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Docket Nos. 50-275
and 50-323

Mr. John C. Morrissey
Vice President & General Counsel
Pacific Gas & Electric Company
77 Beale Street
San Francisco, California 94106

Dear Mr. Morrissey:

SUBJECT: PROPOSED REVISION 2 TO REGULATORY GUIDE 1.97 "INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT" - (DIABLO CANYON, UNITS 1 & 2)

The Regulatory staff is revising Regulatory Guide 1.97 to provide more specific guidance than that contained in the guide now being used. A draft of proposed revision 2 of this guide is enclosed (Enclosure 1). The Advisory Committee on Reactor Safeguards has reviewed proposed Revision 2 of the guide and has agreed to its issuance for comment.

Enclosure 1 is based on the results of Task Action Plan A-34, "Instrumentation for Monitoring Radiation and Process Variables During an Accident" which was initiated in June 1977 to develop more specific guidance concerning implementation of Regulatory Guide 1.97. A major addition to the current Regulatory Guide 1.97 is the identification of specific parameters to be measured, the range of the measurements and design criteria for the instruments. The enclosed draft guide also incorporates the applicable recommendations in NUREG-0578 "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The proposed Revision 2 of this guide will have a significant impact on the design of plants such as yours that are currently under review for an operating license. Therefore, we are requesting early comments on proposed Revision 2 from applicants for these plants. We have arranged meetings in Bethesda, Maryland to discuss these comments. We will meet with nine applicants for pressurized water reactors on December 13, 1979 and with five applicants for boiling water reactors on December 14, 1979.

We request that you attend the meeting as indicated in the enclosed meeting notice (Enclosure 2) to discuss your comments on the feasibility of designing and installing instruments meeting the requirements of proposed Revision 2 to Regulatory Guide 1.97 in your plant. An advance copy of major comments and the associated rationale should be given to V. Benaroya, Chief, Auxiliary Systems Branch, DSS, NRC prior to the meeting.

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OFFICE						
SURNAME					7912180	
DATE						

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If there are any questions regarding the meeting, call L. L. Kintner (301) 492-8344.

Sincerely,

/S/

S. A. Varga, Acting Assistant
Director for Light Water Reactors
Division of Project Management

Enclosures:
1. Draft of Proposed Revision 2 to Regulatory Guide 1.97
2. Meeting Notice

cc w/Encls: See next page

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Enclosure 1

Draft 1
October 15, 1979

PROPOSED REVISION 2 TO REGULATORY GUIDE 1.97

INSTRUMENTATION FOR LIGHT-WATER-COOLED NUCLEAR POWER PLANTS TO ASSESS PLANT AND ENVIRONS CONDITIONS DURING AND FOLLOWING AN ACCIDENT

A. INTRODUCTION

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," includes a requirement that instrumentation be provided to monitor variables and systems for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents and that equipment at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor including necessary instrumentation.

Criterion 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50 includes a requirement that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant.

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B. DISCUSSION

Indications of plant variables and status of systems important to safety are required by the plant operator (licensee) during accident situations to (1) provide information required to permit the operator to take pre-planned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered-safety-feature systems, and manually initiated systems are performing their intended functions, i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity; (3) provide information to the operator that will enable him to determine the potential for causing a breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundry and containment) and if a barrier has been breached; (4) furnish data for deciding on the need to take unplanned action if an automatic or manually initiated safety system is not functioning properly or the plant is not responding properly to the safety systems in operation; (5) allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of the impending threat.

At the start of an accident, it may be difficult for the operator to determine immediately what accident has occurred or is occurring and, therefore, determine the appropriate response. For this reason, reactor trip and certain other safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) have been designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant parameters required to enable the operation of manually initiated safety systems and other appropriate operator actions involving systems important to safety.

Instrumentation is also needed to provide information about some plant parameters that will alert the operator to conditions that have degraded beyond those postulated in the accident analysis so that the operator can take actions that are available to mitigate the consequences. It is not intended that the operator be encouraged to circumvent systems important to safety prematurely, but that he be adequately informed in order that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade beyond those assumed in the Final Safety Analysis Report are loss-of-coolant accidents (LOCAs), overpressure transients, ATWSs reactivity excursions, and releases of radioactive materials. Such events require that the operator understand, in a short time period, the ability of the barriers to limit radioactivity release, i.e., the potential for breach of a barrier, or an actual breach of a barrier by an accident in progress.

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required as defined by ANS-4.5, Section 3.0. It could therefore either be designed to withstand the accident environment or be protected by a local protected environment. If the environment surrounding an instrument component is the same for accident and normal operating conditions (e.g., some instrumentation components outside of containment or those in the main control room powered by a Class 1E source), the instrumentation components need no special environmental qualification.

It is important that accident-monitoring instrumentation components and their mounts that cannot be located in other than non-Seismic Category I buildings be conservatively designed for the intended service.

Parameters selected for accident monitoring can be selected so as to permit relatively few instruments to provide the essential information needed by the operator for postaccident monitoring. Further, it is prudent that a limited number of those parameters (e.g., containment pressure, primary system pressure) be monitored by instruments qualified to more stringent environmental requirements and with ranges that extend well beyond that which the selected parameters can attain under limiting conditions. It is essential that the range selections not be arbitrary but sufficiently high that the instruments will always be on-scale; for example, a range for the containment pressure monitor extending to the burst pressure of the containment in order that the operator will not be blind as to the level of containment pressure. Provisions of such instruments are important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions determined. On the other hand, we should also make sure that when a range is extended, the sensitivity and accuracy of the instrument are within acceptable limits.

Normal power plant instrumentation remaining functional for all accident conditions can provide indication, records, and (with certain types of instruments) time-history responses for many parameters important to following the course of the accident. Therefore, it is prudent to select the required accident-monitoring instrumentation from the normal power plant instrumentation to enable the operator to use, during accident situations, instruments with which he is most familiar. Since some accidents impose severe operating requirements on instrumentation components, it may be necessary to upgrade those instrumentation components to withstand the more severe operating conditions and to measure greater variations of monitored variables that may be associated with the accident if they are to be

used for both accident and normal operation. However, it is essential that instrumentation so upgraded does not compromise the accuracy and sensitivity required for normal operation. In some cases this will necessitate use of overlapping ranges of instruments to monitor the required range of the parameter to be monitored.

Draft Standard ANS-4.5, "Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station," dated September 1979, delineates criteria for determining the variables to be monitored by the control room operator, as required for safety, during the course of an accident and during the long-term stable shutdown phase followng an accident. Draft Standard ANS-4.5 was prepared by ANS 4 Working Group 4.5 with two primary objectives, (1) to address that instrumentation which permits the operator to monitor expected parameter changes in an accident period, and (2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

The standard defines four classifications of variable types for the purpose of aiding the designer in his selection of accident monitoring instrumentation and applicable criteria. (A fifth type (Type E) has been added by this regulatory guide.) The types are, (1) Type A - those variables that provide information needed for pre-planned operator actions, (2) Type B - those variables that provide information to indicate whether plant safety functions are being accomplished, (3) Type C - those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release, i.e., fuel cladding, primary coolant pressure boundary, and containment, (4) Type D - those variables that provide information to indicate the performance of individual safety systems, and (5) Type E - those variables to be monitored as required to provide defense-in-depth and for diagnosis and

other useful purposes. Type A variables have not been included in the listings of variables to be measured because they are plant specific and will depend upon the operations that the designer chooses for pre-planned manual action. The five classifications are not mutually exclusive in that a given variable (or instrument) may be included in one or more types, as well as for normal power plant operation. Where such multiple listing occurs, it is essential that instrumentation be capable of meeting the most stringent requirements.

