Superseded

Revision 2*
December 1980

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.97
(Task RS 917-4)

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 over their anticipated ranges 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 Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

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

B. DISCUSSION

Indications of plant variables are required by the control room operating personnel 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 safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity); and (3) provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and to determine if a gross breach of a barrier has occurred. In addition to the above, indications of plant variables that provide information on operation of plant safety systems and other systems important to safety are required by the control room operating personnel during an accident to (1) furnish data regarding the operation of plant systems in order that the operator can make appropriate decisions as to their use and (2) provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of any 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 variables required to enable the operation of manually initiated safety systems

and other appropriate operator actions involving systems important to safety.

Independent of the above tasks, it is important that operators be informed if the barriers to the release of radioactive materials are being challenged. Therefore, it is essential that instrument ranges be selected so that the instrument will always be on scale. Narrow-range instruments may not have the necessary range to track the course of the accident; consequently, multiple instruments with overlapping ranges may be necessary. (In the past, some instrument ranges have been selected based on the setpoint value for automatic protection or alarms.) It is essential that degraded conditions and their magnitude be identified so the operators can take actions that are available to mitigate the consequences. It is not intended that operators be encouraged to prematurely circumvent systems important to safety but that they be adequately informed in order that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated operational occurrences that become accidents such as anticipated transients without scram (ATWS), and reactivity excursions that result in releases of radioactive materials. Such events require that the operators understand, within a short time period, the ability of the barriers to limit radioactivity release, i.e., that they understand the potential for breach of a barrier or whether an actual breach of a barrier has occurred because of 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. It could therefore either be designed to withstand the accident environment or be protected by a local protected environment.

It is desirable that accident-monitoring instrumentation components and their mounts that cannot be located in seismically qualified buildings be designed to continue to function, to the extent feasible, following seismic events. An acceptable method for enhancing the seismic resistance of this instrumentation would be to design it to meet the seismic criteria applicable to like instrumentation installed in seismically qualified locations although a lesser overall qualification results.

Variables for accident monitoring can be selected to provide the essential information needed by the operator to determine if the plant safety functions are being performed. It is essential that the range selections be sufficiently great to keep instruments on scale at all times. Further, it is prudent that a limited number of those variables that are functionally significant (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 variables can attain under limiting conditions; for example, a range for the containment pressure monitor extending to the burst pressure of the containment in order that the operators will not be uninformed as to the pressure inside the containment. The availability of such instruments is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. It is also necessary to be sure that when a range is extended, the sensitivity and accuracy of the instrument are within acceptable limits for monitoring the extended range.

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 variables 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 operators to use, during accident situations, instruments with which they are most familiar. Since some accidents could impose severe operating requirements on instrumentation components, it may be necessary to upgrade those normal power plant instrumentation components to withstand the more severe operating conditions and to measure greater variations of monitored variables that may be associated with an accident. It is essential that instrumentation so upgraded does not degrade 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 variable to be monitored, possibly with different performance requirements in each range.

ANSI/ANS-4.5-1980,[1] "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," 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 following an accident. 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 operators 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. ANS-4.5 references a revision to IEEE Standard 497 as the source for specific instrumentation design criteria. Since the revision to IEEE Standard 497 has not been completed, its applicability cannot yet be determined. Hence, specific instrumentation design criteria have been included in this regulatory guide.

ANS-4.5 defines three types of variables (definitions modified herein) for the purpose of aiding the designer in selecting accident-monitoring instrumentation and applicable criteria. The types are: Type A, those variables that provide primary information[2] needed to permit the control room

[1]Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

[2]Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events; Type B, those variables that provide information to indicate whether plant safety functions are being accomplished; and 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 (modified to reflect NRC staff position; see regulatory position 1.2). The sources of potential breach are limited to the energy sources within the barrier itself. In addition to the accident-monitoring variables provided in ANS-4.5, variables for monitoring the operation of systems important to safety and radioactive effluent releases are provided by this regulatory guide. Two additional variable types are defined: Type D, those variables that provide information to indicate the operation of individual safety systems and other systems important to safety, and 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.

A minimum set of Type B, C, D, and E variables to be measured is listed in this regulatory guide. Type A variables have not been listed because they are plant specific and will depend on the operations that the designer chooses for planned manual action. Types B, C, D, and E are variables for following the course of an accident and are to be used (1) to determine if the plant is responding to the safety measures in operation and (2) to inform the operator of the necessity for unplanned actions to mitigate the consequences of an accident. The five classifications are not mutually exclusive in that a given variable (or instrument) may be applicable to one or more types, as well as for normal power plant operation or for automatically initiated safety actions. A variable included as Type B, C, D, or E does not preclude that variable from also being included as Type A. Where such multiple use occurs, it is essential that instrumentation be capable of meeting the more stringent requirements.

