**Westinghouse Electric Corporation**

**Water Reactor Divisions**

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

July 21, 1981
CAW-81-43

Docket 50-247
DPR-26

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

SUBJECT: "Summary Report - Westinghouse Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling (7300 System)," for Consolidated Edison Company of New York, Inc. Indian Point Unit 2, July 1981 (Proprietary)

REF: Consolidated Edison Company Letter Application for Withholding Proprietary Information from Public Disclosure, O'Toole to Eisenhut, August 31, 1981

Dear Mr. Eisenhut:

The proprietary material for which withholding is being requested by Consolidated Edison Company is of the same technical type as that proprietary material submitted by Westinghouse previously in Application for Withholding AW-77-18, and is accompanied by an affidavit signed by the owner. Further, the Application for Withholding AW-77-18 was approved by the Commission on October 28, 1977 and is equally applicable to this material.

Westinghouse Electric Corporation previously transmitted similar information for utility use that was accompanied by affidavits CAW-80-75 and AW-77-18. This plant specific submittal is accompanied by CAW-81-43 and AW-77-18.

Accordingly, withholding the subject information from public disclosure is requested in accordance with the previously submitted affdavit and Application for Withholding AW-77-18 dated April 6, 1977, a copy of which is attached.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference CAW-81-43 and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek
Attachment

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

50-247

NONPROPRIETARY & PROPRIETARY VERSIONS
OF
SUMMARY REPORT

Received wth ltr dtd 9/04/81

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AW-77-18

# AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 20 day
of April 1977.

Notary Public

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rule-making proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition in those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is attached to Westinghouse Letter Number NS-CE-1403, Eicheldinger to Stolz, dated April 6, 1977. The letter and attachment are being submitted in support of the Westinghouse emergency core cooling system evaluation model.

Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of Westinghouse, taking into account the value of the information to Westinghouse, the amount of effort and money expended by Westinghouse in developing the information, and considering the ways in which the information could be acquired or duplicated by others.

Further the deponent sayeth not.

ATTACHMENT A

Response to 7/10/81 Confirmatory Order
and Revised Design Details on Item II.B.1
(NUREG-0737)

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September, 1981

II.B.1 REACTOR COOLANT SYSTEM VENTS

The following information revises and supplements that information provided in Consolidated Edison's December 31, 1979 and February 15, 1980 submittals to the Staff. The revised Indian Point Unit No. 2 Remote Reactor Head Vent (RRHV) System is shown schematically in the Figure SI-1, Rev 1. The system uses two motor operated valves (in series) which are operated from the control room. Most of the lines in the system are 3/4 inch in diameter, stainless steel tubing with 0.095 inch thick wall. Two joints in the tubing run are made up with compression fittings ("Grayloc" or equal) to permit removal of the vessel head. All remaining joints in the non-isolable portion of the tubing run are welded. An acoustic flow detector, which has the capability of distinguishing between gaseous and liquid flow is used for flow indication and valve position indication. The readout for the detector is located in the control room.

An extended "spool piece" runs from the connection at the reactor head to a point at the steam generator shield wall. This "spool piece" and its major supporting structure is removed as an assembly prior to reactor vessel head removal. When in use the supporting structure is bolted to existing steel and the elevation 95' - 0" floor. The RRHV line continues from the "spool piece" at the steam generator shield wall up to the top of the wall where a tee connection permits flow to be directed to the normal vent via the Pressurizer Relief Tank (PRT) or to the emergency vent to containment atmosphere. Except for startup and shutdown, the normal vent is disconnected and capped; the emergency vent is always connected during reactor operation.

# REMOTE REACTOR HEAD VENT SCHEMATIC

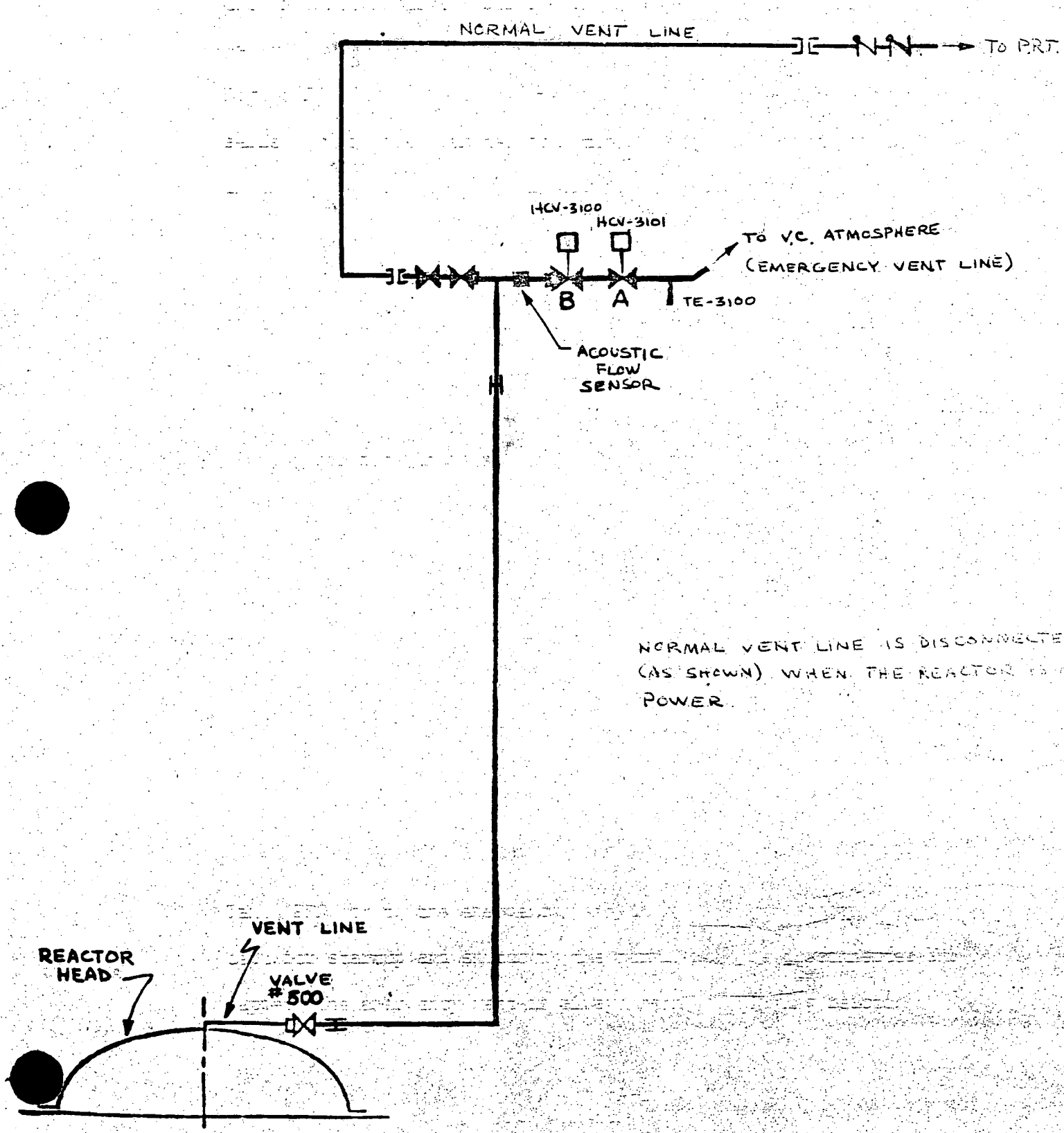


FIGURE S1-1 REV-1

The discharge point of the emergency reactor head vent to the containment atmosphere is located approximately seventy-five (75) feet from the nearest hydrogen recombiner precluding the possibility of slugs of high hydrogen concentration from reaching them. If the PRT is used as a vent path, the discharge would vent into the containment atmosphere at approximately El - 64'. Containment Fan Cooler Unit (FCU) No. 25 is located on El - 68' above the PRT and is capable of circulating approximately 70,000 cfm during normal operations and approximately 65,000 cfm under accident conditions. The FCU No. 25 intake accesses the PRT area below via the floor grating and edge clearance between the El - 68' floor and the containment liner. This design feature prevents the buildup of trapped pockets of non-condensible gases. Since the hydrogen recombiners are aligned with and supplied from containment fan coolers other than FCU 25, the non-condensibles will be thoroughly mixed with containment atmosphere prior to entering the recombiners.

II.F.2 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

A non-proprietary version of the Westinghouse summary report on "Reactor Vessel Level Instrumentation System for Monitoring Inadequate Core Cooling" for Indian Point Unit No. 2 is presented in Enclosure 1. The proprietary version of this report is presented in Enclosure 3. A summary of key operator action instructions in the current emergency procedures for ICC and a description of how these procedures will be modified when the final monitoring system is implemented is presented in Enclosure 2. The completion of the reactor vessel level indication system will be accomplished by the required implementation date of January 1, 1982.

The above information and the information on ICC instrumentation contained in our previous submittals dated October 17, 1979, November 20, 1979, December 31, 1979, February 15, 1980, and February 26, 1981 provide the documentation required by this NUREG-0737 Task Item.

III.D.3.4 CONTROL ROOM HABITABILITY

A study of the effects of an accidental release of radioactive gases and toxic chemicals on the Indian Point Unit No. 2 control room was performed and documented in Consolidated Edison's May 12, 1981 submittal. As a result of that study, additional dampers are planned for installation in the control room charcoal filtration system to insure redundancy and chemical monitors to detect anhydrous ammonia, chlorine and hydrogen cyanide are planned for installation in the control room air intake. The chemical monitors will be alarmed in the control room and automatically isolate the control room outside air intake at predetermined setpoints.

These modifications will be completed, as required, by January 1, 1983.

Enclosure 1

Westinghouse Summary Report on
"Reactor Vessel Level Instrumentation System for Monitoring
Inadequate Core Cooling" (Non-Proprietary)
(NUREG-0737, Item II.F.2)

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September, 1981

SUMMARY REPORT

# WESTINGHOUSE REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM FOR MONITORING INADEQUATE CORE COOLING

(7300 SYSTEM)

FOR CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT UNIT 2

July, 1981

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# 1.0 INTRODUCTION

## 1.1 NRC REQUIREMENTS

The NRC has established requirements (items I.C.1 and II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements") to provide the reactor operator with instrumentation, procedures, and training necessary to readily recognize and implement actions to correct or avoid conditions of inadequate core cooling (ICC).

Under certain plant accident conditions, the potential exists for the formation of voids in the reactor coolant system (RCS). Under these conditions, it would be advantageous for the reactor operator to monitor the water level in the reactor vessel or the approximate void content during forced circulation conditions in order to assist him in subsequent actions. Therefore, a reactor vessel level instrumentation system (RVLIS) has been incorporated to provide readings of vessel level which can be used by the operator. Vessel level as measured by the RVLIS is the collapsed liquid level in the vessel.

The RVLIS provides a relatively simple and straight-forward means to monitor the vessel level. This instrumentation system neither replaces any existing system nor couples with any safety system; however, it does act to provide additional information to the operator during accident conditions. The RVLIS utilizes differential pressure (d/p) measuring devices to indicate relative vessel level or relative void content of the circulating primary coolant system fluid.

## 1.2 DEFINITION OF ICC

ICC as defined in References 1 and 2, is a high temperature condition in the core such that operator action is required to cool the core before damage occurs.

## 1.3 CONDITIONS OR EVENTS WHICH DESCRIBE THE APPROACH TO ICC

The most obvious failure that would lead to ICC during a small-break loss of coolant accident (LOCA), although highly unrealistic since multiple failures are required, is the loss of all high pressure safety injection. The approach to ICC conditions and the analyses for this event sequence are provided in References 1 and 2.

# 2.0 FUNCTIONAL REQUIREMENTS

## 2.1 PARAMETERS CRITICAL TO ICC

The analysis provided in References 1 and 2 delineates those parameters critical for the detection of and the necessary mitigation actions for the recovery from an ICC condition.

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

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

## 2.2 INSTRUMENTATION ACCURACIES, RANGES, AND TIME RESPONSE

### Accuracy

An accuracy of 6 percent is required on both types of reactor vessel level instruments. This should be a statistical combination of all uncertainties including those due to environmental effects (if any) on instrumentation. For the narrow range instrument this corresponds to an allowable deviation of about $\pm$ 2.5 feet elevation head. This is required to: 1) provide adequate margin against inadvertant use of the ICC operating guideline ($E^2$OI-1, see Section 5.1), 2) assure that the vessel level reading can be reasonably used to aid in the detection of

the onset of ICC conditions, 3) derive useful information reguarding vessel level behavior during the vessel refill period of a LOCA transient.

