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PRELIMINARY TWO-PHASE INSTRUMENTATION EVALUATION

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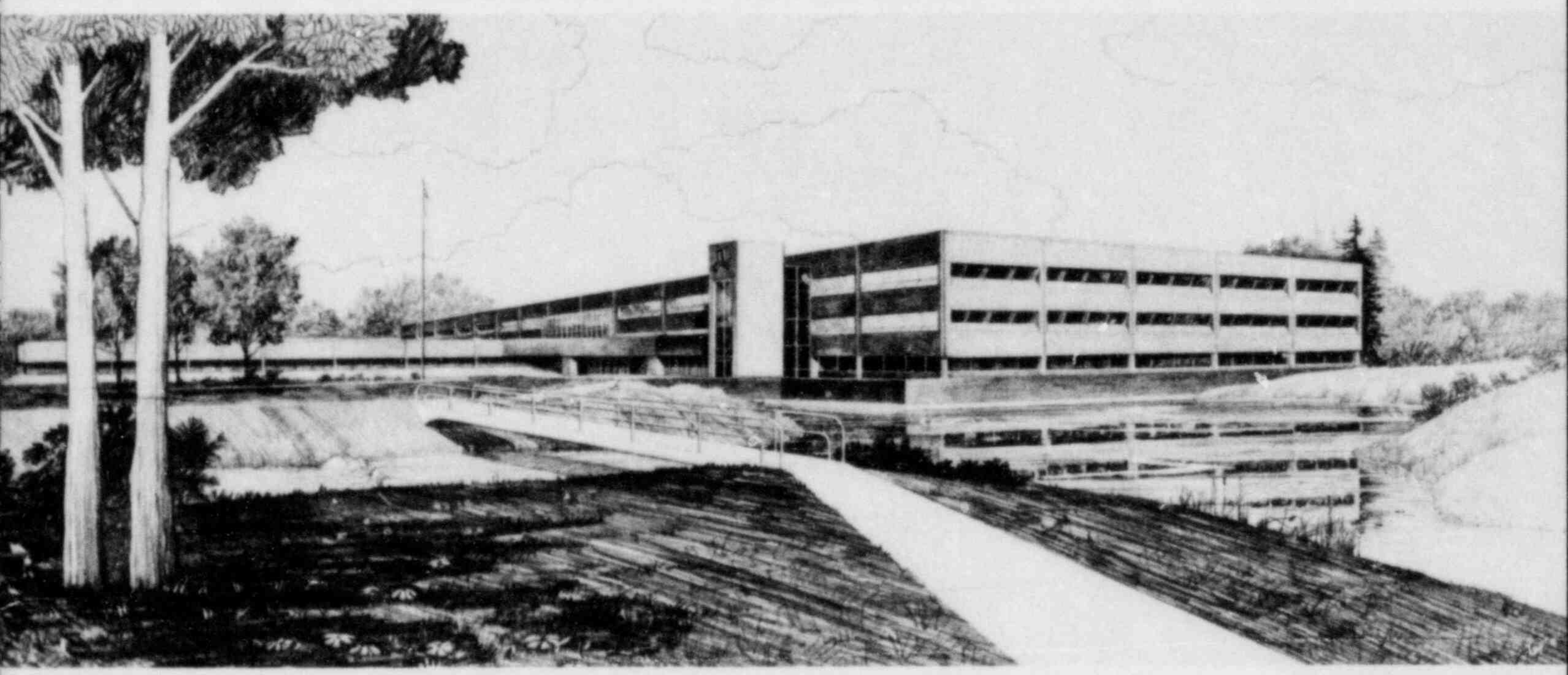

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INTERIM REPORT

PRELIMINARY TWO-PHASE INSTRUMENTATION EVALUATION

By

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CONTENTS

1. INTRODUCTION

The objective of this research is to evaluate and test instruments used in the detection and measurement of parameters which characterize two phase (water-steam) presence during normal and accident conditions.

