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March 10, 1983

Docket Nos. 50-348
50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Units 1 and 2
Inadequate Core Cooling Instrumentation System

Reference: (1) Alabama Power Company letter dated August 3, 1982
(2) Alabama Power Company letter dated February 1, 1983
(3) NRC Generic Letter No. 82-28, "Inadequate Core Cooling Instrumentation," dated December 10, 1982

Gentlemen:

In references 1 and 2, Alabama Power Company committed to provide the NRC with the status of its ongoing review of the commercially available reactor vessel level systems. This review included an evaluation of NUREG-0737 provisions, available systems, equipment lead times, design interfaces, installation considerations, outage schedules, system capabilities and inadequacies, etc. In addition, the NRC issued Generic Letter No. 82-28, "Inadequate Core Cooling Instrumentation" (reference 3). In accordance with these references, Alabama Power Company hereby provides the results of the reactor vessel level system review and addresses NRC Generic Letter No. 82-28.

The inadequate core cooling instrumentation presently installed at the Farley Nuclear Plant - Units 1 and 2 consists of the reactor coolant system subcooling margin monitor and the core-exit thermocouples. Based on the review of the current design of this inadequate core cooling instrumentation and as discussed in Supplement 5 of J. M. Farley Safety Evaluation Report, Alabama Power Company concludes that the subcooling margin monitor satisfies all applicable provisions of NUREG-0737. Alabama Power Company also concludes that the core-exit thermocouples satisfy the provisions of NUREG-0737 with the exception of the environmental qualification criteria specified by Appendix B to NUREG-0737. In letter dated February 9, 1981, Alabama Power Company has committed to

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monitor the Westinghouse program for upgrading the environmental qualification of the core-exit thermocouples. This program is currently scheduled for completion in May 1983. Upon completion of the Westinghouse program, Alabama Power Company will evaluate and compare the results to the NUREG-0737, Appendix B environmental qualification criteria in order to develop an approach regarding qualification of the core-exit thermocouples. Further details regarding the conformance of the presently installed inadequate core cooling instrumentation to NUREG-0737 is provided in Attachment 1.

The results of the ongoing review of the commercially available reactor vessel level systems are summarized in Attachment 2 and indicate that, at this time, no commercially available reactor vessel level system has been accepted by the NRC for operational use. Alabama Power Company has determined that the commercially available systems have potential problems affecting their design, installation and operation. These problems, presented in Attachment 3, range from system accuracy during accident conditions to incomplete generic designs. Consequently, as stated in NRC Generic Letter No. 82-28, the presently installed systems may only be used for training. Alabama Power Company has been assured by vendors and utilities with installed reactor vessel level systems that these problems are expected to be resolved in a timely manner. These installed systems, with possible modifications, are expected to obtain NRC approval for operational use.

Alabama Power Company concurs with and is committed to the objectives of NUREG-0737 to ensure that the Farley Nuclear Plant - Units 1 and 2 can detect the approach to inadequate core cooling. This commitment has been demonstrated by Alabama Power Company's participation in a pilot project for the non-invasive reactor vessel level system. The unsuccessful demonstration of the non-invasive reactor vessel level system led Alabama Power Company to take the initiative to conduct the detailed review of the commercially available reactor vessel level systems discussed above. It is now Alabama Power Company's intention to select the design of a reactor vessel level system that has been operationally proven following the completion of a post-implementation review and approval for operation by the NRC.

Alabama Power Company believes that this course of action is prudent since several utilities and the NRC are actively evaluating the capabilities of presently installed reactor vessel level systems. Following NRC operational acceptance of a reactor vessel level system, Alabama Power Company could therefore install a system that has been verified to be of adequate design, operationally proven and sufficiently integrated with existing plant systems, procedures and training. Alabama Power Company may thereby minimize costs that are presently being incurred by other utilities to resolve design, installation or operational anomalies. Additionally, the installation of an NRC operationally accepted system would allow for prompt operator training and use and preclude a situation similar to that now experienced at the Farley

Nuclear Plant with the reactor vessel head vent system. This system was installed more than two years ago at a cost of over $900,000 and has never received NRC approval for operation.

Attachment 4 provides a typical implementation schedule (start date based on purchase order issuance) for the engineering, procurement and installation of either of the two commercially available reactor vessel level system at the Farley Nuclear Plant - Units 1 and 2. This schedule is based on typical five-week durations for refueling outages, 85% capacity factor for fuel cycles, and engineering, procurement and installation durations unimpeded by other licensing requirements and/or plant initiatives. Alabama Power Company conservatively estimates that it would take up to 47 months including 3 refueling outages to complete the implementation of a reactor vessel level system at each unit. The implementation costs for either commercially available system is estimated to be on the order of $4,500,000 to $5,000,000 for both units. The actual schedule for the installation and upgrading of inadequate core cooling instrumentation, as necessary, will be integrated into the activities to satisfy Supplement 1 to NUREG-0737.

The safe operation of Farley Nuclear Plant - Units 1 and 2 will not be jeopardized during the period that the commercially available reactor vessel level systems are demonstrated operationally feasible and functionally accepted by the NRC. Safe operation will also not be jeopardized while an integration schedule is being developed and installation of inadequate core cooling instrumentation system modifications is completed, as necessary. The present emergency response capabilities at Farley Nuclear Plant - Units 1 and 2 will not be impaired since the implementation of any modifications for inadequate core cooling instrumentation will be fully integrated in accordance with NUREG-0737, Supplement 1. Reactor vessel level instrumentation is to enhance the present emergency response capabilities and is not to provide the sole safety function. The Westinghouse Owners Group has submitted emergency operating guidelines to the NRC for review that do not include the use of reactor vessel level instrumentation such that the upgrading of emergency operating procedures at Farley Nuclear Plant - Units 1 and 2 will not be impeded during the operational acceptance of the reactor vessel level systems by the NRC. Such assurance of safe operation is provided since the Farley Nuclear Plant has installed all instrumentation required to develop plant-specific procedures based on the upgraded Westinghouse Owners Group Generic Emergency Operating Guidelines.

If there are any questions regarding this matter, please advise.

Yours very truly,

RBMcDonald

for E. L. Clayton, Jr.

FLCJr/MAL:lsh-D39
Attachments
cc: Mr. R. A. Thomas
Mr. G. F. Trowbridge
Mr. J. P. O'Reilly
Mr. E. A. Reeves
Mr. W. H. Bradford

SWORN TO AND SUBSCRIBED BEFORE ME
THIS 10th DAY OF MARCH, 1983

Robert W. Stewart
Notary Public

My Commission Expires:
10/27/85

ATTACHMENT 1

STATUS OF CONFORMANCE OF THE PRESENTLY INSTALLED
FARLEY NUCLEAR PLANT ICC INSTRUMENTATION
WITH NUREG-0737, II.F.2 CRITERIA

NRC Request:

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.

APCo Response:

The Inadequate Core Cooling (ICC) Instrumentation System presently installed at Farley Nuclear Plant Units 1 and 2 is comprised of reactor coolant system subcooling margin monitors, core exit thermocouples and upgraded Emergency Operating Procedures (EOPs). The existing EOPs, which were modified in response to post-TMI requirements (I.E. Bulletin 79-06), provide the operator with the necessary guidance required to respond adequately to emergency conditions. As stated in Alabama Power Company's letter of June 4, 1982, the Farley Nuclear Plant EOPs will be upgraded to address the concerns that have been subsequently identified in Generic Letter 82-33 to allow the operator to prevent critical safety function challanges based on plant conditions without need of diagnosing the cause of the transient. The RCS subcooling margin monitors compare coolant temperature to reactor system pressure and indicate the degrees of subcooling or saturation margin available. A decrease in saturation margin provides an indication to the operator that the core is approaching an ICC condition. The core exit thermocouples provide indication of a decrease in the coolant inventory below the top of the core.

An evaluation of conformance with NUREG-0737, item II.F.2, Attachment 1 and NUREG-0737, Appendix B for the reactor coolant system subcooling margin monitor and the core exit thermocouples has been performed. By letters dated August 6, 1980, September 22, 1980, January 14, 1981 and February 9, 1981, Alabama Power Company has provided the NRC information on the conformance of the subcooling monitor and core exit thermocouples to the design and qualification criteria of NUREG-0737, item II.F.2, Attachment 1 and NUREG-0737, Appendix B. An NRC evaluation of this information is contained in Supplement 5 of the Joseph M. Farley Nuclear Plant Unit 2 Safety Evaluation Report which concludes that the subcooling monitor meets all applicable requirements of NUREG-0737 and Regulatory Guide 1.97. The core exit thermocouples were also found to meet the applicable requirements of NUREG-0737 and Regulatory Guide 1.97 with the exception of environmental qualification.

A summary of the information requested on the subcooling monitor is contained in the attached Tables 1 and 2. A summary of the information requested for core exit thermocouples in NUREG-0737, item II.F.2, Attachment 1 is provided below.

The thermocouple system utilizes 39 thermocouples in Unit 1 and 51 thermocouples in Unit 2 positioned to measure fuel assembly coolant outlet temperature at the preselected core locations shown in Figures 1 and 2 (FSAR Figures 4.4-16 and 4.4-17). The thermocouples are the chromel-alumel type and have an accuracy of ±2°F.

As a result of Alabama Power Company's actions taken in response to I & E Circular 80-15, a modification was implemented for realignment of certain thermocouples to the upper head region as a means of indicating upper head voiding during natural circulation. The present design calls for 16 thermocouple inputs for the core subcooling meter (eight per channel), two of which are upper head thermocouples.

The primary means of monitoring incore thermocouple temperature is the core subcooling monitor system. Each channel of the subcooling monitor receives inputs from 8 thermocouples (2 per core quadrant per channel, for a total of 16 thermocouples). A digital readout of any of the 16 single thermocouple temperatures may be obtained at the subcooling monitor panel located in the control room. The upper limit of the readout is in excess of 2300°F.

All of the control equipment for the thermocouple system is located on a rack in the control room. A multipoint precision indicator has been provided to indicate the temperature sensed by the thermocouples. Switches are provided on the control room panel to select the thermocouple desired to be read.

