U.S. NUCLEAR REGULATORY COMMISSION

Revision 3
May 1983

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

IC 268-4

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 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.

Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0011.

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 or that one of a set of overlapping instruments will be 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 Std 497-1977, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,"[2] as the source for specific instrumentation design criteria. Since the revision to IEEE Std 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

[*] Lines indicate substantive changes from Revision 2.

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

[2] Copies may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, New York 10017.

primary information[3] needed to permit the control room 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 1.3). 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 position 1.4 (Table 1) of this guide provides design and qualification criteria for the instrumentation used to measure the various variables listed in Table 2 (for BWRs) and Table 3 (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

[3] 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.

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 Section 1 of ANS-4.5 references IEEE Std 497-1977. The specific applicability or acceptability of this standard has not yet been determined.

1.2 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[3] 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.

1.3 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.

1.4 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 design and qualification criteria categories in Table 1 should be used for the variables in Tables 2 and 3.

In general, Category 1 provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Category 2 provides for qualification but is less stringent in that it does not (of itself) include seismic qualification, redundancy, or continuous display and requires only a high-reliability power source (not necessarily standby power). Category 3 is the least stringent. It provides for high-quality commercial-grade equipment that requires only offsite power.

1.5 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 2 and 3 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. Instrumentation for Monitoring Systems Operation and Effluent Release

2.1 Type D variables are those that provide information to indicate the operation of individual safety systems and other systems important to safety. Type E variables are those that are to be monitored as required for use in determining the magnitude of the release of radioactive materials and in continuously 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 continuously assess such releases.

c. Obtain required information through a backup or diagnosis channel if 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.

TABLE 1

DESIGN AND QUALIFICATION CRITERIA FOR INSTRUMENTATION

Category 1	Category 2	Category 3
1. Equipment Qualification	**1. Equipment Qualification**	**1. Equipment Qualification**
The instrumentation should be qualified in accordance with Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," and the methodology described in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."[4]	Same as Category 1	No specific provision
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.	Same as Category 1	No specific provision
Qualification applies to the complete instrumentation channel from sensor to display where the display is a direct-indicating meter or recording device. If the instrumentation channel signal is to be used in a computer-based display, recording, or diagnostic program, qualification applies from the sensor up to and including the channel isolation device.	Same as Category 1	No specific provision
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.	No specific provision	No specific provision
2. Redundancy	**2. Redundancy**	**2. Redundancy**
No single failure within either the accident-monitoring instrumentation, its auxiliary supporting features, or its power sources concurrent with the failures that are	No specific provision	No specific provision

[4]Copies are available from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

TABLE 1 (Continued)

Category 1	Category 2	Category 3
2. (Continued) 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. Within each redundant division of a safety system, redundant monitoring channels are not needed except for steam generator level instrumentation in two-loop plants.		
3. Power Source 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.	**3. Power Source** 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.	**3. Power Source** No specific provision

TABLE 1 (Continued)

Category 1	Category 2	Category 3
4. Channel Availability The instrumentation channel should be available prior to an accident except as provided in paragraph 4.11, "Exception," as defined in IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations,"[2] or as specified in the technical specifications.	**4. Channel Availability** 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.	**4. Channel Availability** No specific provision
5. Quality Assurance 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 (Safety Guide 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"	**5. Quality Assurance** Same as Category 1 as modified by the following: 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.	**5. Quality Assurance** The instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment.

TABLE 1 (Continued)

Category 1

5. (Continued)

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 revision to Regulatory Guide 1.28 that is under development (Task RS 002-5) and that will endorse ANSI/ASME NQA-1-1979, "Quality Assurance Program Requirements for Nuclear Power Plants."[5]

6. Display and Recording

Continuous real-time display should be provided. The indication may be on a dial, digital display, CRT, or stripchart recorder.

Recording of instrumentation readout information should be provided for at least one redundant channel.

Category 2

6. Display and Recording

The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand.

Signals from effluent radioactivity monitors and area monitors should be recorded.

Category 3

6. Display and Recording

Same as Category 2

Signals from effluent radioactivity monitors, area monitors, and meteorology monitors should be recorded.

[5]Copies may be obtained from the American Society of Mechanical Engineers, 345 East 47th Street, New York, New York 10017.