The initial phase of this program is to assess the current capabilities of U.S. Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs) in detection and measurement of two-phase system parameters. In particular the goal was to assess the current practice and capability of measuring low flow rates in the primary system corresponding to velocities of <0.1 m/s. The detection/measurement of the onset of boiling or voiding in the reactor vessel, steam generator u-tubes and the primary pumps were examined in particular.

2. DESCRIPTION OF METHODS USED TO OBTAIN INFORMATION

The method used in attempting to obtain information for the assessment was to select several operating utilities with plants supplied by the four primary reactor vendors. These comprised utilities operating a total of nineteen plants from the major vendors. The selection included three plants from Combustion Engineering, six plants from the General Electric Company, three Babcock & Wilcox plants and seven Westinghouse plants. The four reactor vendors were contacted along with the utilities by letter requesting any information regarding instruments and methods pertinent to this research. Response to these requests has been slow and sporadic, comprising one phone response from a vendor, three phone responses from utilities promising future letter response and one letter response from a utility answering the questionnaire.

In connection with the letter inquiry to vendors and utilities an extensive literature search was begun utilizing the INEL technical library, EPRI, other industry reports and publications from National Laboratories doing research in related areas of interest. Published proceedings of professional societies and personal contacts in industry were also utilized in the information gathering process. One foreign utility was also contacted with safety related research in progress. This research was related to pump vibration to indicate the onset of primary pump cavitation.

Interviews were conducted with operators and instrumentation groups involved with the planning and operation of research reactors and testing facilities located at the Idaho Nationl Engineering Laboratory, this includes LOFT, PBF, ATR and Semiscale. Current and proposed measurement methods and instruments were discussed and noted.

Commercially available instruments which could possibly have potential use in this study either off the shelf or with modifications to facilitate use in a reactor environment were also considered. The Instrument Summary based on information received to date is shown in Table 1.

TWO-PHASE INSTRUMENTATION SURVEY

TABLE 1

PRIMARY COOLANT FLOW

Sensor(s) Type	Range	Comments
Venturi, Measured by DP cell or Transmitter	Typically displayed as 0 to 120% of range	High range device, Accuracy suffers below 30-40% of design range full flow. Not suitable for two-phase measurement.
Orifice, Measured by DP cell or transmitter.	Typically displayed or recorded as 0-100+% of range	High range device, Accuracy suffers below 30-40% of design range full flow. Not suitable for two-phase measurement.
Differential pressure measured across steam generator or other part of the RCS.	High. Plant and steam generator specific.	High range method. Accuracy poor for low flow rates. Measurement possibly affected by flow blockage or the condition of the steam generator. Not suitable for two-phase measurement.
Elbow or Corner taps	Pipe size and flow rate specific.	High range method. Accuracy poor for low flow rates. Measurement stability often questioned[1]. Not suitable for two-phase.
Gentille tube as measured by a DP cell or transmitter.	High	Head or differential pressure producing device similar to a venturi. Not suitable for low flow rate measurement or two-phase flow measurement.

Two-Phase Detection

Not directly detected or measured	Not Applicable	Operating plants monitor existing instruments including "Saturation Meter", RCS flow rate, T_{hot}, T_{cold}, Core exit temperature, RCS pressure and administratively set operating limits or alarms.

TWO-PHASE INSTRUMENTATION SURVEY

TABLE 1(continued)

Reactor Vessel Voiding

Sensor(s) Type	Range	Comments
Not directly measured	System dependent	Operating plants are now installing or planning to install vessel liquid level systems approved per RG 1.97. Plants presently monitor other system parameters to indicate presence of voids in the system. Research is underway on other methods of making the measurement. See Discussion Section.

Steam Generator U-Tube Voiding

Sensor(s) Type	Range	Comments
No current on line measurement system indicated from Utility responses.	N/A	No current development projects found. See Proposal section.

