UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 17, 1980

MEMORANDUM TO: Distribution

FROM: A. S. Hintze
Reactor Systems Standards Branch
Division of Engineering Standards
Office of Standards Development

SUBJECT: REVISION 2 TO REGULATORY GUIDE 1.97

This memorandum is a summary report of the meeting held September 5, 1980 between industry and NRC staff personnel in which the scope of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," was discussed. Those on the distribution list were in attendance. After considerable discussion, the following conclusions were recorded from the meeting:

1. The regulatory guide should address requirements of the control room operating personnel. It was concluded that the development of the emergency response facilities (SPDS, TSC, EOF, NDL) were not sufficiently along to include identification of specific variables in Regulatory Guide 1.97 as being used by the emergency response facilities. Additionally, the emergency response facilities are to be designed by the licensee using general criteria provided by NRC, which is inconsistent with NRC specifying the variables to be used by those facilities. Bill Coley of Duke Power Company supplied some words for consideration. (See Attachment A).

2. The definition of Design Basis Accident Events should be as defined in ANS-4.5. Regulatory Guide 1.97 should not take exception by including anticipated operational occurrences in the guide definition.

3. The definition of Type A variables should be modified to more specifically identify the operator actions for Type A consideration. John Gallagher of Westinghouse supplied some words for consideration.(See Attachment B).

4. It should be made clear that the potential for breach of the fission product barriers be limited to the energy sources within the barrier itself.

5. The Type D & E variables should be made a separate part or section of the regulatory guide.

6. The discussion of the guide should explain how the design and qualification criteria category (Category 1, 2 or 3) was selected for each variable.

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The discussion of the meeting was limited (by time) to the scope of the guide. Time did not permit any discussion of the criteria which should be included in Categories 1, 2 and 3, or the list of variables in Tables 2 and 3, or the assignment of a category to a given variable.

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Enclosure: Distribution List.
Attachment A
Attachment B

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Distribution List

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Patrick Higgins	HIF
Alan S. Hintze	NRC/OSD
E. C. Wenzinger	NRC/OSD
Henry Martin	Nuclear Safety Associates, Inc.

Form 184 (5-76) Attachment A

Dev./Station ______ Unit ____ File No. ______

Subject: Understanding of the Scope of Reg. Guide 1.97 envisioned by the NRC

Date: 9-5-80

Sheet No. ___ of ___ Problem No. ______ Checked By ______ Date ______

Regulatory Guide 1.97 specifies the minimum parameters required by the control room operating staff during and following an accident. These parameters are used by the control room operating staff to perform their role in the emergency plan in the evaluation, assessment, monitoring, and execution of control room functions until the other emergency facilities ~~[illegible]~~ are effectively manned. Parameters are also defined to permit the operator to perform his long-term monitoring and execution responsibilities after the emergency facilities are manned.

Regulatory Guide 1.97 does not ~~[illegible]~~ address variables to be monitored in the other emergency facilities

Attachment B

Attachment 2: Inappropriateness of Class IE Emergency Response Facilities

The Regulatory Position in the Regulatory Guide references ANS 4.5's definition of type A, B, and C variables and the associated general criteria. As currently defined, both types A and B cover many functions that are performed by the Emergency Response Facilities (Technical Support Center, etc.) and consequently can lead to the application of requirements in section 6.0 of ANS 4.5 and Table 1 of draft Regulatory Guide 1.97 Revision 2 to the Emergency Response Facilities. These requirements would impose inappropriate Class 1E qualification and design criteria on these facilities.

In addition the definition of type A variables can lead to the application of these requirements to any instrumentation circuits which provide information to the operator that are identified in written procedures (pre-planned manual actions), independent of whether the action is required for safety purposes.

We believe that these potential problems can be corrected by the following modifications:

a. Modify the definition of type A variables to read:

   Type A variables are 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 functions for design basis accident events.