TABLE 1 (Continued)

Category 1	Category 2	Category 3
6. (Continued)		
If direct and immediate trend or transient information is essential for operator information or action, the recording should be continuously available on redundant 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.	Same as Category 1	Same as Category 1
7. Range	**7. Range**	**7. Range**
If two or more instruments are needed to cover a particular range, overlapping of instrument span should be provided. If the required range of monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.	Same as Category 1	Same as Category 1
8. Equipment Identification	**8. Equipment Identification**	**8. Equipment Identification**
Types A, B, and C instruments designated as Categories 1 and 2 should be specifically identified with a common designation on the control panels so that the operator can easily discern that they are intended for use under accident conditions.	Same as Category 1	No specific provision
9. Interfaces	**9. Interfaces**	**9. Interfaces**
The transmission of signals 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.	Same as Category 1	No specific provision
10. Servicing, Testing, and Calibration	**10. Servicing, Testing, and Calibration**	**10. Servicing, Testing, and Calibration**
Servicing, testing, and calibration programs should be specified to maintain the capability of the monitoring instrumentation. If the required interval between	Same as Category 1	Same as Category 1

TABLE 1 (Continued)

Category 1	Category 2	Category 3
10. (Continued)		
testing is less than the normal time interval between plant shutdowns, a capability for testing during power operation should be provided.		
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.	Same as Category 1	Same as Category 1
The design should facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.	Same as Category 1	Same as Category 1
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.)	Same as Category 1	Same as Category 1
The location of the isolation device should be such that it would be accessible for maintenance during accident conditions.	Same as Category 1	No specific provision
11. Human Factors	**11. Human Factors**	**11. Human Factors**
The instrumentation should be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.	Same as Category 1	Same as Category 1
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.	Same as Category 1	Same as Category 1

TABLE 1 (Continued)

Category 1	Category 2	Category 3
11. (Continued)		
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.	Same as Category 1	Same as Category 1
12. Direct Measurement	**12. Direct Measurement**	**12. Direct Measurement**
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.	Same as Category 1	Same as Category 1

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 environments 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.

2.4 The design and qualification criteria for syster operation monitoring and effluent release monitorin instrumentation should be taken from the criteria provide in regulatory position 1.4 of this guide. Tables 2 and 3 o this regulatory guide should be considered as the minimun number of instruments and their respective ranges fo systems operation monitoring (Type D) and effluent releas monitoring (Type E) instrumentation for each nuclear power plant.

D. APPLICABILITY

This revision in combination with §50.49 of 10 CFR Part 50 provides acceptable guidance for design of new plants and for plant redesign in response to TMI-2 Action Plan (NUREG-0737) and its subsequent clarifications and generic letters.[6]

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[6]NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, may be obtained from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Supplement 1 (Generic Letter No. 82-33) is available for inspection or copying for a fee at the NRC Public Document Room, 1717 H Street, NW., Washington, D.C.

TABLE 2

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 1.4 and Table 1)	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 (Grab Sample)	0 to 1000 ppm	3	Verification
Core Cooling			
Coolant Level in Reactor Vessel	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 Temperature[1,2]	200°F to 2300°F		To provide diverse indication of water level
Maintaining Reactor Coolant System Integrity			
RCS Pressure[2]	0 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] Provision still being considered, subject to further development.

[2] If 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 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE B (Continued)			
Drywell Sump Level[2]	Top to Bottom	1	Function detection; accomplishment of mitigation; verification
Maintaining Containment Integrity			
Primary Containment Pressure[2]	-5 psig 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	Purpose
Fuel Cladding			
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	1/2 Tech Spec limit to 100 times Tech Spec limit	1	Detection of breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/ml to 10 Ci/ml or TID-14844 source term in coolant volume	3[4]	Detail analysis; accomplishment of mitigation; verification; long-term surveillance
BWR Core Temperature[1,2]	200°F to 2300°F		To provide diverse indication of water level
Reactor Coolant Pressure Boundary			
RCS Pressure[2]	0 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

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[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 N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," should be applied. For liquids, provisions should be made for sampling from well-mixed turbulent zones, and sampling lines should be designed to minimize plateout or deposition. For safe and convenient sampling, the provisions should include:

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

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

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

[7] Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with a dose rate response accuracy within a factor of 2 over the entire range.