An additional selector switch located on the front of the panel allows either the low (100-400°F) or high (400-700°F) range measuring circuit to be used. Besides being directed to the indicator, the thermocouple outputs are also applied to the plant computer (up to 1900°F).

The instrumentation required for displaying the thermocouple readouts on the Main Control Board is powered from the 1E inverter. By manual means through use of an RTD bridge, the hot junction box temperature may be accurately determined without reliance on power.

The thermocouple system is presently qualified to IEEE 323-1971. Alabama Power Company has committed to monitor the Westinghouse generic program to upgrade the qualification of the present thermocouples. This program is scheduled for completion in May 1983.

A description of the computer, software and display functions associated with ICC monitoring in the plant was provided to the NRC by Alabama Power Company letters dated September 22, 1980, January 14, 1981 and February 9, 1981. An NRC evaluation of this information is contained in Supplement 5 of the Joseph M. Farley Nuclear Plant Unit 2 Safety Evaluation Report which concludes that the test programs are acceptable for assuring continued adequate performance of the subcooling monitor, incore thermocouple readout panel and plant process computer. This information, which is summarized below, is also applicable to Unit 1 due to similarity of design.

Core Subcooling Monitor - Initial Testing

The installed core subcooling monitors have been tested by a calibration and functional test procedure for both units. These tests are a comprehensive software and hardware performance verification which includes initial calibration of inputs and functional testing of microprocessor self test features, calculation outputs, display capabilities and alarm outputs. The maximum thermocouple indication error found during testing of the Unit 1 subcooling monitor was 7°F at 2000°F in a conservative (high) direction. The maximum error found during the Unit 2 test was 6°F.

Periodic Retesting

Periodic retesting of the subcooling monitor will be performed at refueling intervals under the Preventive Maintenance Program, utilizing an Instrument Maintenance Procedure (IMP). The IMP is essentially a repeat of the initial calibration and functional test procedure used for acceptance testing in the initial set-up of the subcooling monitor. This test is a full functional checkout of proper input conversion, calculational

output accuracy, Main Control Board (MCB) meter indication accuracy, MCB annunciation and local alarm outputs, temperature and pressure auctioneering, display function outputs and diagnostic output verification. Testing is accomplished following input calibration verification by using analog test signals generated from calibrated plant test equipment to simulate various combinations of temperature and pressure conditions which encompass the normal operating (various subcooled conditions), approach to saturation, saturation and superheated ranges. For each temperature and pressure test point combination, the subcooling monitor calculational outputs and MCB meter indications are verified to be in agreement with expected values as determined from the steam tables. Alert and alarm conditions are simulated to verify proper MCB meter indication, annunciation actuation and local subcooling monitor alarm indications. In addition, all diagnostic and self testing features and display function outputs are verified to be functional.

Self-test features are used to test alarm circuitry and MCB meter operation on a routine basis. Inputs which are common to the subcooling monitor and process computer are routinely verified to be in agreement. Subcooling monitor display functions are also routinely used to check monitor operability.

Diagnostic Capabilities

The subcooling monitor has a range of diagnostic capabilities including channel self-test, channel processor failure, power failure, sensor input failure and calculational output failure. These diagnostics, along with vendor-recommended maintenance, will be utilized to ensure the subcooling monitor channels are operable.

Incore Thermocouple Readout Panel - Initial Testing

Testing of the Incore Thermocouple Readout Panel was performed for Unit 2 by functional test procedure. Thermocouple inputs and readout panel performance was verified to be within expected tolerances. Input verification was performed on each thermocouple input. Maximum thermocouple indication error over the temperature range tested was 4.5°F in the conservative (high) direction on Unit 2.

Periodic Retesting

Retesting is to be accomplished under the Preventive Maintenance Program and will be performed at refueling intervals.

Diagnostic Capabilities

There are no specific diagnostics internal to the Incore Thermocouple Readout Panel. Good industry practice will be utilized to ensure operability of the readout panel.

Plant Process Computer (P2500) - Initial Testing

The plant process computers for both units have been tested using a combined input verification and acceptance test procedure. These tests are a comprehensive software and hardware performance verification which includes calibration of inputs, individual program performance testing, and calculated value, display, and alarm outputs verification. The thermocouple program package and incore thermocouple inputs are verified and calibrated at this time. Unit 1 and Unit 2 preoperational test results indicated errors of less than 4°F over the range tested.

Periodic Retesting

Incore thermocouple input verification to the process computer will be performed at refueling intervals.

Diagnostic Capabilities

Incore thermocouple inputs are monitored and alarmed by the process computer for sensor input failure and temperature alarm limits. Thermocouple program outputs are also monitored by the computer for alarm limits. Preventive maintenance and good industry practice are utilized to ensure process computer system operability.

Conclusion:

It is therefore concluded that the core subcooing monitor meets all requirements of NUREG-0737, Item II.F.2 and the incore thermocouples meet all requirements except for environmental qualification which is currently being evaluated as a part of NUREG-0737, Supplement 1 (Generic Letter 82-33) issued December 17, 1982.

TABLE 1

INFORMATION REQUIRED FOR THE SUBCOOLING MONITOR

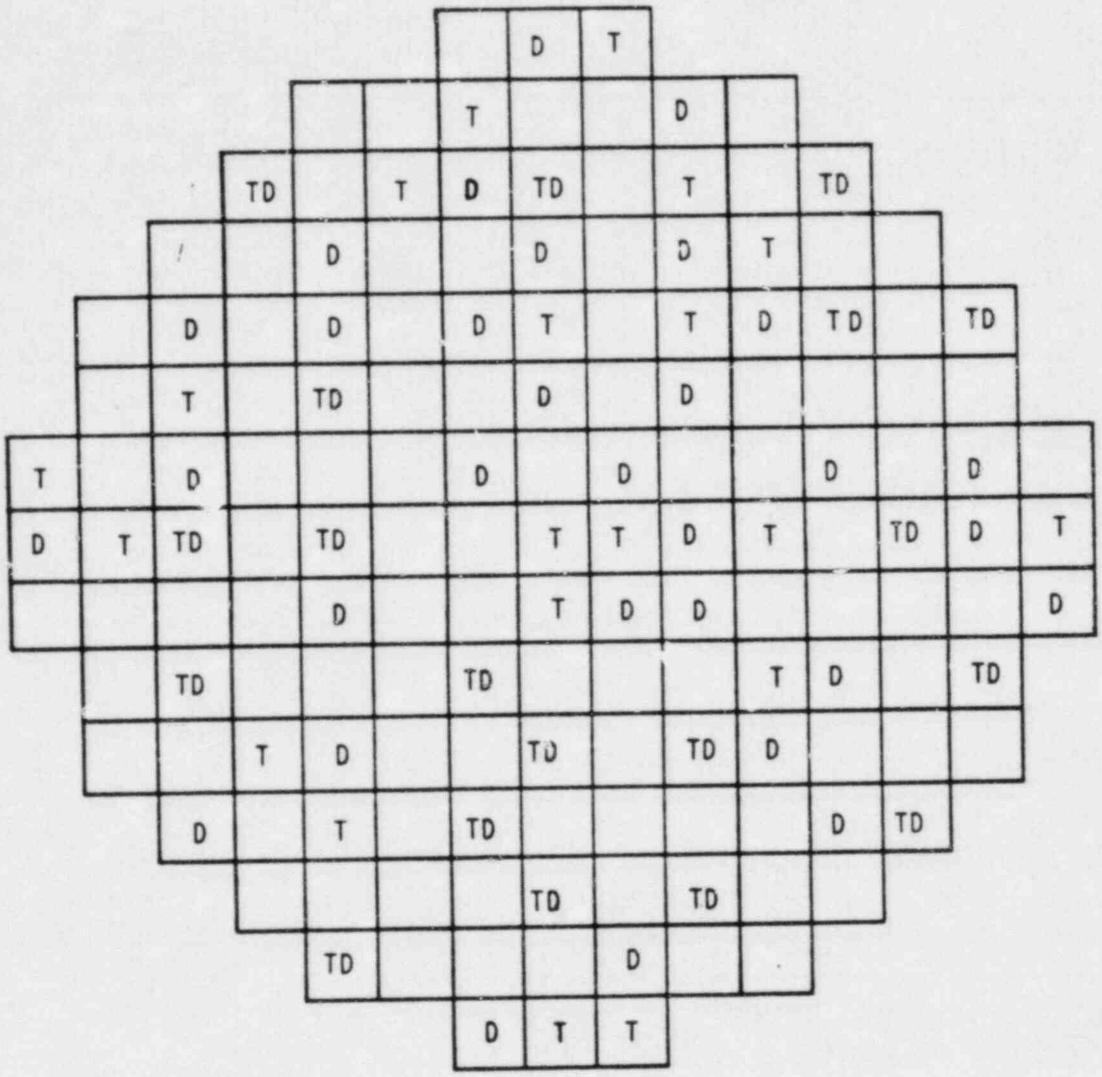
DISPLAY		
1.	Information displayed	T-Tsat subcooled T-Tsat superheat
2.	Display type	Analog and Digital
3.	Continuous or on demand	Analog - continuous Digital - on demand
4.	Single or redundant	Redundant
5.	Location of display	Meter - main control board Microprocessor - main control room instrument racks
6.	Alarms (include setpoints)	Caution: 25°F subcooled for RTD 15°F subcooled for T/C Alarm: 0°F subcooled for RTD and T/C
7.	Overall uncertainty	Digital - 4°F for T/C; 3°F for RTD Analog - 5°F for T/C; 5°F for RTD
8.	Range of display	Calibrated region - 250°F subcooled to 2000°F superheat overall; never offscale
9.	Qualifications	Qualified for main control room mild environment
CALCULATOR		
1.	Type	Dedicated digital
2.	If process computer is used, specify availability	N/A
3.	Single or redundant calculators	Redundant
4.	Selected logic	Highest Temperature for RTD or T/C and lowest pressure
5.	Qualifications	None at present
6.	Calculational technique	Functional fit - ambient to critical point

