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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

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In the matter of Alabama Power Company

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Other ____ Witness ____

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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 [illegible] 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[3] required to permit the control room operators to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events.

1.2 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;

(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

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[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 10^3 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.

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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

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TABLE 1 (Continued)

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

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

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

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

TABLE 1 (Continued)

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

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

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

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

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

TABLE 2

PWR VARIABLES

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

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

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

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

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

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

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

TABLE 2 (Continued)

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

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

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

TABLE 2 (Continued)

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

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

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

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

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

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

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

TABLE 2 (Continued)

Variable	Range	Category (see Regulatory Position 1.3)	Purpose
TYPE C (Continued)			
Containment			
RCS Pressure[1]	0 to 3000 psig (4000 psig for CE plants)	1^2	Detection of potential for breach; accomplishment of mitigation
Containment Hydrogen Concentration	0 to 10% (capable of operating from 10 psia to maximum design pressure[4]) 0 to 30% for ice-condenser-type containment	1	Detection of potential for breach; accomplishment of mitigation; long-term surveillance
Containment Pressure[1]	10 psia pressure to 3 times design pressure[4] for concrete; 4 times design pressure for steel (5 psia for subatmospheric containments)	1	Detection of potential for or actual breach; accomplishment of mitigation
Containment Effluent Radioactivity - Noble Gases from Identified Release Points[4]	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

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[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^{-} 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 Multipurpose Vent Discharging Any of Above Releases (if containment purge is included)	10^{-6} μCi/cc to 10^{3} μCi/cc 0 to 110% vent design flow[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 then 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.

A-19
50-348/364-CIVP
2/14/92

APCo Exhibit 19

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

DOCKETED
USNRC

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#2

February 4, 1983

Docket No. 50-364

Attachment To Be
Withheld From Public Disclosure

Bollard
8302140200
LC# 1703
KWRD 4/7/87
4/16/87

NRC- ICENSING
RPM FILES

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

SUBJECT: SAFETY EVALUATION REPORT FOR ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

New R: 3331

RE: Joseph M. Farley Unit No. 2

This letter transmits the Safety Evaluation Report for the Environmental Qualification of Safety-Related Electrical Equipment at your facility. This evaluation is based on your response to our previous Safety Evaluation Report, dated March 1981 (Supplement No. 6 to NUREG-75/34, Appendix B) and subsequent submittal(s) dated July 1, 1981 and March 2, April 23 and June 25, 1982. This Safety Evaluation Report presents the results of the Environmental Qualification Review for safety-related electrical equipment, exposed to a harsh environment, in accordance with NRC requirements. We request that you provide your plans for qualification or replacement of the unqualified equipment and the schedule for accomplishing your proposed corrective actions to us within ninety (90) days of the receipt of this letter.

As indicated in the conclusion section of the Safety Evaluation Report, we request that you reaffirm the justification for continued operation and within thirty days (30) days of receipt of this letter, submit information for items in NRC categories I.B, II.A and II.B (presented in the enclosed Technical Evaluation Report) for which justification for continued operation was not previously submitted to the NRC. We suggest that the clarification set forth in item 3 of Generic Letter No. 82-09, "Clarification Questions and Answers on Environmental Qualification Requirements," should be considered in your justification for continued operation.

The Technical Evaluation Report contains Proprietary Information from manufacturers' proprietary test reports and should be withheld from public disclosure. We request that you inform us as indicated in the proprietary section of the Safety Evaluation Report whether any portions of the identified pages require proprietary protection.

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FEB 1983
V.P.-NUCLEAR

Attachment To Be
Withheld From Public Disclosure

NUCLEAR REGULATORY COMMISSION

Docket No. ________ Official Exh. No. 19

In the matter of ALABAMA POWER CO.

Staff ________ IDENTIFIED 2/11/92

Applicant APCO RECEIVED 2/19/92

Intervenor ________ REJECTED ________

Cont'g Off'r ________

Contractor ________ DATE 2-4-83

Other ________ Witness ________

Reporter L Eslop

At your option, the staff will be available to discuss the findings in the Safety Evaluation Report as augmented by the Technical Evaluation Report. Questions regarding this letter should be directed through the NRC Project Manager for your plant.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Safety Evaluation Report
2. Technical Evaluation Report

cc w/o TER:
See next page

0053948

Mr. F. L. Clayton, Jr.
Alabama Power Company

cc: Mr. W. O. Whitt
Executive Vice President
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Birmingham, Alabama 35291

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0053949

SAFETY EVALUATION REPORT BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
FOR ALABAMA POWER COMPANY
FARLEY 2
DOCKET NO. 50-364

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC EQUIPMENT

INTRODUCTION

General Design Criteria 1 and 4 specify that safety-related electrical equipment in nuclear facilities must be capable of perfo[illegible] its safety-related function under environmental conditions associated [illegible]th all normal, abnormal, and accident plant operation. In order to ensure compliance with the criteria, the NRC staff required all licensees of operating reactors to submit a re-evaluation of the qualification of safety-related electrical equipment which may be exposed to a harsh environment.

BACKGROUND

On February 8, 1979, the NRC Office of Inspection and Enforcement (IE) issued to all licensees of operating plants (except those included in the systematic evaluation program (SEP)) IE Bulletin (IEB) 79-01, "Environmental Qualification of Class IE Equipment." This Bulletin, together with IE Circular 78-08 (issued on May 31, 1978), required the licensees to perform reviews to assess the adequacy of their environmental qualification programs.

On January 14, 1980, NRC issued IE Bulletin 79-01B which included the DOR guidelines and NUREG-0588 as attachments 4 and 5, respectively. Subsequently, on May 23, 1980, Commission Memorandum and Order CLI-80-21 was issued and stated the DOR guidelines and portions of NUREG-0588 form

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the requirements that licensees must meet regarding environmental qualification of safety-related electrical equipment in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. Supplements to IEB 79-01B were issued for further clarification and definition of the staff's needs. These supplements were issued on February 29, September 30, and October 24, 1980.

In addition, the staff issued orders dated August 29, 1980 (amended in September 1980) and October 24, 1980 to all licensees. The August orde required that the licensees provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment. The October order required the establishment of a central file location for the maintenance of all equipment qualification records. The central file was mandated to be established by December 1, 1980. The staff subsequently issued Safety Evaluation Reports (SERs) on enviromental qualification of safety-related electrical equipment to licensees of all operating plants in mid-1981. These SERs directed licensees to "either provide documentation of the missing qualification information which demonstrates that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action (re-qualification, replacement (etc.))." Licensees were required to respond to NRC within 90 days of receipt of the SER. In response to the staff SER issued 1981, the licensee submitted additional information regarding the qualification of safety-related electrical equipment.

EVALUATION

The acceptability of the licensee's equipment environmental qualification program was resolved for the Division of Engineering by the Franklin Research Center (FRC) as part of the NRC Technical Assistance Program in support of NRC operating reactor licensing actions. The consultant's review is documented in the report "Review of Licensees' Resolutions of Outstanding Issues from NRC Equipment Environmental Qualification Safety Evaluation Reports," which is attached.

We have reviewed the evaluation performed by our consultant contained in the enclosed Technical Evaluation Report (TER) and concur with its bases and findings.

The staff has also reviewed the licensee's justification for continued operation regarding each item of safety-related electrical equipment identified by the licensee as not being capable of meeting environmental qualification requirements for the service conditions intended.

CONCLUSIONS

Based on the staff's review of the enclosed Technical Evaluation Report and the licensee's justification for continued operation, the following conclusions are made regarding the qualification of safety-related electrical equipment.

Continued operation until completion of the licensee's environmental qualification program has been determined to not present undue risk to the public health and safety. Furthermore, the staff is continuing to review the licensee's environmental qualification program. If any additional qualification deficiencies were identified during the course of this review, the licensee would be required to reverify the justification for continued operation. The staff will review this information to ensure that continued operation until completion of the licensee's environmental qualification program will not present undue risk to the public health and safety. In this regard, it is requested that the licensee do the following:

- o Resolve any deficiencies identified in Appendix D of the FRC TER regarding justification for continued operation. If as a result of resolving these deficiencies, the previous justification for continued operation is changed, provide within thirty (30) days of receipt of this SER the new justification for continued operation regarding each affected item.

The major qualification deficiencies that have been identified in the enclosed FRC TER (Tables 4-1, 4-2, 4-3 and 4-4) must be resolved by the licensee. Items requiring special attention by the licensee are summarized below:

- o Submission of information within thirty (30) days for items in NRC categories 1B, 2A and 2B for which justification for continued operation was not previously submitted to NRC or FRC,

- o Resolution of deficiencies associated with Equipment Items 18 and 20 that have been assigned to NRC Category II.B (Equipment Not Qualified).
- o Resolution of the deficiencies identified in Section 4.3.2 of the FRC TER regarding the containment spray system.

The licensee must provide the plans for qualification or replacement of the unqualified equipment and the schedule for accomplishing its proposed correction action in accordance with 10 CFR 50.49.

PROPRIETARY REVIEW

Enclosed in the FRC Technical Evaluation Report (TER) are certain identified pages on which the information is claimed to be proprietary.

During the preparation of the enclosed TER, FRC used test reports and other documents supplied by the licensee that included material claimed to be proprietary by their owners and originators. NRC is now preparing to publicly release the FRC TER and it is incumbent on the agency to seek review of all claimed proprietary material. As such, the licensee is requested to review the enclosed TER with their owner or originator and notify NRR within seven (7) days of receipt of this SER whether any portions of the identified pages still require proprietary protection. If so, the licensee must clearly identify this information and the specific rationale and justification for the protection from public disclosure, detailed in a written response within twenty (20) days of receipt of this SER. The level of specificity necessary for such continued protection should be consistent with the criteria enumerated in 10 CFR 2.790(b) of the Commission's regulations.

A-18
50-348/364-CivP
2/19/92

#1

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

APCo Exhibit 18

DOCKETED
USNRC

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February 4, 1983

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Docket No. 50-348

Attachment To Be
Withheld From Public Disclosure

Ballard
8302140192
LC# 1702
KWRD 4/13/87
4/16/87

Mr. F. L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

NRC-LICENSING
RPM FILES

Dear Mr. Clayton:

SUBJECT: SAFETY EVALUATION REPORT FOR ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

New R: 3332
LC# 1703

RE: Joseph M. Farley Unit No. 1

This letter transmits the Safety Evaluation Report for the Environmental Qualification of Safety-Related Electrical Equipment at your facility. This evaluation is based on your response to our previous Safety Evaluation Report, dated May 21, 1981 and subsequent submittal(s) dated August 25, 1981 and February 19, March 8, April 23, June 23 and 25, 1982. This Safety Evaluation Report presents the results of the Environmental Qualification Review for safety-related electrical equipment, exposed to a harsh environment, in accordance with NRC requirements. We request that you provide your plans for qualification or replacement of the unqualified equipment and the schedule for accomplishing your proposed corrective actions to us within ninety (90) days of the receipt of this letter.

As indicated in the conclusion section of the Safety Evaluation Report, we request that you reaffirm the justification for continued operation and within thirty (30) days of receipt of this letter, submit information for items in NRC categories I.B, II.A and II.B (presented in the enclosed Technical Evaluation Report) for which justification for continued operation was not previously submitted to the NRC. We suggest that the clarification set forth in item 8 of Generic Letter No. 82-09, "Clarification Questions and Answers on Environmental Qualification Requirements," should be considered in your justification for continued operation.

The Technical Evaluation Report contains Proprietary Information from manufacturers' proprietary test reports and should be withheld from public disclosure. We request that you inform us as indicated in the proprietary section of the Safety Evaluation Report whether any portions of the identified pages require proprietary protection.

FEB 1983
V.P.-NUCLEAR

Attachment To Be
Withheld From Public Disclosure

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At your option, the staff will be available to discuss the findings in the Safety Evaluation Report as augmented by the Technical Evaluation Report. Questions regarding this letter should be directed through the NRC Project Manager for your plant.

Sincerely,

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Safety Evaluation Report
2. Technical Evaluation Report

cc w/o TER:
See next page

0053940

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Alabama Power Company

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Atlanta, Georgia 30303

0053941

SAFETY EVALUATION REPORT BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
FOR ALABAMA POWER COMPANY
FARLEY 1
DOCKET NO. 50-348

ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC EQUIPMENT

INTRODUCTION

General Design Criteria 1 and 4 specify that safety-related electrical equipment in nuclear facilities must be capable of performing its safety-related function under environmental conditions associated with all normal, abnormal, and accident plant operation. In order to ensure compliance with the criteria, the NRC staff required all licensees of operating reactors to submit a re-evaluation of the qualification of safety related electrical equipment which may be exposed to a harsh environment.

BACKGROUND

On February 8, 1979, the NRC Office of Inspection and Enforcement (IE) issued to all licensees of operating plants (except those included in the systematic evaluation program (SEP)) IE Bulletin (IEB) 79-01, "Environmental Qualification of Class IE Equipment." This Bulletin, together with IE Circular 78-08 (issued on May 31, 1978), required the licensee to perform reviews to assess the adequacy of their environmental qualification programs.

On January 14, 1980, NRC issued IE Bulletin 79-01B which included the DOR guidelines and NUREG-0588 as attachments 4 and 5, respectively.

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Subsequently, on May 23, 1980, Commission Memorandum and Order CLI-80-21 was issued and stated the DOR guidelines and portions of NUREG-0588 form the requirements that licensees must meet regarding environmental qualification of safety-related electrical equipment in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. Supplements to IEB 79-01B were issued for further clarification and definition of the staff's needs. These supplements were issued on February 29, September 30, and October 24, 1980.

In addition, the staff issued orders dated August 29, 1980 (amended in September 1980) and October 24, 1980 to all licensees. The August order required that the licensees provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment. The October order required the establishment of a central file location for the maintenance of all equipment qualification records. The central file was mandated to be established by December 1, 1980. The staff subsequently issued Safety Evaluation Reports (SERs) on enviromental qualification of safety-related electrical equipment to licensees of all operating plants in mid-1981. These SERs directed licensees to "either provide documentation of the missing qualification information which demonstrates that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action (re-qualification, replacement (etc.))." Licensees were required to respond to NRC within 90 days of receipt of the SER. In response to the staff SER issued May 21, 1981, the licensee submitted additional information regarding the qualification of safety-related electrical equipment.

EVALUATION

The acceptability of the licensee's equipment environmental qualification program was resolved for the Division of Engineering by the Franklin Research Center (FRC) as part of the NRR Technical Assistance Program in support of NRC operating reactor licensing actions. The consultant's review is documented in the report "Review of Licensees' Resolutions of Outstanding Issues from NRC Equipment Environmental Qualification Safety Evaluation Reports," which is attached.

We have reviewed the evaluation performed by our consultant contained in the enclosed Technical Evaluation Report (TER) and concur with its bases and findings.

The staff has also reviewed the licensee's justification for continued operation regarding each item of safety-related electrical equipment identified by the licensee as not being capable of meeting environmental qualification requirements for the service conditions intended.

CONCLUSIONS

Based on the staff's review of the enclosed Technical Evaluation Report and the licensee's justification for continued operation, the following conclusions are made regarding the qualification of safety-related electrical equipment

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Contined operation until completion of the licensee's environmental qualification program has been determined to not present undue risk to the public health and safety. Furthermore, the staff is continuing to review the licensee's environmental qualification program. If any additional qualification deficiencies were identified during the course of this review, the licensee would be required to reverify the justification for continued operation. The staff will review this information to ensure that continued operation until completion of the licensee's environmental qualification program will not present undue risk to the public health and safety. In this regard, it is requested that the licensee do the following:

- o Resolve any deficiencies identified in Appendix D of the FRC TER regarding justification for continued operation. If as a result of resolving these deficiencies, the previous justification for continued operation is changed provide within thirty (30) days of receipt of this SER the new justification for continued operation regarding each affected item.

The major qualification deficiencies that have been identified in the enclosed FRC TER (Tables 4-1, 4-2, 4-3 and 4-4) must be resolved by the licensee. Items requiring special attention by the licensee are summarized below:

- o Submission of information within thirty (30) days for items in NRC categories 1B, 2A and 2B for which justification for continued operation was not previously submitted to NRC or FRC,

0053945

- Resolution of the deficiencies associated with Equipment Items 23 & 25, that have been assigned to NRC Category II.B (Equipment Not Qualified).

The licensee must provide the plans for qualification or replacement of the unqualified equipment and the schedule for accomplishing its proposed correction action in accordance with 10 CFR 50.49.

<u>PROPRIETARY</u> <u>REVIEW</u>

Enclosed in the FRC Technical Evaluation Report (TER) are certain identified pages on which the information is claimed to be <u>proprietary</u>.

During the preparation of the enclosed TER, FRC used test reports and other documents supplied by the licensee that included material claimed to be proprietary by their owners and originators. NRC is now preparing to publicly release the FRC TER and it is incumbent on the agency to seek review of all claimed proprietary material. As such, the licensee is requested to review the enclosed TER with their owner or originator and notify NRR within seven (7) days of receipt of this SER whether any portions of the identified pages still require proprietary protection. If so, the licensee must clearly identify this information and the specific rationale and justification for the protection from public disclosure, detailed in a written response within twenty (20) days of receipt of this SER. The level of specificity necessary for such continued protection should be consistent with the criteria enumerated in 10 CFR 2.790(b) of the Commission's regulations.

A-35 50-348/364-CIVP APCo Exhibit 35 2/20/92

Revision 1*
June 1984

U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE

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REGULATORY GUIDE 1.89
(Task EE 042-2)

ENVIRONMENTAL QUALIFICATION OF CERTAIN ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

The Commission's regulations in 10 CFR *Part* 50, "Domestic Licensing of Production and Utilization Facilities," require that structures, systems, and components important to safety in a nuclear power plant be designed to accommodate the effects of environmental conditions (i.e., remain functional under postulated accident conditions) and that design control measures such as testing be used to check the adequacy of design. These general requirements are contained in General Design Criteria 1, 2, 4, and 23 of Appendix A, "General Design Criteria for Nuclear Power Plants," to Part 50; in Criterion III, "Design Control," Criterion XI, "Test Control," and Criterion XVII, "Quality Assurance Records," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50; and in § 50.55a.

Specific requirements pertaining to qualification of certain electric equipment important to safety are contained in § 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," of 10 CFR Part 50. Section 50.49 requires that three categories of electric equipment important to safety be qualified for their application and specified performance and provides requirements for establishing environmental qualification methods and qualification parameters. These three categories are (1) safety-related electric equipment (Class 1E), (2) non-safety-related electric equipment (non-Class 1E) whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions by safety-related equipment, and (3) certain postaccident monitoring equipment. This regulatory guide applies only to these three categories of electric equipment important to safety.

Section 50.49 does not include requirements for seismic and dynamic qualification, protection of electric equipment against other natural phenomena and external events, and equipment located in a mild environment.

This regulatory guide describes a method acceptable to the NRC staff for complying with § 50.49 of 10 CFR Part 50 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design basis accident.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position. Any guidance in this document related to information collection activities has been cleared under OMB Clearance No. 3150-0011.

B. DISCUSSION

IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,"[1] published February 28, 1974, was prepared by Subcommittee 2, Equipment Qualification, of the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers (IEEE) and was approved by the IEEE Standards Board on December 13, 1973. The standard describes basic procedures for qualifying Class 1E equipment and interfaces that are to be used in nuclear power plants, including components or equipment of any interface whose failure could adversely affect any Class 1E equipment.

For the purposes of this guide, "qualification" is a verification of design limited to demonstrating that the electric equipment is capable of performing its safety

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

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

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NUCLEAR REGULATORY COMMISSION

Docket No. 50-348/364-CivP Official Exh. No. 35

In the matter of Alabama Power Company

Staff ______ IDENTIFIED 3:40 p.m. 2/20/92

Applicant ✓ RECEIVED 3:41 p.m. 2/20/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______

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Reporter L. Eskep

function under significant environmental stresses resulting from design basis accidents in order to avoid common-cause failures. Paragraph 50.49(e)(5) calls for equipment qualified by test to be preconditioned by natural or artificial (accelerated) aging to its end-of-installed-life condition and further specifies that consideration must be given to all significant types of degradation that can have an effect on the functional capability of the equipment. There are considerable uncertainties regarding the processes and environmental factors that could result in such degradation. Oxygen diffusion, humidity, and accumulation of deposits are examples of such effects. Because of these uncertainties, state-of-the-art preconditioning techniques are not capable of simulating all significant types of degradation, and natural pre-aging is difficult and costly. As the state of the art advances and uncertainties are resolved, preconditioning techniques may become more effective. Experience suggests that consideration should be given, for example, to a combination of (1) preconditioning of test samples employing the Arrhenius theory and (2) surveillance, testing, and maintenance of selected equipment specifically directed toward detecting those degradation processes that, based on experience, are not amenable to preconditioning and that could result in common-cause functional failure of the equipment during design basis accidents.

It is essential that safety-related electric equipment be qualified to demonstrate that it can perform its safety function under the environmental service conditions in which it will be required to function and for the length of time its function is required and that non-safety-related electric equipment covered by paragraph 50.49(b)(2) be able to withstand environmental stresses caused by design basis accidents under which its failure could prevent the satisfactory accomplishment of safety functions by safety-related equipment. This concept applies throughout this guide. The specific environment for which individual electric equipment must be qualified will depend on the installed location and the conditions under which it is required to perform its safety function.

The following are examples of considerations to be taken into account when determining the environment for which the equipment is to be qualified: (1) equipment outside containment would generally see a less severe environment than equipment inside containment; (2) equipment whose location is shielded from a radiation source would generally receive a smaller radiation dose than equipment at the same distance from the source but exposed to its direct radiation; (3) equipment required to initiate protective action would generally be required for a shorter period of time than instrumentation required to follow the course of an accident; and (4) analyses taking into account arrangements of equipment and radiation sources may be necessary to determine whether equipment needed for mitigation of design basis accidents other than loss-of-coolant accidents (LOCA) or high-energy line breaks (HELB) could be exposed to a more severe environment than the LOCA or HELB environments delineated in this guide.

Electric equipment to be qualified in a nuclear radiation environment should be exposed to radiation that simulates the calculated integrated dose (normal and accident) that the equipment must withstand prior to completion of its intended safety function. Regulatory Position C.2.c proposes the use of source terms that are consistent with previous guidance in the original edition of this guide, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,"[2] and the DOR Guidelines, "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors."[3]

Item (8) of Regulatory Position C.2.c addresses qualification of equipment exposed to low-level radiation doses. Numerous studies that have compiled radiation effects data on all classes of organic compounds show that compounds with the least radiation resistance have damage thresholds greater than 10^4 rads and would remain functional with exposures somewhat above the threshold value. Thus, for organic materials, radiation qualification may be readily justified by existing test data or operating experience for radiation exposures below 10^4 rads. However, for electronic components, studies have shown failures in metal oxide semiconductor devices at somewhat lower doses. Therefore, radiation qualification for electronic components may have a lower exposure threshold.

The regulatory positions delineated in this guide reflect the state of the art. Research programs currently in progress are investigating such concerns as the effects of oxygen in a LOCA environment, the validity of sequential versus simultaneous applications of steam and radiation environments, and fission product releases following accidents. The staff recognizes that the results of research programs may lead to revisions of the regulatory positions.

C. REGULATORY POSITION

The procedures described by IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,"[1] are acceptable to the NRC staff for satisfying the Commission's regulations pertaining to the qualification of electric equipment for service in nuclear power plants to ensure that the equipment can perform its safety functions subject to the following:

1. Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear

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

[3] Available for inspection or copying at the U.S Nuclear Regulatory Commission Public Document Room, 1717 H Street NW., Washington, D.C., as Enclosure 4 to IE Bulletin No. 79-01B, January 14, 1980.

Power Plants," of 10 CFR Part 50 requires that safety-related electric equipment (Class 1E) as defined in paragraph 50.49(b)(1) be qualified to perform its intended safety functions. Typical safety-related equipment and systems are listed in Appendix A to this guide. Paragraph 50.49(b)(2) requires that non-safety-related electric equipment be environmentally qualified if its failure under postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by safety-related equipment. Typical examples of non-safety-related electric equipment are included in Appendix B to this guide. Paragraph 50.49(b)(3) requires that certain postaccident monitoring equipment also be environmentally qualified. These are specified as "Categories 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."

2. Paragraph 50.49(d) and Section 6.2 of IEEE Std 323-1974 require equipment specifications to include performance and environmental conditions. For the requirements called for in item (7) of Section 6.2 of IEEE 323-1974 and paragraph 50.49(d)(3), the following should be included:

a. Temperature and Pressure Conditions Inside Containment for LOCA and Main Steam Line Break (MSLB). The following methods are acceptable to the NRC staff for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified:

(1) Methods for calculating mass and energy release rates for LOCAs and MSLBs are referenced in Appendix C to this guide. The calculations should account for the time dependence and spatial distribution of these variables. For example, superheated steam followed by saturated steam may be a limiting condition and should be considered.

(2) For pressurized water reactors (PWRs) with a dry containment, calculate LOCA or MSLB containment environment using CONTEMPT-LT or equivalent industry codes.

(3) For PWRs with an ice condenser containment, calculate LOCA or MSLB containment environment using LOTIC or equivalent industry codes.

(4) For boiling water reactors (BWRs) with a Mark I, II, or III containment, calculate LOCA or MSLB environment using CONTEMPT-LT or equivalent industry codes.

Since the test profiles included in Appendix A to IEEE Std 323-1974 are only representative, they should not be considered an acceptable alternative to using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the applicability of those profiles.

b. Effects of Sprays and Chemicals. The effects of containment spray system operation should be considered. This consideration should include, as appropriate, the effects of demineralized water spray or chemical spray systems.

c. Radiation Conditions Inside and Outside Containment. The radiation environment for qualification of electric equipment should be based on the radiation environment normally expected over the installed life of the equipment plus that associated with the most severe design basis accident during or following which the equipment must remain functional. The accident-related environmental conditions should be assumed to occur at the end of the installed life of the equipment. Methods acceptable to the NRC staff for establishing radiation doses for the qualification of equipment for BWRs and PWRs are provided in Appendix D and the following:

(1) The source term to be used in determining the radiation environment associated with a design basis LOCA should be taken as an instantaneous release to the containment of 100% of the noble gas activity, 50% of the halogen activity, and 1% of the remaining fission product activity. The fission product solids should be assumed to remain in the primary coolant and to be carried by the coolant to the containment sump(s).

(2) For all other design basis accidents (e.g., non-LOCA high-energy line breaks or rod ejection or rod drop accidents), the qualification source term calculations should use the percentage of fuel damage assumed in the plant-specific analysis (provided in the Final Safety Analysis Report (FSAR)). The nuclide inventory of the breached fuel elements should be calculated at the end of core life assuming continuous full-power operation. The inventory of the fuel rod gap should be assumed to be 10% of the total rod activity inventory of iodine and 10% of the total activity inventory of noble gases (except for krypton-85, for which a release of 30% should be assumed). All the gaseous constituents in the gaps of the breached fuel rods should be assumed to be instantaneously released to the primary system. When substantial fuel damage is postulated, 100% of the noble gases, 50% of the halogens, and 1% of the remaining fission product solids in the affected fuel rods should be assumed to be instantaneously released to the primary system.

(3) For a limited number of accident-monitoring instrumentation channels with instrument ranges that extend well beyond the values the selected variables can attain under limiting conditions as specified in Regulatory Guide 1.97, Revision 2, the environmental qualification should be consistent with Regulatory Positions C.1.3.1.a and C.1.3.2.a of Regulatory Guide 1.97, Revision 2.

(4) The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of the containment and auxiliary structures.

(5) Electric equipment that could be exposed to radiation should be environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment should withstand prior to completion of its required safety functions. Such qualification should consider that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification should factor in doses from all potential radiation sources at the equipment location. Plant-specific analysis should be used to justify any reductions in dose or dose rate resulting from component location or shielding. The qualification environment at the equipment location should be established using an analysis similar in nature and scope to that included in Appendix D to this guide and incorporating appropriate factors pertinent to the actual plant design (e.g., reactor type, containment design).

(6) Shielded components need be qualified only to the gamma radiation environment provided it can be demonstrated that the sensitive portions of the component or equipment are not exposed to significant beta radiation dose rates or that the effects of beta radiation, including heating and secondary radiation, have no deleterious effects on component performance. If, after considering the appropriate shielding factors, the total beta radiation dose contribution to the equipment or component is calculated to be less than 10% of the total gamma radiation dose to which the equipment or component has been qualified, the equipment or component is considered qualified for the beta and gamma radiation environment.

(7) Electric equipment located outside containment that is exposed to the radiation from a recirculating fluid should be qualified to withstand the radiation penetrating the containment plus the radiation from the recirculating fluid.

(8) Electric equipment that may be exposed to low-level radiation doses should not generally be considered exempt from radiation qualification testing. Exceptions may be based on qualification by analysis supported by test data or operating experience that verifies that the dose and dose rates will not degrade the operability of the equipment below acceptable values.

d. **Environmental Conditions for Equipment Outside Containment.** Electric equipment that is subjected to the effects of pipe breaks and is required to mitigate the consequences of the breaks or to bring the plant to safe shutdown should be qualified for the expected environmental conditions. The techniques to calculate the environmental conditions should employ a plant-specific model.

3. Section 6.3, "Type Test Procedures," of IEEE Std 323-1974 should be supplemented with the following:

a. Electric equipment that could be submerged should be identified and qualified by testing in a submerged condition to demonstrate operability for the duration required. Analytical extrapolation of results for test periods shorter than the required duration should be justified.

b. Electric equipment located in an area where rapid pressure changes are postulated simultaneously with the most adverse relative humidity should be qualified to demonstrate that the equipment seals and vapor barriers will prevent moisture from penetrating into the equipment to the degree necessary to maintain equipment functionability.

c. The parameters to which electric equipment is being qualified (e.g., temperature, pressure, radiation) by exposure to a simulated environment in a test chamber should be measured sufficiently close to the equipment to ensure that actual test conditions accurately represent the environment characterized by the test.

d. Performance characteristics that demonstrate the operability of equipment should be verified before, after, and periodically during testing throughout its range of required operability. Variables indicative of momentary failure that prevent the equipment from performing its safety function, e.g., momentary opening of a relay contact, should be monitored continuously to ensure that momentary failures (if any) have been accounted for during testing. For long-term testing, however, monitoring during periodic intervals may be used if justified.

e. Chemical spray or demineralized water spray that is representative of service conditions should be incorporated during simulated event testing at pressure and temperature conditions that would occur when the spray systems actuate.

f. Cobalt-60 or cesium-137 would be acceptable gamma radiation sources for environmental qualification.

4. The suggested values in Section 6.3.1.5, "Margin," of IEEE Std 323-1974, except time margins, are acceptable for meeting the requirements of paragraph 50.49(e)(8). Alternatively, quantified margins should be applied to the environmental parameters discussed in Regulatory Position C.2 to ensure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any conservatism applied during the derivation of local environmental conditions of the equipment unless these conservatisms can be quantified and shown to contain appropriate margins. The margins should account for variations in commercial production of the equipment and the inaccuracies in the test equipment.

Some electric equipment may be required by the design to perform its safety function only within the first ten hours of the event. This equipment should remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis unless a time margin of less than one hour can be justified. This justification must include, for each piece of equipment, (1) consideration of a spectrum of breaks, (2) the potential need for the equipment later in an event or during recovery operations, (3) a determination that failure of the equipment after performance of its safety function will not be detrimental to plant safety or mislead the operator, and (4) a determination that the margin applied to the minimum operability time, when combined with the other test margins, will account for the uncertainties associated with the use of analytical techniques in the derivation of environmental parameters, the number of units tested, production tolerances, and test equipment inaccuracies. For all other equipment (e.g., postaccident monitoring, recombiners), the 10% time margin identified in Section 6.3.1.5 of IEEE Std 323-1974 should be used.

5. Section 6.3.3, "Aging," of IEEE Std 323-1974 and paragraph .0.49(e)(5) should be supplemented with the following:

a. If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program. Synergistic effects known at this time are dose rate effects and effects resulting from the different sequence of applying radiation and (elevated) temperature.

b. The expected operating temperature of the equipment under service conditions should be accounted for in thermal aging. The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging within the limitation of state-of-the-art technology. Other aging methods will be evaluated on a case-by-case basis.

c. The aging acceleration rate and activation energies used during qualification testing and the basis upon which the rate and activation energy were established should be defined, justified, and documented.

d. Periodic surveillance and testing programs are acceptable to account for uncertainties regarding age-related degradation that could affect the functional capability of equipment. Results of such programs will be acceptable as ongoing qualification to modify designated life (or qualified life) of equipment and should be incorporated into the maintenance and refurbishment/replacement schedules.

6. Replacement electric equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of § 50.49 unless there are sound reasons to the contrary. The NRC staff considers the following to be sound reasons for the use of replacement equipment previously qualified in accordance with the DOR Guidelines or NUREG-0588 in lieu of upgrading:

a. The item of equipment to be replaced is a component of equipment that is routinely replaced as part of normal equipment maintenance, e.g., gaskets, o-rings, coils; these may be replaced with identical components.

b. The item to be replaced is a component that is part of an item of equipment qualified as an assembly; these may be replaced with identical components.

c. Identical equipment to be used as a replacement was on hand as a part of the utility's stock prior to February 22, 1983.

d. Replacement equipment qualified in accordance with the provisions of § 50.49 does not exist.

e. Replacement equipment qualified in accordance with the provisions of § 50.49 is not available to meet installation and operation schedules. However, in such case, the replacement equipment may be used only until upgraded equipment can be obtained and an outage of sufficient duration is available for replacement.

f. Replacement equipment qualified in accordance with § 50.49 would require significant plant modifications to accommodate its use.

g. The use of replacement equipment qualified in accordance with § 50.49 has a significant probability of creating human factor problems that would negatively affect plant safety and performance, for example:

(1) Knowledge, skills, and ability of existing plant staff would require significant upgrading to operate or maintain the specific replacement equipment;

(2) The use of the replacement equipment would create a one-of-a-kind application; or

(3) Maintenance, surveillance, or calibration activities would be unnecessarily complex.

7. In addition to the requirements of paragraph 50.49(j) of 10 CFR Part 50 and Section 8, "Documentation," of IEEE Std 323-1974, documentation should address the information identified in Appendix E to this guide. A record of the qualification should be maintained in an auditable file to permit verification that each item of electric equipment is qualified to perform its safety function under its postulated environmental conditions throughout its installed life.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described herein will be used in the evaluation of the qualification of electric equipment for all operating plants and plants that have not received an operating license subject to the following:

In accordance with paragraph 50.49(k), applicants for and holders of operating licenses are not required to requalify electric equipment important to safety (replacement equipment excepted) in accordance with the provisions of § 50.49 and in accordance with this guide if the NRC has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." These applicants and licensees may continue to use the criteria in these documents for qualifying electric equipment important to safety in the affected plants, with the exception of replacement equipment.

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APPENDIX A

TYPICAL SAFETY-RELATED ELECTRIC EQUIPMENT OR SYSTEMS*

Engineered Safety Feature Actuation
Reactor Protection
Containment Isolation
Steamline Isolation
Main Feedwater Shutdown and Isolation
Emergency Power
Emergency Core Cooling
Containment Heat Removal
Containment Fission Product Removal
Containment Combustible Gas Control
Auxiliary Feedwater
Containment Ventilation
Containment Radiation Monitoring
Control Room Habitability System (e.g., HVAC, Radiation Filters)
Ventilation for Areas Containing Safety Equipment
Component Cooling
Service Water
Emergency Systems to Achieve Safe Shutdown

*Paragraph 50.49(b)(1) identifies safety-related electric equipment as a subset of electric equipment important to safety and defines it as the equipment that is relied upon to remain functional during and following design basis events to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) 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.

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APPENDIX B

TYPICAL EXAMPLES OF NON-SAFETY-RELATED EQUIPMENT

Associated circuits, as defined in Regulatory Guide 1.75, "Physical Independence of Electric Systems," need only be qualified to ensure that they will not fail under postulated environmental conditions in a manner that could prevent satisfactory accomplishment of safety functions by safety-related equipment.

The equipment identified in Examples 1, 2, and 3 has typically been classified as safety-related on recently licensed plants. However, some operating plants were licensed using less definitive safety classification criteria than those applied to recent designs, and they may contain non-safety-related equipment such as that in Examples 1, 2, and 3. The provisions of § 50.49 require that the licensee provide appropriate environmental qualification for equipment described in these examples regardless of the safety classification of that equipment.

Example 4 applies to some plants, depending on the specific location of control system components.

Example 1

The injection of emergency feedwater (EFW) for PWRs and high-pressure coolant injection (HPCI) for BWRs are safety-related functions. The EFW system and the HPCI system are initiated upon detection of low water level. Automatic termination of these systems upon detection of high water level may also be provided. The high-level trip in some cases has been considered an equipment protection device; however, the inadvertent termination of EFW or HPCI due to misoperation of the level sensing equipment when subjected to a harsh environment could defeat the safety-related injection function. Thus the electric equipment associated with automatic termination of the injection must be environmentally qualified.

Example 2

In some cases, the electrical control system for a pump (for example, a charging pump or an emergency core cooling system pump) will include termination commands on loss of lubrication oil pressure or low suction pressure. These features are provided for equipment protection. Failure of these features, however, would defeat the safety-related function. They must therefore be environmentally qualified.

Example 3

A safety-related fluid system may have non-safety-related portions of the system that are isolated from the safety-related portions of the system upon the generation of a safety feature actuation signal. Isolation may be performed by motor-operated valves. These valve operators must be environmentally qualified.

Example 4

Harsh environments associated with HELBs could cause control system malfunctions resulting in consequences more severe than those for the HELBs analyzed in the FSAR (Chapter 15) or beyond the capability of operators or safety systems. In these cases, the control system failures could prevent satisfactory accomplishment of the safety functions required for the HELBs. Typical examples of control systems that could fail as a result of an HELB and whose consequential failure may not be bounded by HELBs analyzed in the FSAR are:

1. The automatic rod control system,

2. The pressurizer power-operated relief valve control system,

3. The main feedwater control system,

4. The steam generator power-operated relief valve control system, and

5. The turbine generator control system.

Based on the above, it may be necessary to environmentally qualify components associated with various control systems.

APPENDIX C

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

LOSS-OF-COOLANT ACC[illegible]NT

Acceptable methods for calculating the mass and energy release to determine the loss-of-coolant accident environment for PWR and BWR plants are described in the following:

1. Topical Report WCAP-8312A for Westinghouse plants.

2. Section 6.2.1 of CESSAR System 80 PSAR for Combustion Engineering plants.

3. Appendix 6A of B-SAR-205 for Babcock & Wilcox plants.

4. NEDO-10320 and Supplements 1 and 2 for General Electric plants. NEDO-20533 dated June 1974 and Supplement 1 dated August 1975 for GE Mark III.

MAIN STEAM LINE BREAK

Acceptable methods for calculating the mass and energy release to determine the main steam line break environment are described in the following:

1. Topical Report WCAP-8822 (MARVEL/TRANSFLA) for Westinghouse plants. Use of this method is acceptable for all Westinghouse plants with the exception that a plant-specific containment temperature analysis will be required for ice condenser containments.

2. Appendix 6B of CESSAR System 80 PSAR for Combustion Engineering plants.

3. Section 15.1.14 of B-SAR-205 for Babcock & Wilcox plants.

4. Same as item 4 above for General Electric plants.

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APPENDIX D

METHODOLOGY AND SAMPLE CALCULATION FOR QUALIFICATION RADIATION DOSE

This appendix illustrates the staff model for calculating dose rates and integrated doses for equipment qualification purposes. The doses shown in Figure D-1 include contributions from airborne and plateout radiation sources in the containment and cover a period of one year following the postulated fission product release. The dose values shown are provided for illustration only and may not be appropriate for plant-specific application for equipment qualification levels. The dose levels intended for qualification purposes should be determined using the maximum time the equipment is intended to function. It should be noted, however, that for equipment that must be qualified for more than thirty days, a source term that incorporates considerable quantities of cesium as suggested by the accident at Three Mile Island Unit 2 (TMI-2) may produce doses greater than those estimated by the present source term.

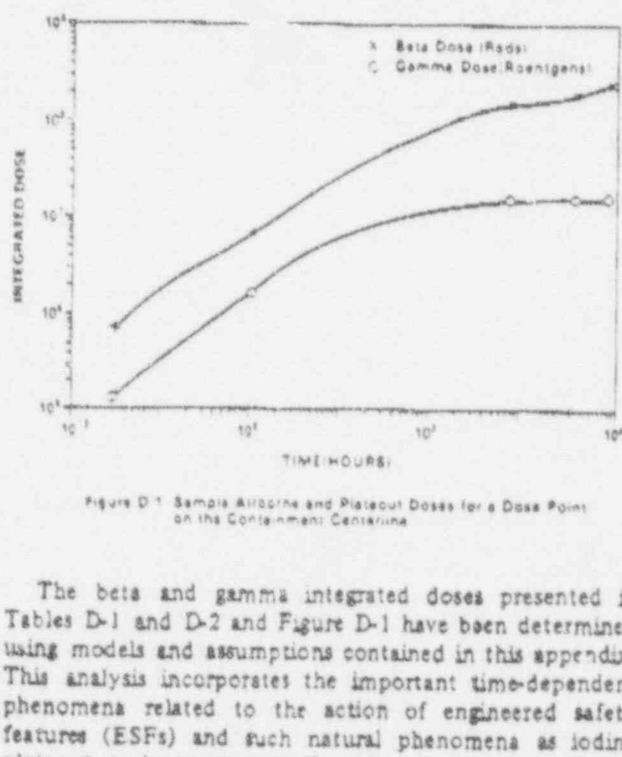

Figure D-1 Sample Airborne and Plateout Doses for a Dose Point on the Containment Centerline

The beta and gamma integrated doses presented in Tables D-1 and D-2 and Figure D-1 have been determined using models and assumptions contained in this appendix. This analysis incorporates the important time-dependent phenomena related to the action of engineered safety features (ESFs) and such natural phenomena as iodine plateout, as in previous staff analyses.

Doses were calculated for a point inside the containment (at the midpoint of the containment) taking sprays and plateout mechanisms into account. The doses presented in Figure D-1 are values for a PWR plant having a containment free volume of 2.5 million cubic feet and a power rating of 4100 MWt.

1. BASIC ASSUMPTIONS USED IN THE ANALYSIS

Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) activity suspended in the containment atmosphere, (2) activity plated out on containment surfaces, and (3) activity mixed in the containment sump water. A given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

Following the TMI-2 accident, the staff concluded that a thorough examination of the source term assumptions for equipment qualification was warranted. It is recognized, however, that the TMI-2 accident represents only one of a number of possible accident sequences leading to a release of fission products and that the mix of fission products released under various core conditions could vary substantially.

Research under way may lead to modifications in source term assumptions. The research will consider the experience from the TMI-2 accident of 1979, contemporary fission product release phenomenology, the transport and attenuation of fission products in primary coolant systems and containments, and distinctions between design basis accidents and events beyond the design basis. This research may result in revision of this guide.

2. ASSUMPTIONS USED IN CALCULATING FISSION PRODUCT CONCENTRATIONS

This section discusses the assumptions used to simulate the PWR and BWR containments for determining the time-dependent and location-dependent distribution of the airborne noble gas and iodine activity within the containment atmosphere, the activity plated out on containment surfaces, and the activity in the sump water.

The staff used a computer program, TACT, to model the time-dependent behavior of iodine and noble gases within a nuclear power plant. The TACT code or other equivalent industry codes would provide an acceptable method for modeling the transfer of activity from one containment region to another and for modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 1), is used to calculate the removal rates of elemental iodine by plateout and

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IMAGE EVALUATION TEST TARGET (MT-3)

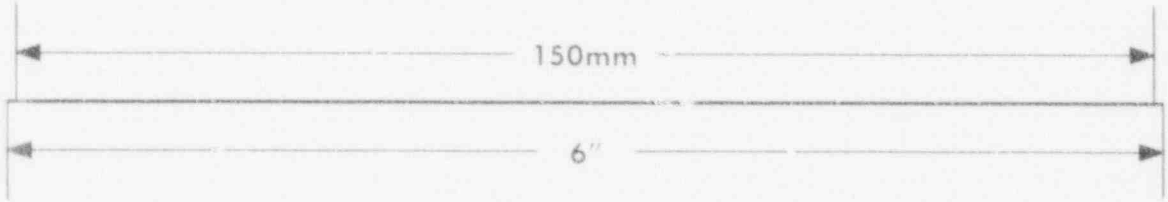

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IMAGE EVALUATION
TEST TARGET (MT-3)

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2

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

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sprays. These codes were used to develop the source term estimates. The assumptions in the following sections were used to calculate the distribution of radioactivity within the containment following a design basis LOCA.

2.1 PWR Dry Containments

The following methods and assumptions were used by the staff for calculating the radiation environment in PWR dry containments:

1. In the analysis of the accident radiation environment, the staff assumed that 50% of the iodine core activity inventory and 100% of the core noble gas activity inventory were released instantaneously to the containment atmosphere. One percent of the remaining "solids" activity inventory was assumed released from the core and carried with the primary coolant directly to the containment sump.

2. The containment free volume was taken as 2.52 x 10^6 ft^3. Of this volume, 74% or 1.86 x 10^6 ft^3 was assumed to be directly covered by the containment sprays, leaving 6.6 x 10^5 ft^3 of the containment free volume unsprayed. The latter includes regions within the main containment space under the containment dome and compartments below the operating floor level. (Plants with different containment free volumes should use plant-specific values.)

3. The initial distribution of activity within the containment should be based on realistic assumptions. The staff's examples assumed a relatively open (non-compartmented) containment with a large release uniformly distributed in the containment. This is a reasonable simplification for dose assessment in a large dry PWR containment and it is realistic in terms of specifying the time-dependent radiation environment in most areas of the containment.

4. The ESF fans were assumed to have a design flow rate of 220,000 cfm in the post-LOCA environment. Mixing between all major unsprayed regions and compartments and the main sprayed region was assumed.

5. Effects of the ESF systems that remove airborne activity or redistribute activity within containment (e.g., containment spray and containment ventilation systems) should be evaluated using assumptions consistent with previous licensing practice. For example, the air exchange between the sprayed and unsprayed regions was assumed to be one-half of the design flow rate of the ESF fans. Good mixing of the containment activity between the sprayed and unsprayed regions is ensured by natural convection currents and ESF fans.

6. The containment spray system was assumed to have two equal-capacity trains each designed to inject 3000 gpm of boric acid solution into the containment.

7. Trace levels of hydrazine were assumed to be added during the injection phase to enhance the removal of iodine. Further, this model assumes that during the recirculation phases, the pH of the sump water is maintained above 8.5.

8. The spray removal rate constant (λ) was calculated using the staff's SPIRT program, conservatively assuming the operation of only one spray train and an instantaneous partition coefficient (H) for elemental iodine of 5000. The calculated value of the spray removal constant for elemental iodine was 27.2 hr^{-1}.

9. Natural deposition (i.e., plateout) of airborne activity should be determined using a mechanistic model (see Reference 1). In the staff's example, plateout of iodine on containment internal surfaces was modeled as a first-order rate removal process, and best estimates for model parameters were assumed. Based on an assumed total surface area within containment of approximately 5.0 x 10^5 ft^2, the calculated value for the overall plateout constant for elemental iodine was 1.23 hr^{-1}. The assumption that 50% of the activity is instantaneously plated out should not be used.

10. The spray removal and plateout processes were modeled as competing iodine removal mechanisms. Removal of iodine from surfaces by the flow of condensed steam or by washoff by the containment spray may be assumed if such effects can be verified and quantified by analysis or experiment.

11. A spray removal rate constant (λ) for particulate iodine concentration was calculated using the staff's SPIRT program (Ref. 1). The staff calculated a value of λ = 0.43 hr^{-1} and allowed the removal of particulate iodine to continue until the airborne concentration was reduced by a factor of 10^4. The organic iodine concentration in the containment atmosphere is assumed not to be affected by either the containment spray or plateout removal mechanisms.

12. The sprays were assumed to remove elemental iodine until the instantaneous concentration in the sprayed region was reduced by a factor of 200. This is necessary to achieve an equilibrium airborne iodine concentration consistent with previous LOCA analyses.

13. The analysis assumed that more than one species of radioactive iodine is present in a design basis LOCA. The calculation of the post-LOCA environment assumed that, of the 50% of the core inventory of iodine released, 5% is associated with airborne particulate materials, 4% forms organic compounds, and 91% remains as elemental iodine. For conservatism, this composition was assumed present at time t = 0. (These assumptions concerning the iodine form are obtained from Regulatory Guides 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," when a plateout factor of 2 is assumed for the elemental form.)

14. The staff analysis conservatively assumed that no leakage from the containment building to the environment occurred.

15. Removal of airborne activity by engineered safety features may be assumed when calculating the radiation environment following other non-LOCA design basis accidents provided the safety features systems are automatically activated as a result of the accident.

16. The radiation environment resulting from normal operation should be based on the conservative source term estimates reported in the plant's Safety Analysis Report or should be consistent with the primary coolant specific activity limits contained in the plant's technical specifications. The use of equilibrium primary coolant concentrations based on 1% fuel cladding failures would be one acceptable method.

2.2 PWR Ice Condenser Containments

The assumptions and methods presented for calculating the radiation environment in PWR dry containments are appropriate for use in calculating the radiation environment for ice condenser containments following a design basis LOCA with the following modifications:

1. The source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the ice beds as functions of time.

2. Credit may be taken for iodine removal via the operation of the ice beds and the spray system. A time-dependent removal efficiency consistent with the steam/air mixture for elemental iodine may be assumed.

3. Removal of airborne iodine in the upper compartment of the containment by the action of both plateout and spray processes may be assumed provided these removal processes are evaluated using conditions and assumptions consistent with items 6 through 12 in Section 2.1 and plant-specific parameters.

2.3 BWR Containments

The assumptions and methods presented for calculating the radiation environment in PWR dry containments are appropriate for use in calculating the radiation environment for BWRs following a design basis LOCA with the following modifications:

1. A decontamination factor (DF) of 10 may be assumed for both elemental and particulate iodine as the iodine activity passes through the suppression pool. No credit should be taken for the removal of organic iodine or noble gases in the suppression pool.

2. For Mark III designs, all of the activity passing through the suppression pool should be assumed instantaneously and uniformly distributed within the containment. For the Mark I and Mark II designs, all of the activity should be assumed initially released to the drywell area and the transfer of activity from these regions via containment leakage to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

3. Removal of airborne iodine in the drywell or reactor building by the action of both plateout and spray processes may be assumed provided the effectiveness of these competing iodine removal processes are evaluated using conditions and assumptions consistent with items 6 through 12 in Section 2.1 and plant-specific parameters.

4. The removal of airborne activity from the reactor building by operation of the standby gas treatment system (SGTS) may be assumed.

3. MODEL FOR CALCULATING THE DOSE RATE OF AIRBORNE AND PLATEOUT FISSION PRODUCTS

The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere were calculated for the midpoint in the containment. The containment was modeled as a cylinder with the height and diameter equal. Containment shielding and internal structures were neglected because they would involve a degree of complexity beyond the scope of the present work. The calculations of Reference 2 indicate that the specific internal shielding and structure would be expected to reduce the gamma doses and dose rates by factors of two or more depending on the specific location and geometry.

Because of the short range of the betas in air, the airborne beta doses presented in Tables D-1 and D-2 were calculated using an infinite medium approximation. This is shown in Reference 3 to result in only a small error. Beta doses for equipment located on the containment walls or on large internal structures may be calculated using the semiinfinite beta dose model.

The staff recognizes that this approach is conservative and that, for most plant-specific calculations, a semiinfinite beta dose model may be more appropriate. The use of the semiinfinite model is acceptable provided there is sufficient justification for its use (such as location, shielding, minimal thickness). Further, the staff recognizes that for some equipment the use of a finite-cloud beta dose model may be warranted. Because the use of the finite-cloud model would result in beta doses much smaller than the values presented in Table D-2, a case-by-case justification for use of the finite-cloud model will be required.

The gamma dose rate contribution from the plated-out iodine on containment surfaces to the point on the centerline was also included. The model calculated the plateout activity in the containment assuming only one spray train and one ventilation system were operating. It should be noted that washoff of the plated-out iodine activity by the sprays was not addressed in this evaluation.

Finally, all gamma doses were multiplied by a correction factor of 1.3 as suggested in Reference 3 to account for the omission of the contribution from the decay chains of the isotopes.

4. MODEL FOR CALCULATING THE DOSE RATE OF SUMP FISSION PRODUCTS

The staff model assumed the washout of airborne iodine from the containment atmosphere to the containment sump. For a PWR containment with sprays and good mixing between the sprayed and unsprayed regions, the elemental iodine (assumed to constitute 91% of the released iodine) is very rapidly washed out of the atmosphere to the containment sump (typically 90% of the airborne iodine in less than 15 minutes).

The dose calculations may assume a time-dependent iodine source. (The difference between the integrated dose calculated on the assumption of 50% of the core iodine immediately available in the sump and that calculated on the assumption of a time-dependent sump iodine buildup is not significant.)

The "solid" fission products should be assumed to be instantaneously carried by the coolant to the sump and uniformly distributed in the sump water. The gamma and beta dose rates and the integrated doses should be computed for a center point located at the surface of the large pool of sump water, and the dose rate calculation should include an estimate of the effects of buildup.

5. CONCLUSION

The values given in Tables D-1 and D-2 and Figure D-1 for the various locations in the containment provide an estimate of expected radiation qualification values for a 4100 MWt PWR design.

The NRC Office of Nuclear Regulatory Research is continuing its research efforts in the area of source terms for equipment qualification following design basis accidents. As more information in this area becomes available, the source terms and staff models may change to reflect the new information.

