Draft 3
October 8, 1980
Division 1
Task RS 917-4

Contact: A. S. Hintze, (301) 443-5913

# [~~PROPOSED~~] REVISION 2 TO REGULATORY GUIDE 1.97

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

## A. INTRODUCTION

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

## 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 operator that will enable him 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 if a gross breach of a barrier has occurred. In addition to the above, indications of plant variables which 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 operator of manually initiated safety systems and other appropriate operator actions involving systems important to safety.

[~~Instrumentation is also needed to provide information about some plant parameters that is currently not available using present technology will alert the operator to conditions that have degraded beyond those postulated in the accident analysis. In particular, it is important that the operator be informed~~

~~regarding that status of coolant level in the reactor vessel or the existence of core voiding thus providing indication of potential degraded core cooling and imminent fuel damage. Direct indication of coolant level in the reactor vessel is not currently available in pressurized water reactors. However, it is imperative that this capability be developed within a reasonable time in order to provide the operator with this vital information in a positive, unambiguous manner.~~]

Independent of the above tasks, it is important that the operator be informed if the barriers to radioactive materials release are being challenged. Therefore, it is essential that instrument ranges be selected such 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 set-point value for automatic protection or alarms.) It is essential that degraded conditions and their magnitude be identified so that the operator can take actions that are available to mitigate the consequences. It is not intended that the operator be encouraged to prematurely circumvent systems important to safety but that he be adequately informed in order that unplanned actions can be taken when necessary.

Examples of serious events that could threaten safety if conditions degrade are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated operational occurrences which become accidents such as anticipated transients without scram (ATWS), reactivity excursions which result in releases of radioactive materials. Such events require that the operator understand, within a short time period, the ability of the barriers to limit radioactivity release, i.e., the potential for breach of a barrier, or an actual breach of a barrier by an accident in progress.

It is essential that the required instrumentation be capable of surviving the accident environment in which it is located for the length of time its function is required. It could therefore either be designed to withstand the accident environment or be protected by a local protected environment.

It is important that accident-monitoring instrumentation components and their mounts that cannot be located in Seismic Category I buildings be designed to continue to function, to the extent feasible, during seismic events. Consequently, it it is essential that they be designed to resist the effects of seismic excitation. An acceptable method for demonstrating the adequacy of the seimsic resistance of this instrumentation would be to qualify it to meet the seismic criteria applicable to instrumentation installed at other locations in the plant.

Variables selected 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 that the instruments will always be on scale. Further, it is prudent that a limited number of those variables which 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 operator will not be unaware as to the pressure inside containment. Provisions of such instruments are important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions determined. 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 the operator to use, during accident situations, instruments with which he is 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 compromise the accuracy and sensitivity required for normal operation. In some cases, this will necessitate use of overlapping ranges of instruments to monitor the required range of the variable to be monitored, possibly with different performance requirements in each range.

Standard ANS-4.5,* "Criteria for Accident monitoring Functions in a Light-Water-Cooled Nuclear Power Generating Station," dated ________ 1980, delineates criteria for determining the variables to be monitored by the control room operator, as required for safety, during the course of an accident and during the long-term stable shutdown phase followng an accident. Standard ANS-4.5 was prepared by Working Group 4.5 of Subcommittee ANS-4 with two primary objectives: (1) to address that instrumentation that permits the operator to monitor expected parameter changes in an accident period and (2) to address extended range instrumentation deemed appropriate for the possibility of encountering previously unforeseen events. ANS-4.5 references a revision to IEEE Std 497 as the source for specific instrumentation design criteria. Since the revision to IEEE Std 497 has not yet been completed, its applicability cannot yet be determined. Hence, specific instrumentation design criteria have been included in this regulatory guide.

The ANS standard defines three variable types (definitions modified herein) for the purpose of aiding the designer in his selection of accident-monitoring instrumentation and applicable criteria. The types are: Type A - those variables that provide primary** information needed to permit the control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and which 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 Position C.1.2). The sources of potential breach are limited to the energy sources within the barrier

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*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, LaGrange Park, Illinois 60525.

**Primary information is that which is essential for the direct accomplishment of the specified safety functions and does not include those variables which are associated with contingency actions that may also be identified in written procedures.

itself. In addition to the accident monitoring variables provided in ANS-4.5 standard, 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. They are: 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 Types 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 (a) to determine if the plant is responding to the safety measures in operation, (b) 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 being included as Type A also. Where such multiple use occurs, it is essential that instrumentation be capable of meeting the most 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.

