ENCLOSURE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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REGULATORY GUIDE DISTRIBUTION LIST (DIVISION 1)

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," is being developed to describe a method acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following accidents in a nuclear power plant. This proposed revision is being released for comment to encourage public participation in its development.

The proposed revision of Regulatory Guide 1.97 was initiated as a result of the lessons learned from an evaluation of the investigation of TMI-2 and is being developed on a high-priority basis. The scope of proposed Revision 2 has been expanded to include all accident-monitoring instrumentation needed by the plant operator (licensee) to protect the health and safety of the public, including that required for emergency planning.

The guide also includes consideration of degraded core cooling conditions to a greater extent than was considered in Revision 1 to Regulatory Guide 1.97, which was issued in August 1977.

The guide endorses, with certain exceptions, a standard which is still under development, draft ANS-4.5, "Functional Requirements for Accident Monitoring in a Nuclear Power Generating Station," Draft 4, dated November 1979. The draft standard has not yet been approved as an American National Standard, but permission has been granted to use Draft 4 of the standard with proposed Revision 2 of Regulatory Guide 1.97 during the public review and comment period. Comments received on the draft standard will be transmitted to the ANS-4 working group that is developing the standard in addition to being given consideration by the staff for inclusion in the regulatory guide. Since the standard and the guide are being developed in parallel, comments can be resolved, as appropriate, by modification to either the standard or the guide.

Concurrent with the public review and comment period, the NRC staff will arrange meetings with the various owners' groups and/or utilities to obtain input on backfitting recommendations and impact.

It would be helpful if comments on the guide and the standard are accompanied by specific word changes to the guide in order to avoid any misunderstanding as to the thrust of the comment.

Sincerely,

[signature] for

Guy A. Arlotto, Director
Division of Engineering Standards
Office of Standards Development

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U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF STANDARDS DEVELOPMENT
DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

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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, "Domestic 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, including the necessary instrumentation, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor.

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.

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

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 preplanned 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 boundary, 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; and (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, to 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. In particular, it is important that the operator be informed regarding that status of coolant level in the reactor vessel or the existence of core voiding that would indicate degraded core cooling. Direct indication of coolant level in the reactor vessel is not currently available in pressurized water reactors. However, it is imperative that this capability be developed within a reasonable time in order to provide the operator with this vital information in a positive, unambiguous manner. It is essential that degraded conditions be identified so that the operator can take actions that are available to mitigate the consequences. It is not intended that the operator be encouraged to prematurely circumvent systems important to safety 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, anticipated transients without scram (ATWS), reactivity excursions, and releases of radioactive materials. Such events require that the operator understand, within 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 Section 3.0 of Draft Standard ANS-4.5,* "Functional Requirements for Accident Monitoring in a Nuclear Power Generating Station," Draft 4 dated November 1979. 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

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525. Although this standard has been balloted by the responsible subcommittee and reviewed by the responsible consensus body, Draft 4 does not reflect the resolution of all comments. A subsequent draft is intended to address the comments that formed the basis of the negative subcommittee ballots.

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, it is also necessary to 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, Draft 4 dated November 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 Working Group 4.5 of subcommittee ANS-4 with two primary objectives: (1) to address that instrumentation that 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~~ THREE 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 preplanned operator actions, (2) Type B - those variables that provide information to indicate whether plant safety functions are being accomplished, AND (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 for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases, for providing defence in depth, and for diagnosis~~. Type A variables have not been included in the listings of variables to be measured because they are plant specific and will depend on the operations that the designer chooses for preplanned manual action. The ~~five~~ THREE 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 or for automatically initiated safety actions. Where such multiple listing or use occurs, it is essential that instrumentation be capable of meeting the most stringent requirements.

The time phases (Phases I, AND II, ~~and 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 (identified as important to safety) contained in Draft Standard ANS-4.5, "Functional Requirements for Accident Monitoring in a Nuclear Power Generating Station," Draft 4 dated November 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 ANS-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 is 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 that 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 (e.g., Sections 4.2, 5.0, 5.1.3, 5.2, 6.1, 6.3) should include this expanded definition.

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 that 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 those of an additional type, Type E, included in this regulatory guide. (See Tables 1, 2, and 3.)