The time phases (Phases I, II, & III) delineated in ANS-4.5 are not specified for each variable in this regulatory guide. These considerations are plant specific. It is important that the required instrumentation survive the accident environment and function as long as the information it provides is needed by the plant operator.

C. REGULATORY POSITION

The criteria, requirements, and recommendations contained in Draft Standard ANS-4.5, "Functional Requirements for Post Accident Monitoring Capability for the Control Room Operator of a Nuclear Power Generating Station," dated September 1979, are considered by the NRC staff to be generally acceptable for providing instrumentation to monitor variables and systems for accident conditions and for monitoring the reactor containment, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released during and following an accident from a nuclear power plant subject to the following:

(1) Section 2.0 of ANSI-4.5, defines the scope of the standard as containing criteria for determining the variables to be monitored by the control room operator during and following an accident that will need some operator action.

Consideration should be given to the additional requirements (e.g., emergency planning) of variables to be monitored by the plant operator (licensee) during and following an accident. Instrumentation selected for use by the plant operator for monitoring conditions of the plant are useful in an emergency situation and for other purposes and therefore should be factored into the emergency plans action level criteria.

(2) In Section 3.0 of ANS-4.5, the definition of "Type C" includes two items, (1) and (2). Item (1) includes those instruments that indicate the extent to which parameters, which indicate the potential for a breach in the containment, have exceeded the design basis values. In conjunction with the parameters that indicate the potential for a breach in the containment, the parameters that have the potential for causing a breach in the fuel cladding (e.g., core exit temperature) and the reactor coolant pressure boundary (e.g., reactor coolant pressure) should also be included. References to Type C instruments, and associated parameters to be measured, in Draft Standard ANS-4.5 should include this expanded definition, e.g., Section 4.2, Section 5.0c, Section 5.1.3, Section 5.2.2, Section 6.3.

(3) Section 3.0 of ANS-4.5 defines design basis accident events. In conjunction with the design basis accident events delineated in the standard, those events which are expected to occur one or more times during the life of a nuclear power unit and include but are not limited to loss of power to all recirculating pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, should be included.

(4) Section 4.2 of ANS-4.5, discusses the various types of variables. With regard to the discussion of Type D variables, Type D variables and instruments are within the scope of Accident Monitoring Instrumentation, although

they are not addressed in Draft Standard ANS-4.5. They are, however, along with an additional type, Type E, included in this regulatory guide. (See Tables 1, 2 and 3)

(5) Section 5.2.1(5) of ANS-4.5 pertains to the delineation of the local environment in which instruments must operate. Section 5.2.1(5) should be understood to require identification of the range of the local physical and electrical environments (e.g., normal, abnormal, accident, and post-accident) in which all of the various instrumentation components are required to operate (e.g., sensors, cables, signal conditioning equipment, indicators).

(6) Section 5.2.2 of ANS-4.5 pertains to the performance requirements for Type C instrumentation. In conjunction with Section 5.2.2, there should be:

(1) Identification of the range of the process variable. (Note - the range selected should extend well beyond that which the variable value can attain under limiting conditions)

(2) Identification of the required accuracy of measurement

(3) Identification of the required response characteristics

(4) Identification of the time interval beginning with initiation of an accident to as long as the measurement is needed

(5) Identification of the local environment (including energy supply) in which the various instrumentation components are required to operate.

(7) Section 6.1.1 of ANS-4.5, pertains to seismic qualification criteria. In conjunction with Section 6.1.1, those instrumentation components which should be seismically qualified are identified in Table 1 of this regulatory guide.

(8) Section 6.1.1 of ANS-4.5, pertains, in part, to the consideration of vibrational loads. In conjunction with Section 6.1.1, those instrumentation

components which are subjected to vibrational loads that occur as a result of plant system operation during any phase for which the instrumentation is required should be qualified to function during and/or following such vibrational loads.

(9) Section 6.1.2 of ANS-4.5, pertains to the duration that instrumentation is qualified to function. In conjunction with Section 6.1.2, Phase II instrumentation should be qualified to function for not less than 200 days unless a shorter time, based on need or component accessability for replacement or repair, can be justified.

(10) Section 6.1.6 of ANS-4.5 pertains to instrumentation location and identification. In conjunction with Section 6.1.6, accident monitoring instrumentation displays should be located in direct view of the plant operator and be distinguished from other displays. Other accident monitoring instrumentation components should be accessible to the plant operator for maintenance and repair although this may not be possible for some components in some accident conditions.

(11) Section 6.2.1 of ANS-4.5 pertains to general requirements for Type B instruments. In conjunction with Section 6.2.1, Type B instruments are essential to meeting the requirements of Criterion 13 and Criterion 64 of Appendix A to 10 CFR Part 50 and are not considered to be an "extra set of instruments which result in an additional layer of protection." Type B instruments are essential to the monitoring of variables and systems during accident conditions and in following the course of an accident.

(12) Section 6.2.2, 6.2.3, 6.2.4, 6.2.5, 6.2.6, 6.3.2, 6.3.3, 6.3.4, and 6.3.5 of ANS-4.5 pertain to variables and variable ranges for monitoring. In conjunction with the above sections, Tables 1, 2 and 3 of this regulatory guide (which includes those parameters mentioned in the above sections) should be used in developing the minimum set of instruments and their respective ranges for accident monitoring instrumentation for each nuclear power plant.

(13) Sections 6.3.2.3, 6.3.3.3, 6.3.3.4, 6.3.4.3, 6.3.4.4, 6.3.5.2, 6.3.5.3, and 6.3.5.4 of ANS-4.5 pertain, in part, to instrument transient response and relate this to compatability with recorder capabilities. In conjunction with the above sections, the transient response requirements of each measurement should be determined on a case-by-case basis by analysis of the event and operator response capabilities.

(14) Sections 6.3.3.3, 6.3.3.4, and 6.3.4.3 of ANS-4.5, pertain, in part, to measurement accuracy. In conjunction with the above sections, the accuracy of each measurement should be consistent with the requirements as established by analysis of the event being monitored.

(15) Section 6.3.6.1.1 ANS-4.5 pertains, in part, to the qualification of Type C instrumentation components. In conjunction with Section 6.3.6.1.1, the environmental envelope for qualification should be the extreme value of each environmental parameter, except the variable being monitored, as determined by the accident analysis for all accidents evaluated in the safety analysis of the plant.

(16) Table 6.4.1 of ANS-4.5 pertains to design criteria for accident monitoring instrumentation. In conjunction with Table 6.4.1, the provisions as indicated in Table 1 of this regulatory guide should be used.

D. IMPLEMENTATION

This proposed revision has been released to encourage public participation in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of the following applications that are docketed after the implementation date to be specified in the guide:

1. Preliminary Design Approval (PDA) applications and Preliminary Duplicate Design Approval (PDDA) applications.

2. Final Design Approval, Type 2 (FDA-2), applications and Final Duplicate Design Approval, Type 2 (FDDA-2), applications.

3. Manufacturing License (ML) applications.

4. Construction Permit (CP) applications except for those portions of CP applications that reference standard designs (i.e., PDA, FDA-1, FDA-2, PDDA, FDDA-1, FDDA-2, or ML) or that reference qualified base plant designs under the replication option.

In addition, the NRC staff intends to implement part or all of this guide for all operating plants, plants under construction, all PDA's and FDA's, all PDDA's and all FDDA's which may involve additions, elimination, or modification of structures, systems, or components of the facility after the construction permit, or design approval has been issued. All backfitting decisions in accordance with the positions stated in this guide will be determined by the staff on a case-by-case basis.

