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Contact: S. K. Aggarwal (301) 443-5946

PROPOSED REVISION 1 TO REGULATORY GUIDE 1.89

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT 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," and Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50; and in § 50.55a.

Specific requirements for electric equipment important to safety are contained in a proposed amendment to 10 CFR Part 50. Section 50.49, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," would require that each type of electric equipment be qualified for its application and specified performance and would provide requirements for establishing qualification methods and environmental qualification parameters.

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to qualification of electric equipment for service in nuclear power plants to ensure that the equipment can perform its safety function.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by APR 23 1982

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

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 (IEEE), and subsequently 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.

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

It is essential that equipment be qualified to meet its performance requirements under the environmental and operating conditions in which it will be required to function and for the length of time its function is required. 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. The specific environment for which individual equipment must be qualified will depend on the installed location, the conditions under which it is required to function, and the length of time (with margin) it is required to operate.

Electric equipment to be qualified in a nuclear radiation environment should be exposed to a fluence that simulates the conservatively calculated total dose

[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.

and dose rate that the equipment should withstand prior to completion of its intended function. Dose rate, spectrum, and particle type should be simulated as closely as practicable unless it can be shown by analysis that damage is not significantly dependent on dose rate, spectrum, or particle type.

Regulatory Position C.1 calls for the qualification of additional equipment whose malfunction or failure resulting from an accident condition could negate the safety function of essential systems and equipment.

Item (12) of Regulatory Position C.4.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.

Equipment qualification is predicated on the assumption that qualification testing adequately simulated the environment and service conditions throughout the installed life of the equipment. Where routine maintenance is essential to maintaining equipment in the conditions simulated by the qualification test (e.g., cleanness), it is important to establish an adequate program of preventive maintenance and quality assurance that includes minimizing dust accumulation that could degrade the ability of the equipment to function properly.

C. REGULATORY POSITION

The procedures described by IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,"[1] dated February 28, 1974, are acceptable to the NRC staff for qualifying electric equipment for service in nuclear power plants to ensure that the equipment can perform its safety functions subject to the following:

1. Proposed § 50.49, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," of 10 CFR Part 50 would require that essential

electric systems and equipment be qualified to perform their intended functions. Typical essential equipment and functions that mitigate accidents are listed in Appendix A to this guide. Additional equipment should also be qualified for accident conditions if its malfunction or failure due to such conditions will negate the safety function of essential systems and equipment. For example, additional equipment that should be considered for qualification are the associated circuits defined in Regulatory Guide 1.75, "Physical Independence of Electric Systems."

2. Reference is made in Sections 2, 6.3.2, and 6.3.5 of IEEE Std 323-1974 to IEEE Std 344-1971, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations." The specific applicability or acceptability of IEEE Std 344 is covered in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants." However, the testing should be performed on a single prototype in the sequence indicated in Section 6 of IEEE Std 323-1974.

3. Section 5, "Principles of Qualification," of IEEE Std 323-1974 presents various methods for qualifying equipment, including analysis. The NRC generally will not accept analysis in lieu of testing. Experience has shown that qualification of equipment without test data may not be adequate to demonstrate functional operability during design basis event conditions. Analysis may be acceptable if testing the equipment is impractical because of size limitations or the state of the art. Analysis in combination with partial type-test data that adequately supports the analytical assumptions and conclusions is acceptable if the purchase order for this equipment was executed prior to May 23, 1980.

4. Section 6.2 of IEEE Std 323-1974 requires equipment specifications to define performance and environmental requirements. In defining the requirements called for in item (7) of Section 6.2, the following should be used:

a. <u>Temperature and Pressure Conditions Inside Containment for Loss-of-Coolant Accident (LOCA) and Main Steam Line Break (MSLB)</u>

(1) The following methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are acceptable to the NRC staff:

(a) Methods for calculating mass and energy release rates for LOCAs and MSLBs are summarized in Appendix B 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.

(b) For pressurized water reactors (PWRs) with a dry containment, calculate LOCA or MSLB containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Section 6.2.1.1. of NUREG-0800, "Standard Review Plan"[2] (SRP).

(c) For PWRs with an ice condenser containment, calculate LOCA or MSLB containment environment using LOTIC or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.B.

(d) For boiling water reactors (BWRs) with a Mark I, II, or III containment, calculate LOCA or MSLB environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2.1.1.C.