The time phases (Phases I and II) delineated in ANS-4.5 are not used 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 control room operating personnel.

The NRC staff is willing to work with the ANS working group to attempt to resolve the above differences.

Regulatory positions 1.3 and 1.4 of this guide provide design and qualification criteria for the instrumentation used to measure the various variables listed in Table 1 (for BWRs) and Table 2 (for PWRs). The criteria are separated into three separate groups or categories that provide a graded approach to requirements depending on the importance to safety of the measurement of a specific variable. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 provides less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 3 is intended to provide requirements that will ensure that high-quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. It is also used where the state of the art will not support requirements for higher qualified instrumentation.

In general, the measurement of a single key variable may not be sufficient to indicate the accomplishment of a given safety function. Where multiple variables are needed to indicate the accomplishment of a given safety function, it is essential that they each be considered key variables and be measured with high-quality instrumentation. Additionally, it is prudent, in some instances, to include the measurement of additional variables for backup information and for diagnosis. Where these additional measurements are included, the measures applied for design, qualification, and quality assurance of the instrumentation need not be the same as that applied for the instrumentation for key variables. A key variable is that single variable (or minimum number of variables) that most directly indicates the accomplishment of a safety function (in the case of Types B and C) or the operation of a safety system (in the case of Type D) or radioactive material release (in the case of Type E). It is essential that key variables be qualified to the more stringent design and qualification criteria. The design and qualification criteria category assigned to each variable indicates whether the variable is considered to be a key variable or for system status indication or for backup or diagnosis, i.e., for Types B and C, the key variables are Category 1; backup variables are generally Category 3. For Types D and E, the key variables are generally Category 2; backup variables are Category 3.

The variables are listed, but no mention (beyond redundancy requirements) is made of the number of points of measurement of each variable. It is important that the number of points of measurement be sufficient to adequately indicate the variable value, e.g., containment temperature may require spatial location of several points of measurement.

This guide provides the minimum number of variables to be monitored by the control room operating personnel during and following an accident. These variables are used by the control room operating personnel to perform their role in the emergency plan in the evaluation, assessment, monitoring, and execution of control room functions when the other emergency response facilities are not effectively manned. Variables are also defined to permit operators to perform their long-term monitoring and execution responsibilities after the emergency response facilities are manned. The application of the criteria for the instrumentation is limited to that part of the instrumentation system and its vital supporting features or power sources that provide the direct display of the variables. These provisions are not necessarily applicable to that part of the instrumentation systems provided as operator aids for the purpose of enhancing information presentations for the identification or diagnosis of disturbances.

C. REGULATORY POSITION

1. Accident-Monitoring Instrumentation

The criteria and requirements contained in ANSI/ANS-4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors," are considered by the NRC staff to be generally acceptable for providing instrumentation to monitor variables for accident conditions subject to the following:

1.1 Instead of the definition given in Section 3.2.1 of ANS-4.5, the definition of Type A variables should be: Type A, those variables to be monitored that provide the primary information[2] required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events.

1.2 In Section 3.2.3 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 variables that have the potential for causing a breach in the primary reactor containment have exceeded the design basis values. In conjunction with the variables that indicate the potential for causing a breach in the primary reactor containment, the variables that indicate 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. The sources of potential breach are limited to the energy sources within the cladding, coolant boundary, or containment. References to Type C instruments, and associated parameters to be measured, in ANS-4.5 (e.g., Sections 4.2, 5.0, 5.1.3, 5.2, 6.0, 6.3) should include this expanded definition.

1.3 Section 6.1 of ANS-4.5 pertains to General Criteria for Types A, B, and C accident-monitoring variables. In lieu of Section 6.1, the following design and qualification criteria categories should be used:

1.3.1 Design and Qualification Criteria - Category 1

a. The instrumentation should be qualified in accordance with Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," and the methodology described in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. Where the instrumentation channel signal is to be used in a computer-based display, recording, and/or diagnostic program, qualification applies from the sensor to and includes the channel isolation device. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions. The seismic portion of qualification should be in accordance with Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants." Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake. Instrumentation whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable should be qualified using the guidance provided in paragraph 6.3.6 of ANS-4.5.

b. No single failure within either the accident-monitoring instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition or result of a specific accident should prevent the operators from being presented the information necessary for them to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. Where failure of one accident-monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function, additional information should be provided to allow the operators to deduce the actual conditions in the plant. This may be accomplished by providing additional independent channels of information of the same variable (addition of an identical channel) or by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (addition of a diverse channel). Redundant or diverse channels should be electrically independent and physically separated from each other and from equipment not classified important to safety in accordance with Regulatory Guide 1.75, "Physical Independence of Electric Systems," up to and including any isolation device. At least one channel should be displayed on a direct-indicating or recording device. (Note: Within each redundant division of a safety system, redundant monitoring channels are not needed except for steam generator level instrumentation in two-loop plants.)

