BWR OWNERS GROUP

Position on NRC Regulatory Guide 1.97, Revision 2

XA

July 1982

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DISCLAIMER

The positions reported herein are consensus responses to the requirements of NRC Regulatory Guide 1.97, Revision 2, December 1980, and as such do not necessarily express in every particular the several positions of the participating utilities.

CONTENTS

Changes have been made to the following pages:

1, 12, 17, 18, 19, 20.

1. INTRODUCTION

Following the publication of Regulatory Guide 1.97, Revision 2, by the U. S. Nuclear Regulatory Commission in December 1980, the BWR Owners Group (BWROG) established a committee to review and evaluate the regulatory positions described therein.[1]

The intent of RG 1.97 is to ensure that all light-water-cooled nuclear power plants are instrumented as necessary to measure certain prescribed variables and systems during and after an accident. The principal purpose of the BWROG RG 1.97 Committee was to evaluate the safety effects and the feasibility of implementing the proposed regulatory positions--particularly those defined in Table 1, RG 1.97.

Twenty-four (24) domestic and two (2) foreign utilities supported the Committee's efforts. Seventeen (17) of these utilities provided representatives to serve on the committee. A subcommittee of the RG 1.97 committee was formed (Feb. 1982) to address the issue of inadequate Core cooling (ICC) detection.

Meetings of the committee commenced in April 1981 and continued through July 1982. The sponsoring utilities and their representatives who served on the BWROG RG 1.97 Committee are identified at the end of this section.

The committee's work was devoted primarily to discussions of specific task assignments, to presentations of committee- and contractor-generated data related to RG 1.97 requirements, and to the formulation of recommendations based on the committee's reviews and analyses. Besides conducting its own studies, the committee contracted other analytical work to Roy & Associates, Inc.; S. Levy, Inc.; and the General Electric Company.

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[1]As used throughout this report, RG 1.97 refers to RG 1.97, Revision 2, December 1980.

A summary statement of the Owners Group position relative to RG 1.97 requirements is presented in Sec. 2; some proposed Type A variables, which are unspecified in RG 1.97, are defined in Sec. 3; a detailed Owners's position statement on a variable-by-variable basis is provided in Sec. 4; and abstracts of the supporting analyses and studies are contained in Sec. 5. Pertinent contractor reports, a copy of Table 1 from RG 1.97, and a list of abbreviations are presented in the appendices.

Sponsoring Utilities

The sponsoring utilities of the BWROG RC 1.97 Committee, their assigned contacts or committee members, and consultants are identified below.

Committee Membership

(Names of the working members of the committee are in italics.)

Boston Edison Company
RICH ST. ONGE; *JERRY KITKOWSKI*

Cincinnati Gas & Electric Company
WILLIAM COOPER; ROGER THONEY

Cleveland Electric Illuminating Company
RAY TANNEY

Detroit Edison Company
JOHN GREEN

Georgia Power Company
PAUL HERRMANN (ICC chairman) (from Southern Company Services Inc.)

Gulf States Utility Company
MATEE RAHMAN; *PHILLIPS PORTER*

Iowa Electric Light and Power Company
YOSSEF (YOSI) BALAS (Chairman)

Jersey Central Power & Light Company
JAMES CHARDOS; PAUL PROCACCI; ABDUL R. BAIG

Long Island Lighting Company
JOHN RIGERT

Mississippi Power & Light Company
SAM HOBBS; *RUFUS BROWN*

Northeast Utilities
MARIO BLANCAFLOR

Northern Indiana Public Service Company
ADAM SHAHBAZI

Pennsylvania Power & Light Company
JOHN BARTOS; DAN CARDINOBE

Philadelphia Electric Company
WES BOWERS; RICK OGITIS

Power Authority of the State of New York
G. RANGARAO; J. STREET

Public Service Electric and Gas Company
RICHARD O'CONNELL

Tennessee Valley Authority
KATHRYN ASHLEY; *ROBERT BOLLINGER*

Washington Public Power Supply System
ARUN JOSHI; *BUD HUNTINGTON*

<u>Supporting Utilities</u>

Carolina Power & Light Company

Centrales Nucleares Del Norte (S.A.)

Commonwealth Edison Company

Ente, Nazionale per l' Energia Elettrica

Illinois Power Company

Nebraska Public Power District

Niagara Mohawk Power Corporation

Northern States Power Company

EPRI/NSAC
C. Dan Wilkinson, program manager (replaced by Robert Kubik for report coordination in Feb. 1982)

<u>Consultants</u>

General Electric Company

S. Levy, Inc.

Roy and Associates

2. BWR OWNERS GROUP POSITION STATEMENT

The BWROG position on NRC Regulatory Guide 1.97, Revision 2, is presented in the following statement. The statement reflects the intent of the regulatory positions set forth in RG 1.97 but includes alternatives and deviations that relate to specific instrumentation requirements and to the particulars of their implementation.

The statements that follow in this section are general positions on the requirements specified in the designated paragraphs of RG 1.97. A detailed position statement on a variable-by-variable basis is presented in Sec. 4, and supplementary data are provided in Sec. 5 and in the appendices.

General Position Statement

BWROG concurs with the intent of RG 1.97, Revision 2. The intent of the regulatory guide is to ensure that necessary and sufficient instrumentation exists at each nuclear power station for assessing plant and environmental conditions during and following an accident, as required by 10 CFR Part 50, Appendix A and General Design Criteria 13, 19, and 64. Implementation of RG 1.97 requirements is recommended except in those instances in which deviations from the letter of the guide are justified technically and when they can be implemented without disrupting the general intent of the Guide.

In assessing RG 1.97, the Owners Group has drawn upon information contained in several applicable documents, such as ANS 4.5, NUREG/CR-2100, and the BWROG Emergency Procedures Guidelines, and on data derived from other analyses and studies. The Owners Group believes that literal compliance with the provisions of the guide, because of their specific nature, is not appropriate. Some RG 1.97 requirements call for excessive ranges or inappropriate categories. Other requirements could adversely affect operator judgment under certain conditions. For example, research by S. Levy, Inc., shows that core thermocouples will provide ambiguous information to BWR operators. The Owners Group intends to follow the criteria used by the NRC for establishing Category 1, 2, and 3 instruments, although it should be noted that Category 2 instruments could vary widely between utilities, because of various plant-unique features.

The following Owners Group compliance statement is applicable to the regulatory positions defined in RG 1.97, Revision 2 (the paragraph numbers cited correspond to those in RG 1.97).

1. Accident-Monitoring Instrumentation

Par. 1.1: The BWR Owners Group concurs with this definition.

Par. 1.2: The BWR Owners Group concurs with this definition.

Par. 1.3: Instruments used for accident monitoring to meet the provisions of RG 1.97 shall have the proper sensitivity, range, transient response, and accuracy to ensure that the control room operator is able to perform his role in bringing the plant to, and maintaining it in, a safe shutdown condition and in assessing actual or possible releases of radioactive material following an accident. Each utility shall assess its plant accident-monitoring instrumentation system.

Accident-monitoring instruments that are required to be environmentally qualified will be qualified to the requirement of NUREG-0588 and Memorandum and Order CLI-80-21. The seismic qualification of instruments will be based on individual assessments performed by each utility.

Each plant will comply with the quality assurance requirements, using its approved quality assurance program, as described in the FSAR or elsewhere. This would ensure that accident-monitoring instruments comply with the applicable requirements of Title 10 CFR 50, Appendix B.

Each plant program for periodic checking, testing, calibrating, and calibration verification of accident-monitoring instrument channels (RG 1.118) shall be in accordance with the utility's commitment, as specified in the FSAR, or elsewhere.

Par. 1.3.1: A third channel of instrumentation for Category 1 instruments will be provided only if a failure of one accident-monitoring channel results in information ambiguity that would lead operators to defeat or fail to accomplish a required safety function, and if one of the following measures cannot provide the information:

1. Cross-checking with an independent channel that monitors a different variable bearing a known relationship to the variable being monitored.

2. Providing the operator with the capability of perturbing the measured variable to determine which channel has failed by observing the response on each instrument.

3. The use of portable instrumentation for validation.

Category 1 instrument channels, which are designated as being part of a Class IE system, will meet the more stringent design requirements of either the system or the regulatory guide.

The requirements for physical independence of electrical systems (RG 1.75) shall be based on each plant's commitments in the FSAR, or elsewhere.

Par. 1.3.2: The BWR Owners Group concurs with the regulatory position for Category 2 instrumentation, except as modified by Par. 1.3 above.

Par. 1.3.3: The BWR Owners Group concurs with the regulatory position for Category 3 instrumentation.

Par. 1.4: To assist the control room operator, identification of instruments designated as Categories 1 and 2 for variable types A, B, and C should be made with due consideration of human factors engineering. This position is taken to clarify the intent of RG 1.97, which specified that these

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Par. 1.4: To assist the control room operator, identification of instruments designated as Categories 1 and 2 for variable types A, B, and C should be made with due consideration of human factors engineering. This position is taken to clarify the intent of RG 1.97, which specified that these instruments be easily discerned for use during accident conditions (see Issue 1, Sec. 5).

Par. 1.5: The BWR Owners Group concurs with the regulatory position taken in this section, except as modified by Par. 1.3 above.

Par. 1.6: It is the position of BWROG that in terms of accident monitoring at a BWR facility, Table 1 of RG 1.97 does not represent a minimum number of variables and does not necessarily represent correct variable ranges or instrumentation categories.

Each BWR facility shall assess its compliance with the intent of RG 1.97 by establishing a list of accident-monitoring variables applicable to its own plant. The classification of instrumentation used to measure the variables as Category 1, 2, or 3 shall be in compliance with the intent and method used in RG 1.97.

The BWR Owners Group position on the implementation of each variable described in Table 1 of RG 1.97 and in other applicable documents is presented in Sec. 4.

2. Systems Operation Monitoring and Effluent Release Monitoring Instrumentation

The BWR Owners Group position stated in Par. 1.3 above is applicable to the Type D and E variables described in RG 1.97.

Par. 2.1: The BWR Owners Group concurs with these definitions.

Par. 2.2: The BWR Owners Group concurs with this regulatory position.

Par. 2.3: The BWR Owners Group concurs with this regulatory position.

Par. 2.4: The BWR Owners Group concurs with this regulatory position.

Par. 2.5: The BWR Owners Group position as stated in Par. 1.6 above is applicable to this regulatory position.

Implementation of Design Changes

The BWR Owners Group recommends that the implementation into each plant design of additional design changes, as required by RG 1.97, be integrated with the implementation of other control room design improvements.

A relationship exists between identifying accident-monitoring variables, developing operating procedures, reviewing control room human factors engineering, and incorporating design changes into the plant. BWROG believes that an integrated approach precludes the use of a specific implementation date for all BWR plants. In this regard, the Owners Group recommends that implementation dates should be scheduled on a plant-by-plant basis.

3. PROPOSED TYPE A VARIABLES

Regulatory Guide 1.97, Revision 2, designates all Type A variables as plant-specific, thereby defining none in particular. The Guide defines Type A variables as

> Those variables to be monitored that provide primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events.

Regulatory Guide 1.97 defines primary information as "information that is essential for the direct accomplishment of the specified safety functions." Variables associated with contingency actions that may be identified in written procedures are excluded from this definition of primary information.

As part of their review of RG 1.97, the BWR owners undertook the task of developing and analyzing a group of variables that were determined to be potential candidates for inclusion in RG 1.97 as specific Type A variables. The variables identified by the Owners Group are generic in nature, and the applicability of a given variable to a particular facility should be determined on an individual utility basis.

In the summary that follows, two groups of variables are defined: (1) proposed Type A variables and (2) potential Type A variables. The variables listed are based on the BWR Owners Group Emergency Procedure Guidelines (EPG's). Although all of the operator actions specified below may not be required to ensure that safety systems fulfill their safety functions in terms of design-basis events, they are nonetheless included in the interest of completeness.

Variables Identified as Type A

(The variables listed here are also included in the tabulation of Sec. 4.)

Variable A1. RPV Pressure

Operator action: (1) Depressurize RPV and maintain safe cooldown rate by any of several systems, such as main turbine bypass valves, isolation condenser, HPCI, RCIC, and RWCU: (2) initiate isolation condenser; (3) manually open one SRV to reduce pressure to below SRV setpoint if any SRV is cycling.

Safety function: (1) Core cooling; (2) maintain reactor coolant system integrity.

Variable A2. RPV Water Level

Operator action: Restore and maintain RPV water level.

Safety function: Core cooling

Variable A3. Suppression Pool Water Temperature

Operator action: (1) Operate available suppression pool cooling system when pool temperature exceeds normal operating limits; (2) scram reactor if temperature reaches limit for scram; (3) if suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the corresponding limit; and (4) attempt to close any stuck-open relief valve.

Safety function: (1) Maintain containment integrity and (2) maintain reactor coolant system integrity.

Variable A4. Suppression Pool Water Level

Operator action: Maintain suppression pool water level within normal operating limits: (1) transfer RCIC suction from the condensate storage tank (CST) to the suppression pool in the event of high suppression-pool level; and (2) if suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below corresponding limit.

Safety function: Maintain containment integrity.

Variable A5. Drywell Pressure

Operator action: Control primary containment pressure by any of several systems, such as containment pressure control systems, suppression pool sprays, drywell sprays.

Safety function: (1) Maintain containment integrity and (2) maintain reactor coolant system integrity.

Potential Type A Variables

(The following is a list of possible Type A variables to be determined at each plant; they are not included in Sec. 4.)

Variable 1. Condensate Storage Tank Level

Operator action: Transfer HPCI or RCIC suction or both from CST to suppression pool.

Discussion: NRC has recommended automatic suction transfer for HPCI and RCIC. This variable is not a Type A variable if the automatic suction transfer is installed.

Variable 2. Emergency Diesel Generator (EDG) Load

Operator action: Control loading of the EDG's.

Discussion: Some plants have a planned manual action to verify the loading on the EDG's before any other safety-related loads are added. If no planned action is necessary, this variable is not type A.

Variable 3. Reactor Building Flood Level

Operator action: Initiate pump-back of sump to suppression pool.

Discussion: Water can accumulate in the reactor building during long-term cooling with any postulated leakage. The flood-level indication would alert the operator to a problem, but this indication is an aid to and not the accomplishment of a safety function.

Variable 4. Drywell Temperature

Operator action: Initiate sprays, reactor water level compensation.

Discussion: This variable may be needed for reactor-water-level compensation. Note: Although the EPG's mention drywell temperature, the drywell pressure is the key variable for containment integrity; drywell temperature is a secondary consideration. This issue will be addressed by the ICC subcommittee.

Variable 5. Suppression Pool Pressure

Operator action: Initiate suppression pool sprays.

Discussion: The suppression pool sprays are not used in safety analysis. Although the EPG's use suppression pool pressure to initiate suppression pool spray, containment pressure may be used to approximate the suppression pool pressure.

Variable 6. Oxygen or Hydrogen Concentration

Operator action: If containment atmosphere approaches the combustible limits, initiate combustible gas control systems. Oxygen for inerted and hydrogen for non-inerted containments.

Safety function: Maintain containment integrity.

4. PLANT VARIABLES FOR ACCIDENT MONITORING

BWROG positions on the implementation of the variables listed in Table 1 of RG 1.97 and on the assignment of design and qualification criteria for the instrumentation proposed for their measurement is summarized in the tabulation that follows.

In brief, the measurement of the five variable types provides the following kinds of information to plant operators during and after an accident: (1) Type A--primary information, on the basis of which operators take planned specified manually controlled actions; (2) Type B--information about the accomplishment of plant safety functions; (3) Type C--information about the breaching of barriers to fission product release; (4) Type D--information about the operation of individual safety systems; and (5) Type E--information about the magnitude of the release of radioactive materials.

The three categories shown for the variables define the design and qualification criteria for the instrumentation that is to be used for their measurement. Category 1 imposes the most stringent requirements; Categories 2 and 3 impose progressively less stringent requirements.

The categories are also related (in RG 1.97) to "key variables." Key variables are defined differently for the different variable types. For Type B and Type C variables, the key variables are those variables that most directly *indicate the accomplishment of a safety function;* instrumentation for these key variables is designated Category 1. Key variables that are Type D variables are defined as those variables that most directly *indicate the operation of a safety system;* instrumentation for these key variables is usually Category 2. And key variables that are Type E variables are defined as those variables that most directly *indicate the release of radioactive material;* instrumentation for these key variables is also usually Category 2. A complete discussion of the variable types and instrumentation design criteria is presented in RG 1.97.

It should be noted that the Type A variables listed below are being proposed for inclusion in RG 1.97 on the basis of analyses conducted by the Owners Group (Sec. 3). Table 1 of RG 1.97 designates all Type A variables as plant specific and thus defines none in particular.

The variables are listed here in the same sequence used in Table 1, RG 1.97; however, for convenience in cross-referencing entries and supporting data, the variables are designated by letter and number. For example, the sixth B-type variable listed in RG 1.97 is denoted here as variable B6. (A copy of Table 1 from RG 1.97 is provided in Appendix C.)

BWROG's position is shown for each variable and for its instrumentation design criteria and category. (The letters *OG* and *RG* preceding the category numbers identify the Owners Group and RG 1.97, respectively.) In general, there are three kinds of responses or recommendations: (1) implement the variable and required instrumentation in accordance with the regulatory position stated in Table 1, RG 1.97 (2) implement, with qualifying exceptions or revisions; and (3) do not implement.