5. Section 6.1 of ANS-4.5 pertains to General Design Criteria for instrumentation monitoring Types A, B, and C variables. In conjunction with Section 6.1,

~~instrumentation monitoring Types D and E variables should also be included. Noted applicable design criteria are identified in Table I of this regulatory guide.~~

~~6. 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.~~

7. Sections 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 include 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.

~~8. Section 6.4 of ANS-4.5 pertains to specific design criteria for accident-monitoring instrumentation. In conjunction with Section 6.4, 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 PDAs and FDAs, all PDDAs and all FDDAs that 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 April 15, 1980.

Table 1

DESIGN CRITERIA[1]

CRITERIA	A	B	C	D	E
1. Seismic qualification per Regulatory Guide 1.100	yes	yes	yes	no	no[3]
2. Single failure criteria per Regulatory Guide 1.53	yes	yes	yes	no	no
3. Environmental qualification per Regulatory Guide 1.89	yes	yes	yes[4]	yes	no[5]
4. Power source	Emr[6]	CB[7]	CB[7]	Emr[6]	Emr[6]
5. Out-of-service interval before accident	[8]	[8]	[8]	[9]	[10]
6. Portable	no	no	no[11]	no[11]	no[11]
7. Quality assurance level	[12]	[12]	[12]	[12]	[12]
8. Display type[13]	Con[14]	Con[14]	Con[14]	OD[15]	OD[15]
9. Display method	Rec[16]	Rec[17]	Rec[17]	Ind[18]	Ind[18,19]
10. Unique identification	yes	yes	yes	no	no
11. Periodic testing per Regulatory Guide 1.118	yes	yes	yes	yes	no[20]

(Column heading: INSTRUMENTATION TYPES[2])

[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 that provide information required to take preplanned manual actions.

Type B - Those instruments that provide information to monitor the process of accomplishing critical safety functions.

Type C - Those instruments that indicate the potential for breaching or the 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 use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases, for defense in depth, and for diagnosis.

~~[3]Radiation monitors should meet the requirements of ANSI N320-1979, Section 5.14 and/or Section 9.1.15, as appropriate.~~

[4]See paragraph 6.3.6 of Draft Standard ANS-4.5.

(Footnotes continued)

Footnotes continued for Table 1

[5] ~~Qualified to the conditions of its operation and, for radiation monitors, ANSI N320-1979.~~

[6] ~~Emergency power source.~~ NON-CLASS 1E POWER

[7] ~~Critical Instrument Bus~~ Class 1E Power.

[8] Paragraph 4.11, "Exemption," of IEEE Standard 279-1971.

[9] Based on normal Technical Specification requirements on out-of-service for the safety system it serves.

[10] Not necessary to include in the Technical Specifications unless specified by other requirements.

[11] Radiation monitoring outside containment may be portable if so designated in Tables 2 and 3.

[12] Level of quality assurance per Appendix B to 10 CFR Part 50.

[13] 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.

[14] Continuous display.

[15] Indication on demand.

[16] Where trend or transient information is essential to planned operator actions.

[17] Recording.

[18] Dial or digital indication.

[19] Effluent release monitors require recording, including effluent radioactivity monitors, environs exposure rate monitors, and meteorology monitors.

[20] Radiation monitors should be periodically tested in accordance with the requirements of ANSI N320-1979.

Table 3

BWR VARIABLES

Measured Variable	Range	Type	Purpose
CORE			
~~Core Exit Temperature~~	~~150°F to 2300°F~~	~~B,C~~	~~To provide incore temperature measurements to identify localized hot areas. (Approximately 50 measurements)~~
~~Control Rod Position~~	~~Full in or not full in~~	~~B~~	~~To provide position 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~~ 1500 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
Primary System Safety Relief Valve Positions, including ADS or Flow Through or Pressure in Valve Lines	Closed-not closed or 0 to 50 psig	C ~~B,C~~	By these measurements, the operator knows if there is a path open for loss of coolant and if an event may be in progress.

[1]Design flow - the maximum flow anticipated in normal operation.

