May 21, 1990

Mr. Stephen D. Floyd
Chairman BWR Owners' Group
Carolina Power and Light
411 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Floyd:

SUBJECT: POSITION ON NRC REGULATORY GUIDE 1.97, REVISION 3 REQUIREMENTS FOR POST-ACCIDENT NEUTRON MONITORING SYSTEM

I am responding to your letter of February 21, 1990 in which you raised several issues concerning post-accident neutron flux monitoring systems (NFMS). The issues were in the areas of environmental qualification, fire and flood conditions, availability, and range. For a detailed discussion of these issues please see the Enclosure.

The staff agrees that the BWR Owners' Group (BWROG) should develop generic design criteria for post-accident NFMS. It is the staff's position that NFMS should be environmentally qualified for the design basis accident spectrum in accordance with Code of Federal Regulations 10 CFR 50.49 and not other postulated events (e.g., fire). The staff will consider plant specific justifications for deviations in instrumentation range above 10^{-6} percent full power. However, the staff believes that 10^{-6} percent full power can be achieved and is appropriate to monitor shutdown neutron flux.

We look forward to prompt resolution of these issues, certainly no later than July 1990. If you have any questions regarding the above information, please contact Barry Marcus, of my staff on 49-20776.

Sincerely,

/S/

William T. Russell, Associate Director
for Inspection & Technical Assessment
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc: A. Udy (EG&G Idaho)

SEE PREVIOUS CONCURRENCE

SICB:DST*	SICB:DST*	SICB:DST*	SRXB:DST*	SRXB:DST*	SRXB:DST*	D:DST*
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5/ /90	5/ /90	5/ /90	5/ /90	5/ /90	5/ /90	5/ /90

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ENCLOSURE

ISSUES RAISED BY BWROG LETTER DATED FEBRUARY 21, 1990

Regulatory Guide (R.G.) 1.97 includes a Category 1 neutron flux monitoring system (NFMS) to monitor reactivity control during post-accident situations.

Reference 1 submitted the BWR Owners' Group (BWROG) Licensing Topical Report (LTR) for staff review and approval. The LTR proposed functional criteria for NFMS as an alternative to the Category 1 criteria identified in R.G. 1.97. Reference 2 provided the NRC Safety Evaluation Report (SER) which found the LTR to be unacceptable. Reference 3 raised issues in the areas of environmental qualification, fire and flood conditions, availability, and range such as:

1. The staff has not identified specific events on which to base qualification requirements or low end minimum range.

2. The staff's intention is not to require qualification of the NFMS for environments beyond those associated with design basis events.

3. The staff expects licensees to propose and technically justify specific design criteria to resolve open issues. It was suggested by the staff that this work could be performed generically in order to avoid multiple reviews by the staff for each licensee.

4. There appears to be a discrepancy between Reference 2 and Code of Federal Regulations 10 CFR 50.49. 10 CFR 50.49 requires that environmental conditions be established for the most severe design basis accident (DBA) during or following which instrumentation is required to function. However, Reference 2 takes the position that the post-accident NFMS design should consider events that cannot be anticipated by standard event analyses. This is a design issue which could raise 10 CFR 50.49 compliance concerns unless it is resolved with the staff prior to individual plant implementation of the regulatory guide.

5. Reference 2 takes the position that NFMS should be qualified in accordance with 10 CFR 50.49 and also qualified for events that go beyond 10 CFR 50.49 design basis events.

R.G. 1.97 includes a Category 1 NFMS to monitor reactivity control during post-accident situations. 10 CFR 50.49 requires that certain post-accident monitoring equipment (as provided in Revision 2 of R.G. 1.97) be environmentally qualified under conditions existing during and following DBAs. This includes the Category 1 NFMS.

As indicated in Reference 3, the scenarios for which the recommended low end of the range (10^{-6} percent full power) might be needed to provide an early warning of abnormal reactivity conditions and possible return to criticality following shutdown have not been specifically defined. The conditions within and surrounding the reactor to be considered for environmental qualification of the NFMS should be those associated with the typical spectrum of design basis events. Conditions beyond that scope are not required. The appropriate maximum conditions within that envelope are those associated with the large break LOCA. Since the NFMS needs to be qualified to DBA environments there is no conflict with 10 CFR 50.49. Because NFMS are not required in deterministic DBA analyses, does not cause us to conclude that a discrepancy exists between the staff SER and 10 CFR 50.49. It is our view that a NFMS qualified for a DBA environment in accordance with 10 CFR 50.49 would be very likely to survive for a spectrum of accidents "beyond" the DBA. There could be severe accident sequences postulated for which such a system would not survive, however, we would expect that the need to monitor return to criticality and shutdown margin would not be the primary safety concern for these events. It is not the staff's intent to require qualification beyond the environment associated with design basis events.

In Reference 3 the BWROG stated that Reference 2 suggested that fire conditions be considered, and that the inclusion of fire conditions in the design of NFMS appears to be outside the scope of R.G. 1.97. The BWROG also stated that there are other design issues such as flooding that also need to be considered.

The fire conditions referred to in Reference 2 were in reference to conceptual causes for control rod actuations and position information loss and not to environmental conditions used for qualification of the NFMS. The discussion of flooding was in the context of the environmental conditions associated with DBA conditions.

Reference 3 stated that instrumentation availability time (as referenced in Reference 2) along with related design basis issues need to be established.

The time range for required operability of the NFMS is based on the staff judgment of the necessary and sufficient time frame in which it might usefully assist in diagnosis of possible recriticality problems, and not on specific events. In the staff's view this is the order of 60 days under conditions which might exist in the DBA environment indicated above. The staff understands, based on conversations with vendors, that the NFMS being developed are being environmentally qualified for 100 days post-accident for the General Electric system and 6 months post-accident for the Gamma-Metrics system.

Reference 3 stated that the staff is aware of the potential difficulty for plant-specific implementation constraints with respect to achieving the full range specified by R.G. 1.97. The staff is willing to consider plant specific deviations based on technical justifications and the capability of available equipment.