As necessary, the positions of BWROG are elaborated or substantiated in the Supplementary Analyses section (Sec. 5) or in supplementary documents provided in the appendixes. *Note that references to the data in Sec. 5 are made by citing the issue numbers that appear in the upper corner of the pages in Sec. 5.*

Type A Variables

The following Type A variables are recommended by the Owners Group (OG) for inclusion in RG 1.97 as type A. (See Sec. 3.)

A1. Reactor pressure (OG Category 1)
RECOMMENDATION: Implement. See B6, C4, and C9.

A2. Coolant level in reactor (OG Category 1)
RECOMMENDATION: Implement. See B4.

A3. Suppression pool water temperature (OG Category 1)
RECOMMENDATION: Implement. See D6.

A4. Suppression pool water level (OG Category 1)
RECOMMENDATION: Implement. See C7 and D5.

A5. Drywell pressure (OG Category 1)
RECOMMENDATION: Implement. Type A for plants without autostarting drywell spray. See B7, B9, C8, C10, and D4.

Type B Variables

Reactivity Control

B1. Neutron Flux (OG Category 2; RG Category 1)
RECOMMENDATION: Implement, but as Category 2 with alarm and reduced range, in accordance with data in Issue 2.

B2. Control Rod Position (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

B3. RCS Soluble Boron Concentration (sample) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

Core Cooling

B4. Coolant Level in Reactor (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See A2, C3, and Issue 3.

B5. BWR Core Thermocouples (RG Category 1)
RECOMMENDATION: Do not implement. See C3 and Appendix A.

Maintaining Reactor Coolant System Integrity

B6. RCS Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See A1, C4, C9, and Issue 3.

B7. Drywell Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See B9, C8, C10, and D4.

B8. Drywell Sump Level (OG Category 3; RG Category 1)
RECOMMENDATION: Implement as Category 3. See C6 and Issue 4.

Maintaining Containment Integrity

B9. Primary Containment Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See B7, C8, C10, and D4.

B10. Primary Containment Isolation Valve Position (excluding check valves) (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. Redundant indication is not required on each redundant isolation valve.

Type C Variables

Fuel Cladding

C1. Radioactivity Concentration or Radiation Level in Circulating Primary Coolant (RG Category 1)
RECOMMENDATION: Do not implement. See Issue 5.

C2. Analysis of Primary Coolant (gamma spectrum) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

C3. BWR Core Thermocouples (RG Category 1)
RECOMMENDATION: Do not implement. See B5 and Appendix A.

Reactor Coolant Pressure Boundary

C4. RCS Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See A1, B6, and C9.

C5. Primary Containment Area Radiation (OG Category 3; RG Category 3)
RECOMMENDATION: Implement. See E1.

C6. Drywell Drain Sumps Level (identified and unidentified leakage) (OG Category 3; RG Category 1)
RECOMMENDATION: Implement as Category 3. See B8 and Issue 4.

C7. Suppression Pool Water Level (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See A4 and D5.

C8. Drywell Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See B7, B9, C10, and D4.

Containment

C9. RCS Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See A1, B6, and C4.

C10. Primary Containment Pressure (OG Category 1; RG Category 1)
RECOMMENDATION: Implement. See B7, B9, C8, and D4.

C11. Containment and Drywell H_2 Concentration (OG Category 1; RG Category 1)
RECOMMENDATION: Implement

C12. Containment and Drywell Oxygen Concentration (for inerted containment plants) (OG Category 1; RG Category 1)
RECOMMENDATION: Implement.

C13. Containment Effluent Radioactivity--Noble Gases (from identified release points including Standby Gas Treatment System Vent) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

C14. Radiation Exposure Rate (inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located) (RG Category 2)
RECOMMENDATION: Do not implement. See E2, E3, and Issue 6.

C15. Effluent Radioactivity--Noble Gases (from buildings as indicated above) (OG Category 2; RG Category 2)
RECOMMENDATION: Implement

Type D Variables

Condensate and Feedwater System

D1. Main Feedwater Flow (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

D2. Condensate Storage Tank Level (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

Primary Containment-Related System

D3. Suppression Spray Flow (RG Category 2)
RECOMMENDATION: Do not implement. See Issue 7.

D4. Drywell Pressure (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See B7, B9, C8, and C10.

D5. Suppression Pool Water Level (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See A4 and C7.

D6. Suppression Pool Water Temperature (OG Category 2; RG Category 2)
RECOMMENDATION: Implement.
Both local and bulk temperature. See A3.

D7. Drywell Atmosphere Temperature (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See Issue 8.

D8. Drywell Spray Flow (RG Category 2)
RECOMMENDATION: Do not implement. See Issue 7.

Main Steam System

D9. Main Steamline Isolation Valves' Leakage Control System Pressure (OG Category 2; RG Category 2)
RECOMMENDATION: Implement if system is part of plant design.

D10. Primary System Safety Relief Valve Position, Including ADS or Flow Through or Pressure in Valve Lines (OG Category 2; RG Category 2)
RECOMMENDATION: Implement

Safety Systems

D11. Isolation Condenser System Shell-Side Water Level (OG Category 2; RG Category 2)
RECOMMENDATION: Implement if system is part of plant design.

D12. Isolation Condenser System Valve Position (OG Category 2; RG Category 2)
RECOMMENDATION: Implement if system is part of plant design.

D13. RCIC Flow (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See Issue 9.

D14. HPCI Flow (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See Issue 9.

D15. Core Spray System Flow (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See Issue 9.

D16. LPCI System Flow (OG Category 2; RG Category 2)
RECOMMENDATION: Implement. See Issue 9.

D17. SLCS Flow (OG Category 3; RG Category 2)
RECOMMENDATION: Implement as Category 3. Await ATWS resolution. See Issue 9.

D18. SLCS Storage Tank Level (OG Category 3; RG Category 2)
RECOMMENDATION: Implement as Category 3. Await ATWS resolution. See Issue 10.

Residual Heat Removal (RHR) Systems

D19. RHR System Flow (OG Category 2; RG Category 2)
RECOMMENDATION: Implement

D20. RHR Heat Exchanger Outlet Temperature (OG Category 2; RG Category 2)
RECOMMENDATION: Implement

Cooling Water System

D21. Cooling Water Temperature to ESF System Components (OG Category 2; RG Category 2)
RECOMMENDATION: Interpret as main system flow and implement.

D22. Cooling Water Flow to ESF System Components (OG Category 2; RG Category 2)
RECOMMENDATION: Interpret as main system flow and implement.

Radwaste Systems

D23. High Radioactivity Liquid Tank Level (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

Ventilation Systems

D24. Emergency Ventilation Damper Position (OG Category 2; RG Category 2)
RECOMMENDATION: Interpret as meaning dampers actuated under accident conditions and whose failure could result in radioactive discharge to the environment. Control room damper position should be indicated. Implement.

Power Supplies

D25. Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic) (OG Category 2; RG Category 2)
RECOMMENDATION: Implement; on-site sources only.

(Note: The addition of the following D-type variables is recommended by BWROG; see Issue 11, Sec. 5.)

D26. Turbine Bypass Valve Position (OG Category 3)
RECOMMENDATION: Add to RG 1.97. See Issue 11.

D27. Condenser Hotwell Level (OG Category 3)
RECOMMENDATION: Add to RG 1.97. See Issue 11.

D28. Condenser Vacuum (OG Category 3)
RECOMMENDATION: Add to RG 1.97. See Issue 11.

D29. Condenser Cooling Water Flow (OG Category 3)
RECOMMENDATION: Add to RG 1.97. See Issue 11.

D30. Primary Loop Recirculation Flow (OG Category 3)
RECOMMENDATION: Add to RG 1.97. See Issue 11.

Type E Variables

Containment Radiation

E1. Primary Containment Area Radiation--High Range (OG Category 1; RG Category 1)
RECOMMENDATION: Implement in accordance with NUREG-0737 commitment. See C5.

E2. Reactor Building or Secondary Containment Area Radiation (RG Category 2 for Mark I and II containments; OG Category 1 and RG Category 1 for Mark III containments)
RECOMMENDATION: Do not implement for Mark I and II containments. Implement for Mark III containments. See C14, E3, and Issue 12.

Area Radiation

E3. Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety) (OG Category 3; RG Category 2)
RECOMMENDATION: Implement as Category 3, using existing instrumentation. See C14, E2, and Issue 13.

Airborne Radioactive Materials Released from Plant

E4. Noble Gases and Vent Flow Rate (OG Category 2; RG Category 2)
RECOMMENDATION: Implement

E5. Particulates and Halogens (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

Environs Radiation and Radioactivity

E6. Radiation Exposure Meters (continuous indication at fixed locations)
RECOMMENDATION: Deleted. See NRC errata of July 1981.

E7. Airborne Radiohalogens and Particulates (portable sampling with on-site analysis capability) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

E8. Plant Environs Radiation (portable instrumentation) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement (portable equipment)

E9. Plant and Environs Radioactivity (portable instrumentation) (OG Category 3; RG Category 3)
RECOMMENDATION: Implement (portable equipment)

Meteorology

E10. Wind Direction (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

E11. Wind Speed (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

E12. Estimation of Atmospheric Stability (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

Accident-Sampling Capability (Analysis Capability On-Site)

E13. Primary Coolant and Sump (OG Category 3--Primary Coolant only; RG Category 3)
RECOMMENDATION: Implement Primary Coolant. Do not implement Sump. See Issue 14.

E14. Containment Air (OG Category 3; RG Category 3)
RECOMMENDATION: Implement

5. SUPPLEMENTARY ANALYSES

These supplementary analyses support positions cited in Sec. 2 (Issue 1) and Sec. 4 (Issues 2-14).

Contents

ISSUE 1. INSTRUMENT IDENTIFICATION

Issue Definition

Regulatory Guide 1.97 specifies, in par. 1.4.b, the following: "The instruments designated as Types A, B, and C and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions."

Discussion

The objective of this regulatory position is the achievement of good human factors engineering in the presentation of information to the control room operator. This objective is best achieved by evaluating current practices and procedures that provide for identifying instruments in a manner that aids the operator; redundant labels would tend to distract the operator and cause confusion. The Control Room Design Review of the BWR Owners Group has the charter to provide a basis for assuring proper identification of accident instrumentation with consideration for current information for safe plant shutdown, operational training, and procedures.

Conclusion

Instruments designated as Categories 1 and 2 for monitoring variable types A, B, and C should be identified in such a manner as to optimize applicable human factors engineering and presentation of information to the control room operator. This position is taken to clarify the intent of RG 1.97, which specifies that these instruments be easily discerned for use during accident conditions.

ISSUE 2. VARIABLE B1

B1: Neutron Flux

Issue Definition

The measurement of neutron flux is specified as the key variable in monitoring the status of reactivity. Neutron flux is classified as a Type B variable, Category 1. The specified range is 10^{-6} percent to 100 percent full power (SRM, APRM). The stated purpose is "Function detection; accomplishment of mitigation."

Discussion

The lower end of the specified range, 10^{-6} percent full power, is intended to allow detection of an approach to criticality by some undefined and noncontrollable mechanism after shutdown.

In attempting to analyze the performance of the neutron-flux monitoring systems, a scenario was postulated to obtain the required approach to criticality. Basically, it assumes an increase in reactivity from loss of boron in the reactor water.

The accident scenario incorporates the following factors:

1. The control rods fail (completely or partially) to insert, and the operator actuates the standby liquid control system (SLCS).

2. The SLCS shuts the reactor down.

3. A leak in the primary system results in an outgo of borated water and its replacement by water that contains no boron.

4. A range of leak rates up to 20 gpm was considered (see Table 1).

Calculations were made to evaluate the rise in neutron population as a function of different leak rates. The calculations were made for a shutdown neutron level of 5×10^{-8} percent of full power. The choice of 5×10^{-8} is based on measurements at two nuclear plants. The shutdown level was assumed to have a negative reactivity of 10 dollars, an assumption that is representative of a shutdown with all rods inserted. The results of the calculations are presented in Table 1. The numbers in the table refer to the time in hours required to increase the flux by 1 decade. For example, with a leak of 5 gpm, it takes 100 hr to increase the power from 5×10^{-8} percent to 5×10^{-7} percent, and 10 hr to increase it from 5×10^{-7} percent to 5×10^{-6} percent.

The reactor is subcritical and the neutron level is given by

$$\text{Neutron level} = S \times M,$$

where S is the source strength and M is the multiplication, which is given by

$$M = 1/(1 - k).$$

For $k = 0.9$, M is 10; for $k = 0.99$, M is 100 and so forth. For criticality, the denominator approaches 0, as k approaches 1.0. Thus, the calculation model used the above equation to calculate relative neutron flux levels for a subcritical reactor until the reactor was near critical; then the critical equation of power with excess reactivity was used. Reactor power is directly proportional to neutron level.

The increase in reactivity toward criticality can be turned around by actuating the SLCS. *It is assumed that operating procedures provide for refilling the SLCS tank soon after its actuation.* A second actuation of the SLCS would cause a decrease in reactivity because of the high concentration of boron in the injected SLCS fluid relative to that in the leaking fluid (nominally 400 ppm). The sensitivity of the detector must allow adequate time for the operator to act. Ten minutes is considered sufficient time for operator action for accident prevention and mitigation.

Table 1 shows that the detector sensitivity (i.e., lower range) requirement is a function of leak rate and therefore of reactivity-addition rate. On the basis of a 20-gpm leak rate, Table 1 shows that a detector that is on scale within 3 decades of the shutdown power would allow 0.18 hr (10.8 min) for operator action before reactor power increased another decade. A total of 0.36 hr (21.6 min) would be available for operator action from the time the indicator comes on scale to the time reactor power reaches 0.5 percent of full power. An alarm would be provided to warn the operator when the neutron flux starts to increase beyond a plant-specific set-point.

The 20-gpm leak rate, which was assumed to continue for 27.75 hr, was used to define the sensitivity of the detector. It should be noted that the assumed leak rate, extended over the 27.75-hr period, would result in a loss of inventory so large that it could not in reality go undetected by the operator. Moreover, reactivity-addition caused by this gradual boron depletion is unlikely, since boron concentration is sampled and measured periodically. Again, the improbable 20-gpm leak rate was used only to obtain a mechanistic and conservative approach for selection of instrument sensitivity.

An absolute criterion for the lower range must include consideration of the neutron source level. The use of the neutron level 100 days after shutdown is conservative. There is high probability that conditions would be stable and controllable 2 days after the emergency shutdown, for the core-decay heat is at a low level and the boron monitoring system should be functioning by that time. The actual neutron level will vary with fuel design, fuel history, and shutdown control strength. Measurements of shutdown neutron flux (with all rods inserted) at two BWR reactors show readings of 30 to 80 counts/sec (1000 counts/sec corresponds to 10^{-8} of full power). Measurements on other BWR reactors and for different fuel histories would show some variation, but those variations would be small compared with a criterion that is concerned with units of decades.

Regulatory Guide 1.97 classifies the instrumentation for measuring a variable as Category 1 on the basis of (1) whether it is a key variable (defined in Sec. 4), and (2) its importance to safety. Neutron flux is the key variable for measuring reactivity control, thus meeting the requirement of criterion (1). The degree to which this variable is important to safety is another consideration. The large number of detectors (i.e., source-range monitors and intermediate-range monitors) that are driven into the core soon after shutdown makes it highly probable that one or more of the existing NMS detectors will be inserted. On the other hand, there is little probability that there would be, simultaneously, a need for this measurement (in terms of operator action to be taken) and an accident environment in which the NMS would be rendered inoperable. Further, the operator can always actuate the SLCS on loss of instrumentation.

Although some upgrading of the current NMS may be appropriate to improve system reliability and its ability to survive a spectrum of accidents, a rigorous Category 1 requirement is not justified when the purpose and use of the measurement are analyzed as they relate to the criterion of "importance to safety." A Category 2 classification of this variable fully meets the intent of RG 1.97.

Four alternative design approaches to meeting the neutron flux requirements of RG 1.97 have been identified. All four alternatives would provide indication over the range recommended by BWROG, using state-of-the-art electronics for displaying the detector reading. A particular utility can choose a suitable alternative, based on its own design evaluation. The principal features of the four alternatives are presented below.

Alternative 1. The first alternative provides for upgrading two or more of the source-range monitors (SRM's). The upgrading includes the connecting cable inside the drywell and the power source for the SRM drives. At least two SRM's would have dual roles of accident instrumentation and normal start-up; these two SRM's would be withdrawn a lesser distance from the core than the SRM in the current design. It is estimated that in its fully withdrawn position, the current SRM will detect about 10^{-3} or 10^{-5} percent of full power. This sensitivity can be increased by using a withdrawn position that is less than the present 2-2.5 ft from the core. A withdrawn position that produces 10 percent depletion in 5 years was used as a guide to the *maximum* allowed burn-up of the sensor. This position below the core would give the SRM a detection capability of about 2×10^{-7} percent of full power. The SRM drives need not be upgraded, because the upgraded detector system would be adequate, even if the drive did not move the SRM detector. (An upgraded power source for the drives improves the probability of insertion.) The success of this alternative--which uses the four SRM's for normal start-up--depends on a design modification to accommodate the new cable (the key concern is the flexibility of the cable, for the detector moves about 10 ft; this movement is accommodated in the cable loop) and on the design of a limit switch or a detent mechanism to hold the drive tube in the required intermediate position.

Alternative 2. The second alternative is to replace two or more SRM systems with upgraded systems. The full SRM system, including the drives, would be upgraded. This approach would require input from a potential equipment supplier in order to estimate schedules, cost, and overall effect of the upgrading. Whereas the first alternative uses upgraded cables and power supply (which are commercially available), this approach would require additional engineering to achieve an upgraded drive system as well. A Category 1 drive system is a developmental item.