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Table D-1

ESTIMATES FOR TOTAL AIRBORNE GAMMA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

Time (Hr)	Airborne Iodine Dose (R)	Airborne Noble Gas Dose (R)	Plateout Iodine Dose (R)	Total Dose (R)
0.00	–	–	–	–
0.03	4.82E+4	7.42E+4	1.69E+3	1.24E+5
0.06	8.57E+4	1.39E+5	3.98E+3	2.29E+5
0.09	1.09E+5	1.98E+5	7.22E+3	3.14E+5
0.12	1.25E+5	2.51E+5	1.10E+4	3.87E+5
0.15	1.38E+5	3.01E+5	1.52E+4	4.54E+5
0.18	1.47E+5	3.48E+5	1.96E+4	5.15E+5
0.21	1.55E+5	3.92E+5	2.41E+4	5.71E+5
0.25	1.64E+5	4.49E+5	3.03E+4	6.43E+5
0.38	1.87E+5	6.19E+5	5.05E+4	8.57E+5
0.50	2.03E+5	7.61E+5	6.90E+4	1.03E+6
0.75	2.36E+5	1.03E+6	1.06E+5	1.37E+6
1.00	2.66E+5	1.26E+6	1.40E+5	1.67E+6
2.00	3.62E+5	2.04E+6	2.61E+5	2.66E+6
5.00	5.50E+5	3.56E+6	5.40E+5	4.65E+6
8.00	6.63E+5	4.38E+6	7.47E+5	5.79E+6
24.0	1.01E+6	6.26E+6	1.45E+6	8.72E+6
60.0	1.31E+6	7.16E+6	2.10E+6	1.06E+7
96.0	1.45E+6	7.56E+6	2.39E+6	[illegible]
192	1.68E+6	8.29E+6	[illegible]	[illegible]
298	1.85E+6	8.76E+6	[illegible]	[illegible]
394	1.95E+6	8.85E+6	3.41E+6	[illegible]
560	2.07E+6	9.06E+6	3.64E+6	1.48E+7
720	2.13E+6	9.15E+6	3.76E+6	1.50E+7
888	2.16E+6	9.19E+6	3.83E+6	1.52E+7
1060	2.18E+6	9.21E+6	3.87E+6	1.53E+7
1220	2.19E+6	9.21E+6	3.89E+6	1.53E+7
1390	2.20E+6	9.21E+6	3.90E+6	1.53E+7
1560	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1730	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1900	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2060	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2230	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2950	2.20E+6	9.23E+6	3.92E+6	1.54E+7
3670	2.20E+6	9.24E+6	3.92E+6	1.54E+7
4390	2.20E+6	9.24E+6	3.92E+6	1.54E+7
5110	2.20E+6	[illegible]	3.92E+6	1.54E+7
5830	2.20E+6	9.25E+6	3.92E+6	1.54E+7
6550	2.20E+6	9.26E+6	3.92E+6	1.54E+7
7270	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8000	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8710	2.20E+6	9.28E+6	3.92E+6	1.54E+7
			Total	1.54E+7

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Table D-2

ESTIMATES FOR TOTAL AIRBORNE BETA DOSE
CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

Time (Hr)	Airborne Iodine Dose (rads)*	Airborne Noble Gas Dose (rads)*	Total Dose (rads)*
0.00	-	-	-
0.03	1.47E+5	5.48E+5	6.95E+5
0.06	2.62E+5	9.86E+5	1.25E+6
0.09	3.33E+5	1.35E+5	1.68E+6
0.12	3.83E+5	1.65E+6	2.03E+6
0.15	4.20E+5	1.91E+6	2.33E+6
0.18	4.49E+5	2.14E+6	2.59E+6
0.21	4.73E+5	2.35E+6	2.82E+6
0.25	5.00E+5	2.60E+6	3.10E+6
0.28	5.67E+5	3.30E+6	3.87E+6
0.50	6.15E+5	3.86E+6	4.48E+6
0.75	7.13E+5	4.89E+6	5.60E+6
1.00	8.00E+5	5.81E+6	6.61E+6
2.00	1.07E+6	9.02E+6	1.01E+7
5.00	1.58E+6	1.65E+7	1.81E+7
8.00	1.88E+6	2.20E+7	2.39E+7
24.0	2.87E+6	4.08E+7	4.37E+7
60.0	3.89E+6	6.15E+7	6.54E+7
96.0	4.37E+6	7.48E+7	7.92E+7
192	5.14E+6	1.00E+8	1.05E+8
298	5.64E+6	1.17E+8	1.23E+8
394	5.99E+6	1.25E+8	1.31E+8
560	6.34E+6	1.34E+8	1.40E+8
720	6.53E+6	1.39E+8	1.46E+8
888	6.63E+6	1.42E+8	1.49E+8
1060	6.69E+6	1.44E+8	1.51E+8
1220	6.73E+6	1.45E+8	1.52E+8
1390	6.75E+6	1.47E+8	1.54E+8
1560	6.76E+6	1.49E+8	1.56E+8
1730	6.76E+6	1.51E+8	1.58E+8
1900	6.76E+6	1.52E+8	1.59E+8
2060	6.76E+6	1.54E+8	1.61E+8
2230	6.77E+6	1.55E+8	1.62E+8
2950	6.77E+6	1.62E+8	1.69E+8
3670	6.77E+6	1.69E+8	1.76E+8
4390	6.77E+6	1.76E+8	1.83E+8
5110	6.77E+6	1.83E+8	1.90E+8
5830	6.77E+6	1.89E+8	1.96E+8
6550	6.77E+6	1.96E+8	2.03E+8
7270	6.77E+6	2.03E+8	2.10E+8
8000	6.77E+6	2.09E+8	2.16E+8
8710	6.77E+6	2.16E+8	2.23E+8
		Total	2.23E+8

*Dose conversion factor is based on absorption by tissue.

APPENDIX D

REFERENCES

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2. E. A. Warman and E. T. Boulette, "Engineering Evaluation of Radiation Environment in LWR Containments," in *Transactions of the American Nuclear Society*, Vol. 23, pp. 604-605, 1976.

3. M. J. Kolar and N. C. Olson, "Calculation of Accident Doses to Equipment Inside Containment of Power Reactors," in *Transactions of the American Nuclear Society*, Vol. 22, pp. 808-809, 1975.

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Normand, E., and W. R. Determan, "A Simple Algorithm to Calculate the Immersion Dose," in *Transactions of the American Nuclear Society*, Vol. 18, pp. 358-359, 1974.

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Rogovin, M., et al., "Three Mile Island—A Report to the Commissioners and to the Public," NUREG/CR-1250, Volume II, Part 2, April 5, 1979.*

U.S. Nuclear Regulatory Commission, "Technical Basis for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981.*

*Copies are available from the National Technical Information Service, Springfield, Virginia 22161.

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APPENDIX E

QUALIFICATION DOCUMENTATION FOR ELECTRIC EQUIPMENT

In order to ensure that an environmental qualification program conforms to General Design Criteria 1, 2, 4, and 23 of Appendix A, Sections III, XI, and XVII of Appendix B; and § 50.49 of 10 CFR Part 50, the following information on the qualification program should be submitted to NRC for electric equipment within the scope of this guide:

1. Provide a list of all electric equipment within the scope of this guide such as the following:

   a. Switchgear
   b. Motor control centers
   c. Valve operators and solenoid valves
   d. Motors
   e. Logic equipment
   f. Cable
   g. Connectors
   h. Sensors (pressure, pressure differential, temperature, flow and level, neutron, and other radiation)
   i. Limit switches
   j. Heaters
   k. Fans
   l. Control boards
   m. Instrument racks and panels
   n. Electric penetrations
   o. Splices
   p. Terminal blocks

2. For each item of equipment identified in 1, provide the following:

   a. Type (functional designation)
   b. Manufacturer
   c. Manufacturer's type number and model number
   d. Plant ID/tag number and location

3. Categorize the equipment identified in item 1 into one of the following categories:

   a. Equipment that will experience the environmental conditions of design basis accidents through which it must function to mitigate such accidents; it must be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

   b. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of such accidents but through which it must not fail in a manner detrimental to plant safety or accident mitigation; it must be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.

   c. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of such accidents and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation; it need not be qualified for any accident environment.

   d. Equipment that has performed its safety function prior to the exposure to an accident environment and whose failure (in any mode) is deemed not detrimental to plant safety and will not mislead the operator; it need not be qualified for any accident environment.

4. For each item of equipment in the categories of equipment listed in item 3, provide the following:

   a. The system safety function requirements for equipment in categories 3.a, 3.b, and 3.d.

   b. An environmental envelope as a function of time that includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown and design basis accident (including LOCA and MSLB), including postaccident conditions, for equipment in categories 3.a and 3.b.

   c. Length of time equipment in categories 3.a and 3.b must perform its safety function when subjected to any of the limiting environment specified above.

   d. The technical bases that justify the placement of each item of equipment in categories 3.b, 3.c, and 3.d.

5. For each item of equipment identified in categories 3.a and 3.b, state the actual qualification envelope simulated during testing (defining the duration of the environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent "qualification envelope" so derived.

6. Provide a summary of test results that demonstrates the adequacy of the qualification program. If any analysis is used for qualification, justification of all analysis assumptions must be provided.

7. Identify the qualification documents that contain detailed supporting information, including test data, for items 5 and 6.

VALUE/IMPACT STATEMENT

Background

The Commission (in Memorandum and Order CLI-80-21 dated May 23, 1980) directed the staff to use NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," along with a document entitled "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines, January 14, 1980) as requirements that licensees and applicants must meet in order to satisfy the equipment qualification requirements of 10 CFR Part 50. Subsequently, the Commission approved a final rule for electric equipment qualification (§ 50.49 of 10 CFR Part 50). Revision 1 to Regulatory Guide 1.89 will provide an acceptable method for meeting the requirements of § 50.49.

Substantive Changes and Their Value/Impact

The following positions were added in Revision 1 to Regulatory Guide 1.89:

1. Regulatory Position C.1, which adds to the scope of the guide non-safety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions (for example, the associated circuits defined in Regulatory Guide 1.75, "Physical Independence of Electric Systems") and certain postaccident monitoring equipment.

2. Regulatory Position C.2, which provides the staff position on establishing performance and environmental requirements for equipment qualification. Methods for establishing temperature and pressure profiles for a loss-of-coolant accident and main steam line break are provided, and radiological source terms are given.

3. Regulatory Position C.3, which provides the staff position pertaining to test procedures.

4. Regulatory Position C.4, which provides the staff position regarding establishing margin in testing requirements.

5. Regulatory Position C.5, which provides the staff position regarding aging of equipment.

6. Regulatory Position C.6, which provides the staff position regarding qualification of replacement equipment.

7. Regulatory Position C.7, which provides the staff position on the documentation of equipment qualification procedures and results.

Value - This guide provides the staff's views on individual sections of IEEE Std 323-1974 and describes acceptable methods for meeting the requirements of § 50.49 of 10 CFR Part 50. This guide should enhance the licensing process.

Impact - This regulatory guide does not impose any new costs or obligations on licensees or applicants. Thus, no impact will result from issuance of this guide with respect to requirements in effect at this time.

November 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.89

QUALIFICATION OF CLASS IE EQUIPMENT FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

Criterion III, "Design Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that design control measures provide for verifying the adequacy of a specific design feature by design reviews, by calculational methods, or by suitable qualification testing of a prototype unit under the most adverse conditions. This regulatory guide describes a method acceptable to the Regulatory staff for complying with the Commission's regulations with regard to design verification of Class IE equipment for service in light-water-cooled and gas-cooled nuclear power plants.

B. DISCUSSION

IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,"[1] dated February 28, 1974, was prepared by Subcommittee 2, Equipment Qualification, of the Nuclear Power Engineering Committee of the Institute of Electrical and Electronics Engineers, Inc., (IEEE) and subsequently was approved by the IEEE Standards Board on December 13, 1973. The standard describes basic procedures for qualifying Class IE equipment and interfaces that are to be used in nuclear power plants and components or equipment of any interface whose failure could adversely affect any class IE equipment.

The requirements delineated include principles, procedures, and methods of qualification which, when satisfied, will confirm the adequacy of the equipment design for the performance of Class IE functions under normal, abnormal, design-basis-event, post-design-basis-event, and containment-test conditions.

C. REGULATORY POSITION

The procedures described in IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,"[1] dated February 28, 1974, for qualifying Class IE equipment for service in light-water-cooled and gas-cooled nuclear power plants are generally acceptable and provide an adequate basis for complying with design verification requirements of Criterion III of Appendix B to 10 CFR Part 50 to verify adequacy of design under the most adverse design conditions subject to the following:

1. Reference is made in IEEE Std 323-1974, Sections 2, 6.3.2(5), and 6.3.5, to IEEE Std 344-1971, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations." The specific applicability or acceptability of IEEE Std 344 will be covered separately in other regulatory guides, where appropriate.
2. The radiological source term for qualification tests in a nuclear radiation environment should be based on the same source term as that used in Regulatory Guide 1.7 (Safety Guide 7, 3/10/71) for BWRs and PWRs. An equivalent source term (i.e., 100% of the noble gases, 50% of the halogens, and 1% of the remaining solids developed from maximum full-power operation of the core) should be used for HTGRs. The containment size should be taken into account in each case. For exposed organic materials, calculations should take into account both beta and gamma radiation.

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

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the Regulatory staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, this guide will be used by the Regulatory staff in evaluating all construction permit applications for which the issue date of the Safety Evaluation Report (SER) is July 1, 1974, or after.

For those construction permit applications for which an SER was issued prior to July 1, 1974, the Regulatory staff may, subsequent to issuance of the construction permit (or operating license), reevaluate the Safety Analysis Report on a case-by-case basis to assure that acceptable methods for qualification of Class IE equipment have been specified in purchase orders executed for such equipment on or after November 15, 1974.

BDM Zack Dotty Good 50-348/364-CIVP A-21 2/19/92

APCo Exhibit 21

J. RPM

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

DOCKETED USNRC

December 13, 1984

'92 MAR 13 P4:14

0054248

Docket Nos. 50-348
and 50-364

NRC-LICENSING
RPM FILES
Ballard

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

8412280507
8412280508
LC# 1657

DEC 1984 RECEIVED NLTS

Dear Mr. McDonald:

KWRD 4/7/87 4/14/87

New P: 4590
4589

Enclosures 1 and 2 are our Safety Evaluations (SEs) that relate to the environmental qualification of electric equipment important to safety at the Joseph M. Farley Nuclear Plant, Unit Nos. 1 and 2 and to your compliance with the requirements of 10 CFR 50.49. These SEs include proposed resolutions for the deficiencies identified in the earlier SEs dated January 31, 1983, and in the January 14 and 17, 1983 Franklin Research Center (FRC) Technical Evaluation Reports, and to your proposal that justifications for continued operation are not necessary.

On January 11, 1984 a meeting was held between your staff and the NRC staff to discuss your proposed method of resolution for each of the environmental qualification deficiencies identified. Discussions included the general methodology which you used to assure compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," which became effective February 22, 1983. We also discussed your proposed justifications for continued operation for those equipment items for which environmental qualification was not yet complete.

By letter dated February 29, 1984, you addressed the above subjects and documented the discussions held at the January 11, 1984 meeting. By letters dated March 14 and May 20, 1983 you provided additional information and stated that all electric equipment important to safety within the scope of 10 CFR 50.49 at both units is environmentally qualified and justifications for continued operation are not necessary. Based on our reviews, we conclude that the Alabama Power Company Equipment Qualification Program is in compliance with the requirements of 10 CFR 50.49, that the

DEC 1984
SENIOR V.P.

~~8412280507~~ 3pp

NUCLEAR REGULATORY COMMISSION

Docket No. ________ Official Exh. No. 21

In the matter of ALABAMA POWER CO.

Staff ________ IDENTIFIED 2/11/92

Applicant APCO ________ RECEIVED 2/19/92

Intervenor ________ REJECTED ________

Contr'g Off'r ________

Contractor ________ DATE 12-13-84

Other ________ Witness ________

Reporter L Estep

0054249

proposed resolution for each of the environmental qualification deficiencies identified for Farley Units 1 and 2 is acceptable, and that the continued operation of Farley Units I and 2 will not present undue risk to the public health and safety.

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:
As stated

cc w/enclosures:
See next page

0054250

Mr. R. P. McDonald
Alabama Power Company

Joseph M. Farley Nuclear Plant
Units 1 and 2

cc: Mr. W. O. Whitt
Executive Vice President
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Birmingham, Alabama 35291

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ENCLOSURE 1 0054251

SAFETY EVALUATION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
FARLEY UNIT 1
DOCKET NO. 50-348

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY

INTRODUCTION

Equipment which is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" (which supplements IEEE Standard 323 and various NRC Regulatory Guides and industry standards), and "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines).

BACKGROUND

On Feburary 8, 1979, the NRC Office of Inspection and Enforcement (IE) issued to all licensees of operating plants (except those included in the systematic evaluation program (SEP)) IE Bulletin (IEB) 79-01, "Environmental Qualification of Class 1E Equipment." This Bulletin, together with IE Circular 78-08 (issued on May 31, 1978), required the licensees to perform reviews to assess the adequacy of their environmental qualification programs.

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On January 14, 1980, NRC issued IEB 79-01B which included the DOR Guidelines and NUREG-0588 as attachments 4 and 5, respectively. Subsequently, on May 23, 1980, Commission Memorandum and Order CLI-80-21 was issued and stated that the DOR Guidelines and portions of NUREG-0588 form the requirements that licensees must meet regarding environmental qualification of safety-related electrical equipment in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. Supplements to IEB 79-01B were issued for further clarification and definition of the staff's needs. These supplements were issued on February 29, September 30, and October 24, 1980.

In addition, the staff issued orders dated August 29, 1980 (amended in September 1980) and October 24, 1980 to all licensees. The August order required that the licensees provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment. The October order required the establishment of a central file location for the maintenance of all equipment qualification records. The central file was mandated to be established by December 1, 1980. The staff subsequently issued a Safety Evaluation Report (SER) on environmental qualification of safety-related electrical equipment to the licensee on May 21, 1981. This SER directed the licensee to "either provide documentation of the missing qualification information which demonstrates that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action (requalification, replacement (etc.))." The licensee was required to respond to NRC within 90 days of receipt of the SER. In response to the staff SER issued in 1981, the licensee submitted additional information regarding the qualification of safety-related electrical equipment. This information was evaluated for the staff by the Franklin Research Center (FRC) in order to: 1) identify all cases where the licensee's response did not resolve the significant qualification issues, 2) evaluate the licensee's qualification documentation in accordance with established criteria to determine which equipment had adequate documentation and which did not, and 3) evaluate the licensee's qualification documentation for safety-related electrical equipment located in harsh environments required for TMI Lessons Learned Implementation. A Technical Evaluation Report (TER) was issued by FRC on January 14, 1983. A Safety Evaluation Report was subsequently issued to the Alabama Power Company on January 31, 1983, with the FRC TER as an attachment.

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for Farley Unit 1 may be qualified to the criteria specified in either the DOR Guidelines or NUREG-0588, except for replacement equipment. Replacement equipment installed subsequent to February 22, 1983 must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, unless there are sound reasons to the contrary.

A meeting was held with each licensee of plants for which a TER had been prepared for the staff by FRC in order to discuss all remaining open issues regarding environmental qualification, including acceptability of the environmental conditions for equipment qualification purposes, if this issue had not yet been resolved. On January 11, 1984, a meeting was held to discuss Alabama Power's proposed method to resolve the environmental qualification deficiencies identified in the January 31, 1983 SER and January 14, 1983 FRC TER. Discussions also included Alabama Power's general methodology for compliance with 10 CFR 50.49, and justification for continued operation f. those equipment items for which environmental qualification is not yet completed. The minutes of the meeting and proposed method of resolution for each of the environmental qualification deficiencie. are documented in a February 29, 1984 submittal from the licensee.

EVALUATION

The evaluation of the acceptability of the licensee's electric equipment environmental qualification program is based on the results of an audit review performed by the staff of: (1) the licensee's proposed resolutions of the environmental qualification deficiencies identified in the January 31, 1983 SER and January 14, 1983 FRC TER; (2) compliance with the requirements of 10 CFR 50.49; and (3) justification for continued operation (JCO) for those equipment items for which the environmental qualification is not yet completed.

Proposed Resolutions of Identified Deficiencies

The proposed resolutions for the equipment environmental qualification deficiencies, identified in the January 31, 1983 SER, and the FRC TER enc.osed with it, are described in the licensee's February 29, 1984 submittal. During the January 11, 1984 meeting with the licensee, the staff discussed the proposed resolution of each deficiency for each equipment items identified in the FRC TER and found the licensee's approach for resolving the identified environmental qualification deficiencies acceptabl· The majority of deficiencies identified were documentation, similarity, aging, qualified life and replacement schedule. All open items identified in the SER dated January 31, 1983 were also discussed and the resolution of these items has been found acceptable by the staff.

The approach described by the licensee for addressing and resolving th identified deficiencies includes replacing equipment, performing additional analyses, utilizing additional qualification documentation beyond that reviewed by FRC, obtaining additional qualification documentation, and determining that some equipment is outside the scope of 10 CFR 50.49, and therefore not required to be environmentally qualified, e.g., located in a mild environment. We discussed the proposed resolutions in detail on an item by item basis with the licensee during the January 11, 1984 meeting. Replacing or exempting equipment, for an acceptable reason, are clearly acceptable methods for resolving environmental qualification deficiencies. The more lengthy discussions with the licensee concerned the use of additional analyses or documentation. Although we did not review the additional analyses or documentation, we discussed how analysis was being used to resolve deficiencies identified in the FRC TER, and the content of the additional documentation in order to determine the acceptability of these methods. The licensee's equipment environmental qualification files will be audited by the staff during follow-up inspections to be performed by Region II, with assistance from IE Headquarters and NRR staff as necessary. Since a significant amount of documentation has already been reviewed by the staff and Franklin Research Center, the primary objective of the file audit will be to verify that they contain the appropriate analyses and other necessary documentation to support the licensee's conclusion that the equipment is qualified. The inspections will verify that the licensee's program for surveillance and maintenance of environmentally qualified equipment is adequate to assure that this equipment is maintained in the as analyzed or tested condition. The method used for tracking periodic replacement parts, and implementation of the licensee's commitments and actions, e.g., regarding replacement of equipment, will also be verified.

Based on our discussions with the licensee and our review of its submittal, we find the licensee's approach for resolving the identified environmental qualification deficiencies acceptable.

Compliance With 10 CFR 50.49

In its February 29, 1984 submittal, the licensee has described the approach used to identify equipment within the scope of paragraph (b)(1) of 10 CFR 50.49, equipment relied upon to remain functional during and following design basis events. The licensee states that the flooding and environmental (temperature, pressure, etc.) effects resulting from the worst case LOCA and HELB were considered in the IEB 79-01B and NUREG-0588 analyses. The capability of equipment to perform its intended function as a result of flooding in the containment or main steam valve room is documented in the IEB 79-01B and NUREG-0588 submittals. The effects of flooding in areas outside containment other than the main steam valve room were analyzed and found to have no adverse effects on the capability of equipment to perform its intended function as documented in FSAR Appendix 3K.

The harsh environmental condition of the worst-case LOCA and HELB envelops the environmental conditions for all other design-basis events as documented in FSAR Section 6.2. Therefore, the LOCA/HELB accidents are the only design-basis accidents which result in significantly adverse environments to electrical equipment that is required for safe shutdown or accident mitigation. Electrical equipment that could be subject to a harsh environment and is required to mitigate the consequences of design-basis events which result in harsh environments were included in the Master List of equipment.

The licensee's approach for identifying equipment within the scope of paragraph (b)(1) is in accordance with the requirements of that paragraph, and therefore acceptable.

The method used by the licensee for identification of electrical equipment within the scope of paragraph (b)(2) of 10 CFR 50.49, nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions, is summarized below:

1. The Master List was generated for electrical equipment as defined by 10 CFR 50.49(b)(1) that could be exposed to the harsh environments caused by design-basis events and that is required to remain functional during or following a LOCA or HELB. The harsh environmental condition of the worst-case LOCA and HELB envelops the environmental conditions for all other design-basis events as documented in FSAR Section 6.2. Therefore, the LOCA/HELB accidents are the only design-basis events that result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The Master List was developed by a review of design and as-built documentation, the FSAR, Technical Specifications, Emergency Operating Procedures, P&IDs, and electrical distribution diagrams to determine the systems and components required to perform the functions of reactor trip, containment isolation, and accident mitigation. Such electrical components that could be exposed to harsh environments resulted in the Master List. These electrical components include safety-related and nonsafety-related components and electrical components associated with plant auxiliary systems (e.g., Component Cooling Water) that are required for the operation of safety-related systems and equipment.

2. Elementary wiring diagrams of safety-related electrical equipment identified by the methods described in Item 1 above were reviewed to identify any auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment (e.g. automatic trips) where failure due to postulated environmental conditions could prevent

required operation of the safety-related equipment. If an adverse effect could result, the connected (interlocked) components (safety-related or nonsafety-related) were added to the Master List.

3. The operation of safety-related systems and equipment were reviewed to identify any directly mechanically connected auxiliary systems with electrical components which are necessary for the required operation of the safety-related equipment. None of the electrical equipment identified in the Master List requires the operation of directly mechanically connected auxiliary systems that depend on electrical components for operation. Plant auxiliary systems that are directly mechanically connected to and required for the operation of mechanical safety-related equipment (e.g., Component Cooling Water) were also reviewed to identify electrical components required to be environmentally qualified as discussed in Alabama Power Company's response to Item 1 above.

4. All nonsafety-related electrical circuits directly or indirectly associated with the electrical equipment identified in Step 1 by a common power supply are properly isolated by design through coordinated protective relays, circuit breakers, and fuses for electrical fault protection. The Farley Nuclear Plant original design criteria provided electrical fault protection devices to protect components connected to a common power supply. The electrical fault protection devices for equipment within the scope of 10 CFR 50.49 that are required to achieve a safe shutdown condition at FNP and within a potential harsh environment resulting from design-basis events are environmentally qualified. An electrical fault on the load side of a power supply feeder breaker or fuse would be isolated without effecting the remaining loads on the common power supply. The electrical design criteria included the use of applicable industry standards (e.g., IEEE, NEMA, ANSI, UL and NEC) and was reviewed and accepted by the NRC prior to receipt of the Farley Nuclear Plant operating license.

   The physical proximity of nonsafety-related electrical circuits associated with electrical equipment identified in Step 1 would not cause an environmental failure. In the judgment of Alabama Power Company, there is no known scenario for the failure of nonsafety-related electrical circuits whose close physical proximity would adversely impact the capabilities of the electrical equipment identified in Item 1 to perform their intended function in a harsh environment resulting from design-basis events.

We find the methodology being used by the licensee is acceptable since it provides reasonable assurance that equipment within the scope of paragraph (b)(2) of 10 CFR 50.49 has been identified.

With regard to paragraph (b)(3) of 10 CFR 50.49, the licensee has been granted an extension request by letter dated April 16, 1984, to the end of the sixth refueling outage scheduled to start in April 1985, but in any event no later than November 30, 1985. As stated in letter dated February 22, 1984, Alabama Power Company has interpreted the scope of 10 CFR 50.49(b)(3) to be those equipment items:

(a) defined as Category 1 and 2 instruments in Alabama Power Company's R.G. 1.97 Compliance Report, and

(b) not addressed by 10 CFR 50.49(b)(1) and (b)(2), and

(c) located in a harsh environment.

We find the licensee's approach to identifying equipment within the scope of paragraph (b)(3) of 10 CFR 50.49 acceptable since it is in accordance with the requirements of that paragraph.

<u>Justification for Continued Operation</u>

As stated in letters dated March 14, 1983 and May 20, 1983, it is the judgement of Alabama Power Company that all electric equipment important to safety within the scope of 10 CFR 50.49 at Farley Unit 1 is environmentally qualified and Justifications for Continued Operation (JCO's) are not necessary.

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CONCLUSIONS

Based on the above evaluation, we conclude the following with regard to the qualification of electric equipment important to safety within the scope of 10 CFR 50.49.

- Alabama Power's electrical equipment environmental qualification program complies with the requirements of 10 CFR 50.49.

- The proposed resolutions for each of the environmental qualification deficiencies identified in the January 31, 1983 SER and FRC TER are acceptable.

- Continued operation will not present undue risk to the public health and safety.

ENCLOSURE 2

0054260

SAFETY EVALUATION REPORT
OFFICE OF NUCLEAR REACTOR REGULATION
EQUIPMENT QUALIFICATION BRANCH
FARLEY UNIT 2
DOCKET NO. 50-364

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY

INTRODUCTION

Equipment which is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" (which supplements IEEE Standard 323 and various NRC Regulatory Guides and industry standards), and "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines).

BACKGROUND

On February 8, 1979, the NRC Office of Inspection and Enforcement (IE) issued to all licensees of operating plants (except those included in the systematic evaluation program (SEP)) IE Bulletin (IEB) 79-01, "Environmental Qualification of Class 1E Equipment." This Bulletin, together with IE Circular 78-08 (issued on May 31, 1978), required the licensees to perform reviews to assess the adequacy of their environmental qualification programs.

0054261

On January 14, 1980, NRC issued IEB 79-01B which included the DOR Guidelines and NUREG-0588 as attachments 4 and 5, respectively. Subsequently, on May 23, 1980, Commission Memorandum and Order CLI-80-21 was issued and stated that the DOR Guidelines and portions of NUREG-0588 form the requirements that licensees must meet regarding environmental qualification of safety-related electrical equipment in order to satisfy those aspects of 10 CFR 50, Appendix A, General Design Criterion (GDC) 4. Supplements to IEB 79-01B were issued for further clarification and definition of the staff's needs. These supplements were issued on February 29, September 30, and October 24, 1980.

In addition, the staff issued orders dated August 29, 1980 (amended in September 1980) and October 24, 1980 to all licensees. The August order required that the licensees provide a report, by November 1, 1980, documenting the qualification of safety-related electrical equipment. The October order required the establishment of a central file location for the maintenance of all equipment qualification records. The central file was mandated to be established by December 1, 1980. The staff subsequently issued a Safety Evaluation Report (SER) on environmental qualification of safety-related electrical equipment to the licensee on May 21, 1981. This SER directed the licensee to "either provide documentation of the missing qualification information which demonstrates that safety-related equipment meets the DOR Guidelines or NUREG-0588 requirements or commit to a corrective action (requalification, replacement (etc.))." The licensee was required to respond to NRC within 90 days of receipt of the SER. In response to the staff SER issued in 1981, the licensee submitted additional information regarding the qualification of safety-related electrical equipment. This information was evaluated for the staff by the Franklin Research Center (FRC) in order to: 1) identify all cases where the licensee's response did not resolve the significant qualification issues, 2) evaluate the licensee's qualification documentation in accordance with established criteria to determine which equipment had adequate documentation and which did not, and 3) evaluate the licensee's qualification documentation for safety-related electrical equipment located in harsh environments required for TMI Lessons Learned Implementation. A Technical Evaluation Report (TER) was issued by FRC on January 17, 1983. A Safety Evaluation Report was subsequently issued to the Alabama Power Company on January 31, 1983, with the FRC TER as an attachment.

0054262

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, Section 50.49 of 10 CFR 50, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for Farley Unit 2 may be qualified to the criteria specified in either in DOR Guidelines or NUREG-0588, except for replacement equipment. Replacement equipment installed subsequent to February 22, 1983 must be qualified in accordance with the provisions of 10 CFR 50.49, using the guidance of Regulatory Guide 1.89, unless there are sound reasons to the contrary.

A meeting was held with each licensee of plants for which a TER had been prepared for the staff by FRC in order to discuss all remaining open issues regarding environmental qualification, including acceptability of the environmental conditions for equipment qualification purposes, if this issue had not yet been resolved. On January 11, 1984, a meeting was held to discuss Alabama Power's proposed method to resolve the enviornmental qualification deficiencies identified in the January 31, 1983 SER and January 14, 1983 FRC TER. Discussions also include Alabama Power's general methodology for compliance with 10 CFR 50.49, and justification for continued operation for those equipment items for which environmental qualification is not yet completed. The minutes of the meeting and proposed method of resolution for each of the environmental qualification are documented in a February 29, 1984 submittal from the licensee.

EVALUATION

The evaluation of the acceptability of the licensee's electrical equipment environmental qualification program is based on the results of an audit review performed by the staff of: (1) the licensee's proposed resolutions of the environmental qualification deficiencies identified in the January 31, 1983 SER and January 17, 1983 FRC TER; (2) compliance with the requirements of 10 CFR 50.49; and (3) justification for continued operation (JCO) for those equipment items for which the environmental qualification is not yet completed.

0054263

Proposed Resolutions of Identified Deficiencies

The proposed resolutions for the equipment environmental qualification deficiencies, identified in the January 31, 1983 SER, and the FRC TER enclosed with it, are described in the licensee's February 29, 1984 submittal. During the January 11, 1984 meeting with the licensee, the staff discussed the proposed resolution of each deficiency for each equipment items identified in the FRC TER and found the licensee's approach for resolving the identified environmental qualification deficiencies acceptable. The majority of deficiencies identified were documentation, similarity, aging, qualified life and replacement schedule. All open items identified in the SER dated January 31, 1983 were also discussed and the resolution of those items has been found acceptable by the staff.

The approach described by the licensee for addressing and resolving the identified deficiencies includes replacing equipment, performing additional analyses, utilizing additional qualification documentation beyond that reviewed by FRC, obtaining additional qualification documentation, and determining that some equipment is outside the scope of 10 CFR 50.49, and therefore not required to be environmentally qualified, e.g., located in a mild environment. We discussed the proposed resolutions in detail on an item by item basis with the licensee during the January 11, 1984 meeting. Replacing or exempting equipment, for an acceptable reason, are clearly acceptable methods for resolving environmental qualification deficiencies. The more lengthy discussions with the licensee concerned the use of additional analyses or documentation. Although we did not review the additional analyses or documentation, we discussed how analysis was being used to resolve deficiencies identified in the FRC TER, and the content of the additional documentation in order to determine the acceptability of these methods. The licensee's equipment environmental qualification files will be audited by the staff during follow-up inspections to be performed by Region II, with assistance from IE Headquarters and NRR staff as necessary. Since a significant amount of documentation has already been reviewed by the staff and Franklin Research Center, the primary objective of the file audit will be to verify that they contain the appropriate analyses and other necessary documentation to support the licensee's conclusion that the equipment is qualified. The inspections will verify that the licensee's program for surveillance and maintenance of environmentally qualified equipment is adequate to assure that this equipment is maintained in the as analyzed or tested condition. The method used for tracking periodic replacement parts, and implementation of the licensee's commitments and actions, e.g., regarding replacement of equipment, will also be verified.

Based on our discussions with the licensee and our review of its submittal, we find the licensee's approach for resolving the identified environmental qualification deficiencies acceptable.

Compliance With 10 CFR 50.49

In its February 29, 1984 submittal, the licensee has described the approach used to identify equipment within the scope of paragraph (b)(1) of 10 CFR 50.49, equipment relied upon to remain functional during and following design basis events. The licensee states that the flooding and environmental (temperature, pressure, etc.) effects resulting from the worst case LOCA and HELB were considered in the IEB 79-01B and NUREG-0588 analyses. The capability of equipment to perform its intended function as a result of flooding in the containment or main steam valve room is documented in the IEB 79-01B and NUREG-0588 submittals. The effects of flooding in areas outside containment other than the main steam valve room were analyzed and found to have no adverse effects on the capability of equipment to perform its intended function as documented in FSAR Appendix 3K.

The harsh environmental condition of the worst-case LOCA and HELB envelops the environmental conditions for all other design-basis events as documented in FSAR Section 6.2. Therefore, the LOCA/HELB accidents are the only design-basis accidents which result in significantly adverse environments to electrical equipment that is required for safe shutdown or accident mitigation. Electrical equipment that could be subject to a harsh environment and is required to mitigate the consequences of design-basis events which result in harsh environments are included in the Master List of equipment.

The licensee's approach for identifying equipment within the scope of paragraph (b) (1) is in accordance with the requirements of that paragraph, and therefore acceptable.

The method used by the licensee for identification of electrical equipment within the scope of paragraph (b)(2) of 10 CFR 50.49, nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions, is summarized below:

1. The Master List was generated for electrical equipment as defined by 10 CFR 50.49(b)(1) that could be exposed to the harsh environments caused by design-basis events and that is required to remain functional during or following a LOCA or HELB. The harsh environmental condition of the worst-case LOCA and HELB envelops the environmental conditions for all other design-basis events as documented in FSAR Section 6.2. Therefore, the LOCA/HELB accidents are the only design-basis events that result in significantly adverse environments to electrical equipment which is required for safe shutdown or accident mitigation. The Master List was developed by a review of design and as-built documentation, the FSAR, Technical Specifications, Emergency Operating Procedures, P&IDs, and electrical distribution diagrams to determine the systems and components required to perform the functions of reactor trip, containment isolation, and accident mitigation. Such electrical components that could be exposed to harsh environments resulted in the Master List. These electrical components include safety-related and nonsafety-related components and electrical components associated with plant auxiliary systems (e.g., Component Cooling Water) that are required for the operation of safety-related systems and equipment.

2. Elementary wiring diagrams of safety-related electrical equipment identified by the methods described in Item 1 above were reviewed to identify any auxiliary devices electrically connected directly into the control or power circuitry of the safety-related equipment (e.g. automatic trips) where failure due to postulated environmental conditions could prevent

required operation of the safety-related equipment. If an adverse effect could result, the connected (interlocked) components (safety-related or nonsafety-related) were added to the Master List.

3. The operation of safety-related systems and equipment were reviewed to identify any directly mechanically connected auxiliary systems with electrical components which are necessary for the required operation of the safety-related equipment. None of the electrical equipment identified in the Master List requires the operation of directly mechanically connected auxiliary systems that depend on electrical components for operation. Plant auxiliary systems that are directly mechanically connected to and required for the operation of mechanical safety-related equipment (e.g., Component Cooling Water) were also reviewed to identify electrical components required to be environmentally qualified as discussed in Alabama Power Company's response to Item 1 above.

4. All nonsafety-related electrical circuits directly or indirectly associated with the electrical equipment identified in Step 1 by a common power supply are properly isolated by design through coordinated protective relays, circuit breakers, and fuses for electrical fault protection. The Farley Nuclear Plant original design criteria provided electrical fault protection devices to protect components connected to a common power supply. The electrical fault protection devices for equipment within the scope of 10 CFR 50.49 that are required to achieve a safe shutdown condition at FNP and within a potential harsh environment resulting from design-basis events are environmentally qualified. An electrical fault on the load side of a power supply feeder breaker or fuse would be isolated without effecting the remaining loads on the common power supply. The electrical design criteria included the use of applicable industry standards (e.g., IEEE, NEMA, ANSI, UL and NEC) and was reviewed and accepted by the NRC prior to receipt of the Farley Nuclear Plant operating license.

The physical proximity of nonsafety-related electrical circuits associated with electrical equipment identified in Step 1 would not cause an environmental failure. In the judgment of Alabama Power Company, there is no known scenario for the failure of nonsafety-related electrical circuits whose close physical proximity would adversely impact the capabilities of the electrical equipment identified in Item 1 to perform their intended function in a harsh environment resulting from design-basis events.

We find the methodology being used by the licensee is acceptable since it provides reasonable assurance that equipment within the scope of paragraph (b)(2) of 10 CFR 50.49 has been identified.

With regard to paragraph (b)(3) of 10 CFR 50.49, the licensee has been granted an extension request by letter dated October 21, 1983 until March 31, 1985. As stated in letter dated February 22, 1984, Alabama Power Company has interpreted the scope of 10 CFR 50.49(b)(3) to be those equipment items:

(a) defined as Category 1 and 2 instruments in Alabama Power Company's R.G. 1.97 Compliance Report, and

(b) not addressed by 10 CFR 50.49(b)(1) and (b)(2), and

(c) located in a harsh environment.

We find the licensee's approach to identifying equipment within the scope of paragraph (b)(3) of 10 CFR 50.49 acceptable since it is in accordance with the requirements of that paragraph.

Justification for Continued Operation

As stated in letters dated March 14, 1983 and May 20, 1983, it is the judgement of Alabama Power Company that all electric equipment important to safety within the scope of 10 CFR 50.49 at Farley Unit 2 is environmentally qualified and Justifications for Continued Operation (JCO's) are not necessary.

CONCLUSIONS

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Based on the above evaluation, we conclude the following with regard to the qualification of electric equipment important to safety within the scope of 10 CFR 50.49.

- Alabama Power's electrical equipment environmental qualification program complies with the requirements of 10 CFR 50.49.

- The proposed resolutions for each of the environmental qualification deficiencies identified in the January 31, 1983 SER and FRC TER are acceptable.

- Continued operation will not present undue risk to the public health and safety.

A-22
50-348/364-CIVP
2/19/92

IEB79-0

APCo Exhibit 22

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USNRC

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OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

Mailing Address
Alabama Power Company
600 North 18th Street
Post Office Box 2641
Birmingham, Alabama 35291
Telephone 205 [illegible]

R. P. McDonald
Senior Vice President
[illegible] Building

Alabama Power

January 28, 1985

Docket Nos. 50-348
50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Units 1 and 2
Environmental Qualification of Electrical Equipment
Response to Generic Letter 84-24

Gentlemen:

Generic Letter 84-24 dated December 27, 1984 requested that Alabama Power Company submit, under oath or affirmation, a certification that: (a) an Environmental Qualification Program is in place and being implemented that satisfies the requirements of 10CFR50.49 within the currently approved schedule for the plant without further extension; (b) the plant has at least one path to safe shutdown using fully qualified equipment, or has submitted a justification for continued safe operation pending full qualification of any equipment not fully qualified; and (c) all other equipment within the scope of 50.49 is either fully qualified or a justification for continued operation has been submitted pending full qualification. In addition, Generic Letter 84-24 stated that the certification described in (a), (b), and (c) above address IE Bulletins and Information Notices that identify environmental qualification problems.

Alabama Power Company has an Environmental Qualification Program in place that satisfies the requirements of 10CFR50.49 as stated in the NRC Safety Evaluations dated December 13, 1984.

The Farley Nuclear Plant - Units 1 and 2 have at least one path to safe shutdown using fully qualified equipment in accordance with 10CFR50.49(b)(1) and (b)(2) as stated in letters dated May 20, 1983 and May 23, 1984 and approved by the NRC Safety Evaluations dated December 13, 1984. Alabama Power Company has developed Master Lists of equipment which require environmental qualification for Farley Nuclear Plant - Units 1 and 2. These Master Lists were developed by a systematic review of design and as-built documentation, the FSAR, Technical Specifications and Emergency Operating Procedures to determine the systems required to

~~850204 0437~~

NUCLEAR REGULATORY COMMISSION

Docket No. 50-348/364-CivP Official Exh. No. 22

In the matter of Alabama Power Company

Staff ______ IDENTIFIED 3:14 p.m. 2/19/92

Applicant ✓ RECEIVED 3:15 p.m. 2/19/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______ DATE 2/19/92

Contractor ______ Witness ______

Other L. Estep

Reporter ______

0605116

perform a safety-related function which includes equipment required to achieve safe shutdown. All of the equipment identified in the Master Lists have been environmentally qualified and, as a result, a justification for continued operation with unqualified equipment is not required.

The only other equipment within the scope of 10CFR50.49 is defined in Section (b)(3). The scope of 10CFR50.49(b)(3) that is subject to the schedule of 10CFR50.49(g) was discussed with justification for continued operation in a letter to the NRC dated February 22, 1984. This scope of equipment is either fully qualified or will be qualified by the end of the Unit 1 sixth and Unit 2 third refueling outages but no later than November 30, 1985 and March 31, 1985 respectively. This completion schedule was approved by the NRC in letters dated April 16, 1984 for Unit 1 and October 21, 1983 for Unit 2, and NRC Safety Evaluations dated December 13, 1984.

In response to I.E. Bulletin No. 82-04 dated January 3, 1983, Alabama Power Company stated that no Bunker Ramo electrical penetrations are installed or planned to be installed in safety-related systems at Farley Nuclear Plant. Responses to IE Information Notices are not required to be submitted to the NRC. However, it is Alabama Power Company policy that all notices are reviewed for applicability to Farley Nuclear Plant and formally documented in the plant files for permanent retention.

If there are any questions, please advise.

Yours truly,

R. P. McDonald

R. P. McDonald

SWORN TO AND SUBSCRIBED BEFORE ME
THIS 28th DAY OF January, 1985

Notary Public

My Commission Expires: 10/27/85

RPM/DHJ:bdv-D6
cc: Mr. L. B. Long
Mr. J. P. O'Reilly
Mr. E. A. Reeves
Mr. W. H. Bradford

0605110

bc: Mr. W. O. Whitt
Mr. H. O. Thrash
Mr. W. G. Hairston, III
Mr. J. D. Woodard
Mr. J. W. McGowan
Mr. C. D. Nesbitt
Mr. R. G. Berryhill
Mr. D. E. Mansfield
Mr. J. A. Ripple
Mr. W. G. Ware
Mr. D. E. Dutton
Mr. B. J. George
Mr. J. R. Crane
Mr. K. C. Gandhi

A-84 50-348/364-CIVP 2/12/92

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 23, 1985

RPM

Licensee Exhibit 84

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OFFICE OF SECRETARY DOCKETING & SERVICE BRANCH

Docket No. 50-364

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Chestron

✓ 8506060592
LC# 1349

KWRD 3/18/87

3/25/87

New R: 4922

Mr. R. P. McDonald
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. McDonald:

SUBJECT: EVALUATION AND STATUS OF LICENSE CONDITIONS FOR JOSEPH M. FARLEY UNIT 2

By letters dated February 8 and October 19, 1982, and January 7, 1983, which superseded the October 19, letter, you requested that certain license conditions be formally closed by the NRC. By letter dated October 22, 1982, you noted that another license condition was satisfied. We have completed our reviews of these submittals.

The enclosure to this letter indicates the current evaluation and status of our review of your submittals relating to the identified license conditions for Facility Operating License No. NPF-8 dated March 31, 1981. No response to this letter is required. However, you may contact the NRC Project Manager, Mr. Edward A. Reeves, at 301-492-7386, should you have any questions.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

JUN 1985
SENIOR V.P.

MILLER - HOT DOCS

-8506060592

NUCLEAR REGULATORY COMMISSION

Docket No. ________ Official Exh. No. 84
In the matter of ALABAMA POWER CO.
Staff ________ IDENTIFIED 2/11/92
Applicant APCO RECEIVED 2/12/92
Intervenor ________ REJECTED ________
Cont'g Off'r ________
Contractor ________ DATE 5-23-85
Other ________ Witness ________
Reporter L Eskp

cc: Mr. W. O. Whitt
Executive Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Mr. Louis B. Long, General Manager
Southern Company Services, Inc.
Post Office Box 2625
Birmingham, Alabama 35202

Houston County Commission
Dothan, Alabama 36301

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N.W.
Washington, DC 20036

Chairman
Houston County Commission
Dothan, Alabama 36301

Robert A. Buettner, Esquire
Balch, Bingham, Baker, Hawthorne,
Williams and Ward
Post Office Box 306
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Resident Inspector
U.S. Nuclear Regulatory Commission
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Columbia, Alabama 36319

State Department of Public Health
ATTN: State Health Officer
State Office Building
Montgomery, Alabama 36104

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
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D. Biard MacGuineas, Esquire
Volpe, Boskey and Lyons
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Washington, DC 20006

Charles R. Lowman
Alabama Electric Corporation
Post Office Box 550
Andalusia, Alabama 36420

Dr. J. Nelson Grace
Regional Administrator - Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 2900
Atlanta, GA 30303

Ira L. Myers, M.D.
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36130

0059476

EVALUATION AND STATUS OF CERTAIN LICENSE CONDITIONS

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2

DOCKET NO. 50-364

INTRODUCTION

Alabama Power Company (APCo) requested that the NRC formally close out ten license conditions which it considers completed. The requests were by letter dated January 7, 1983, which superseded its letter dated October 19, 1982. Also, APCo by letter dated February 8, 1982, requested deletion of the license condition relating to the main steam turbine rotor replacements. By letter dated October 22, 1982, APCo advised the NRC that PAD 3.3 was applicable to subsequent fuel cycles thus satisfying another license condition. Our evaluation of your submittals and status of each of these license conditions follows:

DISCUSSION AND EVALUATION

1. Environmental Qualification (EQ) of Electrical Equipment - License Condition 2.C.(18)

   The license condition required certain remedial actions or alternative actions no later than June 30, 1982. Commission regulation 10 CFR 50.49 negated the June 30, 1982 completion date. By letter dated December 13, 1984, we provided a safety evaluation which concludes that the EQ Program is in compliance with the requirements of 10 CFR 50.49.

   Therefore, License Condition 2.C.(18) has been met.

2. Inspection of Main Steam Turbine Rotor Discs for Cracks or Replacement of Rotors - License Condition 2.C.(19)(d)

   In February 1980, we informed licensees with Westinghouse turbines that stress corrosion cracks were being observed in the keyway and bore regions of low-pressure turbines. Since the mechanisms associated with the initiation and growth of these types of cracks were not well known at that time and because we believed these cracks would increase the probability of disc failure, we requested APCo to perform ultrasonic inspections of the rotors of Unit 1. This unit was inspected during November and December 1980 and found to have significant disc cracks even though the plant had operated for only two fuel cycles (approximately 17,000 hours).

   Because of the similarity of metallurgical and operational characteristics of the turbine discs in Farley Units 1 and 2, we included License Condition 2.C.(19)(d) for Unit 2 to assure that an inspection would be made at the first refueling outage. The inspection would determine if similar cracks occur at an earlier time in machine life.

Thus, License Condition 2.C.(19)(d) required that the low-pressure turbines be inspected for keyway and bore cracks in the turbine discs during the first refueling outage or the turbine discs be replaced. APCo proposed by letter dated February 8, 1982 that these inspections be made on a schedule recommended by the turbine vendor (Westinghouse) using criteria that have been reviewed and approved by the NRC staff. APCo has followed the Westinghouse inspection schedule and criteria for rotor disc inspections since that date. APCo chose to replace the Unit 2 rotors during its first refueling outage (October 1982). During the fifth refueling outage completed April 24, 1984, Unit 1 rotors were again replaced.

Therefore, APCo has met License Condition 2.C.(19)(d).

3. <u>Schedule for Facility to be in Compliance with Regulatory Guide 1.97 Revision 2 - License Condition 2.C.(20)</u>

The license condition required that prior to April 30, 1981, the licensee shall provide a schedule for bringing Unit 2 into compliance with Revision 2 of R.G. 1.97. By letter dated March 30, 1981 APCo provided such a schedule. Subsequently, APCo withdrew the schedule by letter dated November 16, 1982, pending issuance of further NRC guidance. This guidance became a part of Generic Letter 82-33, Supplement 1 NUREG-0737. APCo provided a schedule to us by letter dated March 30, 1984. Subsequently, a Confirmatory Order dated June 12, 1984 was issued to APCO requiring implementation of installation or upgrade requirements with regard to R.G. 1.97 application to Emergency Response Facilities by October 1987.

Therefore, License Condition 2.C.(20) has been superseded by the Confirmatory Order.

4. <u>Upgrading of Emergency Operating Procedures and Operator Training for Transients and Accident (I.C.1) - License Condition 2.C.(21)(a)</u>

Farley Unit 2 was granted an operating license on March 31, 1981, based in part on a pilot monitoring review of some of the emergency operating procedures. The procedures, based on the draft Westinghouse Owners' Group (WOG) guidelines available at the time, were found to be acceptable for a full power license (see Supplement 5 to the SER, NUREG-0117). The SER recognized that the procedures might have to be upgraded when the WOG guidelines were approved by the NRC staff.

On that basis License Condition 2.C.(21)(a) required an upgrade of the emergency operating procedures and associated operator training per NUREG-0737, Item I.C.1, prior to startup following the first refueling outage. However, Generic Letter 82-33, "Supplement 1 to NUREG-0737 Requirements for Emergency Response Capability," issued on December 17, 1982, changed the schedule for items (including Item I.C.1) from industry-wide implementation dates to plant-specific schedules to be negotiated with each licensee. The licensee responded with a proposed integrated schedule on April 15, 1983, which included a commitment to implement procedures based on NRC approved WOG guidelines.

Subsequently, by letters dated August 5, September 22, and December 15, 1983 and April 6, and April 19, 1984 APCo modified several dates in their integrated schedule as a result of negotiations with the NRC staff. A Confirmatory Order was sent to APCo on June 12, 1984 requiring APCo to implement the upgraded EOP's by APCo's commitment date of July 1984.

The requirements of the License Condition 2.C.(21)(a) are fully contained within the scope of Item I.C.1 of Generic Letter 82-33.

Therefore, License Condition 2.C.(21)(a) has been superseded by the Confirmatory Order.

5. Reactor Coolant System Vents (II.B.1) - License Condition 2.C.(21)(b)

This license condition required submittal of a designed description and operating procedures for the reactor coolant system vents by July 1, 1981, and a complete installation by July 1, 1982. Our letter dated November 7, 1983, advised APCo that the implementation schedule has been superseded by 10 CFR 50.44(c)(3)(iii).

Based on our November 7, 1983, letter to APCo License Condition 2.C.(21)(b) has been superseded by NRC regulations.

6. Inadequate Core Cooling Instruments (II.F.2) - License Condition 2.C.(21)(g)

This license condition required that APCo provide detail design information and test results from tests of Farley Unit 1 reactor water level instruments by July 1, 1981. Also the condition further required a planned program to complete instrument development to determine the feasibility of the proposed neutron detector water level instrument by January 1, 1982.

By letter dated June 24, 1981 APCo provided the EPRI test report (part 2 of license condition) of the non-invasive reactor water level instrument completing that part of the license condition. Generic Letter (GL) 82-28 superseded parts 1 and 3 of the license condition. APCo responded to GL 82-28 by letter dated March 10, 1983, which is under NRC staff review.

Therefore, License Condition 2.C.(21)(g) has been met.

7. Analysis of Thermal Mechanical Conditions (II.K.2.13) - License Condition 2.C.(21)(h)(1)

This condition states that prior to January 1, 1982, the licensee is required to submit a detailed analysis of thermal mechanical conditions in the reactor vessel. License Condition 2.C.(21)(h)(1) involves an extended loss of all feedwater, thus, this condition is related to feed and bleed cooling of the core and hence to Unresolved Safety Issue (USI) A-45, Decay Heat Removal. License Condition 2.C.(21)(h)(1) is also related to USI A-49, Pressurized Thermal Shock. The staff will resolve USI's A-45 and A-49 as schedules allow after FY-84.

The staff finds that License Condition 2.C.(21)(h)(1), which is only one part of TMI Action Plan Item II.K.2.13, has been completed by the licensee as part of the submittals of the generic effort by the Westinghouse Owners Group (WOG). In December 1981 WOG submitted to the NRC WCAP 10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants." The NRC staff issued a Safety Evaluation to APCo on June 18, 1984 closing out Item II.K.2.13. However, as stated in the June 18, 1984 letter, should the resolution of USI's A-49 and A-45 result in any changes to the conclusions provided in the Safety Evaluation or require any additional actions related to II.K.2.13, APCo will be notified.

Therefore, License Condition 2.C.(21)(h)(1) has been met.

8. Potential for Voiding in the Reactor Coolant System during Transients (II.K.2.17) - License Condition 2.C.(21)(h)(2)

This condition required that the licensee provided an analysis of the potential for voiding in the reactor coolant system during anticipated transients. Section 22.5 of SER Supplement 5, NUREG-0117 (page 22.5-28) of March 1981 refers to this item in NUREG-0737.

By letter of January 7, 1983, the licensee stated it was in compliance with this requirement by referencing a submittal of April 20, 1981, from the WOG. The staff reviewed that submittal and a supplemental letter from Westinghouse of February 16, 1983. By letter dated January 10, 1984, we advised APCo of acceptance of the Westinghouse transient analysis for Farley Units 1 and 2.

Therefore, License Condition 2.C.(21)(h)(2) is completed.

9. Automatic Trip of Reactor Coolant Pumps (RCP's) (II.K.3) - License Condition 2.C.(21)(i)(2)

This condition related to the B&O Task Force recommendation II.K.3 relating to automatic trip of the RCP's for a small break LOCA. By NRC Generic Letter 83-10d dated February 8, 1983, we established criteria for license considerations based on model comparisons with LOFT test L3-6 results.

In response to this action the licensee provided its plans and schedules in a letter of April 22, 1983, "Generic Letter 83-10d and NUREG-0737 Item II.K.3.5." The plans and schedules include the following:

(1) a generic submittal to be developed by the WOG, and

(2) a plant-specific evaluation to be submittal to NRC within 90 days after licensee receipt of the WOG submittal.