<u>INPUT</u>	
1. Temperature (RTDs or T/Cs)	RTD, T/C, and T_{ref}
2. Temperature (number and location of sensors)	RTD - 2 hot and 2 cold legs per channel 8 in-core T/C per channel
3. Range of temperature sensors	RTD - 0-700°F T/C - 0-1650°F (calibration unit range 0-2300°F)
4. Uncertainty of temperature sensors	±0.7% RTD
5. Qualifications	IEEE 323 1971
6. Pressure (specify instrument used)	RCS Wide Range Pressurizer
7. Pressure (number and location of sensors)	2 wide range - Loops 1 and 3 1 narrow range - Pressurizer (per channel)
8. Range of pressure sensors	Wide range - 0-3000 psi Narrow range - 1700-2500 psi
9. Uncertainty of pressure	Wide range - ±1% Narrow range - ±1.5% Pressurizer - ±1.0%
10. Qualifications	IEEE 323 1971
<u>BACKUP CAPABILITY</u>	
1. Availability of temperature and pressure	Temp - Swap between T/C and RTD Press - Can defeat any of the three inputs. System uses autioneered low pressure.
2. Availability of steam tables	Saturated steam tables and tables to verify required sub-cooled conditions are included in Emergency Procedures.
3. Training and operators	Operators have been trained on the use of the subcooling monitor to determine required subcooling conditions.
4. Procedures	Emergency procedures have been revised to describe the utilization of the subcore cooling monitor

readout and appended portions of the steam tables to determine subcooling conditions. A system operating procedure has been written to guide operators in the operation of the subcooling monitor. Appropriate personnel have been trained in these procedures.

TABLE 2
Appendix B (of NUREG-0737,II.F.2) Information
for the Subcooling Monitor

1. Environmental qualification	Display is environmentally suited for control room. Not seismically qualified. Backup procedures for use of steam tables are provided.
2. Single failure analysis	Redundant displays and calculator
3. Class IE power source	Powered from vital instrument bus.
4. Availability prior to an accident	Continuous main control board indication
5. Quality Assurance	In accordance with Plant QA Procedure
6. Continuous indication	Continuous indication of both channels on main control board
7. Recording of instrument outputs	Inputs recorded by Plant Process Computer. Procedures available to convert readings to degrees subcooling
8. Identification of instruments	Subcooling monitor identified as a Post Accident Instrument
9. Isolation	Inputs to core subcooling monitor picked up on the isolated (control) side of the protection channel

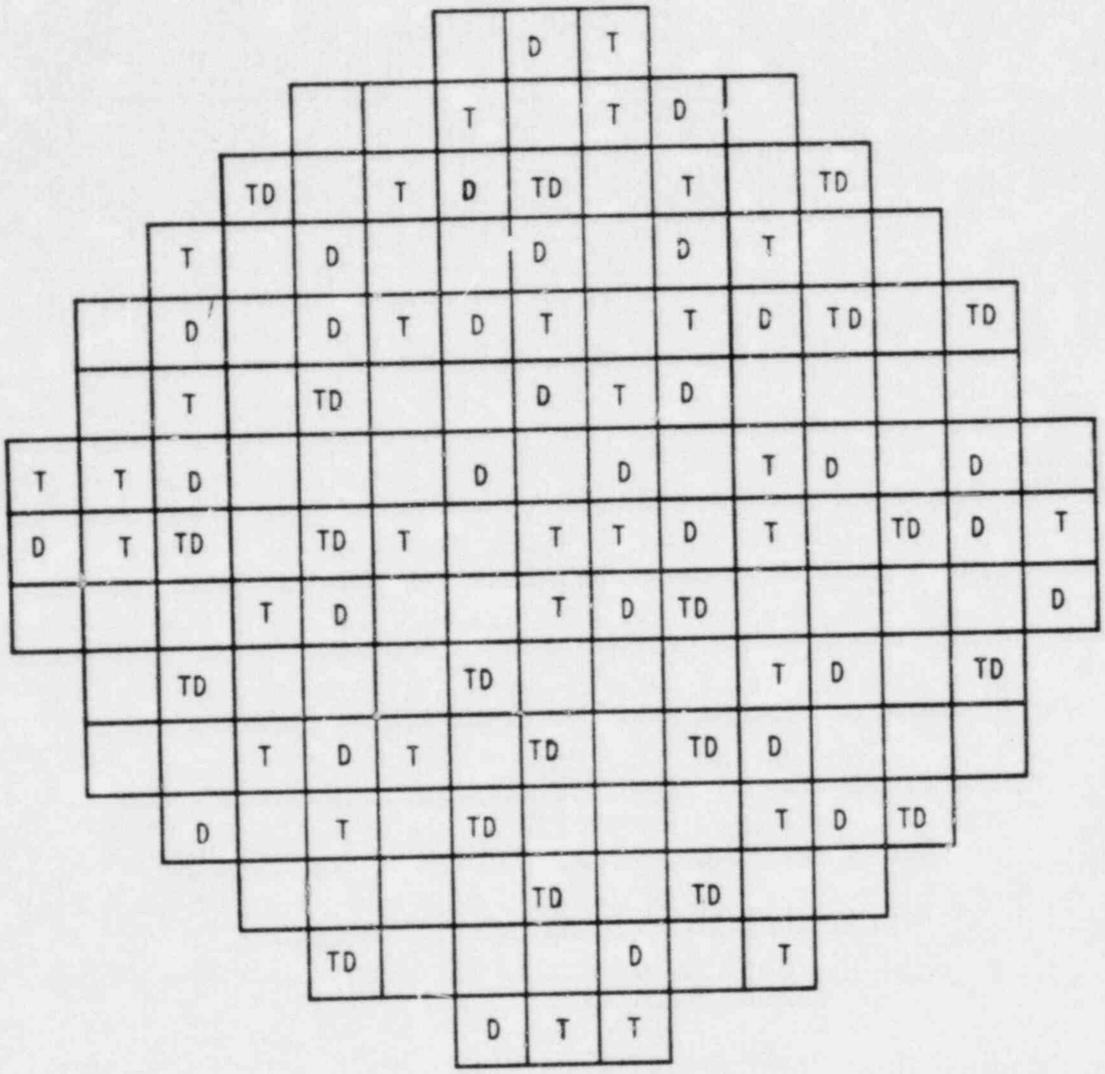


T = THERMOCOUPLE (39)

D = MOVABLE INCORE DETECTOR (50)

Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	UNIT 1 DISTRIBUTION OF INCORE INSTRUMENTATION FIGURE 1

4440-1



T = THERMOCOUPLE (51)

D = MOVABLE INCORE DETECTOR (50)

Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2	UNIT 2 DISTRIBUTION OF INCORE INSTRUMENTATION
		FIGURE 2

4440-1

ATTACHMENT 2

SUMMARY OF THE
JOSEPH M. FARLEY NUCLEAR PLANT
REACTOR VESSEL LEVEL SYSTEM
FEASIBILITY STUDY

I. INTRODUCTION

This report has been prepared to address the feasibility of installing a Reactor Vessel Water Level Instrumentation System at Joseph M. Farley Nuclear Plant - Units 1 and 2. This instrumentation is recommended by the Nuclear Regulatory Commission as part of the system to detect Inadequate Core Cooling (ICC). This recommendation was addressed by the NRC in NUREG-0578 (Short-Term Lessons Learned) and clarified in NUREG-0737 (Clarification of TMI Action Plan Requirements). The purpose of the ICC instrumentation is to provide the operator unambiguous indication of the approach to and existance of inadequate core cooling caused by such various plant phenomena as high-void fraction-pumped flow and stagnant boil-off.

In order to implement the recommendation established by the NRC, instrumentation systems capable of measuring reactor vessel water level have been developed by Westinghouse and Combustion Engineering. These systems employ two entirely different concepts to measure water level and both have received extensive generic design review by the NRC.

The purpose of this report is to outline the feasibility of the procurement and installation of the Westinghouse and Combustion Engineering reactor vessel water level systems at the Farley Nuclear Plant. The report consists of a description and evaluation of the two systems which includes the scope of supply and division of responsibility between the vendor, Bechtel, and other suppliers; cost estimates associated with hardware, engineering and installation; maintenance/operational considerations; licensing considerations; and conclusions. This report has been prepared with input from Westinghouse, Combustion Engineering, Bechtel, Southern Company Services, and Alabama Power Company and is intended to provide Alabama Power Company the information needed to determine the best method available to implement the NRC recommendation.

II. REACTOR VESSEL LEVEL SYSTEM DESCRIPTION

At the present time there are only two systems available to measure reactor vessel water level which have been reviewed by the NRC Staff and approved for installation. These systems are being offered by Westinghouse and Combustion Engineering. The following is a brief description of how the two systems measure vessel level.

Westinghouse Reactor Vessel Level Instrumentation System (RVLIS)

The Westinghouse Reactor Vessel Level Instrumentation System (RVLIS) uses differential pressure (d/p) measuring devices to measure water level and relative void content of the primary coolant system. These measurements are made by using two sets of three d/p cells as shown in Figure 1. These d/p cells measure the pressure drop from the bottom to the top of the reactor vessel, and from the hot legs to the top of the reactor vessel. The system utilizes d/p cells of differing ranges to cover different flow behavior with and without reactor coolant pump operation.

The system is redundant, utilizing separate sets of instrumentation for Train A and Train B. As shown in Figure 2, the system microprocessor accepts inputs from the d/p transmitters, 7 temperature sensors used to measure the temperature of the impulse lines, reactor coolant temperature (T_{hot}) and wide range pressure indications, and the running status of the reactor coolant pumps. The microprocessor unit performs the calculations necessary to compensate for density variations in the impulse lines and reactor coolant density changes. These calculations are used to compensate the d/p transducer outputs for differences in system density and reference leg density, particularly during the change in the environment inside containment following an accident.

The following measurements are provided by the Westinghouse system:

° Reactor Vessel Heat Plenum (ΔP_a) - provides a measurement of reactor vessel water level above the middle of the hot leg pipe when the reactor coolant pump (RCP) in the loop with the hot leg connection is not operating. When the pump in the loop with the P_a connection is in operation, the instrument is ranged such that the level display is offscale.