The sensitivity range of the system, down to 10^{-6} percent of full power, was intended to provide the potential for maximum sensitivity to anomalous shutdown reactivity conditions during the time range indicated above. A normal shutdown (by scram) would typically reach the vicinity of this level (as determined by neutron sources developed during operation) in about 15 to 22 minutes and would remain near that level for an extended period. A less extended lower range, for example, to 10^{-5} percent or possibly 10^{-4} percent, would be somewhat less sensitive, but still useful. The staff would consider, on a plant specific basis, a less extended range for NFMS which cannot meet the 10^{-6} percent level. At the present time WNP-2 has installed a system with a range of 10^{-6} to 100 percent full power and Susquehanna Units 1 and 2 have installed systems with a range of 3.3×10^{-5} to 100 percent full power.

Reference 3 stated that it may be appropriate for the BWROG to develop generic design criteria for post-accident neutron monitoring which could serve as a focal point for further discussion with and review by the staff. It is the staff's position that the environmental qualification, fire and flood conditions, time availability and range questions raised in Reference 3 have been answered. However, these issues appear worthy of some additional review by the BWROG to substantiate our judgment.

NFMS have been installed at BWR plants for which the licensees have certified that the Category 1 criteria of R.G. 1.97 and 10 CFR 50.49 have been met. Therefore the installation of NFMS that comply with the Category 1 criteria of R.G. 1.97 and 10 CFR 50.49 is achievable and has been accomplished.

REFERENCES

1) BWR Owners' Group letter (R. F. Janecek) to NRC (T. E. Murley), "BWR Owners' Group Licensing Topical Report 'Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System,' (General Electric Report NEDO 31558)," April 1, 1988.

2) NRC letter (F. J. Miraglia) to BWR Owners' Group (S. D. Floyd), "BWR Owners' Group Licensing Topical Report 'Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System,' (General Electric Report NEDO 31558)," January 29, 1990.

3) BWR Owners' Group letter (S. D. Floyd) to NRC (F. J. Miraglia), "Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System (NMS)," February 21, 1990.

REGULATORY GUIDE 1.97
BWR CATEGORY 1 VARIABLES
12 VARIABLES

VARIABLE	TYPE	CAT	SYSTEM
PLANT SPECIFIC TYPE A	A	1	
CONTAINMENT & DRYWELL HYDROGEN CONCENTRA	C	1	CONTAINMENT
CONTAINMENT & DRYWELL OXYGEN CONCENTRA	C	1	CONTAINMENT
COOLANT LEVEL IN REACTOR (INVENTORY)	B	1	CORE COOLING
DRYWELL DRAIN SUMP LEVEL	C	1	REACTOR COOLANT PRESSURE BOUNDRY
DRYWELL PRESSURE	B,C	1	MAINTAINING RCS INTEGRITY, REACTOR COOLANT PRESS BOUNDRY
DRYWELL SUMP LEVEL	B	1	MAINTAINING RCS INTEGRITY
NEUTRON FLUX	B	1	REACTIVITY CONTROL
PRIMARY CONTAINMENT AREA RADIATION	E	1	CONTAINMENT RADIATION
PRIMARY CONTAINMENT ISOLATION VALVE POS	B	1	MAINTAINING CONTAINMENT INTEGRITY
PRIMARY CONTAINMENT PRESSURE	B,C	1	MAINTAINING CONTAINMENT INTEGRITY, CONTAINMENT
RCS PRESSURE	B,C	1	MAINT RCS INTEGRITY, REACTOR COOLANT PRESS BOUNDRY, CONT
SUPPRESSION POOL WATER LEVEL	C	1	REACTOR COOLANT PRESSURE BOUNDRY

Revision 2*
December 1980

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.97
(Task RS 917-4)

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

A. INTRODUCTION

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," includes a requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentation, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor.

Criterion 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50 includes a requirement that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

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

B. DISCUSSION

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

At the start of an accident, it may be difficult for the operator to determine immediately what accident has occurred or is occurring and therefore to determine the appropriate response. For this reason, reactor trip and certain other safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) have been designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant variables required to enable the operation of manually initiated safety systems

and other appropriate operator actions involving systems important to safety.

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

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

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

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

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

Normal power plant instrumentation remaining functional for all accident conditions can provide indication, records, and (with certain types of instruments) time-history responses for many variables important to following the course of the accident. Therefore, it is prudent to select the required accident-monitoring instrumentation from the normal power plant instrumentation to enable operators to use, during accident situations, instruments with which they are most familiar. Since some accidents could impose severe operating requirements on instrumentation components, it may be necessary to upgrade those normal power plant instrumentation components to withstand the more severe operating conditions and to measure greater variations of monitored variables that may be associated with an accident. It is essential that instrumentation so upgraded does not degrade the accuracy and sensitivity required for normal operation. In some cases, this will necessitate use of overlapping ranges of instruments to monitor the required range of the variable to be monitored, possibly with different performance requirements in each range.

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

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

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

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

operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events; Type B, those variables that provide information to indicate whether plant safety functions are being accomplished; and Type C, those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release, i.e., fuel cladding, primary coolant pressure boundary, and containment (modified to reflect NRC staff position; see regulatory position 1.2). The sources of potential breach are limited to the energy sources within the barrier itself. In addition to the accident-monitoring variables provided in ANS-4.5, variables for monitoring the operation of systems important to safety and radioactive effluent releases are provided by this regulatory guide. Two additional variable types are defined: Type D, those variables that provide information to indicate the operation of individual safety systems and other systems important to safety, and Type E, those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases.

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

The time phases (Phases I and II) delineated in ANS-4.5 are not used in this regulatory guide. These considerations are plant specific. It is important that the required instrumentation survive the accident environment and function as long as the information it provides is needed by the control room operating personnel.