The implementation date of this guide will in no case be earlier than

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TABLE 1 - DESIGN CRITERIA[1]

CRITERIA	INSTRUMENTATION TYPES[2]				
	A	B	C	D	E
1. Seismic Qualification per Reg Guide 1.100	yes	yes	yes	no	no
2. Single Failure Criteria per Reg Guide 1.53	yes	yes	yes	no	no
3. Environmental Qualification per Reg Guide 1.89	yes	yes	yes[3]	yes	no[4]
4. Consider loss of off-site power	yes	yes	yes	yes	yes
5. Power source	Emr[5]	CB[6]	CB[6]	Emr[5]	Emr[5]
6. Out of service interval before accident	7	7	7	8	9
7. Portable	no	no	no[10]	no[10]	no[10]
8. Quality assurance level	11	11	11	11	11
9. Display type[12]	Con[13]	Con[13]	Con[13]	OD[14]	OD[14]
10. Display method	Rec[15]	Rec[16]	Rec[16]	Ind[17]	Ind[17,18]
11. Unique identification	yes	yes	yes	no	no
12. Periodic testing per Reg Guide 1.118	yes	yes	yes	yes	no

NOTES for Table 1:

(1) Unless different specifications are given in this regulatory guide, the specifications in ANSI N320-1979, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation," apply to the high-range containment area monitors, area exposure rate monitors in other buildings, effluent and environmental monitors, and portable instruments for measuring radiation or radioactivity.

(2) Type A - Those instruments which provide information required to take pre-planned manual actions.
Type B - Those instruments which provide information to monitor the process of accomplishing critical safety functions.
Type C - Those instruments that indicate the potential for breaching or the actual breach of the barriers to fission product release.
Type D - Those instruments that indicate the performance of individual safety systems.
Type E - Those instruments that provide information for defense-in-depth and for diagnosis or other useful purposes.

NOTES for Table 1 continued:

(3) See Paragraph 6.3.6 of Draft Standard ANS-4.5.

(4) Qualified to the conditions of its operation.

(5) Emergency power source.

(6) Critical Instrument Buss - Class 1E Power.

(7) IEEE 279-1971 Paragraph 4.11, "Exemption".

(8) Based on normal tech spec requirements on out-of-service for the safety system it serves.

(9) Not necessary to include in tech specs.

(10) Radiation monitoring outside containment may be portable if as designated.

(11) Level of quality assurance per 10 CFR Part 50, Appendix B.

(12) Continuous indication or recording displays a given variable at all times; intermittent indication or recording displays a given variable periodically; on demand indication or recording displays a given variable only when requested.

(13) Continuous display.

(14) Indication on demand.

(15) Where trend or transient information is essential to planned operator actions.

(16) Recording.

(17) Dial or digital indication.

(18) Effluent release monitors require recording, including effluent radioactivity monitors, environs exposure rate monitors, and meteorology monitors.

TABLE 2 - PWR VARIABLES

Measured Variable	Range	Type	Purpose
CORE:			
Core Exit Temperature	150°F to 2300°F	B,C	ANS-4.5, Section 6.2.3 Provide incore temperature measurements to identify localized hot areas. (Approximately 50 measurements)
Control Rod Position	Full in or not full in	D	Provide positive indication that the control rods are fully inserted. (Minimum 5 days after accident)
Neutron Flux	1 c/s to 1% power (at least one fission counter)	E	ANS-4.5, Section 6.2.2 For indication of approach to criticality.
REACTOR COOLANT SYSTEM:			
RCS Hot Leg Temperature	150°F to 750°F	B	ANS-4.5, Section 6.2.3 To aid in determining reactor system subcooling and to provide indication of natural circulation.
RCS Cold Leg Temperature	150°F to 750°F	B	ANS-4.5, Section 6.2.3 To provide indication of natural circulation; to provide input for heat balance calculations; for direct indication of ECCS injection.
RCS Pressure	15 psia to 4000 psig	B,C	ANS-4.5, Sections 6.2.3 and 6.2.4 For indication of an accident and to indicate that actions must be taken to mitigate an event.
Pressurizer Level	Bottom tangent to top tangent	B,D	ANS-4.5, Section 6.2.3 Level indication is required to assure proper operation of the pressurizer and to assure safe operation of heaters. It is also used in conjunction with changes in reactor pressure to determine leak and void sizes.
Degree of Subcooling	200°F subcooling to 35°F superheat	E	For indication of margin in core cooling and the need for emergency coolant additions or reductions as the margin changes and to obviate the necessity to consult steam tables.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
REACTOR COOLANT SYSTEM CONTINUED:			
Reactor Coolant Loop Flow	0 to 120%], -20% to 20%] design flow[1]	B,D	To provide indication that the core is being cooled.
Primary System Safety Relief Valve Positions or Flow Through or Pressure In Relief Valve Lines	Closed-not closed	B,D	By these measurements the operator knows if there is a path open for loss of coolant and that an event may be in progress.
Radiation Level in Primary Coolant Water	10 μCi/g to 10 Ci/g	C	ANS-4.5, Section 6.3.2 For early indication of fuel cladding failure and estimate of extent of damage.
CONTAINMENT:			
Containment Pressure	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	B,C	ANS-4.5, Sections 6.2.5, 6.3.3, 6.3.4, and 6.3.5 For indication of the integrity of the primary or secondary system pressure boundaries. To indicate the potential for leakage from the containment; to indicate integrity of the containment.
Containment Atmosphere Temperature	40°F to 400°F	E	For indication of the performance of the containment cooling system and adequate mixing.
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[2]	B,C	ANS-4.5, Sections 6.2.5 and 6.3.5 For indication of the need, and to measure the performance of the containment hydrogen recombiner.
Containment Isolation Valve Position	Closed-not closed	B,D	ANS-4.5, Section 6.2.5 To indicate the status of containment isolation and to provide information on the status of valves in process lines which could carry radioactive materials out of containment.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
CONTAINMENT CONTINUED:			
Containment Sump Water Level	Narrow range (sump) Wide range (bottom of containment to 600,000 gallon level equivalent)	B,C	For indication of leakage within the containment and to assure adequate inventory for performance of the ECCS.
High Range Containment Area Radiation	1 to 10^7 R/hr (60 keV to 3 MeV photons with $\pm$20% accuracy for photons of 0.1 to 3 MeV) [10^7 R/hr for photons is approximately equivalent to 10^8 rads per hour for betas and photons]	B,C	For implementation of GDC 64 and to help identify if an accident has degraded beyond calculated values and to indicate its magnitude to determine action to protect the public.
SECONDARY SYSTEMS:			
Steam Generator Pressure	From pressure for safety valve setting to plus 20% of safety valve setting	D	For indication of integrity of the secondary system, and an indication of capability for decay heat removal.
Steam Generator Level	From tube sheet to separators	D	For indication of integrity of the secondary system, and an indication of capability for decay heat removal.
Auxiliary Feedwater Flow	0 to 110% design flow[1]	D	To indicate an adequate source of water to each steam generator upon loss of main feedwater.
Main Feedwater Flow	0 to 110% design flow[1]	E	To indicate an adequate source of water to each steam generator.
Safety/Relief Valve Positions or Main Steam Flow	Closed-not closed	B,D	To indicate integrity of secondary system (vis-a-vis pipe break).
Radiation in Condenser Air Removal System	10^{-7} to 10^5 µCi/cc		To indicate leakage from the primary to the secondary system and measure of noble gas release rate to atmosphere.
Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves	10^{-7} to 10^5 µCi/cc	B,C	An indication of release from the secondary system and measure of noble gas release rate to atmosphere.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
AUXILIARY SYSTEMS:			
Containment Spray Flow	0 to 110% design flow[1]	D	For indication of system operation.
Flow in HPI System	0 to 110% design flow[1]	D	For indication of system operation.
Flow in LPI System	0 to 110% design flow[1]	D	For indication of system operation.
Emergency Coolant Water Storage Tank Level	Top to bottom	D	To determine the amount of water discharged by the ECCS. This provides indication of the nature of the accident, indication of the performance of the ECCS, and indication of the necessity for operator action.
Accumulator Tank Level	Top to bottom	D	To indicate whether the tanks have injected to the reactor coolant system.
Accumulator Isolation Valve Positions	Closed-not closed	D	To indicate state of the isolation valves. (Per Regulatory Guide 1.47)
RHR System Flow	0 to 110% design flow[1]	D	For indication of system operation.
RHR Heat Exchanger Out Temperature	32°F to 350°F	D	For indication of system operation.
Component Cooling Water Temperature	32°F to 200°F	D	For indication of system operation.
Component Cooling Water Flow	0 to 110% design flow[1]	D	For indication of system operation.
Flow in UHS Loop	0 to 110% design flow[1]	D	For indication of system operation.
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	D	For indication of system operation.
Ultimate Heat Sink Level	Plant specific	D	To ensure adequate source of cooling water.
Heat Removal by the Containment Fan Coolers	Plant specific	B	to indicate system operation
Boric Acid Charging Flow	0 to 110% design flow[1]	B	To provide indication of reactor cooling and inventory control and maintain adequate concentration for shutdown margin.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
AUXILIARY SYSTEMS CONTINUED:			
Letdown Flow	0 to 110% design flow[1]	D	For indication of reactor coolant inventory control and boron concentration control.
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	D	To monitor environmental conditions of equipment in closed spaces.
RADWASTE SYSTEMS:			
High Level Radioactive Liquid Tank Level	Top to bottom	E	Available volume to store primary coolant.
Radioactive Gas Hold-up Tank Pressure	0 to 150% of design pressure[2]	E	Available capacity to store waste gases.
VENTILATION SYSTEMS:			
Emergency Ventilation Damper Position	Open-closed status	D	To ensure proper ventilation under accident conditions.
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 180°F	D	To monitor environmental conditions of equipment in closed spaces.
POWER SUPPLIES:			
Status of Class 1E Power Supplies and Systems	Voltages and currents	D	To ensure an adequate source of electric power for safety systems.
Status of Non-Class 1E Power Supplies and Systems	Voltages and currents	E	It indicate an adequate source of electric power.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT:			
Radiation Exposure Rates	10^{-1} to 10^{4} R/hr for photons (permanently installed monitors)	E	For measurement of high-range radiation exposure rates at various locations.
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT:			
Effluent Radioactivity - Noble Gases	(Normal plus accident range for noble gas)	E	ANS-4.5 Section 6.2.6 To provide operator with information regarding release of radioactive noble gases on a continuous basis.
•Containment	10^{-7} to 10^{5} µCi/cc Xe-133 calibration		
•Secondary Containment	10^{-7} to 10^{4} µCi/cc Xe-133 calibration		
•Auxiliary Building including buildings containing primary system gases, e.g., waste gas decay tank	10^{-7} to 10^{3} µCi/cc		
•Other Release Points (including fuel handling area if separage from auxiliary building)	10^{-7} to 10^{2} µCi/cc (permanently installed monitors)		
Effluent Radioactivity - High Range Radiohalogens and Particulates		E	To provide the operator with information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by monitoring (measurements) of samples for radiohalogens and for particulates.
•Untreated Effluents	10^{-3} to 10^{2} µCi/cc		
•HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems	10^{-3} to 10 µCi/cc		
•HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems	10^{-3} to 1 µCi/cc (permanently installed monitors)		