° Reactor Vessel-Narrow Range (ΔP_b) - provides an indication of reactor vessel water level from the bottom of the reactor vessel to the top of the reactor vessel during natural circulation conditions.

- Reactor Vessel-Wide Range (ΔP_c) - provides an indication of pressure drop for the reactor core and internals, regardless of the combination of the 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 instrument will monitor core conditions on a continuing basis.

The set of three d/p cells is used to provide indication to the operator, with or without reactor coolant pump operation. When the reactor coolant pumps (RCPs) are operating, both P_a and P_b will be offscale high, and an indication of the pressure drop across the reactor core and internals will be provided by P_c. This pressure drop can be compared with the normal, single-phase pressure drop to approximate the void content or density of the circulating coolant. With the RCPs off, both P_a and P_b provide an indication of the reactor vessel water level during natural circulation; P_b measures from the bottom to the top of the reactor vessel while P_a provides a more accurate measurement of the level above the middle of the hot leg. This system has been approved for installation by the NRC as a method of tracking coolant inventory when used in conjunction with core exit thermocouple systems and subcooling margin monitors and operated within approved Emergency Operating Procedure Guidelines.

A schematic of the system layout is shown in Figure 3. There are four reactor coolant system penetrations for the d/p cell reference lines; one reactor heat connection at a spare penetration near the center of the head (cannot use the same penetration as the head vent system), one connection to an incore instrument conduit at or near the seal table, and connections into the side of two reactor coolant system hot leg pipes or RTD bypass manifold pipes. Appropriate sensors and hydraulic isolators provide containment isolation in case of a break in any of the sensing lines to prevent the flow of primary coolant system water outside of containment.

Combustion Engineering Heated Junction Thermocouple System (HJTCS)

The Combustion Engineering Heated Junction Thermocouple System (HJTCS) consists of two independent safety channels as shown in Figure 4. Each channel consists of one probe assembly, signal

processing equipment, and an operator interface. Probe assemblies are located in the upper core support structure of the reactor vessel. Signal processing equipment and operator interfaces are located outside containment.

The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube that spans from the top of the core to the reactor vessel head. The basic principle of system operation is the detection of a temperature difference between adjacent heated and unheated thermocouples. As pictured in Figures 5 and 6, the HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel-Alumel thermocouple positioned away from the heater (or unheated junction). In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermocouples is very small. In a fluid with relatively poor heat transfer properties, the temperature differences between the thermocouples is large. Therefore when the HJTC sensor is covered with water, the differential temperature is small and when the HJTC sensor is uncovered, the differential temperature is large. The HJTC probe assembly consists of eight HJTC sensors as shown in Figure 7 and provides an indication of the collapsed liquid level above the upper core alignment plate. The voltage output for each HJTC is processed by a microprocessor-based system performing the following signal processing, surveillance, and heater power control functions:

° Provides indication of the collapsed liquid level above the upper core alignment plate.

° Provides a level output signal for trend recording to the operators.

° Provides temperature indications of coolant in the upper plenum.

° Provides test features for performing HJTC sensor operability and diagnostics.

° Provides on-line surveillance of HJTC sensor to access operabiity.

° Provides control of heater voltage to minimize HJTC internal heating after uncovery.

The CE HJTC system has been installed on other Westinghouse designed plants and can be made compatible with the Emergency Procedure Guidelines (EPGs) developed by the Westinghouse Owners Group by proper selection of the sensor locations in the HJTC probe asssembly. The NRC has completed its review of the Combusion Engineering HJTC system and has found the system acceptable for tracking reactor coolant system inventory and providing an enhanced ICC instrumentation package when used in conjunction with core exit thermocouples and subcooling margin monitors and operated within approved EPGs. The present EPGs would have to be revised to provide guidance to the operators on the meaning and use of the reactor coolant inventory measurement.

III. Scope of Supply

The installation of either the Westinghouse or Combustion Engineering reactor vessel level systems would require a considerable amount of input from the vendor, Bechtel, Southern Company Services, and Alabama Power Company. The basic scope of supply and services offered by Westinghouse are contained in Table 1. In addition, Westinghouse has several options available that can be requested by Alabama Power Company. These options include:

- Installation Engineering - Westinghouse can provide full scope installation engineering services for the RVLIS.
- Installation Services - Westinghouse can provide a wide variety of installation services for the RVLIS.
- Vacuum Fill and Startup - Westinghouse can provide a full-scope vacuum fill, calibration, and startup of the RVLIS.

Additional details of the scope of supply and options offered by Westinghouse are provided in the detailed study.

The basic scope of supply and services offered by Combustion Engineering are contained in Table 2. In addition, Combustion has several options available that can be requested by Alabama Power Company. These options include:

° Dummy Control Rod Drive Mechanism (CRDM) Cooling Shroud

° Thermal Sleeve

° Reactor Core Coolant Guide Tube (RCCGT) Removal and Installation

° Modification of Existing Cooling Shroud

° RCCGT Shroud Hardware

° Cover and Orifice Plate

° Removal of Part Length Control Element Extension Shaft

° LOCA and Seismic Analysis

Additional details of the scope of supply and options offered by Combustion Engineering are provided in the detailed study. In addition to the equipment and services offered by Combustion Engineering, Westinghouse will be required to supply the guide path shrouds needed to house the two HJTC probe assemblies.

It is expected that Bechtel would provide the design engineering required to install either the Westinghouse or Combustion Engineering systems. Engineering Study, ES-038, completed by Bechtel provides design and material scoping information for the installation of the two systems. A copy of this information is provided in the detailed study.

TABLE 1

Westinghouse System Scope of Supply

Equipment Supplied by Westinghouse

Item	Quantity	Description
1	4	3/4 T 78 root valves
2	12	Instrument valves
3	6	Hydraulic isolator
4	6	Special high volume sensors
5	6 lengths (total of 1500 ft max.)	Capillary tubing
6	2	RCS hot leg bosses/or RTD by-pass manifold tee's
7	2	Remote monitor
8	6	Seismically qualified P transmitter
9	14	Impulse line RTDs
10	2	Single bay cabinet including: Microprocessor chassis Termination panel Power supplies Output for serial data link (RS232)
11	1	Three-pen level recorder
12	2	Wide range pressure transmitters
13	2	Loop power supplies (for wide range pressure transmitters) - to be located in process protection sets
14	6	Transmitter access assemblies

Services Supplied by Westinghouse

1. Criteria defined and detailed layout of vessel head piping for level system.

2. Criteria defined and detailed layout of system piping for connections to hot legs or to RTD bypass manifold

3. Criteria defined and detailed layout of seal table conduit connection adaptors.

4. Define criteria for piping installation design.

5. Piping stress and structural analysis for Class 2 components as defined.

6. Schematic drawings for installation of level instrumentation system for piping in Westinghouse scope.

7. Instrument calibration, wiring and operating instruments (system manual).

8. Define criteria for as-built survey of impulse line piping.

9. Instructions for process electronics, sealing calibration and operation.

10. Revision of engineering flow diagrams and appropriate block diagrams.

11. Electrical termination diagrams.

12. Cabinet and display installation drawings.

13. Instruction manual.

14. Defined criteria for hydraulic isolator indication.

15. Design and hardware for obtaining isolated reactor coolant temperature T_{hot} signal and wide range RCS pressure signals for input to the microprocessor.

16. Design of pressure sensing lines from reactor head, seal table and hot leg piping/RTD bypass manifold piping to first isolation valves.

17. Installation of incore detector conduit adapter and head adaptor and R.C. pipe bosses or RTD bypass manifold tee.

Hardware and Service Supplied by Alabama Power Company

1. Reactor coolant pump signal (6).
2. Containment penetrations (2 mechnical, 2 electrical).
3. Calibration fixture (2).
4. Detailed as-built survey of impulse line piping.
5. Installation and procurement of capillary tubing and supports, instrumentation and power cable, conduit, cable trays, breakers, relays, and terminal blocks, hydraulic isolators, contact RTDs and cable terminations.
6. Installation of instrument cabinets, isolation panels and control room panel displays and alarms.
7. Detailed design for the installation of the complete reactor vessel level system including electrical schematics, cable raceway design and drawings, cable routing and cable pulling and connection drawing and piping installation design.
8. Erection/removal of scaffolding and cleanup.
9. Revision of operating procedures.
10. Supervision of vendor installation.
11. Operator training.

TABLE 2

Combustion Engineering System Scope of Supply

Equipment Supplied by Combustion Engineering

Item	Quantity	Description
1	2	HJTC probe assemblies (8 sensors/probe assembly)
2	2	Signal processing units
3	4	Proportional Heater controllers
4	2	Main control board indicator
5	2	Cabinets (30" x 50" x 90")
6	2	HJTC Probe Assembly Holder
7	2	HJTC Probe Pressure Boundary Assembly
8	2	HJTC Handling Canister/Bullet Nose
9	5	Technical Manuals HJTC Processor (per unit)
10	2120 ft.	Mineral Insulated, Metal Sheathed Cable and connectors
11	2	Containment Electrical Penetrations
12	-	HJTC Signal Processors with power supplies, rack and level recorders
13	-	Modems for interfacing Signal Processor Racks to Plasma display
14	130 ft.	Fiber Optic Cable
15	-	Plasma Display Panels complete with Page Control Modules, power supplies and cable, and audible annunciator
16	-	RCC guide tube for guide path to house the Probe Holder

Services Supplied by Combustion Engineering

1. Design and structural analysis of the pressure boundary, taking into consideration loading caused by normal plant operation.

2. Thermal/hydraulic analyses to develop component details.

3. Development of design drawings for use during component procurement and subsequent installation activities.

4. Site technical supervision and installation guidelines for HJTC handling canister.

5. Removal and installation of pressure boundary modifications.

6. Removal of certain reactor internal hardware and installation of HJTC Probe Assembly.

7. Design verification analysis of Probe Holder Assembly.

Hardware and Services Supplied by Alabama Power Company

1. Plant Specific Licensing support

2. Operational guidelines for utilization of HJTCS.

3. Removal of reactor internals hardware.

4. Installation of electrial hardware including containment penetrations, signal processor, modem, plasma display, power supplies, and MI cable.