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

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

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

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

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

C. REGULATORY POSITION

1. Accident-Monitoring Instrumentation

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

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

1.2 In Section 3.2.3 of ANS-4.5, the definition of Type C includes two items, (1) and (2). Item (1) includes those instruments that indicate the extent to which variables that have the potential for causing a breach in the primary reactor containment have exceeded the design basis values. In conjunction with the variables that indicate the potential for causing a breach in the primary reactor containment, the variables that indicate the potential for causing a breach in the fuel cladding (e.g., core exit temperature) and the reactor coolant pressure boundary (e.g., reactor coolant pressure) should also be included. The sources of potential breach are limited to the energy sources within the cladding, coolant boundary, or containment. References to Type C instruments, and associated parameters to be measured, in ANS-4.5 (e.g., Sections 4.2, 5.0, 5.1.3, 5.2, 6.0, 6.3) should include this expanded definition.

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

1.3.1 Design and Qualification Criteria - Category 1

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

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

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

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

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

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

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

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

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

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

1.3.2 Design and Qualification Criteria - Category 2

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

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

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

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

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

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

1.3.3 Design and Qualification Criteria - Category 3

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

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

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

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

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

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

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

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

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

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

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

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

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

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

1.6 Sections 6.2.2, 6.2.3, 6.2.4, 6.2.5, 6.2.6, 6.3.2, 6.3.3, 6.3.4, and 6.3.5 of ANS-4.5 pertain to variables and variable ranges for monitoring Types B and C variables. In conjunction with the above-listed sections of ANS-4.5, Tables 1 and 2 of this regulatory guide (which include those variables mentioned in these sections) should be considered as the minimum number of instruments and their respective ranges for accident-monitoring instrumentation for each nuclear power plant.

2. Systems Operation Monitoring and Effluent Release Monitoring Instrumentation

2.1 Definitions

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

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

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

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

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

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

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

a. For Type D

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

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

b. For type E

(1) The planned paths for effluent release;

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

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

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

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

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

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

D. IMPLEMENTATION

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

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

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

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

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

TABLE 1

BWR VARIABLES

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

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

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

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

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

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

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

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

TABLE 1 (Continued)

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

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

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

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

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

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

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

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

TABLE 1 (Continued)

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

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

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

TABLE 1 (Continued)

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

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

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

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

TABLE 1 (Continued)

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

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

TABLE 1 (Continued)

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

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

TABLE 1 (Continued)

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

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[12] To provide information regarding release of radioactive halogens and particulates. Continuous collection of representative samples followed by onsite laboratory measurements of samples for radiohalogens and particulates. The design envelope for shielding, handling, and analytical purposes should assume 30 minutes of integrated sampling time at sampler design flow, an average concentration of 10^{2} μCi/cc of radioiodines in gaseous or vapor form, an average concentration of 10^{3} μCi/cc of particulate radioiodines and particulates other than radioiodines, and an average gamma photon energy of 0.5 MeV per disintegration.

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

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

TABLE 1 (Continued)

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

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

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

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

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

TABLE 2

PWR VARIABLES

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

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

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

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

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

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

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

TABLE 2 (Continued)

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

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

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

TABLE 2 (Continued)

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

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

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

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

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

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

[8] Monitors should be capable of detecting and measuring radioactive gaseous effluent concentrations with compositions ranging from fres equilibrium noble gas fission product mixtures to 10-day-old mixtures, with overall system accuracies within a factor of 2. Effluent concentr tions 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 devi will have sufficient range to encompass the entire range provided in this regulatory guide and that multiple components or systems will needed. Existing equipment may be used to monitor any portion of the stated range within the equipment design rating.

TABLE 2 (Continued)

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

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

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

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

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

TABLE 2 (Continued)

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

TABLE 2 (Continued)

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE D (Continued)			
Ventilation Systems			
Emergency Ventilation Damper Position	Open-closed status	2	To indicate damper status
Power Supplies			
Status of Standby Power and Other Energy Sources Important to Safety (hydraulic, pneumatic)	Voltages, currents, pressures	2[11]	To indicate system status

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

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
Containment Radiation			
Containment Area Radiation - High Range[1]	1 R/hr to 10^7 R/hr	1[6,7]	Detection of significant releases; release assessment; long-term surveillance; emergency plan actuation
Area Radiation			
Radiation Exposure Rate[1] (inside buildings or areas where access is required to service equipment important to safety)	10^{-1} R/hr to 10^4 R/hr	2[7]	Detection of significant releases; release assessment; long-term surveillance
Airborne Radioactive Materials Released from Plant			
Noble Gases and Vent Flow Rate			
• Containment or Purge Effluent[1]	10^{-6} μCi/cc to 10^5 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Reactor Shield Building Annulus[1] (if in design)	10^{-6} μCi/cc to 10^4 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment
• Auxiliary Building[1] (including any building containing primary system gases, e.g., waste gas decay tank)	10^{-6} μCi/cc to 10^3 μCi/cc 0 to 110% vent design flow[10] (Not needed if effluent discharges through common plant vent)	2[8]	Detection of significant releases; release assessment; long-term surveillance

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

TABLE 2 (Continued)

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

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

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

TABLE 2 (Continued)

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

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

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

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

TABLE 2 (Continued)

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

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

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

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

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The applicant for a license (or licensee) of a nuclear power plant is required by the Commission's regulations to provide instrumentation to (1) monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety and (2) monitor the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents. This revision to Regulatory Guide 1.97 proposes to improve the guidance for plant and environs monitoring during and following an accident, including extended ranges for some instruments to account for consideration of degraded core cooling.

1.2 Need

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

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

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

1.3 Value/Impact of Proposed Action

1.3.1 NRC Operations

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

1.3.2 Other Government Agencies

Not applicable, unless the government agency is an applicant.

1.3.3 Industry

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

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

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

1.3.4 Public

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

No impact on the public can be foreseen.

1.4 Decision on Proposed Action

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

2. TECHNICAL APPROACH

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

3. PROCEDURAL APPROACH

Previously discussed.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

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

4.2 Need for NEPA Assessment

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

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

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

6. SUMMARY AND CONCLUSIONS

Revision 2 to Regulatory Guide 1.97, "Instrumentation For Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," should be issued.