   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.

b. Change the scope of draft Regulatory Guide 1.97 Revision 2 to limit the application of the requirements for equipment to that part of the instrumentation system and its vital supporting features or power sources which provide the direct display of the process variables. Table 1 should contain a note that these requirements are not applicable to instrumentation systems provided as operator aids for the purpose of enhancement of information presentations for the identification or diagnosis of disturbances.

that part of

~~Integrated operational assessment~~ of

accident,

~~ASO~~ & Type A & B & C.

The requirements of the document

Working Paper E
8/20/80

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 and status of systems important to safety are required by the plant operating organization (licensee) during accident situations to (1) provide information required to permit the operator to take preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered-safety-feature systems, and manually initiated systems are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity); (3) provide information to the operator that will enable him to determine the potential for causing a 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; (4) furnish data regarding the operation of plant safety systems in order that the operating organization can make appropriate dicisions as to their use; and (5) 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 the impending threat.

As an aid to the plant operating organization to accomplish the above in carrying out its role to mitigate the consequences of an accident and to protect the health and safety of the public, four emergency response facilities have been identified as being essential. These facilities are: (1) the Safety Parameter Display System (SPDS), (2) the onsite Technical Support Center (TSC), (3) the Emergency Operations Facility (EOF), and (4) the Nuclear Data Link (NDL). The primary function of the SPDS is to help operating personnel in the control room make quick assessments of plant safety status. It will help determine whether plant safety functions are being performed. The primary function of the TSC is to provide overview monitoring of plant safety parameters to ensure that actions are taken promptly to correct or mitigate any abnormalities. The primary function of the EOF is to provide coordination of radiological assessments, manage overall emergency response and recovery operations, and provide assistance in the decision making process to protect the safety of the public. The primary function of the NDL is to transmit data to the NRC in order that the NRC staff can have information to make an independent evaluation of the situation as an aid to carrying out its role in advising the licensee and informing officials and the general public on all pertinent aspects of an accident.

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.

~~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 beyond what was anticipated in the safety analysis. 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 beyond those assumed in the Final Safety Analysis Report are loss-of-coolant accidents (LOCAs), overpressure transients, anticipated transients without scram (ATWS), and 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 the 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 is essential that they be designed to resist the effects of seismic excitation. An acceptable method for demonstrating the adequacy of the seismic 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 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.

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 impose severe operating requirements on instrumentation components, it may be necessary to upgrade those instrumentation components to withstand the more sever 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.

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 following 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 of specific instrumentation design criteria. Since the revision to IEEE Std 497 has not yet been completed, its applicability can not yet be determined. Hence, specific instrumentation design and qualification criteria have been included in this regulatory guide.

The 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. (A fourth and fifth type (Type D and Type E) have been added by this regulatory guide.) The types are: Type A - those variables that provide information needed for preplanned operator actions, Type B - those variables that provide information to indicate whether plant safety functions are being accomplished, 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 caldding, primary coolant pressure boundary, and containment (modified to reflect NRC staff position; see Position C.2),

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

Type D - those variables that provide information to indicate the operation of individual safety systems, and Type E - those variables to be monitored as required for us 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 oper tions 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, and (c) to implement emergency procedures. 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. Since the emergency response facilities (i.e., SPDS, TSC, EOF, NDL) are for the purpose of providing support and backup for detecting abnormalities, mitigating their consequences, and protecting the safety of the public, the variables to be monitored by these emergency facilities are taken from and/or included in the list of variables provided for Types B, C, D, and E. This will avoid duplication of requirements and help assure that information on power plant conditions is derived from the same data base, thus enhancing correlation of data concerning systems status and appropriate operator actions. (See NUREG-0696)

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 plant operating organization.

Regulatory Position C.7 of this guide provides design and qualification criteria for the instrumentation used to measure the various variables listed in Table 2 (for PWRs) and Table 3 (for BWRs). The criteria is grouped into three separate groups or categories which provide a graded approach to requirements depending on the importance to safety of a variable being measured.

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 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. The design and qualification criteria category assigned to each variable indicates whether the variable is considered to be a key variable or for backup or diagnosis.

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.

