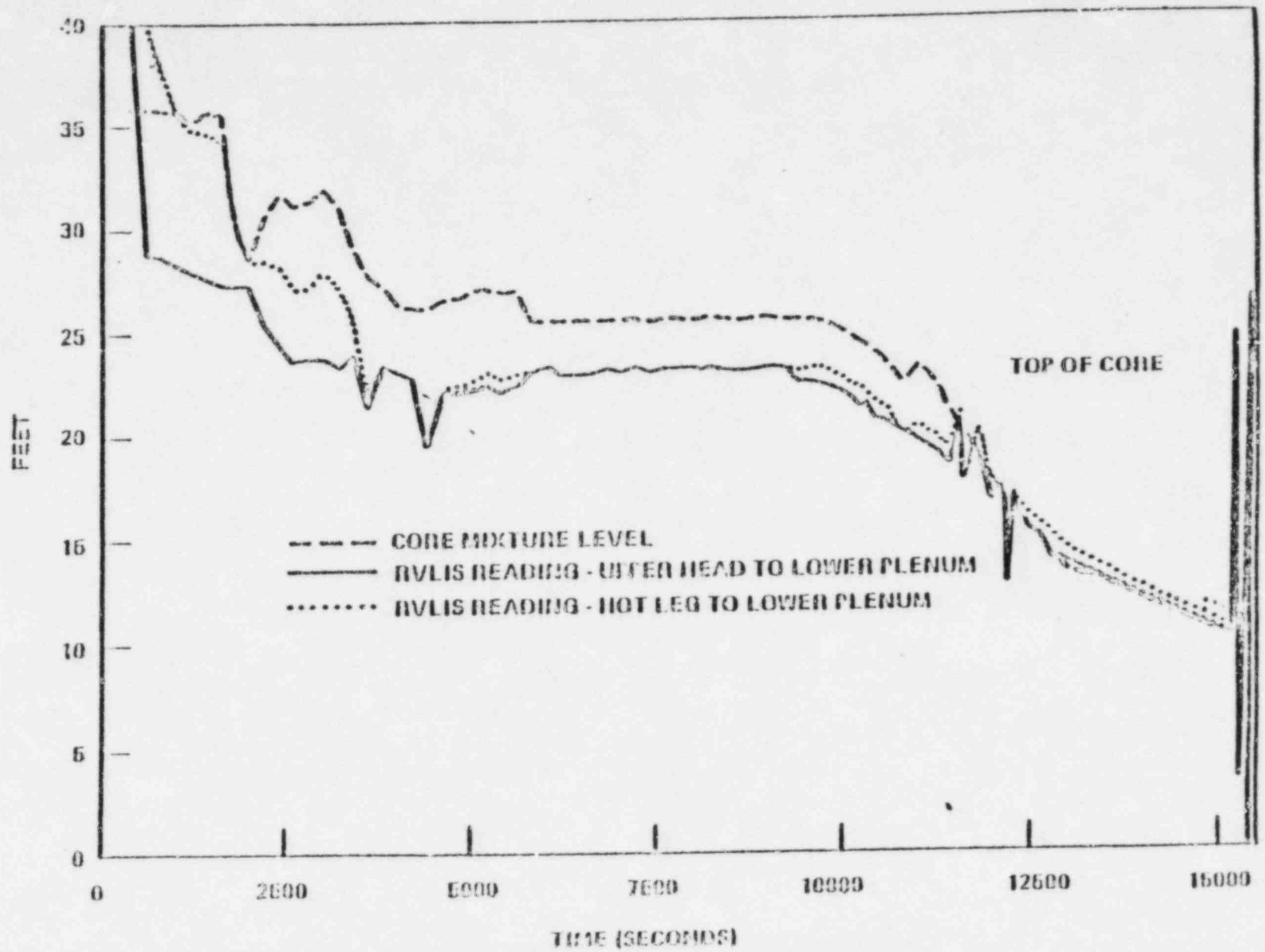

Fig. 26. Case D 1 in. cold leg break, ICC case, void fraction.

ORNL-DWG 82-10197

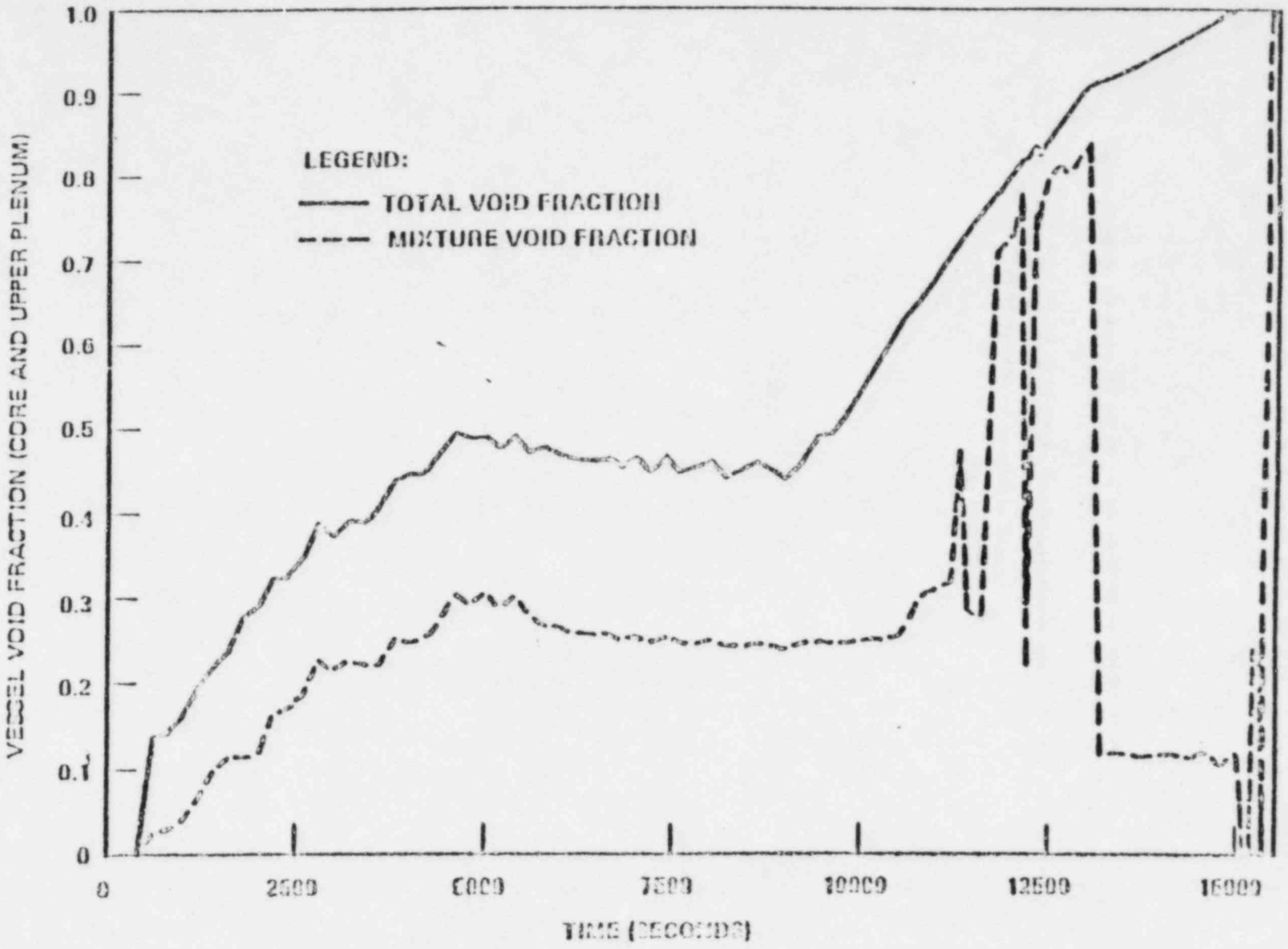

Fig. 27. Case D 1 in. cold leg break, ICC case, void fraction.

Table 3. Transients investigated

Case	Plant	Description
A	3 loop 2775 MWt	3 in. cold leg break - FSAR assumptions*; WFLASH
B	3 loop MWt	3 in. cold leg break - RCPs trip at 750 s - 2775 otherwise, FSAR assumptions; WFLASH
C	4 loop UHI type 3411 MWt	2.5 in. break in top of pressurizer - no UHI - no pumped safety injection - pumps not running; WFLASH
D	4 loop Non-UHI 3411 MWt	1 in. cold leg break - no high head safety injection; NOTRUMP

*RCPs tripped at reactor trip, minimum pumped safety injection is available, minimum auxiliary feedwater is available.

The void fraction plots are for the core and upper plenum fluid volumes. The mixture void fraction includes the volume below the two-phase mixture level while the total void fraction also includes the steam space above the mixture level.

III.B.3 Transients Investigated

Case A

The initiating event for this transient is a 3 in. break in the cold leg. After the break opens, the system depressurizes rapidly to the steam generator secondary safety valve setpoint. Consistent with the FSAR asumptions, the reactor coolant pumps are assumed to trip early in the transient when the reactor trips.

The system pressure hangs up at the secondary setpoint until the loop seal unplugs at approximately 550 s, allowing steam to flow out the break and the depressurization continues. The core uncovers while the loop seal is draining then recovers when the loop seal unplugs. The core then begins to uncover again as more mass is being lost through the break than is being replaced by safety injection. The core begins to recover at about 1500 s when the accumlators begin to inject.

This transient does not represent a condition that would lead to ICC but it does represent a break size in the range that would be most probable if a small-break did occur. The response of the RVLIS for typical conditions for which it would be used can be investigated with this transient.

After the reactor coolant pumps trip the RVLIS reading drops rapidly to the narrow range scale. It falls until the pressure drop due to flow becomes insignificant compared to the static head of the fluid in the vessel. The first dip in the RVLIS reading is due to the behavior of the upper head.

When the upper head starts to drain it behaves like a pressurizer. The pressure in the upper head remains high until the mixture level drops to below the top of the guide tube where steam is allowed to flow from the upper head to the upper plenum. When this occurs the upper head pressure decreases - thereby increasing the vessel d/p - and the RVLIS reading again more accurately reflects the vessel inventory. This phenomenon is more prevalent for large-break sizes and the effect will be of brief duration for breaks in this range. Furthermore, the ICC guidelines require verification of the RVLIS reading through the use of the core exit thermocouples. During this phenomenon, the core exit thermocouples would read near saturation temperature. Therefore, this early phenomena in the upper head will not cause a false indication of ICC.

When the vessel begins to drain during the loop seal uncovery the RVLIS reading trends in the same direction as the vessel level. The RVLIS reading remains below the vessel mixture level and is therfore a conservative indication.

When the vessel mixture level increases after the loop seal unplugs the RVLIS reading follows it. Then, RVLIS readings continue to follow the vessel mixture level throughout the transient while underpredicting the actual two-phase level. The wider difference between the RVLIS level and the two-phase level later in the transient is due to the system being at a lower pressure which allows more bubbles to exist in the mixture.

Case B

This case is the same as Case A except it was assumed that the reactor coolant pumps continued to operate until 750 s. If the reactor coolant pump trip criteria is followed the pumps would be tripped much earlier in the transient. This case is, however, instructive in determining the RVLIS response when the pumps are running.

After the break opens, the system depressurizes rapidly to the secondary safety valve setpoint, and then begins a period of very slow depressurization. During this time the upper portions of the system drain. Due to the reactor coolant pump operation, the two-phase mixture in the vessel remains at the hot leg elevation, although the void fraction of the mixture continues to increase.

At 750 s (12.5 min) the system has drained to the point that steam can be vented through the break and the system begins to depressurize more rapidly. The pumps are also tripped at this time resulting in a collapse of the mixture in the vessel and the core uncovers.

The vessel continues to drain until the accumulators inject at about 1000 s (16.6 min) to recover the core. There is a subsequent uncovery which will be ended when the pressure is low enough for the safety injection to make up for mass lost through the break.

During the early portion of the transient, the wide range RVLIS reading drops fairly smoothly from 100% to about 20%, which is due to the decreasing mass in the vessel and the decreasing pressure drop as the pump performance is degraded. The plot of cold leg mass flowrate is indicative of the pump degradation. The oscillations in this plot are due to alternate steam and two-phase flow predicted by WFLASH. When the flow through the pump becomes mostly steam, the increasing void fraction of the vessel mixture becomes the predominant factor in the decreasing RVLIS reading.

RCP operation keeps the steam and water mixed enough that the mixture level does not fall below the hot legs, although the mixture void fraction is increasing during this time. This loss of inventory is indicated by the continued drop in the RVLIS reading. When the pumps trip, the steam and water in the mixture separate and there is a rapid decrease in the core mixture level and mixture void fraction although the vessel void fraction continues to rise. The fact that mass is being redistributed rather than lost is seen in the RLVIS reading - there is little change in the reading (compared to the change in level) from 750 s to the time that the accumulators come on.

The prolonged reactor coolant pump operation has caused the downcomer to drain so that when the accumulators come on the cold accumulator water condenses steam in the downcomer causing a local depressurization. The downcomer pressure is then temporarily lower than the upper head pressure due to inertia and the RVLIS reading becomes temporarily negative.

This period of erratic indication is brief (one or two min). The pressure will equilibrate and the RVLIS will resume following the vessel mixture level. This phenomenon has only been observed when the accumulators inject when the downcomer is highly voided. There is no apparent discrepancy during accumulator injection when there is a significant amount of water in the downcomer. It is believed that this effect is exaggerated by the modeling techniques used in WFLASH (which utilize a homogeneous equilibrium assumptions at the accumulator injection location). For the remainder of the transient the RVLIS reading follows the vessel level closely.

Case C

The initiating event for this transient is the opening of the pressurizer power operated relief valves (PORVs). The reactor coolant pumps and the reactor trip early in the transient on a low pressurizer pressure signal consistent with FSAR assumptions. Auxiliary feedwater is available in this case but, no pumped safety injection is assumed.

The pressurizer mixture level rises to the top of the pressurizer early in the transient and stays at this level throughout most of the transient. The flow through the PORVs alternates between steam and two-phase mixture while the pressure in the system drops rapidly to the steam generator secondary safety valve setpoint. The pressure hangs up at this value until the upper portion of the system has drained and then continues to decrease. When the upper portions of the primary system (excluding the pressurizer) have drained the vessel mixture level begins to decrease and continues until the core completely uncovers.

The RVLIS reading drops rapidly to the narrow range span after the reactor coolant pumps are tripped. When the vessel level reaches the hot leg elevation, the calculated RVLIS readings begin to oscillate due to the modeling used in WFLASH. In WFLASH, the hot legs are connected to the vessel by point contact connections. This modeling technique causes the hot leg flow to alternate between steam and two-phase flow. The oscillatory behavior of the calculated RVLIS reading continues while the level remains at the hot legs. The average calculated value during this period of time shows that the RVLIS reading is a conservative indication of the mixture level.