Subsequently, by letter dated April 3, 1984, APCo stated that the WOG submittals are complete by letters OG-110, dated December 1, 1983, and OG-117 dated March 12, 1984. Also, APCo advised that the WOG Emergency Response Guidelines for procedure revisions with an appropriate manual trip of the RCP's resolves all issues associated with this generic issue. The schedules for the plant specific emergency operating procedures is included in Confirmatory Order dated June 12, 1984.

Therefore, License Condition 2.C.(21)(1)(2) has been superseded by the Confirmatory Order relating to schedules for NUREG-0737, Supplement 1 items.

10. Revised Small-Break LOCA Analysis (II.K.3.30) - License Condition 2.C.(21)(1)(4)

This condition required the licensee to respond to another recommendation of the B&O Task Force, item II.K.3. A revised small break LOCA analysis was to be submitted, using the revised model, by January 1, 1982. However, APCo confirmed in letters of March 26 and June 4, 1982 and January 7, 1983 that it is participating in an effort by the WOG to resolve Item II.K.3.30 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The WOG has submitted a revised small break LOCA model to the staff which was approved on May 21, 1985. Therefore, the first part of License Condition 2.C.(21)(1)(4) is considered satisfied.

The second part of this license condition required the licensee to submit plant specific calculations, using the NRC approved revised model, by January 1, 1983. However, the NRC revised the schedule to allow all licensees one year after approval of the WOG model to submit specific calculations to the NRC in NUREG-0737, page 3-179 item (4). The staff will determine conformance to 10 CFR 50.46 limits (per NUREG-0737, page 3-180) at that time.

Therefore, License Condition 2.C(21)(1)(4) in its entirety would be considered satisfied upon completion of NUREG-0737 Item II.K.3.31.

11. Fire Protection - License Condition 2.C.(6)

License Condition 2.C.(6) describes the basic elements of the Farley 2 Fire Protection Plan. As a result License Condition 2.C.(6) will be retained in the license.

12. Masonry Walls - License Condition 2.C.(16)

By letter dated November 19, 1982 APCo revised their October 19, 1982 request to delete License Condition 2.C.(16). APCo requested that License Condition 2.C.(16) be revised rather than deleted. Amendment No. 21 to License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit No. 2, revised 2.C.(16) to require modifications to Masonry Wall 2 CBW-34 prior to startup following the second refueling.

The licensee advised the NRC staff that the modification was completed during the second refueling outage which was completed on October 22, 1983. Therefore, License Condition 2.C.(16) has been met.

13. Use of PAD 3.3 Fuel Performance Code - License Condition 2.C.(19)(a)

The condition required that APCo provide additional evaluations of the Westinghouse fuel performance code, PAD 3.3, to demonstrate its applicability during successive fuel cycles. PAD 3.3 was used in the safety analysis of Farley 2. This code was approved with four restrictions described in our safety evaluation of February 9, 1979 sent to Westinghouse. Three of these restrictions deal with numerical limits and have been complied with. The fourth restriction relates to the use of the PAD 3.3 code for the analysis of fission gas release from uranium dioxide (UO_2) for power increasing conditions during normal operation. This restricting applies to the safety analysis of Farley 2. However, Westinghouse stated that this restriction did not adversely affect the results of the safety analyses performed for Farley. In addition, Westinghouse prepared and submitted a detailed evaluation of this restriction in Addendum 1 (September 1979) to WCAP-8720.

At the time the Farley Safety Evaluation Report, Supplement 5, was issued, our review of Addendum 1 to WCAP-8720 had not yet been completed. However, because the fission gas inventory in the fuel is low during the first cycle of operation, this restriction was not expected to have a significant impact early in core life. For this reason, the fuel thermal design for Farley Unit 2 was found acceptable for the first fuel cycle, but a condition was placed on the operating license to require resolution of the issue prior to subsequent cycles of operation.

By letter dated July 20, 1982, we informed Westinghouse that our review of Addendum 1 to WCAP-8720 had been completed and the report was found to be acceptable for reference in license applications. Our evaluation also concluded that the restriction related to fission gas release was unnecessary and should be eliminated from applications involving the PAD 3.3 code.

By letter dated October 22, 1982 APCo cited the approved version of Addendum 1 to WCAP-8720 (including responses to NRC requests for information) as a basis for the continued applicability of the PAD 3.3 code to successive fuel cycles at Farley Unit 2. Since our approval of Addendum 1 results in a less severe set of restrictions than those applied previously in the PAD 3.3 analysis of the subsequent fuel cycles, we agree that Addendum 1 satisfies our concerns and that License Condition 2.C.(19)(a) has been met.

SUMMARY

We conclude that the licensee responses and action taken to the license conditions, noted above, indicates compliance with the conditions as noted. Facility Operating License NPF-8 for Farley Unit 2 will be so annotated at a future date as administrative changes.

Date: May 23, 1985

PRINCIPAL CONTRIBUTORS:

E. A. Reeves
D. Langford
W. G. Kennedy
W. J. Ross
W. Hazelton
M. Slosson
J. Voglewede

(17) Prior to October 1, 1981, Alabama Power Company shall submit to the NRC the design of a modified containment vent and purge system to reduce the use of the 18-inch purge valves during power operation. Prior to startup following the first refueling, Alabama Power Company shall install the modified system.

(18) Alabama Power Company shall take the following remedial actions, or alternative actions, acceptable to the NRC, with regard to the environmental qualification requirements for Class IE equipment:

(a) Complete and auditable records shall be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," dated December 1979. Such records shall be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance no later than June 30, 1982.

(b) Within 90 days of receipt of the equipment qualification safety evaluation (Appendix B to SER Supplement 6, NUREG-0117), Alabama Power Company shall either (i) provide missing documentation identified in Sections 3.0, 4.2 and 4.3 of the equipment qualification safety evaluation which will demonstrate compliance of the applicable equipment with NUREG-0588, or (ii) commit to corrective actions which will result in documentation of compliance of applicable equipment with NUREG-0588 no later than June 30, 1982.

(c) No later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of NUREG-0588.

(19) Prior to resuming power operation following the first refueling, Alabama Power Company shall:

(a) Provide additional evaluations of the Westinghouse fuel performance code (PAD 3.3) to demonstrate its applicability to fuel burnups during successive fuel cycles.

5-6
50-348/364-CIVP
2/12/92

Staff Exh. 6
DOCKETED
USNRC

'92 MAR 13 P12:03

June 18, 1985

POLICY ISSUE
(Notation Vote)

SECY-85-220

FOR: The Commissioners

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: ENVIRONMENTAL QUALIFICATION PROGRAM ACTIONS RESULTING FROM APRIL 2, 1985 COMMISSION MEETING

PURPOSE: To propose staff actions for completing activities relating to Equipment Qualification.

SUMMARY: This paper presents the Commission with the staff's views in the following areas: policy regarding processing of extension requests, enforcement actions for non-compliance discovered now and after the November 30, 1985 deadline, inspection history, and future inspection planning. The staff recommends that the Commission approve the proposed generic letter, including the proposed enforcement policy, relating to 10 CFR 50.49.

BACKGROUND: At the April 2, 1985 Commission meeting on the status of the environmental qualification (EQ) program for electric equipment (10 CFR 50.49), the Commission directed the staff to propose several courses of action related to compliance with the EQ rule. The specific actions requested were described in a staff requirements memorandum dated April 26, 1985. The following discussion contains the staff recommendations regarding these actions.

DISCUSSION: Extension Requests

In the April 26 Staff Requirements Memorandum, the Commission requested the staff to propose two separate courses of action for handling extension requests: (1) for extension requests received between now and late November 1985 and (2) for extension requests received on or about the November 30, 1985 deadline. The Commission directed that the proposed course of action include an analysis of whether the Commission has the authority, the way the rule is worded now, to take the proposed actions. The staff was also directed to prepare criteria which the Commission may use in determining whether extensions should be granted beyond November 30, 1985 for exceptional cases. Finally, the Commission requested that staff proceed to issue a letter to all utilities encouraging them to inform the Commission at an early date if they foresee an exemption request going beyond November 30, 1985.

CONTACT:
R. Karsch, NRR
x28563

8507090004

NUCLEAR REGULATORY COMMISSION

Docket No. ______ Official Exh. No. 6

In the matter of ALABAMA POWER CO.

Staff ✓ IDENTIFIED 2/1/92

Applicant ______ RECEIVED 3/12/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______

Contractor ______ DATE 6/18/85

Other ______ Witness ______

Reporter L Estep

The portion of 50.49(g) governing deadlines and extension requests states:

> This schedule must establish a goal of final environmental qualification...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 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...In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification.

As the Commission was informed at the April 2 meeting, the staff granted a number of extensions under this delegation. Some of these extensions were to dates between March 31 and November 30, but most were to November 30. Each extension was limited to specifically identified pieces of equipment. It is possible that where these extensions terminate prior to November 30, requests may be received to further extend the date to November 30. The staff believes the delegation in 50.49(g) permits the Director of NRR to act on such requests.

A limited number of plants had deadlines prior to March 31, 1985 and were not in compliance on or before the deadline (Calvert Cliffs 2, Sequoyah 1, Cook 1 and 2, and Kewaunee). Once the non-compliance was identified, they filed extension requests. The Director of NRR does not have the legal authority under 10 CFR 50.49(g) to grant the requests because they were not timely.

In these cases, the staff proposes to review the justifications for continued operation from these plants, and use its enforcement discretion not to take escalated enforcement action if adequate justifications for continued operation are provided and if an extension would have been granted if timely filed.

The issue raised by the two categories of requests to go beyond November 30 identified by the Commission is one of timeliness. In general the staff considers requests from licensees for NRC action to be timely if they are filed sufficiently in advance of the applicable deadline to permit adequate review. Some of the extension requests received by the staff prior to March 31 were marginal in this respect, resulting in extraordinary staff efforts to process such requests prior to expiration of the deadline. With regard to requests for Commission-granted extensions beyond the November 30 deadline, the staff believes that requests filed prior to September 30, 1985 would be timely. This guidance would allow one month for staff review and another month for Commission consideration of the staff's evaluation.

Requests received after September 30 should be considered untimely unless the licensee can demonstrate that the untimeliness was entirely the result of events beyond its control. Where a request is untimely, the licensee would be considered in non-compliance, and the matter should be treated in the enforcement context.

The finding of an "exceptional case" will, for the most part, hinge on plant-specific circumstances. For this reason the staff proposed to review requests for extension beyond November 30, 1985 on a case-by-case basis. The staff will consider the following factors, among others, in evaluating the request and forward its recommendation to the Commission:

1. Has the licensee applied best efforts to complete environmental qualification within the prescribed deadline?

2. Does the failure to meet that deadline result from events beyond the licensee's control?

3. Does the failure to meet the deadline result from test failures, procurement difficulties, or installation problems which the licensee could not have reasonably anticipated?

4. Are there compelling reasons for not requiring a plant shutdown on November 30, assuming that operation beyond that date does not present a safety problem?

5. Has the licensee proposed actions which can be expected to result in full compliance within a reasonable time?

6. Issues such as the integrated scheduling of complex plant improvements, of which electrical equipment qualification may be only a small part, may influence the determination of exceptional "good cause."

In each case the staff would provide the Commission with information regarding these factors and other circumstances bearing on the request. The staff will provide a recommendation for Commission action if the Commission so requests.

The proposed generic letter requested by the Commission (Enclosure 2) provides licensees with the above guidance regarding timeliness and criteria for "exceptional cases." The letter also provides guidance on appropriate licensee actions where instances of non-compliance are discovered. For example, if equipment is found to be unqualified, licensees should report the finding if the condition meets the reporting criteria of 10 CFR 50.72 (Prompt Notification) or 10 CFR 50.73 (Licensee Event Reporting System). The staff will issue the letter promptly upon Commission authorization.

Enforcement

Some plants will be found to be in non-compliance after their deadlines expire. Such non-compliance may be within the control of the licensee e.g., inadequate testing or documentation of qualification. Other non-compliance may not be the fault of the licensee, e.g., new test data invalidating previously accepted results of properly conducted tests. The enforcement scheme should distinguish between these two types.

The enforcement policy should also encourage identification and reporting, if required, of safety problems, and corrective actions. In addition, from a technical standpoint, the licensee's corrective actions and justifications for continued operation must be appropriate. Plant shutdowns may be required depending upon the equipment involved.

The Commission directed the staff to consider the range of enforcement options available recognizing that it is difficult to categorize the anticipated types of non-compliance. Criteria should be established to facilitate the determination of appropriate enforcement actions. These actions should be based on the extent, severity, safety significance, duration and justification for non-compliance. Mitigating factors should also be considered to emphasize prompt licensee detection of non-conformance, and also encourage licensee management involvement. The range of enforcement actions the Commission might take between now and November 30, 1985 as well as proposed actions after November 30, 1985 should be addressed. The staff was directed to consider "civil penalties that result from inspections at a much later date beyond the November 30, 1985 requirement and whether such fines should be retroactive on a daily basis from November 30, 1985 to the date of the non-compliance determination." The staff was directed to present these options, which may include issuance of notices of violation, civil penalties, plant shutdown or other orders for example, to the Commission for consideration.

The staff has developed enforcement guidance as directed by the Commission. This guidance is contained in Enclosure 1. Although only a limited number of reviews have been performed in this area to date, the staff believes it has sufficient experience to provide enforcement guidance now. However, as more experience is gained, the guidance may need to be revised. Accordingly, the staff proposes to provide the guidance to the Regions in the form of an Enforcement Guidance Memorandum (EGM) now and place revised guidance in the Enforcement Policy at a later date. The staff suggests that the policy also be provided in the draft generic letter to all licensees (Enclosure 2).

The guidance developed pertains mainly to the severity level categorization of various violations. The existing policy already contains examples at various severity levels which are adequate to cover violations of 10 CFR 50.49 which might arise in the future (see 10 CFR Part 2, Appendix C, Supplement 1, Reactor Operations). In addition, the mitigating and escalating factors currently in the enforcement policy already provide emphasis on prompt detection and correction of non-compliance and also encourage management involvement. Therefore, the guidance developed supplements the existing policy and should be read in conjunction with it.

The failure of a licensee to demonstrate that equipment is qualified could be cause for significant safety concerns. To determine the severity level of a violation of the environmental qualification requirements, the guidance provides that a number of factors must be considered. These include the extent and number of pieces of equipment involved, the safety significance of and function of the equipment affected, the duration of the failure to qualify the equipment, and the reason for the failure to qualify the equipment. However, if non-compliance occurs as a result of events not within the control of the licensee, such as if new test data invalidates previously accepted results of properly conducted tests, the guidance indicates that enforcement action may not be appropriate.

After the severity level of the violation is established, the escalating or mitigating factors contained in the current enforcement policy (10 CFR Part 2, Appendix C, Section IV.B) will then be evaluated.

The staff is proposing that violations involving equipment identified as unqualified between now and November 30, 1985 be evaluated under this guidance. The staff also proposes that violations involving equipment identified as unqualified after November 30, 1985 be evaluated under this guidance but that daily civil penalties for such violations be considered. Daily civil penalties could be proposed if, after November 30, 1985, unqualified equipment is identified and the licensee knew or clearly should have known that the equipment was not qualified. The enclosed enforcement guidance reflects this approach. If a daily civil penalty is warranted the staff proposes that a penalty of $1,[illegible] per violation per day should be proposed for violations categorized at Severity Level III or higher. Since it may take several months to correct violations, particularly if new tests are needed, this could result in substantial civil penalties. Such fines should also be retroactive to November 30, 1985 or to when data became available demonstrating the equipment was unqualified since daily civil penalties will only be proposed for significant violations in which the licensee knew or clearly should have known that the equipment was unqualified. Such a provision will encourage licensees to maintain control over equipment qualification and quickly correct identified problems. Of course, for significant programmatic breakdowns, the policy provides discretion to issue civil penalties of up to $100,000 per violation per day and the staff will recommend such actions where appropriate.

To achieve the maximum deterrent effect from a daily civil penalty, this aspect of the proposed enforcement policy should be published well in advance of the November 30 deadline. This was done in the case of the prompt notification system deadline and resulted in substantial compliance on or before the deadline in that case. Inclusion of the guidance in the generic letter to licensees will serve this purpose.

Inspection Program

The Commission directed that the staff describe the EQ inspection program, including designation of inspection responsibilities, and also prepare an inspection schedule.

The 10 CFR 50.49 equipment qualification inspection program which started with the Calvert Cliffs inspection in October 1984, is in a pilot phase. The objectives of this phase have been to develop an inspection module, to train the regional inspectors, and obtain an indication as to the status of the industry's implementation of 10 CFR 50.49. The inspection module will have the following objectives:

1. Review of licensee's implementation of the program for meeting 10 CFR 50.49 requirements.

2. Review of the licensee's implementation of SER corrective action commitments.

3. Review of the licensee's implementation of the program for maintaining the qualified status of equipment during the life of the plant.

4. Performance of a physical inspection of equipment to determine that the installations agree with SER commitments and qualification requirements.

The inspection module will be issued in the summer of 1985 as a Temporary Instruction (TI) to the IE Inspection Manual.

The pilot phase is scheduled for completion by January 1, 1986 after approximately two inspections have been conducted at each of the five Regions. Based on the results of the staff's experience to date, a transition of the lead role can be made to the Regions after two pilot inspections have been conducted in each Region. IE is now performing the lead role during the pilot phase with participation from NRR, the Regions and consultants (DOE labs). After a region assumes the lead role, support from IE, NRR and contractors will continue, as necessary. Lead responsibilities include the inspection planning, scheduling and coordination, and issuance of the inspection report following the inspection. Depending on their expertise, each of the participating members of the inspection is assigned areas of responsibility by the lead inspector who then coordinates team activities during the inspection.

Regarding the inspection schedule, as of June 1, 1985, five plants have been inspected, one in each Region. These are: Calvert Cliffs 1, Zion 2, Crystal River 3, Ft. Calhoun, and Rancho Seco. Although the identity of the plants remaining to be inspected during the pilot phase has not been finalized, it is intended that an additional plant in each of the Regions will be inspected before January 1, 1986. Reinspections of one or more of the five plants inspected to date may also be scheduled in that time period. Once the lead role transfers to a Region, a coordinated effort between the Regions and the headquarters offices will be necessary to schedule and support the ongoing inspections. IE will retain responsibility to coordinate headquarters and contractor support. The overall schedule will be dependent on the individual Region abilities and resources, the need and availability of headquarters and contractor support and the extent of followup inspections that will be required to close out open issues. The Regions will be expected to complete initial and most followup inspections by the end of FY 87. The proposed FY 86 and FY 87 budgets for the inspection program take this effort into account.

RECOMMENDATION: Approve the proposed generic letter, including the proposed enforcement policy, relating to 10 CFR 50.49.

William J. Dircks
Executive Director for Operations

Enclosure:
As stated

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. Monday, July 8, 1985.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Friday, June 28, 1985, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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APPENDIX

Guidance for Enforcement Actions Concerning Environmental Qualification of Electrical Equipment

The failure of a licensee to demonstrate that equipment is qualified could be cause for significant safety concern. Such equipment may not have been qualified because, for example, 1) it failed the required qualification test; 2) it could not be qualified based upon available information; 3) although it had previously passed a qualification test, as-installed it was not qualified.

The severity level of a violation of the environmental qualification requirements should be determined in accordance with the examples contained in the General Statement of Policy and Procedures for NRC Enforcement Actions, 10 CFR Part 2, Appendix C, Supplement I, Reactor Operations. To determine the severity level of a violation, a number of factors must be considered. These include the extent and number of pieces of equipment involved, the safety significance of and function of the equipment affected, the duration of the failure to qualify the equipment, and the reason for the failure to qualify the equipment.

If non-compliance occurs as a result of events not within the control of the licensee, such as if new test data invalidates previously accepted results of properly conducted tests, and the licensee identifies the problem, enforcement action should not be taken. If, however, the NRC identifies equipment that is clearly not qualified and the equipment is part of a system designed to prevent or mitigate a serious safety event that would not be able to perform its intended safety function under certain conditions, then the violation should be categorized as at least Severity Level III. If, on the other hand, the NRC identifies equipment for which documentation is insufficient to permit NRC verification of qualification but for which there is a sufficient basis to anticipate that the particular equipment can and will be qualified, the violation should be categorized at Severity Level IV. Examples of such deficiencies in documentation may include: 1) Additional testing or analysis is necessary to fully establish qualification; 2) As-installed configuration differs from test configuration to the extent that additional testing or analysis is necessary to maintain equipment qualification. Violations involving procedures which are not sufficiently adequate to satisfy all 50.49 requirements may also be categorized as Severity Level IV violations.

Programmatic breakdowns will be categorized as at least Severity Level III violations.

After the severity level of the violation is established, the escalating or mitigating factors contained in the current enforcement policy (10 CFR Part 2, Appendix C, Section IV.B) should then be evaluated. That policy has been designed to encourage licensee management involvement, prompt reporting of potential safety problems, and prompt and extensive corrective actions.

Violations involving equipment identified as unqualified between now and November 30, 1985 should be evaluated under this guidance. Violations involving equipment identified as unqualified after November 30, 1985 will also be evaluated under this guidance but daily civil penalties for such violations will also be considered. If a daily civil penalty is warranted a penalty of $1,000 per violation per day will be proposed for violations categorized at Severity Level III or higher. Such fines may also be retroactive to November 30, 1985 or to when data became available demonstrating the equipment was unqualified since daily civil penalties will be proposed for significant violations in which the licensee knew or clearly should have known that the equipment was unqualified. Such a provision is intended to encourage licensees to maintain control over equipment qualification and quickly correct identified problems. For significant programmatic breakdowns, the policy provides discretion to issue civil penalties of up to $100,000 per violation per day and such actions may be proposed where appropriate.

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

TO ALL LICENSEES OF OPERATING REACTORS

Gentlemen:

SUBJECT: INFORMATION RELATING TO THE DEADLINES FOR COMPLIANCE WITH 10 CFR 50.49, ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS (Generic Letter 85-)

The deadline for compliance with the Commission rule 10 CFR 50.49, Envrionmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants is specified in the rule as the date of the second refueling outage after March 31, 1982 or March 31, 1985, whichever was earlier. At the request of the Commission, the staff is requesting that licensees identify, at the earliest possible date, the need for any extension of currently applicable deadline extensions beyond November 30, 1985.

Requests for extensions received after September 30, 1985, will be considered non-timely submittals and will be acted upon at the staff's discretion. Extension requests proposing to go beyond November 30, 1985 will be referred to the Commission and the current extensions will only be further extended in "exceptional cases" as stated in 10 CFR 50.49(g). The basis for any extension request beyond November 30, 1985 must clearly identify the exceptional nature of the case, e.g.:

1. Has the licensee applied best efforts to complete environmental qualification within the prescribed deadline?

2. Does the failure to meet that deadline result from events beyond the licensee's control?

3. Does the failure to meet the deadline result from test failures, procurement difficulties, or installation problems which the licensee could not have reasonably anticipated?

4. Are there compelling reasons for not requiring a plant shutdown on November 30, assuming that operation beyond that date does not present a safety problem?

5. Has the licensee proposed actions which can be expected to result in full compliance within a reasonable time?

6. Issues such as the integrated scheduling of complex plant improvements, of which electrical equipment qualification may be only a small part, may influence the determination of exceptional "good cause."

For equipment on which the equipment qualification deadline is passed, (i.e., all equipment not currently covered by an approved extension) and which is discovered (through new test results, NRC inspection, or other means) to be in non-compliance or suspected to be in non-compliance with the requirements of 10 CFR 50.49, licensees should report the finding if the condition meets the reporting criteria of 10 CFR 50.72 (Prompt Notification) or 10 CFR 50.73 (Licensee Event Reporting System). Evaluations of the significance of and corrective action for all potential non-compliances should be documented and retained in appropriate licensee files. If equipment addressed in the plant Technical Specifications is found to be unable to perform its intended function during an accident because of equipment qualification problems, the licensee would be expected to follow the provisions of the Technical Specifications. Decisions relating to enforcement actions will be based on the circumstances relating to non-compliance, the discovery and reporting to NRC of the non-compliance and its safety significance as discussed in the enclosed enforcement policy supplement.

This letter does not require any response and therefore does not need approval of the Office of Management and Budget. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D.C. 20503. Should you have any questions, the staff contact is Rudy Karsch. Mr. Karsch can be reached on (301)492-8563.

Hugh L. Thompson, Jr.

Hugh L. Thompson, Jr., Director
Division of Licensing

Enclosure: As stated

S-7
50-348/364-CIVP
2/12/92

Staff Exh. 7

DOCKETED
USNRC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555

'92 MAR 13 P12:02

AUG 6 1985

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

TO ALL LICENSEES OF OPERATING REACTORS

Gentlemen:

SUBJECT: INFORMATION RELATING TO THE DEADLINES FOR COMPLIANCE WITH 10 CFR 50.49, "ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT IMPORTANT TO SAFETY FOR NUCLEAR POWER PLANTS" (GENERIC LETTER 85-15)

The deadline for compliance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants" is specified in the rule as the date of the second refueling outage after March 31, 1982 or March 31, 1985, whichever was earlier. Some plants have received extensions to these deadlines up to November 30, 1985. Where current extensions terminate prior to November 30, 1985, the delegation in 10 CFR 50.49(g) permits the Director of NRR to act on further requests for extensions as long as the new deadline is not beyond November 30, 1985. Section 50.49(g) states that "in exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification." The purpose of this letter is to advise licensees that it is the Commission's intention that extensions will be granted only in rare circumstances and that enforcement action will be taken against licensees that continue to operate their plants with unqualified equipment 1/ beyond November 30, 1985, without extensions approved by the Commission.

It is the Commission's intention that licensees which are not in compliance on November 30, 1985, and which have not been given extensions either will have to either shut down or, if they have valid staff-approved justifications for continued operation, select to operate and face civil penalties of $5,000 per item 2/ per day for each day after November 30, 1985, on which a licensee operates in noncompliance with the rule. For noncompliance identified after November 30, 1985, such fines may be made retroactive to November 30, 1985 for each day a licensee clearly knew, or should have known, that equipment qualification was incomplete. Some mitigation of any penalty may be considered based upon satisfaction of the following factors:

1/ For purposes of enforcement, "unqualified equipment" means equipment for which there is not adequate documentation to establish that this equipment will perform its intended functions in the relevant environment.

2/ An item is defined as a specific type of electrical equipment, designated by manufacturer and model, which is representative of all identical equipment in a plant area exposed to the same environmental service conditions.

8508070178

NUCLEAR REGULATORY COMMISSION

Docket No. ______ Official Exh. No. 7

In the matter of ALABAMA POWER CO.

Staff ✓ IDENTIFIED 3/11/92

Applicant ______ RECEIVED 3/12/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______

Contractor ______ DATE 8/6/85

Other ______ Witness ______

Reporter ______

1. Did the licensee identify and promptly report the noncompliance with 10 CFR 50.49?

2. Did the licensee apply best efforts to complete environmental qualification within the deadline?

3. Has the licensee proposed actions which can be expected to result in full compliance within a reasonable time?

For equipment which is discovered (through new test results, NRC inspection, or other means) after November 30, 1985, to be in noncompliance or is suspected to be in noncompliance with the requirements of 10 CFR 50.49, licensees should report the finding if the condition found meets the reporting criteria of 10 CFR 50.72 (Prompt Notification) or 10 CFR 50.73 (Licensee Event Reporting System). Evaluations of the significance of and corrective action for all actual and potential noncompliances should be documented as should the circumstances of discovery of the noncompliance or suspected noncompliance. These documents should be retained in appropriate licensee files. If equipment addressed in the plant Technical Specifications is found to be unable to perform its intended function during an accident because of equipment qualification problems, the licensee is required to follow the provisions of the Technical Specifications. A case by case determination will be made whether retroactive enforcement is appropriate for noncompliance identified after November 30, 1985.

Licensees desiring an extension beyond November 30, 1985, must submit an extension request at the earliest possible date to the Commission with a copy to the Director, NRR and the Director, IE. Requests received after September 30, 1985, will be considered untimely, and may be denied on that basis. The basis for any extension request beyond November 30, 1985 must clearly identify the exceptional nature of the case, e.g., why, through events entirely beyond its control, the licensee will not be in compliance with the rule on November 30; the date when compliance will be achieved; and a justification for continued operation until compliance will be achieved.

This letter does not require any response and therefore does not need approval of the Office of Management and Budget. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management Room 3208, New Executive Office Building, Washington, D.C. 20503. Should you have any questions, the staff contacts are Gary Holahan for technical questions and Jane Axelrad for enforcement questions. Mr. Holahan can be reached on (301)492-7415 and Ms. Axelrad can be reached on (301)492-4909.

Hugh L. Thompson

Hugh L. Thompson, Jr., Director
Division of Licensing

cc: List of Generic Letters

JAR ~~JAK~~
JWM
RPM
NGDC
~~A330~~

~~WGW-FNP~~ A-73
50-348/364-CIVP
2/20/92

APCo Exhibit 73

File IEN 86-03

SSINS No.: 6835
IN 86-03

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

January 14, 1986

IE INFORMATION NOTICE NO. 86-03: POTENTIAL DEFICIENCIES IN ENVIRONMENTAL QUALIFICATION OF LIMITORQUE MOTOR VALVE OPERATOR WIRING

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This notice is provided to alert recipients of potential generic problems regarding the environmental qualification of electrical wiring used in Limitorque motor valve operators. It is expected that recipients will review this information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem occurring at their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On September 30, 1985, Commonwealth Edison (Zion Generating Station) reported to the NRC that it had discovered four Limitorque motor valve operators with jumper wires different from those tested by Limitorque in its environmental qualification program. Subsequently, the Tennessee Valley Authority (TVA) notified the NRC that the manufacturer of the internal control wiring of its Limitorque operators at Sequoyah Nuclear Plant either could not be identified or qualification could not be established where the manufacturer was known. Similar circumstances have recently been identified at other nuclear plants.

Discussion:

The results of NRC inspections at Limitorque and TVA have determined that even though Limitorque has conducted environmental qualification testing of motor valve operators, the qualification test reports do not specifically address wiring or wiring qualification. Limitorque has installed wires from several different manufacturers in safety-related operators.

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NUCLEAR REGULATORY COMMISSION

Docket No. 50-348/364-CivP Official Exh. No. 73

In the matter of Alabama Power Company

Staff ______ IDENTIFIED 3:40 pm, 2/20/92

Applicant ✓ RECEIVED 3:41 pm, 2/20/92

Intervenor ______ REJECTED ______

Cont's Off'r ______

Contractor ______ DATE 2/20/92

Other ______ Witness ______

Reporter L. Eter

not true

0070235

Limitorque stated that it can provide or reference documentation to support qualification of wires it has installed; [redacted], licensees, and/or others may have added additional wires that are not qualified by this data. The NRC physical inspection of Limitorque operators at the Sequoyah plant determined that some valve operators contained wires not qualified by the Limitorque tests.

The NRC staff considers the resolution of this issue to be part of the licensee's environmental qualification program to establish and maintain the qualified status of electrical equipment within the scope of 10 CFR 50.49.

Because qualification of Limitorque installed wiring may be based on separate qualification tests of wire, it is important that adequate analyses have been performed and that documentation exists to demonstrate that the separate wire qualification tests encompass the parameters for the valve operator qualification.

If additional wiring has been added or replaced after operator shipment from Limitorque, then additional documentation may be appropriate for establishing qualification of the additional wires and subsequently the valve operators for the valve operator specific application.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office, or this office.

E L Jordan
Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Contact: G. Hubbard, IE
(301) 492-9759

Attachment: List of Recently Issued IE Information Notices

0070236

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
86-02	Failure Of Valve Operator Motor During Environmental Qualification Testing	1/6/86	All power reactor facilities holding an OL or CP
86-01	Failure Of Main Feedwater Check Valve Causes Loss Of Feedwater System Integrity And Water-Hammer Damage	1/6/86	All power reactor facilities holding an OL or CP
85-101	Applicability of 10 CFR 21 To Consulting Firms Providing Training	12/31/85	All power reactor facilities holding an OL or CP
85-100	Rosemount Differential Pressure Transmitter Zero Point Shift	12/31/85	All power reactor facilities holding an OL or CP
85-99	Cracking In Boiling-Water-Reactor Mark I And Mark II Containments Caused By Failure Of The Inerting System	12/31/85	All BWR facilities having a Mark I or Mark II containment
85-98	Missing Jumpers From Westinghouse Reactor Protection System Cards For The Over-Power Delta Temperature Trip Function	12/26/85	All Westinghouse designed PWR facilities holding an OL or CP
85-97	Jail Term For Former Contractor Employee Who Intentionally Falsified Welding Inspection Records	12/26/85	All power reactor facilities holding an OL or CP
85-96	Temporary Strainers Left Installed In Pump Suction Piping	12/23/85	All power reactor facilities holding an OL or CP
85-95	Leak Of Reactor Water To Reactor Building Caused By Scram Solenoid Valve Problem	12/23/85	All BWR facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

S-23
50-348/364-CivP
2/12/92

Staff Exh. 23

NUREG-0588
Rev. 1

DOCKETED
USNRC

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

'92 MAR 13 P12:01

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BRANCH

Including Staff Responses to Public Comments

Resolution of Generic Technical Activity A-24

A. J. Szukiewicz, Task Manager

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

8108310591

NUCLEAR REGULATORY COMMISSION

Docket No. ______ Official Exh. No. 23

In the matter of ALABAMA POWER CO.

Staff ✓ IDENTIFIED 2/12/92

Applicant ______ RECEIVED 2/12/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______

Contractor ______ DATE N/D

Other ______ Witness ______

Reporter L Estep

NUREG-0588
Rev. 1

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

Including Staff Responses to Public Comments

Resolution of Generic Technical Activity A-24

Manuscript Completed: November 1980
Date Published: July 1981

A. J. Szukiewicz, Task Manager

Division of Safety Technology
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

This report provides the NRC interim staff positions on selected areas of environmental qualification of electrical equipment important to safety.

Part I of this report contains the original "For Comment" NUREG, which was published in December 1979. This "For Comment" issue is now endorsed by the Commission, in its May 23, 1980 Memorandum and Order (CLI-80-21) as the staff's interim position, until the final positions, currently being developed in rulemaking, are established.

Part II of this report contains the staff's responses to and resolution of the public comments that were solicited and received before May 1, 1980. Revision 1 of Appendices A through D identifies the additions, modifications, and/or corrections that were made as a result of these comments.

This report completes the staff resolution of the Generic Technical Activity A-24, "Qualification of Class IE Safety-Related Equipment." The information contained in Part I and Part II will be considered and used, in part, by the staff in developing the final positions during rulemaking.

CONTENTS

ACKNOWLEDGEMENT

Many NRC individuals participated in the development of the positions on environmental qualifications of electrical equipment important to safety presented herein. The contributions of the following individuals were particularly helpful in developing the staff responses to the public comments and are acknowledged:

A. J. Szukiewicz
J. Kudrick
L. Ruth
F. M. Akstulewicz
T. Quay

INTRODUCTION

NUREG-0588 was issued for comment in December 1979, to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews for new as well as for the older vintage plants (that is, near term operating license plants). The positions contained in the report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying the equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

The positions in the report do not address all areas of qualification and are intended only to supplement, in selected areas of the qualification issue, the requirements described in the 1971 and the 1974 versions of IEEE Standard 323.

On May 23, 1980, a Commission Memorandum and Order (CLI-80-21) endorsed the positions in the "For Comment" NUREG as the interim positions that shall be satisfied (in order to verify conformance to General Design Criterion No. 4 in Appendix A of 10 CFR 50) until the "final" positions are established in rule-making. The staff is currently developing these positions for rulemaking, and anticipates that the proposed rule (that is, the "final positions") will be issued for public comment in December 1981.

As a result of the above referenced memorandum and order, and the ongoing rule-making activity, the positions developed in the "For Comment" NUREG report have not been modified to reflect the public comments. Staff responses to the public comments and revisions to Appendices A through D are provided in Part II of this report, however. The revised appendices identify additions, modifications, and/or corrections believed necessary to resolve the public comments. The revised appendices are included to provide additional information and guidance to industry and to provide insight into the topics to be considered during rulemaking.

Certain modifications and clarifications to the positions as a result of the TMI-2 event are anticipated, as, for example, in radiation source term requirements described in the staff responses to some of the public comments. In the interim, however, and until the final rule is established, the staff requires that all plants licensed after May 23, 1980 conform to NUREG-0588. In accordance with Regulatory Guide 1.89, all Operating Licenses for facilities whose Construction Permit SER is dated July 1, 1974, or later will be reviewed against IEEE Standard 323-1974. Thus for these licensees, the Operating License applicant is required to qualify equipment to the Category I requirements in NUREG-0588. For Operating Licenses issued after May 23, 1980, whose Construction Permit SER is dated before July 1, 1974, the Operating License applicant is required to qualify equipment to at least Category II requirements in NUREG-0588--unless the licensee made commitments in the Construction Permit application to use the 1974 standard, or unless the Operating License application indicates that the 1974 standard is to be used. In such cases, Category I requirements of NUREG-0588 are to be used. In addition, all parts used to replace installed equipment shall also be qualified to the Category I requirements unless adequate bases are established to justify exceptions.

All reactors with Operating Licenses as of May 23, 1980 will be evaluated by the staff against the DOR guidelines (Division of Operating Reactors - "Guidelines

Part

Part I

NUREG-0588

For Comment

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

December 1979

NUREG-0588
For Comment

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

Resolution of Generic Technical Activity A-24

Manuscript Completed: August 1979
Date Published: December 1979

A. J. Szukiewicz, Task Manager

Division of Systems Safety
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

INTRODUCTION

NUREG-0588 was issued for comment in December 1979, to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews for new as well as for the older vintage plants (that is, near term operating license plants). The positions contained in the report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying the equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

The positions in the report do not address all areas of qualification and are intended only to supplement, in selected areas of the qualification issue, the requirements described in the 1971 and the 1974 versions of IEEE Standard 323.

On May 23, 1980, a Commission Memorandum and Order (CLI-80-21) endorsed the positions in the "For Comment" NUREG as the interim positions that shall be satisfied (in order to verify conformance to General Design Criterion No. 4 in Appendix A of 10 CFR 50) until the "final" positions are established in rule-making. The staff is currently developing these positions for rulemaking, and anticipates that the proposed rule (that is, the "final positions") will be issued for public comment in December 1981.

As a result of the above referenced memorandum and order, and the ongoing rule-making activity, the positions developed in the "For Comment" NUREG report have not been modified to reflect the public comments. Staff responses to the public comments and revisions to Appendices A through D are provided in Part II of this report, however. The revised appendices identify additions, modifications, and/or corrections believed necessary to resolve the public comments. The revised appendices are included to provide additional information and guidance to industry and to provide insight into the topics to be considered during rulemaking.

Certain modifications and clarifications to the positions as a result of the TMI-2 event are anticipated, as, for example, in radiation source term requirements described in the staff responses to some of the public comments. In the interim, however, and until the final rule is established, the staff requires that all plants licensed after May 23, 1980 conform to NUREG-0588. In accordance with Regulatory Guide 1.89, all Operating Licenses for facilities whose Construction Permit SER is dated July 1, 1974, or later will be reviewed against IEEE Standard 323-1974. Thus for these licensees, the Operating License applicant is required to qualify equipment to the Category I requirements in NUREG-0588. For Operating Licenses issued after May 23, 1980, whose Construction Permit SER is dated before July 1, 1974, the Operating License applicant is required to qualify equipment to at least Category II requirements in NUREG-0588--unless the licensee made commitments in the Construction Permit application to use the 1974 standard, or unless the Operating License application indicates that the 1974 standard is to be used. In such cases, Category I requirements of NUREG-0588 are to be used. In addition, all parts used to replace installed equipment shall also be qualified to the Category I requirements unless adequate bases are established to justify exceptions.

All reactors with Operating Licenses as of May 23, 1980 will be evaluated by the staff against the DOR guidelines (Division of Operating Reactors - "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors," dated November 13, 1979). In cases where the DOR guidelines do not provide sufficient detail but NUREG-0588 Category II does, NUREG-0588 will be used.

As noted in the "For Comment" report, seismic qualification is currently being pursued under the equipment qualification program plan and is outside the scope of this document. The staff is also pursuing rulemaking activities in the seismic qualification area and anticipates issuing a proposed rule for public comment in 1982.

Part

Part I

NUREG-0588

For Comment

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

December 1979

NUREG-0588
For Comment

Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment

Resolution of Generic Technical Activity A-24

Manuscript Completed: August 1979
Date Published: December 1979

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ABSTRACT

This report provides the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment in the resolution of Generic Technical Activity A-24, "Qualification of Class 1E Safety-Related Equipment". The positions herein are applicable to plants that are or will be in the construction permit (CP) or operating license (OL) review process and that are required to satisfy the requirements set forth in either the 1971 or the 1974 version of IEEE Standard 323. These positions were developed prior to the Three Mile Island Unit 2 event. Any recommendations resulting from the staff's review of that event will be provided later. The seismic qualification requirements are addressed elsewere and are not included in the scope of this document.

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ACKNOWLEDGMENT

Many NRC individuals participated in the development of the positions on environmental qualification of safety-related electrical equipment presented herein. The contributions of the following individuals were particularly helpful in developing this report and are acknowledged:

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INTERIM STAFF POSITION ON ENVIRONMENTAL
QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

INTRODUCTION

Equipment that is used to perform a necessary safety function must be capable of maintaining functional operability under all service conditions postulated to occur during the installed life for the time it is required to operate. This requirement, which is embodied in General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside containment. More detailed guidance related to the methods, procedures and guidelines for demonstrating this capability has been set forth in IEEE Std. 323 and ancillary daughter standards (e.g., IEEE Stds. 317, 334, 382, 383) and has been endorsed with supplementary material as noted in NRC Regulatory Guides.

As part of the operating license review for each plant, the staff evaluates the applicant's equipment qualification program by reviewing the qualification documentation on selected safety-related equipment. The objective of this review is to provide reasonable assurance that the equipment can perform its intended function in the most limiting environment in which it is expected to function.

The staff review of the documentation submitted by both equipment suppliers and license applicants indicate that some have developed generally acceptable qualification programs. The efforts of others, as compared with the "state of the art," need improvements. This is due in part to the fact that the qualification requirements contained in national standards and other guidance related to equipment qualification have been evolutionary in nature and subject to diverse interpretation.

To promote more orderly and systematic implementation of equipment qualification programs in industry and to provide guidance to be used by the NRC staff for use in the ongoing licensing reviews, the staff has developed a number of positions on selected areas of the qualification issue. These positions, which are presented in this report, provide guidance on the establishment of service conditions, methods for qualifying equipment, and other related matters. They do not address in detail all areas of qualification, since certain areas are not yet well understood and are the subjects of research studies conducted by the NRC and by the industry. For example, the effects of aging, sequential versus simultaneous testing, including synergistic effects, and the potential combustible gas and chloride formation in equipment containing organic materials are being evaluated. It is expected that these studies will lead to the development of more detailed guidance in the future, and may require changes to these positions.

These positions were developed prior to the staff completion of the TMI-2 event evaluation, and any additional requirements or modifications to these positions as a result of this evaluation will be identified later. In addition, seismic qualification is being pursued on a case-by-case basis by the Seismic Qualification Review Team (SQRT) and is outside the scope of this document.

These positions are applicable only to plants that are or will be in the construction permit or operating license review process. These positions do not apply to operating plants. Operating plant licensees have been required by the NRC Office of Inspection and Enforcement to reassess the qualification of safety-related equipment used in those facilities (see IE Bulletin 79-01). Licensee responses are to be evaluated using guidelines being developed specifically for that effort.

DISCUSSION

As part of the staff reviews of operating license applications, a number of positions have been developed on the methods and procedures used to environmentally qualify safety-related electrical equipment. These positions, which are described in the following sections of this report, supplement the requirements found in the 1971 and the 1974 version of IEEE Standard 323*. While alternatives to these positions may be proposed, the positions will be used, together with the standards, as the basis for reviewing all license applications.

The positions are divided into two categories. Category I positions apply to equipment qualified in compliance with IEEE Std. 323-1974 and Category II positions apply to equipment qualified in compliance with IEEE Std. 323-1971.

Section 1 of the the following table contains positions related to the establishment of the service conditions for areas inside and outside containment to which equipment should be qualified. It includes guidance for calculating the pressure and temperature conditions that result from a high energy line break (LOCA and/or MSLB), and also provides guidance for determining the chemical spray and the radiation environments expected to occur during a design basis event condition. Section 2 provides guidance on the selection of qualification methods (that is, testing, analysis, etc.) to be used for equipment located inside and outside containment. Sections 3, 4, and 5 provide guidance on the selection of margins, aging and the preparation of qualification documentation. The appendices supplement the positions and identify specific codes, sample calculations, and procedures that should be used when qualifying equipment. The term "equipment" referred to in the following sections applies to safety-related electrical equipment required for accident mitigation, post-incident monitoring, and safe shutdown.

It should be noted that, although the intent of these positions is to define criteria related to electrical equipment, it is necessary to recognize and address equipment interfaces (e.g., mounting, seals, terminations) in the qualification process to which these positions apply. Also, qualification programs for specific equipment, such as cables, valves, motors, and electrical penetrations, that are designed to conform with the requirements of the daughter standards of IEEE Std. 323-1974 (as endorsed by the NRC Regulatory Guides) are acceptable for demonstrating compliance with the objectives of IEEE Std. 323. The daughter standards include standards such as IEEE Std. 383 for cables,

* IEEE Std. 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

IEEE Std. 323-1971, "IEEE Trial Use Standard: General Guide for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

IEEE Std. 382 for valves, IEEE Std. 334 for motors, and IEEE Std. 317 for electrical penetrations. These standards are endorsed by Regulatory Guides 1.131, 1.73, 1.40, and 1.63 respectively.

INTERIM STAFF POSITION ON ENVIRONMENTAL
QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT

CATEGORY I

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974

1. ESTABLISHMENT OF THE QUALIFICATION PARAMETERS FOR DESIGN BASIS EVENTS

1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)

(1) The time-dependent temperature and pressure, established for the design of the containment structure and found acceptable by the staff, may be used for environmental qualification of equipment.

(2) Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.

Pressurized Water Reactors (PWRs)

Dry Containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP) Section 6.2.1.1.A, NUREG-75/087.

Ice Condenser Containment - Calculate LOCA containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.B, NUREG-75/087.

Boiling Water Reactors (BWRs)

Mark I, II and III Containment - Calculate LOCA environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.C, NUREG-75/087.

(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser types of plants, the generic envelope shown in Appendix C may be used for qualification testing.

(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.

CATEGORY II

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971

1. ESTABLISHMENT OF THE QUALIFICATION PARAMETERS FOR DESIGN BASIS EVENTS

1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)

(1) Same as Category I.

(2) Same as Category I.

Pressurized Water Reactors (PWRs)

Dry Containment - Use the same containment models as in Category I. The assumption of partial revaporization will be allowed. Other assumptions that reduce the temperature response of the containment will be evaluated on a case-by-case basis.

Ice Condenser Containment - Same as Category I.

Boiling Water Reactors (BWRs)

Same as Category I.

(3) Same as Category I.

(4) Same as Category I.

CATEGORY I

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974

1.2 Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)

(1) The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff.

(2) Models that are acceptable for calculating containment parameters are listed in Section 1.1(2).

(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.

(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.

(5) Where qualification has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:

(a) Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in Section 1.1(2).

(b) Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.

(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) requalification testing be performed with appropriate margins, or (ii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For plants that

CATEGORY II

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971

1.2 Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)

(1) Where qualification has not been completed, the environmental parameters used for equipment qualification should be calculated using a plant-specific model based on the staff-approved assumptions discussed in item 1 of Appendix B.

(2) Other models that are acceptable for calculating containment parameters are listed in Section 1.1(2).

(3) Same as Category I.

(4) Same as Category I.

(5) Where qualification has been completed but only LOCA conditions were considered, then it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:

(a) Calculate the peak temperature from an MSLB using a model based on the staff's approved assumptions discussed in item 1 of Appendix B.

(b) Same as Category I Section 1.2(5)(b).

(c) If the calculated surface temperature exceed the qualification temperature, the staff requires that (i) additional justification be provided to demonstrate that the equipment can maintain its required functional operability if its surface temperature reaches the calculated value or (ii) requalification testing be performed with appropriate margins, or (iii) qualified physical protec-

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
are currently being reviewed, or will be submitted for an operating license review within six months from issue date of this report, compliance with items (i) or (ii) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated.	tion be provided to assure that the surface temperature will not exceed the actual qualification temperature.
1.3 Effects of Chemical Spray The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines.	1.3 Effects of Chemical Spray Same as Category I.
1.4 Radiation Conditions Inside and Outside Containment The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life. The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualification. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used. (1) The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design	1.4 Radiation Conditions Inside and Outside Containment Same as Category I.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971

basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable.

(2) The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures.

(3) The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate.

(4) Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section.

(5) Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment.

(6) For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and

CATEGORY I	CATEGORY II
Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971

shielding of the equipment reduces the dose and dose rate.

(7) For unshielded equipment, the beta doses at the surface of the equipment should be the sum of the airborne and plate-out sources. The airborne beta dose should be taken as the beta dose calculated for a point at the containment center.

(8) Shielded components need be qualified only to the gamma radiation levels required, provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to beta radiation or that the effects of beta radiation heating and ionization have no deleterious effects on component performance.

(9) Cables arranged in cable trays in the containment should be assumed to be exposed to half the beta radiation dose calculated for a point at the center of the containment plus the gamma ray dose calculated in accordance with Section 1.4(6). This reduction in beta dose is allowed because of the localized shielding by other cables plus the cable tray itself.

(10) Paints and coatings should be assumed to be exposed to both beta and gamma rays in assessing their resistance to radiation. Plate-out activity should be assumed to remain on the equipment surface unless the effects of the removal mechanisms, such as spray wash-off or steam condensate flow, can be justified and quantified by analysis or experiment.

(11) Components of the emergency core cooling system (ECCS) located outside containment (e.g., pumps, valves, seals and electrical equipment) should be qualified to withstand the radiation equivalent to that penetrating the containment, plus the exposure from the sump fluid using assumptions consistent with the requirements stated in Appendix K to 10 CFR Part 50.

(12) Equipment that may be exposed to radiation doses below 10^4 rads should not be considered to be exempt from radiation qualification, unless analysis supported by test data is provided to verify that these levels will not degrade the operability of the equipment below acceptable values.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
(13) The staff will accept a given component to be qualified provided it can be shown that the component has been qualified to integrated beta and gamma doses which are equal to or higher than those levels resulting from an analysis similar in nature and scope to that included in Appendix D (which uses the source term given in item (1) above), and that the component incorporates appropriate factors pertinent to the plant design and operating characteristics, as given in these general guidelines.	
(14) When a conservative analysis has not been provided by the applicant for staff review, the staff will use the radiation environment guidelines contained in Appendix D, suitably corrected for the differences in reactor power level, type, containment size, and other appropriate factors.	
1.5 Environmental Conditions for Outside Containment	1.5 Environmental Conditions for Outside Containment
(1) Equipment located outside containment that could be subjected to high-energy pipe breaks should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental parameters described in Sections 1.1 through 1.4 above should be applied.	(1) Equipment located outside containment that could be subjected to high-energy pipe breaks should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental parameters described in Sections 1.1 through 1.4 (Category II) above should be applied.
(2) Equipment located in general plant areas outside containment where equipment is not subjected to a design basis accident environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.	(2) Same as Category I.
(3) Equipment not served by Class 1E environmental support systems, or served by Class 1E support systems that may be secured during plant operation or shutdown, should be qualified to the limiting environmental conditions that are postulated for that location, assuming a loss of the environmental support system.	(3) Same as Category I; or, there may be designs where a loss of the environmental support system may expose some equipment to environments that exceed the qualified limits. For these designs, appropriate monitoring devices should be provided to alert the operator that abnormal conditions exist and to permit an assessment of the conditions that occurred in order to determine if corrective action, such as replacing any affected equipment, is warranted.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
2. QUALIFICATION METHODS	2. QUALIFICATION METHODS
2.1 Selection of Methods	2.1 Selection of Methods
(1) Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.	(1) Qualification methods should conform to the requirements defined in IEEE Std. 323-1971.
(2) The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.	(2) Same as Category I.
(3) The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.	(3) Same as Category I.
(a) Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.	
(b) Any equipment (safety-related or non-safety-related) that need not function in order to mitigate any accident, but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.	
(c) Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its non-accident service environment.	

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
Although actual type testing is preferred, other methods when justified may be found acceptable. The bases should be provided for concluding that such equipment is not required to function in order to mitigate any accident, and that its failure in any mode in any accident environment is not detrimental to plant safety.	
(4) For environmental qualification of equipment subject to events other than a DBA, which result in abnormal environmental conditions, actual type testing is preferred. However, analysis or operating history, or any applicable combination thereof, coupled with partial type test data may be found acceptable, subject to the applicability and detail of information provided.	(4) Same as Category I.
2.2 Qualification by Test	2.2 Qualification by Test
(1) The failure criteria should be established prior to testing.	(1) Same as Category I.
(2) Test results should demonstrate that the equipment can perform its required function for all service conditions postulated (with margin) during its installed life.	(2) Same as Category I.
(3) The items described in Section 6.3 of IEEE Std. 323-1974 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.	(3) The items described in Section 5.2 of IEEE Std. 323-1971 supplemented by items (4) through (12) below constitute acceptable guidelines for establishing test procedures.
(4) When establishing the simulated environmental profile for qualifying equipment located inside containment, it is preferred that a single profile be used that envelopes the environmental conditions resulting from any design basis event during any mode of plant operation (e.g., a profile that envelopes the conditions produced by the main steamline break and loss-of-coolant accidents).	(4) Same as Category I.
(5) Equipment should be located above flood level or protected against submergence by locating the equipment in qualified watertight enclosures. Where equipment is located in watertight enclosures, qualification by test or analysis should be used to demonstrate the adequacy of such protection. Where equipment could be submerged, it should be identified and demonstrated to be qualified by test for the duration required.	(5) Same as Category I.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
(6) The temperature to which equipment is qualified, when exposed to the simulated accident environment, should be defined by thermocouple readings on or as close as practical to the surface of the component being qualified.	(6) Same as Category I. If there were no thermocouples located near the equipment during the tests, heat transfer analysis should be used to determine the temperature at the component. (Acceptable heat transfer analysis methods are provided in Appendix B.)
(7) Performance characteristics of equipment should be verified before, after, and periodically during testing throughout its range of required operability.	(7) Same as Category I.
(8) Caustic spray should be incorporated during simulated event testing at the maximum pressure and at the temperature conditions that would occur when the onsite spray systems actuate.	(8) Same as Category I.
(9) The operability status of equipment should be monitored continuously during testing. For long-term testing, however, monitoring at discrete intervals should be justified if used.	(9) Same as Category I.
(10) Expected extremes in power supply voltage range and frequency should be applied during simulated event environmental testing.	(10) Same as Category I.
(11) Dust environments should be addressed when establishing qualification service conditions.	(11) Same as Category I.
(12) Cobalt-60 is an acceptable gamma radiation source for environmental qualification.	(12) Same as Category I.
2.3 Test Sequence	2.3 Test Sequence
(1) The test sequence should conform fully to the guidelines established in Section 6.3.2 of IEEE Std. 323-1974. The test procedures should insure that the same piece of equipment is used throughout the test sequence, and that the test simulates as closely as practicable the postulated accident environment.	(1) Justification of the adequacy of the test sequence selected should be provided. (2) The test should simulate as closely as practicable the postulated environment. (3) The test procedures should conform to the guidelines described in Section 5 of IEEE Std. 323-1971.