Alternative 3. In the third alternative, fixed in-core detectors are used. The system uses SRM-type detectors as stationary detectors that are positioned close enough (as discussed above) to the core to meet the lower range requirements. New cables are needed to meet the requirements of the accident environment. This system would provide dedicated "accident monitors" in two of the intermediate-range monitor (IRM) tubes or in two local-power range-monitor (LPRM) tubes. It may be feasible to put five detectors in the LPRM tube or, if space is limited, the bottom detector of the LPRM string could be replaced with the "accident" detector. With this approach the four movable SRM's would continue to be available for normal functions.

Alternative 4. In the final alternative, out-of-core detectors, which are being qualified for use in pressurized water reactors (PWR's), are used. Considerations of this ongoing PWR qualification program for Category 1 instrumentation and the lack of any effect on the current neutron monitoring system (NMS) make this alternative an attractive one. The key question is whether these out-of-core detectors can meet the lower range requirement, for the detectors are positioned outside the RPV shield wall. A test is needed to demonstrate that the neutron count at this location is adequate. Based on calculations of neutron flux made for a BWR at full power (see Fig. 1) and on current detector design practices, the out-of-core detector may be feasible. Other effects, such as attenuation by water that is at a lower temperature (than the full-power operating temperature) and by boron in the water, need to be considered.

Conclusion

A range from 5×10^{-5} percent of full power (within 3 decades of the neutron flux level 100 days after shutdown) to 100 percent of full power is recommended. An alarm is also recommended that would alert the operator of a rise in neutron flux. It is concluded that a Category 2 classification is responsive to the intent of RG 1.97, as are the four alternatives, provided that the design program resolves the specific design concerns identified in the Discussion.

TABLE 1. RELATIVE NEUTRON FLUX VERSUS TIME[a]

Percent of power	Leakage rate, gpm (ramp rate, ¢/min)[b]					
	1(0.03)		5(0.15)		20(0.60)	
	Σ	Δ	Σ	Δ	Σ	Δ
5×10^{-8}	-555	500	-111	100	-27.75	25
5×10^{-7}	-55	50	-11	10	-2.75	2.5
5×10^{-6}	-5	5	-1	1	-0.25	0.25
5×10^{-5}	0		0		0	
5×10^{-4}	0.8	0.8	0.36	0.36	0.18	0.18
5×10^{-3}	1.33	0.53	0.51	0.15	0.25	0.07
5×10^{-2}	1.59	0.26	0.62	0.11	0.31	0.06
5×10^{-1}	1.80	0.21	0.72	0.10	0.36	0.05
5×10^{0}	1.89	0.09	0.80	0.08	0.40	0.04

[a]Shutdown flux = 5×10^{-8} percent of power.

[b]Σ = total number of hours; Δ = hours for neutron flux to increase by one decade.

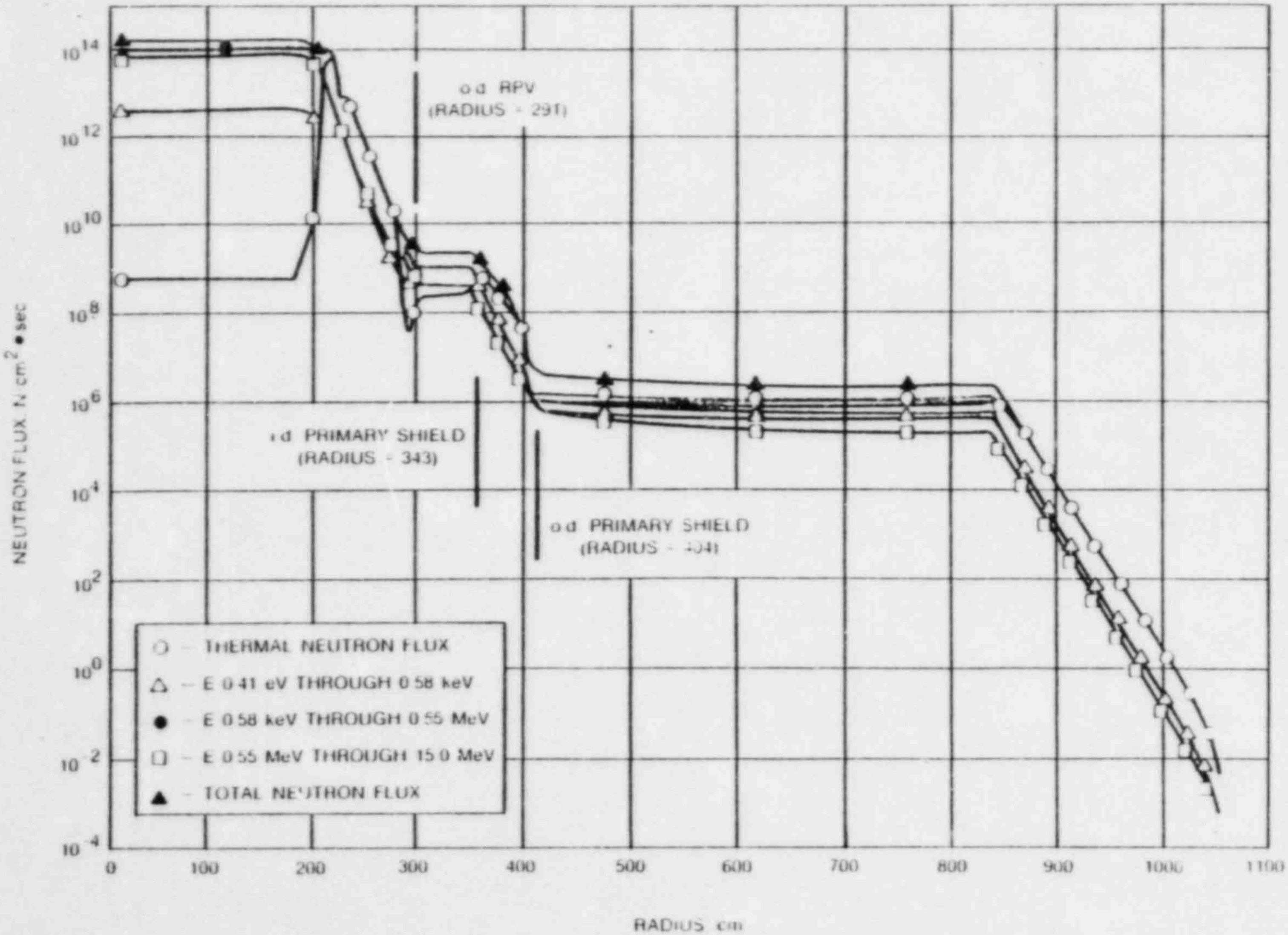

Figure 1. Radial distribution of neutron fluxes: full power.

ISSUE 3. TREND RECORDING

Issue Definition

The purpose of addressing Issue 3 is to determine which variables set forth in RG 1.97 require trend recording.

Discussion

Regulatory Guide 1.97, par. 1.3.2f, states the general requirement for trend recording as follows: "Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available for dedicated recorders." Using the BWR Owners Group Emergency Procedures Guidelines (EPG's) as a basis, the only trended variables *required* for operator action are reactor water level and reactor vessel pressure.

Conclusion

On a generic basis, only reactor water level (variable B4) and reactor vessel pressure (variable B6) *require* trend recording; however, other variables may be necessary on a plant-specific basis.

ISSUE 4. VARIABLES B8 AND C6

B8: Drywell Sump Level
C6: Drywell Drain Sumps Level

Issue Definition

Regulatory Guide 1.97 requires Category 1 instrumentation to monitor drywell sump level (variable B8) and drywell drain sumps level (variable C6). These designations refer to the drywell equipment and floor-drain tank levels. Category 1 instrumentation indicates that the variable being monitored is a key variable. In RG 1.97, a key variable is defined as ". . . that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function. . . ." The following discussion supports the BWR Owners Group alternative position that drywell sump level and drywell drain-sumps levels should be classified as Category 3 instrumentation.

Discussion

The BWR Mark I, II, and III drywells have two drain sumps. One drain is the equipment drain sump, which collects identified leakage; the other is the floor drain sump, which collects unidentified leakage.

Although the level of the drain sumps can be a direct indication of breach of the reactor coolant system pressure boundary, the indication is not unambiguous, because there is water in those sumps during normal operation. There is other instrumentation required by RG 1.97 that would indicate leakage in the drywell:

1. Drywell pressure--variable B7, Category 1
2. Drywell temperature--variable D7, Category 2

3. Primary containment area radiation--variable C5, Category 3

The drywell-sump level signal neither automatically initiates safety-related systems nor alerts the operator to the need to take safety-related actions. Both sumps have level detectors that provide only the following nonsafety indications:

1. Continuous level indication (some plants)
2. Rate of rise indication (some plants)
3. High-level alarm (starts first sump pump)
4. High-high-level alarm (starts second sump pump)

In addition, timers are used in most plants to indicate the duration of sump-pump operation and thereby permit the amount of leakage to be estimated.

Regulatory Guide 1.97 requires instrumentation to function during and after an accident. The drywell sump systems are deliberately isolated at the primary containment penetration upon receipt of an accident signal to establish containment integrity. This fact renders the drywell-sump-level signal irrelevant. Therefore, by design, drywell-level instrumentation serves no useful accident-monitoring function.

The Emergency Procedure Guidelines use the RPV level and the drywell pressure as entry conditions for the Level Control Guideline. A small line break will cause the drywell pressure to increase before a noticeable increase in the sump level. Therefore, the drywell sumps will provide a "lagging" versus "early" indication of a leak.

Conclusion

Based on the above considerations, the BWR Owners Group believes that the drywell-sump level and drywell-drain-sump level instrumentation should be classified as Category 3, "high-quality off-the-shelf instrumentation."

ISSUE 5. VARIABLE C1

C1: Radioactivity Concentration or Radiation Level in Circulating Primary Coolant

Issue Definition

Regulatory Guide 1.97 specifies that the status of the fuel cladding be monitored during and after an accident. The specified variable to accomplish this monitoring is variable C1--radioactivity concentration or radiation level in circulating primary coolant. The range is given as "1/2 Tech Spec Limit to 100 times Tech Spec Limit, R/hr." In Table 1 of RG 1.97, instrumentation for measuring variable C1 is designated as Category 1. The purpose for monitoring this variable is given as "detection of breach," referring, in this case, to breach of fuel cladding.

Discussion

The usefulness of the information obtained by monitoring the radioactivity concentration or radiation level in the circulating primary coolant, in terms of helping the operator in his efforts to prevent and mitigate accidents, has not been substantiated. The critical actions that must be taken to prevent and mitigate a gross breach of fuel cladding are (1) shut down the reactor and (2) maintain water level. Monitoring variable C1, as directed in RG 1.97, will have no influence on either of these actions. The purpose of this monitor falls in the category of "information that the barriers to release of radioactive material are being challenged" and "identification of degraded conditions and their magnitude, so the operator can take actions that are available to mitigate the consequences."

Additional operator actions to mitigate the consequences of fuel barriers being challenged, other than those based on Type A and B variables, have not been identified.

Regulatory Guide 1.97 specifies measurement of the radioactivity of the circulating primary coolant as the key variable in monitoring fuel cladding status during isolation of the NSSS. The words "circulating primary coolant" are interpreted to mean coolant, or a representative sample of such coolant, that flows past the core. A basic criterion for a valid measurement of the specified variable is that the coolant being monitored is coolant that is in active contact with the fuel, that is, flowing past the failed fuel. Monitoring the active coolant (or a sample thereof) is the dominant consideration. The post-accident sampling system (PASS) provides a representative sample which can be monitored.

The subject of concern in the RG 1.97 requirement is assumed to be an isolated NSSS that is shutdown. This assumption is justified as current monitors in the condenser off-gas and main steam lines provide reliable and accurate information on the status of fuel cladding when the plant is not isolated. Further, the post-accident sampling system (PASS) will provide an accurate status of coolant radioactivity, and hence cladding status, once the PASS is activated. In the interim between NSSS isolation and operation of the PASS, monitoring of the primary containment radiation and containment hydrogen will provide information on the status of the fuel cladding.

Conclusion

The designation of instrumentation for measuring variable C1 should be Category 3, because no planned operator actions are identified and no operator actions are anticipated based on this variable serving as the key variable. Existing Category 3 instrumentation is adequate for monitoring fuel cladding status.

ISSUE 6. VARIABLE C14

C14: Radiation Exposure Rate

Issue Definition

Variable C14 is defined in Table 1 of RG 1.97 as follows: "Radiation exposure rate (inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment), which are in direct contact with primary containment where penetrations and hatches are located." The reason for monitoring variable C14 is given as "Indication of breach."

Discussion

The use of local radiation exposure rate monitors to detect breach or leakage through primary containment penetrations is impractical and unnecessary. In general, radiation exposure rate in the secondary containment will be largely a function of radioactivity in primary containment and in the fluids flowing in ECCS piping, which will cause direct radiation shine on the area monitors. Also, because of the amount of piping and the number of electrical penetrations and hatches and their widely scattered locations, local radiation exposure rate monitors could give ambiguous indications. The proper way to detect breach of containment is by using the plant noble gas effluent monitors.

Conclusion

Using radiation exposure rate monitors to detect primary containment breach is neither feasible nor necessary. Other means of breach detection that are better suited to this function (as described above), are available. Therefore, it is the position of the BWR Owners Group that this parameter not be implemented.

ISSUE 7. VARIABLES D3 AND D8

D3: Suppression Spray Flow
D8: Drywell Spray Flow

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of suppression chamber spray (SCS) (variable D3) and drywell spray (variable D8) for monitoring the operation of the primary containment-related systems. Instrumentation for measuring these variables is designated Category 2, with a range of 0 to 110 percent of design flow. These flows relate to spray flow for controlling pressure and temperature of the drywell and suppression chamber.

Discussion

The drywell sprays can be used to control the pressure and temperature of the drywell. The residual heat removal (RHR) system flow element is used for measuring drywell flow in most designs.

The suppression pool sprays can be used to control the pressure and temperature in the suppression chamber. The operator controls pressure and temperature by adjusting suppression chamber spray flow. The RHR system flow element is used for flow indication in most designs. Some plants have a flow element in the branch line to the sprays. The suppression chamber spray operates in parallel with the drywell spray and is regulated with a throttling valve. The flow is determined by the position of the throttling valve that is in the branch line that feeds the containment spray lines. These valve positions are indicated in the control room. The effectiveness of these flows can be verified by pressure and temperature changes of the drywell and the suppression chamber.

Conclusion

The current plant designs, in conjunction with operating practice, provide for operator information that is sufficient for determining the existence of spray flows to the drywell and suppression chamber without the use of a dedicated flow-measuring instrument.

ISSUE 8. VARIABLE D7

D7: Drywell Atmosphere Temperature

Issue Definition

Regulatory Guide specifies drywell atmosphere temperature (variable D7, Category 2) as one of the key variables in monitoring individual safety systems. The temperature range is specified as 40°F to 440°F.

Discussion

The evaluation of this issue addressed requirements that call for direct operator action based on variable D7, that is, temperature and the associated variable of pressure. The BWR Emergency Procedure Guidelines (EPG's) provide guidelines for control of containment pressure and temperature. Classification of this variable should be done on a plant-specific basis with full consideration for EPG requirements.

Temperature-monitoring hardware inside the drywell may not be qualified to the accident conditions specified in RG 1.97; the primary item of concern is the cable inside the drywell.

Conclusion

BWROG recommends implementation of variable D7 requirements as specified in RG 1.97.

ISSUE 9. VARIABLES D13-D17

D13: RCIC Flow
D14: HPCI Flow
D15: Core Spray System Flow
D16: LPCI System Flow
D17: SLCS Flow

Issue Definition

Regulatory Guide 1.97 specifies flow measurements of the following systems: reactor core isolation cooling (RCIC) (variable D13), high-pressure coolant injection (HPCI) (variable D14), core spray (CS) (variable D15), low-pressure coolant injection (LPCI) (variable D16), and standby liquid control (SLC) (variable D17). The purpose is for monitoring the operation of individual safety systems. Instrumentation for measuring these variables is designated as Category 2; the range is specified as 0 to 110 percent of design flow. These variables are related to flow into the reactor pressure vessel (RPV).

Discussion

The RCIC, HPCI, and CS systems each have one branch line--the test line--downstream of the flow-measuring element. The test line is provided with a motor-operated valve that is normally closed (two valves in series in the case of the HPCI). Further, the valve in the test line closes automatically when the emergency system is actuated, thereby ensuring that indicated flow is not being diverted by the test line. Proper valve position can be verified by a direct indication of valve position.

Although the LPCI has several branch lines located downstream of each flow-measuring element, each of those lines is normally closed. Proper valve position can be verified by a direct indication of valve position.

For all of the above systems, there are valid primary indicators other than flow measurement to verify the performance of the emergency system; for example, vessel water level.

The SLC system is manually initiated. Flow-measuring devices were not provided for this system. The pump-discharge header pressure, which is indicated in the control room, will indicate SLC pump operation. Besides the discharge header pressure observation, the operator can verify the proper functioning of the SLCS by monitoring the following:

1. The decrease in the level of the boric acid storage tank

2. The reactivity change in the reactor as measured by neutron flux

3. The motor contactor indicating lights (or motor current)

4. Squib valve continuity indicating lights

5. The open/close position indicators of check valves (available in some plants)

The use of these indications is believed to be a valid alternative to SLCS flow indication.

Conclusion

The flow-measurement schemes for the RCIC, HPCI, CS, and LPCI are adequate in that they meet the intent of RG 1.97. Monitoring the SLCS can be adequately done by measuring variables other than the flow.

ISSUE 10. VARIABLE D18

D18: SLCS Storage Tank Level

Issue Definition

Regulatory Guide 1.97 lists standby liquid-control system (SLCS) storage-tank level as a Type D variable with Category 2 design and qualification criteria.