When the vessel mixture begins to decrease, the RVLIS reading decreases as well. The RVLIS continues to underpredict the two-phase mixture level and to follow the trend.

Case D

This case is one of the transients investigated for the ICC study using NOTRUMP. A more detailed discussion of this transient can be found in ref. 7.

The RVLIS reading is below the vessel mixture level throughout most of the transient and is therefore a conservative indication. The RVLIS reading follows the same trend as the vessel mixture level except for early in the transient when the mixture void fraction is fluctuating.

Included in the plots for this case is a comparison of the mass inventory in the core and upper plenum regions to the RLVIS reading. This comparison shows that the RVLIS reading also corresponds very well with the relative vessel mass inventory. Also included is a comparison for the UHI and non-UHI RVLIS configurations. For the UHI RVLIS configuration, the pressure difference is measured from the hot leg to the lower plenum rather than the upper head to lower plenum. This plot shows a very good comparison between the two systems, indicating that either will give a useful indication.

III.B.4 Observations of the Study

The RVLIS will provide useful information for breaks in the system ranging from small leaks to breaks in the limiting small-break range. For breaks in this range, the system conditions will change at a slow

enough rate that the operator will be able to use the RVLIS information as a basis for some action.

For larger breaks, the response of the RVLIS will be more erratic, due to rapid pressure changes in the vessel, in the early portion of the blowdown. The RVLIS reading will be useful for monitoring accident recovery, when other corroborative indications of ICC could also be observed.

Very few instances have been identified where the RVLIS may give an ambigious indication. These include: (1) a break in the upper head, (2) accumulator injection into a highly voided downcomer, (3) periods of time when the upper head behaves like a pressurizer, (4) upper plenum injection, (5) and periods of void redistribution.

In order to assess the impact of a break in the upper head, a 2 3/4 in. break has been investigated. This break size corresponds to that expected in the event of a control rod ejection accident. This is the largest break size that is plausible in a non-UHI plant. A UHI line break in a UHI plant would result in a larger break, but since the RVLIS narrow range indication for UHI plants is measured from the hot leg to the bottom of the vessel, the RVLIS indication of vessel level is not significantly affected by the upper head conditions.

Immediately after the break occurs, subcooled liquid flows out the break; this is followed by a brief period of two-phase break flow. During this early period, the flow to the upper head is sufficient to cause the RVLIS to read offscale high on the narrow range (there would still be an indication on the wide range after ~2 min). After 4 to 5 min, however, the upper head and upper plenum have drained sufficiently such that steam is flowing through the break, as well as from the upper plenum to the upper head. The system stabilizes in a quasi-steady state mode with the primary pressure slightly above the secondary pressure and the level in the vessel at the hot leg elevation. The RCS remains at these conditions until the upper portions of the RCS have drained. After approximately an hour, the vessel begins to drain.

During the vessel draining the RVLIS trends with the two-phase mixture level. The RVLIS reads higher than it would if the break were located elsewhere in the RCS due to flow pressure drop through the guide tubes.

The RCS pressure remains near the secondary pressure throughout the transient since the secondary is required for decay heat removal. The pressure drop due to steam flow through the guide tubes at 1100 psi system pressure corresponds to an 11% error on the RVLIS indication.

The RVLIS indication would still provide the operator with useful information concerning the trend in vessel level. The operator would still have sufficient information to diagnose the approach to ICC by using the RVLIS indication along with the core exit thermocouples.

This analysis is applicable to all Westinghouse PWR plants, including those plants with upper plenum injection (UPI). The normal condition for continuous UPI occurs only with the operation of the low head safety injection pumps, which does not occur until a pressure of under 200 psi is realized. The RVLIS may not accurately trend with vessel level during the initial start of UPI. During this short period of time, the cold water being injected will mix with the steam in the upper plenum causing condensation to accumulate. This condensation will form faster than the system response. The system will equilibrate after a short period of time. Upon equilibrating, the system will continue to accurately trend reactor vessel level. For the vast majority of small breaks, the condition of upper plenum injection does not cause a significant impact. For the remainder, the impact is very small and within tolerable limits.

As discussed elsewhere in this section, the time when ambiguous indications due to accumulator injection and upper head pressurizer behavior is brief. The situation corrects itself and the RVLIS resumes giving a good indication of the trend in level. Both situations result in an indication of vessel level that is low. The operator must know that a brief period of erratic RVLIS indication may occur when accumulators are injecting. This effect is partially real in that the vessel level may depress for a moment when accumulator injection occurs. Unlike accumulator injection, the operator will not know when the indicated vessel level is being affected by the upper head pressurizer phenomena. However, no premature indication of ICC will occur since the core exit thermocouples will still read saturation temperature.

Rapid void redistributions within the vessel, which will not be detected by the RVLIS, may occur when the pumps are tripped or restarted when the RCS is highly voided. Transient RVLIS response may occur as the RVLIS indication is in the transition between an indication of level with the pumps tripped and an indication of vessel inventory with the pumps running. This transition period will be brief. No other conditions have been identified where void redistribution can occur in the vessel.

Blockage in the core will tend to increase the frictional pressure drop and the total differential pressure across the vessel, resulting in a higher RVLIS indication. The increase in the RVLIS indication would be most significant under forced flow conditions when the reactor coolant pumps are operating.

In order for blockage to be present, the core would have to have been uncovered for a prolonged period of time. A low RVLIS indication along with a high core exit thermocouple indication would have occurred during this time. If the reactor coolant pumps had been operating throughout the transient, there would have been sufficient cooling to prevent core damage and flow blockage. Therefore, for significant blockage to be present with pumps operating, the pumps would have been shut down initially and then restarted after an ICC condition had existed for a period of time. Based on the history of the transient, the operator would expect that the RVLIS indication would be higher due to blockage, and could possibly use the indication to assess the amount of damage to

the core. Although the RVLIS would read high, it would still follow the trend in vessel inventory and monitor the recovery from the accident.

Under natural circulation conditions, the impact of core blockage is not large. At a natural circulation flow of 4.5%, the RVLIS error due to flow would increase from 0.5% of the vessel height with no blockage to about 5% with 2/3 of the fuel assemblies completely blocked from top to bottom. Under an equilibrium boil-off condition, where flow supplied to the core equals the residual heat boil-off, the RVLIS error due to flow blockage is negligible. These sensitivities to flow blockage are illustrated on Fig. 28. Therefore, even with a large amount of flow blockage, the resulting RVLIS error is minimal, and the RVLIS will trend with the vessel inventory and provide useful information for monitoring the recovery from ICC.

III.B.5 Conclusions

1. With the RCPs tripped, the Westinghouse RVLIS will result in an underpredicted indication of vessel level while providing an unambiguous indication of the mass in the vessel. The Westinghouse RVLIS will also measure the vessel level trend reasonably well.

2. With the RCPs tripped, it is feasible to determine a setpoint for the RVLIS to warn the operator that the system is approaching an uncovered core.

3. The RVLIS should be used along with the core exit thermocouples to detect ICC.

4. With the RCPs running, the RVLIS is an indication of the mass in the vessel.

5. When the RCPs are running, and the RVLIS reading drops to the narrow range scale, there is significant voiding in the vessel and the core would just be covered if the pumps were tripped.

6. A break of sufficient size in the upper head could cause the RVLIS to give an incorrect indication of vessel mass. The reading on the narrow range meter with pumps off will be >100%. The core exit thermocouples, however, will provide an indication of ICC if appropriate.

7. Accumulator injection when the downcomer is highly voided could result in a temporarily erratic indication.

8. The RVLIS may significantly underpredict the vessel mass while the fluid in the upper head is flashing. However, use of the core exit thermocouples will preclude a premature entry to the ICC procedures.

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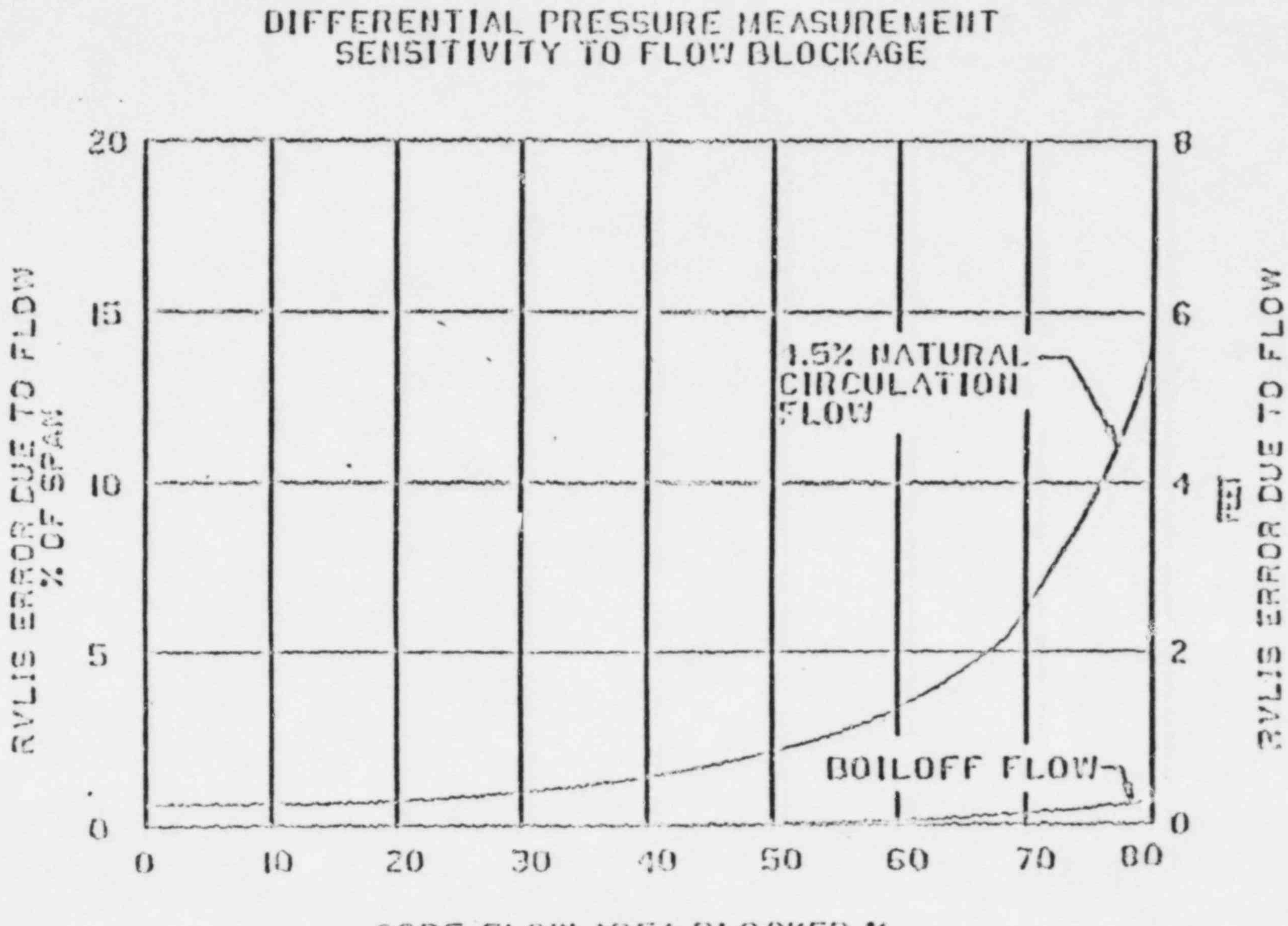

Fig. 28. Differential pressure measurement sensitivity to flow blockage.

9. Rapid void redistributions within the vessel, which will not be detected by the RVLIS, may occur when the pumps are tripped or restarted when the RCS is highly voided. Transient RVLIS response may occur as the RVLIS indication is in the transition between an indication of level with the pumps tripped and an indication of vessel inventory with the pumps running. This transition period will be brief. No other conditions have been indentified where void redistribution can occur in the vessel.

III.C. Qualification

Environmental qualification of the RVLIS shall verify that the system equipment will meet, on a continuing basis, the peformance requirements determined to be necessary for achieving the system requirements as presented above. Verification must include confirmation that those portions of RVLIS equipment which are within the containment will operate during and subsequent to the conditions and events for which the system is required to be operational. Verification will include determination that the system is sufficiently accurate during this time to meet its design basis. The system post-accident environment qualified life requirement for electrical equipment inside containment is 120 d following certain postulated events. The electrical equipment that is installed outside of containment need not meet a qualified life for an extended period of time providing replacement or calibration checks can be made in short enough time commensurate with the reliability goals of the redundant system. Electrical equipment inside containment shall be installed such that it is removed from areas where high energy pipe breaks or pipe whip could cause failure. The d/p transmitters and electronic processing equipment shall be located in a low ambient radiation area.

The RVLIS sensing transmitters and associated electronic processing equipment shall be located in an area whose temperature range is between 40 and 120°F with 0 to 95% ambient relative humidity. Normal operating environment for transmitter locations shall be between 60 and 80°F and 0 to 50% relative humidity. The instrumentation shall be qualified to assure that it continues to operate and read within the required accuracy following but not necessarily during a safe shutdown earthquake. Qualification of the electronic equipment and reactor vessel level sensing transmitters applies to and includes the channel isolation device or where interface with a computer is involved, the input buffer. The location of the electronic isolation device or input buffer should be such that it is accessible for maintenance during accident conditions.

III.C.1 Codes and Standards

The RVLIS is in conformance with the following Codes and Standards:

Regulations

GDC 1 Quality Standards and Records

GDC 2 Design Bases for Protection Against Natural Phenomena

GDC 4 Environmental and Missile Design Bases

IV. EVALUATION OF THE DIFFERENTIAL PRESSURE SYSTEM

The reactor vessel level monitoring systems proposed by Westinghouse (Figs. 2 and 3) uses separate sets of three differential pressure cells in two instrument trains. The differential pressure cells measure the pressure differential between the top and the bottom of the reactor vessel (for UHI plants the measurement is between the bottom of the vessel and the hot leg) and between the level of the hot legs and the top of the vessel. The trains use a spare head penetration for a pressure tap at the top of the vessel. The pressure tap at the bottom of the vessel is made at the seal table to the in-place conduits used for movable in-core detectors. The hot leg pressure taps are connected to the hot leg pipes. Differential pressure cells of differing sensitivity are used to provide wide and narrow range pressure measurements under different flow conditions, i.e. with and without primary pump operation.