Range

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

Time Response

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

## 2.3 QUALIFICATION REQUIREMENTS

Environmental qualification of the RVLIS shall verify that the system equipment will meet, on a continuing basis, the performance requirements determined to be necessary for achieving the system requirements as presented above. Verification must include confirmation that those portions of RVLIS equipment which are within the containment will operate during and subsequent to the conditions and events for which the system is required to be operational. Verification will include determination that the system is sufficiently accurate during this time to meet its design basis. The system post-accident environment qualified life requirement for electrical equipment inside containment is 120 days following certain postulated events. The electrical equipment that is installed outside of containment need not meet a qualified life for an extended period of time providing replacement or calibration checks can be made in short enough time commensurate with the reliability goals of the redundant system. For the resistance temperature detectors (RTDs) environmental requirements for service within the containment, refer to

Section 4.2.3. Electrical equipment inside containment shall be installed such that it is removed from areas where high energy pipe breaks or pipe whip could cause failure. The d/p transmitters and electronic processing equipment shall be located in a low ambient radiation area.

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

## 2.4 CODES AND STANDARDS

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

Regulations

| | |
|---|---|
| GDC 1 | Quality Standards and Records |
| GDC 2 | Design Bases for Protection Against Natural Phenomena |
| GDC 4 | Environmental and Missile Design Bases |
| GDC 13 | Instrumentation and Control |
| GDC 16 | Containment Design |
| GDC 18 | Inspection and Testing of Electric Power Systems |
| GDC 19 | Control Room |
| GDC 24 | Separation of Protection and Control Systems |
| GDC 30 | Quality of Reactor Coolant Pressure Boundary |
| GDC 31 | Fracture Prevention of Reactor Coolant Pressure Boundary |
| GDC 32 | Inspection of Reactor Coolant Pressure Boundary |
| GDC 50 | Containment Design Basis |
| GDC 55 | Reactor Coolant Pressure Boundary Penetrating Containment |

GDC 56 Primary Containment Isolation

10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

Industry Standards

IEEE-308-1971, "IEEE Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"

IEEE-323-1974, "General Guide for Qualifying Class 1 Electric Equipment for Nuclear Power Generating Stations."

IEEE-338-1971, "IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems"

IEEE-344-1975, "Guide for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"

IEEE-384-1977, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits"

ASME BPVC, Section III, Class 2 Nuclear Power Plant Components

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For certain mechanic clients, IEEE-323-1974 is applicable.

ANSI B31.1.0, 1967 including addenda through and including 6/30/71, "Code for Pressure Piping", including nuclear code cases where applicable

Regulatory Guides

R.G. 1.11 Instrument Lines Penetrating Primary Reactor Containment

R.G. 1.22 Periodic Testing of Protection System Actuation Functions

R.G. 1.75 Physical Independence of Electric Systems

## 3.0 ICC INSTRUMENTATION IDENTIFICATION

Adequate instrumentation is necessary to diagnose the approach to ICC and to determine the effectiveness of the mitigation actions taken. During the preparation of the ICC operating instructions, consideration was given to the adequacy of current instrumentation and the benefits derivable from the addition of new instrumentation. The following is a list of existing instrumentation considered (refer to the FSAR for details) and conclusions derived:

1. Current Instrumentation

   a. WIDE RANGE REACTOR COOLANT PRESSURE - present instrumentation is available for determining general RCS pressure trends during the ICC event. The expected accuracy following ICC events is such that this instrument cannot be used for precise determinations of the pressure required to assure onset of low head safety injection flow to the RCS.

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

   c. AUXILIARY FEEDWATER FLOW - present instrumentation is available for assuring the sufficiency of makeup water flow to the steam generators during an ICC event.

   d. WIDE RANGE RESISTANCE TEMPERATURE DETECTORS - present instrumentation is available in determining trends of recovery actions but may not be available in determining the onset of ICC conditions for all break sizes.

   e. CORE EXIT THERMOCOUPLES - present instrumentation is available in determining both the existence of ICC and the trends of recovery actions.

f. CORE SUBCOOLING - does not provide useable information during an ICC condition. Will indicate superheat conditions in core coolant. Will help indicate the approach to ICC by showing saturation conditions. Since the core subcooling monitors may not be described in the FSAR, refer to Table 3.1 for information.

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

h. STEAM GENERATOR LEVEL - present instrumentation is available for determining the availability of a heat sink for the RCS during an ICC condition.

2. New Instrumentation

a. REACTOR VESSEL LEVEL - provides an indication of the approach to ICC and confirms the achievement of adequate core cooling when level in the reactor vessel is restored.

To summarize the above considerations, current plant instrumentation is adequate to determine heat sink availability, to detect the onset of ICC, and to detect the effectiveness of mitigation actions following the onset of an ICC event. The RVLIS is provided to permit a more continuous indication of the approach to ICC.

TABLE 3.1

INFORMATION REQUIRED ON THE CORE SUBCOOLING MONITOR

| | |
|---|---|
| **Display** | |
| Information Displayed($T-T_{sat}$, $T_{sat}$, press, etc.) | All Pressure and Temperature Inputs; $P_{sat}$, $T_{sat}$, $P_{sat}$ Margin and $T_{sat}$ Margin |
| Display Type (analog, digital, CRT) | Digital |
| Continuous or on Demand | Continuous and on Demand |
| Single or Redundant Display | Redundant |
| Location of Display | Flight Panel, Rack |
| Alarms (include setpoints) | Caution - 50°F Margin $T_{sat}$ |
| Overall Uncertainty (°F, psi) | Digital - 5°F for RTD, ± 100 Psi |
| Range of Display | Hot Leg Temp. 0 - 700°F<br>RCS Pressure 0 - 3000 psi |
| Qualifications | Qualification Continuing |
| **Calculator** | |
| Type (process computer, dedicated digital or analog calc.) | Dedicated digital |
| If process computer is used, specify availability (percent of time) | N/A |
| Single or Redundant Calculators | Single |
| Selection Logic (highest T., chosen press) | Highest T for RTD ($T_{hot}$) Lowest Chosen Pressure |
| Qualifications | Qualification Continuing |
| Calculational Technique (steam tables, functional fit, ranges) | Steam Tables |

TABLE 3.1 (Continued)

| | |
|---|---|
| **Input** | |
| Temperature (RTDs) | RTD |
| Temperature (number of sensors and locations) | RTD - 4 hot and 3 cold |
| Range of Temperature Sensors | RTD - 0 - 700°F |
| Uncertainty of Temperature Sensors (°F at 1σ) | 5°F |
| Qualifications | 200 R Continuous<br>200°F |
| Pressure (specify instrument used) | Barton Qualified |
| Pressure (number of sensors and locations) | 2 wide range -<br>2 hot legs |
| Range of Pressure Sensors | Wide range - 0 - 3000 psi |
| Uncertainty** of Pressure Sensors (psi at 1σ) | 21 psi |
| Qualifications | IEEE 323-74<br>and IEEE 344-75<br>pending |
| **Backup Capability** | |
| Availability of Temp and Press | Available on<br>Control Board |
| Availability of Steam Tables etc. | Available to Operator |
| Procedures | Available to Opertor |

** Uncertainties must address conditions of forced flow and natural circulation

# 4.0 REACTOR VESSEL LEVEL INSTRUMENTATON SYST M - SYSTEM DESCRIPTION

## 4.1 GENERAL DESCRIPTION

The reactor vessel level instrumentation system RVLIS) uses differential pressure (d/p) measuring devices to measure vessel level or relative void content of the circulating primary coo ant system fluid. The system is redundant and includes automatic compe sation for potential temperature variations of the impulse lines. Es ential information is displayed in the control room in a form dir ctly useable by the operator.

The functions performed by the RVLIS are:

1. Assist in detecting the approach to ICC

2. Indicate the formation of a void in the RC during forced flow conditions.

## 4.2 DETAILED SYSTEM DESCRIPTION

### 4.2.1 HARDWARE DESCRIPTION

#### 4.2.1.1 Differential Pressure Measurements

The RVLIS (Figure 4-1) utilizes two sets of two /p cells. These cells measure the pressure drop from the bottom of the reactor vessel to the top of the vessel. This d/p measuring system ut lizes cells of differing ranges to cover different flow behaviors ith and without pump operation as discussed below:

1. Reactor Vessel - Narrow Range ($\Delta P_b$)

   This measurement provides an indication of reactor vessel level from the bottom of the reactor vessel to t e top of the reactor during natural circulation conditions.

2. Reactor Vessel - Wide Range ($\Delta P_c$)

This instrument provides an indication of reactor core and internals pressure drop for any combination of operating RCPs. Comparison of the measured pressure drop with the normal, singlephase pressure drop will provide an approximate indication of the relative void content or density of the circulating fluid. This instrument will monitor coolant conditions on a continuing basis during forced flow conditions.

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

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

4.2.1.2 System Layout

A schematic of the system layout for the RVLIS is shown in Figure 4-2. Consolidated Edison has identified two RCS penetrations for the cell reference lines; one reactor head connection at a spare penetration and one connection to an incore instrument conduit at the seal table.

The pressure sensing lines extending from the RCS penetrations will be a combination of 3/4 inch Schedule 160 piping and 3/8 inch tubing and will

include a 3/4 inch manual isolation valve as described in Section 4.2.4. These lines connect to four sealed capillary impulse lines (two at the reactor head, and two at the seal table) which transmit the pressure measurements to the d/p transmitters located outside the containment building. The capillary impulse lines are sealed at the RCS end with a sensor bellows which serves as a hydraulic coupling for the pressure measurement. The impulse lines extend from the sensor bellows through the containment wall to hydraulic isolators, which also provide hydraulic coupling as well as a seal and isolation of the lines. The capillary tubing extends from the hydraulic isolators to the d/p transmitters, where instrument valves are provided for isolation and bypass.

Figure 4-3 is an elevation plan of Indian Point 2 showing the routing of the impulse lines. The impulse lines from the vessel head connection must be routed upward out of the refueling canal to the operating deck, then radially toward the seal table and then to the containment penetration. The connection to the bottom of the reactor vessel is made through an incore detector conduit which is tapped with a T connection at the seal table. The impulse line from this connection is routed axially and radially to join with the head connection line in routing to the penetrations.

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

### 4.2.2 7300 SERIES RVLIS

The 7300 series RVLIS is configured as two trains (protection sets) in the outer bays of a standard three-bay cabinet. The system uses the same components and cabinet that is used in the 7300 series nuclear protection and control systems. The block diagram of the process equipment is shown in Figure 4-4. For displayed information, see Figure 4-5.

Conformance with Regulatory Guide 1.97 for the 7300 display system is given in Table 4.1.

#### 4.2.2.1 RVLIS Inputs

The 7300 series process equipment inputs are as follows. If existing unqualified inputs are used, isolation as required will be provided by the owner.

Hot Leg Wide Range Temperature

Each RVLIS train receives two hot leg wide range temperature signals derived from existing channels in the NSSS Process Protection System.

The hot leg temperature signal is used to compensate the measured reactor vessel d/p to produce an indicated liquid level value during conditions when the liquid is subcooled.

Wide Range RCS Pressure

Each RVLIS train receives one wide range RCS pressure signal derived from existing channels in the NSSS Process Protection System. The RCS pressure signal is used to compensate the measured reactor vessel d/p to produce a liquid level value during conditions when the coolant is saturated. The selection between temperature and pressure compensation is automatic.

Reactor Vessel Narrow Range Differential Pressure

Each train receives one narrow range d/p measurement. This signal is provided from a new transmitter and when compensated, yields the level indication spanning the entire reactor vessel during periods when the reactor coolant pumps are not running.

[ ] a,c

$\Delta P_b$ gives an indication of reactor vessel level when no pumps are running. If one or more pumps are running, $\Delta P_b$ will be off-scale and the reading invalid.

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

Reactor Vessel Wide Range Differential Pressure

Each train receives one wide range d/p measurement. This signal is provided from a new transmitter and when compensated, will yield the relative void content of the circulating primary coolant system fluid during periods when any reactor coolant pumps are running.

[ ] a, c

[ ] The sense of the $\Delta P_c$ output is that 20 ma represents all pumps running and 4 ma is empty vessel. With all pumps running and no void fraction, the $\Delta P_c$ should read 100 percent at zero power. The reading at full power is slightly higher.