c. The instrumentation should be energized from station Standby Power sources as provided in Regulatory Guide 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants," and should be backed up by batteries where momentary interruption is not tolerable.

d. The instrumentation channel should be available prior to an accident except as provided in paragraph 4.11, "Exemption," as defined in IEEE Standard 279 or as specified in Technical Specifications.

e. The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

Regulatory Guide 1.28	"Quality Assurance Program Requirements (Design and Construction)"
Regulatory Guide 1.30	"Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment"
Regulatory Guide 1.38	"Quality Assurance Requirements for Packaging, Shipping,

	Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"
Regulatory Guide 1.58	"Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel"
Regulatory Guide 1.64	"Quality Assurance Requirements for the Design of Nuclear Power Plants"
Regulatory Guide 1.74	"Quality Assurance Terms and Definitions"
Regulatory Guide 1.88	"Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records"
Regulatory Guide 1.123	"Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants"
Regulatory Guide 1.144	"Auditing of Quality Assurance Programs for Nuclear Power Plants"
Regulatory Guide 1.146	"Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants"

Reference to the above regulatory guides (except Regulatory Guides 1.30 and 1.38) is being made pending issuance of a regulatory guide (Task RS 002-5) that is under development and will endorse ANSI/ASME NQA-1-1979, "Quality Assurance Program Requirements for Nuclear Power Plants."

f. Continuous indication (it may be by recording) display should be provided. Where two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided.

g. Recording of instrumentation readout information should be provided. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand. Intermittent displays such as data loggers and scanning recorders may be used if no significant transient response information is likely to be lost by such devices.

1.3.2 Design and Qualification Criteria - Category 2

a. The instrumentation should be qualified in accordance with Regulatory Guide 1.89 and the methodology described in NUREG-0588. Seismic qualification according to the provisions of Regulatory Guide 1.100 may be needed provided the instrumentation is part of a safety-related system. Where the channel signal is to be processed or displayed on demand, qualification applies from the sensor through the isolator/input buffer. The location of the isolation device should be such that it would be accessible for maintenance during accident conditions.

b. The instrumentation should be energized from a high-reliability power source, not necessarily Standby Power, and should be backed up by batteries where momentary interruption is not tolerable.

c. The out-of-service interval should be based on normal Technical Specification requirements on out of service for the system it serves where applicable or where specified by other requirements.

d. The recommendations of the regulatory guides pertaining to quality assurance listed under paragraph 1.3.1e of this guide should be followed. Reference to the above regulatory guides (except Regulatory Guides 1.30 and 1.38) is being made pending issuance of a regulatory guide (Task RS 002-5) that is under development and will endorse ANSI/ASME NQA-1-1979. Since some instrumentation is less important to safety than other instrumentation, it may not be necessary to apply the same quality assurance measures to all instrumentation. The quality assurance requirements that are implemented should provide control over activities affecting quality to an extent consistent with the importance to safety of the instrumentation. These requirements should be determined and documented by personnel knowledgeable in the end use of the instrumentation.

e. The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand by a CRT or by other appropriate means.

f. The method of display may be by dial, digital, CRT, or stripchart recorder indication. Effluent radioactivity monitors, area radiation monitors, and meteorology monitors should be recorded. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand.

1.3.3 Design and Qualification Criteria - Category 3

a. The instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment.

b. The method of display may be by dial, digital, CRT, or stripchart recorder indication. Effluent radioactivity monitors, area radiation monitors, and meteorology monitors should be recorded. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer memory, and displayed on demand.

1.4 In addition to the criteria of regulatory position 1.3, the following criteria should apply to Categories 1 and 2:

a. Any equipment that is used for either Category 1 or Category 2 should be designated as part of accident-monitoring instrumentation or systems operation and effluent-monitoring instrumentation. The transmission of signals from such equipment for other use should be through isolation devices that are designated as part of the monitoring instrumentation and that meet the provisions of this document.

b. The instruments designated as Types A, B, and C and Categories 1 and 2 should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

1.5 In addition to the above criteria, the following criteria should apply to Categories 1, 2, and 3:

a. Servicing, testing, and calibration programs should be specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing will be less than the normal time interval between generating station shutdowns, a capability for testing during power operation should be provided.

b. Whenever means for removing channels from service are included in the design, the design should facilitate administrative control of the access to such removal means.

c. The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

d. The monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator. Human factors analysis should be used in determining type and location of displays.

e. The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

f. To the extent practicable, monitoring instrumentation inputs should be from sensors that directly measure the desired variables. An indirect measurement should be made only when it can be shown by analysis to provide unambiguous information.

g. To the extent practicable, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are most familiar. However, where the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.

h. Periodic checking, testing, calibration, and calibration verification should be in accordance with the applicable portions of Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," pertaining to testing of instrument channels. (Note: Response time testing not usually needed.)