One proposed display at the operator's console will consist of three level indicators (analog, vertical scale voltmeters) and a light to indicate the on/off status of the reactor pumps. The indications are automatically corrected for reference leg densities. Displays are provided for each of the two trains. The signals from the three differential pressure cells can be recorded on a strip chart recorder. In addition, analog input signals and the compensated level outputs are available to the plant computer for monitoring.

After the signals are corrected for reactor coolant and capillary line densities, the display panel level indicators are intended to behave as follows:

1. The narrow range indicator displays collapsed water level in the reactor vessel from 0% to 100% for vessel levels between empty and full when the pumps are not operating. When the pumps are operating, the indication is greater than 100% and would be disregarded.

2. The wide range indicator displays vessel level from 0% with the vessel empty to 100% with the vessel full and all reactor coolant pumps operating. With the pumps shut down, the full vessel indication would be about 33%. (For a UHI Plant the indication with pumps off would be about 15%).

3. The upper range indicator indicates level in the upper portion of the reactor vessel or plenum and displays vessel level from 60% to 100% when the water level is between the level of the hot legs and full and the pumps are not operating. When the pumps are operating, the indication is less than 60% (offscale) and would be disregarded.

A second proposed display system is implemented with a microprocessor and accepts inputs from: the differential pressure cells, the temperature sensors connected to the detector lines, the reactor coolant temperature sensor, the system pressure sensors and the reactor coolant pumps status indicator. A set of algorithms is used to generate water level indication from the values of the measured differential pressures.

The display is alphanumeric and provides a summary of all three differential pressure systems, the validity of the indicated levels and other system status information, i.e., which pumps are on, alarms, etc. In addition, the monitor can display trending information in an alphanumeric mode on demand.

On November 18, 1980, Westinghouse described their proposed system at NRC offices in Bethesda. On this occasion they presented results of calculations of the temporal behavior of pressure differential across the reactor vessel and mass inventory inside the vessel for several postulated loss of coolant events. The results showed that, although pressure differential and mass inventory follow trend well, there were brief periods of time during which they trend in opposing directions. In their conclusions, claims of unambiguity in the indications of mass inventory in the vessel are made for the case in which the reactor pumps are off. For the case in which the pumps are on, voiding in the vessel diagnosis will be provided by the system but no predictable indication of the degree of voiding is expected. Inadequate core cooling diagnosis will necessitate confirmatory indications by other systems. Erratic readings are expected under certain conditions, like accumulator injection, break in upper head and upper plenum injection will require special attention from the operator. Procedures are to be supplied by Westinghouse covering these conditions in particular and the RVLIS in general.

IV.A. Application of NRC Criteria from NUREG-0737, II.F.2 Appendix B

1. Environmentally qualified as per NUREG 0588 based on design basis accident events: The proposed system is stated to conform to the following codes and standards: GDC-1,-2-4,-13,-16,-18,-19,-24,-30,-31,-32,-50,-55-56;, 10CFR50; IEEE-308-1971, IEEE-323-1971, IEEE-338-1971, IEEE-344-1971, IEEE-384-1977; ASME BPVC Sect. III; ANSI B31.1.0 1967; RG 1.11, RG 1.22, and RG 1.75.

2. No single failure of instrument or auxiliary system prevents operator from determining the safety status of the plant: redundancy provided by dual instrument trains in the system should meet this requirement. It should be noted, however, that the two instrument trains have points in common at the penetrations to the reactor vessel head and bottom. It should be documented that these common points do not seriously jeopardize the redundancy of the two instrument trains.

3. Class 1E power source: Provided.

4. An instrument channel should be available prior to accident: proposed system has readouts during normal operation.

5. Relevant Reg. Guides list.
   - 1.28 QA for design and construction
   - 1.30 QA for instrumentation and electrical equipment
   - 1.38 QA for handling equipment and components
   - 1.58 Qualification of inspection, test and examination personnel.
   - 1.64 QA design of plant
   - 1.74 Qa terms and definitions
   - 1.88 Record keeping
   - 1.123 QA for procurement
   - 1.144 QA auditing

   Display systems are stated to conform to 10 CFR 50. Appendix B, the above Reg. Guides have not be addressed.

6. Continuous (temporal) indication: provided.

7. Recording of instrument indications: three-pen chart recorders are provided for logging on one train (which may be selected). In addition, analog inputs and outputs of the level system are available to the plant computer for logging and other functions.

8. Indication of instrument in control room: displays are provided for each train.

9. Isolation of signal channels: provided.

10 Verification-on-line: no detailed information has been provided concerning proposed operational procedures. Plant start-up procedures have been outlined.[20] The system, however, has been designed to provide this capability.

11. Service test and calibration programs: as above, no detailed information has been given concering proposed procedures.

12. Control of removal from service: no information has been provided about proposed procedures.

13. Access to adjustment points: Transducers are to be located outside the containment and check and test points will be readily accessible.

14. Minimize anomalous readings: the microprocessor system is designed to indicate when readings are invalid. The analog system status must be inferred from status lights which may be confusing. Known anomalies are believed to be of short duration (1-2 min) and have been represented to pose no serious problems if indications are properly correlated with the indications from other plant sensors.[21] Refer to IV.D.1-11.

15. Recognition and location of components for repair, adjustment or replacement: No information has been supplied about proposed procedures.

16. Direct measurement of desired variables: The water level is not measured directly but is inferred from hydrostatic head. Voiding will result in error in measurement of actual water level, but in a conservative (lower than actual) direction. That is, in a voiding situation, the indicated water level will be less than the actual level. The differential pressure system does not give direct information about cooling capacity of coolant, but does indicate total coolant inventory.

17. Same instrument should be used for accident monitoring and normal operation: a system of three transducers is required to cover all situations. These are integrated into a single measurement system. Under normal operating conditions, the indication on the wide range readout is 100% on a scale that reads from 0% to 110%.[22] Testing procedure is needed. It is necessary to specify the test procedures and criteria to assure operability of the system.

18. Periodic Testing: no information has been supplied.

We believe the requirement of NUREG-0737, II.F.2 (including Appendix B) are satisfied by the Westinghouse System, except that operating procedures are needed (covering, in particular, anomalous readings) and the plant specific items identified in the conclusions.

IV.B Application of Instrumentation Considerations

A. Installation Specific

1. Requirements on operator: the microprocessor system presents level information with an indication of the validity of the indication. The conditions which lead to ambiguous indications have been identified in IV.D.5-9. It is claimed that other indicators would inform the operator when to start using inadequate core cooling procedures. The analog systems have only indicator lights to display the status of the pumps (on or off). All systems include a means for making temperature corrections to the reactor vessel coolant and capillary line densities.

2. Calibration, procedures, in-situ procedures, frequency, standards, etc.: System start-up calibration procedures are detailed, however, normal maintenance and recalibration schedules have not been specified. The acceptability of these procedures has not yet been confirmed.

3. Redundancy or diversity: redundancy is provided by dual instrument strings. The effect of diversity can be implied by other confirming indications of the core-exit thermocouples and the Saturation Margin Monitor.

4. Useful output during normal operation. During normal operation the indications of the three differential pressure cells are

either 100% (on a scale of 0% to 110%) or off-scale. It is claimed that correlation of the wide range cell with flow rate (i.e., number of pumps running) and coolant temperature will tend to maintain operator confidence in the indications of the system.

5. Ease of retrofit or replacement: no modifications to reactor vessel required for most Westinghouse type reactors. Top and bottom pressure taps use existing penetrations. Hot leg taps must be added. With the exception of the ex-core neutron detector system, the differential pressure level monitoring system requires fewer and less complicated changes to the reactor system than other proposed systems. In addition, applicability to reactors of other manufacturers has been demonstrated.

6. Interference with refueling: interference outside the vessel only. Lack of internal components minimizes interference, but potential problems may arise with respect to the disconnects and refilling the lines. The procedure needs to be specified.

B. Sensor and Transducer Specific Considerations.

1. Expected in-service life: no estimates have been given.

2. Radiation resistance: RTDs specified to withstand a total integrated dose of 1.2 mrads gamma. The differential pressure cells which are located outside the containment are specified to withstand 10 krads.

3. Environmental resistance: Devices in the pressure systems are rated at substantially greater pressure than normal operating pressures. The RTDs are to be tested in a saturated steam environment, under a spray of boric acid and sodium hydroxide and additionally put through a biaxial seismic test.

4. Resistance to temperature damage or effect: Most active components of the system are located outside the containment area. No sensitive electronic components are inside the containment. RTDs have qualification tests specified and hydraulic isolators are stated to be environmentally qualified for containment service.

5. Accuracy and resolution: Overall system accuracy is specified to be ±6%. This will result in an uncertainty of ±305 mm (±1 ft) in the upper head or plenum measurement and ±760 mm (±2.5 ft) in the narrow range indication of total vessel level.

6. Response characteristics: time response is given as 10 s or less for 50% indication of the differential pressure instrument following a 50% step change. (See IV.D.17.)

C. Accident and Post-Accident Monitoring Considerations.

1. Effects of core uncovery: No direct effects.

2. Effects of reactor internals movements: Partial blockage of the core will result in a higher than normal friction drop during natural circulation. This is one case in which the reading of the differential pressure system is not in a conservative direction. Additional friction would increase the head and the system would indicate more inventory than was actually present. The degree and significance of this error is identified in Sect. IV.D.3.

3. Effects of pressure changes: A pressure excursion across the differential pressure transducers during a depresurization can cause a decalibration and zero shift in the transducer. Maintaining the upper head transducer and the narrow range transducer in an over-ranged condition during normal operation may cause decalibration of the transducers.

4. Effects of flow variation: The wide range d/p indicater will read 100% output with pumps on, and will be calibrated during installation. Natural circulation flow may cause some errors as indicted above.

5. Ability to measure water quality: Various combinations of pumps will be part of the calibration procedure during installation, refer to Fig. 9. The wide range indicator should trend water quality. Short term variations should be disregarded; procedures are to be developed concerning this issue. When the pumps are off, all systems will tend to indicate collapsed water level.

IV.C Information Quality

An explicit requirement for the ICC systems is that the information provided by the system shall give the operator "an unabiguous indication of an approach to inadequate core cooling." (NUREG-0737). This has been the central issue in the evaluation of the ICC instrument systems, but in particular the reactor vessel level measurement systems, since they are new instruments. In this section, we discuss the concerns expressed by ourselves and others with respect to the quality of Information derived from the ICC instrumentation systems; that is, possible causes of ambiguity, sources of error, and reliability. In the course of this review and in discussions with the applicant conditions were identified where the RVLIS would indicate other than desired output. It was pointed out by Westinghouse that these indications should be confirmed to other instrumentation in all cases. The whole area of operation procedures is to be addressed by Westinghouse and will cover specifically these conditions. The RVLIS system is to be phased into operation over a period of time allowing modification to these procedures before they become standard.

We believe this is a satisfactory solution to the few ambiguous conditions identified below.

A number of questions arose during the course of our review that were submitted to Wetinghouse for clarification. These are reproduced below with further comments when necessary. Many of these have already been discussed in preceding sections of this paper. To avoid redundancy, the discussions below provide numerous references to relevant points made elsewhere. Table 4 provides an index to locate particular catagories.

IV.D Specific Concerns

IV.D.1 Possible Ambiguous Indication with Pumps on/off

First, the NRC staff has issued recommendations to owners of plants from all three reactor vendors that the pumps be tripped when a small break LOCA is detected[23]. Mass flow anywhere in the reactor coolant system would be limited, therefore, to that within the small break regime (0.1 ft sq max) that is predicted to fall to about 800 lb m/s after 100 s. After the depressurization period and upon reaching saturation, the vessel head may well be filled with a tubulent two-phase mixture. The differential pressure level measurement system is expected to indicate reactor vessel inventory under these conditions even with the pumps running.

Second, the design description of the indication of the wide range indicator given in Sect. II.C.2.1.1, describes how the system behaves when the pumps are running. The initial calibration and set up given in operating procedures in Sect. II.C.1.C, describe how the wide range indicator is calibrated to each specific installation. Further amplification is given in the response to question 22:[17]

"Calculations are performed to obtain an estimate of the differential pressure that the wide range instrument will measure with all pumps operating, from ambient temperature to operating temperature. The calculations employ the same methods used to estimate reactor coolant flow for plant design and safety analysis. The calculations are used primarily to define the instrument span and to provide an estimate for the function that compensates the differential pressure signal over the full temperature range, i.e., that results in the wide range display indicting 100% over the full temperature range with all pumps operating, pumping subcooled coolant. During the initial plant startup following installation of the instrumentation, wide range differential pressure data would be obtained and used to confirm or review the compensation function so that a 100% (the scale is 0% to 110%) output is obtained at all temperatures. Since the calculted compensation function is verified by plant operating data, any uncertainties in the flow and differential pressure estimates are eliminated."

Table 4. Index to Discussion of Specific Concerns

1. Dynamic effects.	
a. Possible ambiguous indication with Pumps on/off	IV.D.1
b. Coolant must be quiescent	IV.D.2
c. Effect of core blockage	IV.D.3
d. Effect of reverse flows	IV.D.4
e. Possible ambiguities	
i. Accumulator injection into highly voided downcomer.	IV.D.5
ii. Upper head acts as pressurizer	IV.D.6
iii. Upper plenum injection	IV.D.7
iv. Periods of void redistribution	IV.D.8
v. Bread in the upper head	IV.D.9
g. Effect of voids in vessel	IV.D.10
h. Indication when voiding in core, upper head solid	IV.D.11
i. Voiding in the upper head	IV.D.12
2. Other error sources	
a. On-line test procedures	IV.D.13
b. How will operational availability be determined:	IV.D.14
c. Location and Display of ICC instruments in control room	IV.D.15
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e. System accuracy	IV.D.17
f. System response time	IV.D.18
g. Effect of overranging	IV.D.19
h. Location of temperature sensor on impulse lines	IV.D.20
i. Density correction,	IV.D.21
j. Draining of impulse lines during accident	IV.D.22
k. Effect of dissolved gases in impulse lines	IV.D.23
3. Man-machine interface	
a. Display of sensor status	IV.D.24
b. Does operator have a well defined set of signals to guide emergency response?	IV.D.25
c. Operator may have too much information (are there too many instruments?)	IV.D.26
d. Do emergency procedures enable operator to avoid misunderstanding?	IV.D.27

Table 4. (Cont'd.)