Regulatory Positions C.1.3 and C.1.4 of this guide provide design and qualification criteria for the instrumentation used to measure the various variables listed in Table 1 (for BWR) and Table 2 (for PWR). The criteria are separated into three separate groups or categories which provide a graded approach to requirements depending on the importance to safety of a variable being measured. Category 1 provides the most stringent requirements and is intended for key variables. Category 2 requires less stringent requirements and generally applies to instrumentation designated for indicating system operating status. Category 3 is intended to provide requirements which will assure that high-quality off-the-shelf instrumentation is obtained and applies to backup and diagnostic instrumentation. It is also used where 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 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 indicate the accomplishment of a safety function (in the case of Types B & C) or the operation of a system safety (in the case of Type D) or radioactive materials 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 variables, 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 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 the operator to perform his 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 which provide the direct display of the variables. These provisions are not necessarily applicable to that part of the intrumentation systems provided as operator aids for the purpose of enhancement of information presentations for the identification or diagnosis of disturbances.

## C. REGULATORY POSITION

### 1. ACCIDENT MONITORING INSTRUMENTATION

The criteria, and requirements, contained in Standard ANS-4.5,"Criteria for Accident Monitoring Functions in a Light-Water-Cooled Nuclear Power Generating Station," dated __________ 1980, 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 In Section 3.2.1 of ANS-4.5, the definition of Type A variables should be modified to be as follows: Type A - those variables to be monitored that provide the primary information required to permit the control room operator to take the specified manually controlled actions for which no automatic control is provided and which are required for safety systems to accomplish their safety function for design basis accident events. (Note: Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which are associated with contingency actions that may also be identified in written procedures.)

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 parameters which have the potential for causing a breach in the primary reactor containment have exceeded the design basis values. In conjunction with the parameters that indicate the potential for causing a breach in the primary reactor containment, the parameters 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. References to Type C instruments, and associated parameters to be measured, in Standard 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

(1) The instrumentation should be qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). 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 to and including 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. 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.

(2) 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 operator from being presented the information necessary for him to determining 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) which could lead the operator to defeat or fail to accomplish a required safety function, additional information should be provided to allow the operator 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 which monitors a different variable which bears a known relationship to the multiple channels (addition of a diverse channel), or by providing the capability, if sufficient time is available, for the operator to perturb the measured variable and determine which channel has failed by observation of the response on each instrumentation channel. Redundant or diverse channels should be electrically independent and physically separated in accordance with Regulatory Guide 1.75 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.)

(3) The instrumentation should be energized from station Standby Power sources.

(4) An instrumentation channel should be available prior to an accident except as provided in Paragraph 4.11, "Exemption", as defined in IEEE Std 279 or as specified in Technical Specifications.

(5) The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

| | |
|---|---|
| Regulatory Guide 1.28 | "Quality Assurance Program Requirements (Design & 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 Insptection, Examination, and Testing Personnel" |
| Regulatory Guide 1.64 | "Quality Assurance Requirements for the Design of Nuclear Power Plants" |

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|---|---|
| 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" |
| Task RS 810-5 | "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (Guide number to be inserted.) |

Reference to the above regulatory guides (except Regulatory Guides 1.30, and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5) which is in progress.

(6) 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.

(7) 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 analog stripchart. Otherwise, it may be continuously updated, computer memory stored, 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

(1) The instrumentation should be qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). 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.

(2) The instrumentation should be energized from a high reliability power source, not necessarily Standby Power, battery backed where momentary interruption is not tolerable.

(3) 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) The recommendations of the following regulatory guides pertaining to quality assurance should be followed:

| | |
|---|---|
| Regulatory Guide 1.28 | "Quality Assurance Program Requirements (Design & 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 Insptection, 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" |
| Task RS 810-5 | "Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants" (Guide number to be inserted.) |

Reference to the above regulatory guides (except Regulatory Guides 1.30, and 1.38) are being made pending issuance of a regulatory guide endorsing NQA-1 (Task RS 002-5) which is in progress. 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, which 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) The instrumentation signal may be displayed on an individual instrument or it may be processed for display on demand by a CRT or other appropriate means.

(6) The method of display may be dial, digital, CRT or stripchart recorder indication. Effluent release monitors should be recorded, including effluent radioactivity monitors, environs exposure rate monitors, and meteorology monitors. Where direct and immediate trend or transient information is essential for operator information or action, the recording should be analog stripchart. Otherwise, it may be continuously updated, computer memory stored, and displayed on demand.

1.3.3 <u>Design and Qualification Criteria - Category 3</u>

(1) High quality commercial grade instrumentation selected to withstand the specified service environment.

1.4 In addition to the criteria of Position C.1.3, the following criteria should apply to Categories 1 and 2:

1.4.1 Any equipment that is used for either Category 1 or Category 2 should be designated as part of accident monitoring 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 monitoring instrumentation and that meet the provisions of this document.

1.4.2 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 should apply to Categories 1, 2 and 3.