1.6 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 Types B and C variables. In conjunction with the above-listed sections of ANS-4.5, Tables 1 and 2 of this regulatory guide (which include those variables mentioned in these sections) should be considered as the minimum number of instruments and their respective ranges for accident-monitoring instrumentation for each nuclear power plant.

2. Systems Operation Monitoring and Effluent Release Monitoring Instrumentation

2.1 Definitions

a. Type D, those variables that provide information to indicate the operation of individual safety systems and other systems important to safety.

b. Type E, those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and in continually assessing such releases.

2.2 The plant designer should select variables and information display channels required by his design to enable the control room operating personnel to:

a. Ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident.

b. Monitor the effluent discharge paths and environs within the site boundary to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and to continually assess such releases.

c. Obtain required information through a backup or diagnosis channel where a single channel may be likely to give ambiguous indication.

2.3 The process for selecting system operation and effluent release variables should include the identification of:

a. For Type D

(1) The plant safety systems and other systems important to safety that should be operating or that could be placed in operation to help mitigate the consequences of an accident; and

(2) The variable or minimum number of variables that indicate the operating status of each system identified in (1) above.

b. For type E

(1) The planned paths for effluent release;

(2) Plant areas and inside buildings where access is required to service equipment necessary to mitigate the consequences of an accident;

(3) Onsite locations where unplanned releases of radioactive materials should be detected; and

(4) The variables that should be monitored in each location identified in (1), (2), and (3) above.

2.4 The determination of performance requirements for system operation monitoring and effluent release monitoring information display channels should include, as a minimum, identification of:

a. The range of the process variable.
b. The required accuracy of measurement.
c. The required response characteristics.
d. The time interval during which the measurement is needed.
e. The local environment(s) in which the information display channel components must operate.
f. Any requirement for rate or trend information.
g. Any requirements to group displays of related information.
h. Any required spatial distribution of sensors.

2.5 The design and qualification criteria for system operation monitoring and effluent release monitoring instrumentation should be taken from the criteria provided in regulatory positions 1.3 and 1.4 of this guide. Tables 1 and 2 of this regulatory guide should be considered as the minimum number of instruments and their respective ranges for systems operation monitoring (Type D) and effluent release monitoring (Type E) instrumentation for each nuclear power plant.

D. IMPLEMENTATION

All plants going into operation after June 1983 should meet the provisions of this guide.

Plants currently operating should meet the provisions of this guide, except as modified by NUREG-0737 and the Commission Memorandum and Order (CLI-80-21), by June 1983.

Plants scheduled to be licensed to operate before June 1, 1983, should meet the requirements of NUREG-0737 and the Commission Memorandum and Order (CLI-80-21) and the schedules of these documents or prior to the issuance of a license to operate, whichever date is later. The balance of the provisions of this guide should be completed by June 1983.

The difficulties of procuring and installing additions or modifications to in-place instrumentation have been considered in establishing these schedules.

Exceptions to provisions and schedules will be considered for extraordinary circumstances.

TABLE 1

BWR VARIABLES

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

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

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

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

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

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

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

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

TABLE 1 (Continued)

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

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

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

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

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

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

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

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

TABLE 1 (Continued)

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

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

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

TABLE 1 (Continued)

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

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

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Condensate and Feedwater System			
Main Feedwater Flow	0 to 110% design flow[10]	3	Detection of operation; analysis of cooling
Condensate Storage Tank Level	Bottom to top	3	Indication of available water for cooling
Primary Containment-Related Systems			
Suppression Chamber Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Drywell Pressure[2]	12 psia to 3 psig 0 to 110% design pressure[3]	2	To monitor operation
Suppression Pool Water Level	Top of vent to top of weir well	2	To monitor operation
Suppression Pool Water Temperature	30°F to 230°F	2	To monitor operation
Drywell Atmosphere Temperature	40°F to 440°F	2	To monitor operation
Drywell Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Main Steam System			
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	2	To provide indication of pressure boundary maintenance
Primary System Safety Relief Valve Positions, Including ADS or Flow Through or Pressure in Valve Lines	Closed–not closed or 0 to 50 psig	2	Detection of accident; boundary integrity indication

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

TABLE 1 (Continued)

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

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

TABLE 1 (Continued)