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shall be implemented within 30 months of NRC approval. Other modifications requiring NRC approval prior to installation shall be implemented within 6 months after NRC approval.

(e) Nuclear power plants licensed to operate after January 1, 1979, shall complete all fire protection modifications needed to satisfy Criterion 3 of Appendix A to this part in accordance with the provisions of their licenses.

[45 FR 76610, Nov. 19, 1980]

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment:[3] This equipment is that relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment.[4]

(c) Requirements for (1) dynamic and seismic qualification of electric equipment important to safety, (2) protection of electric equipment important to safety against other natural phenomena and external events, and (3) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the following information for this electric equipment important to safety in a qualification file:

(1) The performance specifications under conditions existing during and following design basis accidents.

(2) The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

(e) The electric equipment qualification program must include and be based on the following:

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[3] Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323-1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, NY 10017.

[4] Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Copies of the Regulatory Guide may be purchased through the U.S. Government Printing Office by calling 202-275-2060 or by writing to the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082.

(1) *Temperature and pressure.* The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

(2) *Humidity.* Humidity during design basis accidents must be considered.

(3) *Chemical effects.* The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) *Radiation.* The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) *Aging.* Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.

(6) *Submergence* (if subject to being submerged).

(7) *Synergistic effects.* Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

(8) *Margins.* Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

(f) Each item of electric equipment important to safety must be qualified by one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

(g) Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complica-

tions, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification. The schedule in this paragraph supersedes the June 30, 1982, deadline, or any other previously imposed date, for environmental qualification of electric equipment contained in certain nuclear power operating licenses.

(h) Each license shall notify the Commission as specified in § 50.4 of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.

(i) Applicants for operating licenses granted after February 22, 1983, but prior to November 30, 1985, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted, as specified in § 50.4, for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

(1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.

(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section:

(1) Is qualified for its application; and

(2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines), or NUREG-0588 (For Comment version), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

(l) Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.

[48 FR 2733, Jan. 21, 1983, as amended at 49 FR 45576, Nov. 19, 1984; 51 FR 40308, Nov. 6, 1986; 51 FR 43709, Dec. 3, 1986; 52 FR 31611, Aug. 21, 1987]

Issuance, Limitations, and Conditions of Licenses and Construction Permits

§ 50.50 Issuance of licenses and construction permits.

Upon determination that an application for a license meets the standards and requirements of the act and regulations, and that notifications, if any, to other agencies or bodies have been duly made, the Commission will issue a license, or if appropriate a construction permit, in such form and containing such conditions and limitations including technical specifications, as it deems appropriate and necessary.

demand in regulated fresh channels is expected to require about 56 percent of the volume. The remaining 44 percent would be available for utilization in export and processing outlets. The committee indicates that volume and size composition of the crop of navels are such that more than ample supplies of the more desirable larger sizes will be available to satisfy the demand in regulated channels. The committee also reports that when more than ample supplies of larger sizes are available for shipment, disposition of the sizes which would be eliminated by this regulation can be accomplished only at a substantial price discount and this tends to depress the market for all sizes. Navel oranges failing to meet such requirements could be shipped to fresh export markets, left on trees to attain further growth, or utilized in processing. In those circumstances, elimination of sizes smaller than those specified is appropriate in the interest of producers and consumers.

It is further found that it is impracticable and contrary to the public interest to give preliminary notice, engage in public rulemaking, and postpone the effective date until 30 days after publication in the Federal Register (5 U.S.C. 553), because of insufficient time between the date when information became available upon which this regulation is based and the effective date necessary to effectuate the declared purposes of the act. Interested persons were given an opportunity to submit information and views on the regulation at an open meeting. Handlers have been apprised of such provisions and the effective date.

List of Subjects in 7 CFR Part 907

Marketing agreements and orders, California, Arizona, Oranges (Navel).

PART 907—[AMENDED]

Therefore, § 907.860 is added to read as follow: (§ 907.860 expires March 24, 1983, and will not be published in the annual code of Federal Regulations):

§ 907.860 Navel orange regulation 560.

(a) During the period January 21, 1983, through March 24, 1983, no handler shall handle any navel oranges grown in the production area which are of a size smaller than 2.32 inches in diameter: *Provided*, that not to exceed 5 percent, by count, of the oranges in any container may measure smaller than 2.32 inches in diameter.

(b) As used in this section, "handler", "handle" and "production area" mean the same as defined in the marketing order. Diameter shall mean the largest measurement at a right angle to a straight line running from the stem to the blossom end of the fruit.

(Secs. 1–19, 48 Stat. 31, as amended; 7 U.S.C. 601–674)

Dated: January 17, 1983.

D. S. Kuryloski,

Deputy Director, Fruit and Vegetable Division, Agricultural Marketing Service.

[FR Doc. 83-1886 Filed 1-20-83; 8:45 am]

BILLING CODE 3410-02-M

7 CFR Part 910

[Lemon Reg. 395]

Lemons Grown in California and Arizona; Limitation of Handling

AGENCY: Agricultural Marketing Service, USDA.

ACTION: Final rule.

SUMMARY: This regulation establishes the quantity of fresh California-Arizona lemons that may be shipped to market during the period January 23–29, 1983. Such action is needed to provide for orderly marketing of fresh lemons for the period due to the marketing situation confronting the lemon industry.

EFFECTIVE DATE: January 23, 1983.

FOR FURTHER INFORMATION CONTACT: William J. Doyle, Chief, Fruit Branch, F&V, AMS, USDA, Washington, D.C. 20250, telephone 202–447–5975.

SUPPLEMENTARY INFORMATION: This final rule has been reviewed under Secretary's Memorandum 1512–1 and Executive Order 12291, and has been designated a "non-major" rule. William T. Manley, Deputy Administrator, Agricultural Marketing Service, has determined that this action will not have a significant economic impact on a substantial number of small entities. This action is designed to promote orderly marketing of the California-Arizona lemon crop for the benefit of producers, and will not substantially affect costs for the directly regulated handlers.