Discussion

The symptomatic Emergency Procedure Guidelines (EPG), Revision 1, as presently approved do not consider ATWS conditions; however, the EPG committee of the BWR Owners Group has been developing a draft reactivity control guideline in which procedures are described for raising the reactor water level based on the amount of boron injected into the vessel, as indicated by the SLC tank level. Additionally, the operator is required to trip the SLC pumps before a low SLC tank level is reached, thereby preventing damage to the pumps that would render them useless for future injections during the scenario.

Regarding the instrumentation category requirement for variable D18, RG 1.97 indicates that it is a key variable in monitoring SLC system operation. Regulatory Guide 1.97 also states that in general, key Type D variables be designed and qualified to Category 2 requirements.

In applying these requirements of the Guide to this instrumentation, the following are noted:

1. The current design basis for the SLCS assumes a need for an alternative method of reactivity control without a concurrent loss-of-coolant accident or high-energy line break. The environment in which the SLCS instrumentation must work is therefore a "mild" environment for qualification purposes.

2. The current design basis for the SLCS recognizes that the system has an importance to safety that is less than the importance to safety of the reactor protection system and the engineered safeguards systems. Therefore, in accordance with the graded approach to quality assurance specified in RG 1.97, it is unnecessary to apply a full quality-assurance program to this instrumentation.

Based on a graded approach to safety, this variable is more appropriately considered a Category 3 variable.

Conclusion

SLCS storage-tank-level instrumentation should meet Category 3 design and qualification criteria.

It is realized that the resolution of the ATWS issue may include substantial changes to the SLCS design criteria. At that time, the SLCS instrumentation should be reevaluated to ensure adequacy.

ISSUE 11. VARIABLES D26-D30

D26: Turbine Bypass Valve Position
D27: Condenser Hotwell Level
D28: Condenser Vacuum
D29: Condenser Cooling Water Flow
D30: Primary Loop Recirculation

Issue Definition

Regulatory Guide 1.97 states that "The plant designer should select variables and information display channels required by his design to enable the control room personnel to ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation. . . ." The purpose of this analysis was to determine whether certain other D-type variables should be added to Table 1, RG 1.97.

Discussion

Regulatory Guide 1.97 addressed safety systems and systems important to safety to mitigate consequences of an accident. Another list of variables has been compiled for the BWR in NUREG/CR-2100 (Boiling Water Reactor Status Monitoring during Accident Conditions, Apr. 1981). That report and a companion report, NUREG/CR-1440 (Light Water Reactor Status Monitoring during Accident Conditions, June 1980), address plant systems not important to safety, as well as systems that are important to safety. In particular, these reports consider the potential role of the turbine plant in mitigating certain accidents. These two reports were reviewed in determining whether any variables should be added to the RG 1.97 list.

The NUREG evaluations used a systematic approach to derive a variable list. The basic approach of the analysis was to focus on those accident conditions with which the operator is most likely to be confronted and on those accident conditions that result in the most serious consequences, should the operator fail to accomplish his required tasks. These studies used probabilistic event trees and the sequences of the Reactor Safety Study (WASH 1400) and similar studies. The events in each sequence that involved operator action were identified. Also, events were added to the event tree to include additional operator actions that could mitigate the accident. The event tree defines a series of key plant states that could evolve as the accident progresses and as the operator attempts to respond. Thus the operator's informational needs are linked to these plant states.

NUREG/CR-2100 is a BWR evaluation undertaken to address appropriate operator actions, the information needed to take those actions, and the instrumentation necessary--and sufficient--to provide the required information.

The sequences evaluated were

1. Anticipated transient followed by loss of decay-heat removal

2. Anticipated transients without scram (ATWS)

3. Anticipated transient together with failure of HPCI, RCIC, and low-pressure ECCS

4. Large loss of coolant accident (LOCA) with failure of emergency core-cooling systems

5. Small LOCA with failure of emergency core-cooling systems

The RG 1.97 list is based on accidents that result in an isolated NSSS. The NUREG documents considered accidents that could be prevented or mitigated by using water inventory and the heat sink in the turbine plant.

Conclusion

Five of the 15 variables identified in the NUREG, but not in RG 1.97, are recommended as Type D, Category 3 additions to the RG 1.97 list. Four of these variables are in the turbine plant: the turbine bypass valve position, condenser hotwell level, condenser vacuum, and condenser cooling water flow. These variables provide a primary measure of the status of a heat sink or water inventory in the turbine plant. The turbine-plant systems are not to be classed as "safety systems" or as systems important to safety. The addition of reactor primary-loop recirculation flow is also recommended.

ISSUE 12. VARIABLE E2

E2: Reactor Building or Secondary Containment Radiation

Issue Definition

Regulatory Guide 1.97 specifies that "Reactor building or secondary containment area radiation" (variable E2) should be monitored over the range of 10^{-1} to 10^{4} R/h for Mark I and II containments, and over the range of 1 to 10^{7} R/hr for Mark III containments. The classification for Mark I and II is Category 2; for Mark III, the classification is Category 1.

Discussion

As discussed in the variable C14 position statement (Issue 6), Secondary Containment Area Radiation is an inappropriate parameter to use to detect or assess primary containment leakage. However, for the Mark III containment, the reactor building is essentially part of the primary containment and it is appropriate to monitor that building volume as specified in RG 1.97.

Conclusion

It is the position of BWROG that the specified reactor building area radiation monitors be installed on Mark III containments, but that these monitors should not be required for plants with Mark I and II containments.

ISSUE 13. VARIABLE E3

E3: Radiation Exposure Rate

Issue Definition

Regulatory Guide 1.97 specifies in Table 1, variable E3, that radiation exposure rate (inside buildings or areas where access is required to service equipment important to safety) be monitored over the range of 10^{-1} to 10^4 R/hr for detection of significant releases, for release assessment, and for long-term surveillance.

Discussion

In general, access is not required to any area of the secondary containment in order to service equipment important to safety in a post-accident situation. If and when accessibility is reestablished in the long term, it will be done by a combination of portable radiation survey instruments and post-accident sampling of the secondary containment atmosphere. The existing lower-range (typically 3 decades lower than the RG 1.97 range) area radiation monitors would be used only in those instances in which radiation levels were very mild.

Conclusion

It is BWROG's position that unless plant-specific design requires access to a harsh environment area to service safety-related equipment during an accident, this parameter should be modified to allow credit for existing area radiation monitors. That is, this parameter should be reclassified as

Category 3 with a lower range to be selected on a plant-specific basis.

ISSUE 14. VARIABLE E13

E13: Primary Coolant and Sump

Issue Definition

Regulatory Guide 1.97 requires installation of the capability for obtaining grab samples (variable E13) of the containment sump, ECCS pump-room sumps, and other similar auxiliary building sumps for the purpose of release assessment, verification, and analysis.

Discussion

The need for sampling a particular sump must take into account its location and the design of the plant in which it is installed. For all accidents in which radioactive material would be in the primary containment sump of a BWR Mark I or Mark II containment, this sump will be isolated and will overflow to the suppression pool. A suppression pool sample can therefore be used as a valid alternative to a containment-sump sample.

The analysis of ECCS pump-room sumps and other similar auxiliary building sump liquid samples can be used for release assessment, as suggested in RG 1.97 only for those designs in which potentially radioactive water can be pumped out of a controlled area to an area such as radwaste. For designs in which sump pump-out is not allowed on a high-radiation or an LOCA signal, or in which the water is pumped to the suppression pool, a sump sample does not contribute to release assessment. For these designs, the use of the subject sump samples for verification and analysis is of little value; a sample of the suppression pool and reactor water, as required by other portions of RG 1.97 provides a much better measurement for these purposes.

Conclusion

1. A suppression-pool sample can be used as an alternative to a primary containment-sump sample for plants with Mark I or II containments.

2. The analysis of ECCS pump-room sumps and other similar auxiliary building sumps is a consideration only if the water is pumped out of the reactor building (e.g., pumped to radwaste). For designs in which sump pump-out is not allowed on a receipt of an accident signal, or in which the water is pumped to the suppression pool, analysis is not necessary. Provisions for sump sampling and analysis should be in accordance with each utility's response to NUREG-0737.

6. CONCLUSIONS

The BWR Owners Group RC 1.97 Committee completed an extensive analysis of the regulatory positions proposed in NRC Regulatory Guide 1.97, Revision 2. The principal goal of the committee was to formulate the position of the BWR Owners Group relative to RG 1.97 requirements. Toward that end, the committee developed--on the basis of studies conducted by its own representatives and its contractors--a series of positions with respect to interpreting and implementing the various provisions of RG 1.97.

The Owners Group concurs with the intent of RG 1.97, which is to ensure that each BWR facility is sufficiently instrumented to make possible the timely and effective assessment of plant and environmental conditions during and following an accident.

The Owners Group also recommends implementing the particular variables and instrumentation requirements of RG 1.97, except in those instances when deviations from the RG 1.97 positions are indicated, are desirable, are in accord with the intent of RG 1.97, and are technically justifiable. The exceptions noted by the Owners Group are generally derived from the incompatibility of an RG 1.97 requirement with the intent of RG 1.97; from evidence that the implementation of an RG 1.97 position would not accomplish its intended objective or that the consequence of its implementation would be undesirable from a safety point of view; or from the availability of more effective or more practical ways of achieving a particular monitoring activity.

APPENDIXES

SLI - 8121

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APPENDIX A
THERMAL ANALYSES OF IN-CORE THERMOCOUPLES IN BOILING WATER REACTORS
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Abstract

One of the new BWR requirements in Reg. Guide 1.97, in response to the event at Three Mile Island is the requirement for thermocouples located at the top of the core. An analysis was performed of the heat transfer in a BWR fuel bundle during a core uncovery event to determine the nature of the response of thermocouples to core heatup. The thermocouples were assumed to be located in the in-core guide tubes, and are heated primarily by radiation from the fuel channels. The results of this analysis show that for conditions typical of small break loss of coolant accidents, there is a delay of at least 10 minutes between the start of core uncovery and the time when the thermocouple reads 45°F above saturation. It is also probable that operation of relief valves during a small break LOCA would interfere with the thermocouples operation and could render them useless.

Summary and Conclusions.

One of the new BWR requirements in Reg. Guide 1.97, in response to the event at Three Mile Island is the requirement for thermocouples located at the top of the core. The stated purposes of these thermocouples are to provide a backup level gauge, and to provide an assessment of the degree of degradation of the core, should it become uncovered. It has been proposed that these thermocouples be located in the thimbles which house the in-core neutron flux gauges. Based on simple heat transfer analyses of conditions typical of Small Break Loss of Coolant Accidents, it is our conclusion that these thermocouples will not show a temperature 45°F above saturation until at least 13 minutes after the core has started to uncover.

We have also reviewed a calculation by the staff of the Nuclear Regulatory Commission (NRC) of the response of thermocouples in the in-core thimbles. The NRC analysis concludes that the thermocouple response time is on the order of two minutes. We believe that the difference between our analysis and theirs is that we used different, and we believe, more realistic decay power levels and the convective cooling effect of boil-off steam on the fuel rods and channel. Simple calculations show that these elements are important parts of the problem. We have also found that, using the NRC assumptions, our calculation will reproduce their results.

A preliminary look at two alternative locations (upper plenum and steam dome) did not indicate that thermocouples located there would have better response times.

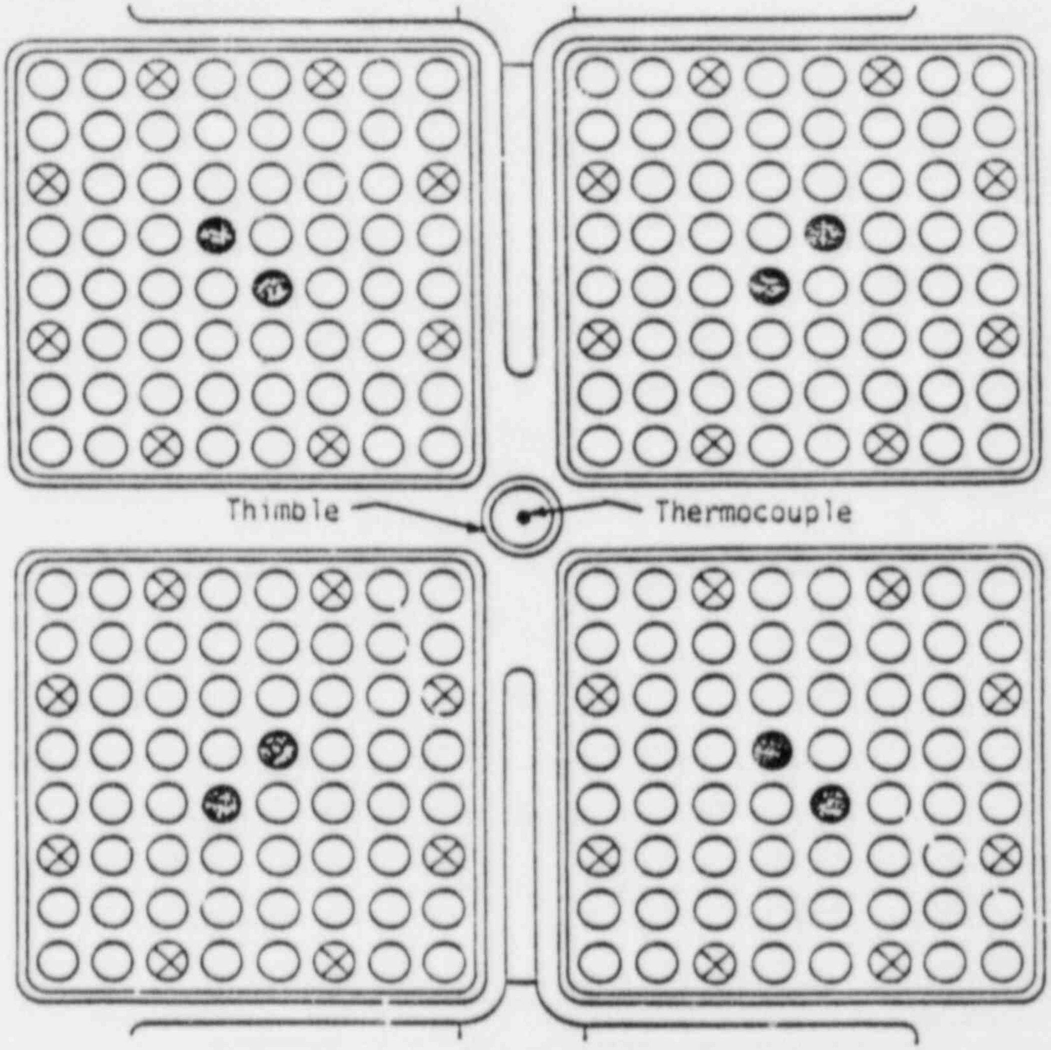


FIGURE 1

THERMOCOUPLE MOUNTED IN THIMBLE AMONG CHANNELS

Section 1

I. Heat Transfer Analysis of In-Core Thermocouples

One of the signals received by plant operations during the accident at Three Mile Island was a high temperature reading - indicating the presence of superheated steam - from the core exit thermocouples. It has now been suggested by the NRC that in-core thermocouples could be used to detect core uncovery by showing high temperatures whenever superheated steam appears. The merits of this idea for Pressurized Water Reactors (PWRs) are being debated elsewhere, only BWRs will be considered here.

II. Physical Description of the Thermocouples Mounted in Flux Monitor Thimbles

After inspection of the BWR design, it has been concluded by the authors, and independently by the NRC staff, that the most logical place (and perhaps the only practical place) to locate in-core thermocouples is in the thimbles which house the in-core neutron flux monitors. A plan view of the physical situation is shown in Figure 1. The fuel rods are surrounded by a square zircalloy channel, and the thimble is at the channel corner. It is assumed that the thermocouple sits in the center of the thimble as shown. The dimensions of parts shown are given in Appendix A.

Questions about the usefulness of the thermocouples mounted in the thimbles have centered on their time of response during a small break LOCA. In that situation the core is initially covered with water and the reactor has been scrammed. The decay heat in the core rods continues to boil the water in the core, and eventually the water level drops to the top of the core. As the water level drops further, to the level of the thermocouple, the rods are uncovered and begin to heat up. Heat then flows outward to the channel wall, to the thimble, and finally to the thermocouple.

III Heat Transfer Analysis of the Response Characteristics of In-Core Thermocouples in Small Breaks

The response characteristics of thermocouples mounted in the thimbles used for in-core neutron monitors was investigated by writing planar energy balance equations for:

i) the rods (the fuel bundle was broken into four subgroups)
ii) the channel
iii) the thimble
iv) the thermocouple.

Also, a heat balance equation was written to calculate the temperature of steam as it rises through the uncovered portion of the core. Together, these equations formed a self-consistent set which determines the temperature-time history of the thermocouple.