4. Other Concerns	
a. Emphasis on RVLM system may have confused real diagnosis requirements.	IV.D.28
b. System is scenario dependent	IV.D.29
c. System is not relevant to TMI-2 type of accident	IV.D.30
c. Definition of ICC	IV.D.31
d. Maximum break size	IV.D.32
e. Survivability after large break LOCA	IV.D.33
f. How do CET's estimate core uncovery?	IV.D.34
g. Single head penetration	IV.D.35
h. Normal in-service live	IV.D.36

IV.D.2 The Proposed Coolant Level Indicators Could Only Have Value Under Quiescent Conditions

This concern is somewhat more general than the question of pumps on/off discussed above. The behavior of the system under other dynamic effects during the small break transient are discussed in more detail under specific events below.

The single serious case of anomalous behavior found in the tests to date occured in the SEMISCALE tests SUT-3 (ref. 19), and -6 (ref. 24) of the Westinghouse d/p level system appears to have been due to a design feature of the SEMISCALE facility which does not accurately model the actual Westinghouse reactors. In S-UT-8,[25] a retest with more accurately modeled upper internals, the Westinghouse reactor vessel measurement system was in excellent agreement (≈1%) with the SEMISCALE level instrumentation. (See Sect. 11.A.3 for more detailed discussion.)

A more specific concern in this catagory has been expressed that, "the main value (sic. of the reactor vessel level instrumentation) would appear to be for conditions where the system has been depressurized and the coolant state is known, for example prior to refueling."

Scenarios of the small break LOCA begin with depressurization, monitored by the saturation margin monitor, followed by establishment of a water level in the upper regions of the reactor vessel. During the repressurization period the saturation margin monitor provides sufficient information about the condition of the coolant. Tests of both the heated junction thermocouple and differential pressure level measuremment systems indicate that the two systems will be useful and in fact essential (see Sec. 11.B) over a wide range of conditions.

IV.D.3 Effect of Core Blockage

There are indications that the TMI-2 core may be at least 2/3 blocked. Westinghouse was asked to estimate the effect of partial blockage in the core on the differential pressure measurements for a range of values from 0 to 2/3 blockage.

The response of Westinghouse to this concern is given below.

"Blockage in the core will tend to increase the frictional pressure drop and the total differential pressure across the vessel, resulting in a higher RVLIS indication. The increase in the RVLIS indication would be most significant under forced flow conditions when the reactor coolant pumps are operating.

In order for blockage to be present, the core would have to have been uncovered for a prolonged period of time. A low RVLIS indication along with a high core exit thermocouple indication would have occurred during this time. If the reactor coolant pumps had been operating throughout the transient, there would have been sufficient cooling to prevent core damage and thus flow blockage. Therefore, for significant blockage to be present with pumps operating, the pumps would have been shut down initially and then restarted after an ICC condition had existed for a period of time. Based on the history of the transient, the operator would expect that the RVLIS indication would be higher due to blockage, and could possibly use the indication to assess the amount of damage to the core. Although the RVLIS would read high, it would still follow the trend in vessel inventory and monitor the recovery from the accident.

Under natural circulation conditions, the impact of core blockage is not large. At a natural circulation flow of 4.5%, the RVLIS error due to flow would increase from 0.5% of the vessel height with no blockage to about 5% with 2/3 of the fuel assemblies completely blocked from top to bottom. Under an equilibrium boil-off condition, where flow supplied to the core equals the residual heat boil-off, the RVLIS error due to flow blockage is negligible. These sensitivities to flow blockage are illustrated on Fig. 28. Therefore, even with a large amount of flow blockage, the resulting RVLIS error is minimal, and the RVLIS will trend with the vessel inventory and provide useful information for monitoring the recovery from ICC."

IV.D.4. Effect of Reverse Flows

Westinghouse was asked to describe the effects of reverse flow within the reactor vessel on the indicated level.

The response of Westinghouse to this concern is given below.

"Reverse flows in the vessel will tend to decrease the d/p across the vessel which would cause the RVLIS to indicate a lower collapsed level than actually exists. The low indication would not cause the operator to take unnecessary actions, since the RVLIS would be used along

with the core exit thermocouples to indicate the approach to ICC. It is important to note that large reverse flows are not expected to occur for breaks smaller than 6 in. in diameter during the time that the core is uncovered. Large reverse flow rates may occur early in the blowdown transient for large diameter breaks but, as is discussed in Sect. IV.D.32,[17] it is not necessary to use the RVLIS as a basis for operator action for breaks in this range."

Five conditins were identified which could cause the d/p level system to give ambiguous indications. Westinghouse was asked to discuss the nature of the ambiguities for: (1) accumulator injection into a highly voided downcomer, (2) when the upper head behaves as a pressurizer, (3) upper plenum injection, (4) periods of void redistribution, and (5) break in the upper head. The following five sections present their response.

IV.D.5 Accumulator Injection into Highly Voided Downcomer

"When the downcomer is highly voided and the accumulators inject, the cold accumulator water condenses some of the steam in the downcomer which causes a local depressurization. The local depressurization will lower the pressure at the bottom of the vessel which will lower the d/p across the vessel, causing an apparent decrease in level indication. The lower pressure in the downcomer also causes the mixture in the core to flow to the lower plenum, causing an actual decrease in level. The period of time when the RVLIS indication is lower than the actual collapsed liquid level will be brief.

An example of when this phenomenon may occur is when the rector coolant pumps are running for a long period of time in a small break transient. After the RCS loops have drained and the pumps are circulating mostly steam, the level in the downcomer will be depressed. A large volume of steam will be present in the downcomer, above the low mixture level, which allows a large amount of condensation to occur. For most small break transients, the reactor coolant pumps will be tripped early in the transient and the downcomer mixture level will remain high, even in cases where ICC occurs. When the downcomer level is high the effect of accumulator injection on the RVLIS indication will be minor."

IV.D.6 Upper Head Acts as Pressurizer

The response of Westinghouse to this concern is given below. "When the upper head begins to drain, the pressure in the upper head decreases at a slower rate than the pressure in the rest of the RCS. This is due to the upper head region behaving much like the pressurizer. The higher resistance across the upper support plate relative to the rest of the RCS prevents the upper head from draining quickly. This situation only exists until the mixture level in the upper head falls below the top of the guide tubes. At this time, steam is allowed to flow from the

upper plenum to the upper head and the pressure equilibrates. While the upper head is behaving like a pressurizer, the vessel differential pressure is reduced and the RVLIS indicates a lower than actual collapsed liquid level.

This phenomenon is discussed in the summary report on the RVLIS* relative to the 3 in. cold leg break. Since that time, the upper head modeling has been investigated in more detail. It was found that the modeling used at that time assumed a flow resistance that was too high for the guide tubes. Subsequent analyses have shown that the pressurizer effect has less impact on the vessel d/p than was originally shown. There is very little impact on the results after the level drains below the top of the guide tubes. The pressurizer effect is still believed to exist and it becomes more significant as break size increases. The interval of time when the upper head behaves like a pressurizer is brief and the RVLIS will resume trending with the vessel level after the top of the guide tubes uncover. The reduced RVLIS indication will not cause the operator to take any unnecessary action, even if a level below the top of the core is indicated since the core exit thermocouples are used as a corroborative indication of the approach to ICC."

IV.D.7 <u>Upper Plenum Injection</u>

"The normal condition for continuous upper plenum injection (UPI) occurs only with the operation of the low head safety injection pumps, which does not occur until a pressure of under 200 psi is realized. The RVLIS may not accurately trend with vessel level during the initial start of UPI. During this short period of time, the cold water being injected will mix with the steam in the upper plenum causing condensation. This condensation will occur faster than the measurement system response. The system will equilibrate after a short period of time. Upon equilibrating, the system will continue to accurately trend with the vessel level.

In the range of break sizes where RVLIS is most useful in detecting the approach to ICC, the system pressure will equilibrate at a level above the pressure where UPI will normally occur. It is important to note that the flow from the low head pumps is sufficient to recover the core and no operator action based on the RVLIS indication will be necessary.

For the vast majority of small breaks, the condition of upper plenum injection does not cause a significant impact. For the reminder, the impact is very small and within tolerable limits."

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*Westinghouse Electric Corporation, "Westinghouse Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling," December 1980.

IV.D.8 Periods of Void Redistribution

"During the time when the distribution of voids in the vessel is changing rapidly, there can be a large change in the two-phase mixture level with very little change in collapsed mixture level. The use of the RVLIS, in conjunction with the core exit thermocouples, is still vald for this situation, however. The only event that has been identified which could cause a large void redistribution is when the reactor coolant pumps are tripped when the vessel mixture is highly voided. After the pump performance has degraded enough that the flow pressure drop contribution to the vessel differential pressure is small, the change in RVLIS indication will be very small when the pumps are tripped. As discussed in the summary report, the approach to ICC would be indicated when the wide range indication reads 33% (15% in a UHI plant). If the pumps were tripped at this time, the core would still be covered. The operator would know that the core may uncover if the pumps were tripped with a wide range indication lower than 33% (15% in a UHI plant). Prior to pump trip, the core will remain adequately cooled due to forced circulation of the mixture. When the pumps trip the two phase level it may equilibrate at a level below the top of the core. The narrow range indication will provide an indiction of core coolability at this time."

IV.D.9 Break in the Upper Head

"In order to assess the impact of a break in the upper head, a 2 3/4 in. break has been investigated. This break size corresponds to that expected in the event of a control rod ejection accident. This is the largest break size that is plausible in a non-UHI plant. A UHI line break in a UHI plant would result in a larger break, but since the RVLIS narrow range indication for UHI plants is measured from the hot leg to the bottom of the vessel, the RVLIS indication of vessel level is not significantly affected by the upper head conditions.

Immediately after the break occurs, subcooled liquid flows out the break; this is followed by a brief period of two-phase break flow. During this early period, the flow to the upper head is sufficient to cause the RVLIS to read offscale high on the narrow range (there would still be an indication on the wide range after approximately 2 min). After 4 to 5 min, however, the upper head and upper plenum have drained sufficiently such that steam is flowing through the break, as well as from the upper plenum to the upper head. The system stabilizes in a quasi-steady state mode with the primary pressure slightly above the

secondary pressure and the level in the vessel at the hot leg elevation. The RCS remains at these conditions until the upper portions of the RCS have drained. After approximately an hour, the vessel begins to drain.

During the vessel draining, the RVLIS trends with the two-phase mixture level. The RVLIS reads higher than it would if the break were located elsewhere in the RCS due to flow pressure drop through the guide tubes.

The RCS pressure remains near the secondary pressure throughout the transient since the secondary is required for decay heat removal. The pressure drop due to steam flow through the guide tubes at 1100 psi system pressure corresponds to an 11% error on the RVLIS indication.

The RVLIS indication would still provide the operator with useful information concerning the trend in vessel level. The operator would still have sufficient information to diagnose the approach to ICC by using the RVLIS indication along with the core exit thermocouples."

IV.D.10 Effect of Voids in Vessel

See Sect. III.B.

IV.D.11 Indication When Voiding in Core, Upper Head Solid

In some tests at SEMISCALE, voiding was observed in the core while the upper head was still filled with water. Westinghouse was asked to discuss the possibility of cooling the core-exit thermocouples by water draining down out of the upper head during or after core voiding with a solid upper head.

The response of Westinghouse to this concern is given below. See also the discussion of the SEMISCALE tests in III.A.3.

"One of the indicators of an approach to an Inadequate Core Cooling (ICC) situation is the response of the core exit thermocouples (CETs) (T/Cs) to the presence of super-heated steam. The core exit thermocouples will not provide an indication of the amount of core voiding. Response of the core exit T/Cs provide a direct indication of the existence of ICC, the effectiveness of ICC recovery actions, and restoration of adequate core cooling. The core is adequately cooled whenever the vessel mixture level is above the top of the core and the core may have a significant void fraction and still be adequately cooled.

Realistically, an indication of an ICC condition would not occur until the primary coolant system has drained sufficiently for the reactor vessel mixture level to fall below the top of the core. Westinghouse has performed analyses which indicate that the upper head will drain below the top of the guide tubes before ICC conditions exist. The guide tubes are the only flow path from the upper head to the upper plenum. In WCAP-9754,[7] "Inadequate Core Cooling Studies of Scenarios With Feedwater Available, Using the NOTRUMP Computer Code," it was found that inadequate core cooling situations would not result for LOCAs of an equivalent size or equal to approximately 6 in. or less without two or more failures in the ECCS. In both specific scenarios examined in WCAP-9754, a 1-in. and 4 in. small LOCA, the upper head and upper plenum had completely drained before the onset of an ICC condition."

In a typical Westinghouse plant, the core exit T/Cs protude slightly from the bottom of the support columns (Fig. 29). In this location, they measure the temperature of the fluid leaving the core region through the flow passages in the upper core plate. Flow from the upper head must enter the upper plenum via the guide tube before being able to enter the upper core plate flow passages. In addition, the LOCA blowdown depressurization behavior must be such that there is a flow reversal for the core exit T/Cs to detect the upper head fluid temperature. The upper head fluid is expected to mix with the upper plenum fluid as it drains from the upper head.

The potential for core exit T/C cooling from colder upper head fluid, while the core has an appreciable void fraction is not viewed as a potential problem for the detection of an inadequate core cooling situation. Although some SEMISCALE tests indicated core voiding while the upper head was liquid solid, that does not imply that the core exit T/Cs would give an ambiguous indication of ICC calculations for a Westinghouse PWR and consideration of the core exit T/C design would not result in ambiguous ICC indications.

In addition Westinghouse was asked to describe the behavior of the level measurement system when the upper head is full, but the lower vessel is not. Their response is given below.

"During the course of a LOCA transient, the upper plenum will experience voiding before the upper head. The voids in the upper plenum will be indicated by a lower RVLIS reading. The RVLIS will not indicate where the voiding is occurring, but at this point in the transient, it is not necessary to know where the region of voiding is. In the early part of the transient when the mixture level is above the top of the guide tube in the upper head, it is sufficient for the operator to know that the vessel inventory is decreasing, irrespective of the region where voiding is occurring. As discussed in Sect. IV.D.6,[17] the fluid in the upper head does not affect the RVLIS indication after the upper head has drained to below the top of the guide tubes. As discussed above,[15] the upper head will drain before the onset of ICC and there will not be an ambiguous indication during the period of time when RVLIS will be used."