This final rule is issued under Marketing Order No. 910, as amended (7 CFR Part 910; 47 FR 50196), regulating the handling of lemons grown in California and Arizona. The order is effective under the Agricultural Marketing Agreement Act of 1937, as amended (7 U.S.C. 601–674). The action is based upon recommendations and information submitted by the Lemon Administrative Committee and upon other available information. It is hereby found that this action will tend to effectuate the declared policy of the Act.

This action is consistent with the marketing policy for 1982–83. The marketing policy was recommended by the committee following discussion at a public meeting on July 6, 1982. The committee met again publicly on January 18, 1983, at Los Angeles, California, to consider the current and prospective conditions of supply and demand and recommended a quantity of lemons deemed advisable to be handled during the specified week. The committee reports the demand for lemons continues easier.

It is further found that it is impracticable and contrary to the public interest to give preliminary notice, engage in public rulemaking, and postpone the effective date until 30 days after publication in the Federal Register (5 U.S.C. 553), because of insufficient time between the date when information became available upon which this regulation is based and the effective date necessary to effectuate the declared purposes of the Act. Interested persons were given an opportunity to submit information and views on the regulation at an open meeting. It is necessary to effectuate the declared purposes of the Act to make these regulatory provisions effective as specified, and handlers have been apprised of such provisions and the effective time.

List of Subjects in 7 CFR Part 910

Marketing agreements and orders, California, Arizona, Lemons.

PART 910—[AMENDED]

Section 910.695 is added as follows:

§ 910.695 Lemon Regulation 395.

The quantity of lemons grown in California and Arizona which may be handled during the period January 23, 1983, through January 29, 1983, is established at 180,000 cartons.

(Secs. 1–19, 48 Stat. 31, as amended; 7 U.S.C. 601–674)

Dated: January 20, 1983.

D. S. Kuryloski,

Deputy Director, Fruit and Vegetable Division, Agricultural Marketing Service.

[FR Doc. 83-1908 Filed 1-20-83; 11:40 am]

BILLING CODE 3410-02-M

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Commission is amending its regulations applicable to nuclear power plants to clarify and strengthen the criteria for environmental qualification of electric equipment important to safety. Specific qualification methods currently contained in national standards, regulatory guides, and certain NRC publications for equipment qualification have been given different interpretations and have not had the legal force of an agency regulation. This amendment codifies the environmental qualification methods and criteria that meet the Commission's requirements in this area.

EFFECTIVE DATE: February 22, 1983.

FOR FURTHER INFORMATION CONTACT: Satish K. Aggarwal, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 443-5946.

SUPPLEMENTARY INFORMATION:

Previous Notice

On January 20, 1982, NRC published in the Federal Register a notice of proposed rulemaking on environmental qualification of electric equipment for nuclear power plants (47 FR 2876). The comment period expired March 22, 1982. A total of 69 comment letters raising 10 major issues were received by April 6, 1982. An additional 10 comment letters were received by April 21, 1982, but no new issues were raised. The major issues are discussed below.

Nature and Scope of the Rulemaking

Nuclear power plant equipment important to safety must be able to perform its safety functions throughout its installed life. This requirement is embodied in General Design Criteria 1, 2, 4, and 23 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; in Criterion III, "Design Control," and Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50; and in paragraph 50.55a(h) of 10 CFR Part 50, which incorporates by reference IEEE 279-1971,[1] "Criteria for Protection Systems for Nuclear Power Generating Stations." This requirement is applicable to equipment located inside as well as outside the containment.

The NRC has used a variety of methods to ensure that these general requirements are met for electric equipment important to safety. Prior to 1971, qualification was based on the fact that the electric components were of high industrial quality. For nuclear plants licensed to operate after 1971, qualification was judged on the basis of IEEE 323-1971. For plants whose Safety Evaluation Reports for construction permits were issued since July 1, 1974, the Commission has used Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Light-Water-Cooled Nuclear Power Plants," which endorses IEEE 323-1974,[2] "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," subject to supplementary provisions.

Currently, the Commission has under way a program to reevaluate the qualification of electric equipment in all operating nuclear power plants. As a part of this program, more definitive criteria for environmental qualification of electric equipment important to safety have been developed by the NRC. A document entitled "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines) was issued in November 1979. In addition, the NRC has issued NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which contains two sets of criteria: the first for plants originally reviewed in accordance with IEEE 323-1971 and the second for plants reviewed in accordance with IEEE 323-1974.

By its Memorandum and Order CLI-80-21 dated May 23, 1980, the Commission directed the staff to proceed with a rulemaking on environmental qualification of safety-related equipment and to address the question of backfit. The commission also directed that the DOR Guidelines and NUREG-0588 form the basis for the requirements licensees and applicants must meet until the rulemaking has been completed. This rule is based on the DOR Guidelines and NUREG-0588. The Commission recognizes the qualification efforts of the industry as a result of CLI-80-21. Therefore, the rule provides that requalification of electric equipment will not be required by applicants for and holders of operating licenses for nuclear power plants previously required by NRC to qualify equipment in accordance with DOR Guidelines or NUREG-0588 (Category I or II). Category I requirements of NUREG-0588, which supplement the recommendations of and apply to equipment qualified in accordance with IEEE 323-1974, apply to nuclear power plants for which the construction permit safety evaluation report was issued after July 1, 1974. Category II requirements, which supplement the recommendations of and apply to equipment qualified in accordance with IEEE 323-1971, apply to nuclear power plants for which the construction permit safety evaluation report was issued prior to July 1, 1974.

In CLI-80-21, the Commission stated that unless there were sound reasons to the contrary, replacement parts should be qualified to the standards set forth in Category I of NUREG-0588 or IEEE 323-1974. The Commission reaffirms that position in this rulemaking. Such qualification constitutes compliance with the provisions of paragraph 50.49(1). The Commission's position is designed to promote the policy of upgrading the environmental qualification and reliability of installed electric equipment. Situations may arise, however, in which such upgrading will not be feasible or compatible with overall plant safety. Licensee must review each situation on a case-by-case basis to determine that "sound reasons to the contrary" do exist to justify an exception from upgrading. Examples of acceptable "sound reasons to the contrary" will be included in Regulatory Guide 1.89.