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

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

TABLE 1 (Continued)

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

[12] To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^{2} μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^{1} μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

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

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

TABLE 1 (Continued)

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

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

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

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

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

TABLE 2

PWR VARIABLES

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

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

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

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

Variable	Range	Category	Purpose
Reactivity Control			
Neutron Flux	10^{-6}% to 100% full power	1	Function detection; accomplishment of mitigation
Control Rod Position	Full in or not full in	3	Verification
RCS Soluble Boron Concentration	0 to 6000 ppm	3	Verification
RCS Cold Leg Water Temperature[1]	50°F to 400°F	3	Verification
Core Cooling			
RCS Hot Leg Water Temperature	50°F to 750°F	1	Function detection; accomplishment of mitigation; verification; long-term surveillance
RCS Cold Leg Water Temperature[1]	50°F to 750°F	1	Function detection; accomplishment of mitigation; verification; long-term surveillance
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Function detection; accomplishment of mitigation; verification; long-term surveillance

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

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE B (Continued)			
Core Cooling (Continued)			
Core Exit Temperature[1]	200°F to 2300°F (for operating plants - 200°F to 1650°F)	3[3]	Verification
Coolant Level in Reactor	Bottom of core to top of vessel	1 (Direct-indicating or recording device not needed)	Verification; accomplishment of mitigation
Degrees of Subcooling	200°F subcooling to 35°F superheat	2 (With confirmatory operator procedures)	Verification and analysis of plant conditions
Maintaining Reactor Coolant System Integrity			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Function detection; accomplishment of mitigation
Containment Sump Water Level[1]	Narrow range (sump), Wide range (bottom of containment to 600,000-gallon level equivalent)	2 1	Function detection; accomplishment of mitigation; verification
Containment Pressure[1]	0 to design pressure[4] (psig)	1	Function detection; accomplishment of mitigation; verification
Maintaining Containment Integrity			
Containment Isolation Valve Position (excluding check valves)	Closed–not closed	1	Accomplishment of isolation
Containment Pressure[1]	10 psia to design pressure[4]	1	Function detection; accomplishment of mitigation; verification

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[3]A minimum of four measurements per quadrant is required for operation. Sufficient number should be installed to account for attrition. (Replacement instrumentation should meet the 2300°F range provision.)

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

TABLE 2 (Continued)

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

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Fuel Cladding			
Core Exit Temperature[1]	200°F to 2300°F (for operating plants - 200°F to 1650°F)	1[3]	Detection of potential for breach; accomplishment of mitigation; long-term surveillance
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	1/2 Tech Spec limit to 100 times Tech Spec limit, R/hr	1	Detection of breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume	3[5]	Detail analysis; accomplishment of mitigation; verification; long-term surveillance
Reactor Coolant Pressure Boundary			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Detection of potential for or actual breach; accomplishment of mitigation; long-term surveillance
Containment Pressure[1]	10 psia to design pressure[4] psig (5 psia for subatmospheric containments)	1	Detection of breach; accomplishment of mitigation; verification; long-term surveillance
Containment Sump Water Level[1]	Narrow range (sump), Wide range (bottom of containment to 600,000-gal level equivalent)	2 1	Detection of breach; accomplishment of mitigation; verification; long-term surveillance
Containment Area Radiation[1]	1 R/hr to 10^4 R/hr	3[6,7]	Detection of breach; verification
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust[1]	10^{-6} μCi/cc to 10^{-2} μCi/cc	3[8]	Detection of breach; verification

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

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

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

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

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE C (Continued)			
Containment			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Detection of potential for breach; accomplishment of mitigation
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[4]) 0 to 30% for ice-condenser-type containment	1	Detection of potential for breach; accomplishment of mitigation; long-term surveillance
Containment Pressure[1]	10 psia pressure to 3 times design pressure[4] for concrete; 4 times design pressure for steel (5 psia for subatmospheric containments)	1	Detection of potential for or actual breach; accomplishment of mitigation
Containment Effluent Radioactivity - Noble Gases from Identified Release Points[1]	10^{-6} μCi/cc to 10^{-2} μCi/cc	2[8,9]	Detection of breach; accomplishment of mitigation; verification
Radiation Exposure Rate (inside buildings or areas, e.g., auxiliary building, reactor shield building annulus, fuel handling building, which are in direct contact with primary containment where penetrations and hatches are located)[1]	10^{-1} R/hr to 10^{4} R/hr	2[7]	Indication of breach
Effluent Radioactivity[1] - Noble Gases (from buildings as indicated above)	10^{-6} μCi/cc to 10^{3} μCi/cc	2[8]	Indication of breach