## Capillary and Conduit Temperature

Each train receives up to 5 temperature measurements from new RTDs. These RTDs provide compensation signals used to cancel out temperature induced d/p effects on the instrumentation system.

A typical arrangement of the reference leg temperature RTDs is shown in Figure 4-6.

The 100 ohm platinum four wire RTDs are capable of operating over the temperature range of -58° to 500°F. Due to temperature restrictions on containment cabling, the useful temperature range is 32°F to 450°F

## Density Calculation

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

## Vessel Liquid Density Calculation

[ ] a, c

## Vessel Vapor Phase Density Calculation

[ ] a, c

Vessel Level Calculation

[ ] a, c

Pump Flow d/p Calculation

[ ] a, c

4.2.2.2 RVLIS Outputs

Plant Operator Interface and Displays

Information displayed to the operator for the RVLIS is intended to be unambiguous and reliable to minimize the potential for operator error or misinterpretation.

## Level Indication

Narrow range and wide range level signals are available from each train for display on standard VX-252 type vertical scale voltage meters. Thus, the indication is compatible with existing control board layouts. The indication signals are electrically isolated from the 1E circuitry and are suitable to serve as either a standard control grade or post-accident monitoring output.

The control board displays provide the following information:

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

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

Redundant displays are provided for the two sets. Level information based on both d/p measurements is presented. Correction for reference leg densities is automatic.

## Level Recording

Narrow range and wide range level signals are available on one of the two trains for trending on the chart recorder. These signals are standard 0 to 10 volt range and are electrically isolated from the protection set. Thus, they are suitable for either control grade or post-accident monitoring applications.

Computer Outputs

Analog inputs to the plant computer are available to monitor each of the RTD and transmitter inputs plus each of the compensated level outputs. These outputs are standard 0 to 10 volt ranges and may be used by the plant computer to do an independent compensation calculation of the reactor vessel d/p.

4.2.2.3 Additional 7300 Series RVLIS Features

1. The 7300 series RVLIS features full systems testing capability without having to lift wires at the termination area. Test injection points and test measurement points are available throughout the system to facilitate ease of calibration and maintenance.

   RVLIS channels are designed to permit maintenance on one channel during power operation. During such operation the active parts of the system need not themselves continue to meet the single failure criterion. As such, monitoring systems comprised of two redundant channels are permitted to violate the single failure criterion during maintenance provided that acceptable reliability of operation for the channel not under maintenance can be demonstrated. The time interval allowed for a maintenance operation will be specified in the plant Technical Specifications. Bypass indication may be applied administratively.

2. The 7300 series RVLIS has card edge adjustments and settings for ease in scaling in modifications due to changes in the installation layout. All systems set-up may be performed by a field technician rather than requiring offsite calibration.

4.2.3 RESISTANCE TEMPERATURE DETECTORS (RTD)

The resistance temperature detectors (RTD) associated with the RVLIS are utilized to obtain a temperature signal for fluid filled instrument

lines inside containment during normal and post-accident operation. The temperature measurement for all separately run vertical instrument lines is used to correct the vessel level indication for density changes associated with the environmental temperature change.

The RTD assembly is a totally enclosed and hermetically sealed strap-on device consisting of a thermal element, extension cable and termination cable as indicated in Figure 4-8. The sensitive portion of the device is mounted in a removable adapter assembly which is designed to conform to the surface of the tubing or piping being monitored. The materials are all selected to be compatible with the normal and post-accident environment. Randomly selected samples from the controlled (material, manufacturing, etc.) production lot will be qualified by type testing. Qualification testing will consist of thermal aging, irradiation, seismic testing and testing under simulation high energy line break environmental conditions. For the qualified life requirements, see Section 2.3. The specific qualification requirements for the RTDs are as follows:

1. Aging

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

2. Radiation

The detectors will be irradiated to a total integrated dose (TID) of $1.2 \times 10^8$ rads gamma radiation using a $Co^{60}$ source at a minimum rate of $2.0 \times 10^6$ rads/hour and a maximum rate of $2.5 \times 10^6$ rads/hour. Any externally exposed organic materials will be evaluated or tested to $9 \times 10^8$ rads TID beta radiation with energy levels of 6 MEV for the first 10 MRad, 3 MEV for 340 MRad and 1 MEV for 550 MRad.

3. Seismic

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

4. High Energy Line Break Simulation

The detectors will be tested in a saturated steam environment using the temperature/pressure curve shown in Figure 4-9.

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

### 4.2.4 REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM VALVES

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

### 4.2.5 TRANSMITTERS, HYDRAULIC ISOLATORS, AND SENSORS

Differential Pressure Transmitters

The d/p transmitters are a seismically qualified design as used in numerous other plant applications. In the RVLIS application, accuracy considerations dictate a protected environment, consequently transmitters are rated for 40 to 130°F and $10^4$ rad TID.

Several special requirements for these transmitters are as follows:

1. Must withstand long term overloads of up to 300 percent with minimal effect on calibration.

2. High range and bi-directional units required for pump head measurements.

3. Must displace minimal volumes of fluid in normal and overrange operating modes.

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

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

Hydraulic Isolator

a, c

High Volume Sensor

a, c

a, c

[ ]

## 4.3 TEST PROGRAMS

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

### 4.3.1 Forest Hills

A breadboard installation consisting of one train of a RVLIS was installed and tested at the Westinghouse Forest Hills Test Facility. The system consisted of a full single train of RVLIS hydraulic components (sensor assemblies, hydraulic isolators, isolation and bypass valves and d/p transmitters) connected to a simulated reactor vessel. Process connections were made to simulate the reactor head, hot leg and seal table connections. Capillary tubing which in one sensing line simulated the maximum expected length (400 feet) was used to connect the sensor assemblies to the hydraulic isolators and all joints were welded. Connections between the hydraulic isolators, valves and transmitters utilized compression fittings in most cases. Resistance temperature

detectors, special large volume sensor bellows and volume displacers inside the hydraulic isolator assemblies which are normally part of a RVLIS installation were not included in the installation since elevated temperature testing was not included in the program.

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

The objectives of the test were as follows:

1. Obtain installation, filling and maintenance experience

2. Prove and establish filling procedures for initial filling and system maintenance.

3. Establish calibration and fluid inventory maintenance procedures for shutdown and normal operation conditions.

4. Prove long term integrity of hydraulic components

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

6. Verify leak test procedures for field use

Reactor Vessel Simulator

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

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

## Installation

The system was installed in the high bay test area of the Westinghouse Forest Hills Test Facility by Westinghouse personnel under the supervision of Forest Hills Test Engineering. All local safety codes were considered in the construction.

## Filling Operation

a, c

[ ] a, c

4.3.2 SEMISCALE TESTS

In order to study the transient response of the RVLIS during a small-break LOCA and other accident conditions, the hydraulic components of the RVLIS have been installed at the Semiscale Test Facility in Idaho. Vessel level measurements will be obtained during the current semiscale test program series which runs from December 1980 to March 1982. The test scheduled to be completed by July 1981 are expected to provide the desired transient response verification; additional data pertaining to the comparison of the d/p system to the gamma densitometer will be obtained from the tests scheduled for completion by November 1981.

The Semiscale Test Facility is a model of a 4-Loop pressurized water reactor coolant system with elevation dimensions essentially equal to the dimensions of a full-size system. The reactor vessel contains an electrically heated fuel assembly consisting of 25 fuel rods with a heated length of 12 feet. Two reactor coolant loops are provided, each having a pump and a steam generator with a full height tube bundle. One loop models the loop containing the pipe break, which can be located at any point in the loop. The other loop models the three intact loops. A blowdown tank collects and cools the fluid discharged from the pipe break during the simulated accident. Over 300 pressure, temperature, flow, level and fluid density instruments are installed in the reactor vessel and loops to record the fluid conditions throughout a test run. Test results are compared with predictions for verification of computer code models of the transient performance.

The Westinghouse level measurements obtained during a test run will be compared with data obtained from existing instrumentation installed on the semiscale reactor vessel. The semiscale facility has two methods of measuring the level or fluid density: d/p measurements are obtained over 11 vertical spans on the reactor vessel to determine level within each span, and gamma densitometers are installed at 12 elevations on the reactor vessel to determine the fluid density at each elevation. This data establishes a fluid density profile within the vessel under any operating condition. This information will be compared with the data obtained from the Westinghouse level instrumentation. Other semiscale facility instruments (loop flows and fluid densities when pumps are operating, and pressure and temperatures for all cases) will provide supplemental information for interpretation of the test facility fluid conditions and the level measurement.

Specific tests included in the semiscale test program during which Westinghouse RVLIS measurements will be obtained are as follows:

1. Miscellaneous steady state and transient tests with pumps on and off, to calibrate test facility heat losses.

2. Small-break LOCA test with equivalent of a 4 inch pipe break.

3. Repeat of small-break LOCA test with test facility modified to simulate a plant with upper head injection (UHI).

4. Several natural convection tests covering subcooled and saturated coolant conditions and various void contents.

5. Tests to simulate a station blackout with discharge through relief valves.

6. Simulation of the St. Lucie cooldown incident.

### 4.3.3 PLANT STARTUP CALIBRATION

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

1. During refilling and venting of the reactor vessel, measurements of all 4 d/p transmitters would be compared to confirm identical level indications.

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

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

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

## 4.4 OPERATING PERFORMANCE

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

1. Reactor Vessel - Narrow Range

   The transmitter span covers the total height of the reactor vessel. With pumps shut down, the transmitter output is an indication of the collapsed water level, i.e., as if the steam bubbles had been separated from the water volume. The actual water level is slightly higher than the indicated water level since there will be some quantity of steam bubbles in the water volume. Therefore, the RVLIS provides a conservative indication of the level effective for adequate core cooling.

   When reactor coolant pumps are operating, the d/p would be greater than the transmitter span, and the transmitter output would be disregarded.

2. Reactor Vessel - Wide Range

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

The output of each transmitter is compensated for the density difference between the fluid in the reactor vessel and the fluid in the reference leg at the initial ambient temperature. The compensation is based on a wide range hot leg temperature measurement or a wide range system pressure measurement, whichever results in the highest value of water density, and, therefore, the lowest value of indicated level. Compensation based on temperature is applied when the system is subcooled, and compensation based on pressure (saturated conditions) is applied if superheat exists at the hot leg temperature measurement point.

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

The corrected transmitter outputs are displayed on meters installed on the control board, one meter for each measurement in each train. A two-pen recorder is also provided on the control board to record the level or relative d/p and to display trends in the measurements.

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

If all pumps are shut down, at any temperature, the narrow range meter would indicate 100 percent, an indication that the vessel is full. The wide range d/p meter would indicate about 33 percent of the span of the meter, which would be the value (determined during the test program) corresponding to a full vessel with pumps shut down.

In the event of a LOCA where coolant pressure has decreased to a predetermined setpoint, existing emergency procedures would require shut-

down of all reactor coolant pumps. In these cases, a level will eventually be established in the reactor vessel and indicated on all of the meters. The plant operator would monitor the meters and the recorder to determine the trend in fluid mass or level in the vessel, and confirm that the ECCS is adequately compensating for the accident conditions to prevent ICC.

Future procedures may require operation of one or more pumps for recovery from certain types of accidents. When pumps are operating while voids are developing in the system, the pumps will circulate the water and steam as an essentially homogeneous mixture. In these cases, there will be no discernible level in the reactor vessel. A decrease in the measured d/p compared to the normal operating value will be an indication of voids in the system, and a continuously decreasing d/p will indicate that the void content is increasing, that mass is being lost from the system. An increasing d/p will indicate that the mass content is increasing, that the ECCS is effectively restoring the system mass content.

## 4.5 RVLIS ANALYSIS

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

This indication corresponds to the RVLIS configuration used for non-UHI plants. The conclusions of the study are expected to be the same for the UHI configuration.

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

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

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

Four small-break transients under a variety of conditions are discussed in the next section. Three of these cases were obtained from WFLASH analyses and the other was obtained from the ICC analysis using NOTRUMP. A description of these codes can be found in References 1 through 6 in Section 6.0.

The transients included in this report are listed Table 4.2 which gives a brief description of the transient, the plant type, and the model used for the analysis. A discussion of each transient is provided in the next section. Figures 4-12 through 4-21 provide plots of vessel two-

phase mixture level, RVLIS narrow range reading, mixture and vessel void fraction, and for Case B with pumps running, RVLIS wide range reading and cold leg mass flowrate..