The dates specified in this rule for completion of environmental qualification of electric equipment important to safety apply to all licensees and applicants and supersede any date previously imposed. No changes to licenses or technical specifications are necessary to reflect these new completion dates.

The scope of the final rule covers that portion of equipment important to safety commonly referred to as "safety-related" (which the Commission interprets as essentially "Class 1E" equipment defined in IEEE 323-1974), and nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of required safety functions by safety-related equipment. Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the

[1] Incorporation by reference approved by the Director of the Office of Federal Register on January 1, 1982. Copies may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, N.Y. 10017.

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

consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences; design basis accidents; external events; and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) above. Also covered in the scope of the final rule is certain postaccident monitoring equipment specified as "Category 1 and 2," in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

Included in the final rule are specific technical requirements pertaining to (a) qualification parameters, (b) qualification methods, and (c) documentation. Qualification parameters include temperature, pressure, humidity, radiation, chemicals, and submergence. Qualification methods include (a) testing as the principal means of qualification and (b) analysis in combination with partial type test data or operating experience. The final rule requires that the qualification program include synergistic effects, radiation, environmental conditions and margin considerations. Also, a record of qualification must be maintained. Proposed Revision 1 to Regulatory Guide 1.89, which has been issued for public comment, describes methods acceptable to the NRC for meeting the provisions of this rule and includes a list of typical equipment covered by it. Revision 1 to Regulatory Guide 1.89 will be issued after resolution of public comments.

NRC will generally not accept analysis alone in lieu of testing. Experience has shown that qualification of equipment without test data may not be adequate to demonstrate functional operability during design basis event conditions. Paragraph 50.49(f) provides four methods for qualification. Testing will be preferred. To ensure integrity of a testing program, the Commission expects that the same piece of equipment will be used throughout the complete test sequence.

The final rule requires that each holder of an operating license provide a list of electric equipment important to safety within the scope of this rule previously qualified based on testing, analysis, or a combination thereof, and a list of equipment that has not been qualified. These lists and the schedule for completion of qualification of electric equipment must be submitted by May 20, 1983.

The general requirements for seismic and dynamic qualification for electric equipment are contained in the General Design Criteria and are not included within the scope of this rule. Further guidance is provided in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants," (Revision 1) and NUREG-0800, "Standard Review Plan." NRC is considering future rulemaking concerning requirements for the environmental qualification of electric equipment important to safety and the requirements for seismic and dynamic qualification of electric equipment.

Comments On The Proposed Rule

The Commission received and considered the comments on the proposed rule contained in the 69 letters received from the public by April 6, 1982. Copies of those letters and a staff response to each comment are available for public inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

The major issues raised by the comments and NRC staff responses are as follows:

(1) Seismic and Dynamic Qualification—Paragraph 50.49(c)

Issue: Seismic and dynamic qualifications are an integral part of environmental qualification. It is therefore inappropriate to codify these requirements separately.

Response: Electric equipment at operating nuclear power plants was generally qualified for environmental and seismic stresses separately, i.e., by using separate prototypes for environmental and seismic qualification tests. The Commission has decided, after considerable deliberation, to pursue the issue of seismic and dynamic qualification separately at a future date. A future seismic rule may not require retesting for environmental stresses because a single prototype was not used during the original qualification. Also, the Commission has concluded that protection of electric equipment important to safety against other natural phenomena and external events should not be within the scope of this rule.

(2) Scope—Cold Shutdown Requirement—Paragraph 50.49(b)

Issue: The rule introduces a new requirement to qualify "equipment needed to complete one path of achieving and maintaining a cold shutdown condition." A change of this magnitude, at this advanced stage of the industry's qualification effort, most certainly introduces significant new costs and obligations with no demonstrated improvement in safety.

Response: Regulatory requirements in effect at the time of licensing of the majority of operating reactors did not require that all electric equipment and systems necessary to bring the reactor to cold shutdown be classified as safety related. However, electric equipment and systems necessary to shut down the reactor and maintain it in a safe shutdown condition are required to be classified as safety related and therefore are covered by the rule.

The Commission is currently studying the requirements for shutdown decay heat removal under Unresolved Safety Issue (USI) A-45. The overall purpose of A-45 is to evaluate the adequacy of current licensing requirements to ensure that failure to remove shutdown decay heat does not pose an unacceptable risk. Under A-45 a comprehensive and consistent set of shutdown cooling requirements for existing and future plants is being developed. The final technical resolution of A-45 is presently scheduled for October 1984.

The Commission believes it would be premature at this time to impose the requirement to environmentally qualify electric equipment and systems necessary to achieve and maintain cold shutdown prior to the final resolution of A-45. Therefore, this requirement is not included in the final rule.

(3) Scope—Equipment in a Mild Environment—Paragraph 50.49(b)

Issue: The rule makes no distinction between equipment located in a harsh or mild environment. The stresses for equipment in a mild environment are less severe than for those in a harsh environment.

Response: The final rule does not cover the electric equipment located in a mild environment. The Commission has concluded that the general quality and surveillance requirements applicable to electric equipment as a result of other Commission regulations, including 10 CFR Part 50, Appendix B (see for example, Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 3) are sufficient to ensure adequate performance of electric equipment important to safety located in mild environments. Since it has been concluded that no further environmental qualification requirements are needed for such equipment provided they fully satisfy all other applicable regulations, the Commission has determined that no additional requirements are necessary

with respect to electric equipment important to safety located in mild environments in order for licensees to satisfy, with respect to such equipment, existing license conditions or technical specifications calling for qualification of safety-related electric equipment in accordance with DOR Guidelines or NUREG-0588.

(4) Scope—Previous Qualification Efforts—Paragraph 50.49(b)

Issue: The rule does not recognize that plants have completed qualification of equipment to the DOR Guidelines or NUREG-0588. Without such recognition, industry efforts, manpower, and billions of dollars will go down the drain.