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE C (Continued)			
Reactor Coolant Pressure Boundary (Continued)			
Drywell Drain Sumps Level[2] (Identified and Unidentified Leakage)	Top to Bottom	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]	0 to 1500 (psig)	1[5]	Detection of potential for breach; accomplishment of mitigation
Primary Containment Pressure[2]	-5 psig 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 vol-% (capability of operating from -5 psig 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 vol-% (capability of operating from -5 psig 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

[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 gaseous effluent radioactivity 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 radioactivity may be expressed in terms of concentrations of Xe-133 equivalents, in terms of concentrations of any noble gas nuclides, or in terms of integrated gamma MeV per unit time. It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this regulatory guide and that multiple components or systems will be needed. Existing equipment may be used to monitor any portion of the stated range within the equipment design rating.

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE C (Continued)			
Containment (Continued)			
Effluent Radioactivity[2] - Noble Gases (from buildings or areas where penetrations and hatches are located, e.g., secondary containment and auxiliary buildings and fuel handling buildings that are in direct contact with primary containment)	10^{-6} μCi/cc to 10^{3} μCi/cc	2[9]	Indication of breach

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

Variable	Range	Category	Purpose
Condensate and Feedwater System			
Main Feedwater Flow	0 to 110% design flow[10]	3	Detection of operation; analysis of cooling
Condensate Storage Tank Level	Top to Bottom	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]	-5 psig to 3 psig (narrow range) and 0 to 110% design pressure[3] (wide range)	2	To monitor operation
Suppression Pool Water Level	Top of vent to top of weir well	2	To monitor operation
Suppression Pool Water Temperature	40°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 (narrow range) and 0 to 5 psid (wide range)	2	To provide indication of pressure boundary maintenance

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE D (Continued)			
Main Steam System (Continued)			
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
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	Top to Bottom	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	40°F to 350°F	2	To monitor operation
Cooling Water System			
Cooling Water Temperature to ESF System Components	40°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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE D (Continued)			
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (electric, hydraulic, pneumatic) (voltages, currents, pressures)	Plant specific	2[11]	To monitor system status
TYPE E Variables: those variables to be monitored as required for use in determining the magnitude of the release of r active materials and continually assessing such releases.			
Containment Radiation			
Primary Containment Area Radiation - High Range[2]	1 R/hr to 10^7 R/hr	1[6,7]	Detection of significant releas 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[7] 1[6,7]	Detection of significant release release assessment; long-term surveillance
Area Radiation			
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	3[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 discharge 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

[11] [illegible]us 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 1.4 and Table 1)	Purpose
TYPE E (Continued)			
• 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 dry well 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
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[13]			
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} μCi/cc to 10^{-3} μCi/cc	3[14]	Release assessment; analysis

[12] To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^{2} μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^{2} μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration. For the purposes of this item only, "collection of representative samples" means obtaining the best samples practicable given the exigencies that attend the accident environment; line losses or line deposition should be empirically predetermined and appropriate loss correction factors should be applied.

[13] It is unlikely that a few fixed-station area monitors could provide sufficiently reliable information to be of use in detecting releases from unmonitored containment release points. However, there may be circumstances in which such a system of monitors may be useful. The decision to install such a system is left to the licensee.

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE E (Continued)			
Environs Radiation and Radioactivity[13] (Continued)			
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)	(Isotopic Analysis)	3[15,16]	Release assessment; analysis
Meteorology[17]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 10°). Starting speed less than 0.4 mps (1.0 mph). Damping ratio greater than or equal to 0.4, delay distance less than or equal to 2 meters.	3	Release assessment
Wind Speed	0 to 22 mps (50 mph). ±0.2 mps (0.5 mph) accuracy for speeds less than 2 mps (5 mph), 10% for speeds in excess of 2 mps (5 mph), with a starting threshold of less than 0.4 mps (1.0 mph) and a distance constant not to exceed 2 meters.	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary meteorological 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

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

[16] A portable multichannel gamma ray spectrometer would provide the earliest capability for scoping the radionuclide content of [illegible] source (see R. C. Ragaini, D. E. Jones, and G. W. Huckabay, "Instrumentation for Off-Site Reactor Plume Studies," in *International Sy[illegible] sium on Environmental Monitoring*, IEEE Catalogue No. 75-CH 1004-1 ICESA, Institute of Electrical and Electronics Engineers, 345 East [illegible] Street, New York, New York 10017, 1976).