Break Identification

Sensor(s) Type	Range	Comments
No current on line measurement system indicated from Utility responses	N/A	See Proposal Section

4. DISCUSSION OF PRIMARY COOLANT FLOW, METHODS AND INSTRUMENTATION

All of the surveyed plants use similar instrumentation to measure primary coolant flow and are head producing devices such as orifices, venturi or differential pressure measured across some portion of the primary system. In all differential pressure devices flow rate varies as the square root of the differential pressure produced by flow passing through the device. Operating plants wish to keep unrecoverable head losses to a minimum; therefore, these devices are sized large, typically 100 to 120% of designed flow through the piping system containing the device. During reduced flow <100% normal, flow derived differential pressure is reduced substantially. Several methods are employed, e.g., different range transmitters or differential pressure cells are switched in to the measurement system, in an attempt to increase accuracy and stability. Drift and stability has sometimes been noted as a problem[1] as differential pressure (flow) decreases.

Measurement of differential pressure across a part of the primary system such as the steam generator also presents a potentially misleading measurement. Differential pressure could change due to a change in the heat exchanger internal condition or a flow blockage at the heat exchanger input.

None of the presently utilized instrumentation found in the assessment provides a viable method of measurement of low velocity flow rates.

Recent developmental work performed on low flow devices include the cooled low flow velocimeter[2, 3, 4] and impedance flow sensor[5] testing. The cooled thermal velocimeter shows excellent response to low velocity flows at variable temperature and pressure and has shown some usable data in two phase flows with void fractions as high as 0.60 in dispersed flows. The two phase testing was performed in an air water loop.

The impedance flow sensor was designed to measure void fraction and two phase velocity. Velocity calculations were valid for only certain flow regimes as listed in a 9-rod heated bundle reflood test facility. Studies have also been made[6] of the relationship of core exit temperature noise to thermal-hydraulic conditions in PWRs. This study utilizes the output of core exit thermocouples and in pile neutron detectors in determining abnormal thermal hydraulic conditions in an operating PWR. Pulsed neutron activation has also been used experimentally[7, 8] to measure both single and two-phase flows. Reference 7 reports the success of using PNA in a high nuclear radiation background. PNA, a nonintrusive nuclear tagging technique for measuring the flow velocity of a dense fluid, has three major advantages. They are:

1. It is a nonintrusive measurement technique requiring no pipe penetration.

2. The measurement process does not perturb the flow.

3. It is an absolute measurement in the sense that it does not require calibration for an acceptable flow measurement.

However, since PNA employs nuclear counting techniques, shielding of both detector and neutron generator becomes critical. Neutron generators have a limited life expectancy and installation and shielding has proved to be expensive and vital to a reliable measurement.

4.1 Two-Phase Detection

None of the plants surveyed had an on-line method of detection or measurement of two-phase conditions present in the system with the exception of reactors used in Research and Development. These reactors typically have one or more gamma densitometers or other experimental/developmental instrumentation in locations of interest to operators or experimenters.

Responding utilities administratively require operators to monitor designated plant instruments to give indications of system status. A saturation meter or indication of saturation exists in responding power plants. Saturation is either indicated by a commercial "saturation meter" or is calculted by the plant computer and displayed on an instrument console and/or CRT display, usually with alarms. Normal inputs to the saturation meter or to the plant computer used for calculating saturation condition include but are not limited to hot leg temperature (T_H), cold leg temperature (T_C), core exit temperature, and reactor coolant system pressure in several places. The Babcock & Wilcox Company offers a saturation meter with capabilities to accept temperature and pressure inputs from (16) sixteen channels and displays margin to saturation. Other options include switch selectable monitoring of either the T_H or T_C loop.

4.2 Reactor Vessel Voiding

Utilities responding to the questionnaire indicated that there are presently no direct methods of indicating or measurement of vessel voiding in use. Indirect indicators exist within the current plant instruments employed in nuclear power plant systems.