The two-phase mixture level plotted is that which was predicted by the codes for the mixture height below the upper support plate. Water in the upper head is not reflected in this plot. The RVLIS reading that would be seen is plotted on the same figure for ease of comparison.

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

### 4.5.1 Transients Investigated

Case A

The initiating event for this transient (Figure 4-12) is a 3 inch break in the cold leg. After the break opens, the system depressurizes rapidly to the steam generator secondary safety valve setpoint. Consistent with the FSAR assumptions, the reactor coolant pumps are assumed to trip early in the transient when the reactor trips.

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

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

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

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

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

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

Case B

This case is the same as case A except it was assumed that the reactor coolant pumps continued to operate until 750 seconds (Figure 4-15). If

the reactor coolant pump trip criteria is followed the pumps would be tripped much earlier in the transient. This case is, however, instructive in determining the RVLIS response when the pumps are running.

After the break opens, the system depressurizes rapidly to the secondary safety valve setpoint, and then begins a period of very slow depressurization. During this time the upper portions of the system drain. Due to the reactor coolant pump operation, the two-phase mixture in the vessel remains at the hot leg elevation, although the void fraction of the mixture continues to increase (Figure 4-16).

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

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

During the early portion of the transient the wide range RVLIS reading drops fairly smoothly from 100 percent to about 20 percent (Figure 4-17), which is due to the decreasing mass in the vessel and the decreasing pressure drop as the pump performance is degraded. The plot of cold leg mass flowrate is indicative of the pump degradation. The oscillations in this plot are due to alternate steam and two-phase flow predicated by WFLASH. When the flow through the pump becomes mostly steam, the increasing void fraction of the vessel mixture becomes the predominant factor in the decreasing RVLIS reading.

RCP operation keeps the steam and water mixed enough that the mixture level does not fall below the hot legs, although the mixture void fraction is increasing during this time. This loss of inventory is indicated by the continued drop in the RVLIS reading. When the pumps trip, the steam and water in the mixture separate and there is a rapid de-

crease in the core mixture level and mixture void fraction although the vessel void fraction continues to rise. The fact that mass is being redistributed rather than lost is seen in the RVLIS reading - there is little change in the reading (compared to the change in level) from 750 seconds to the time that the accumulators come on.

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

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

Case C

The initiating event for this transient (Figures 4-18 and 4-19) is the opening of the pressurizer power operated relief valves (PORVs). The reactor coolant pumps and the reactor trip early in the transient on a low pressurizer pressure signal consistent with FSAR assumptions. Auxiliary feedwater is available in this case but, no pumped safety injection is assumed.

The pressurizer mixture level rises to the top of the pressurizer early in the transient and stays at this level throughout most of the transient. The flow through the PORVs alternates between steam and twophase mixture while the pressure in the system drops rapidly to the steam

generator secondary safety valve setpoint. The pressure hangs up at this value until the upper portion of the system has drained and then continues to decrease. When the upper portions of the primary system (excluding the pressurizer) have drained the vessel mixture level begins to decrease and continues until the core completely uncovers.

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

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

Case D

This case (1 inch cold leg break with no safety injection) is one of the transients investigated for the ICC study using NOTRUMP. A more detailed discussion of this transient can be found in Reference 1.

The RVLIS reading (Figures 4-20 and 4-21) is below the vessel mixture level throughout most of the transient and is therefore a conservative indication. The RVLIS reading follows the same trend as the vessel mixture level except for early in the transient when the mixture void fraction is fluctuating.

Included in the plots for this case is a comparison of the mass inventory in the core and upper plenum regions to the RVLIS reading. This comparison shows that the RVLIS reading also corresponds very well with the relative vessel mass inventory.

### 4.5.2 Observations Of The Study

The RVLIS will provide useful information for breaks in the system ranging from small leaks to breaks in the limiting small-break range. For breaks in this range, the system conditions will change at a slow enough rate that the operator will be able to use the RVLIS information as a basis for some action.

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

Very few instances have been identified where the RVLIS may give an amibiguous indication. These include a break in the upper head, accumulator injection into a highly voided downcomer, periods of time when the upper head behaves like a pressurizer and periods of void redistribution.

A break in the upper head may cause a much lower pressure to exist in the upper head compared to the rest of the RCS. Because of this the pressure difference between the lower plenum and the upper head is much larger than is seen for an equivalent vessel level when the break is located elsewhere in the system. The reading, in fact, may never reach the narrow range scale. If the narrow range reading remains at full scale and the wide range reading is greater than that reading which would indicate a full vessel with the reactor coolant pumps tripped, a break in the upper head is indicated. This situation should not cause a problem in detecting ICC because of the parallel logic for the "kickout" to the ICC procedures. If the RVLIS indication is erroneous due to a

break in the reactor vessel upper head, the operator will begin following the ICC procedure if the selected core exit thermocouples read 1200°F.

This situation only exists, however, when the break discharge is large enough to cause a large d/p through the flow paths connecting the upper head to the rest of the system. These flow paths become the limiting factor in the depressurization rate.

This analysis is applicable to all Westinghouse PWR plants, including Indian Point Unit 2 which is a four loop plant.

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

During periods when the void distribution in the vessel is changing rapidly, there may be a large change in two-phase mixture level with very little change in mass inventory in the vessel. This could happen if the reactor coolant pumps (RCPs) were tripped when the mixture in the vessel was highly voided. This could cause the mixture level to drop from the hot leg elevation to below the top of the core. The operator would expect this to happen based on the fact that the RVLIS reading was within the narrow range indication. The operator should know in general that, for a brief period of time after tripping the RCPs, transient RVLIS response will occur.

Flow blockage is not expected to decrease the usefulness of the RVLIS indication. The increased d/p due to the flow blockage will be small during natural circulation. The RVLIS will continue to follow the trend in vessel level. When the reactor coolant pumps are operating, flow blockage is not expected to occur unless the pumps had previously been tripped and are being restarted after an ICC situation already exists. If flow blockage were present when the pumps were running the RVLIS indication would still be useful and, although the indication would be somewhat higher, would continue to follow the trend in vessel inventory.

### 4.5.3 Conclusions

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

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

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

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

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

6. A break of sufficient size in the upper head could cause the RVLIS to give an ambiguous indication of vessel mass. The core exit thermocouples, however, will provide an indication of ICC if appropriate.

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

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

9. Rapid void redistributions will not be detected by the RVLIS.

TABLE 4.1

CONFORMANCE WITH REGULATORY GUIDE 1.97, DRAFT 2, REV. 2 (6/4/80)
FOR THE 7300 DISPLAY SYSTEM

| | |
|---|---|
| Seismic qualification | Yes |
| Single failure criteria | Yes |
| Environmental quailification<br>*[IEEE-323-1971 applicability] | Pending |
| Power Source | Class 1E |
| Quality Assurance<br>10CFR50 Appendix B applicability | Yes |
| Display type and method | Vertical scale voltage processed in addition to a recorder |
| Unique identification | Yes |
| Periodic Testing | Yes |

* In some cases IEEE-323-1974 is applicable.

TABLE 4.2

TRANSIENTS INVESTIGATED

| CASE | PLANT | DESCRIPTION |
|---|---|---|
| A | 3 loop<br>2775 MWt | 3 inch cold leg break - FSAR assumptions*; WFLASH |
| B | 3 loop<br>2775 MWt | 3 inch cold leg break - RCPs trip at 750 seconds - otherwise, FSAR assumptions; WFLASH |
| C | 4 loop<br>UHI type<br>3411 MWt | 2.5 inch break in top of pressurizer - no UHI - no pumped safety injection - pumps not running; WFLASH |
| D | 4 loop<br>Non-UHI<br>3411 MWt | 1 inch cold leg break - no high head safety injection; NOTRUMP |

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

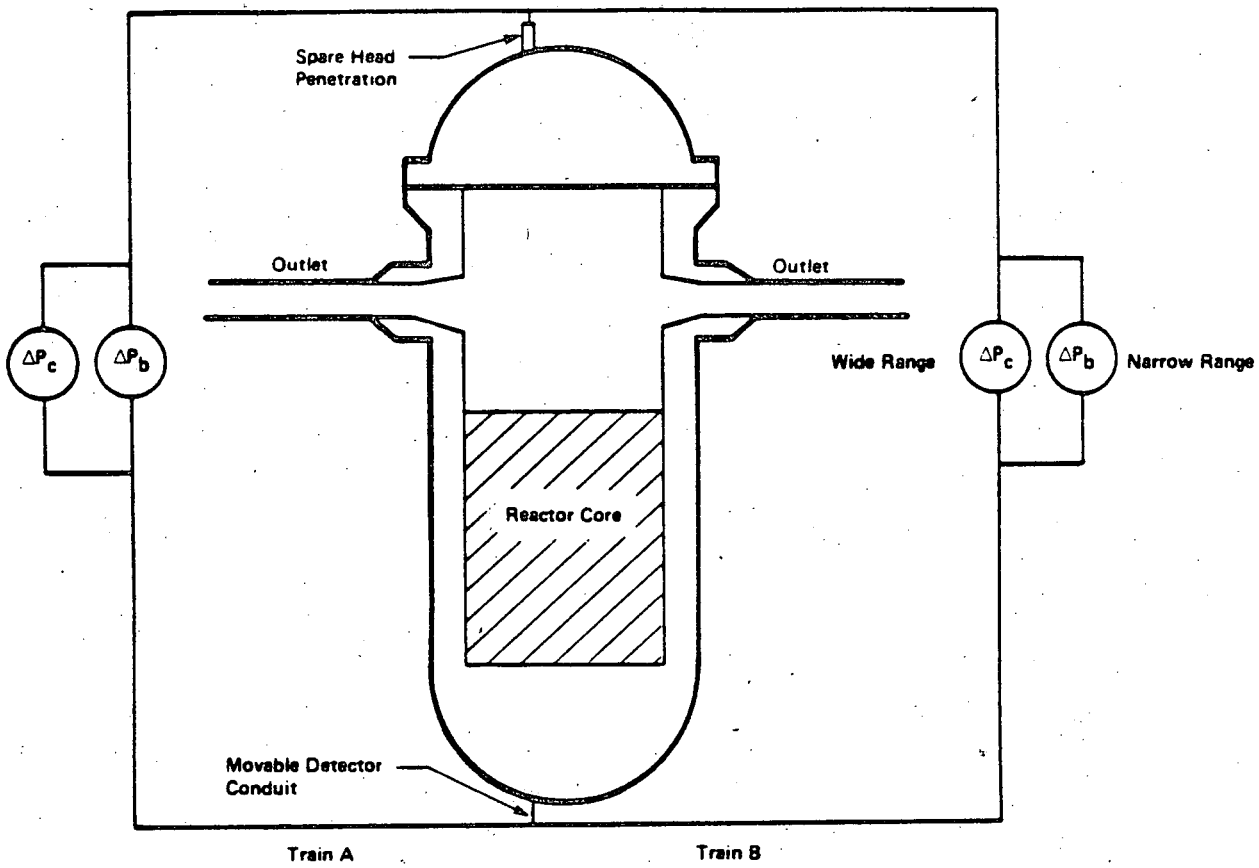
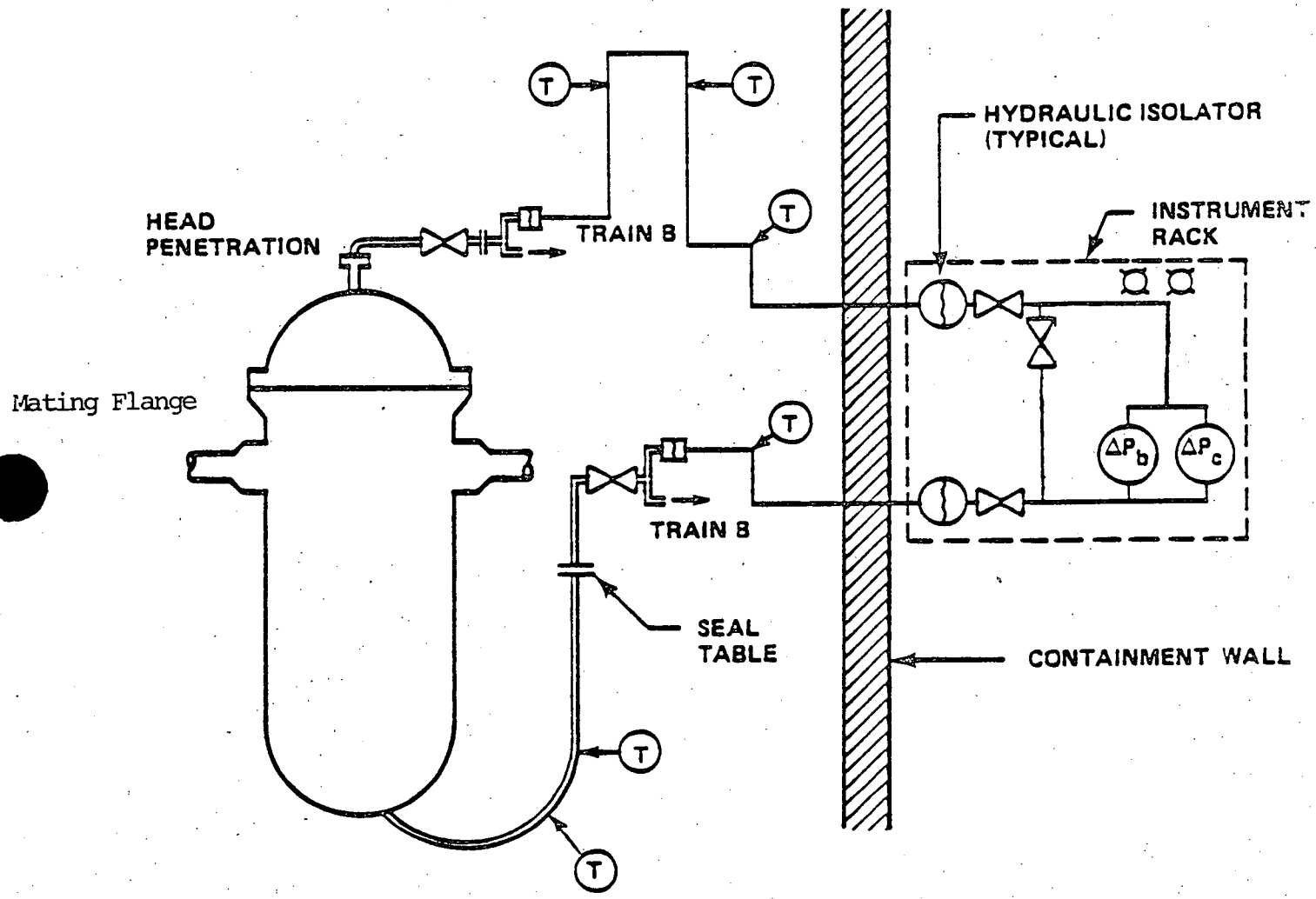