5. ASME stress report for pressure boundary modifications addenda.

6. Detailed design for complete installation of reactor vessel level system including electrical schematics, cable raceway design and drawings, cable routings, pulling and connection information.

7. Procurement and installation of instrument racks, fuses, terminal blocks, instrument and power cable, conduit cable trays and cable terminations.

8. Erection/Removal of scaffolding and cleanup.

9. Supervision of vendor installation.

IV. COST ESTIMATES

In order to estimate the cost of installation of a reactor vessel water system in Farley Nuclear Plant - Units 1 and 2, cost information related to hardware, engineering and installation was obtained from Westinghouse, Combustion Engineering, Bechtel Power Corporation, Alabama Power Company and other utilities. The total cost for installation of a system on both units ranges from 4.5 to 5.0 million dollars. These costs are only estimates since actual bid information was not requested from either Westinghouse or Combustion Engineering. These cost estimates do not contain allowances for replacement power cost if the installation of a reactor vessel level system would delay the return to power from a refueling outage.

V. LICENSING CONSIDERATIONS

The acceptability of both the Westinghouse and Combustion Engineering reactor vessel level systems have been the subject of continued review by the NRC staff. The NRC has approved both the Westinghouse and Combustion Engineering systems for installation and training on a generic basis. The generic approval of the two systems was based upon an evaluation of the systems by the Oak Ridge National Laboratory using documentation supplied by Westinghouse and Combustion Engineering. The results of the Oak Ridge evaluations are contained in NUREG/CR-2628 for the Westinghouse Differential Pressure System and NUREG/CR-2627 for the Combustion Engineering Heated Junction Thermocouple System.

The evaluations performed by Oak Ridge show that the two systems are acceptable on a generic basis but the final approval would require plant specific review. Among the plant specific items which would 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 tranducers outside containment (for W system).

4. Inclusion of hydraulic isolators and sensors in the impulse lines (for W system).

5. Location of the HJTC sensors in the reactor vessel head (for CE system).

In addition, the items listed in NUREG-0737, II.F.2. would be reviewed on a plant specific basis. Deviations from the generic system descriptions contained in NUREG/CR-2627 and NUREG/CR-2628 would have to be justified. The open items from the generic evaluations contained in NUREG/CR-2627 and NUREG/CR-2628 would also have to be reviewed. The major open item from both evaluations is the integration of the ICC system into the plant specific Emergency Operating Procedures (EOPs). These open items would have to be closed during the review of the system by the NRC prior to placing the system into operation.

VI. CONCLUSIONS

This report has been prepared to address the feasibility of installing a reactor vessel water level system at Joseph M. Farley Nuclear Plant - Units 1 and 2. The systems considered for installation at Farley Nuclear Plant were the Westinghouse Differential Pressure Measuring System and the Combustion Engineering Heated Junction Thermocouple System. Both of these systems have been accepted by the NRC Staff for installation and training on a generic basis. To date, neither the Westinghouse nor Combustion Engineering Systems have received final NRC approval for use by the plant operators during normal plant operations or accident conditions. Based upon the information obtained during the feasibility study, we believe that both the Westinghouse and Combustion Engieering Systems would be acceptable for installation at the Farley Nuclear Plant following NRC approval for operational use.

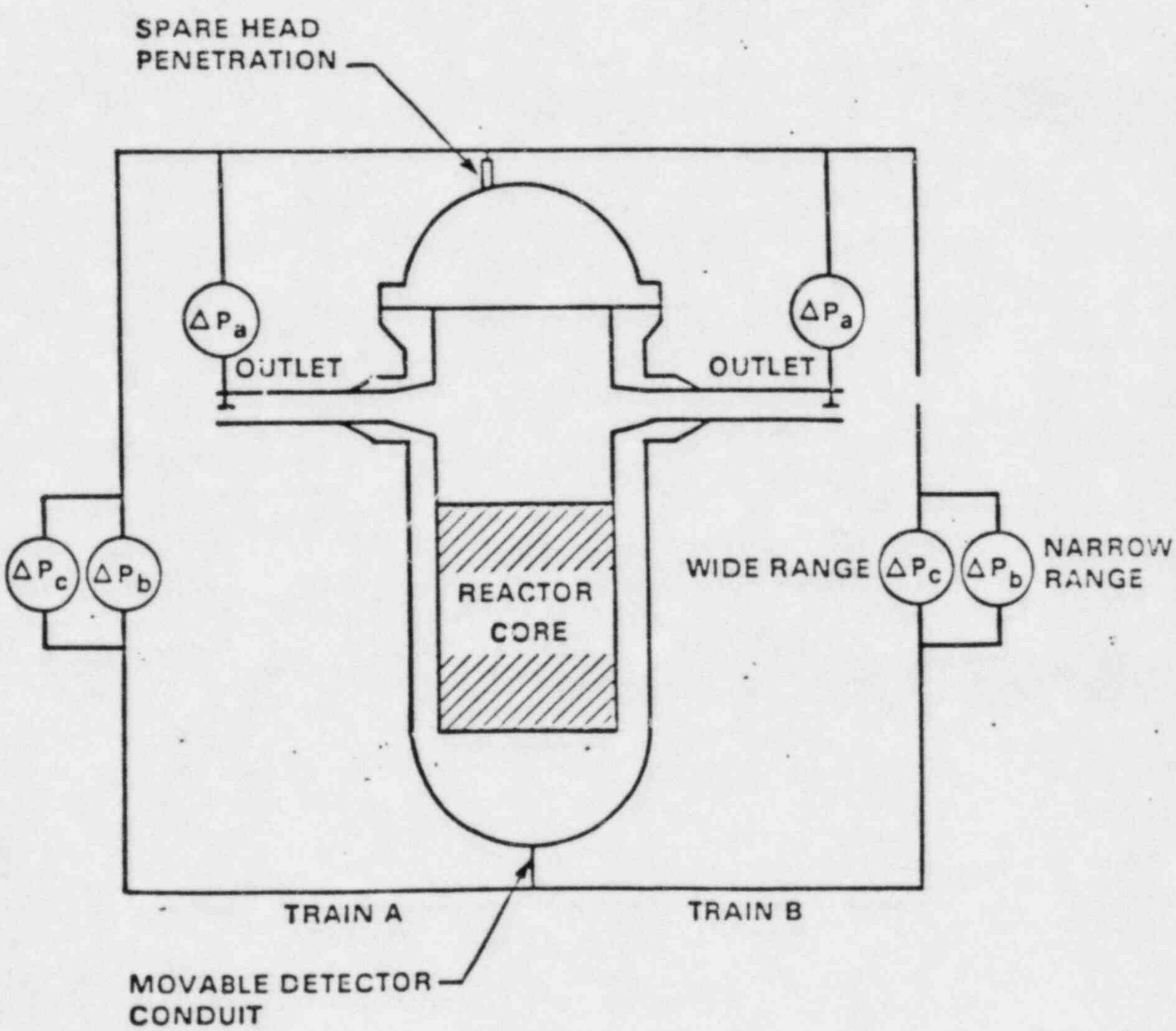


FIGURE 1
Westinghouse Reactor Vessel Water Level Instrumentation System

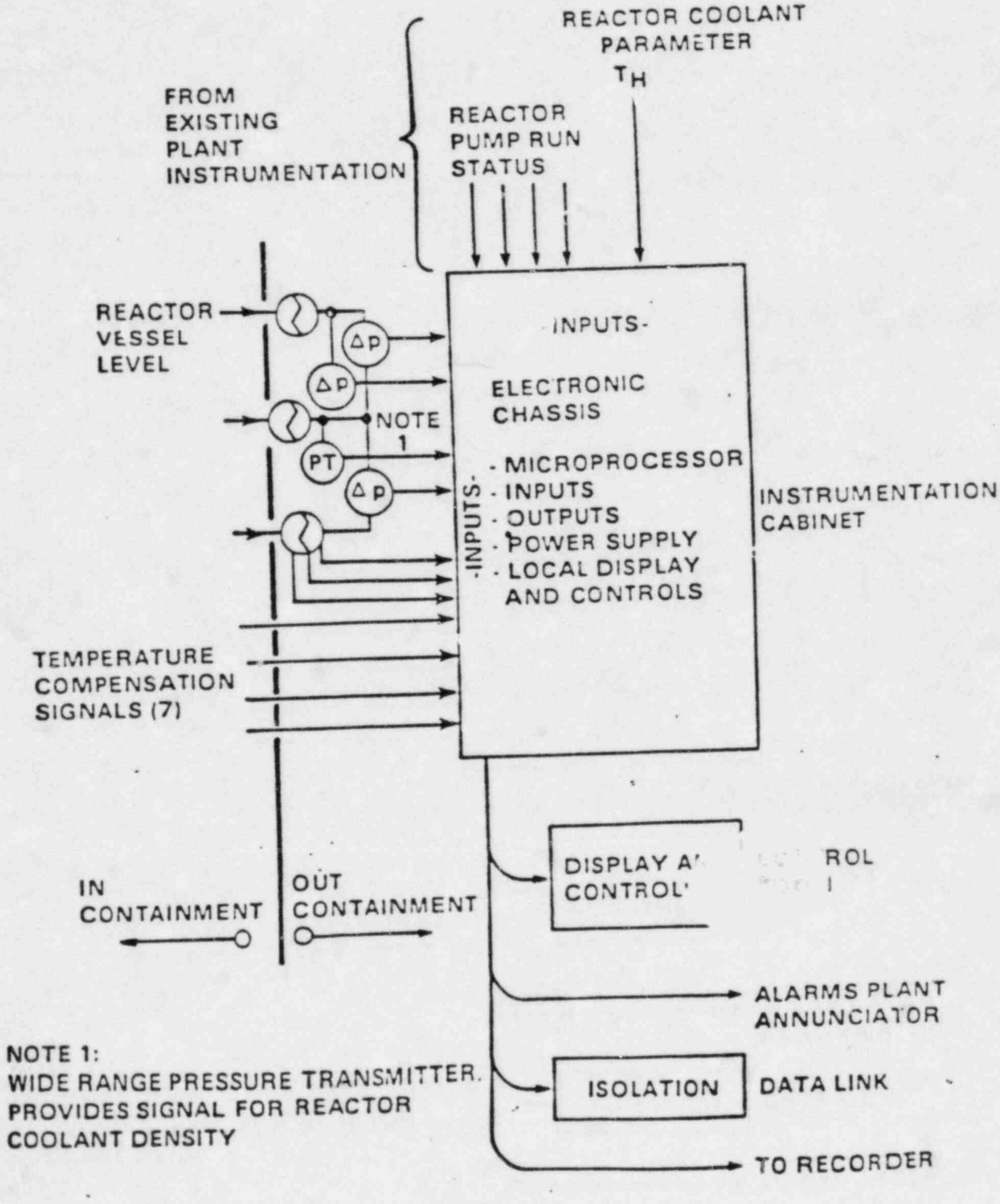


FIGURE 2

Westinghouse Reactor Vessel Level Instrumentation System Block Diagram
(One Channel of Two Redundant Channels Shown)