Most responding plant have either installed or are in the planning stages of installing an NRC approved level system in the reactor vessel, e.g., the Westinghouse Reactor Vessel Level Indicating System,[9] RVLIS, Combustion Engineering, Inc., Reactor Vessel Level Monitoring System,[10] RVLMS.

The Westinghouse RVLIS has been installed and tested at INEL's Semiscale Test Facility. Several tests have been performed[11, 12, 13] with the RVLIS installed in the facility. Independent evaluations of the data were performed at ORNL.[14, 15, 16] The results of this evaluation indicate that some measurement errors occured during the S-UT-3 Test and that Westinghouse feels that differences between the Westinghouse and semiscale reactors causes the errors in the level measurements in the

S-UT-3 Test. S-UT-3 was a communicative break in the cold leg of Semiscale. Further tests were conducted in which Semiscale was modified to more closely resemble a Westinghouse facility. Published reports imply that the data is subject to correct interpretation. The heated thermocouple method of determining reactor vessel liquid level RVLMS, has been tested at ORNL[17, 18] and at Combustion Engineering. Reference 10 shows the design basis for the RVLMS and results from some of the tests performed at ORNL and CE. The RVLMS, (a heated junction thermocouple system) is now offered by Combustion Engineering to nuclear plant operators. Technical evaluations[29] performed by ORNL on Inadequate Core Cooling using Differential Presure for Reactor Vessel Level Measurement describes the complete Westinghouse RVLIS system. The technical evaluation[30] relating to the detection of inadequate core cooling using heated junction thermocouples for reactor level measurement describes the complete Combustion Engineering system designated RVLMS. It is interesting to note that both Westinghouse and Combustion Engineering include as part of their ICC systems the output of selected core exit thermocouples. Recent studies indicate that during three accident simulations at LOFT[31] the core exit thermocouple did not adequately indicate ICC conditions. The ICC conditions that occurred in very different hydraulic conditions in the reactor coolant system and may bring into question the reliance this instrument to detect ICC conditions.

ORNL's ultrasonic probe,[17, 18, 19] was designed to measure simultaneously the level, temperature and density profile of the fluid in which it is immersed.

4.2.1 Research and Development in the Measurement and Detection of Reactor Vessel Voiding

Several research and development projects have been undertaken to improve the measurement or indication of inadequate core cooling, reactor vessel level and primary system voiding.

Both in-core and ex-core neutron detection have been looked at as a source of information of reactor vessel liquid level. Current work at the INEL[20] utilizes both in-core and ex-core neutron detectors to detect reactor vessel liquid level at LOFT.

A report has been published reporting the results of using in-core neutron detectors and shows the result of changing water level within the reactor vessel. The measured data are current plots and as with all SPND data the current values are extremely low levels. The data presentation show trends but the uncertainty levels along with the low levels of the measured current makes the data open to interpretation. The ex-core use of self powered neutron detectors (SPNDs), involves the use of the detectors outside the LOFT reactor vessel in the annulus and as of yet published data is not available.

Another neutron detection technique is being studied at Penn State University.[20] The study at Penn State is investigating the use of BF_3 detectors at the side of the reactor vessel attempting to indicate water level from the detection of neutrons. Recent EPRI funded work[22] utilized ex-core neutron detection of water level from above the reactor head at Farley Unit One. These tests were performed during a normal plant shutdown and looked at the water level during several draindown and refill tests that allowed the water level to reach as low as the core nozzle midplanes. This test data indicates that detection of the water level in the reactor vessel may be possible within two feet of the top of the core. This data again shows the low count rates of neutron detection signals.