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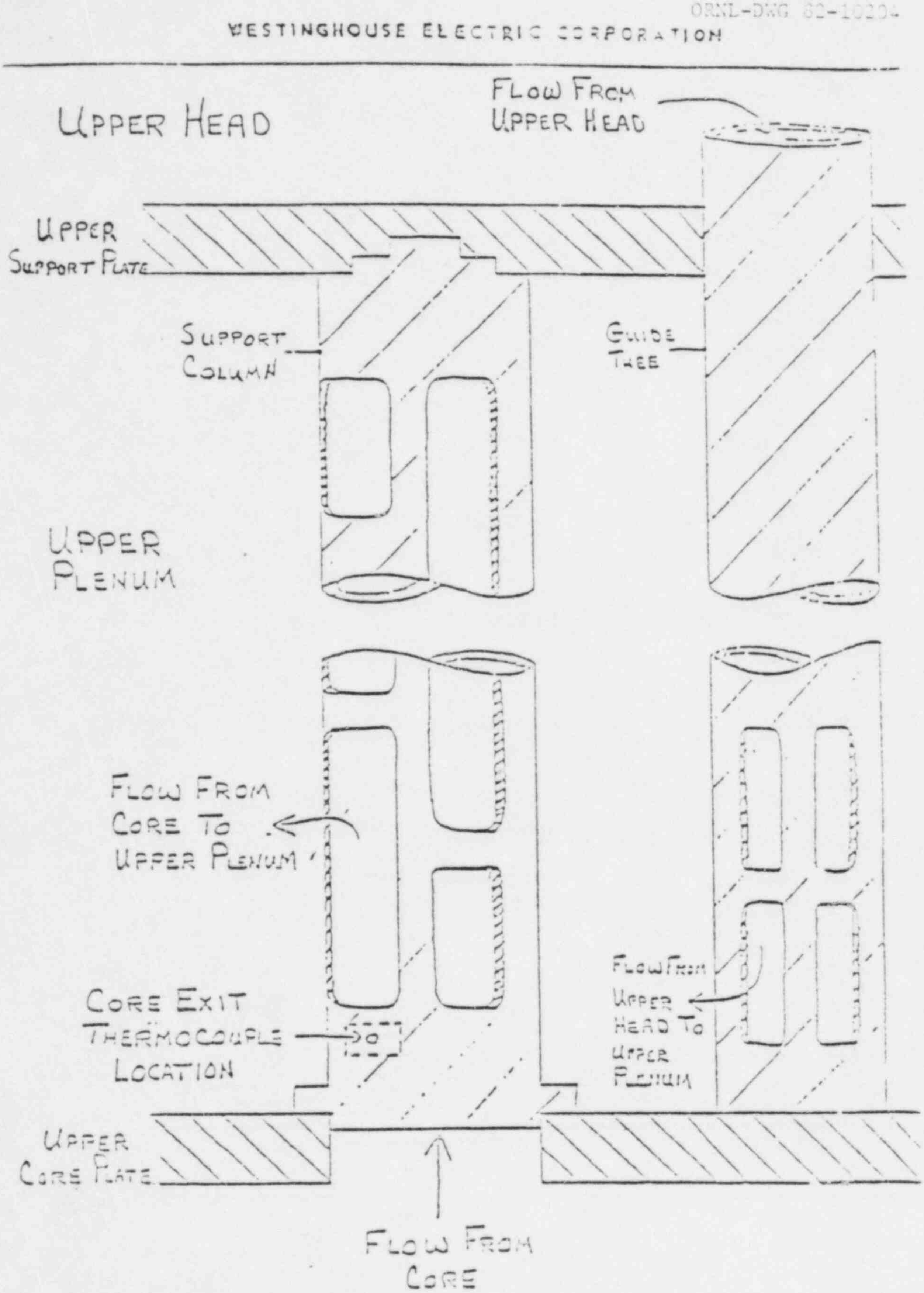

Fig. 29. Sketch indicating location of CETs.

IV.D.12 Voiding in Upper Head

See Sect. IV.D.6 above.

IV.D.13 On-line Test Procedures

Westinghouse was asked to describe the provisions and procedures for on-line verification, calibration and maintenance.

The response of Westinghouse to this concern is given below.

"In general, the system electronics are verified, maintained and calibrated on-line by placing one of the redundant trains into a test and calibrate mode while leaving the other train in operation to monitor inadequate core cooling."

A general verification is performed before shipment, but plant specific data is not used. The capability exists for the operator to verify the operation of the system. This would involve disconnecting the sensors at the RVLIS electronics, providing an artificial input, and observing the response of the system on the front panel and remote display.

On-line calibration of the system is made possible by the "Card Edge" adjustments. The P.C. Cards are calibrated at the factory; however, if the function is changed or a component on the card is replaced, the calibration procedure is given within the equipment reference manual.

The RVLIS system requires the normal maintenance given to other control and protection systems within the plant. On-line maintenance is accomplished by placing only one of the two redundant trains into maintenance at a time; this will allow continued monitoring of inadequate core cooling.

In addition, software programs are provided so that the front panel controls and display can be used to perform a functional test, serial data link tests, calibration tests and deadman timer tests. These tests are considered part of the operator maintenance procedures and should be performed monthly." Although plant specific operating procedures have not yet been written, the generic suggested operating procedures provided by Westinghouse are given in Sect. II.C.1.c. Details regarding on line detection of component malfunction are given in Sect. IV.D.16.

IV.D.14 How Operational Availability will be Determined

With two redundant systems, intercomparisons of the indications displayed by the two systems will provide on-line checking and operability. In addition, see IV.D.13 above.

IV.D.15 Location and Display of ICC Instruments in Control Room

This is plant specific, although indications of all ICC related instrumentation will be located in a single cabinet (or two cabinets, side-by-side).

IV.D.16 Identification of Malfunctioning Components

The response of Westinghouse to this concern is given below.

The cabinet mounted equipment is designed to facilitate periodic tests to identify malfunctioning components and ensure the equipment functional operability is maintained comparable to the original design standards. Component power supply failure is annuniciated in the main control room. Addition details regarding diagnostic techniques and criteria to be used to identify malfunctioning components can be found in Sect. II.C.2.

IV.D.17 System Accuracy

This is discussed in detail in Sect. II.C.3.a.

IV.D.18 System Response Time

Westinghouse was asked to describe how the system response time was estimated. In addition, they were asked to explain how the response times of the various components (differential pressure transducers, connecting lines and isolators) affect the response time.

The response of Westinghouse to this concern is given below. (see also Sect. III.A.2.a.1.

"Testing performed at the SEMISCALE Test Facility in Idaho has confirmed that the response time for the RVLIS hydraulics is less than 3 s. The RVLIS system at SEMISCALE includes sensors, hydraulic isolators, differential pressure transmitters and capillary tubing in lengths typical of a reactor plant installation. An independent analysis by ORNL has established a rise time of less than 3 s, which is consistent with the SEMISCALE test. The RVLIS electronics incorporate an adjustable lag to filter hydraulic noise when reactor coolant pumps are operating. The lag time constant, adjustable up to 10 s, will be set during startup to a value of about 1 to 3 s. The time delays associated with the rest of the electronics are negligible. The total system response time will be well below 10 s."

IV.D.19 Effect of Overranging

Westinghouse was asked to relate their experience, if any, of maintaining d/p cells at 300% overrange for long periods of time.

The response of Westinghouse to this concern is given below.

"Experience in overranging of d/p Instruments has been obtained in previous applications of d/p capsules similar to those used in RVLIS. In Dual Range Flow (d/p) Applications the "Low Flow" transmitter (and/or gauges) are overranged to 300% or greater by normal flow rates yet provide reliable metering when required for startup.

Also, test data exists on the basic transmitter design showing about 0.5% effect on calibration with 24 h exposure to 3000 psig overrange. All units are similarly exposed to this overrange for 5 min in both directions as a part of factory testing.

There have been instances involving accidental overrange of these instruments (including RVLIS) as the result of leakage or operator errors where full line pressure overranges have occurred for up to several weeks with minimal effect on instrument accuracy."

IV.D.20 Location of Temperature Sensor on Impulse Lines

Westinghouse was asked to describe the location of the RTD sensors on each vertical run of the impulse lines. In addition they were asked to describe the expected temperature gradients along each line under normal operating conditions and under a design basis accident. Finally they were asked to describe worst case error that could result from only determining the temperature at a single point on each line.

The response of WestingHouse to this concern is given below.

"RTD sensors are installed on every independently run vertical section of impulse line, to provide a measurement for density compensation of the reference leg. If the vertical section of impulse line runs through two compartments separated by a solid floor, an RTD sensor is installed in each compartment.

The RTD is installed at the midpoint of each vertical section, based on the assumption that the temperature in the compartment is uniform or that the temperature distribution is linear in the vicinity of the impulse line. As stated in the response to question 6, an allowance for the true average impulse line temperature to differ from the RTD measurement by 5°F is included in the measurement uncertainty analysis. This allowance permits a significant deviation from a linear gradient, e.g., 20% of the impulse line could be up to 25°F different from a linear gradient without exceeding the allowance. During normal operation, forced

circulation from cooling fans is expected to maintain compartment temperatures reasonably uniform. During the LOCA, turbulence within a compartment due to release of steam would also produce a reasonably uniform temperature. Note that the impulse lines are protected from direct jet impingement by metal instrument tubing channels."

IV.D.21 Density Correction

Westinghouse was asked to describe the source of the tables or relationships used to calculate density corrections for the level system. The response of Westinghouse to this concern is given below. A discussion is given in Sect. II.C.2.2.2. and is illustrated in Figs. 8 and 9.

"The relationships used in the analog based RVLIS system, to calculate density corrections are from the ASME steam tables, dated 1967. These relationships are implemented within the system by means of P.C. cards that generate an output signal which is a predetermined function of the input signal. The predetermined functions produce specific scopes which are added together to obtain the required input-output relationships."

IV.D.22 Draining of Impulse Lines During Accident

Westinghouse was asked to describe the provisions for preventing the draining of either the upper head or hot leg impulse lines during an accident. They were also asked to describe resultant errors in the level indications should such draining occur.

The response of Westinghouse to this concern is given below.

"The layout of the impulse lines from the upper head and hot leg are arranged to prevent or minimize the impact or drainage during an accident. In general, however, the water in the impulse lines will be cooler than the water in the reactor or hot leg, and there will be sufficient subcooling overpressure in the lines so that very little, if any, of the water would flash to steam during a depressurization or containment heatup. Heat conduction along the small diameter piping and tubing would be insufficient to result in flashing in a significant length of piping.

The connection to the upper head from the vessel vent line drops or slopes down from the highest point of the vessel connection to the sensor bellows mounted on the refueling canal wall, so water would be retained in this piping. Draining of the vertical section immediately above the reactor vessel has no effect on the level measurement, since this section is included in the operating range of the instrument. Draining of the horizontal portion of vessel vent piping above the vessel also has no effect on the measurement since no elevation head is involved.

The connection from the hot leg to the sensor bellows is a horizontal run of tubing, so draining of this tubing has no effect on the measurement since no elevation head is involved.

The majority of the impulse line length is in capillary tubing sealed at both ends with a bellows (sensor bellows at the rector end, hydraulic isolator at the containment penetration end), so water would be retained in this system at all times. The water will be pressurized by reactor pressure, and since the reactor temperature will be higher than containment temperature during an accident, the water in the sealed capillary lines cannot flash."

IV.D.23 Effect of Dissolved Gases in Impulse Lines

Westinghouse was asked to discuss the effect on the level measurements of the release of dissolved, noncondensible gases in the impulse lines in the event of a depressurization.

The response of Westinghouse to this concern is given below.

"The majority of the impulse lines are sealed capillary tubes vacuum filled with demineralized, deaerated water. The lines contain no noncondensible gases and are not in a radiation environment sufficient for the disassociation of water.

The short runs of impulse line connected directly to the primary system will behave as described in Sect. IV.D.22. There would be no error due to gases in the hot leg line since the line is horizontal. Since there is no mechanism for concentration of gases at the top of the reactor vessel during normal operation, this connection to the top of the vessel would contain, at most, the normal quantity of dissolved gases in the coolant, and the subcooling pressure during an accident would maintain this quantity of gas in solution."

IV.D.24 Display of Sensor Status

The microprocessor system is stated to display the status of the sensor input. Describe how is this indicated and what this actually means with respect to the status of the sensor itself and the reliability of the indication. The response of Westinghouse to this concern is given below.

"In the Westinghouse 7300 system, "the status of the hydraulic isolators is indicated by a train status light on the control board. The status of other sensors, such as the wide range RCS pressure and temperature and the strap on RTDs, is not indicated. When this train status light is lit, the operator is instructed to further verify status by checking the dial indicators on the isolators themselves, which are located in an accessible area outside containment."

IV.D.25 Does Operator have a Well Defined Set of Signals to Guide Emergency Response?

The staff has judged the Westinghouse ICC instrumentation system to meet this requirement.

IV.D.26 Operator may have Too Much Information (are there too many instruments?)

See Sect. II.B above.

IV.D.27 Do Emergency Procedures Enable Operator to Avoid Misunderstanding of those Signals Under Circumstances where Accident Diagnosis is Needed in Conjunction with Emergency Actions?

Review of emergency procedures is not within the scope of our evaluation. This is a valid consideration which should be addressed by the staff in conjunction with other procedures work.

IV.D.28 Emphasis on RVLM System may have Confused Real Diagnosis Requirements.

While it may be true that reactor vessel level measurements were emphasized in the early versions of REG GUIDE 1.97 without reguard for situations where no level may actually exist and core cooling is still adequate. A casual reader of the clarification of the requirements for instrumentation for ICC in Sect. II.F.2, p. 2 of NUREG-0737, might possibly intrepret these requirements as placing undue emphasis on level measurement, when, instead, an instrument to measure reactor vessel level is cited as an example of the type of new instrumentation which should receive consideration., The clarifications in Item 1, 3, 4, and 5 of NUREG-0737 emphasize that the ICC instrumentation shall emcompass all necessary instrumentation for indication of the approach, existance and recovery from inadequate core cooling conditions. None of the PWR reactor vendors have taken such a limited interpretation of these requirements as to only propose installation of reactor vessel level instrumentation, but all of the reactor vendors proposed systems include a saturation margin monitor, core-exit thermocouples, and some form of display. Thus, the ICC instrumentation is viewed by the vendors as a "system" that includes water level measurement, but only as a part of the overall ICC system.

In our view the central issue for accident diagnosis requirements is the detection of the approach to ICC and this has been comprehensively covered by the requirements of NUREG-0737 and has been satisfactorily addressed in the Westinghouse ICC instrumentation systems.

IV.D.32 Maximum Break Size

Westinghouse was asked to assume a range of sizes for "small break" LOCA's and describe the relative times available for each size break for the operator to initiate action to recover the plant from the accident and prevent damage to the core. What is the dividing line between a "small break" and a "large break"?

The response of Westinghouse to this concern is given below.