1.5.1 Means should be provided for checking, with a high degree of confidence the operational availability of each monitoring channel, including its input sensor, during reactor operation. This may be accomplished n various ways, for example:

(1) By perturbing the monitored variable

(2) By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or

(3) By cross-checking between channels that bear a known relationship to each other and that have readouts available.

1.5.2 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.

1.5.3 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.

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

1.5.5 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.

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

1.5.7 To the extent [~~practical~~] <u>possible,</u> monitoring instrumentation inputs should be from sensors that directly measure the desired variables.

1.5.8 To the extent practical, the same instruments should be used for accident monitoring as are used for the normal operations of the plant to enable the operator to use, during accident situations, instruments with which he is 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.

1.5.9 Periodic testing should be in accordance with the applicable portions of Regulatory [illegible] 118 pertaining to testi[illegible] f instruments 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 sections, Tables 1, and 2 of this regulatory guide (which include those variables mentioned in the above sections) should be used as the minimum set of instruments and their respective ranges for accident-monitoring instrumentation for each nuclear power plant.

## 2. <u>SYSTEMS OPERATION MONITORING AND EFFLUENT RELEASE MONITORING INSTRUMENTATION</u>

### 2.1 <u>Definitions</u>

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

2.1.2 Type E - those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and 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:

2.2.1 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.

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

2.2.3 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:

2.3.1 For Type D

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

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

2.3.3 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;

(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:

(1) the range of the process variable.
(2) the required accuracy of measurement.
(3) the required response characteristics.
(4) the time interval during which the measurement is needed.
(5) the local environment(s) in which the information display channel components must operate.
(6) any requirement for rate or trend information.
(7) any requirements to group displays of related information.
(8) 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 Positions C.1.3 and C.1.4 of this guide. Tables 1 and 2 of this regulatory guide should be should be used as a minimum set of instruments and their respectives 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 1982 should meet the provisions of this guide.

Plants currently operating or scheduled to be licensed to operate before June 1, 1982 should meet the requirements of NUREG-0578 and NRR letters dated September 13, and October 30, 1979. The provisions of this guide as specified in Tables 1, and 2 for operating plants are compatible with these documents which are to be completed by January 1, 1981. The balance of provisions of the guide are to 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 requirements and schedules will be considered for extraordinary circumstances.

TABLE 1

BWR VARIABLES

Type A - 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 which are required for safety systems to accomplish their safety function for design basis accident events. Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which 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 Position C.1.3) | Purpose |
|---|---|---|---|
| Plant specific | plant specific | 1 | Information required for operator action |

TABLE 1

BWR VARIABLES (continued)

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, (4) containment integrity (which includes 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 (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE B VARIABLES | | | |
| Reactivity Control | | | |
| Neutron Flux | $10^{-6}$ to 5% 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 1000 ppm | 3 | Verification |
| Core Cooling | | | |
| Coolant Level in the Reactor | Bottom of core support plate to above the top of discharge plenum | 1 | Function detection; Accomplishment of mitigation; Long-term surveillance |
| BWR Core Thermocouples | Unresolved[5] | | To monitor core cooling if water level is low, spray is lost, or channels restricted. |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE B - continued | | | |
| Maintaining Reactor Coolant System Integrity | | | |
| RCS Pressure | 15 psia to 2000 psig | 1[4] | Function detection; Accomplishment of mitigation; Verification |
| Drywell Pressure[1] | 0 to design pressure[2] (psig) | 1 | Function detection; Accomplishment of mitigation; Verification |
| Drywell Sump Level[1] | Bottom to top | 2 | Function detection; Accomplishment of mitigation; Verification |
| Maintaining Containment Integrity | | | |
| Primary Containment Pressure (Drywell)[1] | 10 psia to design pressure[2] | 1 | Function detection; Accomplishment of mitigation; Verification |
| Primary Containment Isolation Valve Position (excluding check valves) | Closed - not closed | 1 | Accomplishment of isolation |

TABLE 1 (continued)