Figure 4-1 Reactor Vessel Level Instrument System for IP-2




NOTE: The Mating Flange and the Seal Table are at the same elevation.

Figure 4-2 Process Connection Schematic, Train B for IP-2

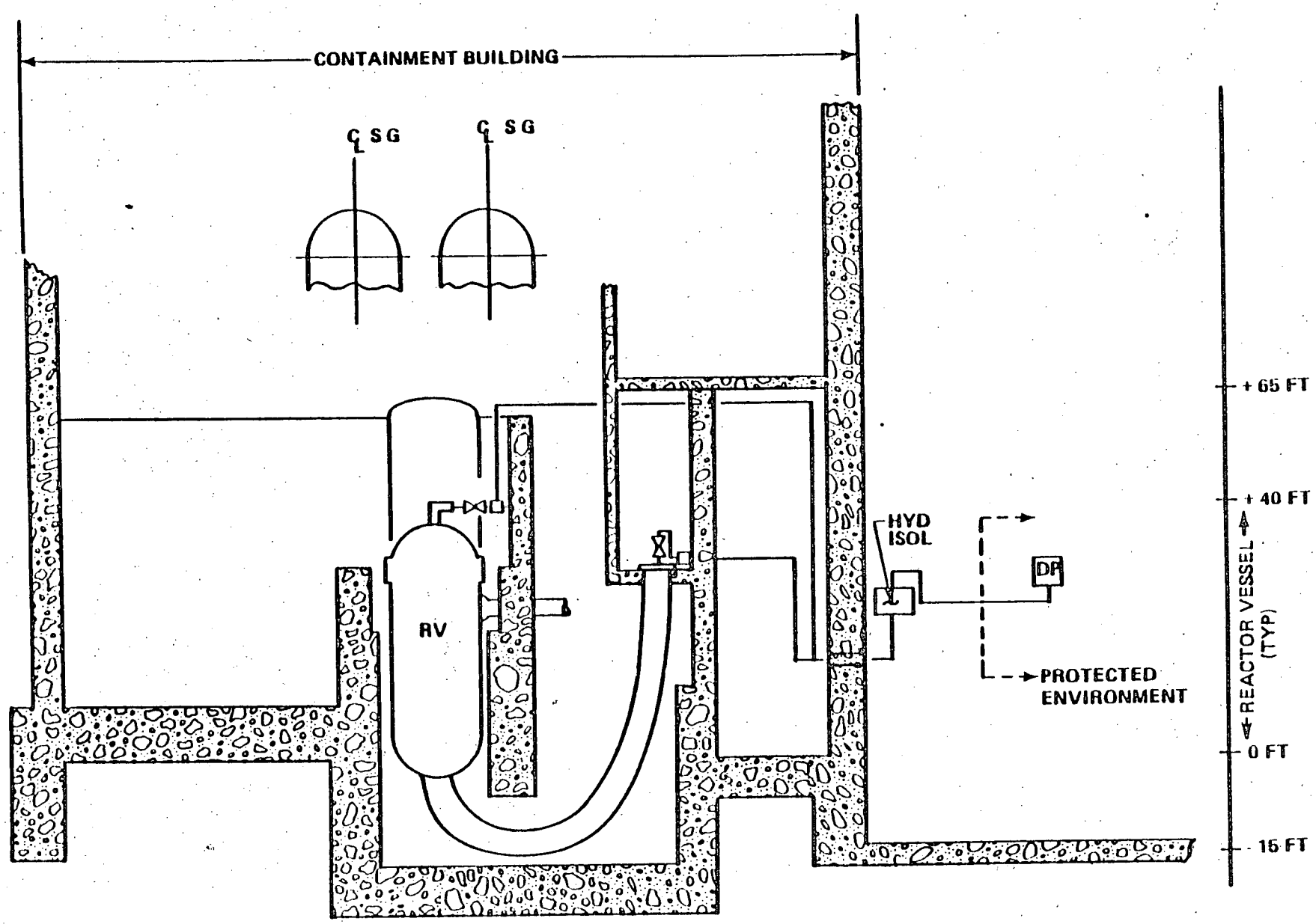



Figure 4-3 TP 2 Plant Arrangement for RVLIS

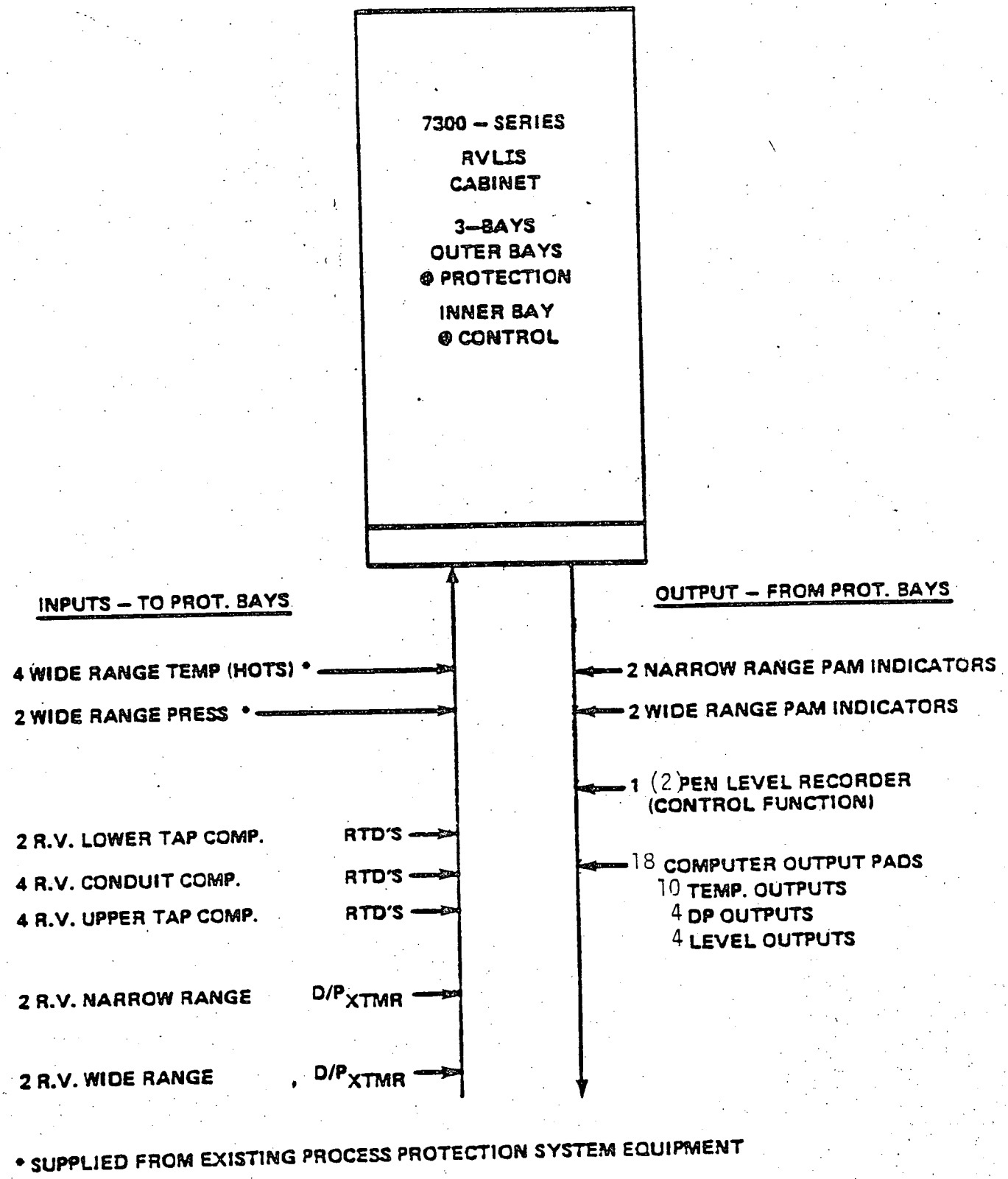
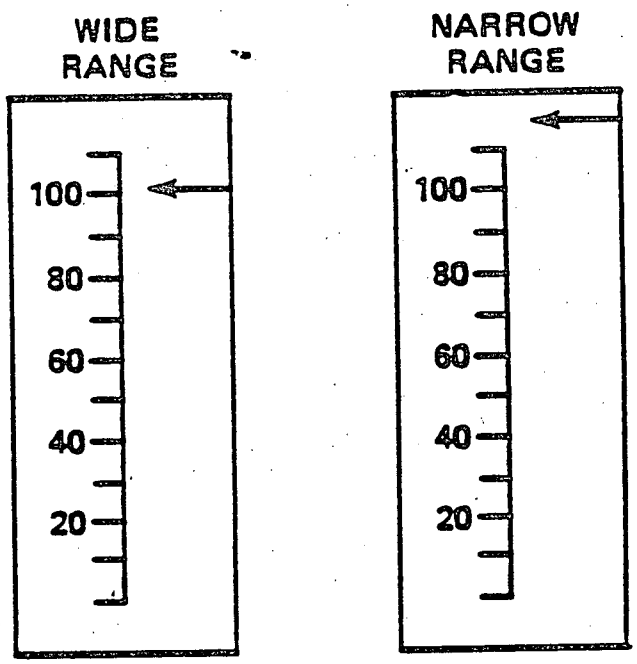



Figure 4–4 7300 Series RVLIS Processing Equipment - Block Diagram for Indian Point Unit 2