A. Energy balance on the thermocouple.

The thermocouple was asssumed to receive heat by radiation from the thimble wall. This is the only method of heat transfer assumed - convection through the air in the thimble was ignored. The energy equation was then:

$$\frac{d\,T_{tc}}{dt} = \frac{1}{MC}\left\{\frac{\sigma}{R_3}\left(T_{tc}^4 - T_{th}^4\right)\right\} \tag{1}$$

where

$$R_3 = \frac{1-\epsilon_{th}}{A_{th}\,\epsilon_{th}} + \frac{1}{A_{tc}} + \frac{1-\epsilon_{tc}}{A_{tc}\,\epsilon_{tc}} \tag{2}$$

B. Energy Balance on the Thimble.

The thimble receives energy by radiation from the channel wall, and loses energy by natural convection to the steam between the channels, and by radiation to the thermocouple. The steam between the channels is assumed to be at the saturation temperature. The energy balance can be written:

$$\frac{dT_{th}}{dt} = \frac{1}{M_{th}C}\left\{\frac{\sigma}{R_2}\left(T_c^4 - T_{th}^4\right) + \frac{\sigma}{R_3}\left(T_{tc}^4 - T_{th}^4\right) + \bar{h}_n A_{th}\left(T_{sat} - T_{th}\right)\right\} \tag{3}$$

where

$$R_2 = \left[\frac{1-\epsilon_c}{A_c\,\epsilon_c} + \frac{1}{A_{th}} + \frac{1-\epsilon_{th}}{A_{th}\,\epsilon_{th}}\right] \tag{4}$$

A relative evaluation of R_2 and R_3 showed that R_3 is two orders of magnitude larger than R_2. Since the temperature differences are about the same, the thimble's heat loss to the thermocouple is neglected.

C. Energy Balance on the Channel Wall.

The channel wall receives energy by radiation from the fuel rods, and loses it both by convection to the steam flow and by radiation to the thermocouple thimble. As discussed below in more detail, the rod bundle is divided into four rod subgroups and energy balance equations are written for each. The radiant heat transfer between each of those rod groups and the channel was calculated using gray body factors (F_{ij}) discussed in section E. The sum of the radiant transfer from all the rod groups to the channel is:

$$Q_{rad} = A_c * \sigma\Big[F_{1c}(T_{r1}^{4}-T_c^{4}) + F_{2c}(T_{r2}^{4}-T_c^{4}) + F_{3c}(T_{r3}-T_c^{4}) + F_{4c}(T_{r4}^{4}-T_c^{4})\Big] \tag{5}$$

The channel convection terms are calculated using a forced convection heat transfer coefficient on the inside of the channel, and a natural heat transfer coefficient on the outside of the channel. These coefficients are calculated from correlations discussed in section E. It is assumed that the steam temperature between the channels is at saturation.

$$Q_{conv} = \bar{h}_F(T_{st}-T_c) + \bar{h}_n(T_{SAT}-T_c) \tag{6}$$

The energy balance equation for the channel is then:

$$\frac{dT_c}{dt} = \frac{1}{(mc)_c}\Big\{A_c\sigma\Big[F_{1c}(T_{r1}^{4}-T_c^{4}) + F_{2c}(T_{r2}^{4}-T_c^{4}) + F_{3c}(T_{r3}^{4}-T_c^{4}) + F_{4c}(T_{r4}^{4}-T_c^{4})\Big] + \bar{h}_F A_c(T_{ST}-T_c) + \bar{h}_n A_c(T_{SAT}-T_c)\Big\} \tag{7}$$

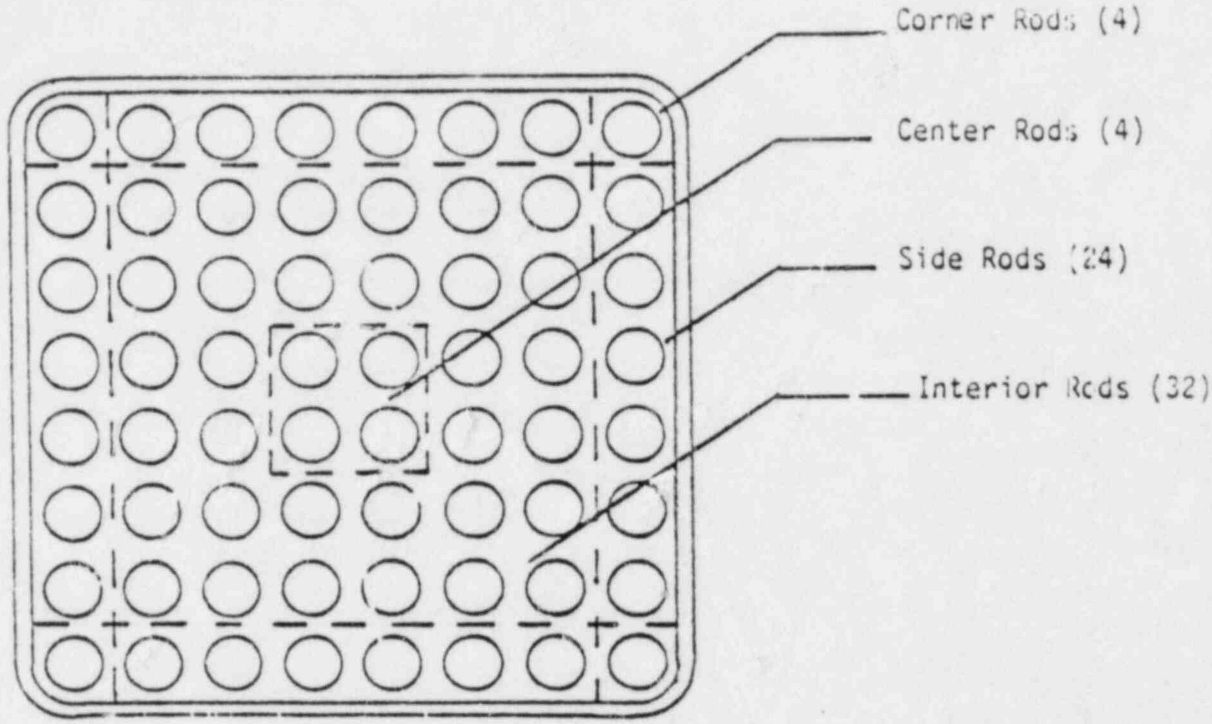


FIGURE 2

DIVISION OF RODS INTO ROD GROUPS FOR RADIATION MODEL

D. Energy Balance Equations for the Four Rod Groups

The 64 rods in a single 8 by 8 fuel rod bundle were divided up into four groups as shown in Figure 2. An energy balance equation was written for each of these rod groups which considered the heat up of the rods by decay heat, the transfer of energy among the rod groups (and channel wall) by radiation, and heat transfer by convection to the steam. Radiation from the rods to the steam was neglected as this has been shown (4) to be a small term.

The four rod group energy balance equations then have the form

$$\frac{dT_{ri}}{dt} = \frac{1}{n_i (mc)_r} \left[Q_{DECAY} - \sum_{j=1}^{4} F_{ij} A_i (T_{ri}^4 - T_{rj}^4) + \bar{h}_F A_i (T_{ri} - T_{st}) \right] \tag{8}$$

The decay heat is determined from the ANS decay heat curve for times between 150 and 10,000 seconds, and the initial power before scram.

$$Q_{Decay}(s,t) = (Q_o) \left[.130(t-t_{scram})^{-.283} \right] \cdot f(s) \qquad (9)$$

where f(x) is the axial power shape, and Q_o is the initial power. The initial power level assumed is 2436 megawatts (thermal). The axial power shape used is:

$$f(x) = 1.387 \left[\cos \left(\frac{x-6}{4.425} \right) \right] \qquad (10)$$

where x is in feet and the computed angle is in radians.

E. Convective Heat Transfer Correlations and Radiation Model

Equations 5 and 7 above use the convective heat transfer coefficients for the rod surface, the inside channel surface and the outside channel surface. When the Reynolds' number for the steam flow through the rod bundle is greater than 2300, the correlation below is used to obtain the Nusselt number for the rod surfaces.

$$Nu_{ir} = \left[0.022\ P_r^{0.5}\ R_e^{0.8} \right] * F\ (s/r) \qquad (11)$$

The Reynolds number in this calculation is defined as:

$$Re = \frac{4\ G_{st} A_{flow}}{\mu_{st}\ P} = \frac{4\ G_{st}\ A_{flow}}{\mu_{st}\ \pi d} \qquad (12)$$

Equation (12) was modified for the parallel rod geometry by the factor F (s/r) which depends, as shown by Reference 1 on the ratio of rod pitch to rod radius (s/r). The resulting heat transfer coefficients ranged between 10 and 17 Btu/hr ft^2 ^{o}F.

When the Reynolds number is below 2300, a constant Nusselt number, given for rod bundles as a function of (s/r) by Sparrow, Loeffler and Hubbard is used. (Ref. 3)

$$Nu = C \cdot f(s/r) \tag{13}$$

For the channel wall, the Nusselt number for turbulent flow is calculated from equation 11, without the F (s/r) correction.

Similarly, for laminar flow, equation (13) is used for the channel without the F (s/r) correction factor.

Radiation heat transfer between the rod groups is calculated using grey body factors, which account for the fact that some of the radiation incident on a surface is absorbed and some is reflected. These factors denoted F_{ij} are defined in terms of the radiant heat transfer between two surfaces as:

$$Q_{ij} = A_j F_{ij} \sigma [T_i^4 - T_j^4] \tag{14}$$

These factors were developed from the emissivities of the surfaces (assumed to be .6) and the geometric view factors for rod to rod and rod to channel radiation given in Reference 5. As in reference 5, it was assumed that all radiation emitted by a rod would be absorbed by its 25 nearest neighbors, and that the fraction of radiation emitted outside the 25 nearest neighbor rods (or channel surface) which arrived at a given rod after multiple reflections was negligible.

F. Calculation of the Steam Temperature and Flow Rate

In equations 5 and 7 the rate of convective heat transfer is determined by the flow rate of boiled-off steam, and its temperature as it moves through the fuel assemblies. The boil-off rate, for a partially-submerged fuel bundle, was calculated by assuming that all the decay heat from the portion of the fuel rods below the waterline goes into producing steam. The water level is determined by integrating the boil-off rate as the calculation proceeds.

When the steam leaves the water's surface, its temperature will be at saturation. As the steam rises through the rod bundle it will be heated by contact with the rods. Thus, steam temperature is both a function of time and elevation. To calculate the steam temperature at any elevation at a given time the following equation is integrated from the liquid surface to the top of the rod bundle.

$$\frac{dT_{st}}{d_x} = \frac{\bar{h}_f A_r}{GA_{flow}Cp} (T_r - T_{st}) \tag{15}$$

This integration is done numerically using a core divided into twelve zones. The rod temperatures are obtained from a heat balance on an average rod in each of the twelve zones.

The above set of ordinary differential equations was integrated forward in time simultaneously using a fourth-order accurate Adams predictor-corrector scheme.

IV Results for Thermocouple in Thimble

The calculations described above was performed for the following starting conditions:

- Reactor power at 2% of full power (2436 MW thermal) - this corresponds to 700 seconds after scram.
- No feedwater supply to reactor pressure vessel or leakage.
- Constant Reactor pressure of 1000 psia.
- 8x8 fuel

These conditions were chosen so that our calculation would correspond to one performed by the NRC which will be discussed later. In the NRC calculations, it was assumed that the operator would not consider the thermocouple signal to be seriously out of line until it read 45°F above saturation. At first glance this seems like a high number. However, it must be remembered that the saturation temperature is not absolutely steady and that during plant transients, it can change by about ± 20°F, so the value of 45°F is reasonable. The fact that the operator has to keep the change of saturation temperature with reactor pressure in mind is another complicating factor which will make successful use of the thermocouples less likely.

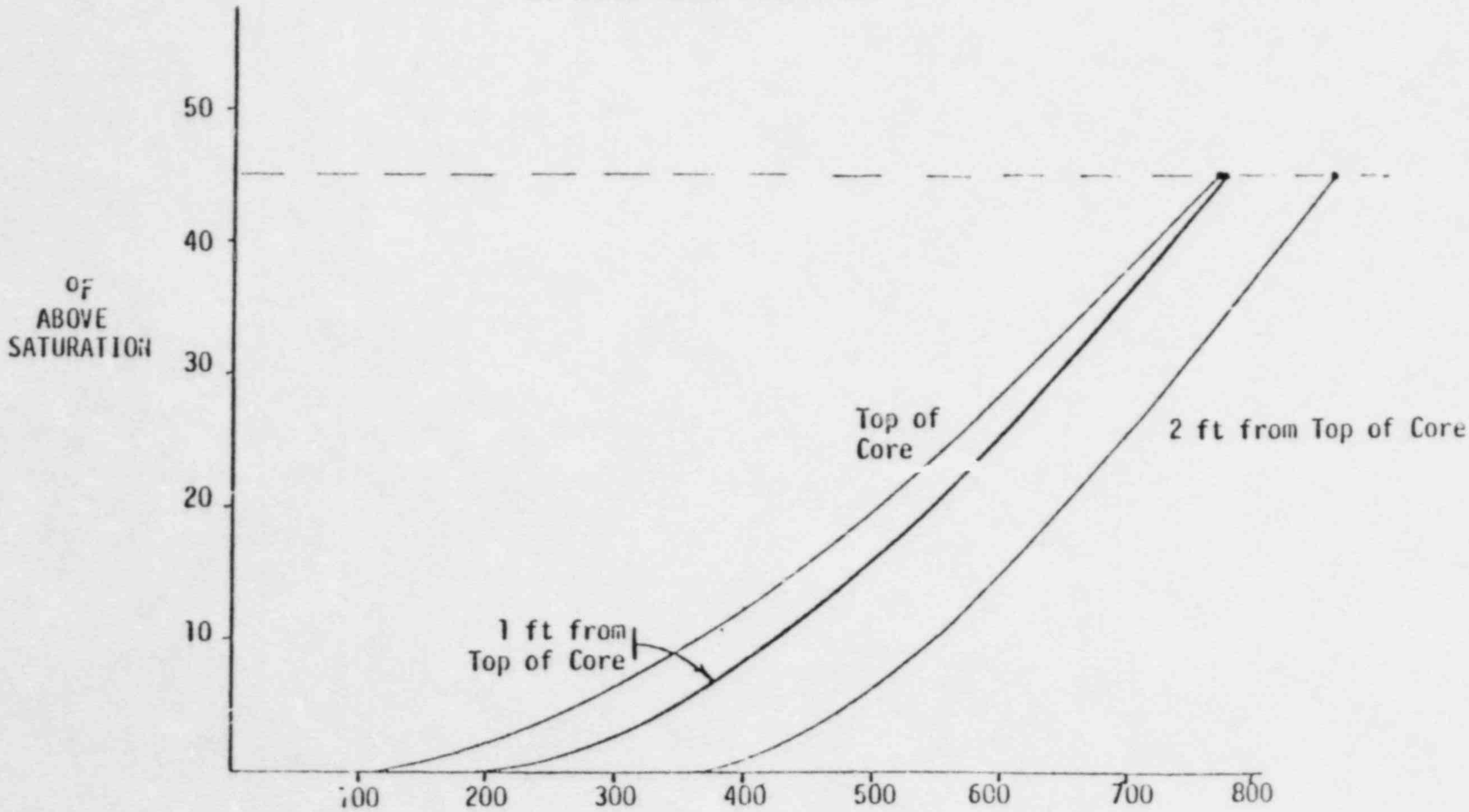
Figure 3 shows the calculated temperature response for three axial thermocouple positions, the top of the core, 11 ft and 10 ft. elevations. This graph shows that the response times are on the order of 13 minutes. Figure 3 also shows that the optimum location for the thermocouple is near to the top of the core, although the response time (measured from the start of core uncovery) is not a strong function of position. After examining Figure 3 it was decided to use a thermocouple location 1 ft from the top of the core for all further calculations.

More detailed information on the response of the system with the thermocouple located one foot below the top of the core is shown in Figure 4.

The plane of the thermocouple is uncovered about 150 sec after the top of the core uncovers. The rods begin to heat up adiabatically, but later the rate temperature rise drops off due to convection and radiation losses. As the foam level in the bundles drops, and more and more of the core below the plane of the thermocouple is uncovered, the temperature of the steam passing the thermocouple location rises. The channel wall, thimble and thermocouple all rise in temperature, and the thermocouple is 45°F above saturation 780 seconds (13 minutes) after the start of core uncovery. Figure 4 also shows that the time lag between the thimble and thermocouple temperatures is extremely small, thus direct contact between the thermocouple and thimble will not significantly reduce the time delay.