Table 3 (Continued)

Measured Variable	Range	Type	Purpose
REACTOR COOLANT SYSTEM (Continued)			
Radiation Level in Coolant	GROSS GAMMA ~~10 µCi/cc to 10 Ci/cc~~ 1/2 THE ALLOWABLE TECHNICAL SPECIFICATION COOLANT ACTIVITY TO 100 TIMES THAT ACTIVITY.	C	ANS-4.5, Section 6.3.2. For early indication of fuel cladding failure and estimate of extent of damage.
CONTAINMENT			
Primary Containment Pressure (DRYWELL)	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 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])	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 and Drywell Oxygen Concentration (for those plants with inerted containments)	0 to 10% (capability of operating from 12 psia to design pressure[2])	C	For indication of the need for and a measurement of the performance of the containment oxygen elimination system.
Primary Containment Isolation Valve Position	Closed-not closed	B,~~C~~	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 that could carry radioactive materials out of containment.
Suppression ~~Pool~~ CHAMBER Water Level	LOWEST ECCS SUCTION ~~Top of vent to top of weir well~~ LEVEL TO 5 FEET ABOVE NORMAL LEVEL.	~~B~~ C	ANS-4.5, Section 6.3.3.
Suppression ~~Pool~~ CHAMBER Water Temperature	50°F to 250°F	~~B~~ C	To ensure proper temperature for NPSH of ECCS. To verify the operation of the makeup system.
~~Drywell Pressure~~	~~12 psia to 3 psig~~ ~~0 to 110% design pressure[2]~~	~~B~~	~~ANS-4.5 Section 6.3.3.~~ ~~Diagnosis of impact of accident on drywell structure.~~

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

Table 3 (Continued)

Measured Variable	Range	Type	Purpose
CONTAINMENT (Continued)			
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 to indicate its magnitude in order to determine action to protect the public.
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.

Table 3 (Continued)

Measured Variable	Range	Type	Purpose
AUXILIARY SYSTEMS (Continued)			
Service Cooling Water Flow	0 to 110% design flow[1]	D	For indication of system operation.
Flow in Ultimate Heat Sink 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.
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	20°F to 130°F	~~D~~ E	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 adequate source of electric power.

Table 3 (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^{2} 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 gas on a continuous basis.
.Containment Exhaust Vent and Standby Gas Treatment System Vent	10^{-7} to 10^{5} μCi/cc Xe-133 calibration		Provisions should be made to monitor all potential pathways for release of gaseous radioactive materials to the environs in conformance with General Design Criterion 64.
.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)		<u>Note</u>: Monitoring of individual effluent streams only is required where such streams are released directly to 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 guide provided such monitoring has a range adequate to measure worst-case releases.
Effluent Radioactivity - High-Range Radiohalogens and Particulates	10^{-3} to 10^{2} μCi/cc (permanently installed monitors)	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.

Table 3 (Continued)

Measured Variable	Range	Type	Purpose
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT (Continued)			
Environs Radioactivity - Exposure Rate	10^{-6} to 10^{2} R/hr (50 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 release 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.)
Plant and Environs Radioactivity (portable instruments)	<u>High Range</u> 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 it 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.
POSTACCIDENT SAMPLING CAPABILITY			
Primary Coolant Suppression Pool Containment Air Standby Gas Treatment System (upstream of each area served)	As required based on Regulatory Guide 1.3 guidelines	E	ANS-4.5, Section 6.3.2. To provide means for safe and convenient sampling. These provisions should include:

Table 3 (Continued)

Measured Variable	Range	Type	Purpose
POSTACCIDENT ANALYSIS CAPABILITY (ON SITE)	1. gamma-ray spectrum 2. pH 3. hydrogen 4. oxygen	E	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, and 5. Pre-arrangement for analysis and interpretation.
METEOROLOGY			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15° Starting speed 0.45 mps (1 mph))	E	For determining effluent 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 than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1 mph))	E	For determining effluent travel speed and dilution for emergency planning, dose assessments, and source estimates.
Temperature	-60°F to 120°F (±0.9°F accuracy)	E	For determining nature of precipitation and ground deposition for emergency planning.
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 ensure accuracy of total accumulation within 10% of recorded value - 0.01" resolution	E	For determining effluent transport and ground deposition for emergency planning.