BWR VARIABLES (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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE C VARIABLES | | | |
| Fuel Cladding | | | |
| Radioactivity Concentration or Radiation Level in Circulating Primary Coolant | ½ Tech Spec limit to 100 Times Tech Spec limit R/hr | 1[6] | Detection of breach |
| Accident Sampling and Analysis of Primary Coolant<br>· Gross Activity<br>· Gamma Spectrum | 10 µCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume | 3[17] | Detail analysis;<br>Accomplishment of mitigation;<br>Verification;<br>Long-term surveillance |
| BWR Core Thermocouples | Unresolved[5] | | To monitor core cooling if water level is low, spray is lost, or channels restricted |
| Reactor Coolant Pressure Boundary | | | |
| RCS Pressure[1] | 15 psia to 1500 psig | 1[4] | Detection of potential for or actual breach;<br>Accomplishment of mitigation;<br>Long-term surveillance |
| Primary Containment Area Radiation[1] | 1 R/hr to $10^5$ R/hr | 3[7 11] | Detection of breach;<br>Verification |
| Drywell Drain Sumps[1] Level (Identified and Unidintified Leakage) | Bottom to top | 2 | Detection of breach;<br>Accomplishment of mitigation;<br>Verification;<br>Long-term surveillance |
| Suppression Pool Water Level (for operating plants) | Bottom of ECCS suction line to 5ft above normal water level | 1 | Same as immediately above |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE C - continued | | | |
| Reactor Coolant Pressure Boundary (continued) | | | |
| Drywell Pressure[1] | 0 to design pressure[2] (psig) | 1 | Detection of breach; Verification |
| Containment | | | |
| RCS Pressure[1] | 15 psia to 1500 psig | 1[4] | Detection of potential for breach; Accomplishment of mitigation |
| Primary Containment[1] Pressure (Drywell) | 10 psia pressure to 3 times design pressure[2] 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[2]) | 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[2]) | 1 | Detection of potential for breach; Accomplishment of mitigation |
| Containment Effluent[1] Radioactivity - Noble Gases (from identified release points including Standby Gas Treatment System Vent) | $10^{-6}$ to $10^{-2}$ μCi/cc | 3[9,10] | Detection of actual breach; Accomplishment of mitigation; Verification |
| Environs Radioactivity - Exposure Rate[1] | $10^{-6}$ to 10 R/hr | 2[8,11] | Detection of breach; Accomplishment of mitigation; Verification |

TABLE 1 (continued)

BWR VARIABLES (continued)

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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D VARIABLES | | | |
| Condensate and Feedwater System | | | |
| Main Feedwater Flow | 0 to 110% design flow[3] | 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[3] | 2 | To monitor operation |
| Drywell Pressure | 12 psia to 3 psig<br>0 to 110% design pressure[2] | 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 |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Main Steam System | | | |
| Main Steamline Flow | 0 to 120% design flow[3] | 1 | To monitor operation |
| Main Steamline Isolation Valves' Leakage Control System Pressure | 0 to 15" of water<br>0 to 5 psid | 1 | 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 | 1 | Detection of accident; boundary integrity indication |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Safety Systems | | | |
| RCIC Flow | 0 to 110% design flow[3] | 2 | To monitor operation |
| HPCI Flow | 0 to 110% design flow[3] | 2 | To monitor operation |
| Core Spray Flow | 0 to 110% design flow[3] | 2 | To monitor operation |
| RHR System Flow (LPCI) | 0 to 110% design flow[3] | 2 | To monitor operation |
| RHR Heat Exchanger Outlet Temperature (LPCI) | 32°F to 350°F | 2 | To monitor operation |
| SLCS Flow | 0 to 110% design flow[3] | 3 | To monitor operation |
| SLCS Storage Tank Level | Bottom to top | 3 | To monitor operation |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Cooling Water System | | | |
| ESF System Component Cooling Water Temperature | 32°F to 200°F | 2 | To monitor operation |
| ESF System Component Cooling Water Flow | 0 to 110% design flow[3] | 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 & Other energy Sources Important to Safety | Voltages, currents, pressures | 2[12] | To monitor operation |

TABLE 1 (continued)