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT CONTINUED:			
Environs Radioactivity - High Range Exposure Rate	10^{-3} to 10^{2} R/hr (60 keV to 3 MeV) (permanently installed monitors)	E	For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability, approximately 16 to 20 locations - site dependent.
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} µCi/cc for both radiohalogens and particulates (permanently installed monitors)	E	For estimating releases rates of radio active materials released during an accident from unidentified release paths (not covered by effluent monitors). Continuous collection of representative samples followed by monitoring (measurements) of the samples. (Approximately 16 to 20 locations)
AIRBORNE RADIOACTIVE MATERIALS RELEASES FROM THE PLANT CONTINUED:			
Plant and Environs Radioactivity (portable instruments	Normal Range 0.1 to 10^{4} mR/hr photons 10^{-9} to 10^{-4} µCi/cc particulates 10^{-9} to 10^{-4} µCi/cc iodine High Range 0.1 to 10^{4} R/hr photons 0.1 to 10^{4} rads/hr betas and low energy photons	E	During and following an accident, to monitor radiation and airborne radioactivit concentrations in many areas throughout the facility where is impractical to install stationary monitors capable of covering both normal and accident levels
	100-channel gamma-ray spectrometer	E	During and following an accident to rapidly scope the composition of gamma-emitting sources.

TABLE 2 - PWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
POST-ACCIDENT SAMPLING CAPABILITY:			
Primary Coolant Sumps Containment Air	As required based on Reg Guide 1.4 guidelines	N/A	Provide means for safe and convenient sampling. These provisions should include: 1. shielding to maintain radiation doses ALARA, 2. sample containers with container-sampling port connector compatibility 3. capability of sampling under primary system pressure and negative pressure 4. handling and transport capability, ar 5. pre-arrangement for analysis and interpretation.
POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE):	1. gamma-ray spectrum 2. pH 3. hydrogen 4. oxygen 5. boron	N/A	
METEOROLOGY:			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1 mph)	E	For determining affluent transport direction for emergency planning, dose assessment, and source estimates.
Wind Speed	0 to 30 mps (67 mph) (±0.22 mps (0.5 mph) accuracy for wind speeds less that 11 mps (25 m ph), with a starting threshold of less than 0.45 mps (1 mph)	E	For determining effluent travel speed and dilution for emergency planning, doses assessments and source estimates.
Vertical Temperature Difference	-9°F to +9°F (±0.3°F accuracy per 164 foot intervals	E	For determining effluent diffusion rates for emergency planning, dose assessments and source estimates.
Precipitation	Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value - 0.01" resolution	E	For determining effluent transport and ground deposition for emergency planning.

Notes for Table 2 -

(1) Design flow - the maximum flow anticipated in normal operation.

(2) Design pressure - that value corresponding to ASME code values which are obtained at or below code allowable material design stress values.

TABLE 3 - BWR VARIABLES

Measured Variable	Range	Type	Purpose
CORE:			
Control Rod Position	Full in or not full in	D	Provide positive indication that the control rods are fully inserted. (Minimum of 2 hours after accident)
Neutron Flux	1 c/s to 1% power (at least one fission counter)	B	ANS-4.5, Section 6.2.2 For indication of approach to criticality.
REACTOR COOLANT SYSTEM:			
RCS Pressure	15 psia to 2000 psig	B,C	ANS-4.5, Sections 6.2.3, 6.2.4, 6.3.3 and 6.3.5 For indication of an accident and to indicate that actions must be taken to mitigate an event.
Coolant Level in the Reactor	Bottom of core support plate to above top of discharge plenum	B	ANS-4.5, Section 6.2.3 For indication of fuel submergency for a LOCA event.
Main Steamline Flow	0 to 120% design flow[1]	B	To provide an indication of the integrity of the pressure boundary.
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	B	To provide an indication of the pressure boundary and containment integrity.
Primary System Safety Relief Valve Positions including ADS or Flow Through or Pressure in Valve Lines	Closed-not closed or 0 to 50 psig	B,D	By these measurements the operator knows if there is a path open for loss of coolant and that an event may be in progress.
Radiation Level in Coolant	10 μCi/g to 10 Ci/g	C	ANS-4.5, Section 6.3.2 For early indication of fuel cladding failure and estimate of extent of damage.