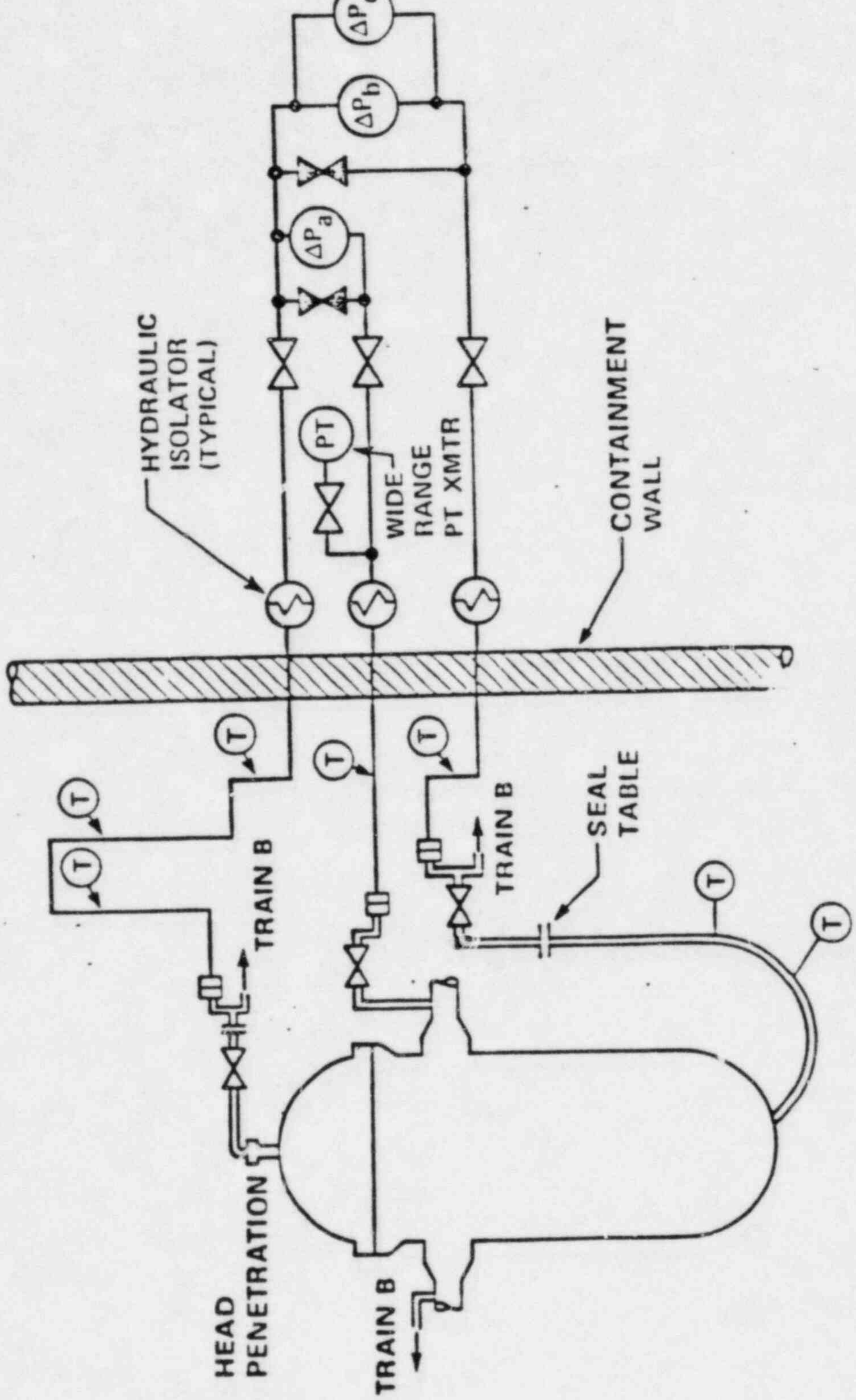


FIGURE 3

Westinghouse Process Connection Schematic, Train A

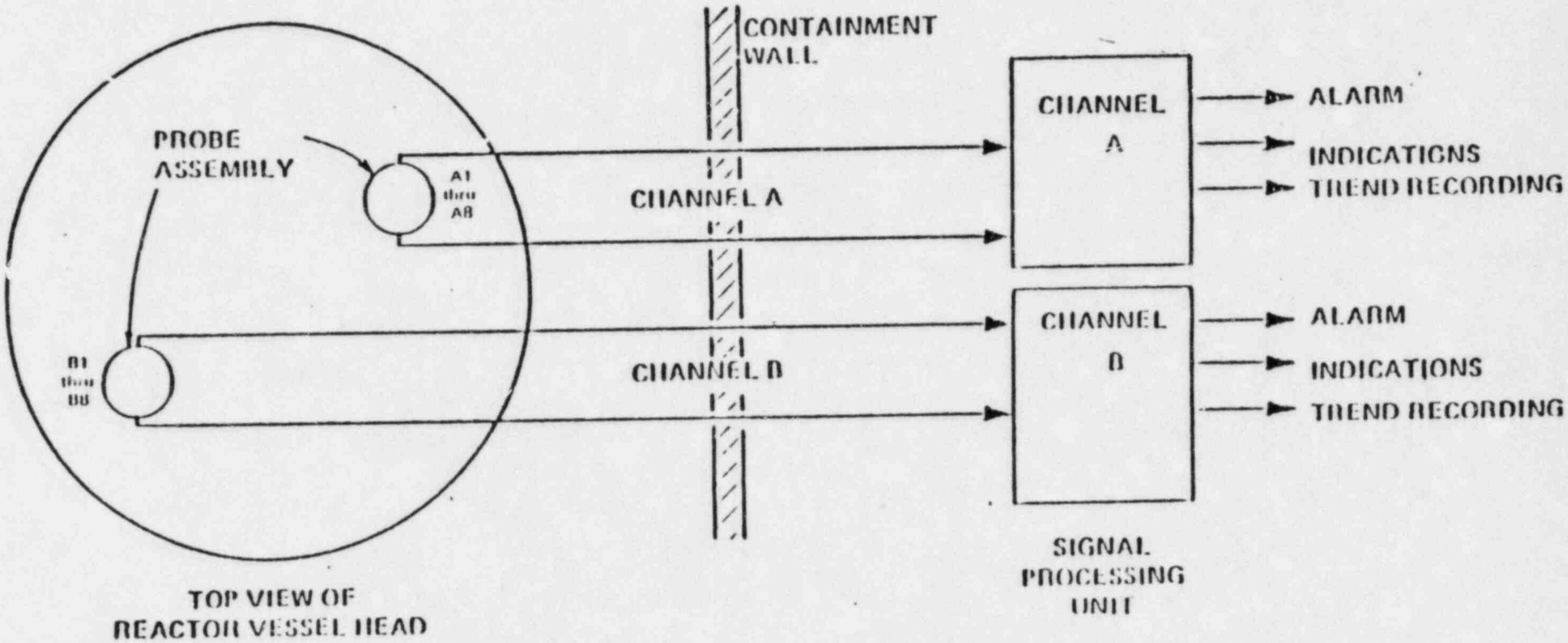


FIGURE 4

Combustion Engineering HJTCS Functional Configuration

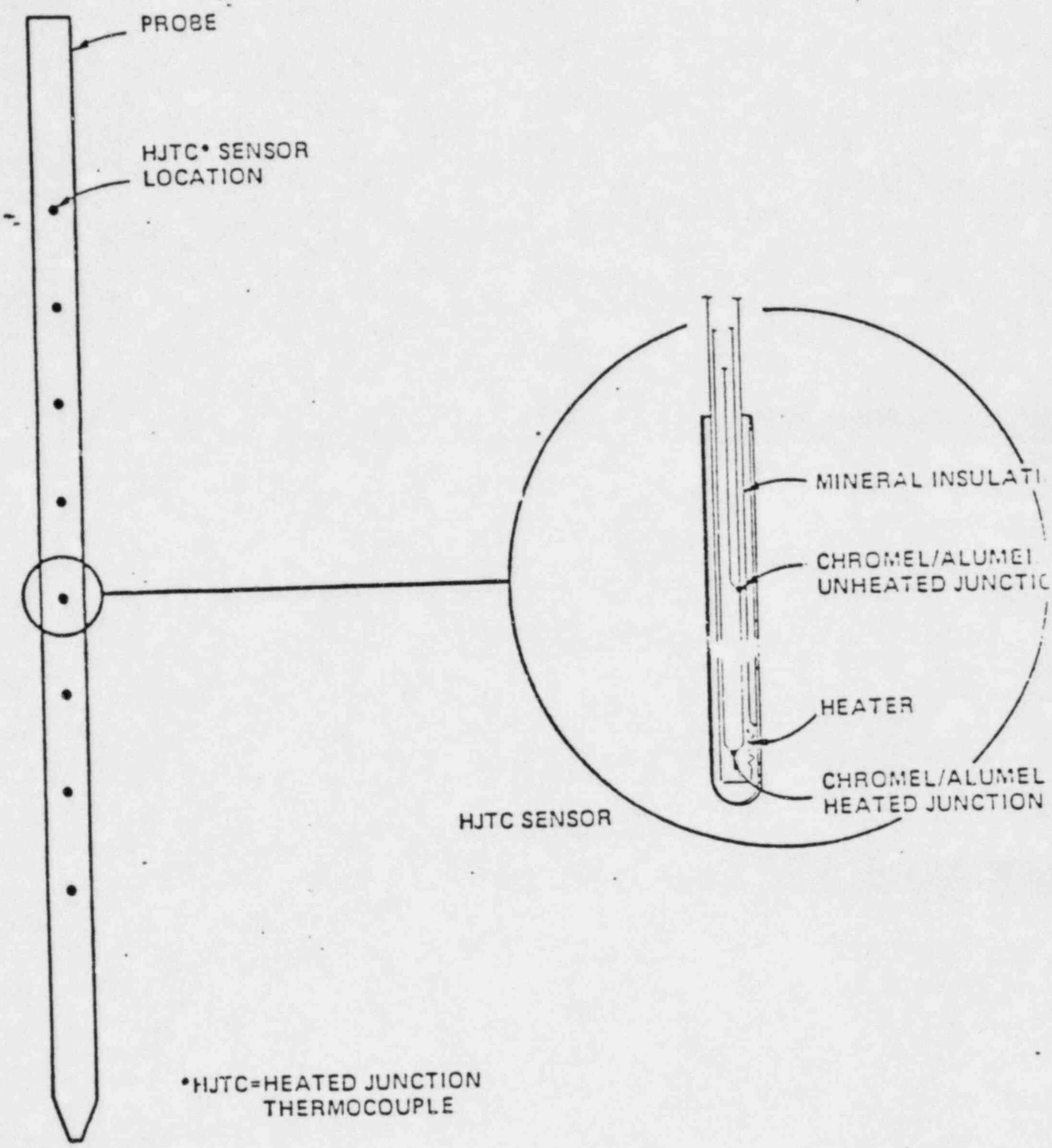


FIGURE 5

Combustion Engineering Probe/Sensor Configuration

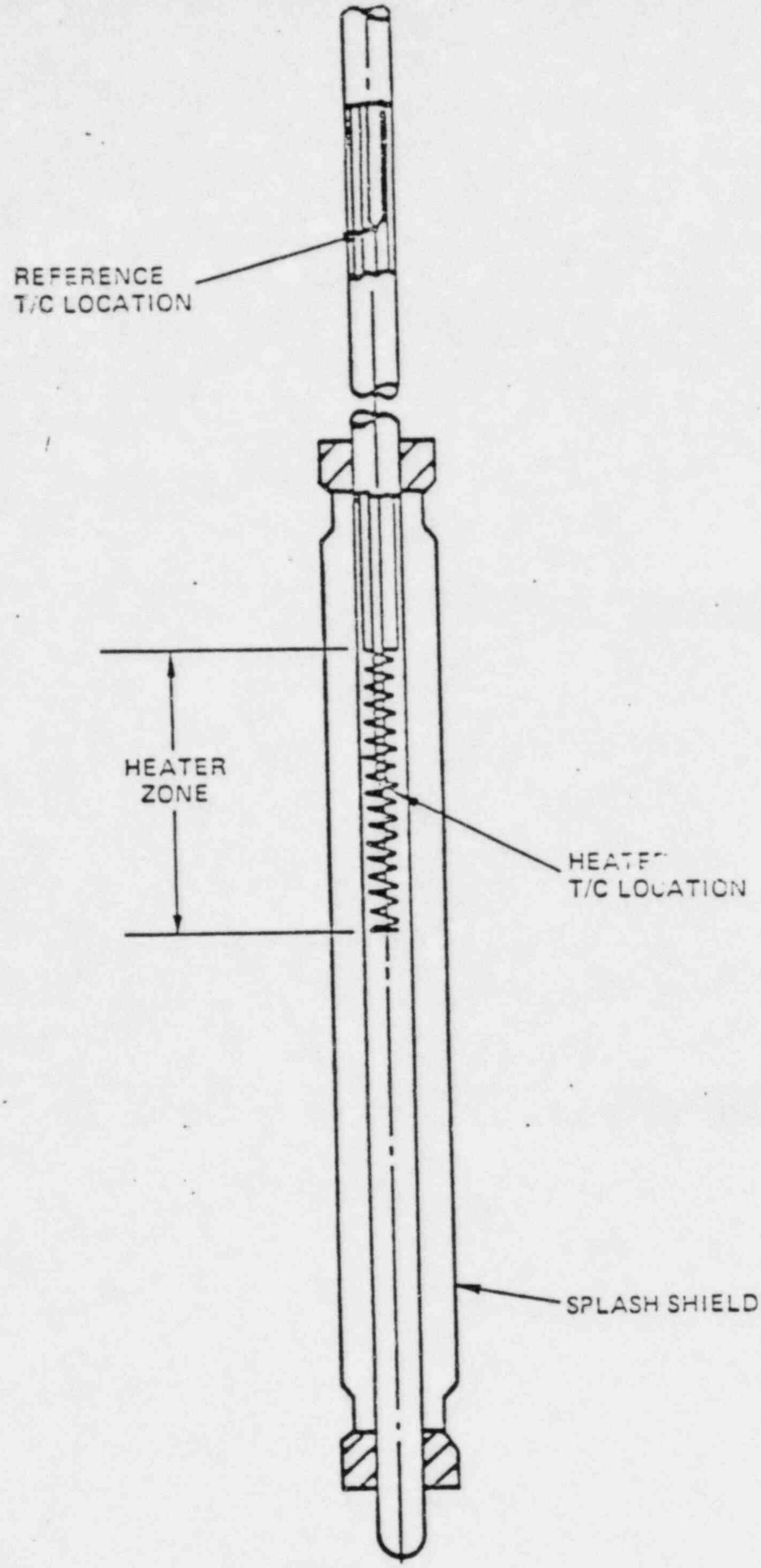


FIGURE 6
Combustion Engineering HJTC Sensor - HJTC/Splash Shield

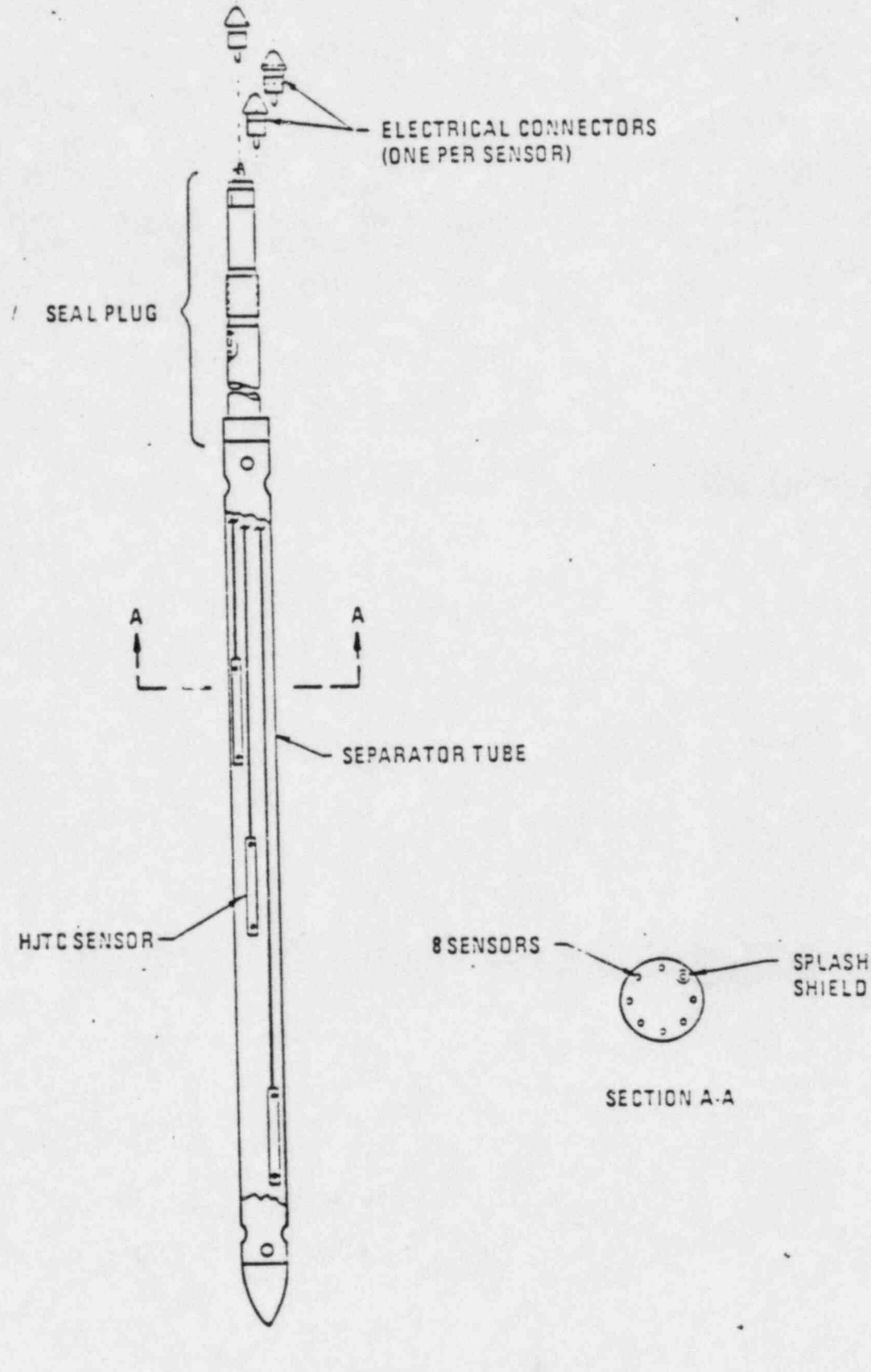


FIGURE 7

Combustion Engineering Heated Junction Thermocouple Probe Assembly

ATTACHMENT 3

DISCUSSION OF PROBLEMS WITH WESTINGHOUSE AND COMBUSTION ENGINEERING REACTOR VESSEL LEVEL SYSTEMS

Westinghouse System Problems

1. Design

a. The microprocessor can accomodate only eight RTDs and the plant specific piping design may require more RTDs. RTDs are attached to the capillary tubing that connects the transmitter to the reactor vessel and detect temperature changes in verticle portions of capillary tubing. Inaccurate temperature measurement of capillary tubing will result in greater inaccuracy of reactor vessel level indication.

b. Magnetic valves do not provide positive valve position indication. Such lack of positive indication could allow inadvertent transmitter isolation and loss of vessel level indication for the operator.

c. Seismic qualification of existing piping may require reanalysis due to connecting new capillary tubing to existing piping. Utility experience has shown that differences between design and as-built conditions during installation could result in major modifications to existing piping and result in an additional iteration of analysis, design and installation. These major modifications, which alter outage durations and manpower requirements, can not be identified prior to the outage and could impact scheduled return to power.

d. LOCA effects have not been considered with respect to system accuracy. Since the primary purpose of such a system is to provide vessel level indication to control room personnel, measurement uncertainty could mislead the operator during post-accident conditions. In addition, level inaccuracy could confuse the operator when compared to other plant parameters.

e. The small clearance (1/32 inch) for the bottom sensing line (incore thimble) could become a crud trap and a point for boron buildup that may affect system operability. Clogging at the bottom sensing line would be gradual and could possibly be undetectable during power operation. In this event, the level system would not be able to perform its intended function during plant transients.