C. REGULATORY POSITION

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 and systems for accident conditions and for monitoring the reactor containment, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released during and following an accident from a nuclear power plant subject to the following:

1. Section 2.0 of ANS-4.5 defines the scope of the standard as containing criteria for determining the variables to be monitored by the control room operator of a light water reactor, as required for safety during the course of an accident. Consideration should be given to the additional requirements (e.g., emergency planning) of variables to be monitored by the plant operating organization during an accident in order that it can perform its role in protecting the health and safety of the public.

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 have the potential for causing a breach in the fuel cladding (e.g., core exit temperature) and the reactor coolant pressure boundary (e.g., reactor coolant pressure) should also be included. References to Type C instruments, and associated parameters to be measured, in 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.

3. Section 3.3 of ANS-4.5 defines design basis accident events. In addition to the design basis accident events delineated in the standard, those events defined as "anticipated operational occurrences" in 10 CFR 50 (which are excluded in the ANS-4.5 definition) should be also included.

4. Section 5.0, "Design Basis," of ANS-4.5 pertains to selection of accident monitoring variable and information display channels. In conjunction

with Items A, B, and C given in Section 5.0, the plant designer should identify information display channels required by his design to enable the operator to:

D. Ascertain the operating status of each individual safety system 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.

E. 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 continuously assessing such releases.

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

5. Section 5.1, "Variable Selection," pertains to the process for selection of accident monitoring variables. In conjuction with the selection of variables for Type A, Type B and Type C in Section 5.1, identification should be made of:

For Type D

1) the plant safety systems which are designed to help mitigate 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.

For Type E

1) the planned paths for effluent release;

2) plant areas or 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 will be detected;

4) the variables that should be monitored in each location identified in 1), 2) and 3) above and additional variables for defense in depth and for diagnosis.

6. Section 5.2, "Performance Requirement," of ANS 4.5 pertains to the determination of performance requirements for accident monitoring information display channels. In conjunction with Section 5.2, determination of performance requirements for Type D and Type E accident monitoring information display channels should be the same as for Type B.

7. 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 are established and should be used:

a. Category 1

(1) The instrumentation should be environmentally qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). Qualification applies to the complete instrumentation channel (end to end) from sensor to display or recording device, including isolation devices. The seismic portion of environmental 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 obtaining the information necessary for him to perform his role in knowing the safety status of the plant and in bringing the plant to and maintaining it in a safe condition following that accident. Where failure of one accident-monitoring channel results in the 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 that are required for him to perform his role. 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 channels should be electrically independent, energized from station Class 1E power, and physically separated in accordance with Regulatory Guide 1.75. (NOTE: Within each redundant division of a safety system, redundant monitoring channels are not required.)

(3) The instrumentation should be energized from station Class 1E power.

(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 following:

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.

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

(7) Recording of instrumentation readout information should be provided. Where trend or transient information is essential for operator information or action, the recording should be analog stripchart. Intermittent displays, such as data loggers and scanning recorders, may be used if no significant transient response can occur inside the recording interval.

(8) The instruments should be specifically identified on the control panels so that the operator can easily discern that they are intended for use under accident conditions.

b. Category 2

(1) The instrumentation should be environmentally qualified in accordance with Regulatory Guide 1.89 (NUREG-0588). Qualification applies from the sensor through the isolator/input buffer if the channel signal is to be processed for display on demand.

(2) The instrumentation should be energized from a high reliability non-Class 1E power source, 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 require recording, including effluent radioactivity monitors, environs exposure rate monitors, and meteorology monitors. Where trend or transient information is essential for operator information or action, the recording should be analog stripchart.

C. Category 3 - high quality commercial grade instrumentation select to withstand service environment.

8. In addition to the criteria of Position C.7, the following criteria should apply:

a. Any equipment that is used for both accident monitoring and non-safety functions should be classified as part of the accident-monitoring instrumentation. The transmission of signals from accident-monitoring equipment for nonsafety system use should be through isolation devices that are classified as part of the accident-monitoring instrumentation and that meet the provisions of the document.

b. Means should be provided for checking, with a high degree of confidence. The operational availability of each accident-monitoring channel, including its input sensor, during reactor operation. This may be accomplished in 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.

c. Servicing, testing, and calibration programs should be specified to maintain the capability of the accident-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.

d. Whenever means for bypassing channels are included in the design, the design should facilitate administrative control of the access to such bypass means.