Response: The final rule has been expanded to alleviate this concern. See Paragraph 50.49(k).

(5) Humidity—Paragraph 50.49(e)(2)

Issue: The effects of time-dependent variations of relative humidity during normal operation cannot be considered for all equipment. There are no detailed standards for how this type of testing should be performed.

Response: The Commission agrees. Humidity variation during normal operation are difficult to predict. It has not been demonstrated that the time-dependent variation in humidity will produce any differences in degradation of electric equipment. The words "Time-dependent variation of relative" have been deleted from Paragraph 50.49(e)(2).

(6) Aging—Paragraph 50.49(e)(5)

Issue: The requirement that ongoing qualifications be done using "prototype equipment natually aged" is overly restrictive. Use of accelerated aging to define a qualified life is not technically feasible.

Response: Preconditioning by accelerated aging is technically feasible for simple electric equipment for plant life and for complex electric equipment for a shorter designated life. The Commission recognizes that state-of-art technology will be utilized in any aging program. Reference to qualified life has been deleted from paragraph 50.49(e)(5).

(7) Margins—Paragraph 50.49(e)(8)

Issue: The margins applied in addition to known conservatisms lead to excessive stress that could lead to failures of equipment in unrealistic qualification tests.

Response: The Commission agrees. This requirement could have caused excessive margins. The paragraph has been modified to recognize conservatisms that can be qualified.

(8) Analysis and partial test data—Paragraph 50.49(j)(4)

Issue: If partial type test data that adequately support the analytical assumptions and conclusions are available, their analysis should be allowed to extrapolate or interpolate these results for equipment, regardless of purchase date.

Response: The Commission agrees. Reference to "purchase date" has been deleted.

(9) Requirement for a central file—Paragraph 50.49(j)

Issue: The requirement for a central file should be deleted since it is not cost effective and has no safety benefit.

Response: The Commission agrees. This requirement has been subject to different interpretations. A record of qualification must be maintained in an "auditable form" but not necessarily in a central file for the entire period during which the covered item is installed in a nuclear power plant. Recordkeeping requirement of 10 CFR Part Appendix B must be met. Certain records can be kept at the vendor's shop.

(10) Justification of continued operation for operating plants.

Issue: The requirement to submit justification for the continued operation of operating plants should be deleted since this information has been previously submitted to NRC.

Response: This requirement has been satisfactorily met and Paragraph 50.49(j) of the proposed rule has been deleted in its entirety from the final rule.

In addition, Paragraph 50.49(g) of the proposed rule has been deleted from the final rule since it is too prescriptive. It will be included in Regulatory Guide 1.89.

Effective Date: This rule replaces the "interim rule" published in the Federal Register on June 30, 1982 (47 FR 28363). The "interim rule" suspended environmental qualification deadlines contained in license conditions or technical specifications of operating plants. On the effective date of this rule (see above), the "interim rule" is superseded and the schedule for environmental qualification contained in this rule takes effect for all plants.

Paperwork Reduction Act

The final rule contains information collection requirements that have been approved by the Office of Management and Budget: OMB approval number is 3150-0011.

Regulatory Flexibility Statement

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not have a significant economic impact on a substantial number of small entities. This final rule affects the method of qualification of electric equipment by utilities. Utilities do not fall within the definition of a small business found in Section 3 of the Small Business Act, 15 U.S.C. 632.

In addition, utilities are required by the Commission's Memorandum and Order CLI-80-21, dated May 23, 1980, to meet the requirements contained in the DOR "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," (November 1979) and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which form the basis of this rule. Consequently, this rule codifies existing requirements and imposes no new costs or obligations on utilities.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, 10 CFR Part 50 is amended as follows:

PART 50—[AMENDED]

1. The authority citation for Part 50 continues to read as follows:

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Sections 50.100-50.102 also issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

For the purposes of sec. 223, 68 Stat 958, as amended (42 U.S.C. 2273), §§ 50.10 (a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10 (b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. Section 50.49 is revised to read as follows:

§ 50.49 Environmental qualification of electric equipment important to safety for nuclear power plants.

(a) Each holder of or each applicant for a license to operate a nuclear power plant shall establish a program for qualifying the electric equipment defined in paragraph (b) of this section.

(b) Electric equipment important to safety covered by this section is:

(1) Safety-related electric equipment.[3] This equipment is that relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) of this paragraph.

(2) Nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (i) through (iii) of paragraph (b)(1) of this section by the safety-related equipment.

(3) Certain post-accident monitoring equipment.[4]

(c) Requirements for (i) dynamic and seismic qualification of electric equipment important to safety, (ii) protection of electric equipment important to safety against other natural phenomena and external events, and (iii) environmental qualification of electric equipment important to safety located in a mild environment are not included within the scope of this section. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences.

[3] Safety-related electric equipment is referred to as "Class 1E" equipment in IEEE 323-1974. Copies of this standard may be obtained from the Institute of Electrical and Electronics Engineers, Inc., 345 East 47th Street, New York, NY 10017.

[4] Specific guidance concerning the types of variables to be monitored is provided in Revision 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Copies of the Regulatory Guide can be obtained from Nuclear Regulatory Commission, Document Management Branch, Washington, DC 20555.

(d) The applicant or licensee shall prepare a list of electric equipment important to safety covered by this section. In addition, the applicant or licensee shall include the following information for this electric equipment important to safety in a qualification file:

(1) The performance specifications under conditions existing during and following design basis accidents.

(2) The voltage, frequency, load, and other electrical characteristics for which the performance specified in accordance with paragraph (d)(1) of this section can be ensured.

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

(e) The electric equipment qualification program must include and be based on the following:

(1) *Temperature and Pressure.* The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

(2) *Humidity.* Humidity during design basis accidents must be considered.

(3) *Chemical Effects.* The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation (e.g., containment spray, emergency core cooling, or recirculation from containment sump). If the composition of the chemical spray can be affected by equipment malfunctions, the most severe chemical spray environment that results from a single failure in the spray system must be assumed.