CATEGORY I

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974

2.4 Other Qualification Methods

Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified. It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods.

3. MARGINS

(1) Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters.

(2) In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Section 6.3.1.5, should be used as a guide. (Note exceptions stated in Section 1.4.)

(3) When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncer-

CATEGORY II

Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971

(4) The staff considers that, for vital electrical equipment such as penetrations, connectors, cables, valves and motors, and transmitters located inside containment or exposed to hostile steam environments outside containment, separate effects testing for the most part is not an acceptable qualification method. The testing of such equipment should be conducted in a manner that subjects the same piece of equipment to radiation and the hostile steam environment sequentially.

2.4 Other Qualification Methods

Same as Category I (except that IEEE Std. 323-1971 and ancillary standards endorsed at the time the CP SER was issued may be used).

3. MARGINS

(1) Same as Category I.

(2) The margins provided in the design will be evaluated on a case-by-case basis. Factors that should be considered in quantifying margins are (a) the environmental stress levels induced during testing, (b) the duration of the stress, (c) the number of items tested and the number of tests performed in the hostile environment, (d) the performance characteristics of the equipment while subjected to the environmental stresses, and (e) the specified function of the equipment.

(3) Same as Category I.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
tainties such as production errors and errors associated with defining satisfactory performance (e.g., when only a small number of units are tested).	
(4) Some equipment may be required by the design to <u>only</u> perform its safety function within a short time period into the event (i.e., within seconds or minutes), and, once its function is complete, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories is required to remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis. For all other equipment (e.g., post-accident monitoring, recombiners, etc.), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std. 323-1974 may be used.	(4) Same as Category I.
4. <u>AGING</u>	4. <u>AGING</u>
(1) Aging effects on all equipment, regardless of its location in the plant, should be considered and included in the qualification program.	(1) Qualification programs that are committed to conform to the requirements of IEEE Std. 382-1972 (for valve operators) and IEEE Std. 334-1971 (for motors) should consider the effects of aging. For this equipment, the Category I positions of Section 4 are applicable.
(2) The degrading influences discussed in Sections 6.3.3, 6.3.4 and 6.3.5 of IEEE Std. 323-1974 and the electrical and mechanical stresses associated with cyclic operation of equipment should be considered and included as part of the aging programs.	(2) For other equipment, the qualification programs should address aging only to the extent that equipment that is composed, in part, of materials susceptible to aging effects should be identified, and a schedule for periodically replacing the equipment and/or materials should be established. During individual case reviews, the staff will require that the effects of aging be accounted for on selected equipment if operating experience or testing indicates that the equipment may exhibit deleterious aging mechanisms.
(3) Synergistic effects should be considered in the accelerated aging programs. Investigation should be performed to assure that no known synergistic effects have been identified on materials that are included in the equipment being	

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
qualified. Where synergistic effects have been identified, they should be accounted for in the qualification programs. Refer to NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), "Qualification Testing Evaluation Quarterly Reports," for additional information.	
(4) The Arrhenius methodology is considered an acceptable method of addressing accelerated aging. Other aging methods that can be supported by type tests will be evaluated on a case-by-case basis.	
(5) Known material phase changes and reactions should be defined to insure that no known changes occur within the extrapolation limits.	
(6) The aging acceleration rate used during qualification testing and the basis upon which the rate was established should be described and justified.	
(7) Periodic surveillance testing under normal service conditions is not considered an acceptable method for on-going qualification, unless the plant design includes provisions for subjecting the equipment to the limiting service environment conditions (specified in Section 3(7) of IEEE Std. 279-1971) during such testing.	
(8) Effects of relative humidity need not be considered in the aging of electrical cable insulation.	
(9) The qualified life of the equipment (and/or component as applicable) and the basis for its selection should be defined.	
(10) Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and in other methods that may be reasonably assumed, coupled with good engineering judgment.	
5. QUALIFICATION DOCUMENTATION	5. QUALIFICATION DOCUMENTATION
(1) The staff endorses the requirements stated in IEEE Std. 323-1974 that, "The qualification documentation shall verify that each type of electrical equipment is qualified for its application and meets its specified performance	(1) Same as Category I.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
requirements. The basis of qualification shall be explained to show the relationship of all facets of proof needed to support adequacy of the complete equipment. Data used to demonstrate the qualification of the equipment shall be pertinent to the application and organized in an auditable form."	
(2) The guidelines for documentation in IEEE Std. 323-1974 when fully implemented are acceptable. The documentation should include sufficient information to address the required information identified in Appendix E. A certificate of conformance by itself is not acceptable unless it is accompanied by test data and information on the qualification program.	(2) Same as Category I, except the guidelines of IEEE Std. 323-1971 may be used.

APPENDIX A

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

APPENDIX A

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

Acceptable methods for calculating the mass and energy release to determine the loss-of-coolant accident (LOCA) environment for PWR and BWR plants are described in the following:

(1) Topical Report WCAP-8312A for Westinghouse plants.

(2) Section 6.2.1 of CESSAR System 80 PSAR for Combustion Engineering plants.

(3) Appendix 6A of B-SAR-205 for Babcock & Wilcox plants.

(4) NEDO-10320 and Supplements 1 & 2 for General Electric plants.

Acceptable methods for calculating the mass and energy release to determine the main steam line break (MSLB) environment are described in the following:

(1) Appendix 6B of CESSAR System 80 PSAR for Combustion Engineering plants.

(2) Section 15.1.14 of B-SAR-205 for Babcock & Wilcox plants.

(3) Same as item (4) above for General Electric plants.

(4) Topical Report WCAP-8822 for Westinghouse plants. (Although this Topical Report is currently under review, the use of this method is acceptable in the interim if no entrainment is assumed. Reanalysis may be required following the NRC staff review of the entrainment model as presently described.)

APPENDIX B

MODEL FOR ENVIRONMENTAL QUALIFICATION FOR

LOSS-OF-COOLANT ACCIDENT AND MAIN STEAM LINE BREAK

INSIDE PWR AND BWR DRY TYPE OF CONTAINMENT

APPENDIX B

MODEL FOR ENVIRONMENTAL QUALIFICATION FOR
LOSS-OF-COOLANT ACCIDENT AND MAIN STEAM LINE BREAK
INSIDE PWR AND BWR DRY TYPE OF CONTAINMENT

1. Methodology to Determine the Containment Environmental Response

a. Heat Transfer Coefficient

For heat transfer coefficient to the heat sinks, the Tagami condensing heat transfer correlation should be used for a LOCA with the maximum heat transfer rate determined at the time of peak pressure or the end of primary system blowdown. A rapid transition to a natural convection, condensing heat transfer correlation should follow. The Uchida heat transfer correlation should be used for MSLB accidents while in the condensing mode. A natural convection heat transfer coefficient should be used at all other times when not in the condensing heat tranfer mode for both LOCAs and MSLB accidents. The application of these correlations should be as follows:

(1) Condensing heat transfer

$$q/A = h_{cond} \cdot (T_s - T_w)$$

where q/A = the surface heat flux
h_{cond} = the condensing heat transfer coefficient
T_s = the steam saturation (dew point) temperature
T_w = surface temperature of the heat sink

(2) Convective heat transfer

$$q/A = h_c \cdot (T_v - T_w)$$

where h_c = convective heat transfer coefficient
T_v = the bulk vapor temperature

All other parameters are the same as for the condensing mode.

b. Heat Sink Condensate Treatment

When the containment atmosphere is at or below the saturation temperature, all condensate formed on the heat sinks should be transferred directly to the sump. When the atmosphere is superheated, a maximum of 8 percent of the condensate may be assumed to remain in the vapor region. The condensed mass should be calculated as follows:

$$M_{cond} = X \cdot q / (h_v - h_L)$$

where M_{cond} = mass condensation rate

X = mass condensation fraction (0.92)

q = surface heat transfer rate

h_v = enthalphy of the superheated steam

h_L = enthalphy of the liquid condensate entering the sump region (i.e., average enthalpy of the heat sink condensate boundary layer)

c. Heat Sink - Surface Area

The surface area of the heat sinks should correspond to that used for the containment design pressure evaluation.

d. Single Active Failure Evaluation

Single active failures should be evaluated for those containment safety systems and components relied upon to limit the containment temperature/pressure response to a LOCA or MSLB accident. This evaluation should include, but not necessarily be limited to, the loss or availability of offsite power (whichever is worse), diesel generator failure when loss of offsite power is evaluated, and loss of containment heat removal systems (either partial or total, whichever is worse).

e. Containment Heat Removal System Actuation

The time determined at which active containment heat removal systems become effective should include consideration of actuation sensors and setpoints, actuation delay time, and system delay time (i.e., time required to come into operation).

f. Identification of Most Severe Environment

The worst case for environmental qualification should be selected considering time duration at elevated temperatures as well as the maximum temperature. In particular, consider the spectrum of break sizes analyzed and single failures evaluated.

2. <u>Acceptable Methodology for Safety-Related Component Thermal Analysis</u>

Component thermal analyses may be performed to justify environmental qualification test conditions that are found to be less than those calculated during the containment environmental response calculation.

The heat transfer rate to component should be calculated as follows:

a. Condensing Heat Transfer Rate

$$q/A = h_{cond} \cdot (T_s - T_w)$$

where q/A = component surface heat flux
h_{cond} = condensing heat transfer coefficient is equal to the larger of 4x Tagami correlation or 4x Uchida correlation
T_s = saturation temperature (dew point)
T_w = component surface temperature

b. Convective Heat Transfer

A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period, a forced convection heat transfer correlation should be used. For example:

$$NU = C\ (Re)^n$$

where Nu = Nusselt number
Re = Reynolds number
C,n = empirical constants dependent on geometry and Reynolds number

The velocity used in the evaluation of Reynolds number may be determined as follows:

$$V = 25\ \frac{M_{BD}}{V_{CONT}}$$

where V = velocity in ft/sec
M_{BD} = the blowdown rate in lbs/hr
V_{CONT} = containment volume in ft^3

After the blowdown has ceased or reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable. However, use of a natural convection heat transfer coefficient must be fully justified whenever used.

APPENDIX C

QUALIFICATION PROFILES FOR BWR AND ICE CONDENSER CONTAINMENTS

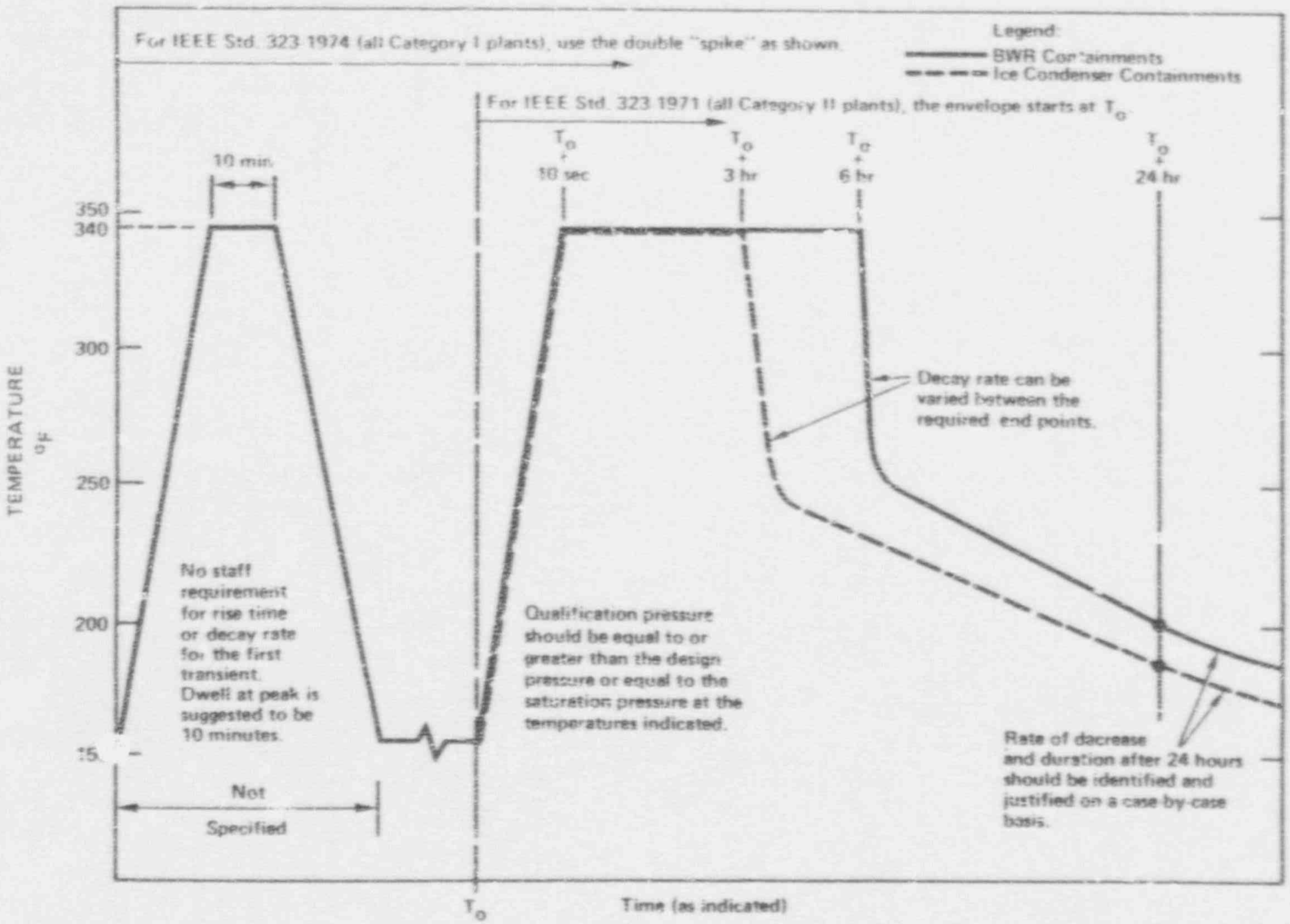

Figure C-1. Qualification Profiles for BWR and Ice Condenser Containments.

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

This appendix illustrates the proposed staff model for calculating dose rates and integrated doses for equipment qualification purposes. The example doses shown below include contributions from several dose point locations in the containment and cover a period of only thirty days following the postulated fission product release. The values shown are not intended for use as appropriate equipment qualification levels. The dose levels intended for qualification purposes should be determined using the maximum time the equipment is intended to function which, for the design basis LOCA event, may well exceed thirty days.

The beta and gamma integrated doses presented in the tables below have been estimated using models and assumptions consistent with those of Regulatory Guides 1.7 and 1.89. This analysis is conservative, but it does not ignore important time-dependent phenomena related to the action of engineered safety features (ESFs) and natural phenomena, such as plate-out, as done in previous staff analyses.

Doses were calculated for points within the containment atmosphere, at the containment surface (taking sprays and plate-out mechanisms into account), and near the sump water.

THIRTY-DAY INTEGRATED DOSES

Location	Integrated Dose (Rad) Beta	Integrated Dose (Rad) Gamma
Containment Atmosphere	1.4×10^8	1.5×10^7
Containment Surface	1.1×10^7	9.1×10^6
Near Sump Water	7.2×10^7	4.4×10^6

1. General Summary of the LOCA Scenario

The accident considered in this report for determining the radiation environment for qualification of safety-related equipment is a design basis LOCA. The following is a description of the events that are postulated to occur. At the time t=0, the pipe break occurs and results in rapid blowdown of the reactor coolant system (RCS). The blowdown of the RCS ends approximately 20 to 40 seconds after the break. Flashing and escape of the coolant during blowdown removes heat rapidly from the primary system and causes the fuel rod cladding temperature to drop. Consequently, only a few fuel rods are expected to fail during the blowdown period.

Following the end of blowdown, the fuel rods are uncovered and the stored heat in the fuel and the decay heat are transferred to the cladding, thus raising the cladding temperature. Some fuel rods may experience cladding failure during this period. The ECCS refills the lower reactor vessel and then refloods the core region within 100 to 300 seconds. causing cladding temperature turnaround. During the initial blowdown, only the radioactive material contained in the coolant from steady-state operation would be released to the containment. During reflood/refill when fuel rod cladding failure may occur, the noble gases would be transported out of the primary system by steam flow and would become airborne within the primary containment of a PWR (or within the drywell of a BWR). Some fraction of the iodines and less volatile fission products that are released as a result of fuel rod failure would also be transported out of the primary system by the steam flow and become airborne, and some fraction would remain in solution in the sump water or would be deposited on surfaces within the primary system. The amount that becomes airborne outside the primary system would be strongly dependent on the time of fuel rod failure and the transport phenomenon for each species within the primary system.

Following the release from the primary system, the fission products would be distributed within the containment. For a PWR containment, the released airborne activity would rapidly disperse and become uniformly distributed within the primary containment. For a BWR, the released activity would be airborne within the drywell. Following initial release to the containment atmosphere, the action of natural convection currents and ESF equipment, such as cooling fans, will cause time-dependent redistribution of the activity within the containment. Natural removal processes, such as deposition on containment surfaces and washout from the containment atmosphere by the containment spray systems, would reduce the airborne activity concentration and would redistribute this activity to the containment surfaces and to the containment sump water.

During the same period of time, leakage of radioactivity from the containment to the atmosphere could take place. This would be processed to some extent, by ESF filters if present, causing a buildup of activity on these filters. In addition, there could be some deposition and plateout of radioactivity (iodine and daughters of noble gases) on surfaces of ductwork or on the walls of secondary containment.

During the longer term, contaminated primary coolant could be circulated through pipes outside of containment (PWR residual heat removal model). The staff usually assumes a failure of a seal in the ECCS equipment, such that significant quantities of coolant could leak into compartments outside of containment. The leaked fluid is either retained in a sealed room or transported to the radwaste system. Some portion of this leaked fluid is volatilized and also transported in the air of these compartments. These sources would be processed to some extent by ESF filters.

2. Basic Assumptions Used in the Analysis

Gamma and beta doses and dose rates were determined for three types of radioactive source distributions: isotopes suspended in the containment atmosphere, plated-out on containment surfaces, or mixed in the containment sump water. Thus, a given piece of equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and effects of shielding.

Previous guidance issued by the staff regarding the source term for equipment qualification was general in nature. Recognizing that implementation of that guidance required a number of assumptions to be made regarding the time-dependent behavior of material within and outside of containment, the staff, in this report, has performed an analysis of the radiation environment that is associated with the source term of position C.2 of Regulation Guide 1.89, using assumptions and methods which were intended to be consistent with staff practices in analyzing the radiological consequences of a design basis LOCA. Position C.2 of Regulatory Guide 1.89 assumes a source term condition associated with a core meltdown. To get a feel for the degree of conservatism in this assumption, calculations using the RELAP-EM (Evaluation Model) program, which uses the conservative assumptions given in Appendix K to 10 CFR Part 50, predict that the peak cladding temperature attained by the hottest fuel rod will be less than 2200°F. Based on the predicted distribution of cladding temperature throughout the core, it is estimated that between 20 and 80 percent of the fuel rods could experience cladding failure for a PWR with a lesser fraction for a BWR. Calculations performed using the more realistic RELAP-BE (Best Estimate) program predicted much lower cladding temperatures than RELAP-EM. Based on the RELAP-BE predictions, the number of fuel rod cladding failures is estimated to be less than 10 percent.

A Sandia Laboratories report (SAND 76-0740, "Radiation Signature Following the Hypothesized LOCA") also analyzed the radiation environment associated with the conditions of position C.2 of Regulatory Guide 1.89. But as noted in the text of that report (ct. Table 1.1, for example), those analyses are based upon calculational assumptions that are not consistent (are overly conservative) with respect to staff recommended practices. Therefore, the results in that report should not be directly applied.

Table D-1 compares the source terms of position C.2 of Regul... ...uide 1.89 to source terms used for other design basis events.

3. Analysis of the Concentration of Fission Products in Air

This section discusses the physical model used to simulate the .WR containment and to determine the time-dependent and location-dependent distribution of noble gases and iodines airborne within the containment atmosphere and plated-out on containment surfaces.

The staff has developed a computer program (TACT) to be published that is used to model the time-dependent behavior of iodine and noble gases within a nuclear power plant. The TACT code is used routinely by the staff for the calculation of the offsite radiological consequences of a LOCA, and is an acceptable method for modeling the transfer of activity from one containment region to another and in modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 1), is used to estimate the removal rates of elemental iodine by plate-out and sprays, and is a needed input to TACT. These codes were used to develop the source term estimates.

The source terms used in the analysis assumed that 50 percent of the core iodines and 100 percent of the core noble gases were released instantaneously to the containment atmosphere. The following assumptions were also used to calculate the distribution of radioactivity within the containment:

a. The representative containment free volume was taken as 2.52×10^6 ft^3. Of this volume, 74 percent or 1.86×10^6 ft^3 is assumed to be directly covered by the containment sprays.

b. 6.6×10^5 ft^3 of the containment free volume is assumed unsprayed, which includes regions within the main containment room near the containment dome and compartments below the operating floor level. Good mixing of the containment activity between the sprayed and unsprayed regions is assured by natural convection currents and ESF fans.

c. The ESF fans are assumed to have a design flow rate of 220,000 cfm in the post-LOCA environment. Since mixing between all major unsprayed regions and compartments and the main sprayed region will occur, the containment was modeled with TACT nodes.

d. Air exchange between the sprayed and unsprayed region was taken as one-half of the design flow rate of ESF fans plus the effect of natural convection.

e. The containment spray system was assumed to have two equal capacity trains, each designed to inject 3000 gpm of boric acid solution into the containment.

f. Trace levels of hydrazine was assumed added to enhance the removal of iodine.

g. The spray removal rate constant (lambda) was calculated using the staff's SPIRT program, conservatively assuming only one spray train operation and an elemental iodine instantaneous partition coefficient (H) of 5000. The calculated value of the elemental iodine spray removal constant was 27.2 hr^{-1}, which represents an elemental iodine residence half-life in the sprayed region of approximately 1.5 minutes.

h. Plate-out of iodine on containment internal surfaces was modeled as a first-order rate removal process and best estimates for model parameters were assumed. Based on an assumed total surface area within containment of approximately 5.0×10^5 ft^2, the calculated value for the overall elemental iodine plate-out constant was 1.23 hr^{-1}

i. The spray removal and plate-out process were modeled as competing iodine removal mechanisms.

4. Departure from Past Practices

Computing the radiological consequences at the exclusion radius and the low population zone, the staff usually assumed that an instantaneous release of 100 percent of the noble gases and 25 percent of the core iodines is available for leakage from the containment. Recognizing that it would take some time before a release of this magnitude could occur, even assuming degraded emergency core cooling system (ECCS) operation, the staff has also assumed, for purposes of estimating offsite dose consequences, that the source is uniformly distributed and that containment sprays activate at the time the large source is available for release (both of which would also take time to occur). Also implicit in the 25 percent release of iodines was the assumption that 50 percent of a 50 percent release of iodine from the fuel is plated-out in a very short period of time.

The staff usually limits credit for element iodine spray removal to no more than 10 hr^{-1}, for an assumed release of 25 percent of the halogens to compensate for the articial assumption of instantaneous plate-out. If a release of 50 percent were assumed (as is implied by Regulatory Guide 1.7 and TID-14844), the actual conservatively calculated spray lambdas would be appropriate. In any event, removal of elemental iodine from the containment atmosphere by spray and plate-out is assumed to cease when the concentration in the sprayed region is reduced by a factor of 200 (when the initial concentration of iodine in the containment is calculated assuming 50 percent of the core inventory of iodines is initially airborne). This reduction factor is obtained by doubling the reduction factor used in the LOCA dose analysis. The intent is to achieve an equilibrium airborne iodine concentration that is consistent with the LOCA analysis. Since the initial (t=0) concentration is assumed to be twice that of the LOCA analysis (50 versus 25 percent), the reduction factor has been doubled.

The staff assumes that more than one species of iodine is present, or will be formed, in a design basis LOCA (see Regulatory Guides 1.3 and 1.4). For our analysis, it is assumed that 2.5 percent of the core inventory of iodine released is associated with airborne particulate material and 2 percent of the core inventory of iodine released forms organic compounds. Even though these values would not be obtained until several hours after the LOCA, it is the staff assumption that the aforementioned composition is present at t=0.

A removal rate constant for particulate iodine was calculated to be 0.43 hr^{-1}. The organic iodine concentration in the containment atmosphere was assumed to be unaffected by containment sprays or plate-out. The action of sprays would not commence at t=0 (e.g., some time would elapse between the onset of the LOCA and the delivery of spray solution to the spray nozzles). Similarly, the assumed large source would not be immediately released from the fuel, and some time would pass before any airborne iodine would be distributed throughout containment.

The assumption of a large release, uniformly distributed in containment (or in the sump water as will be discussed later) is a convenient simplification for purpose of the dose assessment in a PWR containment, and is conservative in terms of specifying the time-dependent radiation environment. Accurate coupling of the various time sequences is beyond the scope of this analysis.

The calculated values of noble gas and airborne iodine activity in the containment as a function of time following the LOCA are presented in Table D-2.

5. Analysis of the Concentration of Fission Products on Surface

The air dose model assumed that only one spray train and one ventilation system train were operable. If both trains of both systems were operable, spray washout would progress more rapidly in the sprayed regions and the "equilibrium" of concentrations between sprayed and unsprayed regions would be reached more quickly. The result would be lower dose rates due to plate-out activity on surfaces or suspended in the air in sprayed regions, and in unsprayed regions during the early phases of the accident.

It has been suggested that the plate-out source used in estimating the radiation environment should assume that 50 percent of the released elemental iodine is instantaneously plated-out on containment and equipment surfaces. This assumption is inconsistent with the time-dependent model used to characterize the concentration of iodines in the air. It is the staff's view that the estimates should be mechanistically consistent. A large margin of conservatism already exists by virtue of the assumed large source term. In any event, the subsequent removal of deposited material by washoff (by sprays or condensate flow) may be important. Ignoring this factor (as was done for this short-term effort) introduces conservatism. Current staff guidelines do not include an acceptable method for estimating this effect. In the absence of such methods, it has been assumed that all plated-out material is retained by the containment surfaces. Table D-3 gives the values calculated for the iodine activity buildup on the plate-out surfaces of the containment.

6. Analysis of the Concentration of Fission Products in the Sump

Regulatory Guide 1.7 (Table D-1) recommends that 50 percent of the iodines and 1 percent of the remaining fission products present in the core are assumed to be intimately mixed with the coolant water. These values stem directly from TID-14844 (and we presume that the 1 percent solids refer to fission products other than halogens and noble gases). No specification of the time dependencies for this source are given. However, for a PWR with containment sprays, the elemental iodine (constituting about 95 percent of released iodine) is rapidly washed out of the containment atmosphere and transported to the containment sump (over 90 percent in less than 15 minutes is a typical result). Table D-4 presents an estimate of buildup of iodine in the sump fluid. There is little difference in the estimated integrated dose from the sump water between these values and values resulting from an assumed instantaneous release of 50 percent of the core iodines into the sump.

The inclusion of solid fission products in the sump source seems to be an artifact from the source of TID-14844. Although it may have applicability to the estimates of hydrogen production per Regulatory Guide 1.7, its applicability to radiation dose estimates has not been fully resolved. Pending this resolution, it should be assumed that the sump fluid contains 1 percent of the solid fission products and that the solid fission products are released and uniformly distributed in the sump fluid at t=0.

7. Estimates of the Radiation Environment Dose and Dose Rates

Previous staff estimates did not take into account the important time-dependent and spatially dependent phenomena. The calculated radiation environment was generally taken as a point on a surface or in the center of containment.

The activities within the containment regions were used as input to calculate the beta and gamma dose rates and integrated doses. One typical location was assumed to be a point located in the center of the main containment region. A second location was assumed to be a point on a containment inner surface. A third location would be adjacent to the sump water. Doses for representative points outside containment were taken from Reference 2 and are also listed for completeness.

The gamma transport calculations were performed in cylindrical geometry. Containment internal geometry was not modeled because this was considered to involve a degree of complexity beyond the scope of the present work. The calculations of both References 3 and 4 indicate that the specific internal shielding and structure would be expected to reduce the gamma doses and dose rates by factors of two or more, depending upon the specific location and geometry.

The beta doses were calculated using the infinite medium approximation. Because of the short range of the betas, this was shown in Reference 5 to result in only small error. The beta doses are not expected to be significantly reduced by the presence of containment internal structures.

Finally, the doses were multiplied by a correction factor of 1.3 as suggested by Reference 5 to account for the neglect of the decay chains with subsequent growing-in of additional daughter products.

a. Containment Atmosphere Doses and Integrated Dose

The beta and gamma dose rates and integrated doses for a point detector on the containment centerline exposed to the airborne activity within the containment atmosphere was calculated. The containment was modeled as an air-filled cylinder whose height equaled the diameter. Containment internal structure and shielding were neglected. The gamma dose rate contribution for the plate-out iodine on containment surfaces to the detector was also modeled and included as a contributor. The gamma dose rates and integrated doses are shown in Table D-5, whereas the beta dose rates and integrated doses are shown in Table D-6. The increased pressure effects in a post-LOCA containment have little shielding importance and therefore was not considered. This results in a small conservatism in the calculated dose.

b. Surface Dose and Dose Rates

The beta and gamma dose rates and integrated doses were computed for containment coatings on which iodine fisssion products were presumed to be plated-out. The containment coatings were assumed to have a thickness of 10 mils (0.0254 cm) with an average density of 2 gms/cm^3.

Removal of plated-out activity with time is expected to be a complex phenomenon dependent upon such conditions as whether the surface is exposed to the sprays and whether moisture condensation and runoff can be expected to remove surface activity. Assuming complete retention of plate-out activity, half of the beta energy from plated-out iodine is assumed directed toward the coated surface. The airborne contribution was added to the plate-out contribution, and all the betas directed toward the coating were assumed to be absorbed in the coating. This is conservative since the maximum range for betas is greater than the coating thickness. Hence, this assumption may overestimate the beta dose for a specific coating, but may be appropriate for a cable insulation layer. The airborne contribution was taken to be one-half the dose rate from an infinite cloud.

The gamma dose rate at the plated-out surface exposed to airborne activity was calculated to be one-half of the dose rate for a detector at the containment centerline. Although half of the gamma energy from plated-out iodine is also directed toward the coating, the coating is calculated to be relatively permeable to gammas with only about 1 percent of the plated-out gammas absorbed by the coating, and this contribution is considered negligible.

The gamma dose rates and integrated doses are therefore half of the centerpoint values for an airborne detector. The gamma dose rates are not significantly affected by the radioactive decay of plated-out activity with time.

The beta dose rates and integrated doses for "well-washed" and "unwashed" surface, respectively, are shown in Table D-7. Note that a plate-out "washoff" model was not used for the "well-washed" example, the plate-out dose rate component was set equal to zero.

c. Dose Near Sump Water

The activity in the sump water was assumed to vary with time, and to be initially free of any iodine fission products. Ultimately, essentially all of the iodine released appears in the sump water. Table D-4 gives the iodine activity in the sump as a function of time. Note that the maximum is reached in about 0.2 hour with radioactive decay reducing the activity afterwards. The beta and gamma dose rates and integrated doses were computed for a detector located at the surface of a large pool of sump water contaminated by iodine and solid fission products. There was 44,200 cubic feet of water that was assumed to cover the bottom of the containment. The containment geometry was simplified to assume a uniform depth of water of about 2.5 feet, and the dose rates were calculated at the sump water surface excluding the effects of buildup. The gamma dose rate and integrated dose from the sump water source are given in Table D-8.

d. Equipment Outside Containment

Although not specifically calculated in this study, several values of dose rates and doses at points outside of containment were taken from Reference 2 for completeness. The method used in this report in arriving at these results are acceptable for plant-specific determination.

The gamma dose rates and integrated doses at a point outside of containment are shown in Table D-9 (taken from Reference 2). The containment source was assumed to be a Regulatory Guide 1.4 source (with a power level of 4000 MWt) and was shielded by 3 feet of concrete. The dose rates at the beginning of recirculation near a pipe containing water contaminated by iodine fission products was also calculated in Reference 2 and the dose rates are shown in Table D-10.

8. Comparison of a PWR and a BWR

A detailed model for a BWR equivalent to the PWR model is not presented in this report. Doses to equipment inside a BWR containment (primarily considering a BWR with a MARK III type of containment structure) would not be expected to differ greatly from the doses calculated for PWR equipment. However, some differences in equipment doses will result due to the compartmented design of BWR containments, and the fact that most BWRs do not have containment sprays designed for rapid iodine removal.

Several of the models and assumptions used in the PWR analysis would not be appropriate for an equivalent analysis for a BWR. Specifically:

a. The assumption of an initial uniformly distributed airborne concentration of activity throughout the containment is not appropriate for a BWR containment.

b. Following the blowdown portion of the LOCA, the air exchange rates between the drywell region and the remainder of the containment free volume will be relatively small.

c. Since any major releases of activity would be initially into the drywell and would occur following the blowdown period, only relatively slow transport would occur to the main containment volume. Consequently, an appropriate model for a BWR containment should consider that all (or most) of the activity is initially released into the drywell region.

d. It is important to correctly estimate the atmospheric mixing rates between the drywell and the main containment regions (including sprayed and unsprayed regions) to adequately estimate the time-dependent and location-dependent distribution of activity. This should include an estimate of the flow between the drywell and the main containment that bypasses the suppression pool. This suggests a relatively detailed multi-node containment model, if overly conservative estimates of the radiation environment are to be avoided.

e. Removal of iodines from the main containment region and from the drywell, by operation of ESF systems such as containment sprays, should be modeled in a manner similar to that used in calculating offsite doses (i.e., single failure etc.).

f. Time-dependent deposition of iodines on surfaces by natural processes should be evaluated using mechanistic models and best estimates for model parameters; this will require a relatively detailed evaluation of potential deposition surfaces within the main containment and drywell.

g. Capture of iodines in the suppression pool, although not currently assumed, may be important and should be evaluated.

Table D-1. Source Terms: Activity Released from the Fuel as a Percentage of the Total Core Inventory

Source Terms	Activity Released (percent) Noble Gases	Iodines	Solids
1. Source term based on TID-14844 required by Reg. Guides 1.3 and 1.4)	100	50	0
2. Source term as required by Regulatory Guides 1.7 and 1.89 Rev. 0 (base case)*	100	50	1
3. Source term based on conservative gap release (Reg. Guide 1.25)	10 (30 of Kr-185)	10	0
4. Best estimates of total activity gap:			
WASH-1400	3	5	
NUREG/CR-0091**	1.27	2.79	

*Case 2 was used in the calculations presented in this appendix.
**Calculated for stable and long half-life isotopes (Ref. 8).

Table D-2. PWR Airborne Activity Distribution Within Containment Versus Time - Base Case, Ci

Time (hours)	Noble Gases	Elemental Iodine	Organic Iodine	Particulate Iodine	Total Iodine	Total Airborne
0.0	1.31 + 9	4.37 + 8	9.15 + 6	1.14 + 7	4.58 + 8	1.77 + 9
0.03	1.19 + 9	4.17 + 8	9.07 + 6	1.13 + 7	4.37 + 8	1.63 + 9
0.50	7.36 + 8	3.56 + 6	7.98 + 6	8.58 + 6	2.01 + 7	7.56 + 8
0.75	6.80 + 8	3.35 + 6	7.51 + 6	7.46 + 6	1.83 + 7	6.98 + 8
1.00	6.41 + 8	3.17 + 6	7.11 + 6	6.52 + 6	1.68 + 7	6.58 + 8
2.00	5.54 + 8	2.66 + 6	5.95 + 6	3.96 + 6	1.26 + 7	5.67 + 8
8.00	3.62 + 8	1.62 + 6	3.62 + 6	3.56 + 5	5.60 + 6	3.68 + 8
24.00	2.33 + 8	9.11 + 5	2.04 + 6	1.21 + 3	2.95 + 6	2.36 + 8
60.00	1.64 + 8	4.84 + 5	1.09 + 6	--	1.57 + 6	1.66 + 8
96.00	1.33 + 8	3.47 + 5	7.78 + 5	--	1.13 + 6	1.34 + 8
192.00	7.84 + 7	2.19 + 5	4.92 + 5	--	7.11 + 5	7.91 + 7
298.00	4.49 + 7	1.48 + 5	3.34 + 5	--	4.82 + 5	4.54 + 7
394.00	2.73 + 7	1.05 + 5	2.37 + 5	--	3.42 + 5	2.76 + 7
560.00	1.20 + 7	5.76 + 4	1.31 + 5	--	1.89 + 5	1.22 + 7
720.00	6.01 + 6	3.23 + 4	7.36 + 4	--	1.06 + 5	6.12 + 6

Table D-3. Total Plate-out Surface Activity in the Containment Versus Time for the Base Case

Time (hours)	Iodine Activity Deposited on Surfaces, Ci
0.0	0.0
0.03	1.57 + 7
0.07	2.96 + 7
0.14	3.92 + 7
0.20	4.23 + 7
0.40	--
0.50	4.23 + 7
0.75	3.98 + 7
1.00	3.77 + 7
2.00	3.15 + 7
8.00	1.92 + 7
24.00	1.08 + 7
60.00	5.76 + 6
96.00	4.13 + 6
192.00	2.61 + 6
298.00	1.77 + 6
394.00	1.25 + 6
560.00	6.91 + 5
720.00	3.90 + 5

Table D-4. Iodine Activity in Containment Sump Versus Time
Iodine Activity in Containment Sump, Ci

Time (hours)	Elemental Iodine	Particulate* Iodine	Total Iodine in Sump
0.0	0.0	0.0	0.0
0.03	0.0	0.0	0.0
0.07	2.04 + 8	- -	2.04 + 8
0.14	3.04 + 8	- -	3.04 + 8
0.20	3.35 + 8	- -	3.35 + 8
0.25	3.44 + 8	- -	3.44 + 8
0.50	3.34 + 8	1.39 + 6	3.35 + 8
0.75	3.15 + 8	1.93 + 6	3.17 + 8
1.00	2.98 + 8	2.36 + 6	3.00 + 8
2.00	2.49 + 8	3.48 + 6	2.52 + 8
8.00	1.52 + 8	4.18 + 6	1.56 + 8
24.00	8.58 + 7	2.54 + 6	8.83 + 7
60.00	4.56 + 7	1.36 + 6	4.70 + 7
96.00	3.27 + 7	9.75 + 6	3.37 + 7
192.00	2.06 + 7	6.15 + 5	2.12 + 7
298.00	1.40 + 7	4.18 + 5	1.44 + 7
394.00	9.43 + 6	2.96 + 5	9.73 + 6
560.00	5.48 + 6	1.63 + 5	5.64 + 6
720.00	3.09 + 6	9.30 + 4	3.18 + 6

*Particulate iodine activity in the containment sump for times less than 0.5 hours is small and, when added to the elemental iodine activity, does not significantly affect the total magnitude of the iodine activity in the sump

Table D-6. Beta Dose Rates and Integrated Doses at the Containment Center Versus Time in Air

Time (hours)	Dose Rate in Containment Air (R/hr)	Integrated Dose in Containment Air (R)
0.0	2.373 + 7	--
0.03	1.951 + 7	8.89 + 5
0.25	5.856 + 6	3.55 + 6
0.5	4.198 + 6	4.93 + 6
0.75	3.671 + 6	6.0 + 6
1.0	3.369 + 6	7.13 + 6
2.0	2.758 + 6	1.03 + 7
8.0	1.538 + 6	2.21 + 7
24.0	7.068 + 5	4.1 + 7
60.0	3.919 + 5	6.1 + 7
96.0	3.117 + 5	7.2 + 7
192.0	1.871 + 5	8.9 + 7
298.0	1.083 + 5	1.03 + 8
394.0	6.807 + 4	1.08 + 8
560.0	3.278 + 4	1.17 + 8
720.0	1.901 + 4	1.26 + 8

Table D-7. Beta Dose Rates and Integrated Doses for Paint on Containment Wall - Washed and Unwashed Cases

Time (hours)	Dose Rate* Unwashed (R/hr)	Dose Rate** Washed (R/hr)	Dose Unwashed (R)	Dose Washed (R)
0.0	1.19 + 7	1.19 + 7	0.0	0.0
0.03	1.01 + 7	9.76 + 6	4.99 + 5	6.46 + 5
0.25	3.79 + 6	2.93 + 6	1.81 + 6	1.69 + 6
0.5	2.92 + 6	2.10 + 6	2.70 + 6	2.32 + 6
0.75	2.60 + 6	1.84 + 6	3.65 + 6	3.0 + 6
1.0	2.39 + 6	1.68 + 6	4.20 + 6	3.25 + 6
2.0	1.94 + 6	1.38 + 6	6.39 + 6	4.77 + 6
8.0	1.07 + 6	7.69 + 5	1.42 + 7	9.9 + 6
24.0	5.05 + 5	3.53 + 5	2.55 + 7	1.77 + 7
60.0	2.60 + 5	1.96 + 5	3.90 + 7	2.73 + 7
96.0	1.96 + 5	1.56 + 5	4.6 + 7	3.3 + 7
192.0	1.16 + 5	9.36 + 4	6.0 + 7	4.4 + 7
298.0	6.90 + 4	5.42 + 4	7.0 + 7	5.2 + 7
394.0	4.45 + 4	3.40 + 4	7.6 + 7	5.6 + 7
560.0	2.22 + 4	1.64 + 4	8.2 + 7	6.1 + 7
720.0	1.28 + 4	9.51 + 3	8.29 + 7	6.33 + 7

*Includes both the containment airborne and plate-out contributions.
**Includes only the containment airborne contribution.

Table D-5. Total Gamma Dose Rates and Integrated Doses at the Containment Center in Air Versus Time - Base Case Unwashed

Time (hours)	Gamma Dose Rate From Airborne (R/hr)	Gamma Dose Rate in Air From Plate-out Source (R/hr)	Total Gamma Dose Rate in Air (R/hr)	Total Integrated Gamma Dose in the Containment Air (R)
0.0	4.92 + 6	1.56 + 4	4.92 + 6	--
0.03	4.43 + 6	5.59 + 4	4.49 + 6	2.06 + 5
0.50	1.33 + 6	1.44 + 5	1.47 + 6	1.18 + 6
0.75	1.16 + 6	1.33 + 5	1.29 + 6	1.55 + 6
1.00	1.05 + 6	1.23 + 5	1.17 + 6	1.82 + 6
2.00	7.75 + 5	9.44 + 4	8.69 + 5	2.80 + 6
8.00	2.37 + 5	4.14 + 4	2.78 + 5	6.0 + 6
24.00	5.19 + 4	1.58 + 4	6.77 + 4	7.1 + 6
60.00	1.70 + 4	6.36 + 3	2.34 + 4	9.2 + 6
96.00	1.30 + 4	4.36 + 3	1.74 + 4	1.0 + 7
192.00	7.66 + 3	2.66 + 3	1.03 + 4	1.15 + 7
298.00	4.38 + 3	1.80 + 3	6.18 + 3	1.20 + 7
394.00	2.67 + 3	1.28 + 3	3.95 + 3	1.25 + 7
560.00	1.14 + 3	7.04 + 2	1.84 + 3	1.30 + 7
720.00	5.14 + 2	3.98 + 2	9.12 + 2	1.36 + 7

Table D-4. Iodine Activity in Containment Sump Versus Time
Iodine Activity in Containment Sump, Ci

Time (hours)	Elemental Iodine	Particulate* Iodine	Total Iodine in Sump
0.0	0.0	0.0	0.0
0.03	0.0	0.0	0.0
0.07	2.04 + 8	- -	2.04 + 8
0.14	3.04 + 8	- -	3.04 + 8
0.20	3.35 + 8	- -	3.35 + 8
0.25	3.44 + 8	- -	3.44 + 8
0.50	3.34 + 8	1.39 + 6	3.35 + 8
0.75	3.15 + 8	1.93 + 6	3.17 + 8
1.00	2.98 + 8	2.36 + 6	3.00 + 8
2.00	2.49 + 8	3.48 + 6	2.52 + 8
8.00	1.52 + 8	4.18 + 6	1.56 + 8
24.00	8.58 + 7	2.54 + 6	8.83 + 7
60.00	4.56 + 7	1.36 + 6	4.70 + 7
96.00	3.27 + 7	9.75 + 6	3.37 + 7
192.00	2.06 + 7	6.15 + 5	2.12 + 7
298.00	1.40 + 7	4.18 + 5	1.44 + 7
394.00	9.43 + 6	2.96 + 5	9.73 + 6
560.00	5.48 + 6	1.63 + 5	5.64 + 6
720.00	3.09 + 6	9.30 + 4	3.18 + 6

*Particulate iodine activity in the containment sump for times less than 0.5 hours is small and, when added to the elemental iodine activity, does not significantly affect the total magnitude of the iodine activity in the sump

Table D-8. Containment Sump Gamma Dose Rates and Integrated Doses Versus Time

Time (hours)	$\bar{E}$ (Mev)	Dose Rate at the Sump Surface From Iodine in Sump (R/hr)	Dose Rate at the Sump Surface From 1% Solids in Sump (R/hr)	Total Dose Rate at the Sump Surface (R/hr)	Total Integrated Gamma Dose at the Surface (R)
0.0	0.887	0.0	5.90 + 4	5.90 + 4	--
0.03	0.887	0.0	3.09 + 4	3.09 + 4	4.65 + 2
0.07	0.886	1.18 + 5	--	--	--
0.14	0.884	1.79 + 5	2.21 + 4	2.01 + 5	1.23 + 4
0.20	0.882	1.94 + 5	--	--	--
0.25	0.880	1.99 + 5	1.90 + 4	2.18 + 5	2.82 + 4
0.50	0.873	1.83 + 5	1.59 + 4	1.99 + 5	7.89 + 4
0.75	0.866	1.71 + 5	--	--	--
1.00	0.860	1.56 + 5	1.25 + 4	1.68 + 5	1.68 + 5
2.00	0.839	1.19 + 5	1.01 + 4	1.29 + 5	3.00 + 5
8.00	0.763	5.08 + 4	--	--	--
24.00	0.569	1.61 + 4	4.99 + 3	2.11 + 4	1.15 + 6
60.00	0.401	6.04 + 3	--	--	--
96.00	0.357	3.81 + 3	3.09 + 3	6.90 + 3	1.95 + 6
192.00	0.332	2.20 + 3	--	--	--
298.00	0.330	1.50 + 3	2.14 + 3	3.64 + 3	2.95 + 6
394.00	0.330	1.06 + 3	--	--	--
560.00	0.330	5.86 + 2	1.61 + 3	2.20 + 3	3.65 + 6
720.0	0.330	3.30 + 2	1.42 + 3	1.75 + 3	3.96 + 6

Table D-9. Gamma Dose Rates Outside Shielded Containment (3-foot Concrete Shield)

Time After Release (hours)	Dose Rate (R/hr)	Integrated Dose (Rads)
0	4.0×10^{2}	0
1	2.5×10^{2}	3.2×10^{2}
3	1.2×10^{2}	6.9×10^{2}
10	2.8×10^{1}	1.2×10^{3}
30	2.4×10^{0}	1.5×10^{3}
100	2.8×10^{-2}	1.6×10^{3}

Table D-10. Gamma Dose Rates at Beginning of Recirculation Near Pipe Containing Iodine Fission Products

Distance	Dose Rate (R/hr)
4 inches	1.6×10^5
1 foot	5.3×10^4
3 feet	1.8×10^4

REFERENCES

1. A. K. Postma and P. Tam, "Technological Bases for Models of Spray Washout and Airborne Contaminants in Containment Vessels," USNRC Report NUREG/CR-0009, November 1978. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

2. A. K. Postma and R. Zavadoski, "Review of Organic Iodide Formation Under Accident Conditions in Water Cooled Reactors," WASH-1233, October 1972, pp. 62-64. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

3. Memorandum from R. B. Minogue, NRC, to R. F. Fraley, October 25, 1975, "Response to Request for Additional Information Concerning Regulatory Guide 1.89 on Qualification of Class 1E Equipment." Available for inspection and copying for a fee from NRC PDR, Washington, DC 20555.

4. E. A. Warman and E. T. Boulette, "Engineering Evaluation of Radiation Environment in LWR Containments," Vol. 23, pp. 604-605 in Transactions of the American Nuclear Society, 1976. Available from technical libraries.

5. M. J. Kolar and N. C. Olson, "Calculation of Accident Doses to Equipment Inside Containment of Power Reactors," Vol. 22, pp. 808-809 in Transactions of the American Nuclear Society, 1975. Available from technical libraries.

6. D. C. Kocher, ed., "Nuclear Decay Data for Radionuclides Occurring in Routine Releases from Nuclear Fuel Cycle Facilities," ORNL/NUREG/TM-102, August 1977. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

7. E. Normand and W. R. Determan, "A Simple Algorithm to Calculate the Immersion Dose," Vol. 18, pp. 358-359 in Transaction of the American Nuclear Society, 1974. Available from technical libraries.

8. R. A. Lorenz, J. L. Collins, and A. P. Malinauskas, "Fission Product Source Terms for the LWR Loss-of-Coolant Accident: Summary Report," USNRC Report NUREG/CR-0091, May 1978. Available for purchase from National Technical Information Service, Springfield, Virginia 22161.

APPENDIX E

STANDARD QUESTION ON ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT

APPENDIX E

STANDARD QUESTION ON ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT

In order to ensure that your environmental qualification program conforms with General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, and to the national standards mentioned in Part II "Acceptance Criteria" (which includes IEEE Std. 323) contained in Standard Review Plan Section 3.11, the following information on the qualification program is required for all Class 1E equipment.

1. Identify all Class 1E equipment, and provide the following:
   a. Type (functional designation)
   b. Manufacturer
   c. Manufacturer's type number and model number
   d. The equipment should include the following, as applicable:
      (1) Switchgear
      (2) Motor control centers
      (3) Valve operators
      (4) Motors
      (5) Logic equipment
      (6) Cable
      (7) Diesel generator control equipment
      (8) Sensors (pressure, pressure differential, temperature and neutron)
      (9) Limit switches
      (10) Heaters
      (11) Fans
      (12) Control boards
      (13) Instrument racks and panels
      (14) Connectors
      (15) Electrical penetrations
      (16) Splices
      (17) Terminal blocks

2. Categorize the equipment identified in item 1 above into one of the following categories:

   a. Equipment that will experience the environmental conditions of design basis accidents for which it must function to mitigate said accidents, and that will be qualified to demonstrate operability in the accident environment for the time required for accident mitigation with safety margin to failure.

   b. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, but through which it must not fail in a manner detrimental to plant safety or accident mitigation, and that will be qualified to demonstrate the capability to withstand any accident environment for the time during which it must not fail with safety margin to failure.

c. Equipment that will experience environmental conditions of design basis accidents through which it need not function for mitigation of said accidents, and whose failure (in any mode) is deemed not detrimental to plant safety or accident mitigation, and need not be qualified for any accident environment, but will be qualified for its non-accident service environment.

d. Equipment that will not experience environmental conditions of design basis accidents and that will be qualified to demonstrate operability under the expected extremes of its non-accident service environment. This equipment would normally be located outside the reactor containment.

3. For each type of equipment in the categories of equipment listed in item 2 above, provide separately the equipment design specification requirements, including:

   a. The system safety function requirements.

   b. An environmental envelope as a function of time that includes all extreme parameters, both maximum and minimum values, expected to occur during plant shutdown, normal operation, abnormal operation, and any design basis event (including LOCA and MSLB), including post-event conditions.

   c. Time required to fulfill its safety function when subjected to any of the extremes of the environment envelope specified above.

   d. Technical bases should be provided to justify the placement of each type equipment in the categories 2.b and 2.c listed above.

4. Provide the qualification test plan, test setup, test procedures, and acceptance criteria for at least one of each group of equipment of item 1.d as appropriate to the category identified in item 2 above. If any method other than type testing was used for qualification (operating experience, analysis, combined qualification, or ongoing qualification), describe the method in sufficient detail to permit evaluation of its adequacy.

5. For each category of equipment identified in item 2 above, state the actual qualification envelope simulated during testing (defining the duration of the hostile environment and the margin in excess of the design requirements). If any method other than type testing was used for qualification, identify the method and define the equivalent "qualification envelope" so derived.

*6. A summary of test results that demonstrates the adequacy of the qualification program. If analysis is used for qualification, justification of all analysis assumptions must be provided.

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*For applications for construction permits, it is acceptable to state that items 6 and 7 will be supplied in the initial application for an operating license.

*7. Identification of the qualification documents which contain detailed supporting information, including test data, for items 4, 5 and 6.

In addition, in accordance with the requirements of Appendix B of 10 CFR 50, the staff requires a statement verifying that (1) all Class 1E equipment has been qualified for an operating license (OL) or will be qualified for a construction permit (CP) to the program described above, and (2) the detailed qualification information and test results are (or will be) available for an NRC audit.

*For applications for construction permits, it is acceptable to state that items 6 and 7 will be supplied in the initial application for an operating license.

Part I

Part II

Staff Responses to Public Comments

Including Appendices A Through D

PART TWO
CONTENTS

LIST OF COMMENTORS

Public comments to the "For Comment" NUREG-0588, dated December 1979, were received from those organizations and individuals listed below. The comment period was extended to May 1980 in order to factor in the majority of the comments received. A discussion of the comments and their resolutions appears in the following pages.

T. M. Anderson, Westinghouse Electric Corporation

J. T. Boettger, Nuclear Power Engineering Committee (J. T. Bauer - IEEE Standards, SC-2 Chairman)

R. H. Buchholz, General Electric Company

N. W. Curtis, Pennsylvania Power and Light Company

S. H. Howell, Atomic Industrial Forum, Inc.

W. O. Parker, Jr., Duke Power Company

H. W. Pielage, Entor Corporation

D. L. Renberger, Washington Public Power Supply System

H. C. Schmidt, Texas Utilities Services, Inc.

F. Sillag, Bailey Controls Company

J. H. Taylor, Babcock and Wilcox Company

E. E. VanBrunt, Jr., Arizona Public Service Company

G. E. Wuller, Illinois Power Company

COMMENT NO. 1: (General)

Please don't refer to cable as "equipment." Wire and cable are components which may become part of equipment but are not of themselves "equipment."

Resolution

The term "equipment" as used in this report includes all types of equipment (i.e, components, subassemblies, etc.) essential for plant safety. No attempt is made in this report to differentiate between components, subassemblies, and so forth. Cable--unlike other components such as resistors, capacitors, or wires that are integral parts of other equipment--is a unique and major item that may be qualified independently of any other component and can be treated as a specific piece of equipment.

COMMENT NO. 2A: (General)

In several places in the (Discussion) section, mention is made of valve qualification. We believe this should be "valve actuator qualification." Valve qualification including valve actuators is a recent project by the ASME which has not yet been completed.

COMMENT NO. 2B:

P3 Typo - reference to 382 should be for valve actuators, not valves.

Resolution

The staff agrees with the comments.