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

f. The accident-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.

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

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

i. 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 accident-monitoring instrumentation results in a loss of instrumentation sensitivity in the normal operating range, separate instruments should be used.

j. Periodic testing should be in accordance with the applicable portions of Regulatory Guide 1.118 pertaining to testing of instruments channels.

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

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 2, and 3 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 2 (Revised

PWR VARIABLES

Variable	Range	Design & Qualification Criteria Category	Purpose
TYPE A - Variables for Preplanned Manual Actions	Plant specific	1	
TYPE B - Variables Indicating Critical Safety Functions			
Reactivity Control			
Control Rod Position	Full in or not full in	3 (for 2 hr minimum)	
Neutron Flux	1 c/s to 1% power	1	SPDS
Soluble Boron Content	0 to 6000 ppm (continuous indication)	3	
Boric Acid Charging Flow	0 to 110% design flow[1]	3	
Core Cooling			
Steam Genarator Level	From tube sheet to separators	1 (2 for B&W plants)	
RCS Hot Leg Temperature	50°F to 750°F	1	SPDS
Coolant Level in Reactor		1	
RCS Cold Leg Temperature	50°F to 750°F	1	SPDS
Reactor Coolant Loop Flow	0 to 120%] design -12% to 12%] flow[1]	1	
Degree of Subcooling	200°F subcooling to 35°F superheat	2	
Condensate Storage Tank Level	Plant specific	1 (3 if not primary source of AFW. Then whatever is primary source of AFW should be listed and should be Category 1)	

POOR ORIGINAL

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

TABLE 2 (continued) (Revised

Variable	Range	Category	Purpose
TYPE B - (continued)			
Maintaining Reactor Coolant System Integrity			
RCS Pressure	15 psia to 3000 psig (for CE plants, 4000 psig)	1^3	SPDS
Pressurizer Level	Bottom to top	1	SPDS
Primary System Safety Relief Valve Positions (including PORV and code valves) or Flow Through or Pressure in Relief Valve Lines	Closed-not closed	1	
Containment Sump Water Level	Narrow range (sump). Wide range (bottom of containment to 600,000-gallon level equivalent)	1	SPDS
Maintaining Containment Integrity			
Containment Pressure	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	1	SPDS
Containment Isolation Valve Position (excluding check valves)	Closed-not closed	1	
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[2]) 0 to 30% for ice condenser type containment	1	

POOR ORIGINAL

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

TABLE 2 (continued) (Revised

Variable	Range	Category	Purpose
TYPE C - Variables for Potential for or Breach of Barriers			
Fuel Cladding			
Core Exit Temperature	150°F to 2300°F	1[4]	SPDS
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	Normal to 10 Ci/gm	3[20]	
Reactor Coolant Pressure Boundary			
Containment High-Range Area Radiation	1 to 10^7 R/hr	1[5, 19]	SPDS
Effluent Radioactivity - Noble Gas Effluent from Condenser Air Removal System Exhaust	10^{-6} to 10^5 μCi/cc	2[16, 18]	SPDS
Steam Turbine Driven Auxiliary Feedwater Pump Vent	10^{-6} to 10^3 μCi/cc	2[16, 18]	
Containment			
Effluent Radioactivity - Noble Gases	10^{-6} to 10^5 μCi/cc	2[6, 16, 18]	SPDS
Environs Radioactivity - Exposure Rate	10^{-6} to 10 R/hr	2[7, 19]	