[17] Guidance on meteorological measurements in the context of emergency response is provided in Regulatory Guide 1.101, "Emer[illegible] Planning and Preparedness for Nuclear Power Reactors." Guidance on meteorological instrumentation is contained in Regulatory Guide [illegible] "Onsite Meteorological Programs." A proposed revision to this guide has been issued for comment as Task SS 926-4.

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE E (Continued)			
Accident Sampling[18] Capability (Analysis Capability On Site)			
Primary Coolant and Sump	Grab Sample	3[4,19]	Release assessment; verification; analysis
• Gross Activity	1 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[20]	0 to 2000 cc(STP)/kg		
• Dissolved Oxygen[20]	0 to 20 ppm		
• pH	1 to 13		
Containment Air	Grab Sample	3[4]	Release assessment; verification; analysis
• Hydrogen Content	0 to 10 vol-% 0 to 30 vol-% for inerted containments		
• Oxygen Content	0 to 30 vol-%		
• Gamma Spectrum	(Isotopic analysis)		

[18]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 on sea or brackish water sites. Plants on fresh water sites should perform analysis within 96 hours.

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

[20]Within the first 30 days after an accident, oxygen analysis need not be performed until chloride analysis indicates a chloride concentration greater than 0.15 ppm. Once the chloride concentration exceeds this value, oxygen should be determined within 3 hours. For this 30-day period, it is acceptable to verify that dissolved oxygen is less than 0.1 ppm if the measured dissolved hydrogen residual is 10 cc/kg or less. However, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended. This applies only to primary coolant, not to sump.

TABLE 3

PWR VARIABLES

TYPE A Variables: those variables to be monitored that provide the primary information required to permit the con room operator to take specific manually controlled actions for which no automatic control is provided and that are requ for safety systems to accomplish their safety functions for design basis accident events. Primary information is infor tion that is essential for the direct accomplishment of the specified safety functions; it does not include those varia 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 1.4 and Table 1)	Purpose
Plant specific	Plant specific	1	Information required for operato action

TYPE B Variables: those variables that provide information to indicate whether plant safety functions are being accomplish Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (maintaining containment integrity (including radioactive effluent control). Variables are listed with designated ranges a 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; accomplishme 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 700°F	1	Function detection; accomplishment of mitigation; verification; long-term surveillance
RCS Cold Leg Water Temperature[1]	50°F to 700°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
Core Exit Temperature[1]	200°F to 2300°F	3[3]	Verification

[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.

[3] Instrumentation that is a part of the final ICC detection system should meet the design requirements specified in Item II.F.2 of NUREG-0737. (When Type K thermocouples become part of the system, they are considered to meet the requirements. However, the remainder of the detection system that is outside the reactor vessel should meet the requirements specified.)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE B (Continued)			
Core Cooling (Continued)			
Coolant Inventory	Bottom of hot leg to top of vessel[4]	1	Verification; accomplishment of mitigation
Degrees of Subcooling	200°F subcooling to 35°F superheat	2[3] (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 (plant specific)	2 1	Function detection; accomplishment of mitigation; verification
Containment Pressure[1]	0 to design pressure[5] (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]	-5 psig to design pressure[5]	1	Function detection; accomplishment of mitigation; verification

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	Purpose
Fuel Cladding			
Core Exit Temperature[1]	200°F to 2300°F	1[3]	Detection of potential for breach; accomplishment of mitigation; long-term surveillance

[4]A measurement to detect the *trend of voids in the* reactor coolant system with reactor coolant pumps running should also be provided for all plants. For B&W reactors, a measurement should be provided to detect voids in the hot leg candy cane when reactor coolant pumps are not running.

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

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE C (Continued)			
Fuel Cladding (Continued)			
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	1/2 Tech Spec limit to 100 times Tech Spec limit	1	Detection of breach
Analysis of Primary Coolant (Gamma Spectrum)	10 μCi/ml to 10 Ci/ml or TID-14844 source term in coolant volume	3[6]	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]	-5 psig to design pressure[4] (-10 psig for subatmospheric containments)	1	Detection of breach; accomplishment of mitigation; verification; long-term surveillance
Containment Sump Water Level[1]	Narrow range top to bottom (sump), wide range (plant specific)	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[7,8]	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[9]	Detection of breach; verification
Containment			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1[2]	Detection of potential for breach; accomplishment of mitigation

[6]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 N13.1-1969, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," 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.