BWR VARIABLES ( 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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E VARIABLES | | | |
| Containment Radiation | | | |
| Primary Containment Area Radiation - High Range[1] | 1 R/hr to $10^7$ R/hr | 1[7,11] | Detection of significant releases; Release assessment; Long-term surveillance; Emergency plan actuation |
| Reactor Bldg or Secondary Containment Area Radiation | $10^{-6}$ to $10^4$ μCi/cc | 2[10] | Detection of significant releases; Release assessment Long-term surveillance |
| Area Radiation | | | |
| Radiation Exposure Rate (Inside bldgs or areas where access is required to service equipment important to safety) | $10^{-1}$ R/hr to $10^4$ R/hr | 2[11] | Detection of significant releases; Release assessment; Long-term surveillance |
| Airborne Radioactive Materials Released from the Plant | | | |
| Noble Gases and Vent Flow Rate | | | |
| o Drywell Purge, Standby Gas Treatment System Purge (for Mark I, II, III plants) & Secondary Containment Purge (for Mark I plants) | $10^{-6}$ to $10^5$ μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |
| o Secondary Containment Purge (for Mark I, II, III plants) | $10^{-6}$ to $10^4$ μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - continued | | | |
| Airborne Radioactive Materials Released from the Plant | | | |
| Noble Gases and Vent Flow Rate (continued) | | | |
| o Secondary Containment (reactor shield bldg annulus, if in design) | $10^{-6}$ to $10^{4}$ μCi/cc<br>0 to 110% vent design flow[3]<br>(Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |
| o Auxiliary Building (including any bldg containing primary system gases, e.g., waste gas decay tank) | $10^{-6}$ to $10^{4}$ μCi/cc<br>0 to 110% vent design flow[3]<br>(Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |
| o Common Plant Vent or Multi-purpose Vent Discharging Any of the Above Releases | $10^{-6}$ to $10^{3}$ μCi/cc<br>0 to 110% vent design flow[3] | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |
| o All Other Identified Release Points | $10^{-6}$ to $10^{2}$ μCi/cc<br>0 to 110% vent design flow[3]<br>(Not needed if effluent discharges thru other monitored plant vents) | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |
| Particulates and Halogens | | | |
| o All Identified Plant Release Points. Sampling, with Onsite Analysis Capability | $10^{-3}$ to $10^{2}$ μCi/cc<br>0 to 110% vent design flow[3] | 3[13] | Detection of significant releases; Release assessment; Long-term surveillance |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - continued | | | |
| Environs Radiation and Radioactivity | | | |
| Radiation Exposure Rate[1] (Installed instrumentation) | $10^{-6}$ R/hr to 10 R/hr | 2[11] | Detection of significant releases; Verification; Release assessment; Long-term surveillance |
| Airborne Radiohalogens and Particulates (Sampling, with onsite analysis capability) | $10^{-9}$ to $10^{-3}$ $\mu$Ci/cc | 3[14] | Release assessment; Analysis |
| Plant and Environs Radiation (Portable Instrumentation) | 0.1 to $10^4$ R/hr, photons<br>0.1 to $10^4$ rads/hr, beta radiations and low-energy photons | 3[15]<br>3[15] | Release assessment; Analysis |
| Plant and Environs Radioactivity (Portable Instrumentation) | Multi-channel Gamma-Ray spectrometer | 3 | Releases assessment; Analysis |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - Continued | | | |
| 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 backup system. | 3 | Release assessment |

TABLE 1 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - (continued) | | | |
| ACCIDENT SAMPLING CAPABILITY* (Analysis Capability Onsite) | | | |
| Primary Coolant & Sump | Grab Sample | 3[17,18] | Release assessment; Verification; Analysis |
| o Gross Activity | 10 μCi/ml to 10 Ci/ml | | |
| o Gamma Spectrum | (Isotopic Analysis) | | |
| o Boron Content | 0 to 1 000 ppm | | |
| o Chloride Content | 0 to 20 ppm | | |
| o Disolved Oxygen | 0 to 20 ppm | | |
| o pH | 1 to 13 | | |
| Containment Air | Grab Sample | 3[17] | Release assessment; Verification; Analysis |
| o Hydrogen Content | 0 to 10%<br>0 to 30% for inerted containments | | |
| o Oxygen Content | 0 to 30% | | |
| o Gamma Spectrum | (Noble gas analysis) | | |

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

NOTES

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

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

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

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

[5]The number of thermocouples, their range and location to be determined.

[6]Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit-volume. System accuracy should be ±½ order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, and should not be a line or loop subject to isolation, e.g., main steam line. While such an instrument may not be currently available off-the-shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.

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

[8]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability. (Approximately 16 to 20 locations - site dependent.)

[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 only is 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 guide provided such monitoring has a range adequate to measure worst-case releases.

[10]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 of ±½ decade. 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 guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. Additional extended range instrumentation should overlap the range of existing instrumentation by a least a factor of 2.

[11]Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ±½ decade over the entire range.

TABLE 1 (continued)

NOTES - continued

[12]Status indication of all StandBy Power A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[13]To provide information regarding release of radioactive halogens and particulates. Continous 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 concentrations 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.

[14]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples. (Approximately 16 to 20 locations - site dependent.)

[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]Meteorological measurements should conform to the provisions of the forthcoming revision of Regulatory Guide 1.23, "Onsite Meteorological Programs".

[17]Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures 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 compatability,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Pre-arrangement for analysis and interpretation.

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

TABLE 2

PWR VARIABLES

Type A - 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 which are required for safety systems to accomplish their safety function for design basis accident events. Primary information is that which is essential for the direct accomplishment of the specified safety function and does not include those variables which 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 Position C.1.3) | Purpose |
|---|---|---|---|
| Plant specific | plant specific | 1 | Information required for operator action |

TABLE 2

PWR VARIABLES (continued)