COMMENT NO. 3: (General)

It is not clear whether the Category I subparagraphs apply to Category II. It is, therefore, recommended that the subparagraphs applicable (if any) to Category II be individually identified.

Resolution

The staff concurs with this comment. If the main section is identified as being applicable to Category II, then all the subsections associated with it are also applicable to Category II unless otherwise noted.

COMMENT NO. 4: (General)

IEEE-323 is entitled "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." NUREG-0588 is entitled "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," yet is stated to address a method acceptable to the NRC for implementing the requirements of IEEE-323. The NRC needs to explain/define their interpretation of the difference, if any, between "Class IE Equipment" and "Safety-Related Equipment." GE is particularly concerned that "Safety-Related Electrical Equipment" may refer to non-safety grade components assumed in mitigating a transient. If our assumption is correct such an expansion of IEEE-323 is unjustified.

Resolution

Electrical equipment important to safety (that is, safety-related) is a broad category of equipment and includes the well-defined subset identified in the national standards as Class IE equipment. Equipment important to safety, however, also includes other equipment addressed in the Standard Review Plan Sections 7 and 8 such as equipment required for reactor shutdown and post-accident monitoring. In addition, certain equipment may be required and classified as important to safety because it functions as a supporting system for Class IE equipment, or simply because of its association with Class IE systems. Equipment in this latter category (for example, anticipatory trips) although not essential for accident mitigation, may be considered important to safety if by its association with Class IE equipment may render the Class IE equipment inoperable.

Recognizing that functional requirements differ for different equipment important to safety, the staff is in the process of attempting to establish several categories of safety equipment. However, until these categories are defined, the existing two-category systems (equipment important to safety, and non-safety equipment) will be used.

Mitigation of transients may be considered a safety function for which non-safety grade equipment has been and may be used, as long as it can be shown that failure of that equipment does not significantly impact the mitigation of the transient or adversely affect public health and safety.

COMMENT NO. 5A: (General)

We recognize the intent of the section and, in general, agree with it; however, in the interest of accuracy it should be noted that the IEEE Standards and Regulatory Guides referenced constitute a mixed bag that may not provide the coverage expected.

Not all of them are derived from or contain the basic requirements of IEEE 323-1974. For example R.G. 1.73 endorses IEEE 382-1972 and R.G. 1.40 endorses IEEE 334-1971. Perhaps a better approach would be to state that these older standards in combination with IEEE 323-1974 constitute the bases for an acceptable approach.

COMMENT NO. 5B:

The discussion (page 2, paragraph 5) implies that conformance with the daughter standards and endorsing regulatory guides as specified will provide assurance that the equipment being qualified meets the requirements of NUREG-0588. Not all of the specified standards are related to IEEE 323-1974, such as:

Regulatory Guide 1.73 endorses IEEE 382-1972, which is related to IEEE 323-1971.

Regulatory Guide 1.40 endorses IEEE 334-1971, which is related to IEEE 323-1971.

Furthermore, it is the opinion of Westinghouse that the qualification program recommended by the pre-1974 versions of these standards do not meet the requirements of IEEE 323-1974 and therefore, reference to these Regulatory Guides and Standards should be deleted from NUREG-0588.

Resolution

The staff concurs in part with the comments. It should be recognized, however, that when a standard which has been previously endorsed by the staff is significantly revised to reflect the "state-of-the-art" technology, a revised Regulatory Guide will follow. The staff is or will be in the process of updating and issuing revisions to the above referenced guides.

COMMENT NO. 6:
(General)

We are concerned that qualification of some components may take an extended period of time. Large or heavy components requiring testing may be subject to the restrictions inherent in the very limited number of facilities in which such testing can be performed. Industry, with need for access to such a testing facility, faces a significant extension of time before all components are tested and qualified. Early implementation of the staff philosophy espoused in NUREG-0588 would have a significant impact on the issuance of CPs and OLs for those facilities awaiting component qualification.

Resolution

The staff has been implementing, in part, the positions in NUREG-0588 through its endorsement of related Regulatory Guides and individual positions on a case-by-case basis for quite some time. Therefore, on plants that are currently under review for a construction permit (CP) or operating license (OL) applications, the long lead times for qualification purposes should have been accounted for.

Recognizing that there may be equipment for which qualification may not be completed by the time a plant is ready to start up, it is incumbent on the applicant to provide justification of the adequacy of the existing design on a short-term basis until the qualification program is complete.

Methods such as ongoing qualification may be designed to resolve the long-term qualification programs. Other methods (than those described in NUREG-0588) that are designed to satisfy the requirements of the General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50 may be proposed, and may be found acceptable on a case-by-case basis.

COMMENT NO. 7A:
(General)
(Category II)

NUREG-0588, particularly as it applies to IEEE 323-1971, is not a reasonable interpretation of the standard. The NUREG, in actuality, extends the standard into new areas rather than interpreting existing criteria. In three specific areas (aging, margin and qualification by analysis) the NUREG has either added to or deleted from the standard. Neither the words nor the intent of aging and margin have ever before been included as part of IEEE 323-1971. To include them at this time is to revise the standard nine years after its issuance and to negate the actions of the NRC and the work of the nuclear industry during that period.

COMMENT NO. 7B: (Category II)

There are a number of substantially completed plants that will be affected by NUREG-0588, from those with construction activities well advanced to those in the "near term operating license" category. Changes in qualification and documentation requirements have significant cost and schedule impacts on such plants. We strongly question the benefit of across-the-board application of the document in its current form, especially in regard to plants committed to meeting IEEE 323-1971 (Category II). These plants are currently being handled on a case-by-case basis in this area, as is appropriate. Changes in requirements should only be made where there is demonstrable significant additional protection of public health and safety.

Resolution

It is not the intent of the NUREG to interpret IEEE Standard 323-1971 but rather to supplement it and to focus attention and activity on areas where additional improvements and guidance in qualification are deemed essential to satisfy the applicable General Design Criteria of Appendix A to 10 CFR Part 50.

Although the 1971 version of the standard does not uniquely identify aging and margin as parameters that have to be addressed with any defined degree of rigor, aging has been a requirement in the national standards for selected equipment since 1971 and has been incorporated in other ancillary standards since that time. Providing margins during testing (or when analysis is used) has always been considered standard and good engineering practice to ensure that the design conditions under consideration have been enveloped.

Therefore, the staff does not agree with the suggestion that the positions for Category II should be omitted on the basis given. Implementation and the degree of conformance of these positions will be evaluated on a case-by-case basis. On the older plants, backfitting an acceptable degree of conformance to the positions will be made where it is demonstrated that additional assurance is warranted.

COMMENT NO. 8: (General)

The introduction to the document states that the staff position developed prior to the TMI-2 event and any additional requirement or modifications will be identified later. This position is unacceptable from the standpoint that the data available today (Reference: "Technical Staff Analysis Report on Alternate Event Squences to President's Commission on the Accident at Three Mile Island,' by William R. Stratton, et al., October 1979, Washington, D. C.) demonstrates a significant difference in airborne activity available for release from the containment from those assumed by the staff in a DBA. The Kemeny Report notes that the evaluation of the consequences o reactor accident have, in the past, been dominated by the iodine doses. TMI-2 demonstrates that in this type accident, at least, those estimates have been grossly and conservatively pessimistic. The difference between the design basis LOCA and the Kemeny Report notes that the evaluation of the consequences of reactor accident have, in the past, been dominated by the iodine doses. TMI-2 demonstrates that in this type accident, at least, those estimates have been grossly and conservatively pessimistic. The difference between the design basis LOCA and the Kemeny Report estimates range up to three orders of magnitude. Such a range would have a significant impact on the qualification of components. Prior to the implementation of the staff qualification program, these differences need to be resolved since there appears to be a wide difference between the assumed NRC source terms and the White House approved Kemeny Report estimated values.

Resolution

The staff agrees that its consequence calculations for the site boundary and the low population zone have been dominated by conservative estimates of airborne iodine concentrations. However, for inside containment, the noble gas contribution to the gamma and beta doses is substantially greater and therefore dominates any dose contribution resulting from airborne radioactive idoine. (Refer to NUREG-76-6521 and Sandia Report No. 78-0091 for additional information.)

The NRC staff recently prepared two reports, NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," and NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions." These studies reflect not only the TMI-2 accident experience but also the results of recent research and improved methods of analysis. The findings of these studies will be factored into the rulemaking. In the interim, the source terms for equipment qualification shall remain as defined in position 1.4(1).

COMMENT NO. 9A: (Section 1.2)

In Section 1.2(5) the staff position restricts the use of the calculation model (Appendix B) to deriving the peak surface temperature. This is unnecessarily restrictive in that the item of interest is the temperature of the critical components inside the equipment under test as compared to estimated temperatures under DBE conditions. Westinghouse believes that the method documented in WCAP 8936 continues to be valid and conservative. The following change to Item 1.2(5) is, therefore, recommended:

(a) "Show that the peak internal temperature of the component to be qualified does not exceed the LOCA qualification internal temperature using the method discussed in Item 2 of Appendix 2 as a boundary condition."

(b) "If the calculated internal temperature. . ."

COMMENT NO. 9B: The many comments on this item (1.2(5)) questioned the wisdom of stressing surface temperature and of the lack of adequate guidance on modeling intrinsic heat capacity in order to obtain a measure of the thermal lag and the resultant effect on the internals of the equipment. A suggested rewording is as follows:

(b) "...or show that the peak internal temperature of the component to be qualified does not exceed the LOCA qualification internal temperature using the method of Appendix B, Item 2 as a boundary condition."

(c) "If the calculated internal temperature (or the calculated surface temperature if internal temperatures are not calculated)..." SC 2 assumes the methodology of Appendix B has an auditable basis.

COMMENT NO. 9C: In Section 1.2(5)(b), the main point of the analysis should be to show that critical internal components do not reach higher temperatures during MSLB than during LOCA. Ideally, the surface temperature is indicative of the internal temperature. This may not apply under the required analysis method, leading to an erroneous conclusion.

COMMENT NO. 9D: This Section (1.2(5)(b) and (c)) indicates that if the calculated surface temperature exceeds the qualification temperature, the component must be requalified or protection must be provided. The qualification temperature should be that which applies in the critical part of the component and not the surface temperature of the component. The peak surface temperature may exceed the required qualification temperature but the component would still function correctly. Furthermore, time-at-temperature is an important consideration which should be factored into any qualification evaluations.

COMMENT NO. 9E: The requirement 1.2(5) should be revised to allow component testing for steam line break environmental parameters as an option to analysis utilizing what are judged to be overconservative heat transfer coefficients given in Appendix B, Item 2.

Resolution

An important consideration in qualifying a piece of equipment is the identification of the various failure modes of the component. This information is necessary prior to the determination of the critical element or elements. For components containing only a few elements with symmetrical geometry the above determination may be achievable. For more complex components, however, all failure modes may not be identifiable. As a result of the difficulty in identifying failure modes, surface temperature was selected as a generic parameter. If additional information can be provided to ensure that the specific failure modes can be identified and justified, then consideration may be given on a case-by-case basis to the use of a temperature other than at the surface of the equipment. Internal component temperature would be considered only on a case-by-case basis for Category II equipment, if supporting justification can be provided. Component testing for qualifying to the steam line break environment certainly can be performed using the actual approved temperature profile. However, in addition to using the correct pressure-temperature profile, the containment turbulence and air content must be properly taken into account. (See also staff response to Comment No. 58.)

COMMENT NO. 10: (Section 1.2)

In item 1.2(5)(a), Category I requires calculation of the envelope of peak temperature for MSLB while Category II requires only a single point (this is being inferred) peak temperature based on different ground rules.

Why is there a difference in requirements for the categories?

In item 1.2(5)(c) the ordering of the items listed was changed from those given for Category I to those for Category II. Is there any reason or significance to be associated with this?

Resolution

The intent of Section 1.2(5)(a) Category II is to require a calculation for the peak temperature envelope (not a single-point calculation).

With regard to the second point, Section 1.2(5)(c) for Category I requires that testing be the principal qualification method. These plants are in the early stages of design and have the opportunity for such equipment qualification testing at the anticipated bounding design conditions. Category II, which applies to near-term operating license (NTOL) applications and operating reactors, recognizes the vintage of the equipment and allows additional justification to be provided, by analytical means, to demonstrate that the equipment can maintain its required functional operability if the calculated MSLB temperature at or near the surface of the equipment exceeds the LOCA test temperature (for which the equipment has already been qualified). This type of qualification has been and will be applied on older NTOL plants.

COMMENT NO. 11: (Section 1.1)

Consider referencing CSB BTP 6-1 for one manner in determining LOCA qualification temperature. Consider surface thermocouple measurements as one way for determining LOCA qualification temperature. Consider mentioning that if the component is temperature soaked in a LOCA qualification test for a period of time resulting in justifiable quasi-equilibrium temperature conditions (example: 3 or 4 temperature time constants) then the LOCA qualification temperature would be equal to the test chamber temperature.

Resolution

Other methods may be used to measure surface temperature of the component provided that it can be shown that thermal equilibrium exists in the test chamber and at the equipment under test. Simulating and monitoring the rise time of the temperature transient should not be ignored. (See also staff response to Comments No. 9 and 58.)

COMMENT NO. 12: (Section 1.1) Section 1.2(2) implies that Appendix A contains the only models acceptable for calculating contain ment environmental parameters. The NRC should clarify that other models are also acceptable if approved by the staff.

Resolution

Sections 1.1(1) and 1.2(1) state that other models approved by the staff may be found acceptable.

COMMENT NO. 13: (Section 1.3) The containment spray system is not the only source of chemicals under high energy line break conditions; boric acid should also be addressed. The following change to Item 1.3 is therefore, recommended:

The sentence:

"The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system."

Should be changed to:

"The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system, during both the initiation and recirculation phases."

Resolution

The staff concurs. This change will be considered in the proposed rulemaking and/or the revision to Regulatory Guide 1.89 to be issued for public comment in December 1981.

COMMENT NO. 14: (Section 1.3) Any chemical change resulting from a malfunction of equipment would be addressed by spraying during testing with the required solution or one that correctly simulates its effects. It would not be necessary to use a different solution as its effects could be different than those of the actual solution.

Resolution

The staff concurs. If a spray solution is used in a simulated test--the effects of which could be different to those provided by the solution in the actual plant--the testing would be considered unacceptable. However, if a more concentrated form of the solution is used during testing as a bounding condition, the adequacy of such testing, if justified, may be found acceptable. No change was proposed as a result of this comment.

COMMENT NO. 15: (Section 1.3) Add words "where applicable" to this article.

Resolution

The staff does not agree that the position should be modified to include these words. The applicability of all the positions in the text is a function of the design. As indicated in the discussion, alternatives (or exceptions) may be proposed and, if justified, may be found acceptable.

COMMENT NO. 16A: (Section 1.4) (Category II) The concept of a qualified life is not a requirement for Category II plants. The following change is, therefore, recommended:

The words:

"over the equipment qualified life"

should be deleted from the first sentence at Section 1.3 for use in the Category II column.

COMMENT NO. 16B: (Category II) By statement in Section 1.4, "qualified life" is not applicable to Category II. The statement "over the qualified life" should be deleted from 1.4 for Category II.

Resolution

With the exceptions noted in Section 4(1) of the NUREG, the staff does not require that a qualified life be established for all Category II equipment.

The words "qualified life" may be interpreted as "installed life" for Category II equipment.

COMMENT NO. 17A: (Section 1.4(1)) In item 1.4(1), the LOCA source term utilized should reflect the multi-level term in the proposed revision to Regulatory Guide 1.89, Revision 1, November 1, 1976. For non-LOCA accidents, gap release from 100% of the fuel rods is the principal basis and, in accordance with NUREG/CR-0091, a total release fraction of 1-2% of the noble gases is sufficient.

COMMENT NO. 17B: The staff position in Item 1.4(1) of requiring 100% of the gap activity (approximately 10% LOCA) released to the containment for all other non-LOCA design basis accidents cannot be justified since:

a. Many design basis accidents do not result in a breach of either the primary or secondary systems. Thus, for equipment that is only required to protect against such contained faults, the application of any accident related dose is illogical and unnecessary provided that the equipment can be shown to have no adverse effect under high energy line break conditions, as required by Item 2.1(3)(d) and Appendix E.

b. For equipment that is only required to function following a secondary side break, the application of the dose that would result from the release of 100% of the gap activity is grossly conservative. Westinghouse dose calculations have conservatively assumed 1% clad damage (1% gap activity release) and considering the fraction of the core activity in the RCS as 0.003 Kr-85, 0.001 halogens, and 0.001 of other noble gases. It was also conservatively assumed that all of RCS inventory was instantaneously released into the containment atmosphere at the initiation of the incident. This method is documented in WCAP-8587 Section 6.8.4. The following change to item 1.3(1) is therefore, recommended:

(1) "The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For secondary side break design basis accident conditions, a source term involving an instantaneous release from the fuel to the containment atmosphere of 0.1 percent of the noble gases (except for Kr-85 for which a release of 0.3 percent should be assumed) and 0.1 percent of the iodines is acceptable. For design basis accidents that do not result in a breach of either the primary or secondary systems only the normal operational dose need be considered.

COMMENT NO. 17C: In Item 1.4(1) the position required excessively conservative and unrealistic assumptions in determining the source terms for design basis accidents. For all design basis accidents, any core damage and the subsequent release of radioactive material will not occur instantaneously, but instead will occur over some period of time. Consideration of time dependent release of radioactive material should be permitted in the determination of accident radiation environments.

The use of approximately 10% of the LOCA source terms for all other non-LOCA design basis accidents has no apparent basis and is overly conservative. Equipment which is required to function during or after a non-LOCA design basis accident need only be qualified to the radiation environment resulting from that accident (with adequate margin).

It is recommended that an additional sentence be added to this position as follows:

"The time-dependent release of radioactivity and the use of alternate source terms may be found acceptable when supported by conservative analysis for the specific accident of concern."

COMMENT NO. 17D: Appendix D should reflect the multi-level source term as reflected in the proposed revision to R.G. 1.89, Revision 1, November 1, 1976.

COMMENT NO. 17E: Appendix D provides "Sample Calculations and Type Methodology for Radiation Qualification Dose." In the section on the basic assumptions used in the analysis it is stated that "between 20 and 80% of the fuel rods could experience cladding failure for a PWR and a lesser fraction for a BWR." The current GE licensing basis model calculates that fuel perforations occur only beyond 15,000 to 20,000 MWD/T exposure and then only in the high power bundle. Our best estimate model does not calculcate any fuel perforations for a LOCA. In determining the source term to be used for equipment qualification, the vendor should be able to use as a basis his staff approved Appendix K model in determining the number of rods which are calculated to have failed. This comment also pertains to the statements in Section 1.4(1) of NUREG-0588.

Resolution

An NRC-sponsored research effort is investigating the use of a multi-level and time-dependent release of radioactivity from the fuel following the design basis accident LOCA. Until the results of this effort are available, the staff will continue to use the source terms presented in this interim report. The final rulemaking will factor in the results of any additional findings identified in these ongoing investigations.

The staff maintains the position that in some non-LOCA accidents--in particular those that are power-increasing transients--the inventory in the fuel rod gaps may be larger than predicted by NUREG/CR-0091. Therefore, to be conservative, the value of 10% of the rod inventory in the gaps will be retained. The staff agrees with the comment that the 100% cladding failure assumption may be overly conservative. As part of the proposed rulemaking, the staff is considering using the conservatively calculated estimates of fuel damage for non-LOCA transients instead of using the current assumption.

In response to comment 17E, the source terms in position 1.4(1) are not based on best estimate calculations. The fuel damage estimates in Appendix D were intended to show the amount of conservatism provided by position 1.4(1). The degree of conservatism, however, does not appear to be quite as high as initially envisioned in light of the data of the TMI-2 accident. To avoid misinterpretation of this intent, the discussion of the best estimate models of fuel damage following a DBA has been deleted from Appendix D.

COMMENT NO. 18: (Section 1.4)

In Item 1.4(1), the fission product release assumptions to the containment atmosphere are different from those which have been traditionally used in Regulatory Guide 1.3 for the design basis accident. Furthermore, we have not assumed, nor in the past has the staff assumed, the fission product releases identified in NUREG-0588 for "all other non-LOCA design basis accident conditions." For the fuel drop accident, it appears that the NUREG is inconsistent with the Regulatory Guide 1.25. The document needs to be modified to reflect currently approved fission product transport models.

Resolution

The values for the iodine and noble gas portions of the source term used in NUREG-0588 are identical to the source term identified in TID-14844, which is also the starting point for the source-term assumptions for Regulatory Guides 1.3 and 1.4. The staff is currently evaluating the adequacy of the source terms in light of the TMI-2 event, and will factor in the results of their study in the final rulemaking to be issued to public comment in December 1981. One significant finding of the TMI-2 event is the significant amount of cesium in the coolant (between 40 and 60%). The incorporation of cesium in the source term appears to be a more appropriate treatment of fission products in the coolant (other than iodine) in addition to the current assumption of the 1% solids.

The comments about the source-term assumptions for all other non-LOCA accidents have been addressed in response to Comment No. 17.

COMMENT NO. 19: (Section 1.4 and Appendix D)

Paragraph 2 of Appendix D should be rewritten to state clearly the basic assumption of 100% fuel clad failure. A simple statement that the source term is that given in position C2 of Regulatory Guide 1.89 is appropriate. It is suggested that all wording in the first paragraph after the words "core meltdown" be deleted.

Resolution

Position 1.4(1) and Appendix D describe the current staff positions on source terms for equipment qualification.

See also the response to Comment No. 17 for source terms for non-LOCA accidents.

COMMENT NO. 20: (Section 1.4(1)

A conflict exists between the postulated source term values in NUREG-0588 and NUREG-0578 (TMI Short Term Lessons Learned). The use of NUREG-0578 source terms will result in even higher values than those presently given in NUREG-0588.

Resolution

The "For Comment" version of NUREG-0588 provided the methods for determining the radiation source term when considering LOCA events inside containment (100% noble gases/50% iodine/1% particulates). These methods considered the radiation source term resulting from an event which completely depressurizes the primary system and assumes the release of the source term inventory instantaneously to the containment.

The "For Comment" version of NUREG-0588 also provides the radiation source term to be used for qualifying equipment following non-LOCA events both inside and outside containment (10% noble gases/10% iodine/0% particulates).

NUREG-0578 provided the radiation source term to be used for determining the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment as a result of LOCA. This method considered a LOCA event in which the primary system may not depressurize and the source term inventory remains in the coolant.

The apparent conflict between NUREG-0588 and NUREG-0578 has been resolved and reported as clarification for item II.B.2 in NUREG-0737. The incorporation of all source term assumptions for equipment qualification will be provided in final rulemaking to be issued for public comment in December 1981.

Reduction of the noble gas contribution in the source term (assumed in the reactor coolant system per NUREG-0578) may be warranted for those designs (systems) where the primary coolant system is depressurized before the reactor coolant flow through these designs (systems) is initiated (for example, the residual heat removal system outside containment).

COMMENT NO. 21: (Section 1.4(1))

(1) In recent requirements imposed by the NRC on the Near Term Operating License Plants, the staff has required a change in the assumptions used for the calculation of post-accident radiation dose for equipment internal to the RCS.

(2) In recent drafts of Regulatory Guide 1.97, the staff is requiring a much lengthened post-accident monitoring time period.

These current staff positions should be included in NUREG-0588 and issued for comment, as part of the NUREG.

Resolution

The staff concurs with item no. 1. Position 1.4(1) contains the current staff requirements for source terms for equipment inside the containment. Regarding equipment internal to the RCS, the source terms to be used have been provided to the Near Term Operating License (NTOL) plants as clarification to Item II.B.2 of NUREG-0737. (See also response to Comment No. 20.)

With regard to item 2, Regulatory Guide 1.97 provides guidance for instrumentation used to assess plant conditions during and following an accident. A limited number of instruments covered by Regulatory Guide 1.97 are to be designed for worst-case conditions, that is, total core meltdown.

The radiation source terms in NUREG-0588 represent a partial core meltdown and should not be used for that limited group. Use of the staff positions in NUREG-0588 for the qualification of the remaining instrumentation covered by the Regulatory Guide depends on an individual functional requirement for each instrument. This determination should consider the type of accident, the function of the instrument during and following that accident, and the portion(s) of that instrument located within a harsh environment caused by the accident. The specific positions for postaccident monitoring are provided in Regulatory Guide 1.97 and are outside the scope of this generic report. Final rulemaking, however, will address the postaccident monitoring requirements.

COMMENT NO. 22: (Section 1.4) Item 1.4.14 on page 10 states that qualification levels given in Appendix D are adequate. However, the Appendix D analysis ignores the normal operation dose which is required in Item 1 on page 7. This should be resolved.

Resolution

Section 1.4 requires that the qualification dose should be the sum of the normal and accident doses. Appendix D addresses only the accident doses. It should be noted that the dose values in Appendix D are provided for illustrative purposes and they may not be appropriate for plant-specific application. Any modifications to position 1.4 (14) will be incorporated in the final rulemaking to be issued for public comment in December 1981.

COMMENT NO. 23: (Section 1.4) In Section 1.4(1) it is stated that 1% of the remaining fission products are released instantaneously to the atmosphere. In contrast, Section 3 of Appendix D ignores these other fission products when determining the airborne sources. Elsewhere in Appendix D, it is stated that these other fission products are released instantly to the sump fluid at T=0. We recommend that this inconsistency be resolved with the other fission products being released to the sump fluid only.

Resolution

The staff agrees with the comment. The intent of the position is that the 1% solids are assumed to be instantaneously released from the fuel to the coolant and are carried by the primary coolant to the containment sump. See also response to comment No. 18.

COMMENT NO. 24: (Section 1.4) In two places of the wording Section 1.4(1) "... instantaneous release from all the fuel..." is suggested for clarity.

Resolution The staff agrees with the comment. The suggested accommodation will be considered in the final rulemaking.

COMMENT NO. 25: (Section 1.4) In Section 1.4(1) the requirement to assume an instantaneous, non-mechanistic release of activity from the fuel is inconsistent with the time-dependent, mechanistic approach required for radioactivity redistribution analyses in containment and auxiliary building volumes. As briefly discussed in Appendix D to NUREG-0588, any core damage and subsequent release of activity will require a significant amount of time which would depend on the accident scenario. Since this NUREG is establishing more realistic and rational bases for estimated radioactivity levels after release from the fuel the same approach should be applied to fuel releases themselves. This time-dependent fuel release fraction is particularly significant for equipment which is required to function for only a short time following a LOCA/MSLB. Enforcement of this requirement will cause significant equipment replacement for Category II plants. We do not believe enforcement of this position can be defended on a cost/benefit basis.

Resolution

See the staff response to Comment No. 17 regarding time-dependent fuel releases.

Regarding the comments on equipment replacement: Category II plants in NUREG-0588 (applicable to equipment qualified in accordance with IEEE Standard 323-1971) should already have qualified equipment using the source terms previously acceptable to the staff (that is, instantaneous fission product release per Regulatory Guide 1.3 and 1.4). In areas where the "solids" contribution is significant, equipment requalification may be warranted unless appropriate justification is provided to demonstrate the adequacy of previous qualification methods (such as the use of shielding, and so forth).

For plants that did not qualify equipment using the source terms acceptable to the staff, equipment replacement or requalification may also be warranted, unless the adequacy of the qualification methods used is justified on some defined basis.

COMMENT NO. 26: (Section 1.4)

(1) Section 1.4, "Radiation Conditions Inside and Outside Containment" does not indicate that Appendix D is a sample calculation for a PWR, but that the general approach is applicable to BWR. Additional wording to this effect would enhance the clarity of the section.

(2) Appendix D itself should indicate very early on that it is a sample calculation for a PWR.

Resolution

The staff concurs with the comment. This recommended change will be considered in the proposed rulemaking and or revision to Reg. Guide 1.89 to be issued for public comment in December 1981. In addition, Appendix D was modified to include the assumptions for modeling radiation environments for BWR (as well as the PWR) containments.

With regard to Item 2, paragraph 3 of the revised Appendix D clearly indicates that the numbers presented are strictly the results of a sample calculation for a PWR which uses the methods and assumptions in the appendix.

COMMENT NO. 27A: (Section 1.4(3)) With respect to 1.4(3) GE does assume uniform distribution of activity throughout the containment at time 0. The mechanistic treatment of fission product transport for the non-mechanistic accident event is thought to be a significant deviation from past staff acceptances of the GE design. The staff needs to provide greater explanation for the need for this change.

COMMENT NO. 27B: In Appendix D(3), the multi-mode mechanistic fission product transport mechanisms should not be considered in the analysis of the BWR. It was thought that by using the conservative real world source terms, such refinements as defined on page D.9 could and should be avoided.

Resolution

Appendix D was modified to provide the appropriate assumptions for distribution of activity within the containment, following the accident, for both PWRs and BWRs. See also response to comment No. 28.

COMMENT NO. 28: (Section 1.4(3)) The last sentence of 1.4(3) appears to be in conflict with Appendix D.8.a which appears to restrict the assumption to PWR while 1.4(3) restricts it in all cases. Which is correct?

Resolution

The intent of Position 1.4(3) is to prevent a situation where an assumed uniform distribution of activity throughout the entire containment could result in a nonconservative estimate of the qualification dose or dose rate. This position by itself does not preclude the use of a uniform distribution when that assumption is appropriate. Section 2.2 and 2.3 of the revised Appendix D provides appropriate assumptions for the initial distribution of activity inside containment.

COMMENT NO. 29: (Section 1.4(5)) GE analysis considers radiation at the centerpoint of any given compartment rather than the NUREG position which specifies radiation at the centerpoint of containment It seems unnecessarily conservative not to account for the presence of internal structures.

Resolution

The staff does not preclude the dose calculation within compartments. Further, the staff position 1.4(6) allows for reduction of the calculated beta and gamma doses if the dose point is such that internal structures or shielding contribute to the reduction of the dose. However, any claims for reduction in the doses due to either internal structures or shielding must be clearly documented and justified.

COMMENT NO. 30: (Section 1.4(7))

With respect to Section 1.4(7) and 1.4(9) in the NUREG, the GE analyses for radiation are based on a semi-infinite medium analagous to Regulatory Guide 1.3 - doses to people. The staff is apparently taking the position that an infinite concept is unacceptable.

Resolution

The use of the infinite cloud assumption is in connection with the airborne beta radiation dose only. The assumed dose point on the containment centerline is surrounded on all sides by the containment atmosphere thus an infinite cloud assumption is appropriate. Also, positions 1.4(7) and 1.4(9) do not preclude the use of a semi-infinite cloud assumption if it can be adequately justified.

COMMENT NO. 31: (Section 1.4(7) Appendix D)

Any justification of the assumption in Appendix D Section 7(b), that, "all betas directed toward the coating were assumed to be absorbed in the coating," would be analytically difficult. We feel that it would be more appropriate for the actual beta dose at a designated depth to be evaluated; the 10-mil depth where adhesion occurs would probably be most appropriate.

COMMENT NO. 32:

Also in Appendix D, Section 7(b), the method of dose evaluation to be applied to cable insulation layers is vague. Is it intended that the total absorbed energy be distributed throughout the mass of the insulation or that the dose determined for the coating be applied to the entire cable insulation? The first method would underestimate while the second would be an overestimate. Once again, we recommend that it would be more appropriate to determine the actual beta dose at a predetermined critical depth. It should also be noted that item 1.4.9 on page 9 implies that the beta dose from plate-out on cables can be ignored, but this contradicts item 1.4.7 and page 9.

Resolution

The doses calculated using the methods of Appendix D are estimates at the surface of the equipment. The staff does not wish to use the approach of specifying a dose at a predetermined critical depth because the critical depth doses are dependent on the absorbing materials (which is different for different equipment).

The beta dose from plate-out on cables cannot be ignored. The intent of position 1.4(2) is to explicitly require the consideration of all radiation sources when calculating the qualification doses, which includes the beta dose from plate-out sources on cables.

COMMENT NO. 33: (Section 1.4(7) Appendix D)

In Section 7(b) it is stated that the gamma dose for coatings due to plate-out is negligible because the absorbed dose in the coatings is small. Since the purpose of the model in Appendix D is to determine the radiation environment to which the coatings should be subjected in qualification tests rather than the absorbed dose in the coatings, the gamma dose in Rads (C) or Rads (air) should be determinted on the basis of the total dose due to both airborne and plated-out sources at the surface of the coatings.

Resolution

The staff concurs. Position 1.4(2) shall be interpreted to include all potential sources when calculating qualification doses (which would include both airborne and plated-out sources).

COMMENT NO. 34: (Section 1.4(9))

The argument given in Section 1.4(9) for reducing, by a factor of at least 2, the beta dose for qualification of cables arranged in trays based on localized or self shielding effects can be extended to other components. Any exposed components will be sufficiently massive to attenuate beta radiation from the containment atmosphere on the opposite side. Hence, the beta dose at the surface of unshielded equipment should, in general, be half the beta dose calculated at the containment center.

Resolution

Implementation of this assumed dose reduction may be warranted for a piece of equipment in a specific location. Sufficient justification should be presented to warrant the above-mentioned reduction in calculated beta doses and should be evaluated on a case-by-case basis. A beta dose reduction due to shielding from large internal structures is acceptable.

COMMENT NO. 35A: (Section 1.4)

With respect to Sections 1.4(7), (8), (10), (14), requiring qualification to beta radiation at this date for Category II could be traumatic and needlessly so. We recommend that the requirement be tied to results of the IEEE 323-1974 qualification programs such that material which evidences adverse beta effects be addressed or replaced for Category II.

COMMENT NO. 35B:

With respect to item 1.4(7), (8), (9), (10, (14), the qualification for the effects of beta radiation was not a requirement for Category II plants and systematic enforcement of this requirement at this stage will have major impact. A reasonable alternative would be to require that any significant adverse experience gained during qualification testing of equipment, for the effects of beta radiation, to IEEE 323-1974 be considered for applicability to Category II plants.

Resolution

The staff agrees with the comment. Any modification to the positions will be considered during the final rulemaking to be issued for public comment in December 1981.

It is the staff's belief that the qualification dose should account for all types of radiation present at the equipment location. The staff permits the reduction of calculated beta doses to account for localized shielding (that is, component and/or structural shielding) and has provided additional guidance in the DOR guideline document. When a significant beta dose reduction can be justified, the staff expects the equipment qualification dose to equal or exceed the gamma radiation dose calculated using assumptions and models similar to those in Appendix D. For any safety-related component not meeting the calculated qualification values, justification for the adequacy of the design should be provided, or a modification of the design to satisfy the above radiation requirements may be warranted. Replacement equipment or equipment that has not yet been qualified should conform to the Category I requirements.

COMMENT NO. 36: (Section 1.4(10)) Position 1.4(11) discussed the need to consider the exposure received by ECCS equipment located outside containment from sump fluids. This need is understood but the reference to Appendix K to 10 CFR Part 50 is unclear This reference should be made more specific or removed from the position.

Resolution

The staff agrees with the comment. Any modification to the positions will be considered during the final rulemaking.

COMMENT NO. 37A: (Section 1.4(11)) The implication that radiation qualifications must be performed for low level doses (no matter how small) on all equipment (no matter how radiation tolerant) is unfortunate. Surely some guidance can be given of a more practical nature for equipment located in regions of trivial integrated dose (present consensus is that the threshold of triviality occurs at approximately 1×10^4 rads).

COMMENT NO. 37B: The implication that radiation qualification must be performed for low level doses (no matter how small) on all equipment (no matter how radiation tolerant) is unfortunate. Surely some guidance can be given of a more practical nature for equipment located in regions of trivial integrated dose (present consensus is that the threshold of triviality occurs between 5×10^3 and 1×10^4 rads). The guidance might take the form of the tabulation in Appendix C of the NRC's recent "Guidelines for Evaluating Qualification of Class IE Electrical Equipment on Operating Reactors," November 1979.

COMMENT NO. 37C: This position (item 1.4(11)) requires that test data are required to exempt equipment from radiation qualification, even if the integrated dose is less than 10^4 rads. There is no apparent basis for this requirement. Numerous tests have provided data that show that radiation damage thresholds are greater than 10^4 rads. To require radiation testing below 10^4 rads will only add significantly to the cost of testing programs without adding to plant safety. It is recommended that the position be rewritten as follows:

Equipment that shall not be exposed to an integrated dose greater than 10^4 rads may be exempted from radiation qualification.

Changes to this position should be made for specific materials if any, with a radiation damage threshold below 10^4 rads, are identified.

COMMENT NO. 37D: Inclusion of equipment qualification testing for equipment with radiation doses below 10^4 rads would require substantial expenditures of time and money for qualification testing with no corresponding benefit to health and safety. Of the general classes of materials or components (organic compounds, ceramics, metallics, electronic components), only organic compounds and electronic components are susceptible to damage from moderate amounts of gamma or beta radiations. Numerous studies have compiled radiation effects data on all the classes of organic compounds and show that the least radiation resistant compounds have damage thresholds greater than 10^4 rads and would remain functional with exposures substantially above the threshold value. Thus, for organic materials, an exposure level of 10^4 rads is reasonable threshold value below which proper qualification is assured without adding the substantial costs of testing.

For electronic components, studies have shown failures of metal-oxide-semi-conductor devices at 3.5×10^3 rads. Therefore, a lower minimum qualification value should be assigned probably in the range of 1×10^3 rads. This would also provide adequate margin for safety without an unreasonable qualification test requirement.

Resolution

It is not the staff's intent to imply that testing is the only acceptable method for demonstrating qualification adequacy of equipment that will be exposed to low-level radiation. Other methods (such as analysis), if they are supported by test data, literature search (on identical or sufficiently similar material and/or equipment where extrapolation of the data to the actual equipment being qualified is feasible), or operating history (if it is supported by test data) may also be found acceptable.

The staff concurs that there may be information available to indicate that many materials used today have radiation dose and dose rate damage thresholds greater than 10^4 rads. However, there are also components that may be made of materials susceptible to low level radiation dose and dose-rate damage (for example, Telecon TFE and integrated circuits). Therefore, low-level radiation should not be dismissed on a generic basis and should be evaluated on a case-by-case basis.

COMMENT NO. 38A: (Section 1.4(11)) (Category II)

It has long been the industry practice to ignore radiation in zones where the total dose was less than $10^3/10^4$ rads. To impose this requirement without technical basis is arbitrary and without adequate cost/benefit consideration.

COMMENT NO. 38B: (Category II)

Enforcement of this requirement for Category II plants will have a major impact. The systematic addressment of radiation doses below 10^4 rads constitutes addressment of low-level inservice radiation aging effects, which was never a requirement for these plants. Westinghouse believes this effort cannot be justified on a cost/benefit basis and should be limited to consideration of any radiation sensitive materials identified during qualification of equipment to IEEE 323-1974.

Resolution

See staff response to Comment No. 37.

COMMENT NO. 39: (Section 1.5(3))

Where Class IE equipment is served by redundant environmental support systems, such as the main control room, this section should not be interpreted to mean the loss of both redundant support systems.

Resolution

If the redundant systems are separate and independent so that a single failure will not render both systems inoperable, the interpretation of the comment is correct. There may be designs and/or procedures, however, that may shut down both redundant and independent environment support systems during plant outages. For those designs, a loss of both redundant support systems should be assumed.

COMMENT NO. 40: (Section 1.5(2)) (Category II)

For equipment not subject to a design-basis-event accident environment, documentation of environmental qualification to the limits of normal and abnormal environments was not required for plants committed to IEEE 323-1971. Rather, equipment specifications included such environmental limits to be considered in the design and purchase of the equipment. A requirement to document qualification by test or analysis would constitute a major impact for Category II plants.

The following change is, therefore, recommended:

The word "qualified" in both subparagraphs should be changed to "qualified or designed" for use in the Category II column.

Resolution

IEEE Standard 323-1971 states that the service conditions required to be addressed include "Environmental conditions expected as a result of normal operating requirements, expected extremes in operating requirements (i.e., abnormal environments) and postulated conditions appropriate for the design basis events of the station." Therefore, the staff does not agree with the conclusions reached in the comment.

The main purpose of qualification is to verify the performance adequacy and/or the capability of a design. A specification alone does not provide this verification. A purchase specification supported by a certificate of compliance which is based test or test and analysis could constitute acceptable qualification documentation. It should also be noted that the position in Section 2.1(4) does not limit the qualification to only test or analysis.

COMMENT NO. 41: (Section 1.5)

Section 1.5(1) requires that equipment located in areas that could be subjected to high energy pipe breaks (HEPB) should be qualified to the condition resulting from the accident for the durations required.

Comment: Only that equipment necessary to mitigate the consequences of the postulated HEPB accident need be qualified to the respective HEPB conditions.

Resolution

The staff concurs in part. If any equipment failure resulting from an HELB will be detrimental to safety (even though this equipment is not necessary for mitigating the consequence of the accident), that equipment should also be qualified to the HELB conditions (see Section 2.1(3)).

COMMENT NO. 42A: (Section 1.5)

There is a significant lack of evidence of consideration of the "systems analysis method" required in the guidelines accompanying IE Bulletin 79-01B. This is especially notable in the treatment of high energy line breaks (HELB) outside containment. The words in the introduction to the NUREG indicate that all equipment is required to meet the worst environments resulting from all events. The NRC Branch Technical Position on HELB outside containment clearly indicates that only that equipment required to mitigate the HELB, that is to achieve safe plant shutdown, is required to be qualified to the HELB environment. The introduction and the body of the NUREG (e.g., paragraphs 1 and 2 of Section 1.5) should be revised accordingly. We suggest the Supplement to IE Bulletin 79-01B provides some clarification in this area.

Along the lines indicated above, analytical approaches to determine HELB environment should be clearly identified. HELB outside containment, in some cases, is calculated in a different manner from HELB inside containment. Longer time frames and multicompartment steam migration can be considered, and accordingly, different computer codes are often used.

COMMENT NO. 42B: The acceptable methods referred to in Appendix A are all used for PBIC analysis of DBA LOCAs with ECCS. These may not be reasonable for equipment qualification purposes, especially outside containment.

Resolution

The staff concurs; the techniques to calculate the environmental parameters should employ plant specific models reviewed and approved by the staff. The reference to Appendix A for outside containment qualification purposes will be modified in the final rulemaking. (See also staff response to Comment No. 41.)

COMMENT NO. 43A: (Section 2.1(2)) The statement in Section 2.1(a) is not strictly true. It should be changed to read as follows:

Second sentence - "Experience...without test data may not be adequate..."

Third sentence - "In general,...size limitations, (b) ..., (c) capability to perform the required function can be readily analyzed (such as mechanical support, simple conductivity, etc.), and (d) especially in aging, where components or devices can be shown not to be limiting to the overall performance of the function."

As written, this section is far too narrow and restrictive. For example, in aging a piece of equipment which contains many materials it is sufficient to eliminate those not affected using analysis (evaluation of activation energies) in order to determine which materials are controlling by being most susceptible to aging effects.

COMMENT NO. 43B: The statement that, "in general the staff will not accept analysis in lieu of test data...," imposes a severe limitation on the industry that is not present in IEEE 323-1971 and 1974. IEEE 323-1971 states that, while type tests are preferred, other methods may be used "when size or other practical requirements limit or preclude type tests." At this late date, the Commission, by its action in NUREG-0588 proposes the deletion of a phrase from the standard, thereby invalidating a great deal of work that has been done and accepted up to this point in time.

We believe those sections of the NUREG dealing with aging, margin and qualification by analysis should be revised to reflect the standards as they are now written and have been interpreted since their issuance.

Resolution

See staff response to Comment Nos. 46 and 51. With regard to "other practical requirements" that may limit or preclude type testing, the staff has stated (see "Discussion") that alternatives (or exceptions) to the interim positions may be proposed and, if justified, may be found acceptable. These exceptions should be identified and evaluated on a case-by-case basis.

COMMENT NO. 44: (Section 2.1(3)) It is not clear whether the term "safety margins" is intended to be the same as the term "margin" used in IEEE 323-1974 or a new undefined term. If the former, delete "safety;" if the latter, it should be defined.

Resolution

The staff agrees with the comment. The term "safety margins" is intended to be the same as the term "margin" used in IEEE 273-1974.

COMMENT NO. 45: (Section 2.1(3)) With respect to the last sentence, does the term "operability" mean safety function?

To what factors or conditions should "safety margin," as used in the last sentence, be applied?

Resolution

Operability refers to assuring that the performance characteristics of the equipment that are necessary to perform a safety function are satisfied, including but not limited to accuracy, response time, and so forth. Test margin should be applied as described in Section 3.0 of the document.

COMMENT NO. 46: (Section 2.1(3) The need to qualify non-safety-related equipment by test is overly stringent. In many cases, analysis is adequate to determine whether or not a failure mechanism exists which can result in reduced plant safety. The wording should be changed to, "...should either be shown by analysis to not fail in a manner detrimental to plant safety or be qualified to demonstrate such capability."

Resolution

The staff concurs in part; clarification of this requirement will be considered in final rulemaking. The staff maintains, however, that for active electrical equipment subjected to a DBA environment, type testing is the preferred qualification method. Other methods may be justified and will be evaluated on a case-by-case basis.

COMMENT NO. 47: (Section 2.1(3)) This section (Section 2.1(3)(c)) deals, in part with qualification of equipment that has been shown and justified to be unrelated to accident mitigation or plant safety. This goes beyond the bounds of IEEE 323-1971 and 1974 which state that they are for the qualification of Class IE equipment not non-safety-related equipment. Therefore, the requirement to qualify non-safety-related equipment should be deleted.

Resolution

Equipment that has been shown and justified to be unrelated to accident mitigation or plant safety is exempt from qualification.

COMMENT NO. 48: (Section 2.1(3))

The inclusion of non-safety-related equipment in this section is incompatible with the scope of this document. Furthermore, the effects and consequences of adverse environments on non-safety-related equipment has been raised as a Category I item on NUREG-0585 "TMI-2 Lessons Learned Task Force Final Report." We strongly recommend that the staff delete this requirement under NUREG-0588 to permit orderly resolution of this generic issue.

Resolution

The position addresses designs where equipment or systems have been incorrectly classified as non-Class IE strictly on the basis that they did not have to perform a specific safety function (such as actuation). The designs may have not factored in the broader function of determining whether or not their failure or improper actuation could also effect safety. It is the staff's intent to assure that this broader functional scope is factored into the design and as such, the qualification of some previously classified non-Class IE equipment may be warranted. It is not the staff's intent to require qualification of all non-safety-related equipment. (See also the staff response to Comment 4 and Comment 47.) Regarding the second point in the comment, the staff does not agree with the recommendation to delete this requirement from NUREG-0588. Resolution and implementation of the generic issue identified in NUREG-0588 are independent of the requirement stated herein.

COMMENT NO. 49: (Section 2.1(3))

This paragraph (Section 2.1(3)(c)) should be clarified to indicate applicability to safety-related equipment only. Non-safety-related equipment is not environmentally qualified unless it falls into Category 2.1(3)(b).

Resolution

The staff concurs. NUREG-0588 is only applicable to safety-related equipment.

COMMENT NO. 50: (Section 2.1(3))

Paragraph 2.1(3)(c) on page 11 seems to suggest "qualification" requirements for non-IE equipment, which is beyond the subject of this staff position document.

COMMENT NO. 50B:

This position (Item 2.1(3)(b)) requires the qualification of non-safety-related equipment to show that the equipment will not fail in a manner detrimental to plant safety. This requirement does not fall under the scope of Environmental Qualification of Safety-Related Electrical Equipment. This requirement is being addressed elsewhere under such headings as "Systems Interaction" and "Consequential Failure" and in other documents such as NUREG-0578, and NUREG-0660. The overall program of "System Interaction" is very large and must be part of a long-term program as defined in some of the references mentioned above. The ultimate results of this long-term program may have some impact on the environmental qualification of safety-related electrical equipment but this should not be forced into NUREG-0588. It is recommended that the reference to non-safety-related equipment be deleted from this paragraph since these studies are addressed elsewhere and a parallel review under equipment qualification would detract from the remainder of the qualification program and would not add to plant safety.

COMMENT NO. 50C: The word "qualified" in this section (Section 2.1(3)(c) presents problems. We do not "qualify" non-Class IE equipment. We recommend deletion of both paragraphs of this section as, otherwise, we may have to obtain documentation for items of no safety significance.

COMMENT NO. 50D: It should be clarified that this paragraph (Item 2.1(3) (c)) applies only to safety-related equipment.

Resolution

Refer to the staff response to Comment Nos. 46, 47, 48, and 49.

COMMENT NO. 51A: (Section 2.1(3)) The requirement to qualify such equipment by test only is incompatible with the alternatives recognized under paragraph 2.1(2).

Therefore, delete the words "by test" from Section 2.1(3)(a) and the words contained in brackets in Section 2.1(3)(b).

COMMENT NO. 51B: Methods of qualification other than type testing should be applicable to item 2.1(3)(a)(b) also.

COMMENT NO. 51C: The requirement to demonstrate by test that the equipment will not fail in a manner detrimental to plant safety should be expanded to allow demonstration by analysis as well as test.

COMMENT NO. 51D: The words "qualified by test" should read "qualified by test or analysis." Otherwise, the testing program would expand considerably to no apparent benefit, expecially for non-safety-related materials and equipment.

Resolution

For electrical equipment located inside or outside containment that may be exposed to high energy line breaks (for example, LOCA, MSLB, feedwater line rupture), analysis alone is generally inadequate to demonstrate functional operability such as accuracy or response time, or to verify seal integrity (as in connectors), or even to detect intermittent or spurious failures. Although some analysis may be used when the testing is the principal qualification method, that analysis should be limited to extrapolations of data or to analyzing similarities in equipment or materials. In either case, analytical assumptions should be verifiable or supported by test data.

Recognizing the complex interaction of the environment on materials and equipment (such as aging or simultaneous vs. sequential effects) the staff does not agree that analysis by itself is an acceptable alternative for qualifying equipment required to function in the above-mentioned hostile environments. (See exceptions in Section 2.4.)

COMMENT NO. 52: (1) The implication in this section (Section 2.1(3)(a)) is that equipment that must function at any time during an accident must be shown to be capable of operating for the entire duration of accident conditions. This fails to differentiate between those items that must function throughout the accident and those that must perform some specific task at a given point in time during the accident. This paragraph should be changed to reflect these different classes of items.

Resolution

Refer to Section 3(4) which addresses this issue.

COMMENT NO. 53: (Section 2.1(3)) We interpret the equipment referred to in these sections to be that which is subjected to the environment of a LOCA or MSLB.

Resolution

The interpretation is correct in part. Environments caused by other high energy line breaks such as feedwater line rupture should also be considered.

Section 2.1(3) was modified for clarity.

COMMENT NO. 54A: (Section 2.1(3)) Throughout Section 2.1, reference is made to "accident" and "DBA." These terms should be defined. Comments pertaining to this section are predicated on the assumption that these terms mean LOCA or MSLB.

COMMENT NO. 54B: The term "event" (in Section 2.1) is not obvious. This should not be clarified (i.e., LOCA, MSLB, etc.).

COMMENT NO. 54C: Introduction of term "DBA" at this point (Section 2.1(4)) is inconsistent as it has not previously been used nor defined and its meaning is not clear.

Resolution

Accidents in this category include the complete spectra of break sizes for loss-of-coolant accidents (LOCAs) and for other high energy line breaks (HELBs) such as main steam line breaks (MSLBs) or feedwater line breaks.

For a listing of accidents required to be analyzed, refer to the Standard Review Plan Section 15.

The term "event" refers to occurrences such as natural phenomena, including but not limited to earthquakes and flooding resulting from other than pipe breaks. "Event" also refers to occurrences such as loss of ventilation which may occur as a result of a single failure.

COMMENT NO. 55A: (Section 2.2(12)) If only a source for the simulation of gamma is to be recommended in Section 2.2(12), and not a source for beta simulation, we recommend deletion of the item to avoid possible misunderstanding that Co-60 is acceptable for simulation of all radiation.

COMMENT NO. 55B: On page 13, item 1.1(12), from a commercial standpoint, it is encouraging to see that NRC considers cobalt simulation of the radiation environment; it has not yet been technically accepted and is in fact being questioned by WG 2.6 of IEEE/NPEC/SC-2. Unless the NRC is privy to some information that is unavailable to the rest of the industry this item should be removed. Furthermore, it implies that acceptable accounts for radiation can be considered by cobalt simulation, which case has just not been demonstrated and is also being pursued by the above mentioned WG 2.6.

COMMENT NO. 55C: In Section 2.2(12) sources other than Cobalt-60, e.g., Cesium-137, should be acceptable as qualification sources.

Resolution

The staff currently has a research effort with Sandia Laboratory to investigate the adequacy of qualifying equipment for both gamma and beta radiation environments by using only a gamma radiation source. While the results are very preliminary, there does not seem to be any significant problem in using only a gamma source to qualify certain types of equipment for a beta/gamma environment provided the gamma dose rate during the qualification tests is consistent with the expected beta and gamma dose rates (energy deposition rates) during the LOCA. It appears therefore that a gamma source (only) may be used for qualification testing, provided an analysis or test data indicates that the dose and dose rate produces damage similar to that which could be produced under accident exposure (i.e., combined gamma and beta environment), or a beta and gamma qualification dose and dose rates may be determined separately and the testing may be performed using both a beta and a gamma test source. The staff notes that the research effort is still continuing and that the preliminary findings may change, but until such time as other evidence is presented, the use of either Co-60 or Cs-137 for equipment qualification would seem appropriate.

COMMENT NO. 56: (Section 2.2(1))

The requirement to establish "failure criteria" would appear to be an unfortunate choice of words in that it imposes an unbounded set. We recommend changing to "acceptance criteria" as being not only more practical but more correct. Note the conflict with Appendix E, item 4.

Resolution

The staff concurs. Any modifications to the positions will be considered during final rulemaking.

COMMENT NO. 57A: (Section 2.2(4))

By stating a preference for an environmental profile that envelopes any design basis event there is a strong implication that equipment is to be designed to withstand more than one event. We know of no such requirement, and suspect that equipment exposed to the environment of an MSLB would not be allowed to be returned to operation without extensive inspection and, perhaps, refurbishing. While such enveloping should certainly be allowed, we fail to see the technical basis for the preference (especially since experience with NRC's "preferences" is that they eventually become requirements). There is an additional problem with the "preferred" approach when one considers the margin.

Current practice of doubling the number of transients would mean that a given equipment would have to be designed to withstand two MSLB events and two LOCAs. This may be excessive. We recommend the following wording: "...located inside containment, a single profile may be used that envelopes the environmental...loss-of-coolant accidents, but in any case, the equipment shall be shown to operate correctly under the environmental conditions of any design basis event for which it must perform a safety function."

COMMENT NO. 57B:

Use of separate profiles for LOCA and SLB should remain an acceptable option. While it may be convenient to test with one profile, the test is unnecessarily more severe than separate profiles.

COMMENT NO. 57C:

Requiring a single profile to envelope the worst case environmental conditions is neither practical nor realistic in terms of previously established acceptable qualification test methods. This procedure implies that one item of equipment could be subjected to both a LOCA and an MSLB, and this is not part of any accident analysis scenario nor is it consistent with previously acceptable practice.

In addition, in order to comply with the margin application requirements of IEEE 323-1974 the question of margin on peak transients is raised with respect to the number and severity if indeed this combined profile is to simulate a LOCA and MSLB case which would result in four transients at elevated temperature.