2. Operations

a. Analog and CRT displays require a procedure for the operator to understand the several ranges of displayed data. A system that requires procedure guidance to understand the display may be an impediment to the operator. During stressful situations data should be presented to the operator in such a manner that plant status can be quickly and accurately ascertained.

b. Since the operators are not allowed by the NRC, according to Generic Letter No. 82-33, to utilize this instrumentation until operationally accepted by NRC, there is less confidence in the system. Greater operator acceptance could be obtained if the system were operationally proven as required by technical specifications and system design, and operational characteristics could be incorporated into plant procedures and training immediately upon installation.

c. The system is not debugged and is complicated due to capillary tubing design. Additional calibration data of installed systems may be required to determine calibration criteria. These criteria have yet to be finalized for all installed systems. Significant manpower will be required to maintain the system operable. Containment entry into possible high radiation areas will be required for calibration during each refueling outage. Transmitter location outside containment results in long lengths of capillary tubing which must be filled and vented utilizing several vacuum pumps. System filling and venting can be a tedious evolution especially when technicians are working in radiation protective clothing. In addition, the plant ALARA program must be considered when planning calibration durations and manpower requirements.

d. Reconnection of capillary tubing at the seal table and systems calibration is one of the last activities prior to plant startup. Any problems incurred during system reconnection and calibration could delay plant startup.

3. Installation

a. Deviation from the designed location of bellows and slope of capillary tubing may occur during installation. Inability to comply with design criteria could affect system operational reliability.

b. The length of capillary tubing required to install a transmitter outside containment increases the installation schedule and dose received by the installation personnel.

c. Cabinet location for operating plants could be a problem. Plant modifications may be required to locate and maintain these cabinets.

d. Existing main control rooms will require modification to incorporate display equipment. Installation of reactor vessel level displays into the main control room should be coordinated with efforts associated with Emergency Response Capabilities identified in Generic Letter 82-33.

COMBUSTION ENGINEERING SYSTEM PROBLEMS

1. Design

a. The total generic system has not been completed. A picture of the display is not available for utilities to incorporate into the operator training and plant procedures.

b. Mineral Insulated (MI) cable is fabricated into design lengths at the factory. Any field deviations will require the purchase of new fabricated cable (steel tubing). Such repurchase of MI cable after initiation of installation could lengthen the outage and delay return to power.

c. Even though MI cable is seismically qualified, cable tray support may be required to prevent damage to this cable from either people standing on it or placing heavy objects upon it.

d. Design support is required for installation of the CE probe. A platform will be placed on the upper internals for personnel to perform required reactor internal modifications.

e. MI cable and containment penetration interface design must be resolved. The Architect Engineer must obtain drawings from CE to determine how to design the electrical connection at the containment penetration. A new containment penetration may be required if spares are not available or if the existing penetrations design is not compatible.

f. Electrical cabinets are large and location could be a problem. Also the electrical cabinets may require new support systems such as humidity and air conditioning control.

2. Operations

a. Storage of MI cable and new cable trays installed near the reactor head must be considered during subsequent refueling outages. Prior to removing the reactor head to support refueling, personnel must determine where to store MI cable and new cable trays so that they will not interfere with outage activities.

b. Shipment of contaminated material offsite should be included in the installation planning. The orifice plates and a spare CRDM cap must be shipped offsite and buried.

c. Operators may not trust the system since there are no criteria to demonstrate operability. Other equipment required to be operable by technical specifications have specified criteria such that operability can be determined.

3. Installation

a. Field personnel are not familiar with installation of MI cable. This unfamiliarity will require additional time for implementation and any cable rendered unuseable durin[illegible]llation or improperly manufactured must be reor[illegible]om the factory. MI cable is a long lead time procurement item and each length must be specifically manufactured (i.e., spare cable lengths can not be stockpiled).

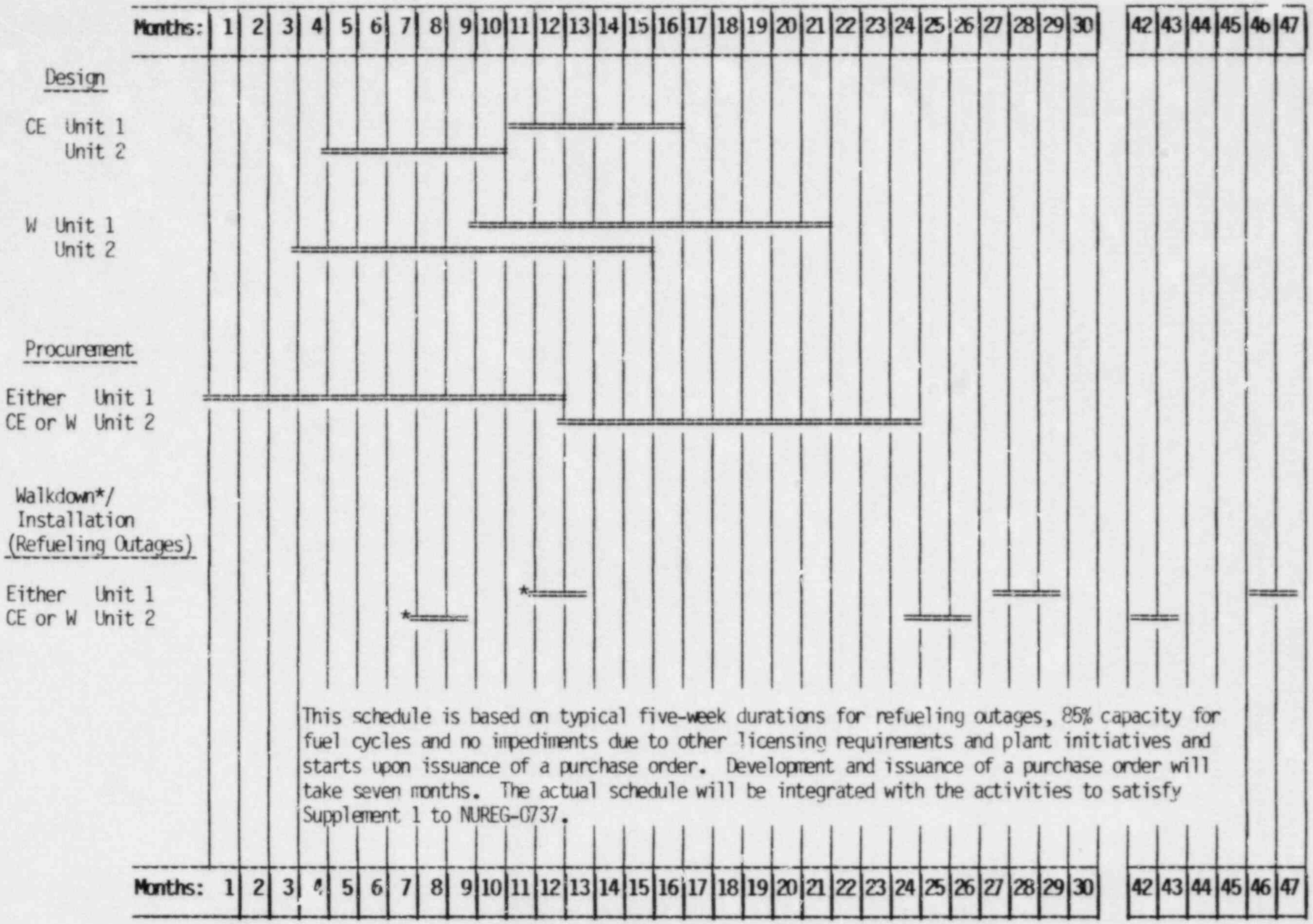
b. Temporary shielding to support installation could require significant manpower. To reduce exposure of personnel making the pressure boundary modifications, temporary shielding may be required on the reactor head.

c. Decontamination of orifice plates and spare penetration caps must be considered. By utilizing spare CRDM penetrations for the CE probe, the orifice plates, which control the flowrate into the upper internals and head area, and the pressure boundary cap must be removed and decontaminated.

d. Reworking of vendor supplied material may be required. For example, the bullet nose that guides the reactor head over the CE probe may require field modification.

e. Special tools which are not normally maintained by utilities are required to remove the orifice plates. If these can not be obtained from another utility or the NSSS vendor, special purchase orders would be placed with the vendor which could add to the cost of modification and manufacturer lead time.

Implementation Schedule of Vessel Level System
Attachment 4



This schedule is based on typical five-week durations for refueling outages, 85% capacity for fuel cycles and no impediments due to other licensing requirements and plant initiatives and starts upon issuance of a purchase order. Development and issuance of a purchase order will take seven months. The actual schedule will be integrated with the activities to satisfy Supplement 1 to NUREG-0737.