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

[8]Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with a dose rate response accuracy within a factor of 2 over the entire range.

[9]Monitors should be capable of detecting and measuring gaseous effluent radioactivity 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 radioactivity may be expressed in terms of concentrations of Xe-133 equivalents, in terms of concentrations of any noble gas nuclides, or in terms of integrated gamma MeV per unit time. It is not expected that a single monitoring device will have sufficient range to encompass the entire range provided in this regulatory guide and that multiple components or systems will be needed. Existing equipment may be used to monitor any portion of the stated range within the equipment design rating.

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE C (Continued)			
Containment (Continued)			
Containment Hydrogen Concentration	0 to 10 vol-% (capable of operating from -5 psig to maximum design pressure[5]) 0 to 30 vol-% for ice-condenser-type containment	1	Detection of potential for breach; accomplishment of mitigation; long-term surveillance
Containment Pressure[1]	-5 psig pressure to 3 times design pressure[5] for concrete; 4 times design pressure for steel (-10 psig 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[9,10]	Detection of breach; accomplishment of mitigation, verification
Effluent Radioactivity[1] - Noble Gases (from buildings or areas where penetrations and hatches are located, e.g., secondary containment and auxiliary buildings and fuel handling buildings that are in direct contact with primary containment)	10^{-6} μCi/cc to 10^{3} μCi/cc	2[9]	Indication of breach

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

Variable	Range	Category	Purpose
Residual Heat Removal (RHR) or Decay Heat Removal System			
RHR System Flow	0 to 110% design flow[11]	2	To monitor operation
RHR Heat Exchanger Outlet Temperature	40°F to 350°F	2	To monitor operation and for analysis
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

[10] 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.

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

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE D (Continued)			
Safety Injection Systems (Continued)			
Boric Acid Charging Flow	0 to 110% design flow[11]	2	To monitor operation
Flow in HPI System	0 to 110% design flow[11]	2	To monitor operation
Flow in LPI System	0 to 110% design flow[11]	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	Top to bottom	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[5]	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[11]	3	To monitor operation

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE D (Continued)			
Auxiliary Feedwater or Emergency Feedwater System			
Auxiliary or Emergency Feedwater Flow	0 to 110% design flow[11]	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[11]	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[11]	2	To monitor operation
Letdown Flow - Out	0 to 110% design flow[11]	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	40°F to 200°F	2	To monitor operation
Component Cooling Water Flow to ESF System	0 to 110% design flow[11]	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[5]	3	To indicate storage capacity

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	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 (electric, hydraulic, pneumatic) (voltages, currents, pressures)	Plant specific	2[12]	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.

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
Containment Radiation			
Containment Area Radiation - High Range[1]	1 R/hr to 10^7 R/hr	1[7,8]	Detection of significant releases; release assessment; long-term surveillance; emergency plan actuation
Area Radiation			
Radiation Exposure Rate (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	3[8]	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[11] (not needed if effluent discharges through common plant vent)	2[9]	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[11] (not needed if effluent discharges through common plant vent)	2[9]	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[11] (not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases; release assessment; long-term surveillance

[12] Status indication of all standby power a.c. buses, d.c. buses, inverter output buses, and pneumatic supplies.

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	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[11] (not needed if effluent discharges through common plant vent)	2[9]	Detection of significant releases; release assessment
• Common Plant Vent or Multipurpose 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[11] 10^{-6} μCi/cc to 10^{4} μCi/cc	2[9]	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[13]	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[11] (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 (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[11]	3[14]	Detection of significant releases; release assessment; long-term surveillance

[13] 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.

[14] 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. For the purposes of this item only, "collection of representative samples" means obtaining the best samples practicable given the exigencies that attend the accident environment; line losses or line deposition should be empirically predetermined and appropriate loss correction factors should be applied.

