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OFFICE OF NUCLEAR REGULATORY RESEARCH

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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.

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 50.49(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.

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

*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.

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

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 ACCIDENT

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.

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.

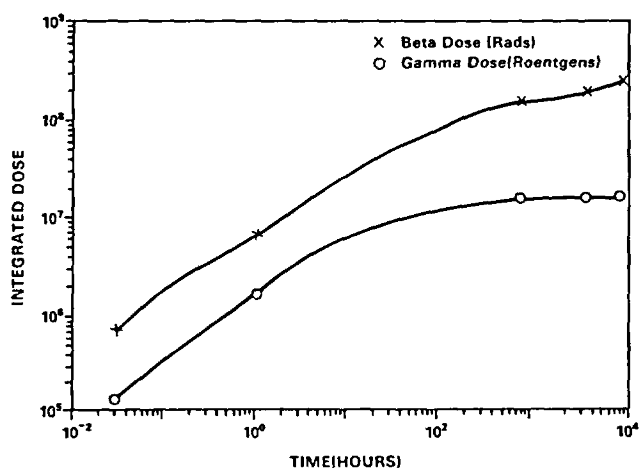

Figure D-1 Sample Airborne and Plateout Doses for a Dose Point on the Containment Centerline

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

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.

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	1.14E+7
192	1.68E+6	8.29E+6	2.86E+6	1.28E+7
298	1.85E+6	8.76E+6	3.19E+6	1.38E+7
394	1.95E+6	8.85E+6	3.41E+6	1.42E+7
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	9.25E+6	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

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
0.38	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

1. A. K. Postma, R. R. Sherry, and P. Tam, "Technological Bases for Models of Spray Washout and Airborne Contaminants in Containment Vessels," U.S. Nuclear Regulatory Commission, NUREG/CR-0009, November 1978.*

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.

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 an 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.