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# INSTRUMENT ACCURACY IN REACTOR VESSEL INVENTORY TRACKING SYSTEMS

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## I. INSTRUMENTATION NEEDS FOR DETECTION OF INADEQUATE CORE COOLING

Studies of the Three Mile Island accident identified the need for additional instrumentation to detect inadequate core cooling (ICC) in nuclear power plants. Industry studies by plant owners and reactor vendors[1] supported the conclusion that improvements were needed to help operators diagnose the approach to or existence of ICC as well as to provide more complete information for operator control of safety injection flow to minimize the consequences of such an accident. In 1980, the U.S. Nuclear Regulatory Commission (NRC) required further studies by the industry[2] and described ICC instrumentation design requirements that included human factors and environmental considerations.[3] On December 10, 1982, NRC issued to Babcock & Wilcox (B&W) licensees orders for Modification of License and transmitted to pressurized water reactor licensees Generic Letter 82-28 to inform them of the revised NRC requirements. The instrumentation requirements include upgraded subcooling margin monitors (SMM), upgraded core exit thermocouples (CET), and installation of a reactor coolant inventory tracking system. NRC Regulatory Guide 1.97, which covers accident monitoring instrumentation,[4] was revised (Rev. 3) to be consistent with the requirements of item II.F.2 of NUREG-0737.[3]

Some of the more significant requirements specified in item II.F.2 of NUREG-0737 are that

1. Instrumentation should provide an unambiguous indication of the approach to and existence of ICC;

2. Reactor water level measurement is to be considered;

3. The system must indicate the existence of ICC caused by various phenomena (i.e., high void fraction pumped flow as well as stagnant boil-off);

4. The presence of an unrelated phenomenon must not cause the system to erroneously indicate ICC;

5. Advance warning of the approach of ICC must be given;

6. The instrumentation must conform to Appendix B (Class 1E) of NUREG-0737;

7. Alarms and displays should be selected based on a human factors analysis; and

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8. Instrumentation indications must be integrated into emergency procedures and operator training programs.

## II. REVIEW OF PROPOSED PROGRAMS

The NRC, with assistance from Oak Ridge National Laboratory, reviewed generic ICC instrumentation systems proposed by reactor vendors, instrument manufacturers, and individual utilities. Plant-specific systems are still under review to assure that their designs meet NRC requirements. Two water level measurement methods have been developed and approved generically by NRC for application in nuclear power plants: the Westinghouse differential pressure (dp) system and the Combustion Engineering heated junction thermocouple (HJTC) system. Both of these systems were tested extensively under simulated accident conditions, including extremes of temperature and two-phase flow, and the results have been reviewed.[5,6]

## III. INSTRUMENT PERFORMANCE

Subcooling margin monitoring provides early indication of potential voiding but does not of itself provide any additional information about the possible approach of ICC. In some cases, the reactor vessel head water level monitor provides an initial indication of voiding at the same time that the hot-leg resistance temperature detectors (RTDs), or even the CETS, are indicating that subcooling still exists. This occurred in the Ginna steam generator tube rupture event and could happen in any overcooling event that results in loss of pressurizer level. The upper head can be the region of highest temperature, thus acting as the system pressurizer. In addition, a small-break loss-of-cooling accident (SBLOCA) with a leak in the upper head of the reactor vessel can result in upper head voiding.

The CETs provide perhaps the most reliable indication of the existence of ICC during stagnant boil-off when superheated steam conditions are present. However, some loss-of-fluid tests (LOFT)[7] suggest that during reflood or coolant injection the CETs may be subcooled while the core remains voided. (There is some indication in LOFT tests that CETs can be cooled by water falling back from the steam generator.) A serious condition can occur if loss of pumps in a highly voided situation permits the water level to collapse below the bottom of the core. The core can then heat up without detection because there is no coolant in the core to boil, and thus no superheated steam is generated to flow past and heat the CETs. A diverse measurement system that includes measurements of other parameters such as coolant inventory or water level is needed to complement CET measurements under abnormal conditions. When the primary coolant pumps are running, the voids tend to be distributed throughout the system, and the resulting "froth" can provide adequate cooling when high void fractions exist. However, if pumping is continued, high void fraction mixtures are likely to cause pump damage.

The problems of possible ambiguities in water level and the difficulties of achieving adequate accuracy with water level or inventory tracking systems are considerable. The remainder of this paper will discuss the current status of industry efforts to achieve reliable and unambiguous water level indication.