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE E (Continued)			
Environs Radiation and Radioactivity[15]			
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	10^{-9} μCi/cc to 10^{-3} μCi/cc	3[16]	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[17] 3[17]	Release assessment; analysis
Plant and Environs Radioactivity (portable instrumentation)	(Isotopic Analysis)	3[17,18]	Release assessment; analysis
Meteorology[19]			
Wind Direction	0 to 360° (±5° accuracy with a deflection of 10°). Starting speed less than 0.4 mps (1.0 mph). Damping ratio greater than or equal to 0.4, delay distance less than or equal to 2 meters.	3	Release assessment
Wind Speed	0 to 22 mps (50 mph). ±0.2 mps (0.5 mph) accuracy for speeds less than 2 mps (5 mps), 10% for speeds in excess of 2 mps (5 mph), with a starting threshold of less than 0.4 mps (1.0 mph) and a distance constant not to exceed 2 meters.	3	Release assessment
Estimation of Atmospheric Stability	Based on vertical temperature difference from primary meteorological 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

[15] It is unlikely that a few fixed-station area monitors could provide sufficiently reliable information to be of use in detecting releases from unmonitored containment release points. However, there may be circumstances in which such a system of monitors may be useful. The decision to install such a system is left to the licensee.

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

[17] 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.

[18] A portable multichannel gamma ray spectrometer would provide the earliest capability for scoping the radionuclide content of the source (see R. C. Ragaini, D. E. Jones, and G. W. Huckabay, "Instrumentation for Off-Site Reactor Plume Studies," in *International Symposium on Environmental Monitoring*, IEEE Catalogue No. 75-CH 1004-1 ICESA, Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, New York 10017, 1976).

[19] Guidance on meteorological measurements in the context of emergency response is provided in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors." Guidance on meteorological instrumentation is contained in Regulatory Guide 1.23, "Onsite Meteorological Programs." A proposed revision to this guide has been issued for comment as Task SS 926-4.

TABLE 3 (Continued)

Variable	Range	Category (see Regulatory Position 1.4 and Table 1)	Purpose
TYPE E (Continued)			
Accident Sampling[20] Capability (Analysis Capability On Site)			
Primary Coolant and Sump	Grab Sample	3[6,21]	Release assessment; verification; analysis
• Gross Activity	1 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[22]	0 to 2000 cc(STP)/kg		
• Dissolved Oxygen[22]	0 to 20 ppm		
• pH	1 to 13		
Containment Air	Grab Sample	3[6]	Release assessment; verification; analysis
• Hydrogen Content	0 to 10 vol-% 0 to 30 vol-% for ice condensers		
• Oxygen Content	0 to 30 vol-%		
• Gamma Spectrum	(Isotopic analysis)		

[20] 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 for plants that use sea or brackish water in essential heat exchangers (i.e., shutdown cooling) that have only a single barrier from the reactor coolant. Other plants have 96 hours to perform a chloride analysis.

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

[22] Within the first 30 days after an accident, oxygen analysis need not be performed until chloride analysis indicates a chloride concentration greater than 0.15 ppm. Once the chloride concentration exceeds this value, oxygen should be determined within 3 hours. For this 30-day period, it is acceptable to verify that dissolved oxygen is less than 0.1 ppm if the measured dissolved hydrogen residual is 10 cc/kg or less. However, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended. This applies only to primary coolant, not to sump.

REGULATORY ANALYSIS

1. STATEMENT OF THE PROBLEM

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 modify and update the guidance previously given. The modification is based on the results of studies pertaining to radiation monitors, further evaluation of meteorological measurements, and initial input from independent evaluation of the overall clarity of the guide.

Regulatory Guide 1.97, Revision 2, was issued as an active guide in December 1980. The guide was issued with an outstanding question raised by the industry and supported by the Advisory Committee on Reactor Safeguards regarding the practicality of deploying at fixed locations environs radiation monitors capable of detecting radioactive material releases from an unidentified breach of the containment. These monitors were listed in the guide but implementation of these provisions of the guide was delayed pending the outcome of a study that was to develop guidance as to their number and location. Additionally, shortly after the guide was issued, a research program was initiated with INEL to identify any modifications to the guide that might make the intent more clear.