R. V. LEVEL SYSTEM INDICATION

* DISPLAY REPRESENTS ALL PUMPS OPERATIONAL

Figure 4–5 7300 Series RVLIS Display (One Set Shown - Other Set Same) for IP-2

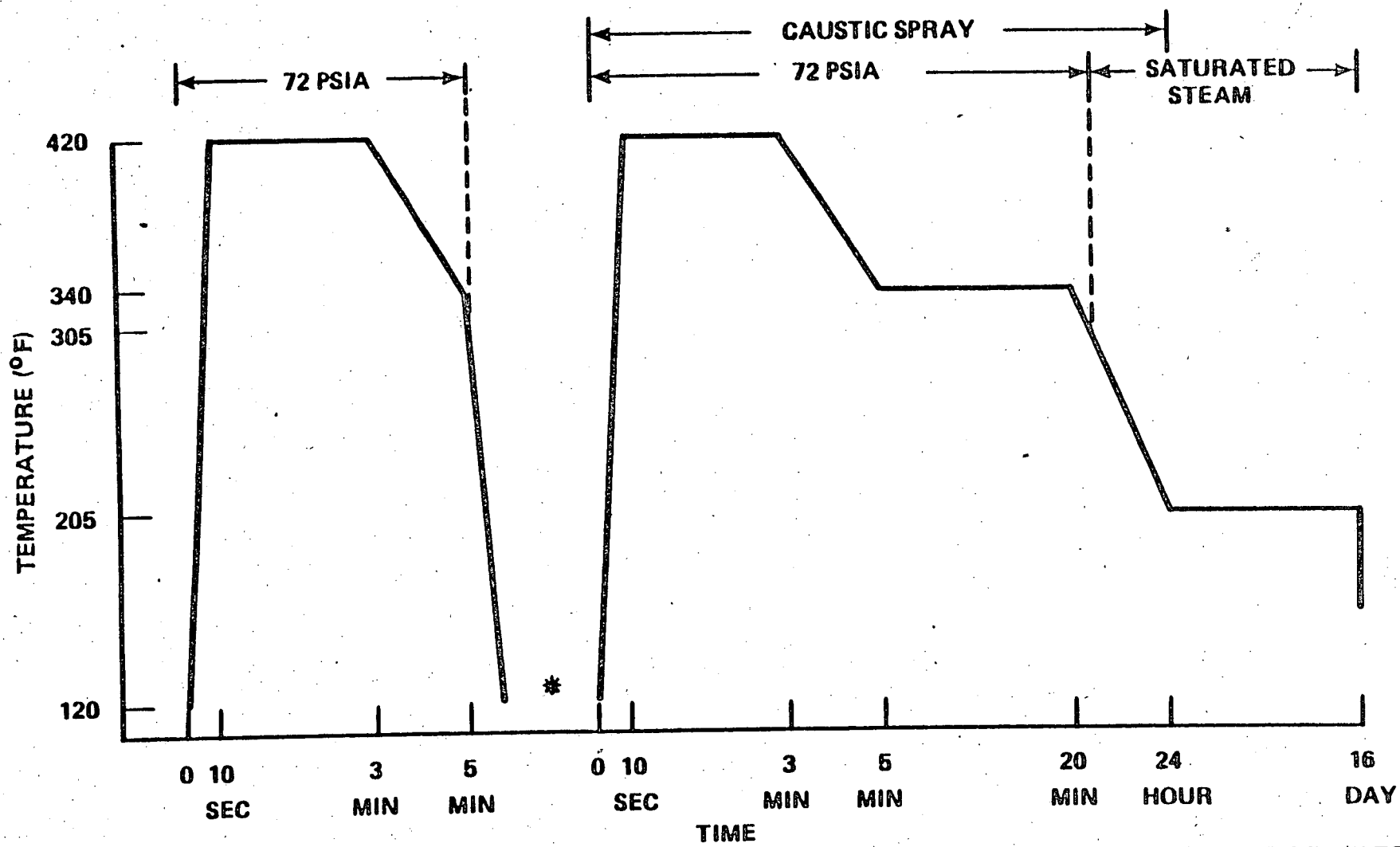
a,b,c

FIGURE 4-6 IP-2 Plant Arrangement for RVLIS

Figure 4–7 Block Diagram of Compensation Function for Indian Point 2

a,b,c

Figure 4-8 Surface Type Clamp-On Resistance Temperature Detector




*TIME BETWEEN TEMPERATURE TRANSIENTS MUST BE AT LEAST ONE HOUR OR UNTIL TEST UNITS RETURN TO A STEADY STATE OUTPUT. TIME ABOVE 340°F MUST BE FIVE MINUTES OR LESS.

Figure 4-9 HELB Simulation Profile

17917-13

a,b,c

Figure 4-10 ITT Barton Hydraulic Isolator Internal Scheme

17917-18

a,b,c

Figure 4-11 ITT Barton "High Volume" Sensor Bellows Check Valve

Figure 4-12 Case A 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip with Reactor Trip, RVLIS Reading and Vessel Mixture Level

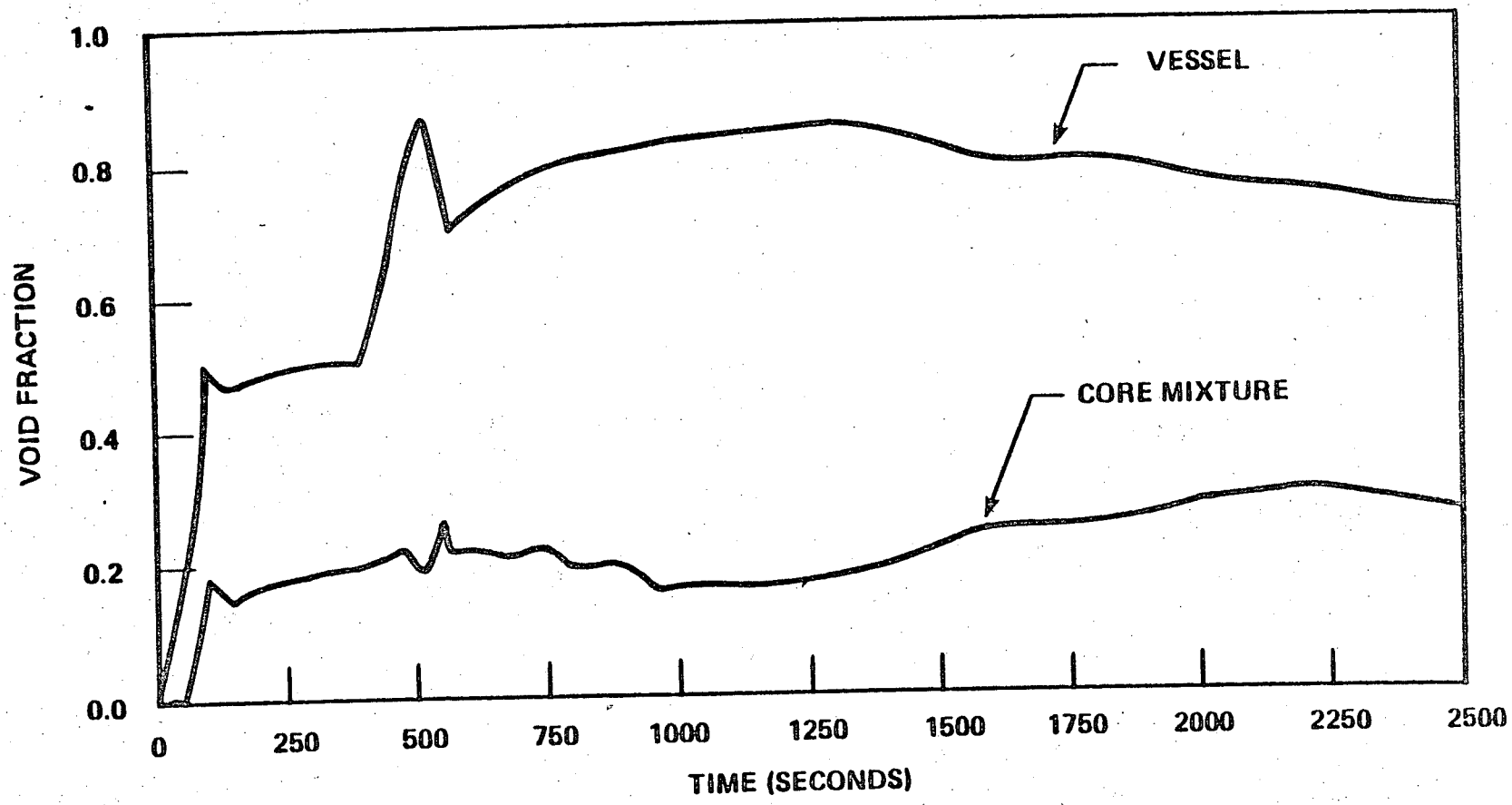



Figure 4-13 Case A 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip with Reactor Trip, Void Fraction

17917-23

a.b.c

Figure 4-14 Case B 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip at 750 Seconds, Wide Range Reading

a,b,c

Figure 4-15 Case B 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip at 750 Seconds, RVLIS Reading and Mixture Level

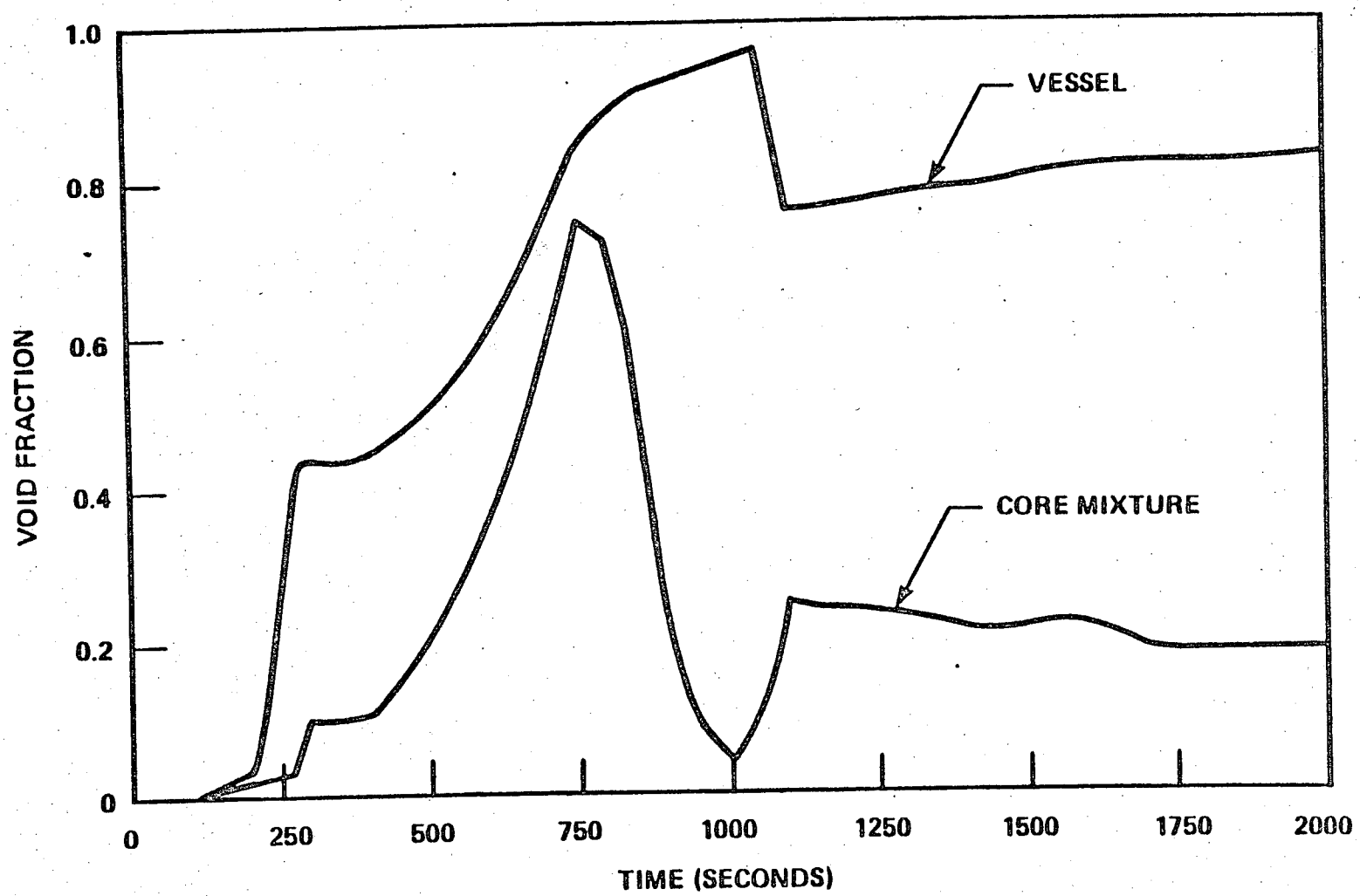



Figure 4-16 Case B 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip at 750 Seconds, Void Fraction.