### III.A Generic ICC Systems

The Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) uses redundant sets of three dp cells to measure pressure drop from the bottom to the top of the reactor vessel and from the hot legs to the top of the vessel. A wide-range transducer includes the pump dynamic head and is used to infer void fraction with the pumps running. A narrow-range transducer is calibrated to indicate full-scale with the static head across the vessel with pumps off, and the output is conditioned to display the equivalent collapsed liquid level in the vessel. The head-to-hot-leg measurement is used for head venting operations during long-term recovery. The Westinghouse RVLIS dp system uses sealed lines compensated for temperature and density effects inside containment, with the transmitters located outside containment to achieve an accuracy of better than ±6% (about ±2.5 ft) under degraded environmental conditions in containment.

The Combustion Engineering HJTC system measures reactor coolant liquid inventory using discrete HJTC sensors located at different levels within a separator tube that extends from the reactor vessel head to the top of the core. The separator tube allows any steam-water mixture to collapse and hence provide a steam-water interface at the collapsed liquid level. Heated thermocouple sensors at discrete axial levels within the tube indicate the presence of water at the measurement level. The HJTC system boasts very high accuracy (within a few inches), but resolution is limited to 2 to 4 ft because of the spacing between the limited number of discrete measurement points (typically 8).

III.B Installation Experience

A total of 30 operating nuclear power plants have installed one of the two generic systems. The limited operating experience obtained so far has been generally satisfactory, although a number of early failures and problems occurred in both system types, some of which necessitate field design changes. Twenty-one plants have received final NRC approval and have declared their systems operational. Two plants have opted to use a gamma-thermometer (GT) level measurement scheme very similar in principle to the HJTC system. However, testing and qualification of the GT system is incomplete. A number of plants have chosen, with varying degrees of success, to design their own dp measurement systems. Most of these systems use dp transmitters located inside the containment vessel, where they tend to give large errors when subjected to small-break LOCA conditions and can trap noncondensable gases in their impulse lines. Measurement uncertainties claimed for these systems range from ±10% to ±30%, even with density compensation schemes.

IV. ACCURACY GUIDELINES

NRC requirements have not included an absolute accuracy specification for the necessary water level measurements. Analyses based on small-break LOCA scenarios suggest that ±6% is acceptable but that ±15% probably is not. Fifteen percent is typically equivalent to more than one-half the core height, and this amount of uncertainty may leave inadequate time for an operator to respond appropriately. Also, too large an uncertainty will have a negative impact on system credibility and may lead operators to ignore or distrust the inventory measurement. Since the objective of the coolant inventory system is to alert the operator to an approach to ICC during a transient event, a reasonable criterion for an acceptable uncertainty of level measurement should be described in terms of the effect on the margin of advance warning provided an operator after initiation of a transient.

The maximum break size that falls into the small-break category is 0.1 $ft^2$. In the case of a postulated small-break transient with high-pressure safety injection unavailable, 15-20 min is the estimated time for the collapsed level to drop below the top of the core and for the CETs to indicate a temperature of 1200°F. This is considered to be sufficient time for the operator to recognize the small-break symptoms and to initiate emergency procedures to recover the plant. However, the analysis assumes that the actual level is known. Instrument errors can reduce the time between recognition of loss of inventory and onset of ICC. Figure XXV.4-1 is a schematic diagram of a typical PWR vessel in which the height of the water above the top of the core is typically 17 to 20 ft. Figure XXV.4-2 illustrates the predicted loss of inventory during a 0.1-$ft^2$ break and the effect of level measurement uncertainties on identifying uncovery of the core. The actual coolant inventory predicted by analysis for this transient is indicated by the heavy line at the center of the two error bands. The inner error band of ±6% would result in an uncertainty of about ±5 min of an available 20 min in indicating the onset of core uncovery. With an uncertainty of ±15% as denoted by the outer bands, warning may be nonexistent as the core may be almost completely uncovered when the instrument is indicating that the level is above the top of the core.