The study pertaining to the environs radiation monitors has been completed and published in NUREG/CR-2644, "An Assessment of Offsite, Real-Time Dose Measurement Systems for Emergency Situations."[1] The conclusion was that it is unlikely that a few fixed-station area monitors could provide sufficiently reliable information to be of use in detecting releases from unmonitored containment release points. The NRC staff agrees with the conclusion of this study, and the environs radiation monitors have been deleted from the PWR and BWR tables of variables of the guide.

Another evaluation by the NRC staff concluded that the function of exposure rate monitors inside auxiliary buildings and other buildings adjoining the containment (which were intended to measure releases caused by potential breaches in the containment) could be just as effectively performed by the effluent monitors installed at release points from those buildings. Therefore, the exposure rate monitors inside buildings for the purpose of detecting containment breach were deleted from the guide. Exposure rate monitors inside buildings where access is required to service equipment important to safety have been retained.

[1]Copies may be obtained from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

The NRC staff also agreed that the high accuracy specified in Revision 2 of Regulatory Guide 1.97 for the containment radiation monitors is unnecessary and should be reduced, since correction factors can be applied to compensate for the energy spectrum.

An additional change agreed to by the NRC staff pertains to meteorology measurements. During the Committee To Review Generic Requirements (CRGR) review of proposed Revision 1 of Regulatory Guide 1.23, "Meteorological Programs in Support of Nuclear Power Plants" (Task SS 926-4), the CRGR noted that several of the instrument range specifications on meteorology variables updated those presented in Revision 2 of Regulatory Guide 1.97 and recommended that both guides provide the same specifications. Regulatory Guide 1.97 has been modified to agree with Proposed Revision 1 of Regulatory Guide 1.23.

Of the clarifying modifications that have thus far been identified by the INEL evaluation, those that can be readily agreed to by the NRC staff are also included. These modifications include (1) listing the provisions for the design and qualification criteria for Categories 1, 2, and 3 in a different format that was recommended as being more understandable, (2) changing the "range" provisions in the tables of variables to make them consistent, (3) correcting editing and printing errors, and (4) clarifying the intent of the discussion and regulatory position of the guide.

The value to NRC operations and industry is that many of the questions regarding radiation monitoring will be resolved. Additionally, questions on guide intent frequently asked by industry will be settled by this revision.

2. OBJECTIVES

The above-mentioned deletions represent a substantive change in the NRC staff position regarding accident monitoring that could represent a reduction in cost to the users of Regulatory Guide 1.97 with no reduction in safety since the environs radiation monitors were found not to be needed, as discussed above, and the function of the exposure rate monitors inside buildings can be effectively performed by effluent monitors. It is desirable that the users of the guide be notified as soon as possible to prevent unnecessary costs being applied to meet a provision no longer recommended by the NRC staff. Since the guide is being revised to accomplish the above objectives, it is prudent to also include those modifications that have been identified as being essential to make the guide more understandable. Consequently, the guide is being revised to reflect these changes.

3. ALTERNATIVES

The alternative is to take no action to revise the guide but to inform licensees on an individual basis as interchanges between the licensee and the staff pertaining to accident monitoring occur.

4. CONSEQUENCES

If no action is taken, licensees and vendors may continue to incur costs to meet a provision that is no longer a recommendation of the staff. Time will be lost in answering questions that could be avoided by issuing a revision.

5. DECISION RATIONALE

The revision of the guide should be issued to inform its users of the current staff position, to clarify the staff position, and to eliminate or reduce unnecessary costs incurred by trying to meet provisions that are no longer recommended.

6. IMPLEMENTATION

The implementation for this revision of Regulatory Guide 1.97 does not alter the implementation of Revision 2 as outlined in §50.49 of 10 CFR Part 50 and Supplement 1 to NUREG-0737, "Clarification of TMI Action Plan Requirements."[2] Since there are no new recommendations, there is no adverse impact on cost or schedule.

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[2]NUREG-0737 may be obtained from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Supplement 1 is available for inspection or copying for a fee at the NRC Public Document Room, 1717 H Street, NW., Washington, D.C.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL. (Perry Nuclear Power Plant, Units 1 and 2)	)))))))	Docket No. 50-440 OL 50-441 OL

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSE IN SUPPORT OF APPLICANTS' MOTION FOR SUMMARY DISPOSITION OF ISSUE #14" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 4th day of February, 1985:

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