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
CONTAINMENT:			
Primary Containment Pressure	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	B,C	ANS-4.5, Sections 6.2.5, 6.3.3, 6.3.4, and 6.3.5 For indication of the intergrity of the primary containment pressure boundary; to indicate the potential for leakage from the containment.
Containment and Drywell Hydrogen Concentration	0 to 10% (capability of operating from 12 psia to maximum design pressure[2]	B,C	ANS-4.5, Sections 6.2.5, and 6.3.5 For indication of the need for, and a measurement of the performance of the containment hydrogen recombiner and to verify the operation of the mixing system.
Containment Isolation Valve Position	Closed-not closed	B,D	ANS-4.5, Section 6.2.5 To indicate the status of containment isolation and to provide information on the status of valves in process lines which could carry radioactive materials out of containment.
Suppression Pool Water Level	Top of vent to top of weir well	B	ANS-4.5, Section 6.3.3
Suppression Pool Water Temperature	50°F to 250°F	B	To assure proper temperature for NPSH of ECCS. To verity the operation of the makup system.
Drywell Pressure	12 psia to 3 psig 0 to 110% design pressure	B E	ANS-4.5. Section 6.3.3 Diagnosis of impact of accident on drywell structure.
Drywell Drain Sumps Level (Identified and Unidentified Leakage)	Bottom to top	B,C	ANS-4.5, Section 6.3.3
High Range Containment Area Radiation	1 to 10^7 R/hr (60 keV to 3 MeV photons with ±20% accuracy for photons of 0.1 to 3 MeV)[10^7 R/hr for photons is approximately equivalent to 10^8 rads/hr for betas and photons]	B,C	To help identify if an accident has degraded beyond calculated values and indicate its magnitude and to determine action to protect the public.

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
POWER CONVERSION SYSTEMS:			
Main Feedwater Flow	0 to 110% design flow[1]	E	To indicate an adequate source of water to the reactor.
Condensate Storage Tank Level	Bottom to top	E	To indicate available water for core cooling.
AUXILIARY SYSTEMS:			
Containment Spray Flow	0 to 110% design flow[1]	D	For indication of system operation.
Steam Flow to RCIC	0 to 110% design flow[1]	E	To verify that adequate steam is available for the system to perform its function.
RCIC Flow	0 to 110% design flow[1]	D	For indication of system operation.
RHR System Flow	0 to 110% design flow[1]	D	For indication of system operation.
RHR Heat Exchanger Outlet Temperature	32°F to 350°F	D	For indication of system operation.
Service Cooling Water Temperature	32°F to 200°F	D	For indication of system operation.
Service Cooling Water Flow	0 to 110% design flow[1]	D	For indication of system operation.
Flow in UHS Loop	0 to 110% design flow[1]	D	For indication of system operation.
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	D	For indication of system operation.
Ultimate Heat Sink Level	Plant specific	D	To ensure adequate source of cooling water.
SLCS Storage Tank Level	Bottom to top	E	To provide indication of inventory for boron injection for shutdown.
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	D	To monitor potential for failure of equipment in closed spaces due to flooding.

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
RADWASTE SYSTEMS:			
High Radioactivity Liquid Tank Level	Top to bottom	E	Available volume to store primary coolant.
Charcoal Delay Gas System Gas Flow or Radioactivity Level	As required	E	To monitor performance of system.
VENTILATION SYSTEMS:			
Emergency Ventilation Damper Position	Open-closed status	D	To ensure proper ventilation under accident conditions.
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 130°F	B	To monitor environmental conditions of equipment in closed spaces.
POWER SUPPLIES:			
Status of Class 1E Power Supplies and Systems	Voltages and currents	D	To ensure an adequate source of electric power for safety systems
Status of Non-Class 1E Power Supplies and Systems	Voltages and currents	E	To indicate an adequage source of electric power.
RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY RELATED EQUIPMENT:			
Radiation Exposure Rates	10^{-1} to 10^{4} R/hr for photons (permanently installed monitors)	E	For measurement of high-range radiation exposure rates at various locations.

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
AIRBORNE RADIOACTIVE MATERIALS RELEASES FROM THE PLANT:			
Effluent Radioactivity - Noble Gases	(Normal plus accident range for noble gas)	E	ANS-4.5, Section 6.2.6 To provide operator with information regarding release of radioactive noble gases on a continuous basis.
.Containment Exhaust Vent and Standby Gas Treatment System Vent	10^{-7} to 10^{5} µCi/cc Xe-133 calibration		
.Other Release Points [including fuel handling building, auxiliary building, and turbine building]	10^{-7} to 10^{2} µCi/cc Xe-133 calibration (permanently installed monitors)		
Effluent Radioactivity - High Range Radiohalogens and Particulates		E	To provide the operator with information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by monitoring (measurements) of samples for radiohalogens and for particulates.
.Untreated Effluents	10^{-3} to 10^{2} µCi/cc		
.HEPA Filters, minimum of 2" of TEDA impregnated charcoal, non-ESF systems	10^{-3} to 10 µCi/cc		
.HEPA Filters, minimum of 4" of TEDA impregnated charcoal, ESF systems	10^{-3} to 1 µCi/cc (permanently installed monitors)		
Environs Radioactivity - High Range Exposure Rate	10^{-3} to 10^{2} R/hr (60 keV to 3 MeV) (permanently installed monitors)	E	For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability, approximately 16 to 20 locations - site dependent.
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} µCi/cc for both radiohalogens and particulates (permanently installed monitors)	E	For estimating releases rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors). Continuous collection of representative samples followed by monitoring (measurements) of the samples. (Approximately 16 to 20 locations)

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
AIRBORNE RADIOACTIVE MATERIALS RELEASES FROM THE PLANT CONTINUED:			
Plant and Environs Radioactivity (portable instruments)	Normal Range 0.1 to 10^4 mR/hr photons 10^{-9} to 10^{-4} μCi/cc particulates 10^{-9} to 10^{-4} μCi/cc iodine High Range 0.1 to 10^4 R/hr photons 0.1 to 10^4 rads/hr betas and low energy photons	E	During and following an accident, to monitor radiation and airborne radioactivity concentrations in many areas throughout the facility where is impractical to install stationary monitors capable of covering both normal and accident levels.
	100-channel gamma-ray spectrometer	E	During and following an accident to rapidly scope the composition of gamma-emitting sources.
POST-ACCIDENT SAMPLING CAPABILITY:			
Primary Coolant Suppression Pool Containment Air	As required based on Reg Guide 1.3 guidelines	N/A	Provide means for safe and convenient sampling. These provisions should include: 1. shielding to maintain radiation doses ALARA, 2. sample containers with container-sampling port connector compatibility 3. capability of sampling under primary system pressure and negative pressure 4. handling and transport capability, an 5. pre-arrangement for analysis and interpretation.
POST-ACCIDENT ANALYSIS CAPABILITY (ONSITE):	1. gamma-ray spectrum 2. pH 3. hydrogen 4. oxygen	N/A	
METEOROLOGY:			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°. Starting speed 0.45 mps (1 mph)	E	For determining affluent transport direction for emergency planning, dose assessment, and source estimates.
Wind Speed	0 to 30 mps (67 mph) (±0.22 mps (0.5 mph) accuracy for wind speeds less that 11 mps (25 m ph), with a starting threshold of less than 0.45 mps (1 mph)	E	For determining effluent travel speed and dilution for emergency planning, doses assessments and source estimates.

TABLE 3 - BWR VARIABLES continued -

Measured Variable	Range	Type	Purpose
METEOROLOGY CONTINUED:			
Vertical Temperature Difference	-9°F to +9°F (±0.3°F accuracy per 164 foot intervals	E	For determining effluent diffusion rates for emergency planning, dose assessments and source estimates.
Precipitation	Recording rain gage with range sufficient to assure accuracy of total accumulation within 10% of recorded value - 0.01" resolution	E	For determining effluent transport and ground deposition for emergency planning

Notes for Table 3 -

(1) Design flow - the maximum flow anticipated in normal operation.