17917-26

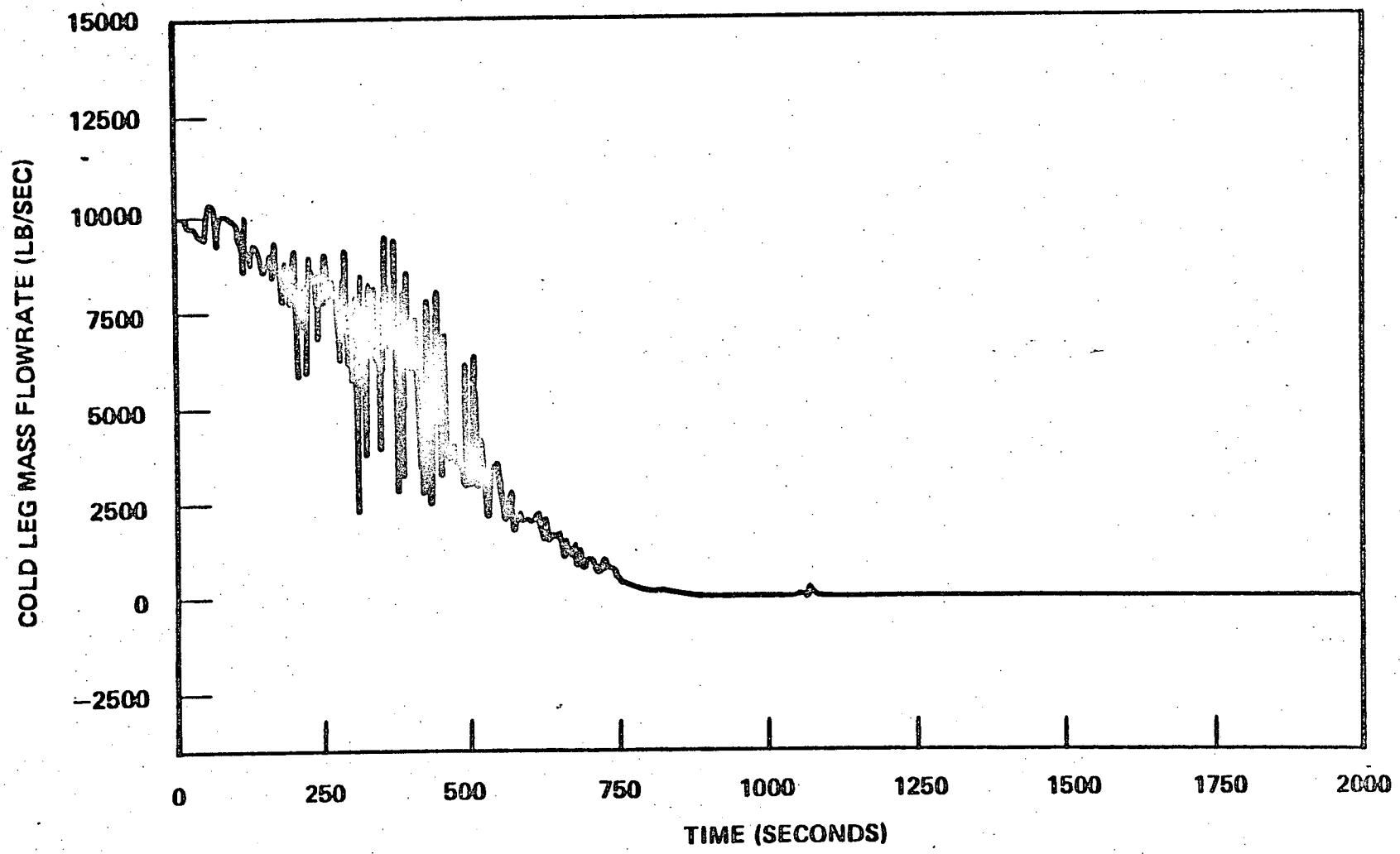



Figure 4-17 Case B 3-Loop Plant, 3 Inch Cold Leg Break, Pump Trip at 750 Seconds, Cold Leg Mass Flowrate (LB/Sec)

17917-27

a.b.c.

Figure 4-18 Case C 2.5 Inch Pressurizer Break, No SI, RVLIS Reading and Mixture Level.

a,b,c

Figure 4-19 Case c 2.5 Inch Pressurizer Break, No SI, Void Fraction

17917-29

a,b,c,

Figure 4-20 Case D 1 Inch Cold Leg Break, ICC Case, Mixture Level, RVLIS Reading and Measured Inventory

17917-31

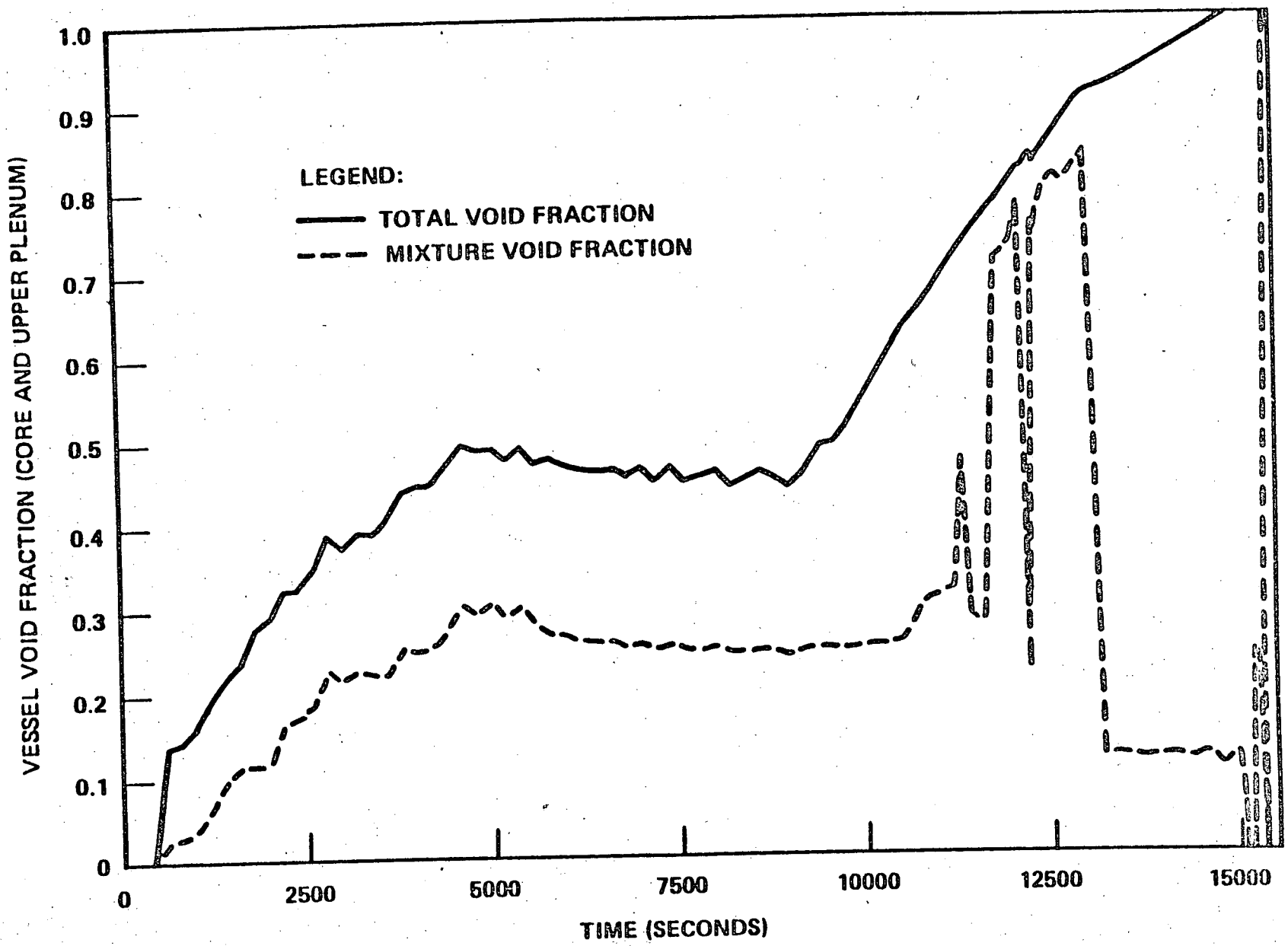



Figure 4-21 Case D 1 Inch Cold Leg Break, ICC Case, Void Fraction

17917-33

# 5.0 GUIDELINES FOR THE USE OF ICC INSTRUMENTATION

## 5.1 REFERENCE OWNERS GROUP PROCEDURES

Based on the analyses defined in Sections 1.3 and 4.5 of this report, Westinghouse and the Westinghouse Owners Group have developed a Reference Emergency Operating Instruction to address recovery from ICC conditions caused by a small-break LOCA without high head safety injection. This instruction has been transmitters to the NRC via Westinghouse Owners Group Lett r, OG-44, dated November 10, 1980. It should be noted that this instruc ion was developed on a generic basis as a technical reference for implementing plant specific rocedures, and must be tailored to meet plant specific needs.

## 5.2 SAMPLE TRANSIENT

The response of the vessel level indications, other ICC instrumentation and system response during these ICC events and recovery actions are described in References 1 and 2.

## 6.0 REFERENCES


1. Thompson, C. M., et al., "Inadequate Core Cooling Studies of Scenarios with Feedwater Available, Using the NOTRUMP Computer Code," WCAP-9753 (Proprietary) and WCAP-9754 (Non-Proprietary), July 1980.

2. Mark, R. H., et al., "Inadequate Core Cooling Studies of Scenarios with Feedwater Available for UHI Plants, Using the NOTRUMP Computer Code," WCAP-9762 (Proprietary) and WCAP-9763 (Non-Proprietary), June 1980.

3. "Report on Small Break Accidents for Westinghouse Nuclear Steam Supply System," WCAP-9600 (Proprietary) and WCAP-9601 (Non-Proprietary), June 1979.

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

5. Skwarek, R., Johnson, W., and Meyer, P., "Westinghouse Emergency Core Cooling System Small Break October 1975 Model," WCAP-8970 (Proprietary) and WCAP-8971 (Non-Proprietary), April 1977.

6. "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss of Coolant Accident for Westinghouse NSSS," WCAP-9584 (Proprietary) and WCAP-9585 (Non-Proprietary), August 1979.

Enclosure 2

Modifications to Emergency Procedures for ICC
(NUREG-0737, Item II.F.2)

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
September, 1981

## II.F.2 - MODIFICATION TO EMERGENCY PROCEDURES FOR ICC

The following pages contain a summary of the key operator action instructions in the current emergency procedures for ICC. The modifications to these action instructions upon implementation of the final monitoring system are as follows:

1) Under "Immediate Operator Action", an item will be added to direct the operator to monitor the reactor vessel level indication system for a subcooled condition.

2) Under "Subsequent Actions", words will be added to the note in 5.1.a. to direct the operator that the Reactor Coolant System status can also be determined by observing the reactor vessel level indication.

Decay Heat Removal Using Natural Circulation
(Inadequate Core Cooling)

## 1.0 Description of Conditions

Below are the maximum typical amounts of decay heat that must be removed by convection heat transfer.

| Time after reactor trip | %Power | MWT |
|---|---|---|
| 0 | 100.0 | 2758 |
| 1 sec | 8.5 | 235 |
| 10 sec | 6.2 | 170 |
| 1 min | 4.7 | 130 |
| 1 hr | 1.6 | 44 |
| 1 day | 0.60 | 16.5 |
| 1 week | 0.30 | 8.3 |
| 1 month | 0.15 | 4 |
| 1 year | 0.04 | 1.1 |

Inadequate Core Cooling is a condition during which core temperatures cannot be reduced or cannot be stabilized below the saturation temperature corresponding to the existing pressure. This condition maybe the result of loss of all outside power causing a coincident loss of condenser steam dump and the ability to run 6900 volt equipment (e.g. Reactor Coolant Pumps).