Because the ultimate indication of the approach to ICC is expected to come from the CETs, it is argued that diagnosis of the transient does not depend on the absolute accuracy of the level measurement system and that only a correct indication of the trend of the coolant inventory is required. In most cases, the operator is not instructed to take action on the basis of the level measurement system indication alone. However, some additional concerns arise if an uncertainty of ±15% is permitted. With an error of +15% (~6 feet), the actual water level could be well down into the core when the inventory system is still indicating that the core is covered. Conversely, with an error of -15%, the inventory system could indicate much less water in the core than actually exists. Increasing CET temperatures may give the operator adequate warning of approach to ICC. However, by the time the CET temperature indications begin to increase, the operator might react with actions far more drastic than necessary--such as depressurization and activation of low-pressure injection--with the attendant possibility of overcooling and resultant thermal shock to the vessel. System accuracies of ±6% appear to be attainable; therefore, accepting greater measurement uncertainties would require strong justification.

## V. PROCEDURES DEALING WITH INSTRUMENT UNCERTAINTY

Some licensees have proposed justification for in-containment dp systems with uncertainties of about ±15% under adverse containment conditions. Procedures were developed with "conservative" decision points which, coupled with appropriate operator training, purportedly would not mislead operators into initiating inappropriate action. "Conservative," in this case, means that the measurement uncertainty would be added to the water levels requiring operator action to mitigate the consequences of accidents or transients. The operator uses the dp measurement system to determine the reactor vessel water level. A number of the Emergency Operating Procedures (EOPs) use this information to confirm actions at levels corresponding to 3.5 ft above the bottom of the core (collapsed water level), the top of the core, the top of the reactor vessel outlet nozzles, and a full reactor vessel.

Problems have been identified with proposed procedures in conjunction with the corresponding generic guidelines, considering the large band of uncertainty in the level indications. Although the proposed level readings for initiating operator action would in some procedures be acceptable, in certain cases the water level action points may be misleading and, therefore, the intent of the emergency response procedure could not be accomplished using the level action points proposed. In particular, in the case of events that call for a strategy of keeping the core covered while maintaining the water level below the vessel nozzles in order to reduce the flow of water out of the nozzles, the proposed use of the instrumentation would not accomplish the objective. If the proposed conservative water level setpoints for operator action were used and a +15% error existed, attempts to keep the water level above the top of the core would result in the level actually being maintained in or above the hot leg nozzle. In the case of breaks in the hot leg, additional makeup flow would be required to compensate for the greater leakage from the break. In other words, the proposed set point does not permit the operator to conserve reactor coolant by maintaining the water level below the vessel nozzles, a significant objective of such procedures. Minimizing the flow out of the reactor coolant system is especially important in the event of a LOCA outside containment such as might result from a steam line break coupled with ruptured steam generator tubes.

## VI. NRC POSITION

The position taken by NRC was that the proposed procedures do not adequately accommodate high-level measurement system errors and therefore are unacceptable. The established NRC guideline for level measurement system uncertainty is ±6%. In the case exemplified above, even the smaller uncertainty of 6% could result in some additional flow out the hot leg. Although the NRC position has not ruled out an alternative procedural approach to dealing with large uncertainties in level measurement, there remains the concern that operators will not have confidence in a system with uncertainties larger than the ±6% guideline.

Most B&W type plants propose to use dp measurements for both the hot leg and the reactor vessel head. The hot leg measurement represents about two-thirds of the coolant available for cooling the core. The emphasis of the design is on the detection of voids in the hot leg "candy cane" and monitoring of coolant inventory from the top of the candy cane to the bottom of the hot leg. A generic approach to inventory monitoring in B&W type plants has not been proposed. Some applicants proposed hot leg measurements only, but NRC has taken the position that both hot leg and vessel head measurements are needed. Accuracies in the range of 6% appear achievable for these systems.

## VII. REFERENCES


1. P. G. Bailey and M. Merilo, A Review of Proposed Instrumentation for Measurement of Water Level as a Means of Detecting Inadequate Core Cooling in Pressurized Water Reactors, EPRI Report NP-2727-SR, November 1982.

2. Laurence E. Phillips, Approach to Inadequate Core Cooling Detection, ANS Topical Meeting on Anticipated Transients in Light Water Reactors, Jackson, Wyoming, December 1983.

3. U.S. NRC, Clarification of TMI Action Plan Requirements, NUREG-0737, November 1980.

4. U.S. NRC, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, NRC Regulatory Guide 1.97, Revision 3, May 1983.

5. G. N. Miller, J. L. Anderson, and R. L. Anderson, Inadequate Core Cooling Instrumentation Using Differential Pressure for Reactor Vessel Level Measurement, NUREG/CR-2628, ORNL/TM-8269, March 1982.