"Inadequate core cooling (ICC) was defined in WCAP-9754,[7] "Inadequate Core Cooling Studies of Scenarios with Feedwater Available using the NOTRUMP Computer Code", as a high temperature condition in the core such that the operator is required to take action to cool the core before significant damage occurs. During the design basis, small loss of coolant accident, the operator is not required to take any action to recover the plant other than to verify the operable status of the safeguards equipment, trip the reactor coolant pump (RCPs) when the primary side pressure has decreased to a specific point, and initiate cold and hot leg recirculation procedures as required. In the design basis, small LOCA, a period of cladding heatup may occur prior to automatic core recovery by the safeguards equipment. The heat up period is dependent upon the break size and ECCS performance.

An ICC condition may arise if there is a failure of the safeguards equipment beyond the design basis. In that case, adequate instrumentation exists in Westinghouse plants to diagnose the onset of ICC and to determine the effectiveness of the mitigation actions taken. The instrumentation which may be used to determine the adequacy of core cooling consists of a subcooling meter, Core Exit Thermocouples (T/Cs), and the Reactor Vessel level Instrumentaion System (RVLIS).

For a LOCA of an equivalent size equal to approximately 6 in. or less, an ICC condition can only occur if two or more failures occur in the ECCS. As indicated in WCAP-9754, an ICC condition can be calculated by hypothesizing the failure of all high head safety injection (HPSI) for LOCAs of approximately 1 in., in size. For a 4 in. equivalent size LOCA, one can hypothesize an ICC condition by assuming the failure of all HPSI as well as the failure of the passive accumulator system (a truly incredible sequence of events).

For LOCAs of sizes of 6 in. or less, the approach to ICC is unambiguous to the reactor operators. The first indication of a possible ICC situation is the indication that some of the ECCS pumps have failed to start or are not delivering flow. The second indication of a possible ICC situation is the occurrence of a saturation condition in the primary coolant system as indicted on the subcooling monitor. Shortly after the second indication, the RVLIS would start to indicate the presence of steam voids in the vessel. At some point in time, the RVLIS will indicate a collapsed liquid level below the top of the core. The core exit thermocouples will begin to indicate superheated steam conditions. If appropriate, the RVLIS and core exit T/C behavior will provide unambiguous indications to the operator to follow the ICC mitigation procedure.

WCAP-9754 indicates that the selected core exit T/Cs will read 1200°F at approximately 11000 s after the initiation of a 1 in. LOCA with the loss of all HPSI. The generic Westinghouse Emergency Operating Procedures (EOPs) Guidelines instruct the operator to pursue ICC mitigation procedures when these conditions are reached after initiation of a 4 in. LOCA with the loss of all HPSI and the accumulator system. 1200°F is indicated to occur at about 1350 s. By following the Westinghouse recommended EOP, the operators will have earlier indication of a possible ICC situation. Recovery procedures to depressurize the primary system below the low pressure safety injection shutoff head may be followed. These procedures include correction of the HPSI failure, opening steam dump, or opening pressurizer PORVs. The RCPs may be restarted to provide additional steam cooling flow. Large break LOCAs consists of LOCAs in which the fluid behavior is inertially dominated. Small break LOCAs, on the other hand, have the fluid behavior dominated by gravitational effects. For LOCAs which are significantly larger than an equivalent 6 in. break, the ECCS has the maximum potential for flow delivery since the primary coolant system is at low pressure.

No early manual action is useful in recovering from ICC. Analyses for LOCAs in this range indicate ambiguous behavior of the core exit T/Cs and RVLIS early in the accident due to dynamic blowdown effects. This behavior is temporary and the core exit T/Cs and the RVLIS will indicate the progress being made by the ECCS in recovering the core. When the core exit T/Cs and RVLIS may be temporarily providing ambiguous indication, no manual action is needed or useful. Later in the accident when manual action may be useful, the core exit T/Cs and RVLIS will provide an unambiguous indication of ICC if it exists. This unambiguous indication may be present as early as 30 s after the initiation of the LOCA for a double ended guillotine rupture or a main coolant pipe.

It follows from the above discussion that, for ICC considerations, a reasonable definition of large breaks are breaks that are significantly larger than an equivalent 6 in. break. All other breaks are small breaks."

IV.D.33 Survivability After Large Break LOCA

Westinghouse was asked to estimate the expected accuracy of the system after an ICC event.

The response of Westinghouse to this concern is given below.

"The accuracy of the system as described in the response to concern IV.D.17 would be the same for any LOCA-type incident, including an ICC event, causing a temperature increase within the reactor containment. Uncertainties due to reference leg temperature measurements and sensor and hydraulic isolator displacements are included in the accuracy analysis."

IV.D.34 How do CET's Estimate Core Uncovery?

This is covered extensively in Westinghouse's small break analyses in WCAP-9600. Vol. II.

IV.D.35 Single Head Penetration

This question is related to the single failure requirement of NUREG-0737, Appendix B, Item 2. Westinghouse was asked to justify that the single upper head penetration meets the single failure requirement of NUREG-0737 and show that it does not negate the redundancy of the two instruments trains.

The response of Westinghouse to this concern is given below.

"Redundancy is not compromised by having a shared tap since it is highly unlikely that the tap will fail either from plugging or breaking. Freedom from plugging is enhanced by, (1) use of stainless steel connections with preclude corrosion products and, (2) absence of mechanism, such as flow for concentrated boric acid. It is also unlikely that the tap will break because it is in a protected area. It should also be pointed out that in other cases where sharing of a tap occurs in the RCS, we know of no prior experience reporting deleterious malfunctions of the shared tap. In the unlikely event the shared tap does fail, it should be recognized that RVLIS is not a Protection System initiating automatic action, but a monitoring system with adequate and redundant backup monitoring such as by core exit thermocouples for operator correlation."

IV.D.36 Normal In-Service Life

Westinghouse was asked to estimate the in-service life under conditions of normal plant operations and describe the methods used to make the estimate, and the data and sources used. Their response is given below. Based on the assumption of normal conditions and proper maintenance of the components, the only limitation to the in-service life will be the availability of replacement parts. It is estimated that in 20 20 y, some of the components will be technically obsolete and no longer produced. Consequently, the cards may have to be modified in the future to accommodate the current technology. Thus, any individual component failures are regarded as maintenance considerations and their replacement is necessary to prolong in-service life.

In-Service life which is different than Design Life and Qualified Life (*) is dependent upon implementing a scheduled preventative maintenance program including periodic overhaul of the equipment. In this

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(*) The system is, of course, qualified to IEEE-323-74, including the requirement for testing to establish a qualified life. Available aging information from Westinghouse is contained in Attachment B.

manner, the equipment is restored to a level that continual operability is ensured. In developing the maintenance program, repair costs may necessitate replacement of the equipment.

If the maintenance program is followed there is no apparent reason that operation of the equipment cannot be extended.

Some of the equipment is similar to equipment installed in present Westinghouse plants that have been operating for 10-15 y.

ATTACHMENT B

COMPONENT AGING INFORMATION

Component	Qualified Life	References	Comments
7300 Process Equipment	5 y	1	A
ΔP Transmitters	5 y	2	
Indicators	5 y	3	A
Recorders	5 y	4	A
High Volume Sensors			B
Strap on RTDs			B

References:

1. Equipment Qualification Data Package EQDP-ESE-13 Rev. 3, 7/81, "Process Protection System"

2. Equipment Qualification Data Package EQDP-ESE-4 Rev. 4, 6/81, "Differential Pressure Transmitters, Qualification Group B."

3. Equipment Qualification Data Package EQDP-ESE-14 Rev. 3, 7/81, "Indicators: Post Accident Monitoring."

4. Equipment Qualification Data Package EQDP-ESE-15, Rev. 3, 7/81. "Recorders: Post Accident Monitoring."

Notes:

A) Westinghouse is planning an extension of Subprogram C of their Aging Evaluation Program (Appendix B to WCAP-8687) to extend this demonstrated qualified life.

B) Qualification testing to establish a qualified life is not yet completed by Westinghouse.

The following valves are typical of those that have been supplied by Westinghouse for the Reactor Vessel Level Instrumentation Systems.

W Valve ID	Qty	Manufacturer	W Design Specification	Code Applicability
3/4 T 78	4	Rockwell	G-952855; Rev 0	ASME B&PV Clas
1/4 X 28I	10	Autoclave Engineers	G-955230; Rev 2	N&S
1/4 N 28I*	6	Autoclave Engineers	G-955230; Rev 2	N&S

*Shut off valve which is part of the transmitter access assembly.

The 3/4 T78 valve is a stainless steel, manually operated globe valve whose basic function is to isolate the flow of fluid. The valve is designed for a cycle life of 4000 cycles over the 40 y design life, which satisfies the normal plant operating requirements established in the above references specification. The valve is a hermetically sealed valve, designed to be maintenance free with no consumable materials making a pressure boundary seal.

The instrumentation valves (W Valve ID's 1/4 x 28I and 1/4 N28I) are stainless steel, manually operated valves, designed to meet the requirements of the above references specification, which calls for zero leakage (environmentally and across the seats), minimal fluid displacement during stoke and a 1000 cycle life. For normal plant operating conditions, the metallic parts are designed for a 40 y service life. The consumable items, where applicable, are identified in the appropriate drawings and instruction manuals, with recommended maintenance schedules.

V. CONCLUSIONS

Analyses have been presented by the Westinghouse Owner's Group in WCAP-9753, of the system behavior with 1 and 4 in. diam breaks. Summary reports describing the generic analog and microprocessor based differential pressure level measurement system together with the Saturation Margin Monitor and core exit thermocouples assert that these systems are adequate for detecting an approach to inadequate core cooling for breaks up to 4 in. diam. Tests of the differential pressure system were added to the regular testing program at SEMISCALE and the results reported in EGG-SEMI-5494 and EGG-SEMI-5552. Additional analysis of these results are forthcoming in ORNL-TMs. Some differences between indications of the Westinghouse system and the SEMISCALE differential pressure level system were noted in the upper head. Westinghouse claims that this difference is mainly a result of differences in the configurations between the full-sized Westinghouse reactor and SEMISCALE upper head regions. Indications of other Westinghouse differential pressure level measurements were in good agreement with the SEMISCALE instruments in the same range. On August 8, 1981, the NRC requested additional information from the utilities proposing to use the Westinghouse differential pressure system. Most of these questions have been resolved to the staff's satisfaction, but a few outstanding questions remain to be answered. The generic description of the system along with the clarification supplied appear to be adequate for approval of the system for trial installation and use. Plant specific features, however, will still require review on a plant by plant basis.

In summary, the systems proposed by Westinghouse do provide an unambiguous indication of water level above the core when, in fact, such a level exists. For some rapid transients the RVLIS has an ambiguous indication, but these conditions are of short duration. For cases where the reactor vessel is filled with a two-phase mixture, experimental evidence indicates that the differential pressure systems will indicate collapsed liquid level or the trending of the reactor vessel coolant inventory. The conclusion of this evaluation is that the Inadequate Core Cooling Instrumentation system which includes the Differential Pressure Reactor Vessel Level Indicating System (RVLIS) proposed by Westinghouse will meet the requirements of NUREG-0737 to provide the plant operator with an unambiguous indication of the approach to adequate core cooling in small break LOCA transients. Furthermore, the system will provide the plant operator a valuable indication of the effect of the recovery measures. Final approval is contingent on resolution of the open items listed below.

V.A.1.a Open items to be resolved. Generic emergency operating procedures have not been provided in the descriptions of the Westinghouse ICC systems. Detailed emergency operating procedures, however, are considered plant specific and must treat ambiguities; these will be reviewed separately for each plant. There are no other significant open issues to be resolved with respect to the generic Westinghouse ICC instrumentation system.

V.A.1.b Plant specific items which must be reviewed individually. The generic differential pressure reactor vessel level measurement system has been evaluated on the basis of documentation supplied by the Westinghouse Co. Any generic description of reactor vessel level measurement systems is necessarily incomplete when applied to a specific plant because of differences in the individual plants. In the course of the evaluation of submittals by the individual licensees, it has become apparent that some utilities have chosen not to install the complete system offered by Westinghouse, particularly with respect to the use of hydraulic isolators and sensors and installation of the differential pressure transducers outside the containment area. Specific plant differences must, therefore, be evaluated on a plant-by-plant basis in cases which are clearly not covered by the generic Westinghouse description or testing program.

Among the plant specific items which will be reviewed on a case-by-case basis are:

1. Location of the display system in the control room.
2. Integration of the ICC displays into console or rack.
3. Location of the differential pressure transducers outside containment.
4. Inclusion of hydraulic isolators and sensors in the impulse lines.

Items listed in NUREG-0737, II.F.2 (see Appendix A, this document) are to be reviewed on a plant specific basis for all plants. Deviations from the generic descriptions in this document must be justified.

1a. Description of display or deviations of instrumentation from generic descriptions in this document.
1b. Description of existing instrumentation systems.
1c. Planned modifications.
2. A design analysis and evaluation of ICC detection instrumentation. Any deviations from generic descriptions in this document, or from instruments tested by the Westinghouse testing program.
3. Additional testing programs, including qualification tests planned.
4. Evaluation of conformance with NUREG-0737: II.F.2, Attachment 1 and Appendix B.
5. Description of computer, software and display functions in plant.
6. Schedule for installation, testing and calibration.
7. Guidelines for use of additional instrumentation, and analyses used to develop procedures.
8. Operator instructions in emergency operating procedures for ICC.
9. Schedule for additional submittals required.

All items required in NUREG-0737, II.F.2, Attachment 1, related to core exit thermocouples are to be reviewed on a plant specific basis.

1. Number and arrangement of core exit thermocouples
2a. Description of display map for all CETs
2b. Description of selected CETs and how selected.
2c. Readout, direct and hardcopy, capability and ranges
2d. Trending capabilities
2e. Alarm capabilities
2f. Operator-display interface
3. Backup displays
4. Type and location of displays

   a. use in normal and abnormal conditions
   b. integration into emergency procedures
   c. integration into operator training
   d. alarms and prioritization of alarms

5. Evaluation with respect to Appendix B
6. Display redundancy, power supply, isolation
7. Environmental qualification
8. Availability
9. Quality Assurance procedures

All items required in NUREG-0737, Appendix B, will be reviewed on a plant specific basis.