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Variable	Range	Category	Purpose
TYPE D - Variables Indicating Operation of Individual Safety Systems			
Secondary Systems			
Steam Generator Pressure	From atmospheric pressure to 20% above the lowest safety value setting	2	SPDS
Emergency Auxiliary Feedwater Flow	0 to 110% design flow[1]	2 (1 for B&W plants)	SPDS
Main Feedwater Flow	0 to 110% design flow[1]	3	SPDS
Safety/Relief Valve Positions or Main Steam Flow	Closed-not closed	2	
Radioactivity in Effluent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves	10^{1} to 10^{3} μCi/cc	2[17]	
Auxiliary Systems			
Sump Water Temperature	50°F to 250°F	2	
Containment Spray Flow	0 to 110% design flow[1]	2	
Containment Atmosphere Temperature	40°F to 400°F	3	
Heat Removal by the Containment Fan Heat Coolers Removal System	Plant specific	2	

TABLE 2 (continued) (Revised)

Variable	Range	Category	Purpose
TYPE D - (continued)			
Auxiliary Systems (continued)			
Flow in HPI System	0 to 110% design flow[1]	2	
Flow in LPI System	0 to 110% design flow[1]	2	
Emergency Refueling Coolant Water Storage Tank Level	Top to bottom	2	
Accumulator Tank Level or Pressure	10% to 90% volume or 0 to 750 psi	2	
Accumulator Isolation Valve Positions	Closed-not closed	2	
RHR System Flow	0 to 110% design flow[1]	2	
RHR Heat Exchanger Out Temperature	32°F to 350°F	2	
Component Cooling Water Temperature	32°F to 200°F	2	
Component Cooling Water Flow	0 to 110% design flow[1]	2	
Flow in Ultimate Heat Sink Loop	0 to 110% design flow[1]	2	
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	2	
Ultimate Heat Sink Water Level	Plant specific	2	
Letdown Flow - In	0 to 110% design flow[1]	2	
Letdown Flow - Out	0 to 110% design flow[1]	2	
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	3	
Steam Flow to Auxiliary Feedwater Pumps	0 to 110% design flow[1]	2	

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TABLE 2 (continued) Revised

Variable	Range	Category	Purpose
TYPE D - (continued)			
RADWASTE SYSTEMS			
High-Level Radioactive Liquid Tank Level	Top to bottom	3	
Radioactive Gas Hold-up Tank Pressure	0 to 150% of design pressure[2]	3	
VENTILATION SYSTEMS			
Emergency Ventilation Damper Position	Open-closed status	2	
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 180°F	3	
POWER SUPPLIES			
Status of Class 1E Power ~~Supplies~~ and Systems Sources	Voltages currents pressures	2[8]	
Status of Non-Class 1E Power ~~Supplies~~ and Systems Sources	Voltages currents pressures	3[9]	
TYPE E - Variables Which Indicate Magnitude and Direction of Dispersion of Released Radioactive Materials			
RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY-RELATED EQUIPMENT			
Radiation Exposure Rates	10^{-1} R/h to 10^{4} R/hr	3[19]	

Variable	Range	Category	Purpose
TYPE E - (continued)			
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT			
Gaseous Effluent Volumetric Flow Rate	0 to 110% design Flow flow[1]	2	
Effluent Radioactivity - Noble Gases			
.Secondary Containment (Reactor shield building annulus)	10^{-6} to 10^{4} µCi/cc	2[16] [18]	
.Auxiliary Building including buildings containing primary system gases, e.g., waste gas decay tank	10^{-6} to 10^{3} µCi/cc	2[16] [18]	
.Other Release Points (including fuel handling area if separate from auxiliary building)	10^{-6} to 10^{2} µCi/cc	2[16] [18]	
Effluent Radioactivity - ~~High Range~~ Radiohalogens and Particulates	10^{-3} to 10^{2} µCi/cc	2[10]	
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} µCi/cc for both radiohalogens and particulates	2[11]	
Plant and Environs Radioactivity & Radiation (portable instruments)	High Range 0.1 to 10^{4} R/hr photons 0.1 to 10^{4} rads/hr betas and low-energy photons	3[12]	
	multi-channel gamma-ray spectrometer	3	