6. R. L. Anderson, J. L. Anderson, and G. N. Miller, Inadequate Core Cooling Instrumentation Using Heated Junction Thermocouples for Reactor Vessel Level Measurement, NUREG/CR-2627, ORNL/TM-8268, March 1982.

7. EG&G LOFT Test L2-5 Data Report, NUREG/CR-2826, EGG-2210, August 1982.

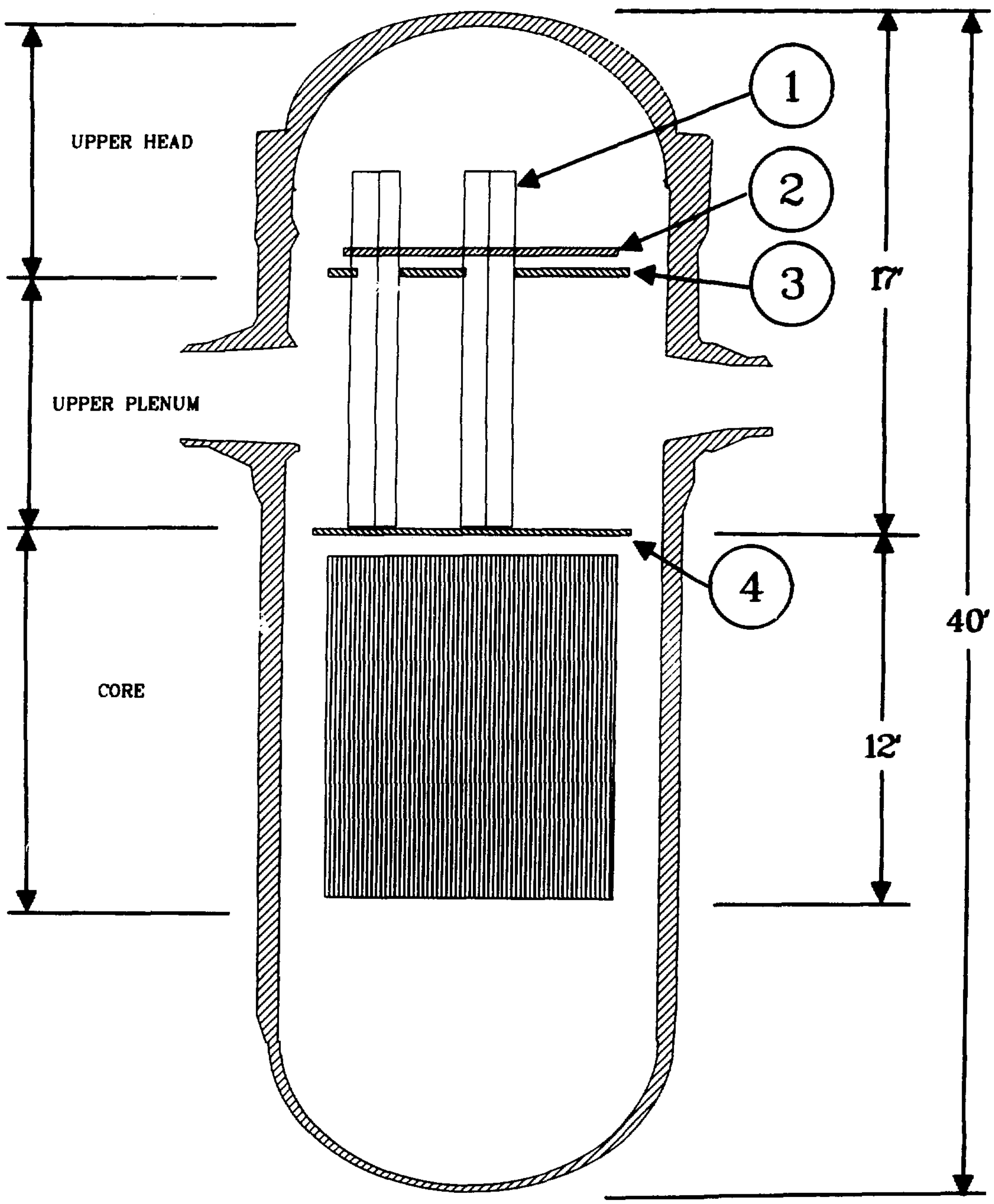


Fig. XXV.4-1, EFFECT OF LEVEL MEASUREMENT ERRORS ON COOLANT INVENTORY IN PRESSURIZED WATER REACTOR

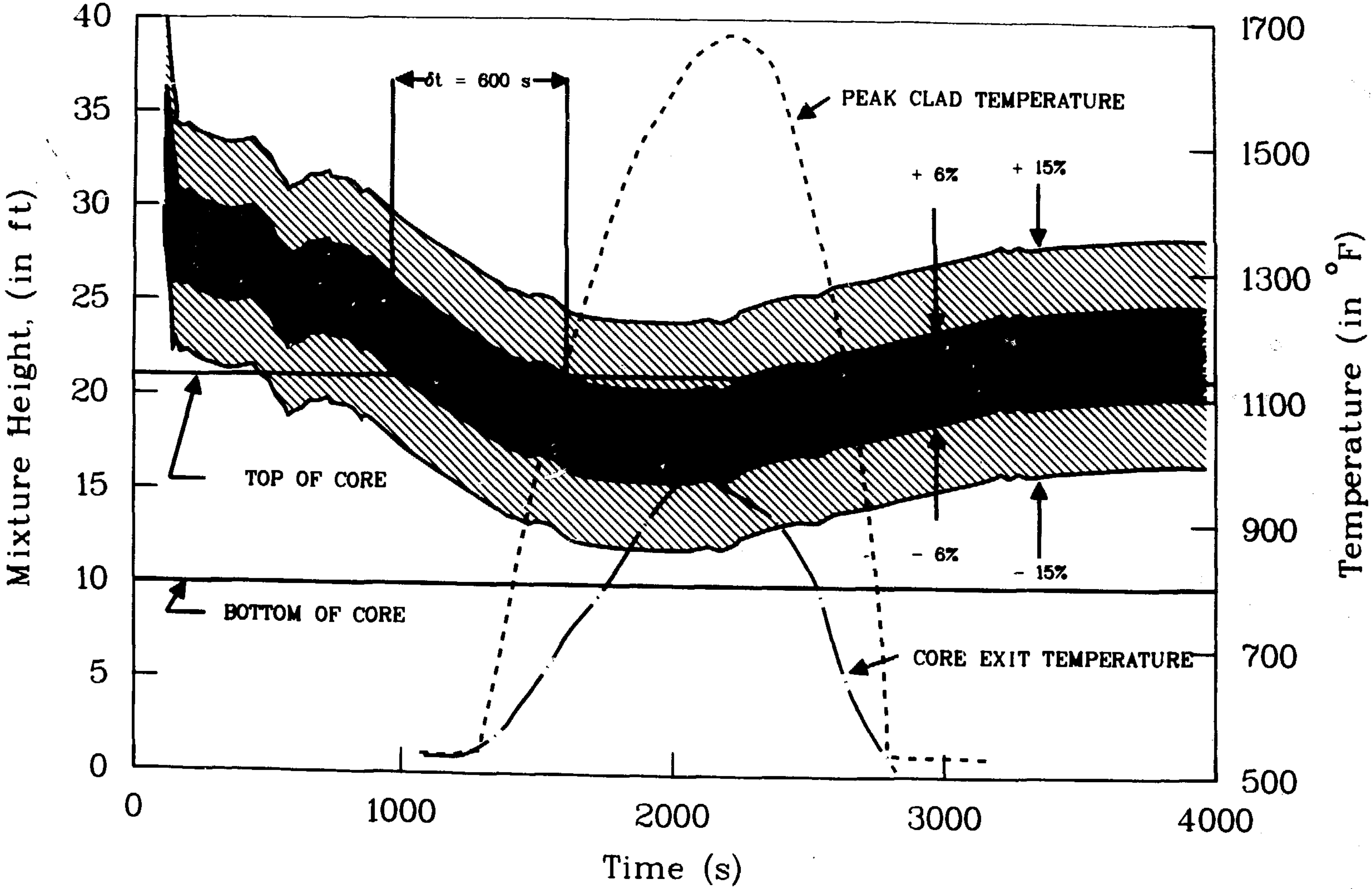


Fig. XXV-4.2, PREDICTED RESPONSE OF VESSEL LEVEL TO A 0.1-FT² BREAK