In order to expect equipment to operate successfully during this type of test it would almost certainly require substantial redesign and retesting to new conditions which would obviate the usefulness of any prior test performed.

This item should carefully be re-thought and substantially revised to allow the continuation of past type-test practice.

Resolution

There are components and equipment inside containment that are important to safety and are required to operate in both a LOCA or an MSLB environment. There may also be equipment or components (such as cable, penetrations, connectors, valves) that may not be required to perform a specific function but whose failure or inadvertent operation in a LOCA or an MSLB environment may be detrimental to safety. For such equipment, although a single bounding profile used to qualify the equipment is prefereable, other envelopes used in testing for a LOCA and an MSLB, either separately or sequentially, may also be used. The staff's preference for a single bounding envelope is to minimize the review effort by reducing the documentation and the analysis that would be required to demonstrate qualification to both environments.

With regard to margins, the staff considers that exposing the same component or equipment to a combined or sequential LOCA and MSLB envelope is sufficiently conservative to justify omitting the additional requirements of doubling the number of transients. These options will be evaluated on a case-by-case basis.

COMMENT NO. 58A: (Section 2.2(6)) We believe this to be an incorrect requirement that appears to miss a fundamental concept of qualification. The equipment is to work in the required environment and the qualification test should so demonstrate. The actual real time temperature of any portion of the surface boundary of the equipment is of no consequence in meeting this requirement (except to the equipment designers) and, may be misleading and non-conservative. We recommend the following wording:

> "The temperature to which equipment is tested to demonstrate qualification shall be measured and recorded throughout the test. The thermal capacity of the environment simulation shall be shown to provide an adequate simulation."

COMMENT NO. 58B: Suggest the NRC should keep away from designing tests by requiring thermocouple readings. It would be sufficient to request that the component temperature be determined by suitable means. The temperature to which equipment is qualified does not have to be defined as the surface temperatures but on the basis of its ability to perform as specified in a bulk ambient environment expected to occur as defined for the design basis. Whether or not the surface temperature ever reaches this value is immaterial, particularly with respect to short-term high peak temperatures, such as could occur for an HELB.

COMMENT NO. 58C: In general, surface temperature is not monitored directly during testing. Instead, the ambient air temperature at various locations within the test chamber is monitored. We assume the implication behind requesting measurements of surface temperature is to ensure that the device has stabilized at the test temperature prior to timing its exposure. If so, revise this section to so state the above. If not, revise Section 1.2(5) to clarify the reason for requesting component surface temperature to be monitored.

Resolution

The staff agrees with the intent of the comment. It should be noted that the objective of the position is to ensure, by independent verification, that the equipment or component was exposed to the bulk temperature equivalent to or more severe than that temperature assumed in the bounding envelope derived from the accident analysis. Temperature sensors (not necessarily limited to thermocouples) located only on the inlet piping of the test chambers may not be indicative of the bulk temperature at the component being tested.

The intent is to ensure that temperature sensors are located as close as practical to the components being qualified.

It may also be prudent to provide temperature sensors that in addition to monitoring bulk temperature would also monitor the surface temperature of the equipment. This would facilitate the comparative studies discussed in Section 1.2(5)(b) of the NUREG. Without these readings, the use of the more conservative comparison to the "bulk" LOCA test temperature would be warranted. See also staff response to Comment No. 9.)

COMMENT NO. 59: (Section 2.2(5)) Full duration testing for extended periods of submergence is impractical and unnecessary. Short duration testing to demonstrate seal integrity plus an addressment of potential corrosion mechanisms by test or analysis are adequate. The following change to item 2.2(5) is therefore, recommended:

The sentence:

"Where equipment could be submerged, it should be identified and demonstrated to be qualified by test."

Should be changed to:

"Where equipment could be submerged, it should be identified and demonstrated to be qualified by test to demonstrate seal integrity. The effects of corrosion mechanisms for the duration required should be addressed by test or analysis."

Resolution

The staff concurs in part. Shorter test periods and analytical extrapolation may be found acceptable if adequately justified. This justification should be included as part of the qualification documentation. Analysis by itself, however, may not be adequate see staff response to Comment No. 51 for additional information.

COMMENT NO. 60A: (Section 2.2(9)) There appears to be a conflict between this requirement and that of 2.2(7) in that (7) allows periodic verification of operability and (9) requires continuous verification with justification necessary for periodic. This conflict should be resolved.

COMMENT NO. 60B: The requirement for continuous monitoring seems inconsistent with the requirement for periodic performance verification stated in 2.2(7). The same thing is being required differently. This should be revised.

COMMENT NO. 60C: These paragraphs appear to be requiring in-plant testing for qualification acceptance. The paragraph should be rewritten to clearly identify that this is not the staff's position.

COMMENT NO. 60D: Does continuous monitoring of equipment operability status mean that equipment is to be exercised throughout the test (e.g., coils energized, motors energized...)? If so, the statement is appropriate when actual environmental conditions are simulated. However, if accelerated aging temperatures are being used, the operability should only be checked at discrete intervals with components at anticipated ambient conditions.

Resolution

The intent of Section 2.2(9) is to ensure that intermittent failures in equipment--such as momentary change of state of bistables (that is, contact chatter), a cyclic variation in a transmitter output or a valve position variation--have been accounted for in the qualification testing program. Where intermittent failures in equipment can negate a safety function, the test program should include provisions to monitor selected parameters on a continuous basis in order to detect these failures (if any). It is recognized that certain equipment requir[illegible] long-term testing (for example, postaccident monitoring equipment) where [illegible] the clock monitoring is difficult to accomplish. For this cate[illegible] of equipment, continuous monitoring for spurious or intermittent operation [illegible]ring periodic intervals may be justified.

COMMENT NO. 61: (Section 2.2(8)) The application of spray at the maximum ambient condition is unrealistic because during actual initiation in a plant it is at much lower temperature prior to any recirculation. This condition has generally been simulated during type-testing and has also been shown to produce the same results where spray has been at elevated temperature, which in itself is difficult to attain during type-testing and is not necessary.

Resolution

Chemical spray ingress (if any) is one area of concern addressed by the position. Pressure is considered a driving force influencing ingress into vital components through materials such as seals, jackets, and so forth. It is therefore prudent during testing to ensure that chemical (or demineralized water) sprays are introduced at or as close to the simulated maximum containment peak pressure conditions (if not already introduced before the maximum peak pressure conditions are reached).

COMMENT NO. 62: (Section 2.2(8)) Add words, "where applicable" to this article.

Resolution

The staff does not agree. See staff response to Comment No. 15.

COMMENT NO. 63A: (Section 2.2(10)) This is only applicable when the design range of voltage and frequency is significant. For Class IE devices fed from a guaranteed stabilized power source, such a demonstration is unnecessary. The following change to item 2.2(10) is, therefore, recommended:

> "The aspects of the expected extremes in power supply voltage range and frequency need only be considered during simulated event environmental testing if there is a significant design range for these parameters."

COMMENT NO. 63B: When would it be required to demonstrate performance under expected extremes in operating characteristics? This has been delineated for valve actuation in IEEE 382-1980 and its omission permitted by suitable justification to demonstrate that it does not upgrade the equipment's ability to perform its specified safety function.

COMMENT NO. 63C: If simulated event environment is accelerated, then voltage and frequency ranges should be applied at discrete intervals with components at anticipated ambient conditions.

Resolution

The loss of offsite power is assumed concurrently with a design basis accident (as in the case of a LOCA, MSLB, and so forth). As a result of sequencing the loads onto the diesel generators, power and frequency variations will be sensed on selected equipment such as valves, motors, and relays, and may affect their performance characteristics (for example, response time) and negate their safety function. If equipment can sense these effects, these variations should be accounted for in the test program.

There are, however, designs where the power supplies remain unchanged (because, for example, of the instantaneous availability of backup power supply systems). In those cases, exceptions to this position would be justified.

COMMENT NO. 64: (Section 2.2(10) — This position is too binding and does not allow analysis to be considered to establish most critical input conditions. Also, simulation of under-voltage and/or frequency is applied during seismic testings and is considered more severe.

Resolution

This position does not exclude the use of analysis to establish critical input conditions. See staff response to Comment No. 51.

Regarding the second point, it is not evident that under-voltage and/or under-frequency simulation during seismic testing is always more severe than in other hostile environment conditions. Where this is the case, the basis for excluding such testing under these other hostile environment conditions should be provided and documented in the qualification reports.

COMMENT NO. 65A: (Section 2.2(11)) — Rather than addressing "dust" in qualification, it would make more sense to improve cleanliness requirements on plant operations. We recommend deleting this non-quantitative item.

COMMENT NO. 65B: — Dust has not been considered in the environmental specification. This is a potential major change for IEEE 323. Technical justification for its inclusion must be supplied. Rather than addressing "dust" in qualification, it would make more sense to improve cleanliness requirements on plant operations. We recommend deleting this non-quantitative item.

COMMENT NO. 65C: — We disagree that this should be a "service condition" specified in the qualification programs. The dust accumulation is primarily a function of housekeeping. If any special cleaning requirements are necessary in order to ensure operability of the equipment, it should be addressed in the operating and maintenance requirements of the equipment and not the qualification service conditions. It should not be the intent of a qualification program to address all possible service conditions that could occur if normal maintenance is not performed.

COMMENT NO. 65D: The paragraph requires that "dust environments" should be addressed when establishing qualification service conditions. NRC should delete or be more definitive.

Resolution

The staff agrees in concept with the comment.

It is not the staff's intent to require quantitative testing to ensure equipment operability in dusty environments, but rather to highlight a potential failure mechanism. Equipment susceptability to dust should be considered when qualifying safety-related equipment and be accounted for in the interface requirements via, for example, in improved periodic maintenance, or by the use of protective covers. The staff is currently in the process of rulemaking and will consider the recommendations expressed in the above comments, in the "Final" position.

COMMENT NO. 66: (Section 2.3(1)) The statement that "the test procedures should...accident environment" is in conflict with the recommendation that the test sequence should conform fully to the guidelines of Section 6.3.2 of IEEE 323-1974. The conflict results from Section 6.3.2(3) permitting the operational performance extremes test to be completed on other, essentially similar equipment. It is, therefore, recommended that the identified sentence be deleted as being inaccurate and redundant.

Resolution

The implementation of Section 6.3.2(3) of IEEE 323-1974 (or the staff's position) establishes a data base during normal environments which should provide a comparison of the performance characteristics at the more severe environments. The staff agrees with the statement in the standard that if a data base is available from other tests "on identical or essentially similar equipment," then there is no need to repeat a test to establish a redundant set of performance characteristics at a normal environment. However, caution should be taken in using data from other than identical equipment, so that extrapolation of data is indeed valid. When exposing equipment to hostile environments, the same piece of equipment should be used in sequence see resolution to Comment No. 80). The staff does not agree that Section 2.3(1) is in conflict with Section 6.3.2(3) of IEEE 323-1974 but does recognize that justified exception may also be found acceptable.

COMMENT NO. 67: (Section 2.3(1)) IEEE Standard 323-1974 permits deviations from the recommended test sequence providing adequate justification can be provided.

Resolution

The staff agrees with the comment. See staff's response to Comment No. 15.

COMMENT NO. 68: (Section 2.3(2)) (Category II) This is incompatible with item 2.2(2). The following change is, therefore, recommended:

(2) "The test should simulate as closely as practicable the combination of postulated environments necessary t meet the requirements of subparagraph 2.2(2)."

Resolution

The intent of the above-referenced section is to ensure that all environmental service conditions expected to occur would be enveloped. Any apparent inc mpatibility will be corrected during the final rulemaking.

COMMENT NO. 69: (Section 2.3(4)) (Category II) Separate effects testing may have been done on penetrations, etc. It may be very difficult to retest such equipment. This requirement should be revised to a "best efforts" basis.

Resolution

Justification for the adequacy of the sequence used should be established and provided as part of the qualification documentation. Exceptions, if justified, may be established and will be evaluated on a case-by-case basis. If the adequacy of the qualification method can not be justified, retesting or equipment replacement may be warranted.

COMMENT NO. 70A: (Section 3(1)) The paragraph calls for margin on margin. Presumably, the staff requires demonstrable margin with respect to accident parameters which have been established employing a calculation model acceptable to the staff. The following change to item 3(1) is therefore, recommended:

(1) "Qualification margins should be applied to the design parameters discussed in Section 1, which are established employing a calculation model acceptable to the staff, to assure that the postulated accident conditions have been enveloped during testing."

COMMENT NO. 70B: The application of margin in addition to the margin applied during derivation of the service conditions would be doubly redundant and not necessary if the previous margins have been quantified.

COMMENT NO. 70C: There is no technical basis for the summary dismissal of margins just because they are part of the plant parameters rather than just of the test parameters. It has always been the intent of IEEE 323-1974 that margin need not be added if it can be shown that adequate margin is already included in the environmental requirements. The position taken in 3(1) is not consistent with the position in 1.4 which states that additional radiation margins are not required if certain procedures are followed.

Resolution

The staff is in agreement that additional margin need not be added if it can be shown that adequate margin (to account for uncertainties identified in IEEE 323) is already included in the environmental requirements. Although claims are made that these margins are included in the calculated envelopes, experience has shown that those margins may not be adequately quantified to facilitate independent verification.

In general qualified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident environmental conditions have been enveloped. The margins should (1) account for uncertainties associated with the use of analytical techniques in deriving environmental parameters; (2) account for uncertainties associated with defining satisfactory performance (e.g., when only a small number of units are tested) (3) account for variations in the commmerical production of the equipment and (4) account for the inaccuracies in the test equipment to assure that the calculated parameters have been enveloped. These margins should be provided in addition to any conservatisms applied during the derivation of the specified plant parameters unless these conservatisms can be quantified and shown to contain sufficient margin. It is the staff's belief that when the temperature and pressure conditions are derived using the methods identified in Section 1.1(2) or the qualification envelope in Appendix C is used, or the radiation methodology described in Appendix D is used the only additional margins to be provided are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for the uncertainties identified in (1) through (3) above.

COMMENT NO. 71A: (Section 3(1),(2)) It appears that three levels of margin are to be employed. The first is that applied during the derivation of plant conditions. The second would be for accident conditions to ensure enveloping postulated accident conditions, and the third would be in accordance with Section 6.3.1.5 of IEEE 323-1974 to account for normal variations in commercial production. Please confirm if the above understanding is correct. There is general concern in the industry regarding regulatory requirements resulting in the cascading of margins. In some instances this leads to unrealistic qualification testing parameters and results.

COMMENT NO. 71B: This position on margin (Section 3(1)) is excessive. The conservatism used in calculating specified plant parameters is a form of margin. Not allowing any credit for this conservatism will only force vendors/engineers to go back and recalculate these parameters in a less conservative manner in order to establish more realistic qualification plans or to validate tests already completed. Such work will do nothing to better show the adequate qualification of electrical equipment.

It is recommended that the second sentence from paragraph 3(1) be dropped. In lieu of this sentence, it is recommended that the following be added to the end of paragraph 3(2):

"Normally, margin is applied to the specified plant parameters; however, credit may be taken for the conservatism applied in calculating specific plant parameters if this conservatism, along with whatever other margin is applied, is shown to provide an overall adequate level of margin."

Resolution

See the staff's response to Comment Nos. 70 and 73.

COMMENT NO. 72A: (Section 3(1)) (Category II) For Category II, the requirement ("Same as Category I") is in conflict with the requirement for Category II on aging, in Section 4. Furthermore, since IEEE 323-1971 did not require margin, and since many pieces of equipment have operated satisfactorily for a long time with qualification that did not include margin, it may be counter to safety to require such equipment to be replaced because the test data did not include margin.

COMMENT NO. 72B: On page 14, Category II, item 3(1) -- "Margin was not a requirement for qualification under the guidelines of IEEE 323-1971 but its incorporation is now required by reference to the requirements for Category I applicability. This could have significant adverse impact on the acceptability of the qualification process particularly if it is determined under the present ground rules that the service conditions have to be changed when the original methodology used during the FSAR preparation for the plant licensing basis was considered adequate. Remove from Category II this requirement to implement margin per the IEEE 323-1974 groundrules."

COMMENT NO. 72C: Item 2.2(1), 3(1) -- Since the application of margin was not a requirement of IEEE 323-1971, it may not be possible to demonstrate margin in all cases. In such cases, lack of documentation demonstrating qualification margin should not constitute unacceptability for Category II plant equipment.

COMMENT NO. 72D: Paragraph 2.2(2) suggest change "...all service conditions postulated (with margin, see Section 3.0) during..." for clarity.

Resolution

Qualification documentation should clearly show that the environmental parameters (to which the equipment may be exposed) have been adequately enveloped. If no margin can be claimed per Section 3.2 of the NUREG (Category II), the adequacy of the design is considered questionable. (See also response to Comment No. 7.)

COMMENT NO. 73: (Section 3(2)) The Nuclear Power Engineering Committee has taken the following position to clarify the intent of the margin requirements in IEEE 323-1974. We feel it would be of use to the industry if this clarification were included in NUREG-0588.

"IEEE 323 requires margins and suggests considerations increasing the test level, increasing the number of test cycles, or increasing the test duration as methods of assuring margin. It was the general consensus that while IEEE 323 tends to promote that all three be used, there are situations where it could be demonstrated that one higher transient is equivalent to or more conservative than, for example, two lower level transients, etc. The choice is, therefore, up to the user and depends upon the type of equipment, method of test, etc. In any event, the user should justify his method."

Resolution

The staff supports the Nuclear Power Engineering Committee position on margins and considers the comment an amplification of the staff positions identified in the NUREG (specifically, Section 3(2) of Category II).

COMMENT NO. 74: (Section 3(3)) The specific reference to inaccuracies in the test equipment should be provided so that the required application of margin can be correctly implemented in the test program.

Resolution

Margins to account for inaccuracies in the test equipment should factor in the accuracy tolerance of the sensors used to monitor the test conditions (for example, pressure or temperature sensors). These margins should be added to the test profiles to ensure that the calculated environments have been enveloped. For example, if the maximum temperature to be sensed is 300°F and the sensor can be in error by 5°F at that value, then the indicated temperature during the test should not be less than 305°F to ensure that maximum conditions have been simulated.

COMMENT NO. 75: (Section 3(1)) For new designs, quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. Where existing designs are being qualified for a new application, margin should be applied as a difference between service conditions and design limits.

Resolution

Margins should be applied to account for test, production, and analytical uncertainties that are identified in IEEE Standard 323 and NUREG-0588, independent of their design chronology.

COMMENT NO. 76A: (Section 3(4)) There is no technical basis for the application of an arbitrary margin of one hour (in Section 3(4)). The Reactor Protection System, for example, is designed to operate in terms of milliseconds. As long as subsequent failure can be shown to not undo the safety action already taken, there is no need to require survival for some totally arbitrary period.

COMMENT NO. 76B: Westinghouse is totally opposed to the arbitrary application (in Section 3(4)) of an additional one-hour time requirement in excess of the calculated worst-case time required to perform the safety function as derived from accident analysis. Implementation of this requirement will negate extensive qualification testing already completed by industry and, furthermore, will severely impact qualification test schedules established for the lead plants committed to IEEE 323-1974.

The staff has indicated that this requirement has arisen from concerns over earlier transmitter tests where failure of some units was noted after a few minutes. Thus, Westinghouse recommends that the sentence:

"Equipment in these categories is required to remain functional in the accident for a period of at least 1 hour in excess of the time assumed in the accident analysis."

Be changed to:

"Equipment in these categories is required to remain operable, in the accident environment, for a period of at least 1 hour in excess of the time assumed in the accident analysis. The equipment performance during this additional 1 hour shall be shown not to negate any prior completed automatic safety functions or, in the case of equipment required for post-accident monitoring, provide misleading information to the operator."

COMMENT NO. 76C: The environmental standard does not imply that equipment should be functional in the accident environment(s) for a period of at least one hour, as required by the staff position. This requirement should be removed. There is no technical basis for the application of an arbitrary margin of 1 hour. The Reactor Protection System, for example, is designed to operate in terms of milliseconds. As long as subsequent failure can be shown to not undo the safety action already taken, there is no need to require survival for some totally arbitrary period.

COMMENT NO. 76D: The "1-hour minimum operability time" following the DBE is a new requirement and will impact present and previous equipment programs. It also is over and above that required by paragraph 2.1(3) and Appendix E, Section 2.

Adding a 1-hour operability requirement to equipment qualification will discourage additional transducer suppliers, whose equipment is designed to function quickly for safety purposes. The "margin" defined in IEEE 323-1974 appears sufficient.

COMMENT NO. 76E: The margin requirements (in Section 3(4)) are excessive for equipment intended to function for less than one hour in an accident environment. A more appropriate margin would be based on a percentage increase above the operability requirements.

COMMENT NO. 76F: This position (in Section 3(4)) states that equipment which is required to only perform its safety function within a short period into the event (i.e., within seconds or minutes) is required to remain functional in the accident environment for a period of at least one hour in excess of the time assumed in the accident analysis. We feel that this qualification requirement is unnecessary for this type of equipment.

COMMENT NO. 76G: The staff should document the concern being expressed to allow industry the opportunity to develop alternative ways to resolve the issue.

COMMENT NO. 76H: This position (in Section 3(4)) will impose harsh requirements on equipment qualification without improving plant safety. Basically this position requires a minimum of one-hour margin on the required operating time for safety-related electrical equipment. This is a harsh requirement on instrumentation which must function early in a LOCA or an MSLB with specific accuracy. Qualifying this equipment with some reasonable margin (including time) is difficult but feasible. Maintaining the required accuracy for one hour beyond the specified time in an (LOCA/MSLE) accident environment will not improve safety. Such a requirement will only invalidate tests, disqualify equipment previously qualified and force users to obtain and qualify equipment (if available) in order to pass a test but not to improve plant safety. It is recommended that this paragraph be deleted or, as a minimum, be replaced with a paragraph that discusses the need to show adequate time margin for equipment with short specified operating times after a DBE.

COMMENT NO. 76I: Implementation of this requirement (Section 3(4)) will negate extensive qualification testing already completed by industry and, furthermore, will severly impact qualification test schedules established for the lead plants committed to IEEE 323-1974. As a minimum, a review of equipment capability to meet these revised requirements will be necessary prior to embarking on an expensive test program and, at worst, an equipment development program may be required to meet this arbitrarily imposed functional requirement. Tests and analyses of Category II equipment in some cases did not include a requirement to remain functional for at least an hour longer than assumed in the accident analysis.

This requirement should be considered on a case-by-case basis, especially for such items as isolation valves.

Resolution

For equipment subjected to hostile environments resulting from pipe breaks, an accepted practice is to qualify that equipment to the most limiting environment (which would envelope the less hostile environments caused by a range of different pipe breaks). Subjecting the equipment to the most severe portion of the hostile environment (maximum pressure, temperature, and radiation) for only a very short time period (seconds or minutes) does not provide adequate assurance that all the environmental service conditions have indeed been enveloped. It is the staff's belief that the additional one hour of demonstrated functional operability for equipment required to operate for only a short period (that is, less than or equal to 10 hours), provides for the most part, the assurance that the equipment will function in any accident environment that can exist during large and small line-break accident scenarios.

There may be some designs where less restrictive margins may be justified and found acceptable on a case-by-case basis (see Category II, Section 3(2)). The staff believes, however, that the general requirement of testing for an additional hour is warranted.

COMMENT NO. 77: (Section 4(1))

The staff position (in Section 4(1)) should be clarified to assure that the requirements are being applied to Class IE equipment only.

Resolution

See staff resolution to Comments Nos. 4 and 48.

COMMENT NO. 78: (Section 4(1)) (Category II)

In Section 4.1, the Category I position far exceeds those established in IEEE 382-1972 or IEEE 334-1971. Compliance to the provisions of these standards should be sufficient for Category II equipments. It is recommended to delete the last sentence of the Category II position 4(1).

Resolution

This area is under staff review. Any modifications to the staff positions will be included in the final rulemaking which is planned to be issued to public comment in December 1981. In general the staff does not require, for Category II plants, the same degree of rigor in the proof testing, analysis, and documentation as it does for Category I equipment. Recognizing the limitations in the state of the art in assessing synergistic effects, the position regarding synergisms for Category I is not applicable to Category II plants unless known synergistic effects have been identified on the materials that are in use in these older plants. With the exception noted above (synergisms), the aging positions identified for Category I are applicable for Category II equipment identified in Section 4(1).

COMMENT NO. 79A: (Section 4(2)) (Category II)

Section 4.2 requires an aging evaluation program be conducted and a periodic replacement schedule be established. This is of major impact. For this category of plants, the staff should specifically state what equipment has been shown susceptible to aging effects. One source could be the NPRD program. Trend studies could be conducted that point to equipment aging at an unacceptable rate. Periodic bulletins could alert the utilities and corrective action taken based on good data rather than engineering guesses. Requiring a reevaluation of aging effects in the Category II equipment is well beyond the licensing commitments.

COMMENT NO. 79B:

For Category II equipment, identification of materials susceptible to aging would require a long list compiled from literature of test data. Also, each manufacturer uses his own formulation and may be reluctant to release information. Going back to manufacturers, and particularly their subsuppliers, of equipment delivered several years ago will be extremely time consuming and probably inconclusive. The benefits, in the form of improvement in safety, do not appear to be commensurate with the potential effort required.

Resolution

As stated in the position of Section 4.2, the staff has and will continue to identify materials and or equipment that may be susceptible to deleterious aging effects. It is, however, incumbent on the user of the equipment (that is, the utility) to ensure that the equipment that has been identified by the staff and by others as being susceptible to significant degradation because of aging is properly accounted for. Data banks established by owners groups are one way of maintaining current information of specific equipment in use today. Ongoing programs should exist at the plant to review surveillance and maintenance records to ensure that equipment which is exhibiting age-related degradation will be identified and replaced as necessary.

COMMENT NO. 80A: (Section 4(3))

In Section 4(3), the term "investigation" with regard to synergistic effects is ambiguous. This could mean an experimental program or a literature search. In any case, the state of the art in aging to a single environmental stress is rudimentary at best. Requiring experimental studies of combined effects exceeds the existing technology. We recommend deleting the second sentence. We further recommend deletion of the last sentence which references partial results of questionable test programs until such time as they are completed, reviewed and verified.

COMMENT NO. 80B:

For item 4(3), it has not yet been determined whether or not synergistic effects (not defined in this NUREG) are necessary for consideration during any phase of the qualification program. Reference is made to a previously established position developed by the above-mentioned WG 2.6 and subsequently endorsed in full, by SC-2 and forwarded to NPEC. NRC should seriously take this statement into consideration before requiring the consideration of unknown in NUREG requirements.

Furthermore, the reference given in the NUREG to partial results of questionable test programs should be totally omitted as there is no technically sound basis for assuming that these results are definitive and represent the state of the art.

COMMENT NO. 80C: In item 4(3), the position concerning synergistic effects is contradictory to the state of the art as discussed in paragraph 4 of the introduction. Thus, Westinghouse recommends that this paragraph be deleted.

Consideration of synergistic effects is a new requirement for which there are no specific guidelines to apply to equipment involved in a qualification program, that including NUREG/CR-0276 and NUREG/CR-0401. Equipment that is properly qualified to the intent and requirements of IEEE 323-1074 demonstrates its ability to survive and perform its safety function.

The evaluations of synergisms for Class IE equipment appear to be more in line with an R&D program that introduces new equipment, but not in line with the qualification of equipment to the requirements of IEEE 323-1974. See our response to Reg. Guide 1.131 for additional detail.

COMMENT NO. 80D: In paragraph 4(3), synergistic effects should be considered in the accelerated aging programs where applicable.

COMMENT NO. 80D: In Section 4(3), to date, contractor qualification procedures have not included testing methods which would establish synergistic effects.

COMMENT NO. 80F: For item 4(3), this position on synergistic effects implies that every qualification report must include documentation to show that synergistic effects were investigated or that at least a document search was conducted. This is an artifical requirement. Synergistic effects are not "testing" parameters but are the subject of research projects. Even the existence of synergistic effects is questionable depending on how the data are evaluated in the limited research conducted thus far.

This position should be dropped or at least modified to say that synergistic effects need only be addressed where they have been identified. The following rewrite of this paragraph is recommended:

4(3) Synergistic effects should be considered in accelerated aging. Synergistic effects need only be addressed, however, if known synergistic effects exist for the materials of concern. See NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), "Qualification Testing Evaluation Quarterly Reports."

Resolution

The staff is aware that some equipment important to safety may contain materials whose aging effects from combined environments (applied either concurrently or sequentially) are more severe than the sum of the effects of each environmental parameter applied separately. Identifying the most limiting combination of environmental parameters in order to establish a qualified life through research programs, however, may be a long-term, on-going process. Therefore, in lieu of research programs, the qualification program should:

(1) Identify potentially significant synergistic effects through a literature search and account for those effects through testing or analysis when establishing a qualified life, or

(2) Establish through a literature search or operating experience the basis for omitting synergistic considerations.

For equipment where, for example, significant radiation and temperature environments may be present (and in lieu of contrary information determined through items 1 or 2), the synergistic effects to these parameters should be considered during the simulated aging portion of the overall test sequence. The testing sequence used to age the equipment (or material) should be justified and the basis documented in the qualification report. For equipment where thermal aging evaluation has been conducted prior to issuance of this document on non-irradiated equipment or materials, the adequacy of the assumptions made and the conclusions reached will be evaluated on a case-by-case basis. Other methods designed to address synergisms (such as ongoing surveillance with additional qualification testing) may also be found acceptable and will be evaluated on a case-by-case basis.

COMMENT NO. 81: (Section 4(4)) Arrhenius is presumably limited to addressment of thermal aging effects. The following change to item 4(4) is, therefore, recommended:

(4) "The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging. Other thermal aging methods tha can be supported by type tests will be evaluated on a case-by-case basis."

Resolution

The staff agrees. Any modifications to the staff position will be included in the final rulemaking which is planned to be issued for public comment in December 1981.

COMMENT NO. 82: (Section 4(4)) If the NRC considers that acceptable methods exist to address aspects of the qualification, then it would be most helpful if they provide examples similar to those given in the Guidelines for Operating Reactor Qualification (Ref. Denton to Stello, dated November 13, 1979).

Resolution

The examples given in the referenced document are compatible with NUREG-0588. Implementation of the examples should be conditioned on their applicability to the vintage plant, and is outside the scope of this document.

COMMENT NO. 83: (Section 4(4))

In Section 4(4), the Arrhenius equation can be linearized by assuming activation energies are independent of temperature. The linear equation can be used to derive an accelerated aging time by inputting an aging temperature, the desired component life, and ambient temperature. The accelerated aging parameters are then used to type test the component. An alternate approach is to cycle material samples at a number of test temperatures until failure occurs. The data are then used to form a linear regression as described in IEEE 101, "IEEE Guide for the Statistical Analysis of Thermal Life Da a." The regression line can be extrapolated to determine a life based on an ambient temperature. Do these approaches meet the NRC's intent of using the Arrhenius methodology?

Resolution

The test procedures, and the assumptions used, should be evaluated on a case-by-case basis and may be found acceptable.

COMMENT NO. 84: (Section 4(4))

Paragraph 4(4) on page 16 speaks of "The Arrhenius methodology" regarding aging. It is suggested that a reference be given with a source of information on this methodology.

Resolution

Numerous references can be found in qualification publications. The reports identified in Comment No. 80 or the IEEE Standard 101-1072 referenced in Comment No. 83 also provide information on this methodology.

COMMENT NO. 85: (Section 4(5))

Item 4(5): This position requires that known phase changes of materials should be defined to ensure that no changes occur during accelerated aging. This is certainly a valid means of supporting an aging program. However, there are other means that are equally valid. For example, some equipment has undergone previous tests (such as UL tests) at elevated temperatures. If these test temperatures exceed the temperatures being used for accelerated aging, the previous test will provide sufficient evidence that no known changes will occur. To more clearly allow for such alternate methods, it is recommended that this paragraph be rewritten as follows:

"(5) Effects of temperatures used in accelerated aging and within the extrapolation limits must be considered to evaluate materials phase changes. Relevant phase changes should be shown not to occur by defining known phase changes, referencing previous testing, or providing other supportive evidence."

Resolution

The use of previous testing to support the claims that conservative extrapolation limits have been implemented in the qualification programs is acceptable, provided the materials used in previous tests are identical or sufficiently similar so that a comparison is valid. The position is general to allow such specific applications.

COMMENT NO. 86: (Section 4(6)) Endorsement of Arrhenius methodologies should be limited to thermal aging only. We agree that this method should be allowed for want of any better approach. Criteria for selecting conservative activation energies should be included for cases where multiple degradation phenomena are operative or where the activation energy is not known.

Resolution

For cases where equipment is composed of different material components having different activation energies, and testing each component separately is not practical, the testing of the equipment should be conducted using the most limiting (lowest) activation energy of the components.

COMMENT NO. 87: (Section 4(6)) We interpret Section 4(6) as being applicable to post-accident environmental thermal age acceleration also.

Resolution

For equipment that is required to function for an extended period of time in a hostile environment, postaccident environmental aging considerations may be warranted.

COMMENT NO. 88: (Section 4(7)) Section 4(7): The staff appears to be requiring that the plant design include procedures for subjecting the equipment to the limiting service environment conditions. Periodic testing of equipment subjected to the most limiting service environmental conditions would undoubtedly result in more rapid equipment aging. The necessity to perform such testing on components already qualified is questionable. This requirement should be removed.

Resolution

This position applies when the choice of qualification is on-going, in order to extend, verify, or provide a more realistic qualified life. It is the opinion of the staff that component degradation due to aging for the most part may not be readily detectable by visual inspection or testing at only the normal service conditions. However, in the hostile environments this degradation, if significant, should be readily apparent.

COMMENT NO. 89A: (Section 4(8))

For item 4(8), the exemption of humidity from aging considerations for cable should also apply to other insulating materials and particularly for cases where there is significant heat generation within the device to cause humidity reduction. The basis for this exemption should be stated so that other materials can be considered for exemption also. At the present time, there is no known aging mechanism in the electrical materials due to humidity and, certainly, no known method of accelerating this unknown mechanism.

4(8) "The exemption of humidity from aging considerations for cable should apply to any insulating material for which there is adequate justification."

COMMENT NO. 89B:

For item 4(8), the basis for exception to humidity effects on cable should be provided as it is not clear how this can be considered acceptable in a NUREG. There are certainly situations where the humidity effects should be accounted for, particularly if the performance requirements specify the need to demonstrate integrity of the insulation if brittleness could be a factor. This particular parameter should be considered in the same context as all other parameters even for cable.

COMMENT NO. 89C:

In item 4(8), this position is not clear and might be interpreted incorrectly. The effect of relative humidity on aging is discussed in neither IEEE Standard 323-1974 nor Regulatory Guide 1.89 of November 1974. This lack of discussion is valid, however, for relative humidity is not recognized as an aging mechanism. Relative humidity recognized as an environmental parameter for equipment performance and should be addressed in that manner.

Your position that the "effects of relative humidity need not be considered in the aging of electrical cable insulation," is correct. However, the possible interpretation that relative humidity must be considered for all other electrical equipment is not correct. The following rewrite of the paragraph is recommended:

4(8) "Effects of relative humidity need not be considered in the aging of electrical equipment."

Resolution

The effects of humidity on equipment should be considered in the qualification program. Justification, however, may be established to limit the testing of selected equipment to the range and the duration of humidity environments expected at a plant site. A literature search of the tests conducted on identical or similar equipment (or materials) or operating experience may be used to establish a basis for not including rigorous humidity testing. As an example, the Sandia Laboratory report SAND 78-0344 (October 1978) on "Aging of

Nuclear Power Plant Safety Cables" provides assurance that humidity effects on the cable insulation materials tested is not a significant aging contributor. Therefore, for qualification of equipment using these materials, the aging effects due to humidity may be omitted. The basis for these exemptions, however, should be documented.

An NRC-funded research program is presently investigating aging mechanisms due to humidity and is developing methods to qualitatively assess these efforts on selected materials (reference NUREG/CR-1466, "Predicting Life Expectancy and Simulating Age of Complex Equipment Using Accelerated Aging Techniques," April 1980). The staff has not, however, endorsed any one specific method of accelerating humidity. At this time, various methods of accelerating humidity effects during the aging portion of the test program or humidity conditioning during a test sequence may be found acceptable.

COMMENT NO. 90: (Section 5(2)) (Category II)

The documentation requirement for Category II plants allows for exclusion of the format and content of the Standard question given in Appendix E and it is not clear why this is being done. The documentation requirements of 323-71 and suggestions of the previously mentioned Guidelines for Operating Reactors would seem to indicate the need to address these same items in a consistent form for both operating plants and those in the CP or OL phase. Otherwise, the documentation requirements of 323-74 should be followed exclusively, as they are considered sufficient.

Resolution

The information in Appendix E is applicable equally to Category II as well as to Category I plants. The only difference in the position is that Category II plants should utilize the documentation guidelines identified in IEEE Standard 323-1971, whereas Category I plants should utilize IEEE Standard 323-1974.

COMMENT NO. 91: (Section 5(2))

The requirement in Section 5(2) to require "test data" on each piece of complex and varied equipment, much of which could be qualified by extension to equivalent or identical pieces, would be extremely cumbersome and expensive to manage. Paragraph 3.0 of IEEE 334-1975 illustrates the difficulty that might be involved.

It appears sufficient for the last sentence to read in effect:

"...unless it is accompanied by information on the qualification program, including test data or comparable test data from equivalent equipment."

Resolution

The staff position does not exclude the use of data from tests conducted on similar equipment as long as independent verification of similarity or equivalence can be established. It is incumbent on the applicant to have the necessary documentation to justify the adequacy of using data from similar or equivalent equipment. Any modification to the staff positions will be included in the final rulemaking which is planned to be used for public comment in December 1981.

COMMENT NO. 92: (Section 5(2)) Page 17, item 5(2), Better defi- .ion of the level of documentation required to s .. port a certificate of conformance should be provided.

Resolution

See response to Comment No. 91 or the documentation requirement identified in IEEE Standard 323-1971 or IEEE Standard 323-1974. Additional information is also provided in applicable ancillary standards on specific equipment.

COMMENT NO. 93: (Appendix A) Appendix A provides acceptable methods for calculating the mass and energy release both for a LOCA and a main steam line break. For GE, an acceptable reference is stated to be NEDO-10320. This reference is incomplete since it pertains solely to the GE Mark I containment concept. Additions of the appropriate references should be:

a. Mark II containment -- NUREG-0487 (Mark II Interim Acceptance Criteria).

b. Mark III -- NEDO-20533 (GE Mark III Pressure Suppression Containment System Analytical Model) dated June 1974.

Resolution

NUREG-0487 does not contain an acceptable method for calculating the mass and energy release for Mark II containments. The staff has requested that the Mark II applicants perform confirmatory analysis using RELAP 4 to confirm the conservatism in the mass and energy release for Mark II containments as calculated by General Electric.

With respect to NEDO-20533, the staff has accepted the methodology included therein only on a case-by-case basis. The staff does not consider this document fully acceptable on a generic basis. The reference used by the staff is as follows: "Mark III-NEDO-20533 (GE Mark III Pressure Suppression Containment System Analytical Model), dated June 1974, and Supplement 1, dated August 1975."

As a result of this comment, Appendix A was modified to include the "Mark I" reference. The Mark II and III methodology will be evaluated on a case-by-case basis.

COMMENT NO. 94: (Appendix B) The velocity equation used in this section (Appendix B item 2b) is overly conservative. If applied as is, it may yield velocities of several hundred feet per second or more in all areas of the containment. While these high velocities may exist in the region very near the pipe break, they are unreasonably high for remote areas of the containment. This equation does not consider the effects of containment geometry which will affect the convective velocity. Certainly, in the case of PBOC, where the compartment being analyzed is downstream from the break compartment, these velocities are inappropriate. The option should be allowed to calculate velocities for components on a case-by-case basis.

Resolution

The interim position has been developed to conservatively predict the velocities within the containment. It is not the staff's intent to mechanistically arrive at a velocity profile throughout the blowdown, because a mechanistic approach would be complex and would require significant justification for the profiles that were developed. It should be noted that these are interim criteria, and that final resolution will occur upon completion of Task A-21. If more detailed analysis indicates that a less conservative value can be justified, this parameter will be modified. However, at this time, there is no other basis for an alternate approach.

COMMENT NO. 95: (Appendix B)

In BTP CSB 6-1, NRC describes an acceptable heat transfer coefficient for use during LOCA blowdown as a linear ramp from 8 BTU/hr-ft^2 - °F at time = 0 to a peak value of four times the Tagami correlation,

$$h_{max} = 72.5 \left(\frac{Q}{V_{tp}}\right)^{0.62}$$

where h_{max} = 4 x Tagami correlation, BTUs/hr - ft^2 - degrees F

Q = primary coolant energy, BTUs

V = net free containment volume, ft^2

tp = time to end of blowdown, seconds

In Appendix B, item 2 -- acceptable methodology for safety-related component thermal analysis requires the use of the largest of either a condensing heat transfer coefficient based on four times the Uchida correlation, a condensing heat transfer coefficient equal to four times the Tagami correlation, or a convective heat transfer coefficient.

It is unclear whether this requirement is referring to the increasing ramp type Tagami correlation (modified Tagami) used in CSB 6-1 or to the final value of the Tagami correlation, h_{max}. In the latter case it is unclear whether tp is properly defined. In both cases, it is unclear that Q is properly defined.

Please clarify the use of the Tagami correlation for SLB analysis, including all points mentioned above.

Resolution

The intent of the criteria is to maximize the heat transfer coefficient and thereby ensure that the heat flux to components is at a maximum rate. The increasing type Tagami correlation (modified Tagami) as defined in BTP CSB 6-1 is being used. In this Technical Position, "Q" is defined as the energy of the primary system input into the containment at the time that the peak pressure occurs. The time (tp) is defined as the time to the end of blowdown, or that time when approximately 90 percent of the energy is input into the system.

To ensure the selection of a conservative heat transfer coefficient over a range of break sizes that include a turbulent level within the containment, the criteria specify that the maximum heat transfer coefficient should be used by considering 4 x Tagami coefficient or 4 x Uchida coefficient and selecting whichever is greater.

It should be noted that the convective heat transfer should not be used initially, but only as the surface temperature begins to approach the saturation temperature.

COMMENT NO. 96: (Appendix B)

In Section 1.a of Appendix B (Heat Transfer Coefficient), when the containment is superheated and $T_s > T_w$, both convective and condensing heat transfer act simultaneously. The convective heat transfer driving potential is the temperature difference between T_v and T_s and the condensing heat transfer driving potential is the temperature difference between T_s and T_w.

Resolution

Assuming that T_v is equivalent to the bulk temperature, the staff agrees that the convective heat transfer driving potential is the temperature difference between T_v and T_s, and the condensing heat transfer driving potential is the temperature difference between Ts and T_w.

COMMENT NO. 97A: (Appendix C)

We feel and have felt that there is inherent danger in publishing a "universal" environmental profile for use by all in qualifying equipment (Figure C-1). This is the reason IEEE 323-1974 listed numeric values as "typical" and to be used with caution. We recommend the same here.

COMMENT NO. 97B:

Delete note on use of "double spike" (Figure C-1) based on our general comments on Section 3. Also, we feel and have felt that there is inherent danger in publishing a "universal" environmental profile for use by all in qualifying equipment. This is the reason IEEE 323-1974 listed numeric values as "typical" and to be used with caution. We recommend the same here.

COMMENT NO. 97C:

Appendix C - Considering the finite number of both operating plants and plants in the license review process and the NRC manpower devoted to this effort in the NUREG it would seem prudent for the NRC to review, or request review from the utilities involved, of the actual DBE environmental conditions so that these profiles could be provided and envelope profiles drawn based on existing plant configurations. By promulgating a single set of service conditions, the NRC gives the appearance, again, of designing or specifying test conditions where their function is clearly that of addressing the adequacy of what is proposed or has been done. By so specifying these conditions, the NRC is exercising an authority that omits the attendant liability incumbent upon such action.

The basic requirement for qualification is to demonstrate performance under specific service conditions and there are numerous acceptable ways of doing this. The guidance in this Appendix does not address this basic qualification element.

Resolution

The bounding qualification profiles in Appendix C have been generated based on a wide spectrum of postulated accidents. In some cases, these profiles can be considered to be overly conservative; however, in the absence of an approved plant-specific profile, this profile may be used and is considered the minimum bounding profile. In general, the profiles may represent 6 hours of superheat conditions followed by 18 hours of saturated conditions. The actual degree of superheat is left as an open parameter for, as a minimum, the test temperature is to be 340°F for the time specified and the test pressure is to be equal to or greater than the containment design pressure. Obviously, the higher the pressure the less superheat that will exist for a fixed temperature.

For this bounding profile the staff requires a rise in temperature and pressure from normal containment conditions to 340°F and a pressure equal to or greater than the containment design in 10 seconds. This rapid increase in pressure and temperature will provide the component with a severe shock representative of the type of conditions which could be found following a major accident. Following the rapid rise in the test chamber temperature and pressure, the chamber should be stabilized for 6 hours at 340°F to envelope the MSLB conditions.

The basis for the temperature of 340°F and time duration of 6 hours follows from work performed by the staff and General Electric. This is the amount of time one assumes to be required following an accident using a normal cooldown rate of 100°F per hour, because steaming is assumed to occur for that length of time. Because alternative criteria do not exist for a faster cooldown rate, the normal cooldown figure is used. In order to approximate and envelope the LOCA conditions, the temperature and pressure should be reduced after 6 hours to approximately 250°F saturated (approximately 30 psia) and held at these conditions for 2 hours to ensure an equilibrium state has been attained. (This reflects the conditions one might expect following a LOCA.) From 8 to 24 hours, the temperature and pressure can be reduced so that the end point conditions are approximately 250°F and atmospheric pressure. This test for 6 hours at superheated conditions and 18 hours at saturated conditions will bound all possible recirculation line breaks and is, therefore, a bounding test. Any testing beyond 24 hours is beyond the scope of this work and should be addressed in conjunction with postaccident monitoring requirements.

One should recognize that the curve in Figure C-1 is provided for those BWR and PWR ice condenser facilities which do not have plant-specific accident profiles available for use in their specific equipment-qualification program. It would be possible for a utility/applicant to provide, on a case-by-case basis, documentation which yields a plant-unique curve. Should a plant owner want to upgrade a currently installed spray system, it may be possible to shorten the time for the superheat qualification test by providing adequate justification that the spray system will indeed actuate and serve to mitigate the accident, thereby yielding a substantially lower environmental profile. Because Sections 1.1(3) and 1.2(3) clearly allow the use of the plant-specific profiles, no modification was proposed as a result of these comments. (See also staff response to Comment No. 57 regarding the use of a "double spike.")

COMMENT NO. 98: (Appendix C)

Appendix C provides DBE qualification profiles for BWR and Ice Condenser Containments only.

Is it the staff's intention to provide the profiles for PWR and other containments at a later date?

Resolution

The staff does not intend at this time to provide generic temperature profiles for PWRs, because there are significant differences between the PWR plants. As a result, no basis can be established to provide a single generic profile.

COMMENT NO. 99: (Appendix D)

Page 3 of Appendix D references position C.2 of Regulatory Guide 1.89 which appears to be some other source term not discussed in the current Regulatory Guide revision. The NRC cannot reference an apparent revision to a Regulatory Guide which has not yet undergone public, industrial, and ACRS peer review.

Resolution

The position 1.4(1) source terms coupled with the methodology of Appendix D produce a source term consistant with the above position and with current regulatory practice. Appendix D has been modified for clarity when referencing Position C.2 of Regulatory Guide 1.89.

COMMENT NO. 100: (Appendix D)

In Appendix D, page 4, paragraphs G and H, refer to methods which can be used for iodine removal for the PWR. This appendix needs to be expanded with the incorporation of an applicable staff approved and GE reviewed BWR method.

Resolution

The SPIRT program calculations are independent of reactor type. The necessary parameters for the calculation of the spray lambdas are clearly spelled out in NUREG/CR-0009. Heretofore, GE has not used or referenced a spray system design which would result in a large enough value for lambda to result in any appreciable amount of iodine removal. If GE were to supply an appropriate design, credit for iodine removal by action of the sprays may be taken in the calculation of the equipment qualification dose inside containment. No changes to the staff positions are proposed as a result of this comment.

COMMENT NO. 101: (Appendix D)

Since "accurate coupling of the various time sequences is beyond the scope of this analysis," the lengthy discussion on time sequence on pages D-1 through D-5 is unnecessary. Thus, Westinghouse recommends that this discussion be replaced by statement that an instantaneous release and dispersal is conservatively assumed.

Resolution

The staff concurs with the recommendation. Appendix D was modified by deleting the General Summary of the LOCA Scenario Section and incorporating the statement that an instantaneous release is conservatively assumed (see Section 2.1 of the Appendix).

COMMENT NO. 102: (Appendix D)

On page D-6, Section 6, the applicability of the inclusion of solid fission products in the sump water, to radiation dose estimates, is in the long-term dose rates in recirculation equipment.

Resolution

The staff concurs that the solid fission products will affect the long-term dose-rate calculcation for recirculation equipment. The original staff concern, however, is related to the magnitude of an appropriate fission products release of solids. See also response to comments No. 17.

COMMENT NO. 103: (Appendix D)

In Appendix D, Section 7, the handling of daughter products by a simple multiplication factor of 1.3 is not a rigorous approach for a contribution of such magnitude. The emphasis in this improved NUREG has been on mechanistic and analytical treatment in such areas as activity redistribution and spray removal. Therefore, explicit treatment of daughter products should be included.

Resolution

The staff concurs that explicit treatment of the daughter products should be included in any calculation of the qualification dose for equipment. The staff calculations in Appendix D were included only to demonstrate the estimation of qualification doses at a point inside containment, and, therefore a very rigorous approach was not employed.

COMMENT NO. 104: (Appendix D)

The discussion in Appendix D, Section 7A, considers the airborne gamma and beta dose to the containment centerpoint plus the gamma dose to that point from plateout on the containment walls. Why has the gamma and beta dose from plateout on centrally located equipment been ignored? In the past we have found this to be a significant source.

Resolution

The staff agrees that plate-out on centrally located equipment may be significant and should not be ignored. Position 1.4(2) should be interpreted to require that all potential radiation sources be considered when calculating qualification doses (which would include plate-out on centrally located equipment).

As a result of this comment, no modifications are proposed.

COMMENT NO. 105: (Appendix D) For beta radiation, the shielding effects of the humidity in the containment atmosphere (i.e., a density greater than that of dry air) can be significant in reducing doses, particularly during steam release and containment spray periods. Credit for these effects should be explicitly allowed.

Resolution

The staff does not preclude the option of using a different atmospheric density in the containment to calculate the beta and gamma doses provided the assumed values for density are appropriately justified. The adequacy of the justification for the assumptions used (if other than the conservative dry air conditions) will be evaluated on a case-by-case basis.

COMMENT NO. 106A: (Appendix D) The discussion in Appendix D, Section 7B ("Surface Dose and Dose Rates") considers the contribution from airborne beta and gamma sources and plated-out beta sources but it dismisses the plated-out gamma dose contribution as not being significant. The argument given for this is that "the coating is calculated to be relatively permeable to gammas with only about 1% of the plated-out gamms absorbed by the coating." This seems to be a case of misunderstanding of the definition of "dose," viz. Although the amount of energy deposited in a thin layer may be small, the mass of that thin layer is correspondingly small so that (attenuation ignored) the absorbed dose due to a given incident gamma field is independent of the coating thickness. (Note: Microdosimetric considerations such as electron equilibrium are second order effects and have no impact on the above mentioned concerns.)

COMMENT NO. 106B: The first and second sentences of page D-8, Section 7(b), paragraph 3, are inconsistent. The first refers to gamma exposure rate due to airborne activity while the second refers to gamma absorption rate due to plate-out activity. Since the absorption properties are a function of gamma ray energy and not the location of the source, Westinghouse recommends that exposure rate be used for consistency.

Resolution

The staff concurs. A model used in the "For Comment" version of NUREG-0588 incorrectly calculated the gamma doses in the vicinity of the wall. The doses near the containment walls should consider the photon flux from all sources. As a result of the comment, Appendix D was modified to remove the model.

COMMENT NO. 107: (Appendix D) In Appendix D, we assume that all doses calculated are for a dose point material of air. We would recommend normalizing dose to rads-carbon. This should be stated explicitly and thereby indicate the appropriate method of dosimetry to be applied when testing.

Resolution

All doses have been calculated at a dose point in air. The staff calculation seeks only to illustrate the model, and the values are not to be used as actual qualification values. For actual qualification values, the radiation dose values can be specified in units of rads for the absorbing material or by normalizing to rads-carbon, if preferred.

COMMENT NO. 108: (Appendix D) Our attempts to reproduce the evaluations of Appendix D lead us to believe that gamma buildup factors were not taken into account. We recommend that this consideration be included.

Resolution

The staff concurs that the dose calculations given in Appendix D did not incorporate the use of gamma buildup factors. A modified Appendix D explicitly cites the use of buildup factors for the sump source. The gamma buildup factors in the containment atmosphere were assumed equal to unity.

COMMENT NO. 109: (Appendix D) A definition of "shielded" as it is used in items 1.4.7 and 1.4.8 on page 9 of Appendix D is needed.

Resolution

As defined in the Radiological Health Handbook. January 1970, a shield is any body of material used to prevent or reduce the passage of particles or radiation.

COMMENT NO. 110: (Appendix D) The effect on radiation qualification of ECCS equipment leakage is mentioned on page 2 of Appendix D. Was this effect ignored in the Appendix D analysis?

Resolution

Appendix D does not address (in the sample calculations) the dose contributions resulting from ECCS leakage. Position 1.4(2) should be interpreted to require that all potential radiation sources be considered when calculating qualification doses, which should include dose contributions from ECCS leakage where appropriate (for example, for selected areas outside containment).

COMMENT NO. 111: (Appendix D) In Appendix D more consideration should be given to the accurate use of dosimetric terminology. Rad and R (Roentgen) are used interchangably in the tables of Appendix D where they shouldn't be. In particular (for example), the use of R (Roentgen) to specify beta-dose is inappropriate. The Roentgen is a unit of "exposure" which is a dosimetric concept reserved for the measurement of ionization of air in a gamma or x-ray field. All doses must be given in rads, and for exactness should be given in rad-carbon, since at the high energies experienced post-accident the "Z" of the receiver material will have a significant effect on the absorbed dose from gammas.

Resolution

The staff agrees. As a result of this comment, the tables of Appendix D have been modified to reflect the recommendations.

COMMENT NO. 112: (Appendix D) In Appendix D, Table D-10 gives the dose rates near an ECCS recirculation pipe. To be useful, it is important to know the size of the pipe and the time post-accident for which the dose rates were determined. Integrated doses would be more useful for radiation qualification purposes than are the dose rates.

Resolution

The staff concurs. The calculation of dose rates in recirculation piping should be performed in a plant-specific manner using guidance contained in NUREG-0578.

COMMENT NO. 113: (Appendix D) The values given in the table on page D-1 do not correspond to those in Tables D-5 through D-8. This inconsistency should be resolved.

Resolution

Appendix D has been corrected to resolve this inconsistency.