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

Variable	Range	Category	Purpose
Residual Heat Removal (RHR) or Decay Heat Removal System			
RHR System Flow	0 to 110% design flow[10]	2	To monitor operation
RHR Heat Exchanger Outlet Temperature	32°F to 350°F	2	To monitor operation and for analysis

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

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Safety Injection Systems			
Accumulator Tank Level and Pressure	10% to 90% volume 0 to 750 psig	2	To monitor operation
Accumulator Isolation Valve Position	Closed or Open	2	Operation status
Boric Acid Charging Flow	0 to 110% design flow[10]	2	To monitor operation
Flow in HPI System	0 to 110% design flow[10]	2	To monitor operation
Flow in LPI System	0 to 110% design flow[10]	2	To monitor operation
Refueling Water Storage Tank Level	Top to bottom	2	To monitor operation
Primary Coolant System			
Reactor Coolant Pump Status	Motor current	3	To monitor operation
Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines	Closed–not closed	2	Operation status; to monitor for loss of coolant
Pressurizer Level	Bottom to top	1	To ensure proper operation of pressurizer
Pressurizer Heater Status	Electric current	2	To determine operating status
Quench Tank Level	Top to bottom	3	To monitor operation
Quench Tank Temperature	50°F to 750°F	3	To monitor operation
Quench Tank Pressure	0 to design pressure[4]	3	To monitor operation
Secondary System (Steam Generator)			
Steam Generator Level	From tube sheet to separators	1	To monitor operation
Steam Generator Pressure	From atmospheric pressure to 20% above the lowest safety valve setting	2	To monitor operation
Safety/Relief Valve Positions or Main Steam Flow	Closed–not closed	2	To monitor operation
Main Feedwater Flow	0 to 110% design flow[10]	3	To monitor operation

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Auxiliary Feedwater or Emergency Feedwater System			
Auxiliary or Emergency Feedwater Flow	0 to 110% design flow[10]	2 (1 for B&W plants)	To monitor operation
Condensate Storage Tank Water Level	Plant specific	1	To ensure water supply for auxiliary feedwater (Can be Category 3 if not primary source of AFW. Then whatever is primary source of AFW should be listed and should be Category 1.)
Containment Cooling Systems			
Containment Spray Flow	0 to 110% design flow[10]	2	To monitor operation
Heat Removal by the Containment Fan Heat Removal System	Plant specific	2	To monitor operation
Containment Atmosphere Temperature	40°F to 400°F	2	To indicate accomplishment of cooling
Containment Sump Water Temperature	50°F to 250°F	2	To monitor operation
Chemical and Volume Control System			
Makeup Flow - In	0 to 110% design flow[10]	2	To monitor operation
Letdown Flow - Out	0 to 110% design flow[10]	2	To monitor operation
Volume Control Tank Level	Top to bottom	2	To monitor operation
Cooling Water System			
Component Cooling Water Temperature to ESF System	32°F to 200°F	2	To monitor operation
Component Cooling Water Flow to ESF System	0 to 110% design flow[10]	2	To monitor operation
Radwaste Systems			
High-Level Radioactive Liquid Tank Level	Top to bottom	3	To indicate storage volume
Radioactive Gas Holdup Tank Pressure	0 to 150% design pressure[4]	3	To indicate storage capacity

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To indicate damper status
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic)	Voltages, currents, pressures	2[11]	To indicate system status
TYPE E Variables: those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.			
Containment Radiation			
Containment Area Radiation - High Range[1]	1 R/hr to 10^7 R/hr	1[6,7]	Detection of significant releases; release assessment; long-term surveillance; emergency plan actuation
Area Radiation			
Radiation Exposure Rate[1] (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[7]	Detection of significant releases; release assessment; long-term surveillance
Airborne Radioactive Materials Released from Plant			
Noble Gases and Vent Flow Rate			
• Containment or Purge Effluent[1]	10^{-6} μCi/cc to 10^5 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Reactor Shield Building Annulus[1] (if in design)	10^{-6} μCi/cc to 10^4 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Auxiliary Building[1] (including any building containing primary system gases, e.g., waste gas decay tank)	10^{-6} μCi/cc to 10^3 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment; long-term surveillance