1. Environmental qualification
2. Single failure analysis
3. Class 1E power source
4. Availability prior to an accident
5. Quality Assurance procedures
6. Continuous indications
7. Recording of instrument outputs
8. Identification of instruments
9. Isolation
10. In-line Checking
11. Servicing, testing and calibration procedures
12. Administrative control of removal from service
13. Administrative control of access to set point, calibration adjustments
14. Unambiguous indications
15. Identification of malfunctioning components or modules
16. Direct measurement of plant variables
17. Same instruments used for accident monitoring as for normal operation (to the extent practical)
18. Periodic Testing

V.A.2 Limitations

The use of the ICC instrumentation is limited to those transients which progress relatively slowly and for which operator action is required to prevent ICC. There is an uncertainty in the measured level associated with the narrow range differential pressure measurement (the most sensitive) of about 5% or ±2.5 ft. The system may not be used during a rapid depressurization. It is believed that these transients would be of short duration(≈100 s), relative to the response time required for the operator to take action in the regime of small breaks of 0.1 sq ft and smaller. It is not expected to be used for very rapid transients associated with large breaks, since recovery actions should be initiated automatically without operator intervention. The system can be used, however, to monitor the recovery from large breaks. This is discussed in detail in Sect. I.B.

REFERENCES

1. "Report of the President's Commission on The Accident at Three Mile Island," Washington, D.C., 1979, pp. 81-141.

2. Ibid., p. 63 (see also p. 72).

3. "TMI-Related Requirements for New Operating Licenses," NUREG-0694, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1980.

4. Interim Report No. 3 on TMI-2, Advisory Committee on Reactor Safeguards, May 16, 1979.

5. "Clarification of TMI Action Plan Requirements," NUREG-0737, U.S. Nuclear Regulatory Commission, 1980.

6. A letter from D. Eisenhut (NRC) to B&W Owners on the Subject of "Status Summary - Staff Evaluation of Submittals Regarding Additional Instrumentation for Detection of Inadequate Core Cooling for B&W Reactors," August 21, 1980.

7. C. M. Thompson, R. H. Mark, G. E. Hill, and S. Kellman, "Inadequate Core Cooling Studies of Scenarios With Feedwater Available, Using the NOTRUMP Computer Code," WCAP-9743 (proprietary), or WCAP-9754 (non-proprietary), Westinghouse Electric Corporation, Pittsburgh, PA, 19809, p. 1.

8. Included in all C-E plant owner submittals, e.g., "Response to NRC Action Plan NUREG 0660, San Onofre 2 & 3, Inadequate Core Cooling Detection System, Summary Status Report," Southern Califonia Edison Co., Dated April 1981, p. 11.F.2-13.

9. "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, U.S. Nuclear Regulatory Commission, July 1979.

10. "Clarification of TMI Action Plan Requirements," NUREG-0737, Nuclear Regulatory Commission, Washington, D.C., November 1980.

11. Anderson, R. L., Banda, L. A., and Cain, D. G., "In-Core Thermocouple Performance under Simulated Accident Conditions," IEEE Trans. Nucl. Sci., NS-28, 773, (1981).

12. "Interim Report on TMI-2 Accident," ACRS, 7 April 1979.

13. "TMI-2 Lessons Learned Task Force Final Report," NUREG-0585, Nuclear Regulatory Commission, Washington, D.C., October 1979, p. A-12.

14. "The NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, Nuclear Regulatory Commission, Washington, D.C., May 1980, p. I.D-6.

15. "Clarification of TMI Action Plan Requirements," NUREG-0737, Nuclear Regulatory Commission, Washington, D.C., November 1980, p. II.F.2-1.

16. "Reactor Vessel Water Level Signal Compensation for SEMISCALE Test", Westinghouse 1981.

17. From "Generic Responses to NRC Questions on the Westinghouse Inadequate Core Cooling Instrumentation," by Westinghouse Owner's Group Included as a part of Westinghouse plant submittals. October 1981.

18. G. N. Miller, W. L. Zabriskie, and K. G. Turnage, "Frequency Response and Heat Transfer in a dP Measurement System with Long Sensing Lines," Proc. 26th Intern. Instrumentation Symp., Seattle, Washington, May 1980, Instrumentation Society of America, pp. 451-464.

19. J. E. Hardy, G. N. Miller, S. C. Rogers, and W. L. Zabriskie, "Advanced Two-Phase Flow Instrumentation Program, Quarterly ProgressReport for July-September 1981," NUREG/CR-2204, Vol. 3, ORNL/TM-8162, Oak Ridge National Laboratory, Oak Ridge, TN, January 1982.

20. Included in all Westinghouse plant owner submittals, e.g., "Response to NRC Action Plan NUREG 0660, Salem Units 1 & 2, Response to Request for Additional Information on Westinghouse R.V.L.I.S. Summary Report (μ Processor), Public Service Electric and Gas Company, Dated October 1981.

21. V. J. Esposito, K. Kesavan, and B. A. Maul, "WFLASH - A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR," WCAP-8200, Revision 2 (Proprietary), and WCAP-8261 Revision 1 (Non-Proprietary), Westinghouse Electric Corporation, Pittsburgh, PA, 1974.

22. R. Skwarek, W. Johnson, and P. Meyer, "Westinghouse Emergency Core Cooling System Small Break October 1975 Model," WCAP-8970 (Proprietary), or WCAP-8971 (Non-Proprietary), Westinghouse Electric Corporation, Pittsburgh, PA, 1977.

23. "Generic Evaluation of Feedwater Transients and Small Break Loss of Cooland Accidents in Westinghouse-Designed Operating Plants," NUREG-0611, U.S. Nuclear Regulatory Commision, 1980.

24. To be published. (Test report on S-UT-6 and S-UT-7.)

25. To be published. (Test report on S-UT-8.)

APPENDIX A

NUREG-0737 II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Changes to Previous Requirements and Guidance

(1) Specify the "Design and Qualification Criteria" for the final ICC monitoring system in section, "Clarification" (items 7, 8, and 9), Attachment 1, and Appendix B.

(2) Specify complete documentation package to allow NRC evaluation of the final ICC monitoring systems to begin on January 1, 1981.

(3) No preimplementation review is required but postimplementation review of installation and preimplementation review before use as a basis for operator decisions are required.

(4) Installation of additional instrumentation is now required by January 1, 1982.

(5) Clarification item (6) has been expanded to provide licensees/applicants with more flexibility and diversity in meeting the requirements for determining liquid level indication by providing possible examples of alternative methods.

Previous guidance on the design and qualification criteria for upgrading of existing instrumentation was based on Regulatory Guide 1.97, which is still being developed. Detailed design requirements for incore thermocouples and additional instrumentation were not specified. The pertinent portions of draft Regulatory Guide 1.97 have now been included as Appendix B. Design requirements for incore thermocouples used in the ICC monitoring system are specified in Attachment 1. The only significant change in design requirements involves a relaxation of qualification requirements for display systems amenable to computer processing. This facilitates procurement of computer systems and makes feasible the use of

cathode ray tube (CRT) displays that may be needed for proper interpretation of some reactor-water-level systems under development. This relaxation can be accomplished without compromise of ICC monitoring reliability by requiring 99% availability for the display systems, by requiring postaccident maintenance accessibility for nonredundant portions of the system, and by relying on diverse methods of ICC monitoring that include completely qualified display systems.

The staff has concluded that the previous installation requirement of January 1, 1981 for additional instrumentation is unrealistic for most licensees, due to procurement and development problems associated with proposed measurement methods. Further, the staff cannot find the proposed methods acceptable for use until development programs have been completed.

Clarification

(1) Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

(2) The evaluation is to include reactor-water-level indication.

(3) Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

(4) The indication of ICC must be unambiguous in that it should have the following properties:

(a) It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnant boil off); and,

(b) It must not erroneously indicate ICC because of the presence of an unrelated phenomenon.

(5) The indication must give advanced warning of the approach of ICC.

(6) The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated to provide indication of the existence of ICC and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any

inherent deficiencies or inaccuracies in the measurement system selected.

(7) All instrumentation in the final ICC system must be evaluated for conformance to Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

(8) If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix B, need not apply to the channel beyond the isolation device if it is designed to provide 99% availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix B, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

(9) Incore thermocouples located at the core exit or at discrete axial levels of the ICC monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.

(10) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions,

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

Applicability

This requirement applies to all operating reactors and applicants for operating license.

Implementation

This requirement must be implemented by January 1, 1982.

Type of Review

A postimplementation review will be performed for installation, and a preimplementation review will be performed prior to use.

Documentation Required

By January 1, 1981, the licensee shall provide a report detailing the planned instrumentation system for monitoring of ICC. The report should contain the necessary information, either by inclusion or by reference to previous submittals including pertinent generic reports, to satisfy the requirements which follow:

(1) A description of the proposed final system including:

(a) a final design description of additional instrumentation and displays;

(b) a detailed description of existing instrumentation systems (e.g., subcooling meters and incore thermocouples), including parameter ranges and displays, which provide operating information pertinent to ICC considerations; and

(c) a description of any planned modifications to the instrumentation systems described in item 1.b above.

(2) The necessary design analysis, including evaluation of various instruments to monitor water level, and available test data to support the design described in item 1 above.

(3) A description of additional test programs to be conducted for evaluation, qualification, and calibration of additional instrumentation.

(4) An evaluation, including proposed actions, on the conformance of the ICC instrument system to this document, including Attachment 1 and Appendix B. Any deviations should be justified.

(5) A description of the computer functions associated with ICC monitoring and functional specifications for relevant software in the process computer and other pertinent calculators. The reliability of nonredundant computers used in the system should be addressed.

(6) A current schedule, including contingencies, for installation, testing and calibration, and implementation of any proposed new instrumentation or information displays.

(7) Guidelines for use of the additional instrumentation, and analyses used to develop these procedures.

(8) A summary of key operator action instructions in the current emergency procedures for ICC and a description of how these procedures will be modified when the final monitoring system is implemented.

(9) A description and schedule commitment for any additional submittals which are needed to support the acceptability of the proposed final instrumentation system and emergency procedures for ICC.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0578, Recommendation 2.1.3.b

Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, dated October 30, 1979.

II.F.2 Attachment 1. Design and Qualification Criteria For Pressurized-Water Reactor Incore Thermocouples

(1) Thermocouples located at the core exit for each core quadrant, in conjunction with core inlet temperature data, shall be of sufficient number to provide indication of radial distribution of the coolant enthalpy (temperature) rise across representative regions of the core. Power distribution symmetry should be considered when determining the specific number and location of thermocouples to be provided for diagnosis of local core problems.

(2) There should be a primary operator display (or displays) having the capabilities which follow:

(a) A spatially oriented core map available on demand indicating the temperature or temperature difference across the core at each core exit thermocouple location.

(b) A selective reading of core exit temperature, continuous on demand, which is consistent with parameters pertinent to operator actions in connecting with plant-specific inadequate core cooling procedures. For example, the action requirement and the displayed temperature might be either the highest of all operable thermocouples or the average of five highest thermocouples.

(c) Direct readout and hard-copy capability should be available for all thermocouple temperatures. The range should extend from 200°F (or less) to 1800°F (or more).

(d) Trend capability showing the temperature-time history of representative core exit temperature values should be available on demand.

(e) Appropriate alarm capability should be provided consistent with operator procedure requirement.

(f) The operator-display device interface shall be human-factor designed to provide rapid access to requested displays.

(3) A backup display (or displays) should be provided with the capability for selective reading of a minimum of 16 operable thermocouples, 4 from each core quadrant, all within a time interval no greater than 6 min. The range should extend from 200°F (or less) to 2300°F (or more).

(4) The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

(a) the use of this information by an operator during both normal and abnormal plant conditions.

(b) integration into emergency procedures,

(c) integration into operator training, and

(d) other alarms during emergency and need for prioritization of alarms.

(5) The instrumentation must be evaluated for conformance to Appendix B, "Design and Qualification Criteria for Accident Monitoring Instrumentation," as modified by the provisions of items 6 through 9 which follow.

(6) The primary and backup display channels should be electrically independent, energized from independent station Class 1E power sources, and physically separated in accordance with Regulatory Guide 1.75 up to and including any isolation device. The primary display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source, battery backed, where momentary interruption is not tolerable. The backup display and associated hardware should be Class 1E.

(7) The instrumentation should be environmentally qualified as described in Appendix B, item 1, except that seismic qualification is not required for the primary display and associated hardware beyond the isolater/input buffer at a location accessible for maintenance following an accident.

(8) The primary and backup display channels should be design to provide 99% availability for each channel with respect to functional capability to display a minimum of four thermocouples per core quadrant. The availability shall be addressed in technical specifications.

(9) The quality assurance provisions cited in Appendix B, item 5, should be applied except for the primary display and associated hardware beyond the isolation device.

APPENDIX B (of NUREG-0737, II.F.2)

DESIGN AND QUALIFICATION CRITERIA FOR ACCIDENT MONITORING INSTRUMENTATION

Applicability

To the extent feasible and practical (in conformance with the stipulations of Appendix A and ancillary requirements), equipment is to be installed by the specified implementation dates. Where equipment is unavailable, precluding conformance with equipment qualification and schedular requirements, the implementation dates are to be met by installation of best available equipment. In such cases, deviations are to be described and a schedule for the feasible installation of equipment in conformance with the stipulations of Regulatory Guide 1.97 (when the guide is used) is to be provided.

Appendix B is consistent with our current draft version of Regulatory Guide 1.97. We expect no further revisions to our requirements.

Criteria

1. The instrumentation should be environmentally qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. Where the instrumentation channel signal is to be used in a computer-based display, recording and/or diagnostic program, qualification applies to and includes the channel isolation device. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions. The seismic portion of environmental qualification should be in accordance with Regulatory Guide 1.100. The instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake. Instrumentation, whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable, should be qualified using the following guidance.

   The qualification environment shall be based on the design basis accident events, except the assumed maximum of the value of the monitored variable shall be the value equal to the maximum range for the variable. The monitored variable shall be assumed to approach this peak by extrapolating the most severe initial ramp associated with the design basis accident events. The decay for this variable shall be considered proportional to the decay for this variable associated with the design basis accident events. No additional qualification margin needs to be added to the extended range variable. All environmental envelopes except that pertaining to the

5. The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

   1.28 "Quality Assurance Program Requirements" (Design and Construction)

   1.30 "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"

   1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

   1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"

   1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants"

   1.74 "Quality Assurance Terms and Definitions"

   1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"

   1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"

   1.144 "Auditing of Quality Assurance Programs for Nuclear Power Plants"

   Task RS 810-5 "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (Guide number to be inserted.)

   Reference to the above regulatory guides (except Regulatory Guides 1.30 and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5), now in progress.

6. Continuous indication (it may be by recording) display should be provided at all times. Where two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided.

7. Recording of instrumentation readout information should be provided. Where trend or transient information is essential for operator information or action, the recording should be analog stripchart or stored and displayed continuously on demand. Intermittent displays, such as data loggers and scanning recorders, may be used if no significant transient response information is likely to be lost by such devices.

8. The instruments should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

9. The transmission of signals from the instrument or associated sensors for other use should be through isolation devices that are designated as part of monitoring instrumentation and that meet the provisions of the document.