PART II
APPENDIX A

Rev

REVISION 1

METHODS FOR CALCULATING

MASS AND ENERGY RELEASE

APPENDIX A

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

Acceptable methods for calculating the mass and energy release to determine the loss-of-coolant accident (LOCA) environment for PWR and BWR plants are described in the following:

(1) Topical Report WCAP-8312A (Revision 2, August 1975), for Westinghouse plants. Rev

(2) Section 6.2.1 of CESSAR System 80 PSAR through Amendment 36 for Combustion Engineering plants. Rev

(3) Appendix 6A of B-SAR-205 through Amendment 15 for Babcock & Wilcox plants. Rev

(4) NEDO-10320 and Supplements 1 & 2 for General Electric Mark I plants. (For GE Mark II and III containments, the methods will be evaluated on a case-by-case basis.) Rev

Acceptable methods for calculating the mass and energy release to determine the main steam line break (MSLB) environment are described in the following:

(1) Appendix 6B of CESSAR System 80 PSAR through Amendment 36 for Combustion Engineering plants. (The analysis should also include single-failure considerations. The justification for the limiting case that is used will be evaluated by the staff on a case-by-case basis.) Rev

(2) Section 15.1.14 of B-SAR-205 through Amendment 15 for Babcock & Wilcox plants. Rev

(3) Same as item (4) above for General Electric plants.

(4) Topical Report WCAP-8822 (Dated September 1976) for Westinghouse plants. (Although this Topical Report is currently under review, the use of this method is acceptable in the interim if no entrainment is assumed. Reanalysis may be required following the NRC staff review of the entrainment model as presently described.) Rev

PART II
APPENDIX B

Rev

REVISION 1

MODEL FOR ENVIRONMENTAL QUALIFICATION FOR

LOSS-OF-COOLANT ACCIDENT AND MAIN STEAM LINE BREAK

INSIDE PWR AND BWR DRY TYPE OF CONTAINMENT

APPENDIX B

MODEL FOR ENVIRONMENTAL QUALIFICATION FOR LOSS-OF-COOLANT ACCIDENT AND MAIN STEAM LINE BREAK INSIDE PWR AND BWR DRY TYPE OF CONTAINMENT

1. Methodology to Determine the Containment Environmental Response

a. Heat Transfer Coefficient

For heat transfer coefficient to the heat sinks, the Tagami condensing heat transfer correlation should be used for a LOCA with the maximum heat transfer rate determined at the time of peak pressure or the end of primary system blowdown. A rapid transition to a natural convection, condensing heat transfer correlation should follow. The Uchida heat transfer correlation should be used for MSLB accidents while in the condensing mode. A natural convection heat transfer coefficient should be used at all other times when not in the condensing heat transfer mode for both LOCAs and MSLB accidents. The application of these correlations should be as follows:

(1) Condensing heat transfer

$q/A = h_{cond}$ " $(T_s - T_w)$

where q/A = the surface heat flux

h_{cond} = the condensing heat transfer coefficient

T_s = the steam saturation (dew point) temperature

T_w = surface temperature of the heat sink

(2) Convective heat transfer

$q/A = h_c$ " $(T_v - T_w)$

where h_c = convective heat transfer coefficient

T_v = the bulk vapor temperature

All other parameters are the same as for the condensing mode.

b. Heat Sink Condensate Treatment

When the containment atmosphere is at or below the saturation temperature, all condensate formed on the heat sinks should be transferred directly to the sump. When the atmosphere is superheated, a maximum of 8 percent of the condensate may be assumed to remain in the vapor region. The condensed mass should be calculated as follows:

$M_{cond} = X$ " $q / (h_v - h_L)$

where M_{cond} = mass condensation rate

X = mass condensation fraction (0.92)

q = surface heat transfer rate

h_v = enthalpy of the superheated steam

h_L = enthalpy of the liquid condensate entering the sump region (i.e., average enthalpy of the heat sink condensate boundary layer) | Rev

c. Heat Sink - Surface Area

The surface area of the heat sinks should correspond to that used for the containment design pressure evaluation.

d. Single Active Failure Evaluation

Single active failures should be evaluated for those containment safety systems and components relied upon to limit the containment temperature/pressure response to a LOCA or MSLB accident. This evaluation should include, but not necessarily be limited to, the loss or availability of offsite power (whichever is worse), diesel generator failure when loss of offsite power is evaluated, and loss of containment heat removal systems (either partial or total, whichever is worse).

e. Containment Heat Removal System Actuation

The time determined at which active containment heat removal systems become effective should include consideration of actuation sensors and setpoints, actuation delay time, and system delay time (i.e., time required to come into operation).

f. Identification of Most Severe Environment

The worst case for environmental qualification should be selected considering time duration at elevated temperatures as well as the maximum temperature. In particular, consider the spectrum of break sizes analyzed and single failures evaluated.

2. Acceptable Methodology for Safety-Related Component Thermal Analysis

Component thermal analyses may be performed to justify environmental qualification test conditions that are found to be less than those calculated during the containment environmental response calculation.

The heat transfer rate to component should be calculated as follows:

a. Condensing Heat Transfer Rate

$$q/A = h_{cond}'' (T_s - T_w)$$

where q/A = component surface heat flux

h_{cond} = condensing heat transfer coefficient is equal to the larger of 4x Tagami correlation or 4x Uchida correlation

T_s = saturation temperature (dew point)

T_w = component surface temperature

b. Convective Heat Transfer

A convective heat transfer coefficient should be used when the condensing heat flux is calculated to be less than the convective heat flux. During the blowdown period, a forced convection heat transfer correlation should be used. For example:

$$NU = C\ (Re)^n$$

where Nu = Nusselt number

Re = Reynolds number

C,n = empirical constants dependent on geometry and Reynolds number

The velocity used in the evaluation of Reynolds number may be determined as follows:

$$V = 25\ \frac{M_{BD}}{V_{CONT}}$$

where V = velocity in ft/sec

M_{BD} = the blowdown rate in lbs/hr

V_{CONT} = containment volume in ft3

After the blowdown has ceased or reduced to a negligibly low value, a natural convection heat transfer correlation is acceptable. However, use of a natural convection heat transfer coefficient must be fully justified whenever used.

PART II   Rev
APPENDIX C

REVISION 1

QUALIFICATION PROFILES FOR

BWR AND ICE CONDENSER CONTAINMENTS

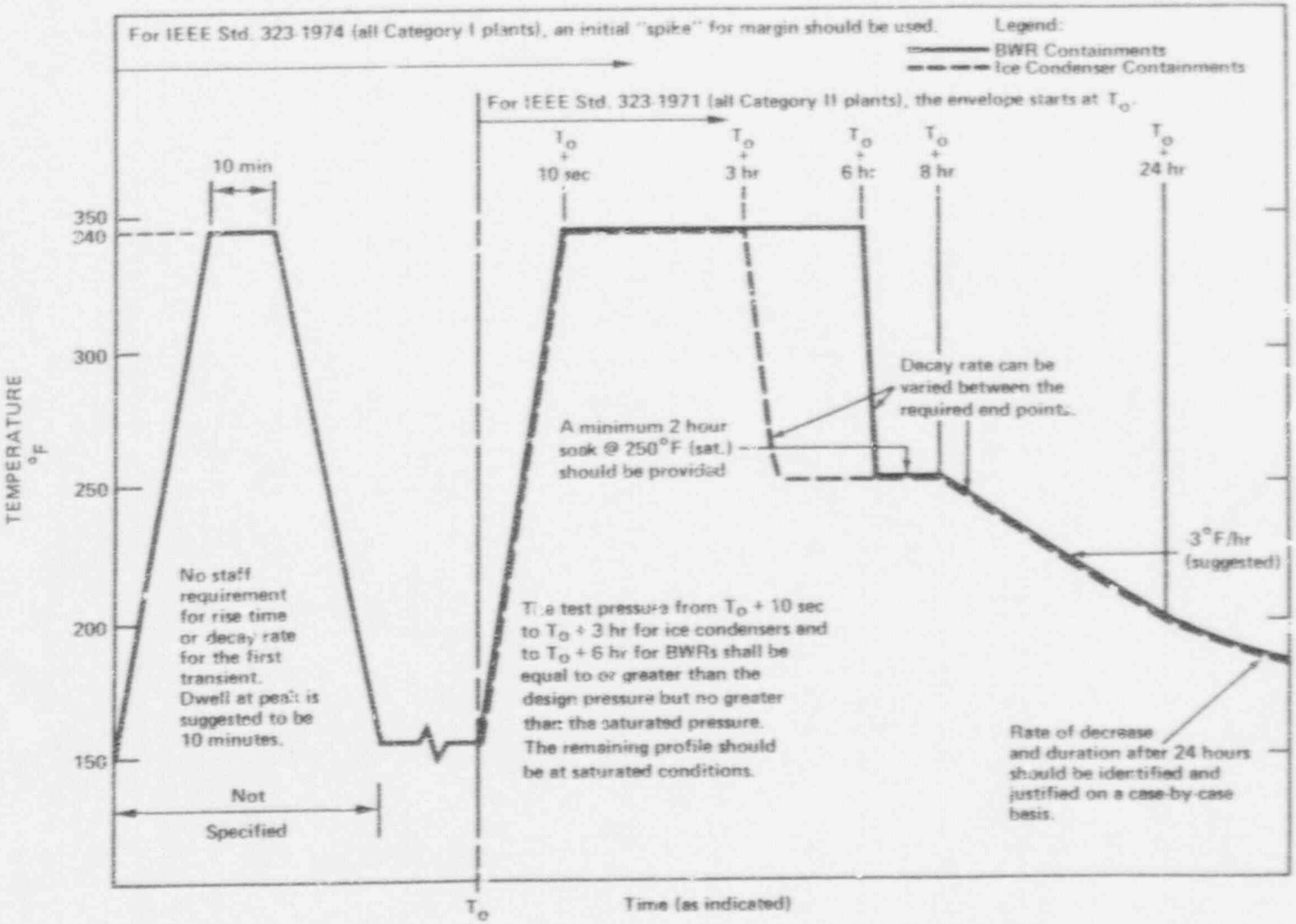

Figure C-1. Qualification Profiles for BWR and Ice Condenser Containments

11C-1

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PART II
APPENDIX D

REVISION 1

SAMPLE CALCULATION AND TYPE METHODOLOGY

FOR RADIATION QUALIFICATION DOSE

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

This appendix illustrates the proposed staff model for calculating dose rates and integrated doses for equipment qualification purposes. The doses shown in Fig. D-1 include contributions from several source locations in the containment and cover a period of one year following the postulated fission product release. Rev
The dose values shown here are provided for illustration only and may not be appropriate for plant-specific application. The dose levels intended for qualification purposes should be determined using the maximum time the equipment is intended to function which, for the design basis LOCA event, may well exceed one year.

The beta and gamma integrated doses presented in the tables and in Figure D-1 have been determined using models and assumptions consistent with those of Regulatory Guide 1.7 and 1.89. This analysis is conservative, and factors in Rev
the important time-dependent phenomena related to the action of engineered safety features (ESFs) and natural phenomena, such as iodine plate-out, as done in the previous staff analyses.

The doses presented in Figure D-1 were calculated at the midpoint of the containment and are expected to be representive values for PWR plants having a contain- Rev
ment free-volume of 2.5 million cubic feet and a power rating of 4100 MWt.

1. <u>Basic Assumptions Used in an Equipment Qualification Analysis</u> Rev

Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) from activity suspended in the containment atmosphere, (2) from activity plated-out on containment surfaces, and (3) from activity mixed in the containment sump water. Thus, a given piece of Rev
equipment may receive a dose contribution from any or all of these sources. The amount of dose contributed by each of these sources is determined by the location of the equipment, the time-dependent and location-dependent distribution of the source, and the effects of shielding.

The source term as set forth in the <u>"For Comment"</u> version of NUREG-0588 is consistent with the C.2 position of Regulatory Guide 1.89 (dated November 1974) and represents the staff licensing position for released fission product activity (i.e., a TID 14844 release).

Although the TMI-2 accident represents only one of a number of possible accident sequences leading to a release of fission products, the staff concluded that a thorough examination of the source term assumptions for equipment qualification was warranted. Current rulemaking proceedings are re-evaluating plant siting policy, degraded cores, minimum requirements for engineered safety features and emergency preparedness. These rulemaking activities also included an examination of fission product releases under degraded core conditions. The final resolution of the source term assumptions is conditioned on the completion of these rulemaking efforts. The staff believes it is prudent to factor in the knowledge of fission product behavior gained from the TMI-2 accident in defining source term assumptions for equipment qualification.

Based upon the fission product release estimates in Volume II, Part 2 of the Rogovin Report (Ref. 8), the staff assumptions for noble gas and iodine releases are still conservative. However, the report estimates that the TMI-2 release contained between 40 and 60 percent of the Cs-134 and Cs-137 core activity in the primary system water, in the containment sump water, and in the auxiliary building tanks.

As part of the effort to incorporate TMI-2 data into the licensing process, the Commission directed the staff to initiate an effort which would investigate the adequacy of the current fission product release assumptions, particularly the past staff assumptions (such as Regulatory Guide 1.3 and Regulatory Guide 1.4) concerning fission product (iodine, cesium, solids, and so forth) behavior following a severe accident. The staff findings from this investigation are presented in two reports, NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions" (Ref.9), which discusses the impact of fission product source term assumptions on past licensing practice, present regulations, and possible future licensing actions, and NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents" (Ref. 10), which contains a description of the most recent technical information currently available for estimating the release of fission products during postulated accidents in commercial LWRs.

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The staff feels that as a first step toward modification of the TID-14844 source term in the direction indicated by the TMI-2 experience, it may be prudent to factor in a cesium release in addition to the previously assumed "1% solids." This change in assumptions would have particular significance for the qualification of equipment in the vicinity of recirculating fluids and for equipment required to function for time periods exceeding 30 days. The conclusions from the reports cited above will form the basis for any revision of source term assumptions, and any changes of the source terms will be factored into the final rulemaking in equipment qualification.

2. Assumptions Used in Calculating Fission Product Concentrations

This section discusses the assumptions used to simulate the PWR and BWR containments for determining the time-dependent and location-dependent distribution of noble gases and iodines airborne within the containment atmosphere, plated-out on containment surfaces and located in the containment sump water.

The staff has developed a computer program TACT (to be published) that models the time-dependent behavior of iodine and noble gases within a nuclear power plant. The TACT code is used routinely by the staff for the calculation of the offsite radiological consequences of a LOCA, and is an acceptable method for modeling the transfer of activity from one containment region to another and in modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 6), is used to calculate the removal rates of elemental iodine by plate-out and sprays. These codes were used to develop the source term estimates. The following assumptions were also used to calculate the distribution of radioactivity within the containment following a design basis LOCA.

2.1 PWR Dry Containments

a. The source terms used in the analysis assumes that 50 percent of the core iodines and 100 percent of the core noble gases were released instantaneously to the containment atmosphere. Also one percent of the remaining core activity inventory of solids should be assumed to be released from the core and carried with the primary coolant to the containment sump. (NOTE: A potential change in this latter assumption is being considered by the staff for the final rulemaking, as noted above.)

b. The containment free volume was taken as 2.52×10^6 ft^3. Of this volume, 74 percent or 1.86×10^6 ft^3 is assumed to be directly covered by the containment sprays. (Plants with different containment free volumes should use plant-specific values.)

c. 6.6×10^5 ft^3 of the containment free volume is assumed unsprayed, which includes regions within the main containment space under the containment dome and compartments below the operating floor level.

d. The ESF fans are assumed to have a design flow rate of 220,000 cfm in the post-LOCA environment. Mixing between all major unsprayed regions and compartments and the main spray region is assured.

e. Air exchange between the sprayed and unsprayed region was assumed to be one-half of the design flow rate of ESF fans. Good mixing of the con- Rev
tainment activity between the sprayed and unsprayed regions is assured by natural convection currents and ESF fans.

f. The containment spray system was assumed to have two equal capacity trains, each designed to inject 3000 gpm of boric acid solution into the containment.

g. Trace levels of hydrazine were assumed added to enhance the removal of iodine.

h. The spray removal rate constant (lambda) was calculated using the staff's SPIRT program (Ref. 6), conservatively assuming only one spray train operation and an elemental iodine instantaneous partition coefficient (H) of 5000. The calculated value of the elemental iodine spray removal constant was 27.2 hr^{-1}.

i. The assumption that 50 percent of the released iodine activity is instantaneously plated out was not used. Plate-out of iodine on containment internal surfaces was modeled as a first-order rate removal process and best estimates for model parameters were assumed. Based on an assumed total surface area within containment of approximately 5.0×10^5 ft^2, the calculated value for the overall elemental iodine plate-out constant was 1.23 hr^{-1}. The assumption that 50 percent of the activity is instantaneously plated-out was not used.

j. The spray removal and plate-out process were modeled as competing iodine removal mechanisms.

k. A spray removal rate constant (λ) for particulate iodine concentration was assumed to have a value of $\lambda = 0.43\ hr^{-1}$ and allowed the removal of particulate iodine to continue until the airborne concentration was reduced by a factor of 10^4. The organic iodine concentration in the containment atmosphere is assumed unaffected by either the containment spray or plate-out removal mechanisms.

l. The spray and plate-out removal processes were assumed to remove elemental iodine until the instantaneous concentration in the sprayed region was reduced by a factor of 200. This is necessary to achieve an equilibrium airborne iodine concentration consistent with previous LOCA analyses.

m. A relatively open (not compartmented) containment was assumed, and the large release was uniformly distributed in the containment. This is an adequate simplification for dose assessment in a PWR containment, and realistic in terms of specifying the time-dependent radiation environment in most areas of the containment.

n. The analysis assumed that more than one species of radioactive iodine is present in a design basis LOCA. The calculation of the post-LOCA environment assumed that 5 percent of the core inventory of the iodine released is associated with airborne particulate materials and 4 percent of the core inventory of the iodine released formed organic compounds. The remaining 91 percent remained as elemental iodine. For conservatism this composition was assumed present at time t=0. Rev

o. For all containments, no leakage from the containment building to the environment was assumed.

p. Removal of airborne activity by engineered safety features may be assumed when calculating the radiation environment following other non-LOCA design basis accidents, provided the safety features systems are automatically activated as a result of the accident.

2.2 <u>PWR Ice Condenser Containments</u>

The assumptions and methods presented for the calculation of the radiation environment in PWR dry containments are appropriate for use in calculating the radiation environment following a design basis LOCA for ice condenser containments with the following modifications:

a. The source should be assumed to be initially released to the lower containment compartment. The distribution of the activity should be based on the forced recirculation fan flow rates and the transfer rates through the ice beds as a function of time.

b. Credit may be taken for iodine removal via the operation of the ice beds and the spray system. A time-dependent removal efficiency consistent with the steam/air mixture for elemental iodine may be assumed.

c. Removal of airborne iodine in the upper compartment of the containment by the action of both plate-out and spray processes may be assumed provided that these removal processes are evaluated using the assumptions consistent with items h through l in Section 2.1 above and <u>plant-specific</u> parameters.

2.3 BWR Containments

The assumptions and methods presented for the calculation of the radiation environment in PWR dry containments are appropriate for use in calculating the radiation environment following a design basis LOCA for BWRs with the following modifications:

a. A decontamination factor (DF) of 10 should be assumed for both the elemental and particulate iodine as the iodine activity is assumed to pass through the suppression pool. No credit should be taken for the removal of organic iodine or noble gases in the suppression pool.

b. For Mark III designs, all the activity passing through the suppression pool should be assumed instantaneously and uniformly distributed within the containment. For the Mark I and Mark II designs, all the activity should be assumed initially released to the drywell area and the transfer rates of activity from these regions to the surrounding reactor building volume should be used to predict the qualification levels within the reactor building (secondary containment).

c. Removal of airborne iodine in the drywell or reactor building by both the plate-out and the spray process may be assumed provided the effectiveness of these competing iodine removal processes are evaluated using the assumptions consistent with items h through l in Section 2.1 above and plant-specific parameters. Rev

d. The removal of airborne activity from the reactor building by operation of the standby gas treatment system (SGTS) may be assumed.

3.0 Model for Calculating the Dose and Dose Rate of Airborne and Plate-Out Fission Products

The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere were calculated for a midpoint in the containment. The containment was modeled as a cylinder of equal height and diameter. Containment shielding and internal structures were neglected because this was considered to involve a degree of complexity beyond the scope of the present work. The calculations of both References 4 and 11 indicate that the specific internal shielding and structure would be expected to reduce the gamma doses and dose rates by factors of two or more, depending upon the specific location and geometry.

Because of the short range of the betas in air, the airborne beta doses were calculated using an infinite medium approximation. This is shown in Reference 2 to result in only a small error. For beta dose calculations for equipment located on the containment walls or on large internal structures, the semi-infinite beta dose model may be used.

The gamma dose rate contribution from the plate-out iodine on containment surfaces to the point on the centerline was also included. The model calculated the plate-out activity in the containment assuming only one spray train and one ventilation system were operating. It should be noted that wash-off by the sprays of the plated-out iodine was not addressed in this evaluation.

Finally, all gamma doses were multiplied by a correction factor of 1.3, as suggested in Reference 2, to account for the omission of the contribution from the decay chains of the isotopes.

4.0 Model for Calculating the Dose and Dose Rate of Sump Fission Products

The staff model assumed the washout of airborne iodine from the containment atmosphere to the containment sump. For a PWR containment with sprays and good mixing between the sprayed and unsprayed regions, the elemental iodine (assumed constituting 91 percent of the released iodine) is very rapidly washed out of the atmosphere to the containment sump (typically, 90 percent of the airborne iodine in less than 15 minutes).

The dose calculations assumed a time-dependent iodine source. (The difference between the integrated dose assuming 50 percent of the core iodine immediately available in the sump versus a time-dependent sump iodine buildup is not significant.)

The "solid" fission products should be assumed instantaneously carried by the coolant to the sump and uniformly distributed in the sump water. The gamma and beta dose rates and the integrated doses should be computed for a centerpoint located at the surface of the large pool of sump water and the dose rates should be calculated including an estimate of the effects of buildup. Rev

5.0 Conclusion

The values given in the tables and in Figure D-1 for the center point in the containment provide an estimate of expected radiation qualification values for Rev a 4100 MWt PWR design at that location.

The NRC Office of Research is continuing its research efforts in the area of source terms for equipment qualification following design basis accidents. As more information in this area becomes available, the source terms and staff models may change to reflect the new information.

TABLE D-1

SUMMARY TABLE OF ESTIMATES FOR
TOTAL AIRBORNE GAMMA DOSE CONTRIBUTORS
IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

TIME (HRS)	AIRBORNE IODINE DOSE (R)	AIRBORNE NOBLE GAS DOSE (R)	PLATE-OUT IODINE DOSE (R)	TOTAL DOSE (R)
0.00	-	-	-	-
0.03	4.82+4*	7.42+4	1.69+3	1.24+5
0.06	8.57+4	1.39+5	3.98+3	2.29+5
0.09	1.09+5	1.98+5	7.22+3	3.14+5
0.12	1.25+5	2.51+5	1.10+4	3.87+5
0.15	1.38+5	3.01+5	1.52+4	4.54+5
0.18	1.47+5	3.48+5	1.96+4	5.15+5
0.21	1.55+5	3.92+5	2.41+4	5.71+5
0.25	1.64+5	4.49+5	3.03+4	6.43+5
0.38	1.87+5	6.19+5	5.05+4	8.57+5
0.50	2.03+5	7.61+5	6.90+4	1.03+6
0.75	2.36+5	1.03+6	1.06+5	1.37+6
1.00	2.66+5	1.26+6	1.40+5	1.67+6
2.00	3.62+5	2.04+6	2.61+5	2.66+6
5.00	5.50+5	3.56+6	5.40+5	4.65+6
8.00	6.63+5	4.38+6	7.47+5	5.79+6
24.0	1.01+6	6.26+6	1.45+6	8.72+6
60.0	1.31+6	7.16+6	2.10+6	1.06+7
96.0	1.45+6	7.56+6	2.39+6	1.14+7
192.	1.68+6	8.29+6	2.86+6	1.28+7
298.	1.85+6	8.76+6	3.19+6	1.38+7
394.	1.95+6	8.85+6	3.41+6	1.42+7
560.	2.07+6	9.06+6	3.64+6	1.48+7
720.	2.13+6	9.15+6	3.76+6	1.50+7
888.	2.16+6	9.19+6	3.83+6	1.52+7
1060	2.18+6	9.21+6	3.87+6	1.53+6
1220	2.19+6	9.21+6	3.89+6	1.53+7
1390	2.20+6	9.21+6	3.90+6	1.53+7
1560	2.20+6	9.22+6	3.91+6	1.53+7
1730	2.20+6	9.22+6	3.91+6	1.53+7
1900	2.20+6	9.22+6	3.92+6	1.53+7
2060	2.20+6	9.22+6	3.92+6	1.53+7
2230	2.20+6	9.22+6	3.92+6	1.53+7
2950	2.20+6	9.23+6	3.92+6	1.54+7
3670	2.20+6	9.24+6	3.92+6	1.54+7
4390	2.20+6	9.24+6	3.92+6	1.54+7
5110	2.20+6	9.25+6	3.92+6	1.54+7
5830	2.20+6	9.25+6	3.92+6	1.54+7
6550	2.20+6	9.26+6	3.92+6	1.54+7
7270	2.20+6	9.26+6	3.92+6	1.54+7
8000	2.20+6	9.27+6	3.92+6	1.54+7
8710	2.20+6	9.28+6	3.92+6	1.54+7
			TOTAL	1.54+7

*+4=10⁴

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TABLE D-2*

SUMMARY TABLE OF ESTIMATES FOR
TOTAL AIRBORNE BETA DOSE CONTRIBUTORS
IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

TIME (HRS)	AIRBORNE IODINE DOSE (RADS)	AIRBORNE NOBLE GAS DOSE (RADS)	TOTAL DOSE (RADS)
0.00	-	-	-
0.03	1.47+5	5.48+5	6.95+5
0.06	2.62+5	9.86+5	1.25+6
0.09	3.33+5	1.35+5	1.68+6
0.12	3.83+5	1.65+6	2.03+6
0.15	4.20+5	1.91+6	2.33+6
0.18	4.49+5	2.14+6	2.59+6
0.21	4.73+5	2.35+6	2.82+6
0.25	5.00+5	2.60+6	3.10+6
0.38	5.67+5	3.30+6	3.87+6
0.50	6.15+5	3.86+6	4.48+6
0.75	7.13+5	4.89+6	5.60+6
1.00	8.00+5	5.81+6	6.61+6
2.00	1.07+6	9.02+6	1.01+7
5.00	1.58+6	1.65+7	1.81+7
8.00	1.88+6	2.20+7	2.39+7
24.0	2.87+6	4.08+7	4.37+8
60.0	3.89+6	6.15+7	6.54+7
96.0	4.37+6	7.48+7	7.92+7
192.	5.14+6	1.00+8	1.05+8
298.	5.64+6	1.17+8	1.23+8
394.	5.99+6	1.25+8	1.31+8
560.	6.34+6	1.34+8	1.40+8
720.	6.53+6	1.39+8	1.46+8
888.	6.[illegible]+6	1.42+8	1.49+8
1060	6.69+6	1.44+8	1.51+8
1220	6.73+6	1.45+8	1.52+8
1390	6.75+6	1.47+8	1.54+8
1560	6.76+6	1.49+8	1.56+8
1730	6.76+6	1.51+8	1.58+8
1900	6.76+6	1.52+8	1.59+8
2060	6.76+6	1.54+8	1.61+8
2230	6.77+6	1.55+8	1.62+8
2950	6.77+6	1.62+8	1.69+8
3670	6.77+6	1.69+8	1.76+8
4390	6.77+6	1.76+8	1.83+8
5110	6.77+6	1.83+8	1.90+8
5830	6.77+6	1.89+8	1.96+8
6550	6.77+6	1.96+8	2.03+8
7270	6.77+6	2.03+8	2.10+8
8000	6.77+6	2.09+8	2.16+8
8710	6.77+6	2.16+8	2.23+8
		TOTAL	2.23+8

*Tables D-3 through D-9 have been deleted.
+Dose conversion factor is based on absorption to tissue.

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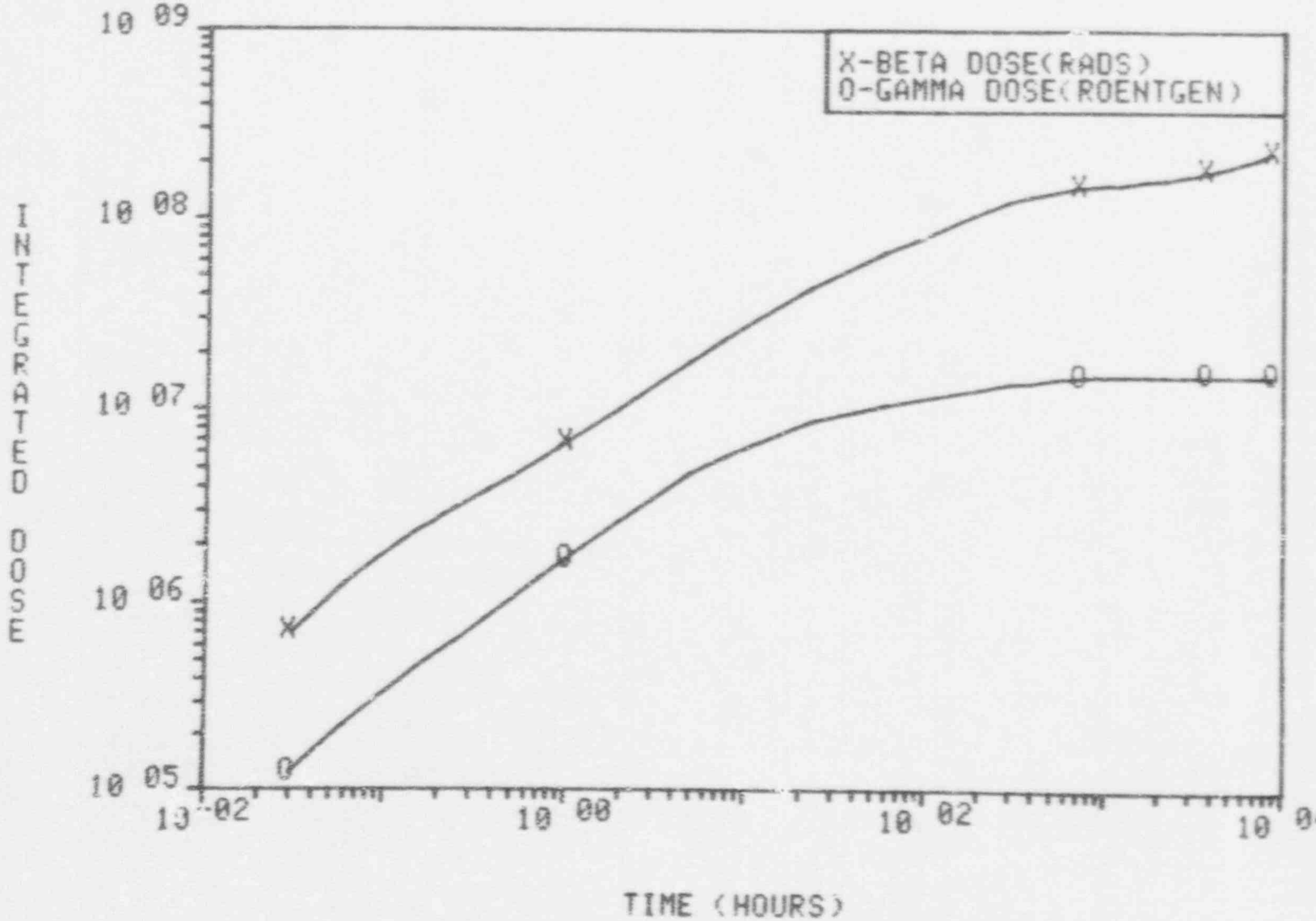

Figure D-1 Sample airborne doses for a dose point on the containment centerline

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BIBLIOGRAPHIC DATA SHEET — NUREG-0588 Rev. 1

Field	Entry
1. TITLE AND SUBTITLE (Add Volume No., if appropriate)	Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment Subtitle: Including Staff Responses to Public Comments
2. (Leave blank)	
3. RECIPIENT'S ACCESSION NO.	
7. AUTHOR(S)	A. J. Szukiewicz and others
5. DATE REPORT COMPLETED	MONTH: November — YEAR: 1980
DATE REPORT ISSUED	MONTH: July — YEAR: 1981
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Safety Technology Office of Nuclear Reactor Regulation U. S. Nuclear Regulatory Commission Washington, D. C. 20555
6. (Leave blank)	
8. (Leave blank)	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Same as 9 above.
10. PROJECT/TASK/WORK UNIT NO.	
11. CONTRACT NO.	
13. TYPE OF REPORT	Regulatory Report
PERIOD COVERED (Inclusive dates)	
15. SUPPLEMENTARY NOTES	This report documents resolution of Unresolved Safety Issue A-24
14. (Leave blank)	

16. ABSTRACT (200 words or less)

This document provides the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment, in the resolution of Unresolved Safety Issue A-24, "Qualification of Class IE Safety-Related Equipment." The positions herein are applicable to plants that are or will be in the construction permit (CP) or operating license (OL) review process and that are required to satisfy the requirements set forth in either the 1971 or the 1974 version of IEEE-323 standard. Part I of this report contains the original "For Comment" NUREG that was published in December 1979. This "For Comment" issue is now endorsed by the Commission, in the May 23, 1980 Memo-
randum and Order (CLI-80-21), as the staff's interim positions until the final positions,
currently being developed in rulemaking, are established. Part II of this reports con-
tains the staff's responses and resolution of the public comments that were solicited
and received before May 1, 1980. Appendices A through D identify the additions, modifi-
cations and/or corrections that were made as a result of these comments.

Field	Entry
17. KEY WORDS AND DOCUMENT ANALYSIS	
17a. DESCRIPTORS	
17b. IDENTIFIERS/OPEN-ENDED TERMS	
18. AVAILABILITY STATEMENT	Unlimited
19. SECURITY CLASS (This report)	Unclassified
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2/14/92

SSINS No.: 6835
IN 83-72

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

OCTOBER 28, 1983

IE INFORMATION NOTICE 83-72: ENVIRONMENTAL QUALIFICATION TESTING EXPERIENCE

Addressees:

All holders of a nuclear power reactor operating license (OL) or construction permit (CP).

Purpose:

This information notice is provided to inform the licensees of environmental qualification test failures. These test failures are based on (1) Construction Deficiency Reports and 10 CFR Part 21 Reports submitted to the NRC, and (2) results from the NRC-sponsored environmental qualification methodology research program. This information notice also serves to inform the licensees of findings that resulted from inspections conducted by the licensee or its agent of equipment that has been environmentally qualified and is being delivered or installed at the sites.

Because of the potential safety significance and related generic implications of these test failures and inspection findings, addressees are expected to review the information for applicability to their facilities. No specific response to this information notice is required.

Description of Circumstances:

The NRC has received a number of Construction Deficiency Reports and 10 CFR Part 21 Reports from licensees and vendors of safety-related equipment. These reports describe a number of test failures and the circumstances under which the equipment failed to function during environmental qualification testing. These reports also indicate that there are a number of instances in which delivered equipment and components contained material that did not conform to standards for safety, thus rendering the qualified equipment and components unqualified. In addition to the monitoring and assessing of environmental qualification information received from the industry, the NRC has also sponsored a number of qualification tests of certain safety-related equipment under its environmental qualification methodology research program, which has resulted in a number of adverse test results. This information notice is published with the following objectives:

1. To disseminate the information on matters related to the environmental qualification of equipment and on test results, as received from the licensees and equipment vendors.

~~8308310048~~

NUCLEAR REGULATORY COMMISSION

Docket No. 50-348/364 Official Exh. No. 56

In the matter of ALABAMA POWER CO.

Staff ✓ IDENTIFIED 2/14/92

Applicant RECEIVED 2/14/92

Intervenor REJECTED

Cont'g Off'r

Contractor DATE 10-28-83

Other Witness

Reporter L. Estep

2. To disseminate the results of NRC-sponsored environmental qualification tests which have been completed.

The enclosed continuing series of Equipment Environmental Qualification Notices (Attachment 1) describes the circumstances of each failure, failure mode, and qualification concerns as described in various reports and sources indicated. Please note that for items in Qualification Notices No. 14 through 19 the vendors have issued service instructions to the affected users regarding corrective action to be taken.

Questions regarding the details of tests described in Attachment 1 should be directed either to the equipment manufacturer or the cognizant design/test agency. If you have other questions regarding this information notice, please contact the Regional Administrator of the appropriate NRC Regional Office, or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness and
Engineering Response
Office of Inspection and Enforcement

Attachments:
1. Series of Equipment Environmental Qualification Notices
2. List of Recently Issued IE Information Notices

Technical Contact: N. B. Le, IE
(301) 492-9673

Equipment Environmental Qualification Notice No. 12

Related Information Notices

1. Information Notices relating to environmental qualification of equipment that have been published since IN 82-52, which included Equipment Environmental Qualification Notices No. 1 through No. 11, are:

   IN 83-40 Need To Environmentally Qualify Epoxy Grouts and Sealers

   IN 83-45 Environmental Qualification Test of General Electric Company "CR-2940" Position Selector Control Switch

2. The Equipment Environmental Qualification Notices issued to date (No. 1 through No. 24) will be updated, if required, in the next periodical information notice.

Equipment Environmental Qualification Notice No. 13

Test Summary Report No. 1

Equipment: Anaconda flexible conduit

Test Facility: Wyle Laboratories

Cognizant Design and/or Test Agency: Washington Public Power Supply System (WPPSS)

Failed Component: Anaconda flexible conduit

Type of Test: Environmental qualification testing at LOCA conditions

Description of Failure(s):

Polyethylene copolymer jacket of the flexible conduit degraded while exposed to the following LOCA conditions:

340°F and 45 psig for 3 hr.
320°F and 45 psig for 3 hr.
250°F and 25 psig for 18 hr.

Failure Mode:

Polyethylene copolymer jacket melted while exposed to LOCA temperature, pressure, and steam environment.

Possible Corrective Action Considerations:

WPPSS is reviewing the failure, and will report corrective action at a later date.

Generic Implications:

Melted polyethylene jacket would allow steam and spray solution to enter otherwise watertight junction boxes, and equipment housings. This in turn will cause degradation and short-circuiting of other safety-related components or electrical circuits inside the junction boxes and equipment housings.

Equipment Environmental Qualification Notice No. 14

Test Summary Report No. 1

Equipment: Rockwell International post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI)
Energy Systems Group (ESG)

Failed Component: ITT Barton pressure transducer, 4-20 ma, part number D4R-29098

Type of Test: Typical IEEE 323 environmental qualification

Description of Failure(s):

ITT Barton pressure transducers, will not withstand radiation dose of 1×10^7 rads (and may not operate satisfactorily after radiation exposures in excess of 10^4 rads TID because of gradual drifting of readings). These pressure transducers are installed in several delivered recombiners and are used to measure recombiner inlet gas flow, total flow, and inlet gas pressure.

Failure Mode:

Pressure transducer did not operate satisfactorily after exposure to radiation dosage in excess of 10^4 rads.

Possible Corrective Action Considerations:

ESG recommended the following corrective action to each affected customer, as a short-term corrective measure for operating plants: Temporary wiring modification in the control cabinet should be made to allow operation of the recombiner in the manual flow control mode, until the transducer can be replaced with a qualified transducer.

Generic Implications:

Rockwell International reported that recombiners with ITT-Barton transducers were shipped to the following plants: Limerick, LaSalle County, Nine Mile Point 2, E. I. Hatch 2, and Fermi 2.

Equipment Environmental Qualification Notice No. 15

Test Summary Report No. 1

Equipment: Rockwell International Post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI) Energy Systems Group (ESG)

Failed Component: Microswitch, DPST toggle switch, rated 15A, 125-250 VAC, 1/2 hp, 125 VAC, part number 12TSI-2.

Type of Test: Typical IEEE 323 environmental qualification

Description of Failure(s):

Both poles of the microswitch failed to close when the switch was activated during the post-LOCA environmental cycling test. It was noted that during the post-seismic functional test, the switch operation appeared to be weak and not as firm as normal. Disassembly and examination of the switch revealed that the plastic slides were missing and there was considerable debris around the switch's operating handle.

Failure Mode:

ESG concluded that the plastic slides were sufficiently degraded during the thermal and radiation aging that when the switch was exposed to the vibration and shock from the seismic test, the plastic slides broke into many pieces (when the switch was operated on with snap action).

Possible Corrective Action Considerations:

Engineering Field Bulletins have been issued by RI informing users to install jumpers bypassing the switch. RI has eliminated the switch in its recent designs.

Generic Implications:

The switch is used in the control circuits of the reversing motor starters for the hydrogen recombiner motor-operated valves. Rockwell International reported that recombiners with microswitches were shipped to the following plants: Fermi 2, Shoreham 1, Limerick, E. I. Hatch 2, Nine Mile Point 2, and LaSalle County.

Equipment Environmental Qualification Notice No. 15

Test Summary Report No. 1

Equipment: Rockwell International post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI)
Energy Systems Group (ESG)

Failed Component: Square D disconnect switch, three-pole nonfusible unit, 30A, 15 hp at 480 VAC or 20 hp at 600 VAC, part number 9422-RC-1.

Type of Test: Typical IEEE 323 environmental qualification

Description of Failure(s):

Square D disconnect switch, failed to operate following the irradiation phase of the EQ test. ESG concluded that the applied radiation dose of 1.1×10^7 rads has sufficiently degraded the plastic component to initiate the failure.

Failure Mode:

The failure is attributed to a plastic component breaking when the switch was mechanically operated.

Possible Corrective Action Considerations:

ESG recommended that the users of the RI recombiner eliminate the disconnect switch from the circuit.

Generic Implications:

The switch is used as the main disconnect for the recombiner skid 480-VAC 3-phase, power bus to the inlet gas, recirculating gas and water valve motor circuits. The switch could fail open during a seismic event following the LOCA, thus disrupting power to the recombiner skid. Rockwell International reported that recombiners with Square D disconnect switches were shipped to the following plants: Limerick, LaSalle County, Nine Mile Point 2, E. I. Hatch 2, and Fermi 2.

Equipment Environmental Qualification Notice No. 17

Test Summary Report No. 1

Equipment: Rockwell International post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI) Energy Systems Group (ESG)

Failed Component: Timetrol SCR power controller, thr[illegible] 1066Z series, part number 2053C-125K

Type of Test: Typical IEEE 323 environmental qualification

Description of Failure(s):

The Timetrol SCR power controller failed the functional testing after the unit was irradiated. The SCR controller controls the recombiner's heaters.

Failure Mode:

The SCR power controller failed after being exposed to radiation dose of 1.62×10^6 rads.

Possible Corrective Action Considerations:

ESG recommended that the users of the RI recombiner replace the Timetrol SCR unit with a unit compatible with ESG current design.

Generic Implications:

Rockwell International reported that recombiners with Timetrol SCR power controllers were shipped to the following plants: Fermi 2, Limerick, Beaver Valley, North Anna, Millstone 3, Three Mile Island 2, and Zion.

Equipment Environmental Qualification Notice No. 18

Test Summary Report No. 1

Equipment Rockwell International post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI) Energy Systems Group (ESG)

Failed Component: Automatic Timing & Control (ATC) time delay relay, 120 VAC, 50-60 Hz, part number 319B006QIC, IEEE No. 21A.

Type of Test: Typical IEEE 323 environmental qualification

Description of Failure(s):

Automatic Timing and Control (ATC) time delay relay failed the final performance test following the post-LOCA environmental cycling test. The relay failed when the unit was subjected to an elevated temperature test with high humidity (RH greater than 90 percent). The relay coil of the timer was energized and after one cycle, it failed to hold closed.

Failure Mode:

The failure was attributed to an oxide film on the relay contacts which occurred during the EQ testing. (NOTE: The timer is not a sealed unit)

Possible Corrective Action Considerations:

Because the ATC time delay is not a sealed unit, ESG recommended the existing timer for BWR Mark 3 and PWR plants be replaced with an EQ qualified, sealed timer. However for BWR Mark 1 and 2, ESG stated that the replacement of the existing timer is not necessary, because the loss of this timer on the BWR Mark 1 and 2 hydrogen recombiners should not result in a loss of safety function.

Generic Implications:

Rockwell International reported that recombiners with ATC time delay relays were shipped to the following plants: Beaver Valley, North Anna, Surry, Millstone 3, Three Mile Island 2, Palo Verde, Hartsville, Phipps Bend, Clinton, Byron, Cherokee, Perkins, Oconee, Marble Hill, Braidwood, and Zion.

Equipment Environmental Qualification Notice No. 19

Test Summary Report No. 1

Equipment: Rockwell International post-LOCA hydrogen recombiner

Test Facility: Wyle-Norco

Cognizant Design and/or Test Agency: Rockwell International (RI) Energy Systems Group (ESG)

Component: ITE Gould circuit breaker, 3-pole, 600 VAC, fully enclosed, thermal magnetic, part number EF3-B015, IEEE No. 79

Test: Typical IEEE 323 environmental qualification

ion of Failure(s):

E Gould circuit breaker opened when temperature reached 165°F (150°F design plus 15°F margin) and also opened at post-test ambient temperature during the post-LOCA phase of the EQ testing. After this failure, a new EF3-B015 circuit breaker was tested under the same test condition as in the previous test and it was observed that the circuit breaker tripped without carrying any load current when the test temperature reached 169°F at high relative humidity. Additional testing was performed and the result was that the same breaker would trip at 145°F. To establish the temperature at which the EF3-B015 circuit breaker would carry the required load without tripping, ESG performed several tests and reported that the circuit breaker did not trip at 140°F while carrying 3.5 amperes, and the breaker remained closed after several days at 130°F with 3.5-ampere load current. ESG also reported that discussion with ITE Gould personnel indicates this potential failure problem may be common to several lines of ITE thermal magnetic circuit breakers at temperatures greater than 120°F. These circuit breakers include type EF2-B, EF3-B, HE3-B, and JL3-B, all of which have been used in recombiner application; however, no testing has yet been performed on these models.

Failure Mode:

The ITE Gould circuit breaker opened when ambient temperature reached 165°F.

Possible Corrective Action Considerations:

The ITE Gould circuit breakers are used to provide 480 VAC to various RI recombiner components and control devices. The failure of these breakers would prevent the operation and cause the recombiner to shut down. ESG has informed the users of RI recombiners about this potential defect and recommended that as an interim fix, the circuit breaker be bypassed and as an alternative fix, the power cabinet be relocated. Supplemental cooling to reduce the cabinet temperature to less than 100°F for PWR recombiners is also recommended.

Equipment Environmental Qualification Notice No. 19 (Cont.)

Test Summary Report No. 1 (Cont.)

Possible Corrective Action Considerations (Cont.):

ESG states that replacement of the "thermal magnetic" breakers with "magnetic only" breakers is being considered as the permanent fix on the RI recombiners.

Generic Implications:

Rockwell International reported that recombiners with ITE Gould circuit breakers were shipped to the following facilities: Fermi 2, Shoreham 1, E. I. Hatch 2, Nine Mile Point 2, LaSalle County, Hartsville, Phipps Bend, Clinton, Byron, and Palo Verde.

Equipment Environmental Qualification Notice No. 20

Test Summary Report No. 1

Equipment: ITT-Barton electronic transmitters

Test Facility: ITT

Cognizant Design and/or Test Agency: ITT-Barton

Failed Component: Barton transmitter, suppressed zero model 763

Type of Test: Performance test

Description of Potential Failu (s):

The potential defect exhibits itself as a negative shift in output during initial exposure to operating pressure. The amount of the shift is a function of the process pressure and the calibrated span of the instrument.

Failure Mode:

ITT-Barton reported that further testing of model 763 pressure transmitter has identified the specific cause for the reported defect. It is due to the combined creep in:

1. The link wire between the pressure bourdon tube and strain-sensing beam, and

2. The material used to attach the link wire to the bourdon tube and to the strain-sensing beam.

Long-term testing on the 763 pressure transmitter is continuing.

Possible Corrective Action Considerations:

The failure is under review by ITT. Users of this instrument are expected to take necessary administrative control and inform their plant staff of the potential defect.

Generic Implications:

ITT-Barton reported that all suppressed zero model 763 transmitters shipped prior to June 30, 1983, may not perform to Barton's specifications.

Equipment Environmental Qualification Notice No. 21

Test Summary Report No. 1

Equipment: Barksdale pressure switches

Test Facility: Sandia National Laboratories (SNL)

Cognizant Design and/or Test Agency: NRC/RES

Failed Components: Barksdale pressure switches models B2T and D2H

Type of Test: Typical IEEE 323 LOCA simulation

Description of Failure(s):

One unit of model B2T shorted out at 5 minutes into the test but recovered on the cooling side of the first temperature transient, then failed again on the next LOCA profile temperature transient. The second unit of model B2T failed at 6 hours, 43 minutes into the test. Two units of model D2H shorted out at 10 minutes into the test but recovered on the cooling side of the first LOCA profile temperature transient, then failed again on the next LOCA profile temperature. These pressure switches were unaged and were exposed to a typical IEEE 323 LOCA simulation profile in the SNL facility test chamber which used saturated steam to achieve a LOCA temperature profile. An evaluation test was also performed on new pressure switches of the same models previously tested. It was observed that at the 10 psig plateau one Barksdale model D2H was shorted out. All other D2H and B2T models failed at the 40 psig plateau.

Failure Mode:

Pressure switches experienced "blown" seals that allowed water to accumulate in the switch housing, and as a result, exhibited electrical shorts across the microswitches.

Possible Corrective Action Considerations:

NRC is planning no further testing on these models at this time. Barksdale pressure switch models B2T and D2H should be replaced with qualified models if their application is for installation in areas where high pressure and steam/spray environment are anticipated.

Generic Implications:

The failure of the pressure switch could prevent proper operation of safety-related functions and could lead to inappropriate operator actions.

Equipment Environmental Qualification Notice No. 22

Test Summary Report No. 1

Equipment: Static-O-Ring pressure switches

Test Facility: Sandia National Laboratories (SNL)

Cognizant Design and/or Test Agency: NRC/RES

Failed Components: Static-O-Ring (SOR) model 5N and 12N

Type of Test: Typical IEEE 323 LOCA simulation

Description of Failure(s):

One unit of model 5N failed in the first 2 minutes, and other units of model 5N and model 12N failed approximately 5 minutes into the LOCA transient. Pressure switch housings were penetrated resulting in diaphragm rupture and steam blowing out through pressure lines. These pressure switches were unaged and were exposed to a typical IEEE 323 LOCA simulation profile in the SNL facility test chamber which used saturated steam to achieve a LOCA temperature profile. An evaluation test was also performed on new pressure switches of same models, and similar failures were experienced.

Failure Mode:

Upon disassembly, the pressure switches were found to have "blown in" gaskets and the elastrometric diaphragms were found to have ruptured.

Possible Corrective Action Considerations:

SOR pressure switch models 5N and 12N (and other models of similar design) should be replaced with qualified models if their installation is in areas where high pressure and steam/spray environment are anticipated. NRC is planning no further testing on these models at the present time.

Generic Implications:

These pressure switches are used in various safety-related applications. The systems affected could be the residual heat removal (RHR), nuclear boiler (NB), reactor core isolation cooling (RCIC), and reactor protection system (RPS). The failure of these pressure switches could adversely affect the safe operation of the plant.

Equipment Environmental Qualification Notice No. 23

Test Summary Report No. 1

Equipment: ITT-Barton electronic transmitters

Test Facility: ITT

Cognizant Design and/or Test Agency: ITT-Barton

Failed Component: Barton transmitter models M-763 and M-764

Type of Test: Performance test

Description of Failure(s):

The potential defect exhibits itself in the form of thermal non-repeatability, and results in performance outside Barton's specification. Transmitters manufactured after September 1, 1981 are suspect. ITT-Barton reported that a total of 474 Barton transmitters (model M-763) and 628 Barton transmitters (model M-764) have been delivered to domestic locations.

Failure Mode:

1 A leakage current path through the shafts of the zero and span potentiometers to the mounting bracket was detected. This resulted in non-repeatability at 320°F, according to ITT-Barton.

Possible Corrective Action Considerations:

ITT-Barton recommends the installation of a fiberglass insulator between the potentiometer shafts and the mounting brackets. Field repair kits are being prepared by Barton for onsite correction of the leakage current problem. ITT-Barton has notified all of its customers of this potential defect.

Generic Implications:

The thermal non-repeatability of these transmitters could impede the intended function of a safety system that is actuated or controlled by these transmitters. In addition, due to the thermal non-repeatability the instrument indication in the control room could become inaccurate. Tests conducted on corrected transmitters have revealed that the test methodology used on all units identified as potentially defective contributed to the out-of-specification performance at normal temperature. Barton has provided its customers with the worst case performance data for their evaluation. Barton also states that some changes to the test methods were made, and presently the specified performance is being achieved in production.

Equipment Environmental Qualification Notice No. 24

GENERIC QA PROBLEMS IN THE EQUIPMENT QUALIFICATION INDUSTRY

The NRC has performed several QA programmatic reviews of companies and laboratory facilities that perform tests to environmentally qualify safety-related electrical equipment used in the harsh environment. This effort was started in late August 1982 to assess the testing organization's establishment and implementation of the QA program based on 10 CFR Part 50, Appendix B criteria.

During these QA programmatic reviews, several generic problems related to the testing industry were identified. They are as follows:

1. Organizations (architect engineers (AEs), manufacturers, suppliers, and testing laboratories) have failed to include 10 CFR Part 50, Appendix B or equivalent quality requirements and 10 CFR Part 21 requirements in purchase orders or contracts. In one case a testing laboratory was performing environmental qualification testing without knowing it was testing safety-related electrical equipment, because the purchase order or contract did not include quality control or 10 CFR Part 21 requirements. NRC inspectors have identified many similar failures to pass on requirements between involved organizations. The NRC considers this to be a serious problem because testing without knowing the criticality of the equipment being tested would likely result in less caution during testing and little concern relative to deviation or defect reporting in accordance with 10 CFR Part 21.

2. A few testing facilities failed to establish and implement a QA program. Other testing facilities failed to properly implement 10 CFR Part 21.

3. The majority of the manufacturing test facilities and testing organizations inspected had QA programs that were deficient or nonconforming. The seriousness of these nonconformances ranged from no QA program to less serious nonconformances such as failing to follow procedures.

4. The generic problems described above have in most cases, been caused by the test organization's customers (manufacturer, AE, or licensee). That is, these upline customers have failed to pass on NRC requirements in procurement documents, failing to ensure that QA programs were established and properly implemented. It is apparent that the audit process (licensee to lowest subcontract tier) is not working properly. A recent example was identified in which three utilities and five AEs had performed audits of the vendor's QA program for manufacturing but had not audited the EQ test program to ensure proper establishment/implementation of the QA program.

Equipment Environmental Qualification Notice No. 24

Equipment: Limitorque valve operators

Reference: Construction Deficiency Report - Part 50.55(e)
Facilities: 50-329, -330

Description of Circumstances:

The Bechtel Associates Professional Corporation, A/E for Consumers Power Company's Midland Plant Units 1 and 2, has recently reported to the NRC the following deficiencies related to the Limitorque valve operators at the Midland plants:

1. The use of underrated terminal blocks in Limitorque operators
2. The use of terminal blocks without proper environmental qualification in Limitorque operators
3. Additional concerns regarding qualification of various Limitorque operator components

These concerns are detailed below.