(4) *Radiation.* The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) *Aging.* Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment may be preconditioned to a shorter designated life. The equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life.

(6) *Submergence* (if subject to being submerged).

(7) *Synergistic Effects.* Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

(8) *Margins.* Margins must be applied to account for unquantified uncertainty, such as the effects of production variations and inaccuracies in test instruments. These margins are in addition to any conservatisms applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins.

(f) Each item of electric equipment important to safety must be qualified by one of the following methods:

(1) Testing an identical item of equipment under identical conditions or under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(2) Testing a similar item of equipment with a supporting analysis to show that the equipment to be qualified is acceptable.

(3) Experience with identical or similar equipment under similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable.

(4) Analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

(g) Each holder of an operating license issued prior to February 22, 1983, shall, by May 20, 1983, identify the electric equipment important to safety within the scope of this section already qualified and submit a schedule for either the qualification to the provisions of this section or for the replacement of the remaining electric equipment important to safety within the scope of this section. This schedule must establish a goal of final environmental qualification of the electric equipment within the scope of this section by the end of the second refueling outage after March 31, 1982 or by March 31, 1985, whichever is earlier. The Director of the Office of Nuclear Reactor Regulatory may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if these requests are filed on

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a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification.

(h) Each licensee shall notify the Commission of any significant equipment qualification problem that may require extension of the completion date provided in accordance with paragraph (g) of this section within 60 days of its discovery.

(i) Applicants for operating licenses that are to be granted on or after February 22, 1983, but prior to November 30, 1985, shall perform an analysis to ensure that the plant can be safely operated pending completion of equipment qualification required by this section. This analysis must be submitted to the Director of the Office of Nuclear Reactor Regulation for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

(1) Accomplishing the safety function by some designated alternative equipment if the principal equipment has not been demonstrated to be fully qualified.

(2) The validity of partial test data in support of the original qualification.

(3) Limited use of administrative controls over equipment that has not been demonstrated to be fully qualified.

(4) Completion of the safety function prior to exposure to the accident environment resulting from a design basis event and ensuring that the subsequent failure of the equipment does not degrade any safety function or mislead the operator.

(5) No significant degradation of any safety function or misleading information to the operator as a result of failure of equipment under the accident environment resulting from a design basis event.

(j) A record of the qualification, including documentation in paragraph (d) of this section, must be maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of electric equipment important to safety covered by this section—

(1) Is qualified for its application; and

(2) Meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life.

(k) Applicants for and holders of operating licenses are not required to requalify electric equipment important to safety in accordance with the provisions of this section if the Commission has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979 (DOR Guidelines), or NUREG-0588 (For Comment version), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

(l) Replacement equipment must be qualified in accordance with the provisions of this section unless there are sound reasons to the contrary.

Dated at Washington, D.C. this 17th day of January, 1983.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 83-1722 Filed 1-20-83; 8:45 am]

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COMMODITY FUTURES TRADING COMMISSION

17 CFR Parts 140 and 145

Commission Headquarters Office and Western and Southwestern Regional Offices; Change of Address

AGENCY: Commodity Futures Trading Commission.

ACTION: Final rule amendments.

SUMMARY: The Commodity Futures Trading Commission is amending its regulations in an attempt to clarify that both the physical location and the mailing address of the Commission's headquarters office are one and the same for all practical purposes. In addition, the Commission is amending its regulations to include new addresses for its recently relocated Western and Southwestern regional offices. The Western Regional office has been moved from San Francisco to Los Angeles, California. The Southwestern Regional office, located in Kansas City, Missouri, has moved to a different suite of offices in the same building.

EFFECTIVE DATE: January 21, 1983.

FOR FURTHER INFORMATION CONTACT: Donald L. Tendick, Acting Executive Director, Commodity Futures Trading Commission, 2033 K Street NW., Washington, D.C. 20581, (202) 254-7556.

SUPPLEMENTARY INFORMATION: Commission regulation § 140.1 currently provides a separate physical location and mailing address for the Commission's headquarters office. The Commission is amending regulation § 140.1 to clarify that there is no meaningful distinction between its physical location and mailing address. The sole address of the Commission's headquarters office as of January 18, 1983 will be 2033 K Street, N.W., Washington, D.C. 20581.

The Commission is amending regulation 140.2 to reflect the fact that the Western Regional office of the Commission has moved from San Francisco to 10850 Wilshire Boulevard, Suite 510, Los Angeles, California 90024. The telephone number for general information is (213) 209-6783. In addition, regulation § 140.2 is being amended to note the Southwestern Regional office has moved from Room 208 to Suite 400 at 4901 Main Street, Kansas City, Missouri 64112. The telephone number for general information remains (816) 374-5425.

Certain other provisions of the Commission's regulations contain references to or addresses of the Commission's Western and Southwestern Regional offices. The appropriate changes have been made to reflect the new addresses in each of those provisions.

Based on the foregoing, pursuant to its authority contained in section 2(a)(11) of the Commodity Exchange Act, 7 U.S.C. 4a(j) (1976), the Commission hereby amends Parts 140 and 145 of the Code of Federal Regulations as follows:

PART 140—[AMENDED]

1. Section 140.1 is revised to read as follows:

§ 140.1 Headquarters Office.

(a) *General.* The headquarters office of the Commission is located at 2033 K Street, NW., Washington, D.C. 20581.

2. Section 140.2 is amended by revising paragraphs (c) and (d) to read as follows:

§ 140.2 Region offices—Regional directors.

* * * * *

(c) The Western Regional office is located at 10850 Wilshire Boulevard, Suite 510, Los Angeles, California 90024, and is responsible for enforcement of the Act and administration of the programs of the Commission in the States of Alaska, Arizona, California, Hawaii, Idaho, Montana, Nevada, Oregon, Utah, Washington, and Wyoming.

(d) The Southwestern Regional office is located at 4901 Main Street, Suite 400, Kansas City, Missouri 64112, and is responsible for enforcement of the Act