10. Means should be provided for checking, with a high degree of confidence, the operational availability of each monitoring channel, including its input sensor, during reactor operation. This may be accomplished in various ways; for example:

    (a) By perturbing the monitored variable.

    (b) By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable.

    (c) By cross-checking between channels that bear a known relationship to each other and that have readouts available.

11. Servicing, testing, and calibrating programs should be specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing will be less than the normal time interval between generating station shutdowns, a capability for testing during power operation should be provided.

12. Whenever means for removing channels from service are included in the design, the design should facilitate administrative control of the access to such removal means.

13. The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

14. The monitoring instrumentation design should minimize the development of conditions that would cause meters annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator.

15. The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

16. To the extent practical, monitoring instrumentation inputs should be from sensors that directly measure the desired variables.

17. To the extent practical, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during accident situations,

instruments with which the operator is most familiar. However, where the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.

18. Periodic testing should be in accordance with the applicable portions of Regulatory Guide 1.118 pertaining to testing of instruments channels.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 or be in HOT SHUTDOWN within the next 12.

b. With the number of OPERABLE accident monitoring cahnnels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channel(s) to OPERABLE status within 48 h or be in at least HOT SHUTDOWN within the next 12.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY
LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

Table 3.3-10. Accident Monitoring Instrumentation

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure - Narrow Range	2	1
2. Containment Pressure - Wide Range	2	1
3. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1
5. Pressurizer Pressure - Wide Range	2	1
6. Pressurizer Water Level	2	1
7. Steam Line Pressure	2/steam generator	1/steam generator
8. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
9. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator
10. Refueling Water Storage Tank Water Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
12. Reactor Coolant System Subcooling Margin Monitor	2	1
13. Safety Valve Position Indicator	2/valve	1/valve
14. Spray System Temperature	2	1
15. Spray System Pressure	2	1
16. LPSI Header Temperature	2	1
17. Containment Temperature	2	1
18. Containment Water Level - Narrow Range	2	1
19. Containment Water Level - Wide Range	[illegible]	[illegible]
20. Core Exit Thermocouples	[illegible]/core quadrant	[illegible]/core quadrant
21. [illegible] - High Range	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

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Table 4.3-7 Accident Monitoring Instrumentation Surveillance Requirements

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
5. Pressurizer Pressure (Wide Range)	M	R
6. Pressurizer Water Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Water Level (Narrow Range)	M	R
9. Steam Generator Water Level (Wide Range)	M	R
10. Refueling Water Storage Tank Water Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. Safety Valve Position Indicator	M	R
14. Spray System Temperature	M	R
15. Spray System Pressure	M	R
16. LPSI Header Temperature	M	R
17. Containment Temperature	M	R
18. Containment Water Level (Narrow Range)	M	R
19. Containment Water Level (Wide Range)	M	R
20. Core Exit Thermocouples	M	R
*21. Containment Area Radiation - High Range	(a)	(a)
*22. Main Steam Line Area Radiation	(a)	(a)
23. Condenser Air Ejector Noble Gas Monitor - Wide Range	M	R
24. Purge/Vent Stack Noble Gas Monitor - Wide Range	M	R
25. Cold Leg HPSI Flow	M	R
26. Hot Leg HPSI Flow	M	R
27. Reactor Vessel Coolant Level Monitor	M	R

(a) In accordance with Table 4.3-3.
*NUREG 0737 item to be operational by January 1, 1982.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/or associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

1. at least HOT STANDBY within the next 6 h,

2. at least HOT SHUTDOWN within the following 6 h, and

3. at least COLD SHUTDOWN within the subsequent 24 h.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

NUREG/CR-2628
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AUTHOR(S)
G. N. Miller, J. L. Anderson, R. L. Anderson

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ABSTRACT (200 words or less)

This document, entitled "Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement," presents a technical review of the Inadequate Core Cooling Instrumentation with a Reactor Vessel Level Monitoring System using a differential pressure measurement system proposed by Westinghouse, Inc., for pressurized water reactors.

Emphasis was placed on evaluation of the generic Inadequate Core Cooling (ICC) Instrumentation System as a whole which includes, besides the differential pressure reactor vessel level measurement, the saturation margin monitor, the core exit thermocouples and the display system (either the 7300, an analog display, or the microprocessor based system with a plasma panel display).

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Attachment 2

December 10, 1982

TO: ALL LICENSEES OF OPERATING WESTINGHOUSE AND CE PWRs (EXCEPT ARKANSAS NUCLEAR ONE - UNIT 2 AND SAN ONOFRE UNITS 2 AND 3)

SUBJECT: INADEQUATE CORE COOLING INSTRUMENTATION SYSTEM (GENERIC LETTER NO. 82-28)

Gentlemen:

On November 4, 1982, the Commission determined that an instrumentation system for detection of inadequate core cooling (ICC) consisting of upgraded subcooling margin monitors, core-exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of pressurized water reactor facilities.

On the basis of analysis of information provided by licensees, meetings with industry groups and independent studies by the NRC Staff, the Commission has found that during a small LOCA, there is a period of time before the core has boiled dry (indicated by core exit thermocouples) when the operators have insufficient information to clearly indicate a void formation in the reactor vessel head or to track the inventory of coolant in the vessel and primary system. The Subcooling Margin Monitor gives early indication of a problem but does not indicate whether the condition is getting worse or better.

The addition of a reactor coolant inventory system will improve the reliability of plant operators in diagnosing the approach of ICC and in assessing the adequacy of responses taken to restore core cooling. The benefit will be preventive in nature in that the instrumentation will assist the operator in avoidance of ICC when voids in the reactor coolant system and saturation conditions result from over cooling events, steam generator tube ruptures, and small break loss of coolant events. The addition of a reactor coolant inventory system, coupled with upgraded in-core thermocouple instruments and a subcooling margin monitor, provides an ICC instrumentation package which could significantly reduce the likelihood of human misdiagnosis and errors for events such as steam generator tube ruptures, loss of instrument bus or control system upsets, pump seal failures, or overcooling events originating from disturbances in the secondary coolant side of the plant. For less frequent events, involving coincidental multiple faults or more rapidly developing small break LOCA conditions, the ICC could also reduce the probability of human misdiagnosis and subsequent errors leading to ICC.

~~8212140103~~

The Nuclear Regulatory Commission has completed its review of several generic reactor level or inventory system instrumentation systems which have been proposed for the detection of ICC in PWRs. The Combustion Engineering Heated Junction Thermocouple (HJTC) system and the Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) are acceptable for tracking reactor coolant system inventory and provide an enhanced ICC instrument package when used in conjunction with core exit thermocouple systems and subcooling margin monitors designed in accordance with NUREG-0737 and operated within approved Emergency Operating Procedure Guidelines. The details of the NRC Staff review of these generic systems are reported in NUREG/CR-2627 and NUREG/CR-2628 for the Combustion Engineering and Westinghouse systems, respectively.

Other differential pressure (d/p) measurement techniques for reactor coolant system inventory tracking are acceptable provided that they meet NUREG-0737 design requirements and monitor the coolant inventory over the range from the vessel upper head to the bottom of the hot leg as a minimum.

In order for the Commission to complete its review of your ICC system to assure that an acceptable system is installed as soon as practicable, the NRC requires additional information.

Accordingly, in order to determine whether your license should be modified, you are required to submit to the Director, Division of Licensing, NRR, the following information in writing and under oath or affirmation pursuant to Section 182 of the Atomic Energy Act and 10 CFR 50.54(f) of the Commission's regulations.

1. Within 90 days of the date of this letter, identify to the Director, Division of Licensing, the design for the reactor coolant inventory system selected and submit to the Director, Division of Licensing, detailed schedules for its engineering, procurement and installation. References to generic design descriptions and to prior submittals containing the required information, where applicable, are acceptable.

2. Within 90 days of the date of this letter review the status of conformance of all components of the ICC instrumentation system, including subcooling margin monitors, core-exit thermocouples, and the reactor coolant inventory tracking system, with NUREG-0737, Item II.F.2 and submit a report on the status of such conformance.

3. The installation of the ICC instrumentation system shall be completed during the earliest refueling shutdown consistent with the existing status of the plant and practical design and procurement considerations. It has become apparent, through discussions with owners' groups and individual licensees, that schedules must adequately consider the integration of these requirements with other TMI related activities. In recognition of this and the difficulty in implementing generic deadlines, the Commission has adopted a plan to establish realistic plant-specific schedules that take into account the unique aspects of the work at each plant. Each licensee is to develop and submit its own plant-specific schedule which will be reviewed by the assigned NRC Project Manager. The NRC Project Manager and licensee will reach an agreement on the final schedule and in this manner provide for prompt implementation of these important improvements while optimizing the use of utility and NRC resources.

Licensees who have completed installation of an approved generic ICC instrumentation system are authorized to make their system operable prior to final NRC approval for purposes of operator training and familiarization. However, the ICC instrumentation system should not be turned on until the licensee has completed the task analysis portion of the control room design review, and should be used with prudence in relation to any operator actions or decisions until the plant specific design and installation has been approved by the staff and instructions on its use and operation have been incorporated in accordance with the Emergency Operating Procedure Guidelines into approved emergency operating procedures.

For your convenience in performing the status review (Item 2) of your conformance with NUREG-0737, a check list of the nine items of documentation cited on pp. II.F.2 - 3 and 4 of that document is provided in an appendix to this letter. Even though you may have provided much of the information required for our review, we have not yet received all of the information required to complete our review of plant specific installations for any licensee. In addition, some licensees have modified their positions during the period when NRC was re-reviewing the II.F.2 requirements. Therefore, your status report should review for completeness and reference those earlier submittals, including generic submittals, which remain valid in response to documentation items on the check list. In addition, you should include a proposed schedule for the remaining submittals. Information items to be addressed in the submittal regarding your review of core exit thermocouples for conformance to NUREG-0737, II.F.2, Attachment 1, and your review of the ICC instrumentation for conformance to NUREG-0737, Appendix B, are also listed in the appendix to this letter. The staff review will focus on deviations from the design criteria.

This request for information was approved by OMB under clearance number 3150-0065 which expires May 31, 1983. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports and Management, Room 3208, New Executive Office Building, Washington, D.C.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
As stated

cc w/enclosure
Service Lists
Westinghouse Electric Corp.
Combustion Engineering

APPENDIX

Checklist
for Plant Specific Review of
Inadequate Core Cooling (ICC) Instrumentation System

For ____ Docket No. ____________

Operated by: ________________________________

____

____

The following items for review are taken from NUREG-0737, pp II.F.2-3, and 4. Responses should be made to full requirements in NUREG-0737, not abbreviated forms below. Applicants should provide reference to either the applicant's submittal or the generic description under the column labled "Reference." These items are required to be reviewed on a plant specific basis by NUREG-0737 for all plants. Differences from the generic descriptions provided by Westinghouse, the Westinghouse Owner's Group, Combustion Engineering, or Combustion Engineering Owner's Group must be indicated by "yes or no" in the column labled deviations and must be justified. Under the Column labeled schedule, either indicate that your documentation of the item is complete or provide a proposed schedule for your submittal.

	Reference	Deviations	Schedule
1. Description of the proposed final system including:			
a. a final design description of additional instrumentation and displays;			
b. detailed description of existing instrumentation systems.			
c. description of completed or planned modifications.			
2. A design analysis and evaluation of inventory trend instrumentation, and test data to support design in item 1.			
3. Description of tests planned and results of tests completed for evaluation, qualification, and calibration of additional instrumentation.			

4. Provide a table or description covering the evaluation of conformance with NUREG-0737: II.F.2, Attachment 1, and Appendix B (to be reviewed on a plant specific basis)* ____________________
5. Describe computer, software and display functions associated with ICC monitoring in the plant. ____________________
6. Provide a proposed schedule for installation, testing and calibration and implementation of any proposed new instrumentation or information displays. ____________________
7. Describe guidelines for use of reactor coolant inventory tracking system, and analyses used to develop procedures. ____________________
8. Operator instructions in emergency operating procedures for ICC and how these procedures will be modified when final monitoring system is implemented. ____________________
9. Provide a schedule for additional submittals required** ____________________

*II.F.2 Attachment 1 (for Core Exit Thermocouples)

In response to item 4 in the above checklist, the following materials should be included to show that the proposed system meets the design and qualification criteria for the core exit thermocouple system.

1. Provide diagram of core exit thermocouple locations or reference the generic description if appropriate.

2. Provide a description of the primary operator displays including:

   a. A diagram of the display panel layout for the core map and description of how it is implemented, e.g., hardware or CRT display.
   b. Provide the range of the readouts.
   c. Describe the alarm system.
   d. Describe how the ICC instrumentation readouts are arranged with respect to each other.

3. Describe the implementation of the backup display(s) (including the subcooling margin monitors), how the thermocouples are selected, how they are checked for operability, and the range of the display.

4. Describe the use of the primary and backup displays. What training will the operators have in using the core exit thermocouple instrumentation? How will the operator know when to use the core exit thermocouples and when not to use them? Reference appropriate emergency operating guidelines where applicable.

5. Confirm completion of control room design task analysis applicable to ICC instrumentation. Confirm that the core exit thermocouples meet the criteria of NUREG-0737, Attachment 1 and Appendix B, or identify and justify deviations.

6. Describe what parts of the systems are powered from the 1E power sources used, and how isolation from non-1E equipment is provided. Describe the power supply for the primary display. Clearly delineate in two categories which hardware is included up to the isolation device and which is not.

7. Confirm the environmental qualification of the core exit thermocouple instrumentation up to the isolation device.

Appendix B (of NUREG-0737, II.F.2)

Confirm explicitly the conformance to the Appendix B items listed below for the ICC instrumentation, i.e., the SMM, the reactor coolant inventory tracking system, the core exit thermocouples and the display systems.

	Reference	Deviations
1. Environmental qualification		
2. Single failure analysis		
3. Class 1E power source		
4. Availability prior to an accident		
5. Quality Assurance		
6. Continuous indications		
7. Recording of instrument outputs		
8. Identification of instruments		
9. Isolation		

**For the users of either Combustion Engineering Heated Junction Thermocouple (HJTC) System or Westinghouse Differential Pressure (dp) system a detailed response to the plant specific items stated below should be provided.

	Reference	Deviations
A. Westinghouse dp System		
1. Describe the effect of instrument uncertainties on the measurement of level.		
2. Are the differential pressure transducers located outside containment?		
3. Are hydraulic isolators and sensors included in the impulse lines?		
B. CE HJTC System		
1. Discuss the spacing of the sensors from the core alignment plate to the top of the reactor vessel head. How would the decrease in resolution due to the loss of a single sensor affect the ability of the system to detect an approach to ICC?		