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Type E (Continued)			
Airborne Radioactive Materials Released from Plant (Continued)			
Noble Gases and Vent Flow Rate (Continued)			
• Condenser Air Removal System Exhaust[1]	10^{-6} μCi/cc to 10^{5} μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Common Plant Vent or Multi-purpose Vent Discharging Any of Above Releases (if containment purge is included)	10^{-6} μCi/cc to 10^{3} μCi/cc 0 to 110% vent design flow[10] 10^{-6} μCi/cc to 10^{4} μCi/cc	2[8]	Detection of significant releases; release assessment; long-term surveillance
• Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves	10^{-1} μCi/cc to 10^{3} μCi/cc (Duration of releases in seconds and mass of steam per unit time)	2[12]	Detection of significant releases; release assessment
• All Other Identified Release Points	10^{-6} μCi/cc to 10^{2} μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through other monitored plant vents)	2[8]	Detection of significant releases; release assessment; long-term surveillance
Particulates and Halogens			
• All Identified Plant Release Points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust). Sampling with Onsite Analysis Capability	10^{-3} μCi/cc to 10^{2} μCi/cc 0 to 110% vent design flow[10]	3[13]	Detection of significant releases; release assessment; long-term surveillance

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[12] Effluent monitors for PWR steam safety valve discharges and atmospheric steam dump valve discharges should be capable of approximately linear response to gamma radiation photons with energies from approximately 0.5 MeV to 3 MeV. Overall system accuracy should be within a factor of 2. Calibration sources should fall within the range of approximately 0.5 MeV to 1.5 MeV (e.g., Cs-137, Mn-54, Na-22, and Co-60). Effluent concentrations should be expressed in terms of any gamma-emitting noble gas nuclide within the specified energy range. Calculational methods should be provided for estimating concurrent releases of low-energy noble gases that cannot be detected or measured by the methods or techniques employed for monitoring.

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Environs Radiation and Radioactivity			
Radiation Exposure Meters (continuous indication at fixed locations)	Range, location, and qualification criteria to be developed to satisfy NUREG-0654, Section II.H.5b and 6b requirements for emergency radiological monitors		Verify significant releases and local magnitudes
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} μCi/cc to 10^{-3} μCi/cc	3[14]	Release assessment; analysis
Plant and Environs Radiation (portable instrumentation)	10^{-3} R/hr to 10^{4} R/hr, photons 10^{-3} rads/hr to 10^{4} rads/hr, beta radiations and low-energy photons	3[15] 3[15]	Release assessment; analysis
Plant and Environs Radioactivity (portable instrumentation)	Multichannel gamma-ray spectrometer	3	Release assessment; analysis
Meteorology[16]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 15°). Starting speed 0.45 mps (1.0 mph). Damping ratio between 0.4 and 0.6, distance constant ≤ 2 meters	3	Release assessment
Wind Speed	0 to 30 mps (67 mph) ±0.22 mps (0.5 mph) accuracy for wind speeds less than 11 mps (25 mph) with a starting threshold of less than 0.45 mps (1.0 mph)	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary system, -5°C to 10°C (-9°F to 18°F) and ±0.15°C accuracy per 50-meter intervals (±0.3°F accuracy per 164-foot intervals) or analogous range for alternative stability estimates	3	Release assessment

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

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

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE E (Continued)			
Accident Sampling[17] Capability (Analysis Capability On Site)			
Primary Coolant and Sump	Grab Sample	3[5,18]	Release assessment; verification; analysis
• Gross Activity	10 μCi/ml to 10 Ci/ml		
• Gamma Spectrum	(Isotopic Analysis)		
• Boron Content	0 to 6000 ppm		
• Chloride Content	0 to 20 ppm		
• Dissolved Hydrogen or Total Gas[19]	0 to 2000 cc(STP)/kg		
• Dissolved Oxygen[19]	0 to 20 ppm		
• pH	1 to 13		
Containment Air	Grab Sample	3[5]	Release assessment; verification; analysis
• Hydrogen Content	0 to 10% 0 to 30% for ice condensers		
• Oxygen Content	0 to 30%		
• Gamma Spectrum	(Isotopic analysis)		

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

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

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

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The applicant for a license (or licensee) of a nuclear power plant is required by the Commission's regulations to provide instrumentation to (1) monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety and (2) monitor 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 revision to Regulatory Guide 1.97 proposes to improve the guidance for plant and environs monitoring during and following an accident, including extended ranges for some instruments to account for consideration of degraded core cooling.

1.2 Need

Regulatory Guide 1.97 was issued as an effective guide in August 1977. At the time the guide was issued, it was recognized that more specific guidance than that contained in the guide would be required. However, the difficulty in developing the guide to the point where it could be initially issued was evidence that experience in using the guide as it then existed was essential before further development of the guide would be meaningful.

Therefore, in August 1977, the staff initiated Task Action Plan A-34, "Instruments for Monitoring Radiation and Process Variables During an Accident." The purpose of the task action plan was to develop guidance for applicants, licensees, and staff reviewers concerning implementation of Regulatory Guide 1.97. Such effort would provide a basis for revising the guide.