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, (4) containment integrity (which includes 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 (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE B VARIABLES | | | |
| Reactivity Control | | | |
| Neutron Flux | $10^{-6}$ to 5% 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 Temperature[1] | 50°F to 400°F | 3 | Verification |
| Core Cooling | | | |
| RCS Hot Leg Temperature | 50°F to 750°F | 1 | Function detection; Accomplishment of mitigation; Verification; Long-term surveillance |
| RCS Cold Leg 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[4] | Function detection; Accomplishment of mitigation; Verification; Long-term surveillance |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE B - continued | | | |
| Core Cooling (continued) | | | |
| Core Exit Temperature[1] | 150°F to 2300°F (for operating plants - 150°F to 1650°F) | 3[5] | Verification |
| Coolant Level in Reactor | Bottom of core to top of vessel | 1 (Direct indicating or recording device not needed) | Verification |
| Degrees of Subcooling | 200°F subcooling to 35°F superheat | 1 (for operating plants - 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[4] | Function detection; Accomplishment of mitigation |
| Containment Sump Water Level | Narrow range (sump),<br>Wide range (bottom of containment to 600,000-gallon level equivalent) | 3<br>1 | Function detection; Accomplishment of mitigation; Verification |
| Containment Pressure[1] | 0 to design pressure[2] (psig) | 1 | Function detection; Accomplishment of mitigation; Verification |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE B - continued | | | |
| Maintaining Containment Integrity | | | |
| Containment Isolation Valve Position (excluding check valves) | Closed-not closed | 1 | Accomplishment of isolation |

TABLE 2 (continued)

PWR VARIABLES (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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE C VARIABLES | | | |
| Fuel Cladding | | | |
| Core Exit Temperature[1] | 150°F to 2300°F (for operating plants - 150°F to 1650°F) | 1[5] | Detection of potential for breach; Accomplishment of mitigation; Long-term surveillance |
| Radioactivity Concentration or Radiation Level in Circulating Primary Coolant | ½ Tech Spec limit to 100 times Tech Spec limit R/hr | 1[6] | Detection of breach |
| Accident Sampling and Analysis of Primary Coolant ·Gross Activity ·Gamma Spectrum | 10 μCi/gm to 10 Ci/gm or TID-14844 source term in coolant volume | 3[18] | 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[4] | Detection of potential for or actual breach; Accomplishment of mitigation; Long-term surveillance |
| Containment Pressure[1] | 10 psia to design pressure[2] psig (5 psia for sub-atmospheric containments) | 1 | Detection of breach; Accomplishment of mitigation; Verification; Long-term surveillance |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE C - continued | | | |
| Reactor Coolant Pressure Boundary (continued) | | | |
| Containment Sump Water Level[1] | Narrow range (sump), Wide range (bottom of containment to 600,000-gallon level equivalent) | 3<br>1 | Detection of breach; Accomplishment of mitigation; Verification; Long-term surveillance |
| Containment Area Radiation[1] | 1 to $10^4$ R/hr | 3[7,11] | Detection of breach; Verification |
| Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust[1] | $10^{-6}$ to $10^{-2}$ $\mu$Ci/cc | 3[10] | Detection of breach; Verification |
| Containment | | | |
| RCS Pressure[1] | 0 to 3000 psig (4000 psig for CE plants) | 1[4] | Detection of potential for breach; Accomplishment of mitigation |
| Containment Hydrogen Concentration | 0 to 10% (capable of operating from 10 psia to maximum design pressure[2])<br>0 to 30% for ice condenser type con[illegible]nment | 1 | Detection of potential for breach; Accomplishment of mitigat[illegible] Long-term [illegible]veillance |
| Containment Pressure[1] | 10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel | 1 | Detection of potential for or actual breach; Accomplishment of mitigation |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE C - continued | | | |
| Containment (continued) | | | |
| Containment Effluent Radioactivity - Noble Gases from Identified Release Points[1] | $10^{-6}$ to $10^{-2}$ μCi/cc | 2[9 10] | Detection of breach; Accomplishment of mitigation; Verification |
| Environs Radioactivity - Exposure Rate[1] | $10^{-4}$ to 10 R/hr | 2[11] | Detection of breach; Accomplishment of mitigation; Verification |

TABLE 2 (continued)

PWR VARIABLES (continued)