(2) Design pressure - that value corresponding to ASME code values which are obtained at or below code allowable material design stress values.

ANS-4.5

Draft 3
September, 1979

DRAFT

CAUTION NOTICE: This Standard is being prepared or reviewed and has not been approved by ANS. It is subject to revision or withdrawal before issue.

DRAFT

American National Standard

FUNCTIONAL REQUIREMENTS FOR POST ACCIDENT MONITORING CAPABILITY FOR THE CONTROL ROOM OPERATOR OF A NUCLEAR POWER GENERATING STATION

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FOREWORD

ANS 4 established Working Group 4.5 in late July 1979 to prepare a draft standard on Accident Monitoring Instrumentation which would complement other standards, but be broader in nature by including economic considerations. Two primary objectives were 1) to address that instrumentation which permits the operator to monitor expected parameter changes in the accident period, and 2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events.

ANS 4.5 began work on July 30th and met for 13 working days in a seven week period. In addition, a Design Criteria subgroup met for two days in this same period.

As presented, this draft standard provides:

1. a list of functions to be performed (design basis section 5.0)

2. a framework to determine those variables to be monitored (design basis section 5.0)

3. an identification of three time periods of interest (definitions 3.0)

4. an identification of four variable types (definitions 3.0)

5. a delineation of applicable design criteria for the variables to be monitored (design criteria section 6.0)

No identification of specific Type A monitored variables is provided in this standard. Recommendations for Type B and Type C monitored variables are provided in Section 6.0.

The significant issues in the development of this standard have been:

1. the scope of the document in terms of applicability to the control room operator or the plant operator (licensee). The work group chose a control room operator scope.

2. the pre-planned operator actions designated by the accident analyses in Chapter 15 of a plant's FSAR and the not previously planned operator action that may be required during unforeseen events. The Working Group established Type A instrumentation for the former, and Type B or C instrumentation for the latter.

3. The monitoring of actual fission product barrier integrity and the potential for breach of a given barrier. The work group chose monitoring of actual breach for the fuel, reactor coolant system, and containment barrier, but only the potential breach of the containment barrier.

4. the degree of alignment of accident monitoring instrumentation with IEEE Class 1E (ANS Class EC-3) and whether an intermediate class is needed between 1E and non-1E.

5. whether a list of variables should be included as an appendix to the standard:

   a. a list of only Type C parameters
   b. a list of Type A, B, C and D parameters

6. the definition of instrument types B and D and whether these types should be included in the standard.

The membership of the Working Group is as follows:

L. Stanley, Chairman
T. Timmons, Vice Chairman and Correspondent

D. Sommers

E. Wenzinger

D. Lambert

R. Bauerle

J. Castanes

M. Wolpert

H. Mumford

X. Polanski

E. Dowling

Additional input has been provided to the Working Group by industry, university, and government participants throughout the meetings. The Work Group is very appreciative of this assistance.

1.0 Introduction

The Code of Federal Regulations requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to assure adequate safety. The purpose of this standard is to establish criteria for the selection of that instrumentation. These criteria are based on the sequence and duration of the phases through which an accident progresses. The control room operator may have different information requirements for each phase of an accident.

This standard presents criteria for monitoring the response of the plant to design basis events. It also presents criteria for monitoring the integrity of fission product barriers under conditions which have degraded beyond the design bases. This fission product barrier monitoring is considered to be an extra set of instrumentation beyond that required for satisfactorily monitoring accident scenarios postulated in the plant safety analysis.

Throughout these criteria, three verbs have been used to indicate the degree of rigor intended by the specific criterion. The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

2.0 SCOPE

This standard contains criteria for determining the variables to be monitored by the control room operator, as required for safety, during the course of an accident and during the long-term stable shutdown phase following the accident. Also included are criteria for determining the requirements for the equipment used to monitor those variables.

The scope of the standard is limited to onsite environment and process monitoring. Emergency preparedness planning is, or will be, covered by other standards.

3.0 DEFINITIONS

Phase I — That portion of the accident extending from the initiation of the accident to that point at which the plant is in a controlled condition.

Phase II — That portion of the accident extending from the time at which the plant is in a controlled condition to that point at which personnel access to the location of the accident is possible.

Phase III — That portion of the accident extending from the time at which personnel access to the location of the accident is possible to the time at which the plant has returned to operating status or been decommissioned.

Type A — Instruments - Those instruments which provide the information required to permit the control room operator to take the pre-planned manual actions to accomplish safe plant shutdown for design basis accident events and to maintain long term plant stability.

Type B — Those instruments which provide to the control room operator information to monitor the process of accomplishing critical safety functions, i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, maintaining containment integrity and radioactive effluent control.

Type C — Those instruments that indicate in the control room (1) the extent to which parameters, which have the potential for causing a breach of the final fission product barrier (i.e., the containment), have exceeded the design basis values, or (2) that a fission product barrier (i.e., fuel clad, reactor coolant pressure boundary or the containment) has been breached.

Type D Those instruments which indicate to the control room operator the performance of individual safety systems.

Design Basis Accident Events

Those events postulated in the plant safety analyses, any one of which may occur during the lifetime of a particular plant, excluding those events which are expected to occur during a calendar year for a particular plant; and those events that are not expected to occur but are postulated in the plant safety analyses because their consequences would include the potential for the release of significant amounts of radioactive material.

4.0 DISCUSSION

It is the philosophy of this Standard that instrumentation is required to monitor plant performance during and after an accident. The purposes of the accident monitoring instrumentation are enumerated in Section 5.0, Design Basis. This Standard specifies the plant safety functions to be performed and the criteria to be used by the designer in selecting the variables to be monitored.

Certain concepts have been established to aid the system designer in the selection of variables to monitor the course of an accident and to arrive at appropriate design criteria for instruments to monitor these variables.

4.1 Planned Versus Unplanned Operator Actions

The plant safety analysis defines the accident scenarios from which the safety system design bases and the planned or anticipated operator actions are derived. Accident monitoring instrumentation is provided to permit the operator to take required actions to address these analyzed situations. However, instrumentation must also be provided for unplanned situations, (i.e., to ensure that, should plant conditions evolve differently than predicted by the safety analysis, the operator has sufficient information to monitor the course of the event). Instrumentation must also be provided to indicate to the operator if fission product barrier integrity has degraded beyond the prescribed limits of the Safety Analysis.

4.2 Variable Types

Four classifications of variables have been identified. Operator manual actions during accidents included in the plant safety analysis are anticipated or pre-planned. Those variables that provide information needed by the operator to perform these manual actions are designated

Type A. Those variables needed to assess that the plant safety functions are being accomplished, as identified in the plant safety analysis, are designated Type B. Variables used to monitor for the actual gross breach of one of the fission product barriers or the potential breach of the final fission product barrier (containment) are designated Type C. Type C variables used to monitor the potential breach of containment have an arbitrarily-determined, extended range. The fourth classification, Type D, consists of those variables monitored to ascertain that the safety systems are performing as designed. Type D variables are less important than Types A, B and C for accident monitoring since safety system performance only infers safety function accomplishment. Type D variables and instruments are not considered to be within the scope of Accident Monitoring Instrumentation. Guidance on the selection of Type D variables and the specification of appropriate design criteria are not given in this standard. This guidance is provided in standards for design of safety systems (e.g. IEEE-603, ANSI N18.2, etc). The four classifications are not mutually exclusive in that a given variable (or instrument) may be included in one or more types. This differentiation by variable type is intended only to guide the designer in his selection of accident monitoring variables and applicable criteria.