A. Underrated Terminal Blocks

While replacing a damaged terminal block on a Limitorque operator, Bechtel determined that some of the terminal blocks used for the termination of the leads from the 460-volt motor were rated less than 460-volt. These Limitorque operators, when used on safety-related valves, must function on an emergency core cooling actuation signal (ECCAS). In addition to being a personnel safety hazard, the potential exists for short circuit/flashover, which could render the valves inoperative.

B. Environmental Qualification

During random inspection for underrated terminal blocks, it was discovered that, in some cases, terminal blocks were used from manufacturers not covered by existing qualification reports.

Limitorque provided the following information on environmental qualification of terminal blocks in its July 31, 1981, letter to Bechtel.

> The Buchanan 0524 has been qualified by analysis. To supplement the qualification by analysis, Limitorque is currently running a type test on the Buchanan 0524 terminal block. The Buchanan 0824 terminal blocks are not qualified and must be replaced.

Equipment Environmental Qualification Notice No. ~~25~~ (Cont.)

?4

Description of Circumstances (Cont.):

Some of the Limitorque operators having Buchanan 0824 type terminal blocks have been used on safety-related valves located inside containment. These operators must function on an ECCAS. The potential exists for a terminal block to fail during its intended service life because of aging and radiation effects, which would render the valve inoperable and prevent proper operation of the safety-related system.

C. Additional Qualification Concerns

During the month of June 1982, a random inspection was made of safety-related Limitorque valve operators supplied through various valve manufacturers and installed inside the reactor building. This inspection resulted in various potential concerns regarding qualification of these Limitorque operators. These concerns are:

1. The motor nameplate ambient temperature rating on various motors installed on some Limitorque operators is 40°C. Limitorque has verbally stated that the Class B insulation motors rated for a 40°C ambient temperature have not undergone qualification testing in accordance with IEEE Std 382-1972 for the specified normal, accident, and postaccident environment. Class H insulation motors are rated for 50°C ambient temperature, but the qualification testing in accordance with IEEE Std 382-1972 for these motors is presently unknown.

2. No identification was evident on certain materials internal to the Limitorque operators (e.g., wiring, insulation, etc.). It is not presently known whether these types are qualified for the service conditions.

3. Various orientations of installed operators were observed. It is not presently known whether the operators are qualified for all installed orientations.

4. Drain plugs on operators were observed to be both in place and removed. Orientation of the operators did not always result in the drain holes being at the lowest point of the installed operator. It is not presently known whether the existence of the drain plug or the orientation of the drain hole is essential to proper operation of the operator or is in conformance with the qualification tests for the operator.

5. Various Limitorque operator limit switch gear frames were observed to be made of a white metal. It is not presently known whether these gear frames are qualified for the service conditions.

Equipment Environmental Qualification Notice No. ~~25~~ (Cont.)
24

Description of Circumstances: (Cont.)

6. Information obtained from purchase order files and qualification files does not agree with the installed components.

7. It is presently not known whether space heaters are qualified or required to be qualified.

8. Various O-rings are located throughout the actuator. It is not presently known whether these components are qualified for the service conditions.

9. Unidentifiable terminal blocks (nonpower lead connectors inside the operators) were observed in other Limitorque operators. It is not presently known whether these components are qualified for the service conditions.

Attachment 2
IN 83-72
October 28, 1983

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
83-71	Defects in Load-Bearing Welds on Lifting Devices for Vessel Head and Internals	10/27/82	All nuclear power facilities holding an OL or CP
83-70	Vibration-Induced Valve Failures	10/27/83	All nuclear power facilities holding an OL or CP
83-69	Improperly Installed Fire Dampers at Nuclear Power Plants	10/21/83	All nuclear power facilities holding an OL or CP
83-68	Respirator User Warning - Defective Self-Contained Breathing Apparatus Air Cylinders	10/11/83	All nuclear power facilities holding an OL or CP; research and test reactors, fuel cycle licensees; Priority 1 material licensees
83-67	Emergency-Use Respirator Material Defect Causes Production of Noxious	10/11/83	All nuclear power facilities holding an OL or CP; research and test reactors, fuel cycle licensees; Priority 1 material licensees
83-66	Fatility at Argentine Critical Facility	10/7/83	All nuclear power facilities holding an OL or CP; non-power reactor , critical facility and fuel cycle licensees
83-65	Surveillance of Flow in RTD Bypass Loops Used in Westinghouse Plants	10/07/83	All Westinghouse facilities holding an OL or CP
83-64	Lead Shielding Attached to Safety-Related Systems Without 10 CFR 50.59 Evaluations	09/29/83	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

DOCUMENT DIVIDER

INSERTED BY DOCUMENT CONTROL SECTION!

-45
50-348/364-CIVP
2/19/92

LIS OI

Staff Exh. 45

DOCKETED
USNRC

'92 MAR 13 P12:09

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AUG 29 1983

'83 SEP 19 P2:33

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USNRC

'92 MAR 13 P12:09

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

MEMORANDUM FOR: Harold R. Denton, Director, NRR
Robert B. Minogue, Director, RES
Richard C. DeYoung, Director, IE
Thomas E. Murley, Regional Administrator, Region I
James P. O'Reilly, Regional Administrator, Region II
James G. Keppler, Regional Administrator, Region III
John T. Collins, Regional Administrator, Region IV
John B. Martin, Regional Administrator, Region V

FROM: C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

SUBJECT: CASE STUDY REPORT - OPERATING EXPERIENCE RELATED TO MOISTURE INTRUSION IN ENVIRONMENTALLY QUALIFIED ELECTRICAL EQUIPMENT AT COMMERCIAL POWER PLANTS

Several occurrences of safety-related equipment failures resulting from moisture intrusion have been found in the licensee event reports (LERs). Primarily involved are electrical components located in high humidity/high temperature areas of the reactor building. Alerted by these events, AEOD has performed a case study on moisture intrusion in environmentally qualified electrical equipment and our preliminary report on the subject is enclosed.

In accordance with our "peer review" process prior to the finalization and distribution of our case study reports, we are providing NRR, IE,RES.INPO, AND NSAC with a copy of a preliminary report for review and comment, particularly regarding accuracy and completeness of the technical details. Changes to the findings, conclusions, and recommendations will be considered only if the underlying information concerning the details of plant design or systems operation is in error. Therefore, comments are being solicited on the technical accuracy of the report. The findings, conclusions, and recommendations are provided for your information in order that you may understand the significance AEOD places on this concern and therefore obtain a more complete picture of the total report. We would welcome comments either informally by phone or formally by memo.

Since we wish to finalize and issue the report shortly, we ask that any comments be received by us within 30 days from receipt of the preliminary copy. Should your office require some additional time beyond that point, please let us know, otherwise it will be assumed that there are no comments.

8309130470 830829
PDR MISC
8309130470 PDR

OFFICE						
SURNAME						
DATE						

NRC FORM 318C (2-82) NRCM 0240

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NUCLEAR REGULATORY COMMISSION

Docket No. 50-348/364-CivP Official Exh. No. 45

In the matter of Alabama Power Company

Staff ______ IDENTIFIED 12:21 p.m. 2/19/92

Applicant ______ RECEIVED 12:23 p.m. 2/19/92

Intervenor ______ REJECTED ______

Cont'g Off'r ______ DATE 2/19/92

Contractor ______ Witness ______

Other ______

Reporter L Gileep

AUG 2 5 1983

We are also placing a copy of the preliminary report in the Public Document Room. If you have any questions regarding this matter, please contact Medhat El-Zeftawy of my staff. Mr. El-Zeftawy can be reached at 301/492-4434.

/S/

C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As stated

cc: w/enclosure
R. J. Mattson, NRR
D. G. Eisenhut, NRR
R. H. Vollmer, NRR
T. Speis, NRR
H. L. Thompson, NRR
S. D. Mackay, NRR
J. M. Taylor, IE
E. L. Jordan, IE
S. A. Schwartz, IE
R. M. Bernero, RES
C. E. Johnson, RES
IE Training Center

Distribution:
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AEOD CF
M. El-Zeftawy, ROAB
S. Rubin, ROAB
K. Seyfrit, C/ROAB
T. Ippolito, DD/AEOD
C. Heltemes, D/AEOD
ROAB R/F

OFFICE	ROAB	ROAB	C/ROAB	AEOD	AEOD	
SURNAME	MEl-Zeftawy:bg	SRubin	KSeyfrit	TIppolito	CHeltemes	
DATE	7/14/83	7/ 4/83	7/14/83	8/25/83	8/ /83	

AUG 29 1983

Mr. David Rossin
Director
Nuclear Safety Analysis Center
3412 Hillview Avenue
P.O. Box 10412
Palo Alto, California 94303

Dear Mr. Rossin:

Subject: Case Study Report - Operating Experience Related to Moisture Intrusion in Environmentally Qualified Electrical Equipment at Commercial Power Plants

Several occurrences of safety-related equipment failures resulting from moisture intrusion have been found in the licensee event reports (LERs). Primarily involved are electrical components located in high humidity/high temperature areas of the reactor building. Alerted by these events, AEOD has performed a case study on moisture intrusion in environmentally qualified electrical equipment and our preliminary report on the subject is enclosed.

The purpose of this letter is to provide you with the opportunity to review the report, particularly with regard to its completeness and accuracy, prior to the issuance of the AEOD final report. Changes to the findings, conclusions, and recommendations will be considered only if the underlying information concerning the details of plant design or systems operation is in error. Therefore, comments are being solicited on the technical accuracy of the report. The findings, conclusions, and recommendations are provided for your information in order that you understand the significance AEOD places on this concern and therefore obtain a more complete picture of the total report.

We would welcome your comments either informally by phone or formally by letter. Since we wish to finalize and issue the report shortly, we ask that any comments you may wish to make be brought to our attention within 30 days from receipt of this letter.

As you may know, AEOD reports do not represent an official NRC position or the position of the responsible NRC program office. Our reports are one input to an ongoing review and evaluation process, and any recommendation contained in our final report will be considered and perhaps modified or eliminated by the responsible NRC office.

OFFICE							
SURNAME							
DATE							

AUG 29 1983

A copy of the preliminary report and this letter are being placed in the Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

If you have any questions regarding this matter, please feel free to contact Mr. Medhat El-Zeftawy of my staff. Mr. El-Zeftawy can be reached at 301/492-4434.

Sincerely,

C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

Enclosure:
As stated

Distribution:
PDR ✓
AEOD RF
AEOD CF
M. El-Zeftawy, ROAB
S. Rubin, ROAB
K. Seyfrit, C/ROAB
T. Ippolito, DD/AEOD
C. Heltemes, D/AEOD

OFFICE	ROAB	ROAB	C/ROAB	AEOD	AEOD		
SURNAME	MEl-Zeftawy	bg SRubin	KSeyfrit	TIppolito	CHeltemes		
DATE	7/14/83	7/14/83	7/14/83	8/25/83	7/ /83		

AUG 2 9 1983

Mr. Stephen L. Rosen, Director
Analysis and Engineering Division
Institute of Nuclear Power Operations
1100 Circle 75 Parkway
Suite 1500
Atlanta, GA 30339

Dear Mr. Rosen:

SUBJECT: CASE STUDY REPORT - OPERATING EXPERIENCE RELATED TO MOISTURE INTRUSION IN ENVIRONMENTALLY QUALIFIED ELECTRICAL EQUIPMENT AT COMMERCIAL POWER PLANTS

Several occurrences of safety-related equipment failures resulting from moisture intrusion have been found in the licensee event reports (LERs). Primarily involved are electrical components located in high humidity/high temperature areas of the reactor building. Alerted by these events, AEOD has performed a case study on moisture intrusion in environmentally qualified electrical equipment and our preliminary report on the subject is enclosed.

The purpose of this letter is to provide you with the opportunity to review the report, particularly with regard to its completeness and accuracy, prior to the issuance of the AEOD final report. Changes to the findings, conclusions, and recommendations will be considered only if the underlying information concerning the details of plant design or systems operation is in error. Therefore, comments are being solicited on the technical accuracy of the report. The findings, conclusions, and recommendations are provided for your information in order that you understand the significance AEOD places on this concern and therefore obtain a more complete picture of the total report.

We would welcome your comments either informally by phone or formally by letter. Since we wish to finalize and issue the report shortly, we ask that any comments you may wish to make be brought to our attention within 30 days from receipt of this letter.

As you may know AEOD reports do not represent an official NRC position or the position of the responsible NRC program office. Our reports are one input to an ongoing review and evaluation process, and any recommendation contained in our final report will be considered and perhaps modified or eliminated by the responsible NRC office.

OFFICE						
SURNAME						
DATE						

AUG 29 1983

A copy of the preliminary report and this letter are being placed in the Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

If you have any questions regarding this matter, please feel free to contact Mr. Medhat El-Zeftawy of my staff. Mr. El-Zeftawy can be reached at 301/492-4434.

Sincerely,

/S/

C. J. Heltemes, Jr., Director
Office for Analysis and Evaluation
of Operational Data

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PRELIMINARY DRAFT

OPERATING EXPERIENCE RELATED TO MOISTURE INTRUSION IN ENVIRONMENTALLY QUALIFIED ELECTRICAL EQUIPMENT AT COMMERCIAL POWER REACTORS

Case Study Report
Reactor Operations Analysis Branch
Office for Analysis and Evaluation
of Operational Data

AUGUST 1983

Prepared by: Medhat El-Zeftawy

NOTE: This report documents the preliminary results of an ongoing study by the Office for Analysis and Evaluation of Operational Data with regard to a particular operational situation. This report is issued for review and comment as part of the "peer review" process used for AEOD case studies. Since the study is ongoing, the content, findings, and recommendations are preliminary and may not represent the final position of AEOD, the responsible program office or the Nuclear Regulatory Commission.

Table of Content

EXECUTIVE SUMMARY

Nuclear power plant equipment important to safety must be able to function throughout its installed life. General Design Criteria 1 and 4 specify that safety-related electrical equipment in nuclear facilities must be capable of performing its safety-related function under environmental conditions associated with all normal, abnormal, and accident plant operation.

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 is for plants originally reviewed in accordance with IEEE 323-1971 and the second is for plants reviewed in accordance with IEEE 323-1974. On January 14, 1980, the NRC issued IE Bulletin 79-01B which included the DOR Guidelines, and NUREG-0588 concerning the environmental qualification of electrical equipment. In order to ensure compliance with the criteria, the NRC staff required all licensees of operating reactors to submit a re-evaluation of the qualification of safety-related electrical equipment which may be exposed to a harsh environment. On June 1982, an environmental qualification of safety-related electric equipment for nuclear power plants rule (50.49) was provided to the NRC commission. However, the licensees of nuclear power plants with operating licenses which were issued prior to this rule are not required to qualify their existing and operable safety-related electrical equipment according to the requirements of this rule.

Numerous occurrences of safety-related equipment failures resulting from moisture intrusion were reported to the NRC after the issuance of IE Bulletin 79-01B.*
The equipment includes, but is not limited to, electrical wiring termination

boxes, junction boxes, and pressure switch instrumentation. The majority of the failures reported have involved BWRs and have been located outside of primary containment in the reactor building basement.

The cause analysis for these operating events indicated two major contributing factors:

(1) Loss of environmental protection boundary provided, e.g. maintenance.

(2) Inadequate protection provided for moisture sources involved, e.g. unsealed conduit.

Analysis indicated that the two modes of component failure are shorting (or grounding) and corrosion. The corrective action which has been taken for the first mode of failure (shorting) was cleaning and drying the equipment. For the second mode of failure (corrosion), the corrective action was a complete replacement of the components involved.

Based on this analysis, some recommendations are provided which could be used to reduce the frequency of safety-related electrical equipment failures resulting from moisture intrusion.

Introduction

The manufacturers and users of electrical equipment in nuclear power application are required to provide assurance that the equipment will meet or exceed its performance requirements throughout its installed life. Equipment that is used to perform a necessary safety function must be capable of maintaining functional operability under all service conditions postulated to occur during the installed life for the time it is required to operate. This requirement, as stated in General Design Criteria 1, 2, 4 and 23 of Appendix A and Sections III and XI of Appendix B to 10 CFR Part 50, is applicable to equipment located inside as well as outside containment.

Numerous occurrences of safety-related equipment failures resulting from moisture intrusion have been reported to the NRC. Primarily involved are electrical components located in high humidity/high temperature areas of the reactor building outside primary containment.

Intensive independent NRC staff reviews covering the importance of equipment qualification have been occurring. The results of these studies have included the issuance of IE Bulletin 79-01B concerning the environmental qualification of electrical equipment; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"; proposed Regulatory Guide 1.89, Rev. 1, "Environmental Qualification of Electrical Equipment Important to Safety for Light-Water-Cooled Nuclear Power Plants"; and a new rule in 10 CFR Part 50, "Environmental Qualification of Safety-Related Electrical Equipment for Nuclear Power Plants."

As part of the operating license review for each plant, the NRC staff evaluates the applicant's equipment qualification program by reviewing the qualification documentation on selected safety-related equipment. The objective of this review is to provide reasonable assurance that the equipment can perform its intended function in the most limiting environment in which it is expected to function.

To provide guidance to be used by the NRC staff for use in the ongoing licensing reviews, a number of positions have been developed on selected areas of the qualification issue. These positions provide guidance on the establishment of service conditions, methods for qualifying equipment, and other related matters. For example, the effects of aging, sequential versus simultaneous testing, and the "as-installed" configuration of the equipment testing with its interface connections have been treated.

This report documents the occurrences of safety-related equipment failures resulting from moisture intrusion in operating light water reactors. The equipment includes, but is not limited to, electrical wiring termination boxes, junction boxes, and pressure switch instrumentation. Also, supplied in this report is the analysis of these collected operational events with the findings and recommendations emphasizing the need for all holders of operating licenses or construction permits to recognize the potential for degradation of environmentally qualified equipment due to improper maintenance or improper use.

Description of Equipment

The components that have failed due to moisture intrusion, wetting caused by water leakage, and corrosion are listed in Table 1. The manufacturer and the component code are also listed. The complete description of each event is provided in Appendix A.

Table 1 - Operational Events

Plant	Equipment Description	Manufacturer	Component Number/Code	Deficiency
Brunswick 2	Pressure switch	Barksdale	E11-PS-N020A & E11-PS-N020C	Corrosion due to moisture
Brunswick 2	Pressure switch	Barksdale	E51-PS-N012A & E51-PS-N012C	Corrosion due to moisture
Brunswick 2	Pressure switch	Barksdale	E11-'S-N016D	Corrosion due to moisture
Brunswick 2	Pressure switch	Barksdale	E!1-PS-N020C	Corrosion due to moisture
Brunswick 2	Level switch	Not listed	E41-LSL-N002	Corrosion
Brunswick 2	Pressure switch	Barksdale	E11-PS-N016D & E11-PS-N020D	Corrosion
Brunswick 2	Temperature switch	Fenwall	E41-TS-3314	Moisture intrusion
Brunswick 2	Neutron source-range monitor	General Electric Co.	SRM (C&D)	Moisture intrusion
E. I. Hatch	Pressure switch	Barksdale	E41-PS-N012D	Corrosion
Oyster Creek	Logic-channel switch	Not listed	RE17D	Wetting
McGuire	Fire detection-data gathering panel	Not listed	DGP, No. 1	Corrosion
Duane Arnold	Pressure switch	Barksdale Valve Co.	PS-8404C	Corrosion
E. I. Hatch	Pressure switch	Barksdale	E41-N001D	Corrosion
Millstone 1	Motor housing	Porter Perless Motors	Not listed	Corrosion
Pilgrim	Electrical contacts in pump trip logic	Limitorque Corp.	MO-1001-7C	Corrosion
Dresden	HPCI valve motor	Limitorque Corp.	210683A2, Model B102953	Moisture
E. I. Hatch	Pressure switch	Barksdale	Instru.	Corrosion
Peach Bottom	Delta press. switch	Dwyer	DPS 0400-16	Corrosion
Brunswick 2	Valve disc	Not listed	Valvex-2	Moisture

Plant	Equipment Description	Manufacturer	Component Number/Code	Deficiency
Pilgrim	HPCI circuitry	Terry Steam Turbine Co.	HTEXCH	Wetting
E. I. Hatch	Inst. switch	Yarway	(Level) Inst. Switch 2	Fouled contact
E. I. Hatch	Pressure switch	Barksdale	2E41-N017B	Comp. failure
Millstone	Breakers	Not listed	MCC-101AB-2	Water intrusion
Brunswick 2	Switch module	SCAM Instrument Corp.	86PTGR-D186	Electronic failure
LaSalle	Radiation detector	General Atomic Co.	2D18-K751A	Rain water
Fitzpatrick	Motor	Porter, H.K.G., Inc.	10-MOV-38A	Water intrusion
Brunswick 2	Monitor transformer	Nuclear Measurements Inc.	F3M3-1AX	Water intrusion
Dresden 3	Fire system	Not listed	HPCI/Fire Sys. Init. Alarm	Humidity/vapor
Prairie Island	Motor leads	Not listed	Wiring/Elecon.	Wetting
Browns Ferry 2	Temp. switch	Fenwal	Model 17002-40 TS-1-29A	Wetting
Browns Ferry 3	Valve/solenoid assembly	Valcor	FSV76-65, Model No. V52630-529-1	Moisture
Quad-Cities	Valve operator	Limitorque Corp.	MO-1-1402-25B	Water leak
Surry 1	Flow transmitter	Fischer & Porter Co.	FT-1495	Water intrusion
Surry 1	Flow transmitter	Fischer & Porter Co.	FT-1494	Water intrusion
Zion Generating Station	Pressure transmitter	Fischer & Porter Co.	2PT-514	Condensation
Surry 2	Defective Coil	ASCO	Coil for TV-MS-201C	Moisture intrusion

Plant	Equipment Description	Manufacturer	Component Number/Code	Deficiency
Sequoyah 1	Junction box	Not listed	Junction Box # 2270	Electrical short
Fitzpatrick	Motor contactor	General Electric Co.	MOV-39B Motor Contactor	Water intrusion

Analysis and Evaluation of Operating Experience

Several moisture intrusion-related equipment failures have occurred. The equipment involved includes, but is not limited to, electrical wiring termination boxes, junction boxes, and pressure switch instrumentation. The majority of the failures reported have involved BWRs and have been located outside of primary containment in the reactor building basement HPCI, LPCI, and RHR/RCIC pump rooms. Appendix A details a number of these failures, as described in Licensee Event Reports (LERs).

For example, at the Brunswick plants, the affected pump rooms normally experience an operational environment of about 95°F and 90% relative humidity. Due to system leakage and sump overflow problems, these rooms may also experience some water accumulation on their floors. This water can have a temperature of up to 130°F. Events reported indicate that failures occur frequently in equipment that has been environmentally qualified. LERs 81-108/03L and 81-139/03T (Appendix A) describe moisture intrusion-related contact corrosion failures of pressure switches associated with the RHR and RCIC systems. These switches were located in the -17 foot elevation pump rooms and were purchased to withstand an environment of 290°F, 16.2 psia, 100% relative humidity and 1.0×10^7 rads. Thus, these switches should have been able to perform correctly in the environment where they were located. Careful investigation and analysis revealed that the following causes for such equipment problems are most likely:

1. Moisture entering into unsealed conduit connections at cable trays. Cable trays are located throughout the reactor building. The moisture enters unsealed conduit and subsequently condenses within the conduit and uses the conduit as a piped pathway to connecting equipment such as termination boxes, switches and instrumentation;

2. Moisture in the local environment entering the equipment at moisture resistant but not moisture tight conduit connections; and

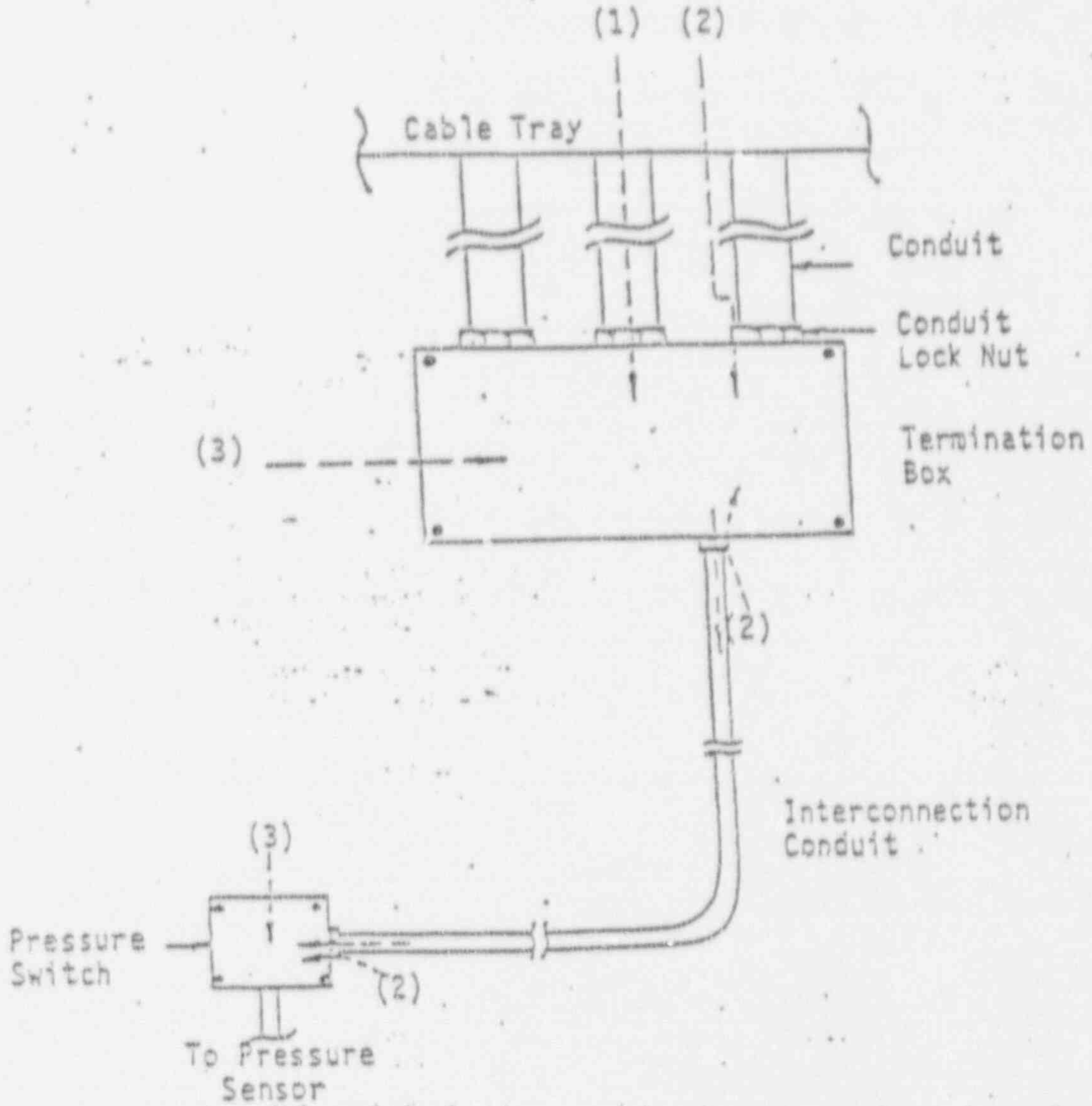
3. Moisture intrusion at improperly sealed equipment enclosure covers.

In items 2 and 3, moisture may condense and enter the connecting conduit, where, as in item 1, other equipment may be affected. (See Figure 1 for a diagram of moisture intrusion pathways.) The cause analysis for these operating events indicated two major contributing factors;

- Loss of environmental protection boundary provided, e.g., as a result of maintenance.
- Inadequate protection boundary provided for moisture sources involved, e.g., unsealed conduit, or other pathways for the entry of moisture into the equipment.

Another contributing factor is the elevational aspect where downward water flow and collection becomes a possible problem. This involves water intrusion into electrical boxes from moisture entering an unsealed conduit located above and leading to such devices. This would apply to a conduit connected to and rising above such devices which are open at some point at a higher elevation, and which may be subject to steam condensation and/or water spray environments at those higher elevations.

Analysis indicates that two modes of failure for these components are shorting (or grounding) due to wetting and water leak, and corrosion. The corrective action taken for the first mode of failure (shorting) was to clean and dry the equipment. For the second mode of failure (corrosion), the corrective action was a complete replacement of the components involved. Also, there were some incidents where the components have failed due to improper re-assembly following maintenance or surveillance activities.



Pathway Notation

(1) Cable Tray/Conduit/Equipment
(2) Environment/Connectors/Equipme
(3) Environment/Enclosure Covers/E

Figure (1) - Steam Electric Plant
Representative Moisture Intrusion Pathways

Findings

1. Corrosion failures of electrical contacts in safety-related switches caused by moisture collection within the switch housing have been a recurring problem at nuclear power plants.

2. Safety-related equipment failures caused by water accumulation within electrical equipment enclosures (e.g., switch housings, terminal boxes, junction boxes, cabinets) have occurred as a result of water/moisture conveyed within connected conduits and/or entering externally through conduit-to-enclosure air gaps.

3. Failures of safety-related electrical equipment caused by corrosion and/or water collection within protective enclosures have occurred predominantly in local environments involving high humidity, steam, water leaks or sprays.

4. The more recently reported failures, caused by moisture effects of safety-related electrical devices located outside primary containment structures have occurred mostly at BWR facilities.

5. Failures have involved devices installed in local environment, for which environmental qualification information sheets would indicate they are qualified for service.

6. Failures of safety-related electrical devices provided with environmental protection as a result of moisture intrusion often can be attributed to improper re-assembly of enclosures following maintenance or surveillance activities.

7. In situations where cabling ends in open conduits at elevated locations within the reactor building, a potential exists for the conduit to collect

moisture at equipment connected to the conduit at lower elevations if the elevated open end is located in a water vapor (or steam) condensation environment or subject to water sprays or leaks.

8. Operating experience shows that failures of safety-related electrical devices, where they have occurred, can often be traced to gaps in the "as-installed" composite assembly of individual components.

Conclusions

Nuclear power plants with operating licenses issued prior to the environmental qualification rule (50.49) are not required to qualify their existing safety-related electrical equipment in accordance with the requirements of this rule. However, an equipment qualification program based on IE Bulletin 79-01B which includes the DOR guidelines and NUREG-0588 recommendations is being used to address the environmental qualification of electrical equipment in harsh environment for operating plants. Based on the analysis and evaluation of the operational events for operating (licensed) plants listed in Appendix A, it is concluded that despite their existing environmental qualification, safety-related equipment fails due to moisture intrusion from various sources. The pathways for moisture entering the various pieces of equipment are through unsealed conduit or openings associated with terminal boxes, junction boxes, and housings, etc. These pathways do exist as a result of inadequate installation or improper or incomplete maintenance activities.

Recommendations

As a result of the findings described in this report, AEOD recommends that the IE staff:

A. Issue a Bulletin or Information notice as appropriate to initiate an evaluation of the equipment installation procedures against the drawings and specifications supplied with the equipment compared to the actual installation in the operating plants.

B. Take appropriate actions to revise the inspection program to include areas needed to ensure:

1. Control of equipment environment during normal plant operation.

2. That maintenance programs include adequate controls to ensure restoration of vapor barriers, gaskets and seals to the environmentally qualified condition following maintenance or testing. For example:

   a. Provide waterproof sealing from the outside for all electrical conduit-to-junction boxes and conduit-to-terminal box connection points for safety-related equipment located in areas of the reactor building actually or potentially subject to high temperature steam or water impingement.

   b. Seal from the inside all electrical conduit-to-junction box or conduit-to-terminal box connection points for safety-related electrical equipment where such boxes are located in areas of the reactor building which may be subject to water intrusion from moisture piped within unsealed conduits.

   c. Provide drain holes in the bottom of electrical boxes. The existence (or lack) of box drain holes should be determined by inspection.

3. Improved quality control of surveillance and/or maintenance activities which involve disassembly/reassembly of components required for

environmental protection of safety-related electrical equipment.

4. Training of the affected personnel in areas which cover electronics and instrumentation and their associated installation techniques and problems.

5. Adequate administrative controls to ensure that equipment which is environmentally qualified is identified prior to maintenance, and that appropriate instructions are included to verify that the equipment is properly protected against moisture intrusion upon reassembly.

References:

1. LER 81-108/03L - Brunswick Steam Electric Plant, Unit 2.

2. LER 81-139/03L - Brunswick Steam Electric Plant, Unit 2.

3. NRC, Memo from Medhat El-Zeftawy to Carlyle Michelson regarding site visit to Brunswick Steam Electric Plant, July 6, 1982.

4. NRC, Memo from D. B. Vassallo, Chief (Operating Reactors Branch #2, Division of Licensing) to E. E. Utley of Carolina Power & Light Co. regarding the Safety Evaluation for Environmental Qualification of Safety-Related Electrical Equipment, December 20, 1982.

5. NRC, IE Circular 80-10, "Failure to Maintain Environmental Qualification of Equipment," April 29, 1980.

6. NRC, Proposed Rule - 10 CFR Part 50.49, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," 1983.

7. NRC, Proposed Revision 1 to Reg. Guide 1.89, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," February 1982.

8. NRC, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979.

9. NRC, IE Bulletin No. 79-01B, "Environmental Qualification of Class 1E Equipment," January 14, 1980.

APPENDIX A

Abstract of Licensee Event Reports* Related to Moisture Intrusion

1. Brunswick Steam Electric Plant, Unit 2

   Event date: 10/18/81
   LER Number: 81-108/03L

   Event Description: During plant operation with no evolutions in progress, the "Core Spray or RHR Pumps Running" annunciator was received. This event occurred again on 10/20/81. Immediately following each event a check of plant operating parameters found no abnormalities and no core spray or RHR pumps running.

   Cause Description: These events resulted from the respective actuation of "A" and "C" RHR pump discharge ADS initiation logic "A" permissive pressure switches 2-E11-PS-N020A on 1/18/81 and N020C on 10/20/81, both due to corrosion of each switches' internals resulting from moisture accumulation in the instruments.

   Corrective Action: A replacement switch was installed in place of N020A and was calibrated; N020C was cleaned and calibrated.

2. Brunswick Steam Electric Plant, Unit 2

   Event date: 12/20/81
   LER Number: 81-139/03L

   Event Description: During the performance of the RCIC system turbine exhaust diaphragm high pressure channel functional test, RCIC turbine exhaust diaphragm instruments, 2-E51-PS-N012A and C-Model Number D2H-M15055, did not actuate when a test signal was applied. The RCIC system was then declared inoperable in accordance with technical specifications.

*This is an actual LERs description as received by the Office of AEOD.

Cause Description: Corrosion from moisture accumulation in the switch internals of both instruments prevented them from actuating during the periodic test.

Corrective Action: Both instruments were cleaned, resealed and the periodic test was satisfactorily completed. A work request authorization was written to seal the attached electrical conduit to these instruments in order to eliminate a suspected moisture intrusion path.

3. Brunswick Steam Electric Plant, Unit 2

   Event date: 11/02/81
   LER Number: 81-120/03L

   Event Description: During plant operation, the "Core Spray or RHR Pumps Running" annunciator was received. An immediate check of plant operating parameters found no abnormalities and no core spray or RHR pumps running.

   Cause Description: The actuation of "D" RHR pump discharge ADS initiation logic "A" permissive pressure switch, 2-E11-PS-N016D, was caused by moisture accumulation in the switch's internals.

   Corrective Action: The moisture was removed from the switch and it was calibrated and returned to service.

4. Brunswick Steam Electric Plant, Unit 2

   Event date: 10/29/81
   LER Number: 81-122/03L

   Event Description: During a reactor startup, the "Core Spray or RHR Pumps Running" annunciator was received. An immediate check of plant operating parameters found no abnormalities and no core spray or RHR pumps running.

   Cause Description: Inadequate maintenance on "C" RHR pump discharge ADS initiation logic "A" permissive pressure switch, 2-E11-PS-N020C, during corrective action to the event reported in LER Number 2-81-108, permitted remaining corrosion in the switch to actuate and cause the annunciation.

Corrective Action: The switch was cleaned, calibrated and returned to service.

5. Brunswick Steam Electric Plant, Unit 2

Event date: 10/14/81
LER Number: 81-094/03L

Event Description: During the performance of condensate storage tank (CST) low water level channel functional test, low level switch 2-E41-LSL-N002, Model Number 83842-A2, would not actuate on a simulated low level. This switch provides a signal to close the HPCI system CST suction valve 2-E41-F004, and open the HPCI system suppression pool suction valves 2-E41-F041 and F042, on low CST level. The suction for the HPCI system was then manually aligned to the suppression pool.

Cause Description: Corrosion buildup at the switch actuator arm prevented it from moving and actuating the switch.

Corrective Action: The corrosion was cleaned from the arm and it was lubricated and exercised to ensure free movement.

6. Brunswick Steam Electric Plant, Unit 2

Event date: 12/16/81
LER Number: 81-136/03L

Event Description: During plant operation, the "Core Spray or RHR Pumps Running" annunciator was received after securing 2D RHR pump from suppression pool pumping operations. On 12/17/81, while performing the channel functional test of the ECCS LPCI pump discharge pressure interlock, it was discovered that the "Core Spray of RHR Pumps Running" annunciator would not initiate as required. In each case, the ADS was declared inoperable in accordance with the technical specifications.

Cause Description: Corrosion in the switch internals of 2-E11-PS-N016D and N020D caused N016D to initiate the annunciation on 12/16/81 and prevented both instruments from initiating the annunciation during the periodic test on 12/17/81.

Corrective Action: Suitable replacement instrumentation was installed in place of both instruments.

7. Brunswick Steam Electric Plant, Unit 2

Event date: 10/07/81
LER Number: 81-109/03L

Event Description: During plant operation, HPCI system seal leak detection ambient temperature high and HPCI logic power failure annunciators were received. An immediate investigation revealed the HPCI system isolation logic power supply fuse, located in distribution panel E614, was blown. The HPCI system was then declared inoperable and was isolated.

Cause Description: Electrical shorting to ground of HPCI steamline tunnel temperature switch 2-E41-TS-3314, Model Number 170002-40, occurred due to moisture accumulation in the switch housing resulting from corrosion of the switch housing caused the event.

Corrective Action: A new temperature switch and logic power supply fuse were installed which returned the HPCI leak detection system to normal. The HPCI system was declared operable and returned to normal.

8. Brunswick Steam Electric Plant, Unit 2

Event date: 07/15/82
LER Number: 82-086/03L

Event Description: During the ongoing 1982 refueling outage, routine surveillance revealed that SRMs C and D were both indicating an erratic, upscale count rate. SRMs A and B indicated a normally expected count rate.

Cause Description: Both monitor detectors experienced moisture intrusion into their detector cable connectors due to a guide tube sealing gland leak on SRM D. SRM C also had a loss of continuity in the monitor instrument high voltage B and C type connector.

Corrective Action: The leak was repaired, the B and C type connector was replaced, both monitor cable connectors were dried and waterproofed, and the monitors were returned to service.

9. E. I. Hatch Nuclear Plant

Event date: 9/29/81
LER Number: 81-096/03L

Event Description: With the reactor at steady state operation and performing HNP-2-3309, HPCI turbine exhaust diaphragm pressure switch functional test and calibration, 2E41-N012D was found out of tolerance, the switch actuated at 12.7 psig. Tech. Spec. section 3.3.2-2 requires actuation at $\leq$ 10 psig.

Cause Description: The cause has been attributed to the switch and its mechanisms being corroded.

Corrective Action: The switch was replaced. The other switches were inspected and found to be acceptable.

10. Oyster Creek

Event date: 2/18/82
LER Number: 82-010/01T

Event Description: On January 9, 1982, the fire protection deluge system for reactor building elevation 51 ft. actuated due to smoke emanating from an overheated bearing in the cleanu system auxiliary pump motor. A DC ground resulting from wetting was traced to switch RE17D which could have rendered a logic channel for the core spray system inoperable. Also, as a result of the wetting, an AC ground fault in the RPS was traced to the position indication switches for the supply valves to the torus, resulting in a partial loss of containment isolation valve position indication.

Cause Description: Cause is attributed to an inadequate safety evaluation. Existing electrical sealing techniques judged adequate were not. Range of deluge spray to instrument rack was underestimated.

Corrective Action: A re-evaluation of the integrity of the reactor building safety-relate equipment with regard to fire protection system wetting was performed. The re-evaluation with designated corrective action will be reviewed by the plant operations review committee and approved by the director of station operations. All items determined to require sealing or drip shield protection will be protected prior to plant startup from the current shutdown.

11. McGuire Nuclear Station, Unit 1

Event date: 07/01/82
LER Number: 82-060/03L

Event Description: While in Mode 5, investigation of fire detection system (EFA) trouble alarms which could not be cleared on July 1, 6, and 13, resulted in EFA zones 63 and 64 (RHR pump room) being declared inoperable on each day. This violates Tech. Specs.

Cause Description: These were the result of water leaking into Data Gathering Panel (DGP) No. 1 causing the failure of several components due to corrosion (water from condensation on the Nuclear Service Water (RN) piping).

Corrective Action: Failed components were replaced, and subsequent efforts to prevent water corrosion included drilling holes in the bottom of DGP-1, sealing panel door and cable penetration, and covering DGP-1 with plastic. The RN piping will be insulated to prevent condensation.

12. Duane Arnold Energy Center

Event date: 04/07/82
LER Number: 82-026/03L

Event Description: During cold shutdown while performing surveillance testing of the main steam isolation valve-leakage control system (MSIV-LCS), main steam line pressure switch PS-8404C failed, rendering the MSIV-LCS on the "C" main steam line inoperable. Following manual system initiation, PS-8404C reisolates "C" steam line LCS if steam line pressure has not decayed to less than 5 psig in one minute. System operability required by Tech. Spec. 3.7.E

Cause Description: Failure of PS-8404C was due to slight corrosion on its terminal strip which resulted in a poor electrical connection.

Corrective Action: The pressure switch and terminal strip were cleaned and lubricated to prevent further corrosion. The pressure switch was functionally tested satisfactory.

13. E. I. Hatch Nuclear Plant

Event date: 7/26/80
LER Number: 80-114/03L

Event Description: On 7/26/80, while taking Hatch Unit 2 to cold shutdown, the HPCI system failed to isolate at greater than or equal to 100 psig (Tech. Spec. Table 3.3.2-2). HPCI was manually isolated at 89.3 psig. The "A" logic auto isolated at 87 psig. The "B" logic auto isolated at 80 psig.

Cause Description: "B" isolation logic pressure switch 2E41-N001D, Barksdale B2T-M1255, was corroded, affecting the instrument setpoint.

Corrective Action: The instrument was replaced. "A" isolation logic pressure instruments were in calibration. Investigation of control room pressure indication has been initiated.

14. Millstone Nuclear Power Station, Unit 1

Event date: 12/3/81
LER Number: 81-040-03L

Event Description: On December 3, 1981, at 1815 hours, while performing containment isolation valve operability demonstration, 1-CU-3 failed to close on an isolation signal. In addition 1-CU-3 did not shut when given a "close" signal from the control switch. Tech. Spec. 3.7.D requires that during reactor power operating conditions, cleanup demineralizer isolation valve, 1-CU-3, be operable.

Cause Description: Investigation revealed that the motor housing was corroded, causing a brush to freeze into its holder, preventing contact with the commutator. This caused an open circuit as well as an arc between the brush and the commutator which damaged the commutator bars. Additionally the motor bearings were corroded. This corrosion resulted from a steam leak in the area that entered the motor through an opening where a plug was missing.

15. Pilgrim Nuclear Power Station

Event date: 12/21/81
LER Number: 81-064/03L

Event Description: During a refueling outage, the "C" RHR pump was observed to be running with no suction flow path. The pump was immediately tripped, the suction valves opened.

Cause Description: This event occurred because electrical contacts in the pump trip logic were corroded to the extent that they had seized in the open position, which is indicative of an open suction path. The situation was aggravated by inadequacies in the implementation of administrative controls.

Corrective Actions: Corrective actions were implemented to repair or replace faulty components, conduct a performance test on "C" RHR pump to determine if any degradation had occurred, and review incident with operations personnel.

16. Dresden Nuclear Power Station

Event date: 7/12/82
LER Number: 82-030/01T

Event Description: During normal operation while performing DOS 2300-1, HPCI motor operated valve operability test, steam supply isolation valve 2301-4 failed to close. Redundant isolation valve 2301-5 was closed in accordance with Tech. Specs. The system was previously declared inoperable due to another unrelated problem.

Cause Description: The cause of this event was due to a valve packing leak. Moisture through the valve packing entered the valve operator and shorted out the motor.

19. Pilgrim Nuclear Power Station

Event date: 08/13/82
LER Number: 82-024/01T

Event Description: During recovery from a scram, with HPCI being used for pressure control, HPCI tripped after five minutes running due to high reactor water level. After a pump restart, attempts to bring pump past idle speed were unsuccessful. Eleven manual SRV actuations were therefore necessary to control pressure.

Cause Description: HPCI Gland Seal Condenser gasket failure caused wetting of the HPCI circuitry.

Corrective Action: The control circuits were dried and calibrated. Gasket repair was accomplished. Four quick-starts under pressure tested both repairs. Modification options for making controls more tolerant of harsh environment are being assessed.

20. E. I. Hatch Nuclear Plant

Event date: 07/30/82
LER Number: 82-065/03L

Event Description: While Unit 1 was operating at normal full load, the Reactor Vessel Level Instrument Switch No. 2 was found inoperative during routine surveillance testing. Tech. Specs. requires both ADS actuation channels to be operable. The redundant ADS permissive channel was operable. In accordance with Tech. Specs. 3.2-4, Unit 1 was placed in a 24-hour LCO.

Cause Description: The cause of this event was a fouled contact on the No. 2 switch.

Corrective Action: The switch was replaced. The reactor vessel level instrument was calibrated and satisfactorily tested. It was returned to service. Yarway level switches are functionally tested mont per Tech. Specs. 4.2-4b.

21. E. I. Hatch Nuclear Plant

    Event date: 05/31/82
    LER Number: 82-058/03L

    Event Description: With Unit 2 in steady state full power operation, the "HPCI turbine exhaust pressure Hi" and "HPCI turbine trip solenoid energized" alarms activated. The HPCI system was determined to be inoperable. Unit 2 was placed in a 14-day LCO in compliance with Tech. Specs. 3.5.1.A.

    Cause Description: The cause of this event has been attibuted to component failure. The failure was due to fouled contacts on the HPCI turbine exhaust pressure switch number 2E41-N017B.

    Corrective Action: The switch was repaired and recalibrated, and then returned to service.

22. Millstone Nuclear Power Station

    Event date: 03/18/82
    LER Number: 82-008/03L

    Event Description: At 1845 hours, breakers that operate a main steam line drain valve, an inboard torus spray stop valve and an inboard drywell spray stop valve were found to be tripped due to water damage.

    Cause Description: The breakers were damaged by water that entered the MCC after soaking through an environmental enclosure penetration seal.

Corrective Action: The breakers were repaired. Penetration seals will be modified to prevent recurrence.

23. Brunswick Steam Electric Plant, Unit 2

Event date: 03/01/82
LER Number: 82-044/03L

Event Description: During plant operation, while performing the channel functional test and calibration of the HPCI steam leak detection isolation system instrumentation, it was discovered that HPCI room ambient temperature switch 2-E41-TS-N602A would not respond to an applied test signal. HPCI system was declared inoperable. At the time of this event, the LPCI and ADS systems were operable.

Cause Description: This event was due to electronic failure of the switch module.

Corrective Action: The module Model No. 86PTGR-D186 was replaced. The periodic test was satisfactorily completed, and the HPCI system was declared operable.

24. LaSalle County Nuclear Station, Unit 1

Event date: 5/15/82
LER Number: 82-021/03L

Event Description: Control room HVAC outside air radiation monitor failed high due to moisture. This was attributed to rain water finding its way into the detector. This "hi" rad signal initiated the emergency make up train. However, the tornado damper was inadvertently closed, shutting off all suction to the make up train. Emergency make up fan was then put in the pull-to-lock control switch position to prevent fan damage from possible motor overload.

Cause Description: Outside air radiation detector 2D18-K751A failed due to rain water finding its way into its circuitry around conduit, junction boxes or metal to metal interfaces.

Corrective Action: All outside air radiation detectors are sealed with weather proof sealant at probable leakage points. A work request had been submitted to provide a permanent "shelter" above these detectors.

25. Fitzpatrick Nuclear Power Plant

Event date: 5/14/82
LER Number: 82-023/03L

Event Description: During normal operation, while conducting surveillance, containment isolation valve 10 MOV 38A was found inoperable and containment spray loop A was declared inoperable. Test of LPCI, Emergency Service Water, Emergency Diesel Generator and other containment spray components was satisfactory.

Cause Description: Failure of the motor on valve 10 MOV 38A was the cause. Disassembly of the motor revealed the presence of water which had apparently entered to motor via the conduit during an earlier unrelated maintenance activity.

Corrective Action: A replacement motor was removed from the spare parts and installed.

26. Brunswick Steam Electric Plant, Unit 2

Event date: 02/09/82
LER Number: 82-031/03L

Event Description: During plant operation, a water leak dripping on the equipment cabinet of primary containment atmospheric monitor

2-CAC-ATH-1263 came in contact with the monitor power supply transformer and resulted in a loss of power to the monitor. Following initial disc[illegible] of this event, the plant fire brigade was dispatched to the sc[illegible]e in case of fire; however, deenergizing the cabinet prevented a fire.

Cause Description: Well water from a small gasket leak on the upstream valve body flange of RHRSW header flushing valve, 2-SW-V140, dripped onto the monitor transformer causing the monitor power failure.

Corrective Action: The transformer was replaced and the monitor, Model No. F3M3-1AX, was returned to service. The leaking V140 gasket was replaced with one of greater durability and the valve was returned to service.

27. Dresden, Unit 3

Event date: 11/30/81
LER Number: 81-039/01T

Event Description: Unit startup was in progress when a HPCI room fire system initiation alarm sounded. It was determined upon investigation that fire protection deluge system had actuated. The HPCI was isolated, an unusual event declared, and a normal unit shutdown initiated.

Cause Description: Cause of the fire system initiation is believed to be a high concentration of humidity/steam vapor in the HPCI room which spuriously actuated the ionization type detector.

Corrective Action: All affected equipment including the fire system detector had been checked for operability and returned to service. Station Fire Marshall will review the HPCI room fire system for possible modifications to improve reliability.

28. Prairie Island Nuclear Generating Plant

Event date: 9/21/81
LER Number: 81-023/03L

Event Description: During surveillance testing, one cooling water header isolation valve did not close fully. Tech. Specs. 3.3.D.1.b applies.

Cause Description: Motor leads in a junction box were wet.

Corrective Action: Wiring was renewed by splicing and the valve tested satisfactorily. Similar junction boxes in the area were inspected.

29. Browns Ferry Nuclear Plant, Unit 2

Event date: 07/11/82
LER Number: 82-020/03L

Event Description: During normal operation while performing surveillance on the main steam tunnel high-temperature switches on Unit 2, temperature switch TS-1-29A failed to operate. Tech. Specs. requires the switch to operate at $\leq$ 200°F. Above the trip setting initiates main steam line isolation.

Cause Description: A valve was found leaking, keeping the switch junction box wet.

Corrective Action: The leak was reduced, the switch was replaced, a new junction box was sealed, and the switch was functionally tested and returned to service.

30. Quad-Cities Nuclear Power Station

Event date: 07/13/82
LER Number: 82-017/03L

Event Description: While taking the inboard core spray injection valve (1-1402-25B) out of service for inspection. the valve failed to open from the control room. The consequences were minimal because the redundant core spray loop, the LPCI mode of the RHR system, and both Diesel Generators associated with Unit 1 were operable. The outboard isolation valve was manually opened making the "B" loop of core spray operable in less than an hour.

Cause Description: This failure was caused by water from a packing leak running into the motor operator. The water caused damage to the rotor and brake of the motor operator.

Corrective Action: The immediate corrective action was to close the outboard valve and manually open the inboard valve. The motor operator will be repaired when parts become available.

31. Surry Power Station, Unit 1

Event date: 9/12/81
LER Number: 81-048/03L

Event Description: While the unit was being returned to service following a rod control problem, "C" steam flow transmitter FT-1495, was found to indicate low. A containment entry was made to inspect the instrument and verify correct valve lineup, at which time it failed high.

Cause Description: Water entered the electronic section of the transmitter via an electrical conduit attached to the transmitter and caused the electronics to fail.

Corrective Action: The transmitter was replaced with one of a different design.

32. Surry Power Station, Unit 1

Event date: 9/20/81
LER Number: 81-052/03L

Event Description: While the unit was being removed from service to repair feedwater leaks inside containment, the steam flow indicator for "C" steam generator, FI-1494, would not indicate less than 2.5 x 10 lbs/hr. The failed channel's bistable was placed in the tripped mode and the redundant channel was verified to be indicating normally.

Cause Description: Water entered the electronic section of the transmitter via an electrical conduit attached to the transmitter and caused the electronics to fail.

Corrective Actions: The transmitter was replaced.

33. Zion Generating Station

Event date: 3/06/82
LER Number: 82-005/03L

Event Description: During cold shutdown, 2A steam generator pressure transmitter 2PT-514 failed high at 4.690 VDC. This condition was non-conservative for the high steam line differential pressure safety injection.

Cause Description: The failure was caused by condensation entry into the transmitter through the signal conduit connection due to a nearby steam leak.

Corrective Action: The steam leak was repaired. The transmitter was dried, cleaned and recalibrated.

34. Surry Power Station, Unit 2

Event date: 5/01/81
LER Number: 81-027/03L

Event Description: With the unit at full power, the performance of PT-8.5 (monthly CLS logic testing) revealed that TV-MS-201C control circuit would not respond to a Train "B" actuation signal.

Cause Description: The cause of this event was shorting of Train B coil due to moisture accumulation in an adjacent junction box.

Corrective Action: The immediate corrective action was to verify the redundant train to be operable. The defective coil was replaced and tested. The other two coils associated with Train B were replaced as a precautionary measure.

35. Sequoyah Nuclear Plant, Unit 1

Event date: 8/28/81
LER Number: 81-113/03L

Event Description: With Unit 1 in Mode 1, the containment isolation and ice bed systems were declared inoperable due to an electrical short in the wiring to ice condenser system isolation valve C1-FCV-61-194.

Cause Description: A junction box was saturated with liquid from the glycol expansion tank located above the junction box, causing the isolation valve to fail close. This resulted in a rise of the ice bed temperature.

Corrective Action: The junction box was cleaned and new terminal strips installed. The junction box was sealed closed.

36. J. A. Fitzpatrick Nuclear Power Plant

Event date: 1/29/81
LER Number: 81-015/03L

Event Description: During normal operation, primary containment isolation valve MOV-39B motor contractor tripped due to water entering the motor control center from an electrical conduit. The isolation valve and other valves in the same line remained closed.

Cause Description: A construction deficiency is considered the cause of water entering the motor control center via a conduit from an underground man hole outside of the building.

Corrective Action: The conduit was sealed and the motor contractor dried out. The valve was tested and returned to service.