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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D VARIABLES | | | |
| Residual Heat Removal or Decay Heat Removal System | | | |
| RHR System Flow | 0 to 110% design flow[3] | 2 | To monitor operation |
| RHR Heat Exchanger Out Temperature | 32°F to 350°F | 2 | To monitor soperation and for analysis |
| Safety Injection Systems | | | |
| Accumulator Tank Level Level or Pressure | 10% to 90% volume<br>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[3] | 3 | To monitor operation |
| Flow in HPI System | 0 to 110% design flow[3] | 2 | To monitor operation |
| Flow in LPI System | 0 to 110% design flow[3] | 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 |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Primary Coolant System - (continued) | | | |
| 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 assure proper operation of pressurizer |
| Pressurizer Heater Status | Electric current | 3 | 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[2] | 3 | To monitor operation |
| Secondary System (Steam Generator) | | | |
| Steam Generator Level | From tube sheet to separators | 2 (Category 1 for 2-loop plants) | 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[3] | 3 | To monitor operation |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Auxiliary Feedwater or Emergency Feedwater System | | | |
| Auxiliary or Emergency Feedwater Flow | 0 to 110% design flow[3] | 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[3] | 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 | 3 | To indicate accomplishment of cooling |
| Containment Sump Water Temperature | 50°F to 250°F | 2 | To monitor operation |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE D - continued | | | |
| Chemical and Volume Control System | | | |
| Makeup Flow - In | 0 to 110% design flow[3] | 2 | To monitor operation |
| Letdown Flow - Out | 0 to 110% design flow[3] | 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 Components | 32°F to 200°F | 2 | To monitor operation |
| Component Cooling Water Flow to ESF System Components | 0 to 110% design flow[3] | 2 | To monitor operation |
| Radwaste Systems | | | |
| High-Level Radioactive Liquid Tank Level | Top to bottom | 3 | To indicate storage volume. |
| Radioactive Gas Hold-up Tank Pressure | 0 to 150% design pressure[2] | 3 | To indicate storage capacity |
| Ventilation Systems | | | |
| Emergency Ventilation Damper Position | Open-closed status | 2 | To indicate damper status |
| Power Supplies | | | |
| Status of Standby Power & Other Energy Sources Important to Safety | Voltages, currents, pressures | 2[13] | To indicate system status |

TABLE 2 (continued)

PWR VARIABLES (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 Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E VARIABLES | | | |
| Containment Radiation | | | |
| Containment Area Radiation - Hi Range[1] | 1 R/hr to $10^7$ R/hr | 1[7 11] | Detection of significant releases; Release assessment; Long-term surveillance; Emergency plan actuation |
| Area Radiation | | | |
| Radiation Exposure Rate (Inside bldgs or areas where access is required to service equipment important to safety) | $10^{-1}$ R/hr to $10^4$ R/hr | 2[11] | Detection of significant releases; Release assessment; Long-term surveillance |
| Airborne Radioactive Materials Released from the Plant | | | |
| Noble Gases and Vent Flow Rate | | | |
| o Containment or Purge Effluent[1] | $10^{-6}$ to $10^5$ μCi/cc<br>0 to 110% vent design flow[3]<br>(Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |
| o Secondary Containment (reactor shield bldg annulus, if in design) | $10^{-6}$ to $10^4$ μCi/cc<br>0 to 110% vent design flow[3]<br>(Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |
| o Auxiliary Building (including any bldg containing primary system gases, e.g., waste gas decay tank) | $10^{-6}$ to $10^4$ μCi/cc<br>0 to 110% vent design flow[3]<br>(not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - continued | | | |
| Airborne Radioactive Materials Release from the Plant (continued) | | | |
| Noble Gases and Vent Flow Rate (continued) | | | |
| o Condensor Air Removal System Exhaust[1] | $10^{-6}$ to $10^{5}$ μCi/cc 0 to 110% vent design flow[3] (Not needed if effluent discharges thru common plant vent) | 2[10] | Detection of significant releases; Release assessment |
| o Common Plant Vent or Multi-purpose Vent Discharging Any of the Above Releases | $10^{-6}$ to $10^{3}$ μCi/cc 0 to 110% vent design flow[3] | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |
| o Vent From Steam Generator Safety Relief Valves or Atmospheric Dump Valves | $10^{-1}$ 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 |
| o All Other Identified Release Points | $10^{-6}$ to $10^{2}$ μCi/cc 0 to 110% vent design flow (Not needed if effluent discharges thru other monitored plant vents) | 2[10] | Detection of significant releases; Release assessment; Long-term surveillance |
| Particulates and Halogens | | | |
| o All Identified Plant Release Points (except Steam Generator Safety Relief Valves or Atmospheric Steam Dump Valves and Condensor Air Removal System Exhaust) Sampling, With Onsite Analysis Capability | $10^{-3}$ to $10^{2}$ μCi/cc 0 to 110% vent design flow[3] | 3[14] | Detection of significant releases; Release assessment; Long-term surveillance |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - continued | | | |
| Environs Radiation and Radioactivity | | | |
| Radiation Exposure Rate[1] (Installed instrumentation) | $10^{-6}$ R/hr to 10 R/hr | 2 [11] | Detection of significant releases; Verification; Release assessment; Long-term surveillance |
| Airborne Radiohalogens and Particulates (Sampling, with onsite analysis capability) | $10^{-9}$ to $10^{-3}$ μCi/cc | 3 [15] | Release assessment; Analysis |
| Plant and Environs Radiation (Portable Instrumentation) | 0.1 to $10^4$ R/hr, photons<br>0.1 to $10^4$ rads/hr, beta radiations and low-energy photons | 3 [16]<br>3 [16] | Release assessment; Analysis |
| Plant and Environs Radioactivity (Portable Instrumentation) | Multi-channel Gamma-Ray spectrometer | 3 | Releases assessment; Analysis |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - Continued | | | |
| METEOROLOGY[17] | | | |
| 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 backup system. | 3 | Release assessment |