Variable	Range	Category	Purpose
TYPE E - (continued)			
POSTACCIDENT SAMPLING* CAPABILITY (Analysis Capability Onsite)			
Primary Coolant & Sump	Grab Sample	3[13 14 21]	
Gross Activity	10 μCi/ml to 10 Ci/ml		
Gamma Spectrum	(Isotopic Analysis)		
Boron Content	0 to 6000 ppm		
Chloride Content	0 to 20 ppm		
Disolved Oxygen	0 to 20 ppm		
Disolved Hydrogen	0 to 1000 cc/kg		
pH	1 to 13		
Containment Air	Grab Sample	3[13 21]	
Hydrogen Content	0 to 10% 0 to 30% for ice condensors		
Oxygen Content	0 to 30%		
Gamma Spectrum	(Noble gas analysis)		
METEOROLOGY[15]			
Wind Direction	0 to 360° (±5° accuracy with a defliction of 15°. Starting speed 0.45 mps (1.0 mph) Damping ratio between 0.4 and 0.6, distance constant ≤2 meters)	2	
Wind Speed	0 to 30 mps (67 mph)(±0.22 mps (0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1.0 mph))	2	
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.	2	

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

NOTES continued -

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

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

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

[6]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 to the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[7]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.)

[8]Status indication of all Class 1E A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[9]Status indication of all non-Class 1E inverter output buses, D-C buses and pneumatic supplies.

[10]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 radioiodine 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.

[11]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.)

[12]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.

[13]To provide means for safe and convenient sampling These provisions should include:

1. Shielding to maintain radiation doses ALARA,
2. Sample containers with container-sampling port connector compatability,
3. Capability of sampling under primary system pressure and negative pressure,
4. Handling and transport capability, and
5. Pre-arrangement for analysis and interpretation.

NOTES continued -

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

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

[16]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 ± 1/2 decade. Calibration should be performed using radiation sources representative of both low and high energy portions of the emission spectrum. Fow low-energy gamma photon calibration, source emission energies should fall within the range of approximately 60 keV to 150 keV (examples - Am-241, Cd-109, Tm-171, and Co-57). For high-energy gamma photon calibration, source emission energies should fall within the range of approximately 500 keV to 1.5 MeV (examples - Cs-137, Mn-54, and Co-60). Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of the equivalent of any noble gas nuclide(s).

[17]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 ±1/2 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-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 which cannot be detected or measured by the methods or techniques employed for monitoring.

[18]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 at least a factor of 2.

[19]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 ±1/2 decade over the entire range.

[20]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 ±1/2 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., PWR letdown line or BWR 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.

[21]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.

TABLE 3

BWR VARIABLES

Variable	Range	Design & Qualification Criteria Category	Purpose
TYPE A - Variables for Preplanned Manual Actions	Plant specific	1	
TYPE B - Variables Indicating Critical Safety Functions			
Reactivity Control			
Control Rod Position	Full in or not full in	3 (for 1 hr minimum)	
Neutron Flux	1 c/s to 1% power	1	SPDS
Core Cooling			
Coolant Level in the Reactor	Bottom of core support plate to above top of discharge plenum	1	SPDS
Main Steamline Flow	0 to 120% design flow[1]	1	
Maintaining Reactor Cooling System Integrity			
RCS Pressure	15 psia to 1500 psig	1[3]	SPDS
Main Steamline Isolation Valves' Leakage Control System Pressure	0 to 15" of water 0 to 5 psid	1	
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	

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

TABLE 3 - (continued)

Variable	Range	Category	Purpose
TYPE B - (continued)			
Maintaining Containment Integrity			
Primary Containment Pressure (Drywell)	10 psia pressure to 3 times design pressure[2] for concrete; 4 times design pressure for steel	1	SPDS
Containment and Drywell Hydrogen Concentration	0 to 30% (capability of operating from 12 psia to maximum design pressure[2])	1	
Containment and Drywell Oxygen Concentration (for those plants with inerted containments)	0 to 20% (capability of operating from 12 psia to design pressure[2])	1	
Primary Containment Isolation Valve Position (excluding check valves)	Closed-not closed	1	
Suppression Chamber Air Temperature	30°F to 230°F	1	
Drywell Temperature	40°F to 440°F	1	
TYPE C - Variables for Potential for or Breach of Barriers			
Fuel Cladding			
Core Exit Temperature	150°F to 2300°F	1[4]	
Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	Normal to 10 Ci/gm	3[19]	SPDS