FIGURE 3

THERMOCOUPLE TEMPERATURE TIME HISTORY
AT THREE AXIAL POSITIONS

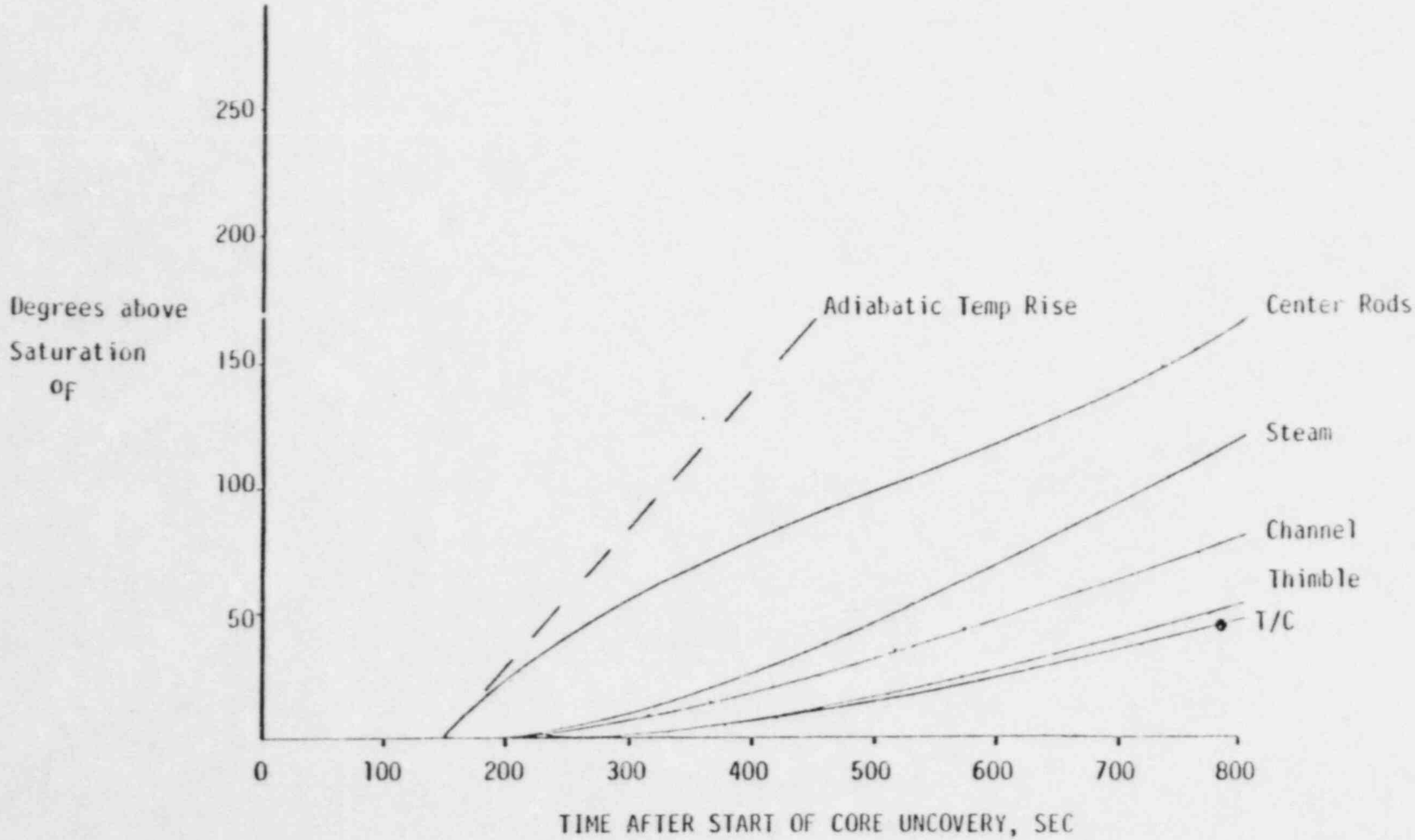


TIME, SECONDS AFTER START OF UNCOVERY

FIGURE 4

TEMPERATURE TIME HISTORIES

FOR THERMOCOUP:E LOCATION 1 ft BELOW TOP OF CORE



V. Verification of Analysis

To check the correctness of the above calculation, two checks were made. First, the initial rate of temperature rise should be consistent with the adiabatic rod heat up rate at 2% power. This rate is

$$\frac{dT_A}{dt} = \frac{\text{average bundle decay power}}{\text{bundle heat capacity}} * \text{axial peaking function}$$

$$= \frac{0.02 * 2438 \text{ megawatts} * 948 \frac{\text{Btu}}{\text{sec. megawatt}} / 560 \text{ assemblies}}{64 * \left[7.37 \text{ Lbm } UO_2 + .911 \text{ Lbm } Z_r\right] * 0.12 \frac{\text{Btu}}{\text{Lbm}^{\circ}\text{F}}} * f(x)$$

$$= 1.30\ ^{\circ}\text{F/sec.} * f(x)$$

A line corresponding to the adiabatic heat up rate at 1 ft. below TAF has been drawn on Figure 4 and it can be seen that the rod temperature rise rate approaches it near its time of uncovery.

A calculation was also conducted to check the correctness of the steam temperature rise calculation. Figure 5 shows the axial distribution of interior group rod temperatures and steam temperatures at 1000 sec after the start of core uncovery. To check the calculated steam temperatures, the rod temperature distribution was approximated with the dashed lines shown. For a linear temperature profile, constant heat transfer coefficient and flow velocity the analytical solution for steam temperature is:

$$T_{st} = a + b(x-1/k) + (T_i + b/k - a)\, e^{-kx} \qquad (16)$$

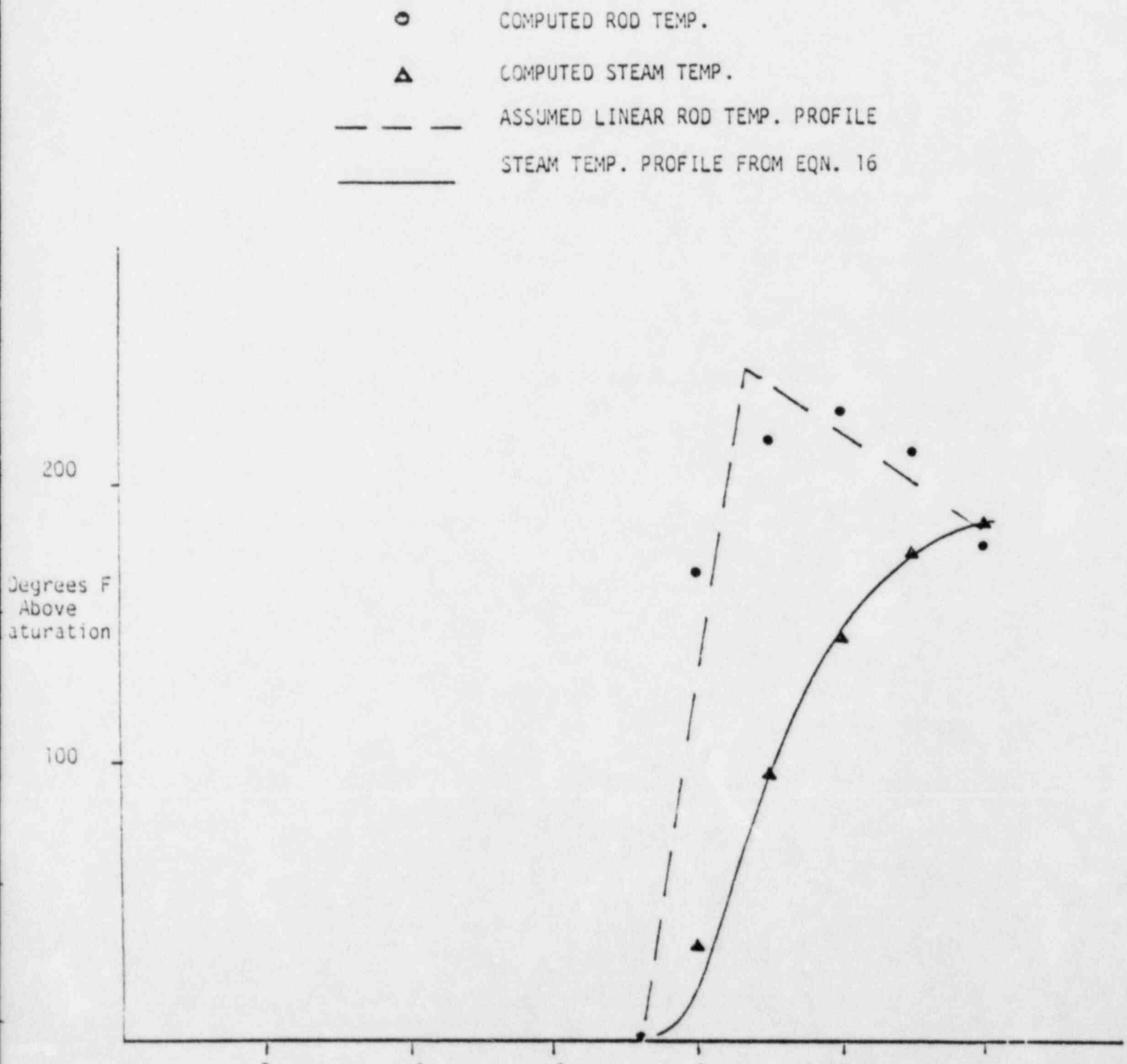
where a and b are the coefficients of the linear temperature profile ($T_r = a+bx$) and k is defined as

$$k = \frac{\bar{h} * n\pi d}{\dot{m}\, C_p} \qquad (17)$$

Using the heat transfer coefficient computed from the correlation given earlier (9.42 $\text{Btu/Hr-Ft}^2 - {}^{\circ}\text{F}$), the steam temperature was calculated using the above formula. Results are plotted on Figure 5 and show close agreement with the machine calculation.

FIGURE 5

CHECK OF STEAM TEMPERATURE CALCULATION



VI <u>Comparison of Present Calculations with a Similar Analysis by the Staff of the NRC</u>.

As part of this project, we have reviewed a calculation of the thermocouple response time by the NRC office of Nuclear Regulatory Research. Their calculation assumed a 2% of rated uniform axial power input and no convective heat transfer. They assumed that convection and radiation losses from the rods would be negligible. Their results are plotted in Figure 6. The adiabatic rod heat up rate which they calculated was about 2.7°F per second at the 80% of core height elevation (about 9.7 ft above B A F) and 3.8°F at the 60% core height elevation. With these heat up rates their results show that a thermocouple at the 60% height would show a 45°F temprature rise 120 sec after the 60% plane is uncovered.

The simple calculation in the last section shows that the adiabatic heatup rate should be on the order of 1°F/sec rather than the 2.7-3.8 that the NRC used. However, in order to compare our calculation to theirs, we adjusted the prescram power in our code (to 8,672 megawatts from 2436) and set the convective heat transfer coefficients equal zero. These results are shown in Figure 7. They agree very well with the NRC results. Using the NRC heatup rate our code predicts that a thermocouple at the 60% height will show 45°F temperature rise 135 seconds after the 60% plane is uncovered. The NRC calculated the 60% plane would uncover after 90 sec and the 80% plane after 210 sec. Our calculations, with their assumptions, shows these planes uncovering at 110 and 242 seconds respctively.

We conclude that the essential differences between our calculation and the NRC's are the extremely high heat up rates they assumed and the fact that they neglected convection to the passing steam. Both of these differences tend to make the calculated core temperatures rise more quickly after uncovery which speeds up thermocouple response. We believe our assumptions are more realistic, and our results more correct.

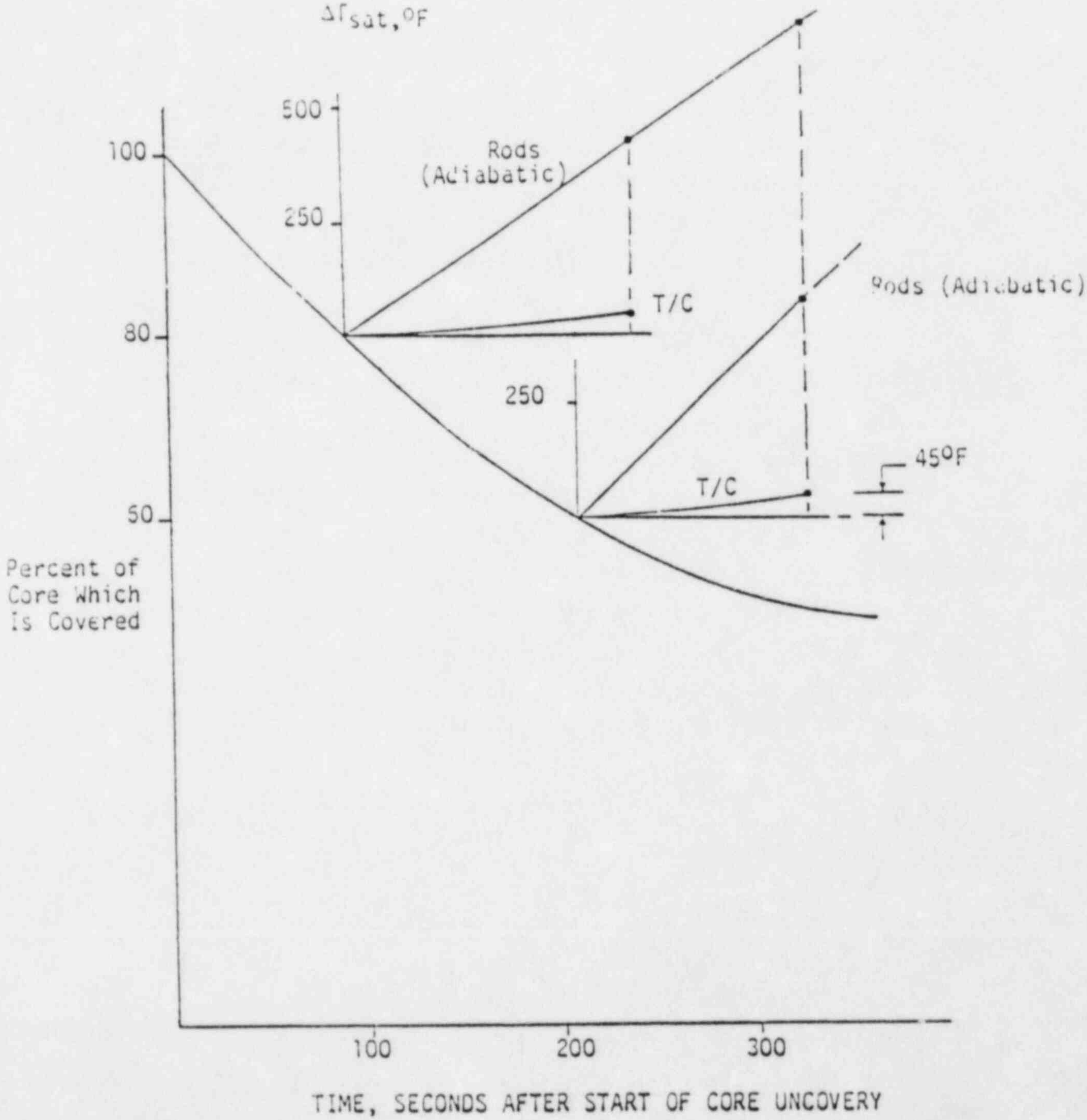
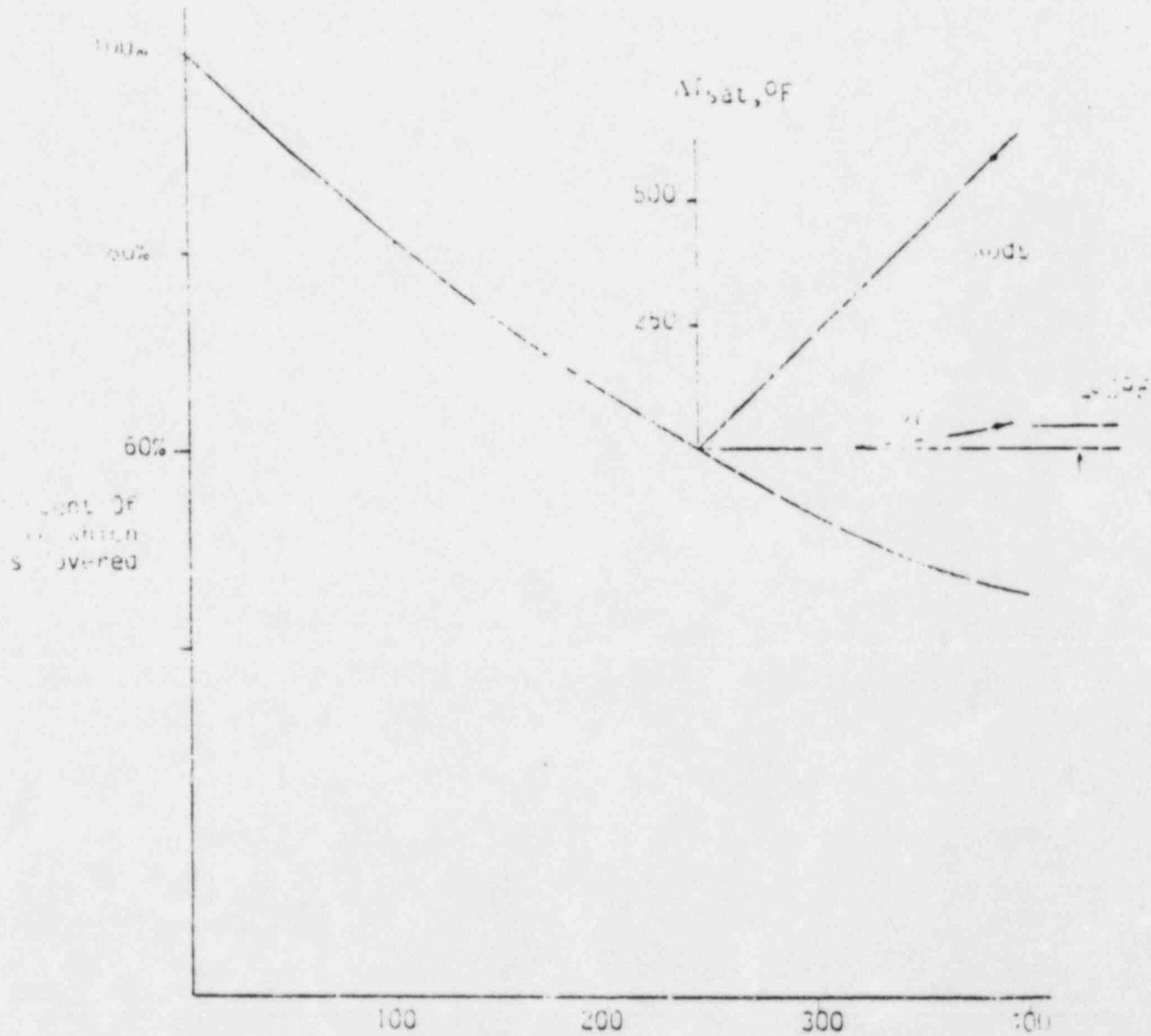


FIGURE 6

RESULTS OF NRC CALCULATION



TIME AFTER START OF CORE UNCOVERY, SECS.