Pump power tests at LOFT[23] indicates a correlation exists between pump power and voiding in the primary system. These findings resulted in a research memorandum[24] being issued to the NRC. This memorandum reports that LOFT experiments show RCP motor input electrical power is a direct function of the coolant density. Therefore the power data can be used to determine the type of transient and thus, the correct RCP operation. Their data shows that pump motor power when displayed with cold leg temperature, provides information which:

1. Allows unambiguous seperation of a rapid cooldown transient from a LOCA.

2. Provides the operator with RCS inventory information when it is needed, that is, when the pumps are on and after the pressurizer is empty.

3. Provides measurement of system voiding as a criterion for both RCP and HPI control.

4. Allows RCP operation when no coolant is being lost from the system and when the HPI can make up the break flow early in a loss of coolant event.

The data gained from these tests confirmed the previously held notion that pump motor power and current are related to RCS coolant density.

Studies have been made and methods proposed for utilizing the RCS pump power data in helping reactor operators to identify and take the proper actions to recover from reactor system transients.

Primary pump vibration measurements are being studied by several groups. Studies are underway to use vibration measurements on the primary pump and primary piping to detect the onset of cavitation[25] to protect the primary pump. Other vibration tests at LOFT[20] indicate that some correlation may exist between primary system vibration and the existence of voids within the primary system. This vibration method has shown that measurements and data are plant specific and are affected by piping geometry.

4.3 Steam Generator U-Tube Voiding

Several schemes have been tested to attempt to detect voiding in the steam generator U-Tubes. Voiding detectors[2] were tested in a mock up of a steam generator. High temperature microphones were placed at various locations on the steam generator tube sheet at LOFT.[26] Both tests had little success.

4.4 Break Identification and Location

Break identification and location is an extremely difficult measurement to make. Responding utilities did not indicate any current studies underway to provide direct indication of break size or location. Literature searches of utility or government funded programs did not indicate any currently funded programs.

5. CONCLUSIONS

The results of this study to date have shown that the responding utilities are very cost and regulation sensitive. The utilities have either installed or are in the process of installing reactor vessel level indicating systems. The most common response to the questionnaire indicated the Westinghouse RVLIS was being installed. Reactor vessel level or voiding continues to be a measurement problem with operating utilities as well as experimental reactors. Several current research programs indicate some potential progress toward the solution of this measurement problem although more research and testing is indicated.

Responding utilities have indicated that there is no direct measurement capability to measure low velocity, less than 0.1 m/s, primary RCS flow. Several projects are currently addressing the problem of inadequate core cooling. The existence of a two phase condition in the primary system is presently indicated by correct interpretation of other reactor measurements available to the operator, e.g., the loss of subcooling at some measured location, pressurizer level, RCS pressure, hot leg temperature or core exit temperature. All of these measurements are subject to incorrect interpretation which could lead to incorrect action by the operator to bring the reactor to a stable condition.

Steam generator u-tube voiding is also a parameter that is not indicated directly although some studies have attempted to design an instrument or method to detect voiding. The literature search has not indicated a reliable method that has been developed to date.

Break location and size identification is another difficult measurement to indicate and there are no current projects or studies toward this end indicated in the literature search.

6. PROPOSED INSTRUMENTS FOR FURTHER EVALUATION

As a result of the literature search and utility/vendor survey the following instruments are proposed for further testing study or evaluation.

1. To measure low velocity flow it is proposed to do further testing/evaluation on the cooled thermal velocimeter.[2, 3, 4] The potential exists to utilize existing penetrations and or thermowells to install such a device in existing plants. Recent scoping tests indicate that specially designed thermal velocimeters can detect low velocity flow when installed inside a standard nuclear grade thermowell.

2. To detect or indicate voiding or thermal hydraulic conditions in the RCS pump and primary system it is proposed to investigate in further depth the primary pump vibration measurements[20, 25] to determine the validity of this measurement as an indicator of the primary system thermal hydraulic condition. This study would utilize the existing study that indicates primary pump power as a measure of fluid density.[23, 24] Experimental data from LOFT indicates that the vibration data recorded from an accelerometer mounted on a RCS primary pump may contain information that could aid in early detection of RCS voiding.