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

TABLE 3 - (continued)

Variable	Range	Category	Purpose
TYPE C - (continued)			
<u>Reactor Coolant Pressure Boundary</u>			
Containment High-Range Area Radiation	1 to 10^7 R/hr	1[5, 18] (for Mark III containments, two redundant monitors are required for primary containment & reactor building)	
Drywell Drain Sumps Level (Identified and Unidentified Leakage)	Bottom to top	1	SPDS
<u>Containment</u>			
Standby Gas Treatment System Vent	10^{-6} to 10^5 µCi/cc	2[16, 17]	SPDS
Effluent Radioactivity - Noble Gases	10^{-6} to 10^5 µCi/cc	2[6, 16, 17]	SPDS
Environs Radioactivity - Exposure Rate	10^{-6} to 10 R/hr	2[7, 18]	
TYPE D - Variables Indicating Operation of Individual Safety Systems			
<u>Power Conversion Systems</u>			
Main Feedwater Flow	0 to 110% design flow[1]	3	
Condensate Storage Tank Level	Bottom to top	3	

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TABLE 3 - (continued)

Variable	Range	Category	Purpose
TYPE D - (continued)			
Containment Systems			
Containment Spray Flow	0 to 110% design flow[1]	2	
Drywell Pressure	12 psia to 3 psig 0 to 110% design pressure[2]	2	SPDS
Suppression Chamber Water Level	Top of vent to top of weir well	2	SPDS
Suppression Chamber Water Temperature	30°F to 230°F	2	SPDS
Auxiliary Systems			
Control Rod Drive System Return Flow	0 to 110% design flow[1]	2	
Steam Flow to RCIC	0 to 110% design flow[1]	2	
HPCI Flow	0 to 110% design flow[1]	2	
RCIC Flow	0 to 110% design flow[1]	2	
Core Spray Flow	0 to 110% design flow[1]	2	
RHR System Flow (LPCI)	0 to 110% design flow[1]	2	
RHR Heat Exchanger Outlet Temperature (LPCI)	32°F to 350°F	2	SPDS
Service Cooling Water Temperature	32°F to 200°F	2	
Service Cooling Water Flow	0 to 110% design flow[1]	2	
Flow in Ultimate Heat Sink Loop	0 to 110% design flow[1]	2	
Temperature in Ultimate Heat Sink Loop	30°F to 150°F	2	
Ultimate Heat Sink Level	Plant specific	2	

TABLE 3 - (continued)

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Variable	Range	Category	Purpose
TYPE D - (continued)			
Auxiliary Systems (continued)			
SLCS Storage Tank Level	Bottom to top	3	
Sump Level in Spaces of Equipment Required for Safety	To corresponding level of safety equipment failure	3	
SLCS Flow	0 to 110% design flow[1]	3	
RADWASTE SYSTEMS			
High Radioactivity Liquid Tank Level	Top to bottom	3	
VENTILATION SYSTEMS			
Emergency Ventilation Damper Position	Open-closed status	2	
Temperature of Space in Vicinity of Equipment Required for Safety	30°F to 180°F	3	
POWER SUPPLIES			
Status of Class 1E Power ~~Supplies~~ and Systems Sources	Voltages currents pressures	2[8]	
Status of Non-Class 1E Power ~~Supplies~~ and Systems Sources	Voltages currents pressures	3[9]	

TABLE 3 - (continued)