4.3 Accident Phases

The typical accident sequence has been subdivided into three phases: Phase I covers the initial portion of the accident, Phase II covers the stable long-term cooling portion of the accident up to the time where personnel access is possible; and Phase III addresses the period following personnel access to the accident area. This sub-division has been made so that variable selection and design criteria application can reflect the differing conditions which characterize these three phases. For example, Phase I can be anticipated to be of relatively short duration, having relatively severe plant conditions, and allowing no personnel access to the accident area. Phase II is expected to be of longer duration, to require a significant number of operator actions, under milder plant conditions, but with still no personnel access to the accident area. Phase III is expected to be of even longer duration where

personnel access is possible. Different design criteria are then appropriate for each of the three phases. In this Standard, guidance and criteria are provided for Phases I and II.

5.0 Design Basis

The plant designer shall perform and document an analysis to select accident monitoring instruments. He shall identify instruments required by his design to enable the control room operator to:

A. Perform pre-planned manual actions.

B. Ascertain the performance of:

(1) Reactivity control
(2) Reactor core cooling
(3) Reactor coolant system integrity
(4) Containment integrity
(5) Radioactive effluent control

C. Ascertain the extent to which parameters, which have the potential for causing a breach of the containment, have exceeded the design basis values and to ascertain that a fission product barrier (i.e. fuel clad, reactor coolant system pressure boundary or the containment) has been breached.

5.1 Variable Selection for Phases I and II

The process for selection of the Accident Monitoring Instrumentation variables shall include:

5.1.1 For Type A

1) Identification of the postulated accidents for which manual action is required.

2) Identification of planned operator actions

3) Identification of the monitored variables needed for planned operator actions.

5.1.2 For Type B

1) Identification of the monitored variables that provide the most direct indication needed to assess the accomplishments of:

   a. Reactivity Control
   b. Reactor Core Cooling
   c. Reactor Coolant System Integrity
   d. Containment Integrity
   e. Radioactive Effluent Control

Guidance on the selection of these variables is provided in Section 6.0.

5.1.3 For Type C

1) Identification of the monitored variables that provide the most direct indication of a gross breach of a fission product barrier or of an approach to breach of the containment. These instruments may have extended ranges. Guidance on the selection of these variables is provided in Section 6.0.

5.1.4 Phase III Access

Prior to the termination of Phase II, the ability to gain access to the location of the accident must be determined. Instrumentation that indicates when conditions are acceptable for personnel access shall be identified.

5.2 PERFORMANCE REQUIREMENTS FOR PHASES I AND II

The process for determining performance requirements of Accident Monitoring Instrumentation shall include, as a minimum, the following considerations:

5.2.1 For Types A and B

1) Identification of the expected range of the process variable.
2) Identification of the required accuracy of measurement.
3) Identification of the required response characteristics.
4) Identification of the time interval during which the measurement is needed.
5) Identification of the local environment in which the instrument must operate.

5.2.2 For Type C

The performance requirements for these instruments are arbitrary. Guidance on these requirements is provided in Section 6.0.

6.0 DESIGN CRITERIA

6.1 GENERAL DESIGN CRITERIA

6.1.1 SEISMIC QUALIFICATIONS

Accident monitoring instrumentation that is to be seismically qualified shall be qualified according to IEEE Standard 344-1975. The instrumentation shall be qualified to continue to function within the required accuracy following, but not necessarily during, a safe shutdown earthquake. Vibration loads which occur as a result of plant system operation during any phase for which the instrument is required shall be considered.

6.1.2 DURATION

Accident monitoring instrumentation shall be qualified for the length of time its function is required. Unless other times can be justified, Phase II instrumentation shall be qualified to function for not less than 100 days. A shorter time may be acceptable if instrumentation equipment replacement or repair can be accomplished within an acceptable out-of-service time, taking into consideration the environment where the equipment is located.

6.1.3 DIRECT MEASUREMENT

To the extent practical, accident monitoring instrumentation inputs shall be from sensors that directly measure the desired variables.

6.1.4 MINIMIZING MEASUREMENTS

To the extent practical, the same instruments shall be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during an accident situation, instruments with which he is most familiar. However, where the required range of accident monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments shall be used.

6.1.5 INSTALLATION

Permanently installed instrument equipment is required for those instruments required to function during Phase I. Permanently installed instrumentation systems need not be provided for those functions required only for Phases II and III providing it can be demonstrated that the instrument components can be installed when required, considering the local environment.

6.1.6 INSTRUMENTATION LOCATION AND IDENTIFICATION

Accident monitoring instrumentation shall be located accessible to the operator and be distinguishable from other displays so that in an accident situation, the operator can rapidly identify the accident monitoring instrumentation.

6.1.7 EQUIPMENT REPAIR

The accident monitoring instrumentation shall be designed to facilitate timely recognition, location, replacement, and repair or adjustment of malfunctioning equipment.

6.1.8 TEST AND CALIBRATION

6.1.8.1 Test

Capability shall be provided for testing, with a high degree of confidence, the operational availability of each instrument channel during plant operation. This may be accomplished in various ways, for example:

1. By observing the effect of perturbing the monitored variable.

2. By observing the effect of introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable.

3. By cross-checking between channels that bear a known relationship to each other.

Where testing during reactor operation is not possible, it must be shown that there is no practical way of implementing such a requirement without adversely affecting plant safety or operability. In addition, it must be shown that the probability of a failure of the component which is not periodically tested is acceptably low and that such testing can be routinely performed when the reactor is shut down.

6.1.8.2 Calibration

Capability shall be provided for calibration of each instrument channel during normal plant operation or during shutdown as determined by the required interval between calibrations. Equipment that does not require periodic calibration is exempt from this requirement.

6.1.9 DIVERSITY

Diversity is preferred in fufilling redundancy requirements.

6.1.10 REDUNDANT READOUT AMBIGUITY

Where a disagreement between redundant displays could lead the operator to defeat or fail to accomplish a required safety function, additional information shall be provided to allow the operator to deduce the actual conditions that are required for him to perform his role. This may be accomplished by providing an independent channel which monitors a different variable bearing a known relationship to the redundant channel or by providing an additional independent channel of instrumentation of the same variable or by providing the capability for the operator to perturb the measured variable and determine by observation of the response which instrumentation display has failed.

6.2 TYPE B INSTRUMENTS

6.2.1 GENERAL REQUIREMENTS

The number of instruments used shall be only that minimum set needed to adequately monitor the accomplishment of the following functions:

a. Reactivity Control
b. Reactor Core Cooling
c. Reactor Coolant System Integrity
d. Containment Integrity
e. Radioactive Effluent Control

Type B instruments provide control room indication beyond that which may be required for any preplanned operator action and as such constitute an extra set of instrumentation which results in an additional layer of protection.

6.2.2 VARIABLES FOR REACTIVITY CONTROL MONITORING

The measured variable shall indicate the accomplishment of control of reactivity in the core. The measured variable should be neutron flux. The range of measurement should extend from one count per second on the source range instrument to the intermediate range instrument value corresponding to 1.% of full reactor power. This range is intended to encompass all neutron flux levels at which the core can be subcritical.

6.2.3 VARIABLES FOR CORE COOLING MONITORING

The measured variables shall indicate the accomplishment of core cooling. For the PWR, the measured variables should be T_H, T_C, pressurizer level, and pressurizer pressure. For the BWR, the measured variable should be reactor vessel water level. Incore thermocouple monitoring should be considered for inclusion as a desireable variable to ascertain cooling.

6.2.4 VARIABLES FOR REACTOR COOLANT SYSTEM INTEGRITY

The measured variable shall indicate the accomplishment of RCS Integrity. The measured variable should be primary system pressure.

6.2.5 VARIABLES FOR CONTAINMENT INTEGRITY

The measured variables shall indicate the accomplishment of containment integrity. The measured variables should be containment hydrogen concentration, containment pressure and containment isolation valve positions.

6.2.6 VARIABLES FOR RADIOACTIVE EFFLUENT CONTROL

The measured variables shall indicate the accomplishment of radioactive effluent control. The measured variables should be noble gas monitoring of the identified plant release points.