FIGURE 7

PRESENT CALCULATION WITH NRC ASSUMPTIONS

VII. Effect of Changing Reactor Pressure Vessel Pressure

All the calculations discussed in the previous sections assumed th[illegible] the reactor vessel pressure was constant during core uncovery and heat up. During the sort of small break loss of coolant accident where core thermocouples are likely to be useful, however, the pressure will most likely not be constant. For example, below in Figure 8 is a computed pressure trace for the Leibstadt plant during a Turbine Trip transient. The reactor vessel pressure rises to the relief valve set point and then drops when the valve is open. During this pressure drop, voids will form in the saturated liquid. These voids will raise the water level as illustrated in Figure [illegible] The amount by which the water level will rise can be determined by a simple approximate, calculation.

According to Reference 5, the fluid filled cross-sectional area [illegible] the Peachbottom II plant between the top and bottom of active fuel is about [illegible] ft^2. The amount of water below the core is about 4100 ft^3. [illegible] density of saturated water near 1000 psia is 46.3 lbm/ft^3, the mass [illegible] water in the reactor is

$$\text{Mass of } H_2O = 190,000 - 10,140\ Z$$

where Z in the height of active core which is covered. The quality change for an incremental change in pressure can be obtained from the chain rule

$$\Delta x = \frac{dx}{dh} \text{ --- } \frac{dh}{dP} \text{ --- } \Delta P$$

Under these conditions

$$\Delta x = 0.004 \left[\frac{in^2}{Lb} \text{ --} \right] \Delta P$$

For the 60 Psi change in pressure which Figure 8 shows takes place [illegible] relief valve opens [illegible] to an increase in volume of

$$\Delta v = 1894 - 101\ Z$$

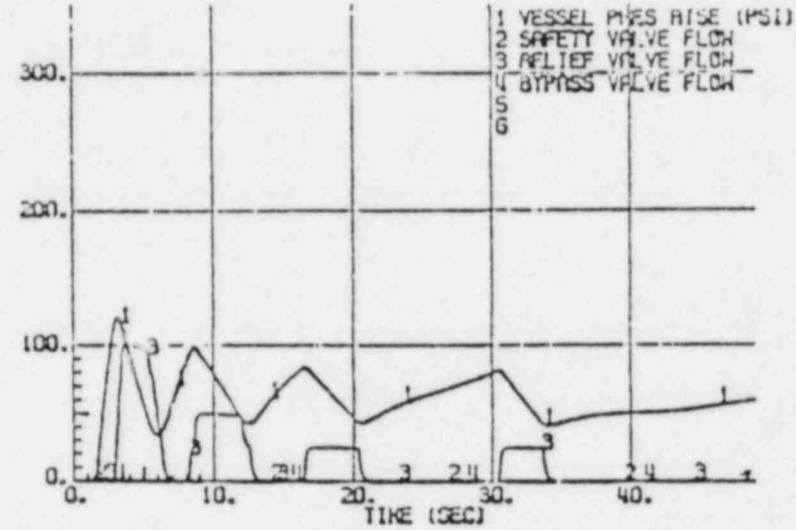


FIGURE 8

SMALL BREAK PRESSURE HISTORY

(Liebstadt)

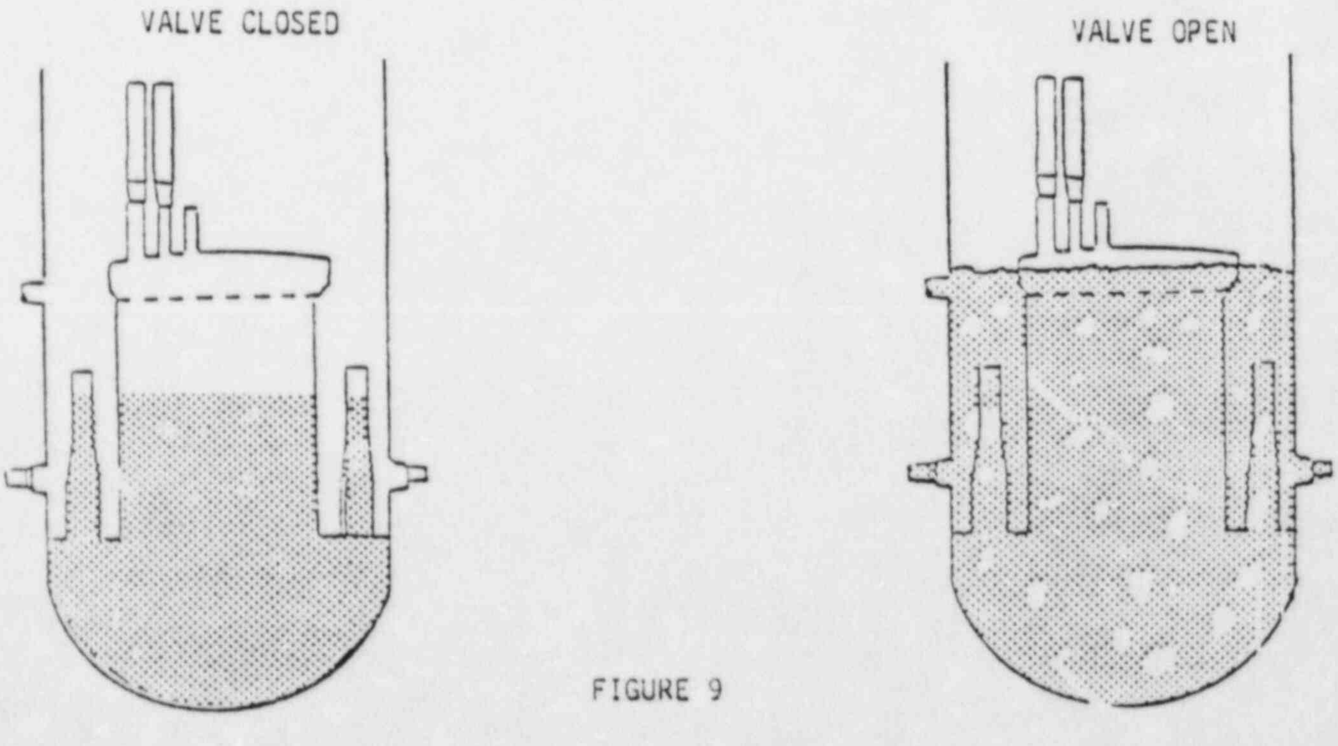


FIGURE 9

LEVEL SWELL EFFECT OF OPEN RELIEF VALVE

FIGURE 10

ANTICIPATED EFFECT OF LEVEL SWELL

Using the core, bypass and annulus fluid cross section of 220 ft^2, this corresponds to a change in water level of

$$\Delta h = 8.64 + 0.462\ Z$$

or 8-12 ft. This will be enough to periodically cover and uncover the thermocouple until the core is almost completely uncovered.

The effect of this periodic swamping of the thermocouple plane is not easy to predict. If the rods are hot enough, then the rod surface <u>will not</u> rewet and very little heat will be lost. On the other hand, even if the channel wall is hot, the fact that it has a high surface to volume ratio means that it (and the thimble) probably <u>will</u> rewet, and its temperature will drop to saturation. In this case, the temperature-time history of the thermocouple would look like Figure 10. The rods would heat up gradually but the thermocouple would never read a temperature very far from saturation.

VIII. Other locations for Thermocouples

A very quick investigation was made of two alternative locations for the thermocouples. The two locations looked into, in the upper plenum and in the steam dome, were chosen on the basis of the following argument. If it is impractical to locate an in-core thermocouple any closer to the fuel cladding than in the in-core flux monitoring tubes, then the only other way to get the information that the core is overheating is to measure the steam temperature after the steam has left the core. The ideal way to do this would be to put a bare thermocouple in the steam flow just above the core exit. Examination of detailed reactor drawings indicates that this would be very difficult to do. An easier alternative would be to put the thermocouple in the steam dome. A thermocouple in the steam dome, however, will not respond immediately to an increase in core steam exit temperature. To get to a thermocouple in the steam dome, the steam will have to pass through relatively cold standpipes, steam separators and dryers before it enters the dome.

The analysis developed to investigate thermocouple response in the in-core tubes was used to determine the response time of thermocouples in these two locations. The temperature drop of the steam as it flows through the dryers and separators was calculated (approximately) by treating these parts as a uniform temperature heat exchanger:

$$T_{st}(exit) = T_{st}(entrance) + [T_{surf} - T_{st}(entrance)] \; e^{-Ntu}$$

where T_{surf} is the temperature of the dryers, separator and standpipes, and Ntu is defined

$$Ntu = \frac{\bar{h}A}{(\dot{m}\ c_p)\ steam}$$

The heat transfer coefficient used was the same one calculated for the rod bundle. The separator-dyer-standpipe temperature was calculated as a function of time by

$$\frac{d(T_{surf})}{dt} = \frac{(\dot{m}\ c_p)}{(m_{sep}C)} \left[T_{st}\ (exist) - T_{st}\ (entrance)\right]$$

The area of the separator-standpipes-dryers was estimated at 20,000 Ft^2, the mass was estimated at 130,000 Lbm.

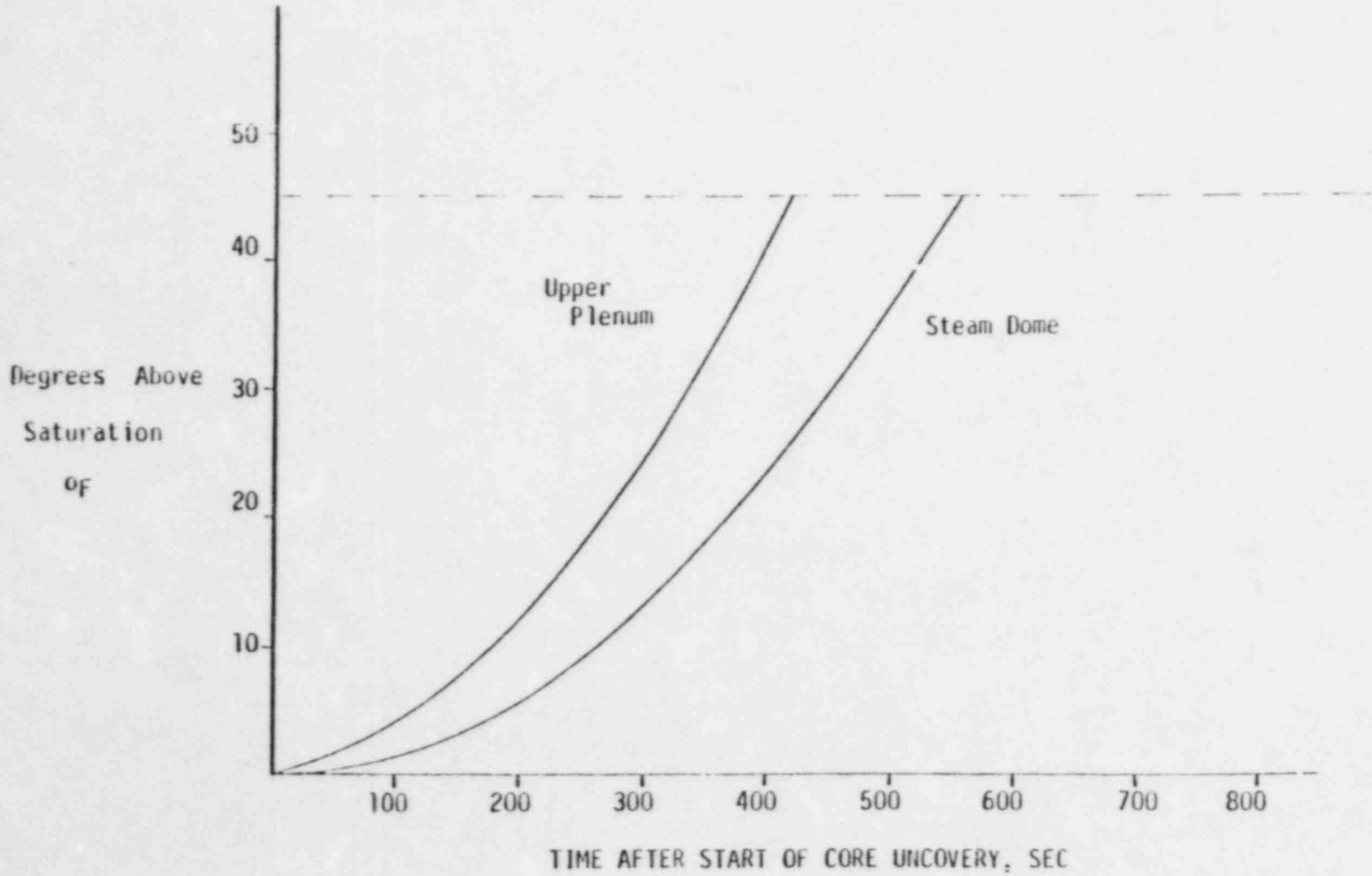
Results shown in Figure 11 do not show the alternative locations to be promising. As Figure 5 showed earlier, the temperature of the steam at the core exit follows the temperature of the top of the rod bundle fairly closely. Since the power is low at the top of the bundle, the temperature there rises fairly slowly. For this reason, a thermocouple in the upper plenum would not read 45°F above saturation for seven minutes after the start of core uncovery. Figure 11 shows that the time delay introduced by the hardware above the upper plenum is not too great, and that a thermocouple in the steam dome would read 45°F above saturation about 9.2 minutes after uncovery.

The two response times calculated above for thermocouples in the upper plenum and steam dome are intended to illustrate the lower limit of how fast they could possibly be under idealized conditions in which an unshielded thermocouple is placed directly in the steam flow out of the core (upper plenum) or directly in the steam flow out of the dryers (steam dome). For other, more realistic, installation positions these times are unrealistically low. In both cases the large volumes of saturation temperature steam in both the upper plenum and steam dome will dilute the superheated steam from the core and will slow the response greatly. Calculations which include this dilution effect in a very approximate manner show the time delay increased by a factor of two.

FIGURE 11

TEMPERATURE TIME HISTORIES FOR ALTERNATE LOCATIONS

- NOT INCLUDING DILUTION BY STAURATED STEAM.



IX. Estimates of Costs & Exposure for Installation of Incore Thermocouples

Tables 1 and 2 show estimated costs and exposures respectively for installation of 16 thermocouples (TC's) in the BWR core for use as a diverse level sensing method. The 16 TC's are to be installed in 16 different LPRM tubes, 4 in each quadrant of the core. Three cases are considered:

Case 1. Installation Prior to Fuel Load

Case 2. Installation During an Outage

Case 3. Differential Cost of TC Installation vs. Normal Failed LPRM Replacement work performed during a refueling outage.

The costs include material, labor, overhead, engineering and A/E fees, contingency and escalation @ 10%/yr. for 3 years. The material costs include $700,000 for 16 strings of qualified LPRM assy. w/TC which results in a cost of $43,750/assy. This compares with estimated cost of $20,000 to $30,000/assy. for a standard replacement LPRM assy. The additional costs includes the TC, and allocated R&D and qualification costs. In calculating the differential cost in Table 1, Case 3 the cost of a typical LPRM assy. wo/TC was taken as $25,000.

The costs and radiation expenses for thermocouple installation can be summarized as:

	Cost	Exposure Man/R Min.	Exposure Man/R Max.
Case 1 (Prior Fuel Load)	$ 2,093,948	N/A	N/A
Case 2 (During Outage)	2,470,220	65	450
Case 3 (vs. Repl. LPRM)	1,697,237	50	250

Table 2 shows a min/max rem exposure expected for installation during an outage. There is a wide variation in expected radiation rates at operating plants which is affected by factors such as:

- History of Fuel Failures
- Water Chemistry
- Reactor Water Clean Up & Polishing Demineralizer Operation History.

Some plants could produce rates 2 or 3 times higher than the highest rate on Table 2. The rates on Table 2 are considered ranges expected for 75% of operating BWR's. The total exposure would be spread over a number of workers so as not to exceed the quarterly allowables for 1 worker.

The following assumptions were used in developing these estimates:

1. Installation of TC's would be accomplished by replacing an LPRM assy. with a new design which includes a TC in the LPRM assy.

2. The existing wiring and connectors for LPRM's need not be altered or replaced.

3. New wiring for 16 TC's is added using existing spare electrical penetrations. No drywell shield or primary containment core drilling is necessary.

4. The TC's are wired back to the relay room to 16 signal conditioners and from there to 2 recorders in the control room. The system is separated into 2 divisions

5. For installation of each of 16 LPRM assy. related cable and conduit runs inside containment, a five man crew including 1 supervisor is used. The four workers require a total of 80 Mandays (per TC) to do the work. Half the 80 Man/days (MD) is spent inside containment. of this 40 MD, 2MD/TC is spent inside the drywell and the remaining 38 MD/TC is spent inside containment.

6. The differential exposure between installing TC's vs. the normal failed LPRM replacement activity is the exposure resulting from cable installation inside primary containment only.

TABLE 1

ITEM	Case 1 Prior to Fuel Load MH Labor	Case 1 Prior to Fuel Load $ Material	Case 2 During Outage MH Labor	Case 2 During Outage $ Material	Case 3 Cost in Addition to Replacement of Failed LPRMS MH Labor	Case 3 Cost in Addition to Replacement of Failed LPRMS $ Material
1. LPRM Strings & Install (16 Strings)	2,560	700,000	6,400	700,000	28	300,000
2. Cable (to Control Room)	1,740	4,800	3,720	4,800	3,720	4,800
3. Penetrations & Assy. (Incl. Seal)	448	140,000	1,380	140,000	1,380	140,000
4. Terminal Boxes	160	4,000	480	4,000	480	4,000
5. Cable Trays & Installation	3,400	48,000	5,000	48,000	5,000	48,000
6. Electronics Installation	550	24,000	550	24,000	550	24,000
Sub Total	8,858	920,800	17,530	920,800	11,258	520,000
Labor @ $20/MH		177,160		350,600		225,160
Distributed Costs (Clerical, Doc. etc.) @ 55% (DL)		97,000		193,000		123,838
Utility Engineering		130,000		140,000		140,000
A/E Fee ≈ 5% (M+L)		55,000		65,000		65,000
Escalation (3 yr @ 10%)		413,988		500,820		322,439
Contingency		300,000		300,000		300,000
Total		$2,093,948		$2,470,220		$1,697,237

Table 2.