Variable	Range	Category	Purpose
TYPE E - Varialbes Which Indicate Magnitude and Direction of Dispersion of Released Radioactive Materials			
RADIATION EXPOSURE RATES INSIDE BUILDINGS OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE SAFETY-RELATED EQUIPMENT			
Radiation Exposure Rates	R/hr 10^{-1} to 10^4 R/hr	2[18]	
AIRBORNE RADIOACTIVE MATERIALS RELEASED FROM THE PLANT			
Effluent Radioactivity - Noble Gases			
Reactor Bldg or Secondary Containment	10^{-6} to 10^4 μCi/cc	2[16 17]	SPDS
.Other Release Points (including fuel handling building, auxiliary building, and turbine building)	10^{-6} to 10^2 μCi/cc	2[16 17]	
Gaseous Effluent Volumetric Flow Rate	0 to 110% design flow[1]	2	
Effluent Radioactivity - ~~High-Range~~ Radiohalogens and Particulates	10^{-3} to 10^2 μCi/cc	2[10]	
Environs Radioactivity - Radiohalogens and Particulates	10^{-9} to 10^{-3} μCi/cc for both radiohalogens and particulates	2[11]	
Plant and Environs Radioactivity & Radiation (portable instruments)	High Range 0.1 to 10^4 R/hr photons 0.1 to 10^4 rads/hr betas and low-energy photons	3[12]	
	multi-channel gamma-ray spectrometer	3	

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Variables	Range	Category	Purpose
TYPE E - (continued)			
POSTACCIDENT SAMPLING* CAPABILITY (Analysis Capability Onsite)			
Primary Coolant & Sump	Grab Sample	3[13 14 20]	
Gross Activity	10 µCi/ml to 10 Ci/ml		
Gamma Spectrum	(Isotopic Analysis)		
Boron Content	0 to 1000 ppm		
Chloride Content	0 to 20 ppm		
Disolved Oxygen	0 to 20 ppm		
Disolved Hydrogen	0 to 1000 cc/kg		
pH	1 to 12		
Containment Air	Grab Sample	3[13 20]	
Hydrogen Content	0 to 30%		
Oxygen Content	0 to 30%		
Gamma Spectrum	(Noble gas analysis)		
METEOROLOGY[15]			
Wind Direction	0 to 360° (±5° accuracy with a defliction of 15°. Starting speed 0.45 mps (1.0 mph) Damping ratio between 0.4 and 0.6, distance constant ≤2 meters)	2	
Wind Speed	0 to 30 mps (67 mph)(±0.22 mps (0.5 mph) accuracy for wind speed less than 11 mps (25 mph), with a starting threshold of less than 0.45 mps (1.0 mph))	2	
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.	2	

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

NOTES continued -

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

[4]Approximately 50 thermocouples should be available, the exact number needed will depend on thermocouple location and other characteristics. In the absence of core spray the thermocouples should detect 5 to 10% core area cross sectional blockage, with high confidence. Sufficient numbers should be installed to account for attrition.

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

[6]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 to the environment. If two or more streams are combined prior to release from a common discharge point, monitoring of the combined stream is considered to meet the intent of this guide provided such monitoring has a range adequate to measure worst-case releases.

[7]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.)

[8]Status indication of all Class 1E A-C buses, D-C buses, inverter output buses and pneumatic supplies.

[9]Status indication of all non-Class 1E inverter output buses, D-C buses and pneumatic supplies.

[10]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 radioiodine 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.

[11]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.)

[12]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.

[13]To provide means for safe and convenient sampling. These provisions should include:

1. Shielding to maintain radiation doses ALARA,
2. Sample containers with container-sampling port connector compatability,
3. Capability of sampling under primary system pressure and negative pressure,
4. Handling and transport capability, and
5. Pre-arrangement for analysis and interpretation.

NOTES continued -

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

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

16 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 ± 1/2 decade. Calibration should be performed using radiation sources representative of both low and high energy portions of the emission spectrum. Fow low-energy gamma photon calibration, source emission energies should fall within the range of approximately 60 keV to 150 keV (examples - Am-241, Cd-109, Tm-171, and Co-57). For high-energy gamma photon calibration, source emission energies should fall within the range of approximately 500 keV to 1.5 MeV (examples - Cs-137, Mn-54, and Co-60). Effluent concentrations may be expressed in terms of Xe-133 equivalents or in terms of the equivalent of any noble gas nuclide(s).

17 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 at least a factor of 2.

18 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 ±1/2 decade over the entire range.

19 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 ±1/2 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., PWR letdown line or BWR 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.

20 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.