6.3 TYPE C INSTRUMENTS

6.3.1 Type C instruments shall meet the following criteria:

6.3.1.1 The number of instruments used shall be only that minimum set needed to adequately monitor the three barriers;

6.3.1.2 Each measurement shall be as direct as possible;

6.3.1.3 Any chosen measurement shall detect a gross breach of one or more barriers (i.e., > 1 percent fuel clad failure, a RCS pressure boundary breach producing a loss of reactor coolant inventory in excess of the normal makeup capability, a containment breach capable of producing radiation releases in excess of 10 CFR 100 at the site boundary using TID-14844 source terms); the ranges established for Type C instruments are not mechanistically related to a postulated accident scenario.

6.3.1.4 During the period of need for Type C instruments, no other failures shall be assumed in the analysis beyond the assumed breach of a barrier coincident with loss of off-site power;

6.3.2 Fuel Clad Barrier Monitoring

6.3.2.1 The measured variable shall detect and alarm the breach of the fuel clad barrier (i.e., > 1 percent fuel clad failure);

6.3.2.2 Operator sampling of reactor coolant shall be used as the means to verify the measured variable alarm.

6.3.2.3 The measured variable should be reactor coolant system radiation. The instrument range should be equivalent to the fuel clad gap activity corresponding to 0.5% to 5% failed fuel. A narrow accuracy band for this measured variable is not significant in achieving this function; for example, ±50% to ±100% accuracy of reading should be acceptable. Instrument transient response should be compatible with its recorder.

6.3.3 Reactor Coolant System Pressure Boundary Monitoring

6.3.3.1 The measured variable(s) shall detect and alarm a breach of the reactor coolant system that produces a loss of coolant inventory in excess of normal makeup capability. The spectrum of RCS pressure boundary breaches extends up to and includes the largest double-ended pipe break.

6.3.3.2 The means used to detect RCS pressure boundary breach should include one RCS pressure boundary variable and one containment variable over the full spectrum of break sizes.

6.3.3.3 The measured PWR variables should be RCS pressure and containment pressure. The instrument range should be the design pressure plus a specified margin ($\leq$ 10%). Normal instrument accuracy is acceptable for these monitors. Instrument transient response should be compatible with its recorder.

6.3.3.4 The measured BWR variables should be drywell pressure and containment sump level. The instrument range should be design values plus a specified margin ($\leq$ 10%). Normal instrument accuracy is acceptable for these monitors. Instrument transient response should be compatible with its recorder.

6.3.4 <u>Containment Pressure Boundary Monitoring</u>

6.3.4.1 The measured variable(s) shall detect and alarm a breach of the containment pressure boundary that is capable of producing radiation releases in excess of 10 CFR 100 at the site boundary using TID-14844 source terms.

6.3.4.2 The means used to detect containment pressure boundary breach should include containment pressure (BWR and PWR), environs radiation monitoring for gross gamma (PWR), and secondary containment air space radiation monitoring for gross gamma (BWR).

6.3.4.3 The instrument range for containment pressure should be design pressure plus a specified margin ($\leq$ 10%). Normal instrument accuracy is acceptable for this monitor. Instrument transient response should be compatible with its recorder.

6.3.4.4 The instrument range for environs radiation monitoring should be 10^{-3} to 10^{2} R/hr. The instrument range for secondary containment air space radiation monitoring should correspond to the 10 CFR 100 value for off-site doses. Instrument accuracy should be $\pm$ 1/2 decade (100 Kev-3 Mev). Instrument transient response should be compatible with its recorder.

6.3.5 <u>Potential Breach of the Final Fission Product Barrier</u>

6.3.5.1 The measured variables should be containment pressure, containment hydrogen concentration, and RCS pressure for indicating the potential for causing a breach of the final fission product barrier (i.e., containment).

6.3.5.2 An arbitrary range of 3 times design pressure for concrete and 4 times design pressure for steel should be used for containment pressure. Instrument accuracy should be ± 10% of span. Instrument transient response should be compatible with its recorder.

6.3.5.3 An arbitrary range of 0-10 volume percent hydrogen should be used for containment hydrogen concentration. Instrument accuracy should be ± 10% of span. Instrument transient response should be compatiable with its recorder.

6.3.5.4 An arbitrary range of 1.5 times design pressure should be used for RCS pressure. Instrument accuracy should be ± 10% of span. Instrument transient response should be compatible with its recorder.

6.3.6 INSTRUMENT QUALIFICATION

6.3.6.1 Type C instruments shall be qualified in the same manner as Type A instruments except:

6.3.6.1.1 For purposes of equipment qualification, the assumed maximum value of the monitored parameter shall be the value equal to the maximum range for the instrument. The monitored parameter shall be assumed to approach this peak by extrapolating the most severe initial ramp associated with the Design Basis Accidents. The decay for this parameter shall be considered proportional to the decay for this parameter associated with the Design Basis Accidents. No additional qualification margin needs to be added to the extended range parameter. See figure 6.3-1. All environmental envelopes except that pertaining to the parameter measured by the instrument shall be those associated with the Design Basis Accidents.

6.4 SPECIFIC DESIGN CRITERIA

Design Criteria specific to particular accident phases and variable types are presented in Table 6.4-1.

TABLE 6.4.1

DESIGN CRITERIA

CRITERION	PHASE I VARIABLE TYPE A	B	C	PHASE II VARIABLE TYPE A	B	C
1. Qualify seismically to IEEE 344-75 (operate after SSE)	Yes	Yes	No	Yes	No	No
2. Meet single failure per IEEE 379-77	Yes	Yes	No	Yes	Yes	No
3. Qualify environmentally to IEEE 323-74	Yes	Yes	Yes[1]	Yes	Yes	Yes[1]
4. Consider loss of off-site power	Yes	Yes	Yes	Yes	No	No
5. Power source	Emergency	Emerg.	Emerg.	Emerg.	Normal[6]	Normal[6]
6. Out of service interval - prior to accident	(2)	(2)	≤72 Hr[3]	(2)	(2)	≤72 Hrs[3]
7. Out of service interval - during accident	None	None	≤2 Hr	(2)	(2)	≤2 Hrs

TABLE 6.4-1 (Continued)

DESIGN CRITERIA

CRITERION	PHASE 1 VARIABLE TYPE A	PHASE 1 VARIABLE TYPE B	PHASE 1 VARIABLE TYPE C	PHASE II VARIABLE TYPE A	PHASE II VARIABLE TYPE B	PHASE II VARIABLE TYPE C
8. Portable instrumentation	No	No	No[7]	Yes	Yes	Yes
9. Level of quality assurance	ANSI N45.2	ANSI N45.2	ANSI N45.2	ANSI N45.2	ANSI N45.2	ANSI N45.2
10. Display type[4]	Continuous	Continuous	Continuous	Continuous	Continuous	On demand
11. Display method	Recording[5]	Recording	Indicator	Recording[5]	Indicator	Indicator
12. Identification as accident monitoring type	Yes	Yes	Yes	Yes	Yes	Yes
13. Periodic Test per IEEE-338-1977	Yes	Yes	Yes	Yes	Yes	Yes

NOTES: (1) See Paragraph 6.3.6 of this Standard.
(2) IEEE 279-1971 Paragraph 4.11 Exemption

NOTES TO TABLE 6.4-1 (Continued)

(3) Based on normal tech spec requirements on out-of-service safety systems.
(4) Continuous indication or recording displays a given variable at all times; intermittent indication or recording displays a given variable periodically; on demand indication or recording displays a given variable only when requested.
(5) Where trend or transient information is essential to planned operator actions.
(6) May be manually connected to emergency buss
(7) Radiation monitoring outside containment may be portable.

Figure 6.3-1. Typical Environmental Qualification Envelope for Type C Instruments