I. Radiation Intensity

Location	Exposure mR/Hr	
Inside Drywell	Min	Max
At LPRM Flange	100	750
Platform (5' Below Flange)	50	300
Inside Primary Containment	10	50

II. Estimated Exposure for 16 LPRM Assy. w/TC.

Min.

Drywell: $\frac{2\ MD}{TC} \times \frac{8\ hrs}{D} \times 16\ TC \times \frac{50\ mR}{HR} =$ 12.8 ManR

Prim Contm: $\frac{38 MD}{TC} \times \frac{8\ hrs}{D} \times 16\ TC \times \frac{10\ mR}{Hr} =$ 48.64 ManR

Min. Total = 61.44

Say 65 Man R

Max

Drywell: $\frac{2\ MD}{TC} \times \frac{8\ hrs}{D} \times 16\ TC \times \frac{750\ mR}{Hr} =$ 192 Man R

Prim Contm: $\frac{38\ MD}{TC} \times \frac{8\ hrs}{D} \times 16\ TC \times \frac{50\ mR}{Hr} =$ 243.2 Man R

TOTAL = 435.2 Man R

Say 450 Man R

MD = Man Day

Table 2 (Contd)

III. Differential Exposure vs. Replacement of 16 Failed LPRM

Min. = Prim. Contm Exp. ≈ 50 Man R

Max. = " " " ≈ 250 Man R.

X. Conclusions

Based on the preceeding analyses we conclude:

1. If thermocouples are mounted in BWR cores for use as core uncovery indicators they will not respond for at least 10 minutes after uncovery in a small break LOCA.

2. Because BWR's have other level gages, the operator will be given conflicting information during this 10-13 minutes. That is, his core thermocouples would say he is not in trouble, while his level gages say he is.

3. The analysis performed by the NRC calculates a quick response of the core thermocouples because of two assumptions made - first that convective heat transfer may be neglected, and second that the uncovered rods (at 2% decay heat) heat up at a rate of 3°/sec. These assumptions are unrealistic, and erroneously lead to the conclusion that core thermocouples are an effective means of determining core water level.

4. The operation of pressure relief valves during a small break LOCA has the potential to render the thermocouples useless. They could read the saturation temperature even while the core heats up. This will further confuse the operator.

5. Locating the thermocouple in the upper plenum or steam dome probably will not reduce the time delay. Furthermore, this has not yet been proven to be a feasible option, due to installation difficulties.

6. The installation cost of in-core thermocouples will be on the order of 2.5 million dollars for four thermocouples per quadrant.

7. The maximum radiation exposure for thermocouple installation will be 450 man-rem.

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APPENDIX A

The dimensions used in this analysis are shown below:

Rod bundle axial length	148 ins.
Rod diameter	0.416 ins.
Cladding thickness	0.034 ins.
Fuel Rods per bundle	64
Rod to wall gap	0.135 ins.
Channel cross section	5.52 x 5.52 ins.
Channel wall thickness	.120 ins.
Rated Reactor Thermal Power	2436 megawatts
Thimble diameter	0.70 ins.
Thimble thickness	0.080 ins.

APPENDIX B
TABLE 1: BWR VARIABLES
(NRC Regulatory Guide 1.97, Revision 2)

TYPE A Variables: those variables to be monitored that provide the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

A variable included as Type A does not preclude it from being included as Type B, C, D, or E or vice versa.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Plant specific	Plant specific	1	Information required for operator action

TYPE B Variables: those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control). Variables are listed with designated ranges and category for design and qualification requirements. Key variables are indicated by design and qualification Category 1.

Variable	Range	Category	Purpose
Reactivity Control			
Neutron Flux	10^{-6}% to 100% full power (SRM, APRM)	1	Function detection; accomplishment of mitigation
Control Rod Position	Full in or not full in	3	Verification
RCS Soluble Boron Concentration (Sample)	0 to 1000 ppm	3	Verification
Core Cooling			
Coolant Level in Reactor	Bottom of core support plate to lesser of top of vessel or centerline of main steam line	1	Function detection; accomplishment of mitigation; long-term surveillance
BWR Core Thermocouples[2]	200°F to 2300°F	1[1]	To provide diverse indication of water level
Maintaining Reactor Coolant System Integrity			
RCS Pressure[2]	15 psia to 1500 psig	1	Function detection; accomplishment of mitigation; verification
Drywell Pressure[2]	0 to design pressure[3] (psig)	1	Function detection; accomplishment of mitigation; verification

[1] Four thermocouples per quadrant. A minimum of one measurement per quadrant is required for operation.

[2] Where a variable is listed for more than one purpose, the instrumentation requirements may be integrated and only one measurement provided.

[3] Design pressure is that value corresponding to ASME code values that are obtained at or below code-allowable values for material design stress.

TABLE 1 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE B (Continued)			
Drywell Sump Level[2]	Bottom to top	1	Function detection; accomplishment of mitigation; verification
Maintaining Containment Integrity			
Primary Containment Pressure[2]	10 psia to design pressure[3]	1	Function detection, accomplishment of mitigation, verification
Primary Containment Isolation Valve Position (excluding check valves)	Closed–not closed	1	Accomplishment of isolation

TYPE C Variables: those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Fuel Cladding			
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	1/2 Tech Spec limit to 100 times Tech Spec limit, R/hr	1	Detection of breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[4]	Detail analysis; accomplishment of mitigation; verification, long-term surveillance
BWR Core Thermocouples[2]	200°F to 2300°F	1[1]	To monitor core cooling
Reactor Coolant Pressure Boundary			
RCS Pressure[2]	15 psia to 1500 psig	1[5]	Detection of potential for or actual breach, accomplishment of mitigation; long-term surveillance
Primary Containment Area Radiation[2]	1 R/hr to 10^5 R/hr	3[6,7]	Detection of breach; verification

[4] Sampling or monitoring of radioactive liquids and gases should be performed in a manner that ensures procurement of representative samples. For gases, the criteria of ANSI N13.1 should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones, and sampling lines should be designed to minimize plateout or deposition. For safe and convenient sampling, the provisions should include:

a. Shielding to maintain radiation doses ALARA,
b. Sample containers with container-sampling port connector compatibility,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Prearrangement for analysis and interpretation.

[5] The maximum value may be revised upward to satisfy ATWS requirements.

[6] Minimum of two monitors at widely separated locations.

[7] Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an energy response accuracy of ±20 percent at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within a factor of 2 over the entire range.

TABLE 1 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE C (Continued)			
Reactor Coolant Pressure Boundary (Continued)			
*Drywell Drain Sumps Level[2] (Identified and Unidentified Leakage)	Bottom to top	1	Detection of breach, accomplishment of mitigation; verification; long-term surveillance
Suppression Pool Water Level	Bottom of ECCS suction line to 5 ft above normal water level	1	Detection of breach; accomplishment of mitigation; verification, long-term surveillance
Drywell Pressure[2]	0 to design pressure[3] (psig)	1	Detection of breach; verification
Containment			
RCS Pressure[2]	15 psia to 1500 psig	1[5]	Detection of potential for breach; accomplishment of mitigation
Primary Containment Pressure[2]	10 psia pressure to 3 times design pressure[3] for concrete, 4 times design pressure for steel	1	Detection of potential for or actual breach, accomplishment of mitigation
Containment and Drywell Hydrogen Concentration	0 to 30% (capability of operating from 12 psia to design pressure[3])	1	Detection of potential for breach; accomplishment of mitigation
Containment and Drywell Oxygen Concentration (for inerted containment plants)	0 to 10% (capability of operating from 12 psia to design pressure[3])	1	Detection of potential for breach; accomplishment of mitigation
Containment Effluent[2] Radioactivity - Noble Gases (from identified release points including Standby Gas Treatment System Vent)	10^{-6} μCi/cc to 10^{-2} μCi/cc	3[8,9]	Detection of actual breach; accomplishment of mitigation; verification
Radiation Exposure Rate[2] (inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located)	10^{-1} R/hr to 10^{4} R/hr	2[7]	Indication of breach

[8]Provisions should be made to monitor all identified pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64. Monitoring of individual effluent streams is only required where such streams are released directly into the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this regulatory guide provided such monitoring has a range adequate to measure worst-case releases.

[9]Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fresh equilibrium noble gas fission product mixtures to 10-day-old mixtures, with overall system accuracies within a factor of 2. Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of any noble gas nuclide(s). It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this regulatory guide and that multiple components or systems will be needed. Existing equipment may be used to monitor any portion of the stated range within the equipment design rating.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE C (Continued)			
Containment (Continued)			
Effluent Radioactivity[2] - Noble Gases (from buildings as indicated above)	10^{-6} μCi/cc to 10^{3} μCi/cc	2[9]	Indication of breach

TYPE D Variables: those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident.

Variable	Range	Category	Purpose
Condensate and Feedwater System			
Main Feedwater Flow	0 to 110% design flow[10]	3	Detection of operation, analysis of cooling
Condensate Storage Tank Level	Bottom to top	3	Indication of available water for cooling
Primary Containment-Related Systems			
Suppression Chamber Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Drywell Pressure[2]	12 psia to 3 psig 0 to 110% design pressure[3]	2	To monitor operation
Suppression Pool Water Level	Top of vent to top of weir well	2	To monitor operation
Suppression Pool Water Temperature	30°F to 230°F	2	To monitor operation
Drywell Atmosphere Temperature	40°F to 440°F	2	To monitor operation
Drywell Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Main Steam System			
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	2	To provide indication of pressure boundary maintenance
Primary System Safety Relief Valve Positions, Including ADS or Flow Through or Pressure in Valve Lines	Closed–not closed or 0 to 50 psig	2	Detection of accident; boundary integrity indication

[10] Design flow is the maximum flow anticipated in normal operation.

TABLE 1 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Safety Systems			
Isolation Condenser System Shell-Side Water Level	Top to bottom	2	To monitor operation
Isolation Condenser System Valve Position	Open or closed	2	To monitor status
RCIC Flow	0 to 110% design flow[10]	2	To monitor operation
HPCI Flow	0 to 110% design flow[10]	2	To monitor operation
Core Spray System Flow	0 to 110% design flow[10]	2	To monitor operation
LPCI System Flow	0 to 110% design flow[10]	2	To monitor operation
SLCS Flow	0 to 110% design flow[10]	2	To monitor operation
SLCS Storage Tank Level	Bottom to top	2	To monitor operation
Residual Heat Removal (RHR) Systems			
RHR System Flow	0 to 110% design flow[10]	2	To monitor operation
RHR Heat Exchanger Outlet Temperature	32°F to 350°F	2	To monitor operation
Cooling Water System			
Cooling Water Temperature to ESF System Components	32°F to 200°F	2	To monitor operation
Cooling Water Flow to ESF System Components	0 to 110% design flow[10]	2	To monitor operation
Radwaste Systems			
High Radioactivity Liquid Tank Level	Top to bottom	3	To monitor operation
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To monitor operation
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic)	Voltages, currents, pressures	2[11]	To monitor system status

[11] Status indication of all Standby Power a.c. buses, d.c. buses, inverter output buses, and pneumatic supplies.

TYPE E Variables: those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Containment Radiation			
Primary Containment Area Radiation - High Range[2]	1 R/hr to 10^7 R/hr	1[6,7]	Detection of significant releases, release assessment; long-term surveillance; emergency plan actuation
Reactor Building or Secondary Containment Area Radiation[2]	10^{-1} R/hr to 10^4 R/hr for Mark I and II containments 1 R/hr to 10^7 R/hr for Mark III containment	2[9] 1[6,7]	Detection of significant releases, release assessment, long-term surveillance
Area Radiation			
Radiation Exposure Rate[2] (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[7]	Detection of significant releases, release assessment; long-term surveillance
Airborne Radioactive Materials Released from Plant			
Noble Gases and Vent Flow Rate			
• Drywell Purge, Standby Gas Treatment System Purge (for Mark I and II plants) and Secondary Containment Purge (for Mark III plants)	10^{-6} μCi/cc to 10^5 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases; release assessment
• Secondary Containment Purge (for Mark I, II, and III plants)	10^{-6} μCi/cc to 10^4 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases, release assessment
• Secondary Containment (reactor shield building annulus, if in design)	10^{-6} μCi/cc to 10^4 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases; release assessment
• Auxiliary Building (including any building containing primary system gases, e.g., waste gas decay tank)	10^{-6} μCi/cc to 10^3 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases, release assessment; long-term surveillance
• Common Plant Vent or Multipurpose Vent Discharging Any of Above Releases (if drywell or SGTS purge is included)	10^{-6} μCi/cc to 10^3 μCi/cc 0 to 110% vent design flow[10] 10^{-6} μCi/cc to 10^4 μCi/cc	2[9]	Detection of significant releases, release assessment; long-term surveillance

TABLE 1 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Airborne Radioactive Materials Released from Plant (Continued)			
Noble Gases and Vent Flow Rate (Continued)			
• All Other Identified Release Points	10^{-6} µCi/cc to 10^{2} µCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through other monitored plant vents)	2[9]	Detection of significant releases; release assessment; long-term surveillance
Particulates and Halogens			
• All Identified Plant Release Points. Sampling with Onsite Analysis Capability	10^{-3} µCi/cc to 10^{2} µCi/cc 0 to 110% vent design flow[10]	3[12]	Detection of significant releases; release assessment; long-term surveillance
Environs Radiation and Radioactivity			
Radiation Exposure Meters (continuous indication at fixed locations)	Range, location, and qualification criteria to be developed to satisfy NUREG-0654, Section II.H.5b and 6b requirements for emergency radiological monitors		Verify significant releases and local magnitudes
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} µCi/cc to 10^{-3} µCi/cc	3[13]	Release assessment; analysis
Plant and Environs Radiation (portable instrumentation)	10^{-3} R/hr to 10^{4} R/hr, photons 10^{-3} rads/hr to 10^{4} rads/hr, beta radiations and low-energy photons	3[14] 3[14]	Release assessment; analysis
Plant and Environs Radioactivity (portable instrumentation)	Multichannel gamma-ray spectrometer	3	Release assessment; analysis

[12] To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^{2} µCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^{2} µCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

[13] For estimating release rates of radioactive materials released during an accident.

[14] To monitor radiation and airborne radioactivity concentrations in many areas throughout the facility and the site environs where it is impractical to install stationary monitors capable of covering both normal and accident levels.

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Meteorology[15]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°). Starting speed 0.45 mps (1.0 mph). Damping ratio between 0.4 and 0.6, distance constant ≤ 2 meters	3	Release assessment
Wind Speed	0 to 30 mps (67 mph) ±0.22 mps (0.5 mph) accuracy for wind speeds less than 11 mps (25 mph) with a starting threshold of less than 0.45 mps (1.0 mph)	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary system, -5°C to 10°C (-9°F to 18°F) and ±0.15°C accuracy per 50-meter intervals (±0.3°F accuracy per 164-foot intervals) or analogous range for alternative stability estimates	3	Release assessment
Accident Sampling[16] **Capability (Analysis Capability On Site)**			
Primary Coolant and Sump	Grab Sample	3[4,17]	Release assessment; verification, analysis
• Gross Activity	10 μCi/ml to 10 Ci/ml		
• Gamma Spectrum	(Isotopic Analysis)		
• Boron Content	0 to 1000 ppm		
• Chloride Content	0 to 20 ppm		
• Dissolved Hydrogen or Total Gas[18]	0 to 2000 cc(STP)/kg		
• Dissolved Oxygen[18]	0 to 20 ppm		
• pH	1 to 13		
Containment Air	Grab Sample	3[4]	Release assessment; verification, analysis
• Hydrogen Content	0 to 10% 0 to 30% for inerted containments		
• Oxygen Content	0 to 30%		
• Gamma Spectrum	(Isotopic analysis)		

[15] Guidance on meteorological measurements is being developed in a Proposed Revision 1 to Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants."

[16] The time for taking and analyzing samples should be 3 hours or less from the time the decision is made to sample, except for chloride which should be within 24 hours.

[17] An installed capability should be provided for obtaining containment sump, ECCS pump room sumps, and other similar auxiliary building sump liquid samples.

[18] Applies only to primary coolant, not to sump.

APPENDIX C
ABBREVIATIONS

ADS	automatic depressurization system
APRM	average-power range monitor
ATWS	anticipated transients without scram
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CRD	control rod drive
CS	core spray
CST	condensate storage tank
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
ESF	engineered safety feature
HPCI	high-pressure coolant injection
IRM	intermediate-range monitor
LOCA	loss of coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPRM	local power range monitor
NMS	neutron monitoring system
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
OG	Owners Group
PASS	post-accident sampling system
PWR	pressurized water reactor
RCIC	reactor core isolation cooling
RCS	reactivity control system
RHR	residual heat removal
RG	Regulatory Guide

RPV	reactor pressure vessel
RWCU	reactor water cleanup unit
SBGT	standby gas treatment
SCS	suppression chamber spray
SGTS	standby gas treatment system
SLCS	standby liquid control system
SRM	source range monitor
SRV	safety relief valve