TABLE 2 (continued)

| Variable | Range | Category (see Position C.1.3) | Purpose |
|---|---|---|---|
| TYPE E - (continued) | | | |
| ACCIDENT SAMPLING CAP-* ABILITY (Analysis Capability Onsite) | | | |
| Primary Coolant & Sump | Grab Sample | 3[18 19] | Release assessment; Verification; Analysis |
| o Gross Activity | 10 μCi/ml to 10 Ci/ml | | |
| o Gamma Spectrum | (Isotopic Analysis) | | |
| o Boron Content | 0 to 6000 ppm | | |
| o Chloride Content | 0 to 20 ppm | | |
| o Disolved Oxygen | 0 to 20 ppm | | |
| o pH | 1 to 13 | | |
| Containment Air | Grab Sample | 3[18] | Release assessment; Verification; Analysis |
| o Hydrogen Content | 0 to 10%<br>0 to 30% for ice condensors | | |
| o Oxygen Content | 0 to 30% | | |
| o Gamma Spectrum | (Noble gas analysis) | | |

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

NOTES

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

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

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

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

[5]A minimum of 4 measurements per quadrant is required for operation. Sufficient number should be installed to account for attrition.

[6]Measurement should be made of the gross gamma radiation emanating from circulating primary coolant, with instrument calibration permitting conversion of readout to radioactivity concentrations in terms of either curies/gram or curies/unit-volume. System accuracy should be ±½ order of magnitude. The point of measurement should be external to a circulating primary coolant line or loop, such as a hot leg, and should not be a line or loop subject to isolation, e.g., letdown line. While such an instrument may not be currently available off-the-shelf, the staff considers that the necessary components are available commercially and have been employed and demonstrated under adverse environmental conditions in high-level hot cell operations for many years.

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

[8]For estimating release rates of radioactive materials released during an accident from unidentified release paths (not covered by effluent monitors) - continuous readout capability. (Approximately 16 to 20 locations - site dependent.)

[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 only is 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 guide provided such monitoring has a range adequate to measure worst-case releases.

[10]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 of ±½ decade. 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 guide and that multiple components or systems will be needed. Existing equipment may be utilized to monitor any portion of the stated range within the equipment design rating. Additional extended range instrumentation should overlap the range of existing instrumentation by a least a factor of 2.

[11]Detectors should respond to gamma radiation photons within any energy range from 60 keV to 3 MeV with an accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within ±½ decade over the entire range.

TABLE 2 (continued)

NOTES - continued

12 Effluent 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 ±½ order of magnitude. Calibration sources should fall within the range of approximately 0.5 MeV to 1.5 MeV (examples: Cs-137, Mn-54, Na-2?, 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 which cannot be detected or measured by the methods or techniques employed for monitoring.

13 Status indication of all Standby Power A-C buses, D-C buses, inverter output buses and pneumatic supplies.

14 To provide information regarding release of radioactive halogens and particulates. Continous 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 concentrations 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.

15 For estimating release rates of radioactive materials released during an accident.from unidentified release paths (not covered by effluent monitors). Continous collection of representative samples followed by laboratory measurements of the samples. (Approximately 16 to 20 locations - site dependent.)

16 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.

17 Meteorological measurements should conform to the provisions of the forthcoming revision of Regulatory Guide 1.23, "Onsite Meteorological Programs".

18 Sampling or monitoring of radioactive liquids and gases should be performed in a manner which assures 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 compatability,
c. Capability of sampling under primary system pressure and negative pressures,
d. Handling and transport capability, and
e. Pre-arrangement for analysis and interpretation.

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

# VALUE/IMPACT STATEMENT

## 1. PROPOSED ACTION

### 1.1 Description

The applicant (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 the 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 completion of 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 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 has 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 plants under construction to assure compliance 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. These instruments have extended ranges and there are others with qualification requirements not previously imposed. There will be additional impact due to a 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 which 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 & 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% 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, there are some concessions made in some of the requirements due to existing licensing committments which brings 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 through the Commission's regulations, in particular, Criterion 13, Criterion 19, and Criterion 64 of Appendix A to 10 CFR Part 50, which 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, emergency planning and its relationship to other agencies is provided by other means. Implementation of the proposed action is discussed in Section D of the proposed revision.

## 6. SUMMARY AND CONCLUSIONS

The revision 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 and implemented according to existing schedules.

## VALUE/IMPACT STATEMENT

### 1. PROPOSED ACTION

#### 1.1 Description

The applicant (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.

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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.

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