When the staff was ready to issue the results of the Task Action Plan A-34 effort, the accident at TMI-2 occurred. Subsequently, the TMI-2 Lessons Learned Task Force has issued its "Status Report and Short-Term Recommendations," NUREG-0578. This report, along with the draft Task Action Plan A-34 report, Draft 1 of Regulatory Guide 1.97 (dated April 12, 1974), and Standard ANS-4.5, provides ample basis for revising Regulatory Guide 1.97.

1.3 Value/Impact of Proposed Action

1.3.1 NRC Operations

Since a list of selected variables to be provided with instrumentation to be monitored by the plant operator during and following an accident has not been explicitly agreed to in the past, the proposed action should result in more effective effort by the staff in reviewing applications for construction permits and operating licenses. The proposed action will establish an NRC position by taking advantage of previous staff effort (1) in completing a generic activity (A-34), (2) in evaluating the lessons learned from the TMI-2 event (NUREG-0578), and (3) in conjunction with effort in developing a national standard (ANS-4.5). For future plants, the staff review will be simplified with guidance contained in the endorsed standard developed by a voluntary standards group and in the regulatory guide, which includes a list of variables for accident monitoring. Efforts by the staff to implement Revision 1 to Regulatory Guide 1.97 have been fraught with frustration and met with delays because the guide was adjudged by licensees to be vague and ambiguous. Revision 2 eliminates the problems encountered with Revision 1 because it provides a minimum set of variables to be measured and hence gives more guidance in the selection of accident-monitoring instrumentation. Consequently, there will be no significant impact on the staff. There will, however, be effort required to review each operating plant and each plant under review to assess conformance with Regulatory Guide 1.97.

1.3.2 Other Government Agencies

Not applicable, unless the government agency is an applicant.

1.3.3 Industry

The proposed action establishes a more clearly defined NRC position with regard to instrumentation to assess plant and environs conditions during and following an accident and therefore reduces uncertainty as to what the staff considers acceptable in the area of accident monitoring. Most of the impact on industry will be in the area of providing instrumentation to indicate the potential breach and the actual breach of the barriers to radioactivity release, i.e., fuel cladding, reactor coolant pressure boundary, and containment. Some instruments have extended ranges and others have higher qualification requirements. There will be additional impact due to heretofore unspecified variables to be monitored (i.e., water level in reactor for PWRs and radiation level in the primary coolant water for PWRs and BWRs) that have been identified during the evaluation of TMI-2 experience and will require development.

Attempts were made during the comment period to determine the cost impact on industry for future plants and for backfitting existing plants. Estimates ranged from \$4,000,000 to over \$20,000,000. The higher estimates undoubtedly charged all accident-monitoring instrumentation to Revision 2 to Regulatory Guide 1.97. This should not be the case. The requirement for accident monitoring has always been a part of the regulations. Consequently the impact of Revision 2 to Regulatory Guide 1.97 should only be the delta added by Revision 2. A conservative estimate of the increase in requirements are the additions of Type C measurements and the upgrading of some of the Type B

measurements to higher qualification of the instrumentation. There are 17 unique Type B and C variables to be measured for PWRs, less for BWRs. A conservative average cost for each measurement is $130,000 making a total cost impact of $2,210,000. If this figure were doubled to account for overhead costs and about a 15 percent contingency added, the cost impact would be about $5,000,000. This cost estimate is the same for operating plants as for plants under construction and future plants. While it is recognized that for operating plants the costs associated with backfitting are generally higher than the costs associated with new plants, some concessions are made in some requirements as a result of existing licensing commitments that bring the cost estimate to about the same value.

1.3.4 Public

The proposed action will improve public safety by ensuring that the plant operator will have timely information to take any necessary action to protect the public.

No impact on the public can be foreseen.

1.4 Decision on Proposed Action

As previously stated, more definitive guidance on instrumentation to assess plant and environs conditions during and following an accident should be given.

2. TECHNICAL APPROACH

This section is not applicable to this value/impact statement since the proposed action is a revision of an existing regulatory guide, and there are no alternatives to providing the plant operator with the required information.

3. PROCEDURAL APPROACH

Previously discussed.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for this guide would be derived from the safety requirements of the Atomic Energy Act. In particular, Criterion 13, Criterion 19, and Criterion 64 of Appendix A to 10 CFR Part 50 require, in part, that instrumentation be provided to monitor variables, systems, and plant environs to ensure adequate safety.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined in paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No conflicts or overlaps with requirements promulgated by other agencies are foreseen. This guide does include the variables to be monitored on site by the plant operator in order to provide necessary information for emergency planning. However, information on emergency planning and its relationship to other agencies is provided elsewhere. Implementation of the proposed action is discussed in Section D of this revision.

6. SUMMARY AND CONCLUSIONS

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," should be issued.