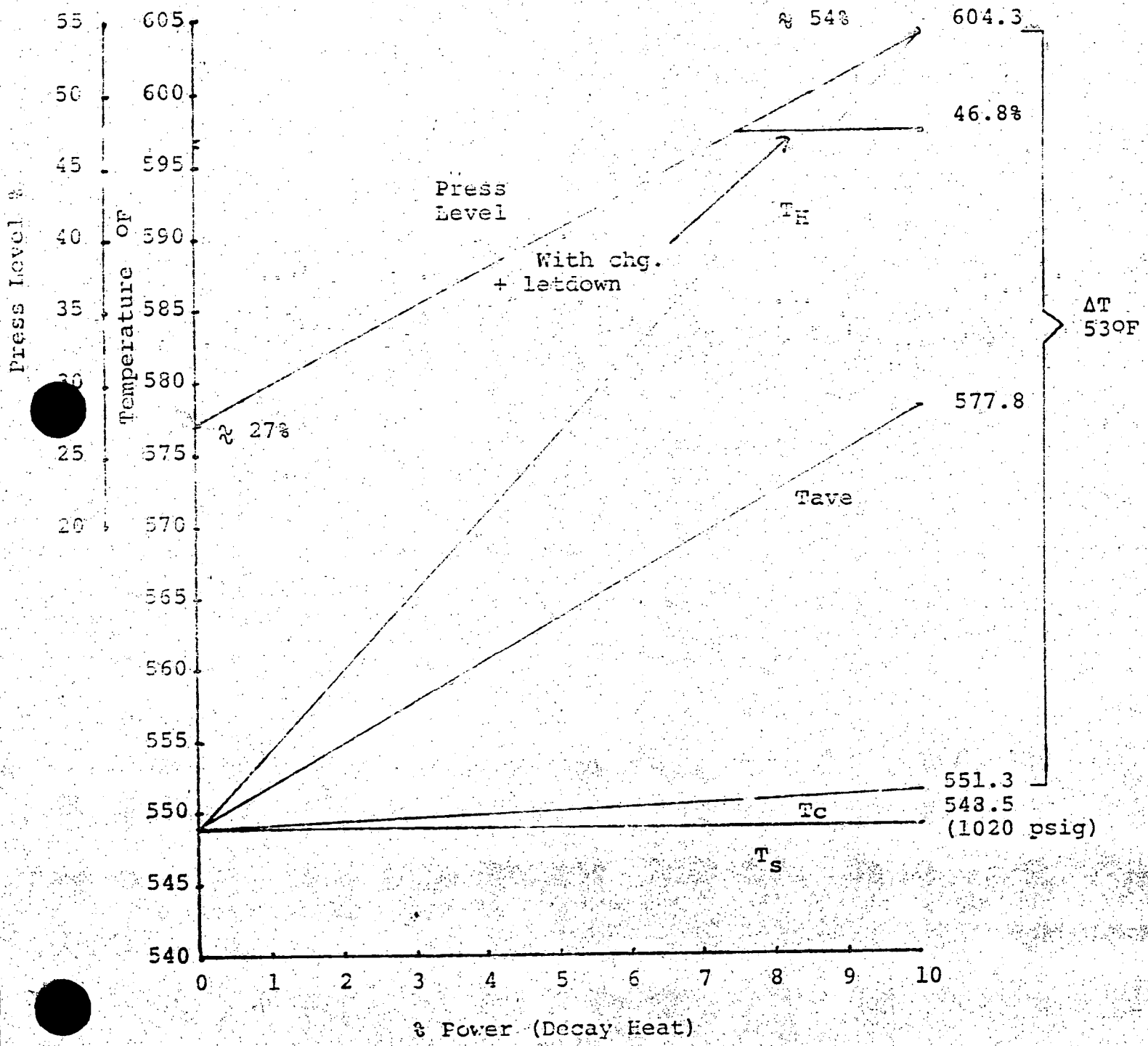
NOTE: Atmospheric steam dump is rated at 10% of Reactor Power and will remove all decay heat requirements after 1 second beyond reactor trip.

NOTE: Inadequate core cooling may result from a failure to deliver auxiliary feedwater to the core when required.

2.0 Indication to Operator

2.1 Actuation of Atmospheric Relief Valves or Safety Valves

2.2 Under the above conditions, plant parameters should trend as follows:



NOTE: 1) The driving force for heat transfer is the RCS $\Delta T$. Due to RTD location there will be a time delay in temperature readings. The peak value TH or T in-core will reach is dependent on decay heat and will be seen prior to the start of convection heat transfer.

2) Actual pressurizer level is indicated on the above graph. Indicated level will depend on pressurizer temperature. See Graph 3A.

2.3 Loss of flow on all RCP's.

2.4 Loss of all CWP's and Heater Drain pumps if in service.

2.5 Loss of condenser vacuum.

2.6 Loss of RCP's spray for pressurizer.

2.7 Possible increasing pressurizer pressure.

3.0 Immediate Automatic Action

3.1 Atmospheric relief actuation at 1020 psig.

4.0 Immediate Operator Action

4.1 Verify all automatic actuations as specified in E-4 have taken place.

4.2 Maintain safety injection and/or charging flow to the reactor coolant system, and attempt to continue feedwater flow to the steam generators.

4.3 Monitor core exit thermocouples to determine effectiveness of cooldown.

NOTE: Core exit thermocouples are the best indicators of an inadequate core cooling condition.

4.4 Begin a depressurization of the reactor coolant system, while maintaining the RCS above saturation. (Refer to Graph RCS-12). Maintain RCS within limits of RCS cooldown curves (Graph RCS-1B), using incore thermocouples and RCS pressure as parameters.

**Caution:** Increases in pressurizer level during any natural circulation cooldown would indicate the creation of voids. An excessive cooldown rate could possibly cause a creation of voids in the vessel head, due to a temperature lag caused by insufficent circulation in the head area. This condition can be corrected by reducing the cooldown rate.

4.4.1 Decrease RCS pressure using the atmospheric relief valves. Steam dump to the condenser will not be available.

4.4.2 Monitor value displayed on saturation meter. If value is decreasing, then RCS is approaching saturation conditions. Action must be taken to control RCS pressure and/or temperature in order to prevent a saturation condition.

4.4.3 If RCS pressure must be increased to maintain an above-saturation condition, pressurizer heaters may be used.

CAUTION: Using steam generators to depressurize the RCS should only be attempted if feedwater is available to maintain an effective water level for heat transfer in the steam generators.

4.4.4 If steam generators cannot be used for heat transfer, actuate PORV's as required to continue depressurization.

CAUTION: Insure pressurizer level is maintained during actuation of PORV's.

4.4.5 Utilize auxiliary spray as required if charging and letdown are available.

4.5 Continue to monitor In-Core Thermocouple Temperatures.

(604°F approximately 1600 lb. saturation pressure)

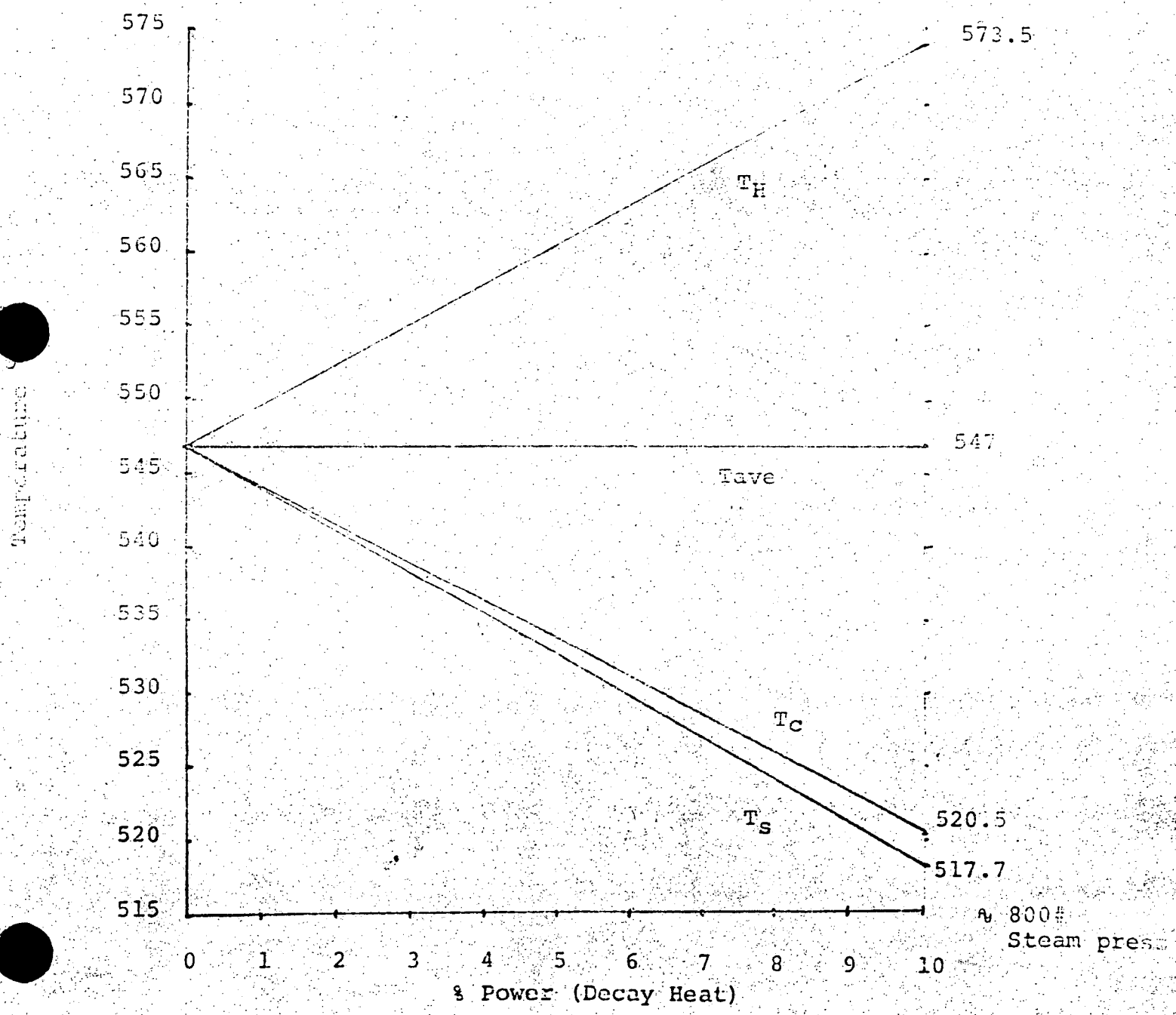
NOTE: Use in-core thermocouple temperatures monitored via Prodac: Digital display or trend typewriter.

Temperatures may also be monitored on the thermocouple readout panel if Prodac is not available.

4.6 Determine that convection heat transfer is taking place by observing in-core temperature stabilized or decreasing. Removal of heat will be evident by steam dump actuation maintained.

NOTE: Initial in-core temperatures may peak as high as 600-605°F.

4.7 Slowly increase atmospheric steam dump to decrease Tave to 547°F. The below indications should exist when no load Tave conditions are reached.



4.8 Place atmospheric steam dumps in automatic at pressure setpoint that will maintain 547° Tave. (Approximately 800#)

NOTE: 1) The above graphs indicate the approximate steady state parameters. (Match % power from decay heat and time after trip).

2) Plant parameters will respond slowly due to the convection process. $\Delta T$ is not directly a function of decay heat. It will respond to steam demand.

3) The 600°F average core exit temperature limit will provide an approximate coolant subcooling of 50°F with pressurizer pressure maintained above 2200 psia. For reactor coolant system pressure above 2000 psia, the 600°F core coolant exit temperature limit assures at least a 25°F margin to saturation.

5.0 Subsequent Actions

5.1 Observe reactor coolant system parameters to verify that the plant is configured to support and enhance natural circulation of reactor coolant.

a. Pressurizer pressure should steady above 2000 psig after initial transient behavior, and increase slowly thereafter towards normal operating pressure under the influence of the backup and steady-state heaters. The desired control band for reactor coolant system pressure is between 2200 psig and 2300 psia.

NOTE: The RC System status can be observed by reviewing the saturation margin monitor.

b. Pressurizer level should approach or exceed the programmed no-load level under the influence of the automatic pressurizer level control system. Use manual control of charging only if the automatic control fails to maintain pressure level above the low level alarm point.

5.2 When power is restored, return the plant to normal hot shutdown conditions.

5.3 Monitor RWST level to determine whether switchover to cold leg recirculation is called for.

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