3. To detect voiding in steam generator u-tubes it is proposed to try to utilize high temperature sensors to detect flow induced acoustic noise from RCS flow through the steam generator with the hope of correlating this flow induced acoustic signal with either flow velocity or a voiding condition in the u-tubes.

4. In an attempt to detect a break in the RCS piping it is proposed to study the currently available pressure instrumen's in the RCS[27] and study the feasibility of detecting a break and determine its location and approximate size by measurement of abnormal pressure drops through the various parts of the primary

piping system. Acoustic signals and detecting location and size by amplitude and propagation time as utilized in state of the art acoustic emission systems[28] may lend itself for use in the task of locating a RCS break. Acoustic detection of leaks in piping systems are being utilized and advances made in recent years in interpreting accoustic signals may aid in the application of this technique in the detection and location of breaks in the RCS piping.

TWO-PHASE INSTRUMENTATION SURVEY

Instruments for Further Evaluation

TABLE 2

Instrument (Method)	1.Low Flow Cooled Velocimeter	2.Primary Pump Power & Vibration
Instrument Status	Moderately developed. More testing required to develop application and installation techniques.	Power measurement already required. High temperature sensors commercially available for installation on primary pump system for vibration measurement.
Power Plant Suitability	Passive device,no moving parts. Requires only cooling water.	Instruments should be very suitable for nuclear plant use.
Reliability	Good: Type K thermocouples have an excellent history in nuclear applications.	Should be good. Power measurement already being made. Other sensors have proven usable in radiation environment.
Retrofit Capability	Plan to try to use nuclear grade thermowells.Existing thermowells or penetrations might be usable.	No vessel or piping modifications should be necessary.
Performance	Performance is good, basic instrument accuracy is good. Accuracy and performance in thermowells would be established in further testing.	Measurement uses techniques and sensors utilized with good results and are currently in use.
Signal Processing	Simple processing is required to obtain direct velocity information	Power data should require simple processing.Vibration data processing could become more complex and require some interpretation.
Signal Usability	Range would have to be established in thermowell. Signal would be usable in RCS decreased flow conditions probably in the velocity range of 0.05 to 2.0 meters/sec.	Information could be directly used by operator as a status indicator and could potentially indicate the onset of other pump or system problems.

TWO-PHASE INSTRUMENTATION SURVEY

Instruments for Further Evaluation

TABLE 2

Instrument (Method)	3.Steam Generator U-Tube Voiding	4.RCS Piping Break (Pressure) [Acoustic Emission]
Instrument Status	Application is developmental Acoustic measurements now in use for reactor applications.	(Application is developmental. Would be plant specific in utilizing existing instruments) [Commercially available systems could be directly applicable to measurement technique.Would be plant specific in transducer location.]
Power Plant Suitability	No penetration into RCS.	(Would attempt to use existing sensors) [Acoustic systems already in use in nuclear plants.(loose parts monitors and valve leak detection)]
Reliability	High temperature and radiation resistant transducers are available.	(Good,utilizing qualified and approved instruments.) [Good,Transducers have a history of operation in nuclear environment.]
Retrofit Capability	Outside pressure boundary installation.Minimal perturbance	(No additional penetrations should be required.) [Outside pressure boundary installation]
Performance	Should be good. Sensitivity may be a problem and discrimination of signal and system generated noise.	(Possibilities are good.Presure resolution and accuracy of existing sensors may be a problem) [Has potential.Techniques are in use in the industry.]
Signal Processing	Processing is required,using proven principals.	(Processing is required.Comparison of normal versus measured pressure readings and establishing limits.) [Processing required.Techniques have been developed.]
Signal Usability	Signal potentially useful in several ways.(loose parts monitor,steam generator status indicator.)	(If method proves feasible,output could be RCS status indicator.) [Potentially to detect loose parts or leaking valves as well as RCS break.]

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