(2) 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 Chemicals

Guidelines for the chemical spray are provided in SRP Section 6.5.2, paragraph II, item (e). Effects of the spray should also be considered for plants that use demineralized water as spray solution.

c. Radiation Conditions Inside and Outside Containment

The radiation environment for qualification of equipment should be based on the radiation environment normally expected over the installed life of the equipment plus that associated with the most severe accident during or following which the equipment must remain functional. It should be assumed that the accident-related environmental conditions occur at the most critical point of degradation during the installed life of the equipment, which may be at the end of its installed life. Methods acceptable to the NRC staff for

[2]Copies may be obtained at current prices from the National Technical Information Service, Springfield, Virginia 22161.

establishing radiation limits for the qualification of equipment for BWRs and PWRs are provided in the sample calculations in Appendix C and the following:

(1) The source term to be used in determining the radiation environment for equipment qualification associated with a LOCA should consider the most limiting environment associated with the following:

(a) For a LOCA in which the primary system cannot be restored, 100% of the core activity inventory of noble gases and 50% of the core activity inventory of halogens should be assumed to be instantaneously released from the fuel to the containment. Fifty percent of the core activity inventory of cesium and 1% of the remaining fission product solids should be assumed to be instantaneously released from the fuel to the primary coolant and carried by the coolant to the containment sump.

(b) For a LOCA in which the primary system integrity can be restored, 100% of the core activity inventory of noble gases, 50% of the core activity inventory of halogens, 50% of the core activity inventory of cesium, and 1% of the remaining fission product solids should be assumed to be instantaneously released (after an initial time delay) and circulated in the primary coolant system. This accident is not expected to produce instantaneous fuel damage. A 30-minute delay may be assumed for fission product release from the fuel. Greater delay times should be justified on the basis of system design that minimizes fission product release. No noble gases should be assumed circulating in the primary system following system depressurization.

(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 FSAR). When only fuel clad perforation is postulated, the nuclide inventory of the fuel elements breached should be calculated at the end of core life, assuming continuous full-power operation. The fuel rod gap inventory 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 Kr-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 instantaneously released to the primary coolant. When fuel melting is postulated, the activity inventory of the melted fuel elements should also be calculated at the end of core life assuming full-power operation.

For this case, 100% of the noble gases, 50% of the halogens, 50% of the cesium, and 1% of the remaining fission product solids in these elements should be assumed to be instantaneously released to the primary coolant.

(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, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," the source term should assume an initial release to the containment that considers the fission product release groups associated with grossly melted fuel. Acceptable assumptions for the fractional release for each group are: noble gases, 100%; I, Br, 100%; Cs, Rb, 100%; Te, 100%; Sr, Ba, 11%; Ru, 8%; and La, 1.3% (individual nuclides are listed in Table VI 3-1 of WASH-1400). The effect of natural and mechanical containment fission product removal may be considered on a best-estimate basis to determine the rate of redistribution of the various groups from the containment atmosphere to other locations.

(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 containment and auxiliary structures.

(5) The initial distribution of activity within the containment should be based on mechanistic assumptions. For example, for compartmented containments such as in some BWRs, it should be assumed that 100% of the source is initially contained in the drywell. For ice condenser containments, it should be assumed that 100% of the source is initially contained in the lower portion of the containment. The assumption of uniform distribution of activity throughout a compartmented containment at time zero may not be appropriate.

(6) Effects of the engineered safety feature systems that act to remove airborne activity and redistribute activity within containment, e.g., containment sprays and containment ventilation and filtration systems, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 and the related sections referenced in the appendices to that section.

(7) Natural deposition (i.e., plateout) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters (see Ref. 3, Appendix C). The assumption of 50% instantaneous

plateout 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 verified and quantified by analysis or experiment.

(8) The qualification dose should be the sum of the calculated doses of the potential radiation sources at the equipment location (i.e., beta and gamma). Plant-specific analysis may be used to justify any reduction in dose or dose rate due to the specific location or shielding. The qualification dose may be established by one of the following:

(a) The total qualification dose should be equivalent to the total calculated dose (beta plus gamma) at the equipment location. A source of gamma radiation only may be used for qualification testing provided analysis or tests indicate that the doses and dose rates produce damage similar to the damage that would occur under accident conditions, i.e., a combination of beta and gamma radiation, or

(b) The beta and gamma qualification doses may be determined separately and the testing may be performed using both a beta and a gamma test source.

(9) Shielded components need be qualified only to the gamma radiation dose or dose rate required provided an analysis or test shows 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 heating and secondary radiation have no deleterious effects on component performance.

(10) Coatings and coverings on electric equipment should be assumed to be exposed to both beta and gamma dose and dose rates in assessing their resistance to radiation. Plateout activity should be assumed to remain on the equipment surface unless the effects of removal mechanisms such as spray washoff or steam condensate flow can be verified and quantified by analysis or experiment.

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

(12) Equipment that may be exposed to low-level radiation doses should not generally be considered exempt from radiation qualification testing. Exemption 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.

(13) A given component may be considered to be qualified provided it can be shown that the component can be subjected, without failing, to integrated beta and gamma doses, taking into account the beta and gamma dose rates, equal to or higher than those levels resulting from an analysis that (a) is similar in nature and scope to that included in Appendix C and (b) incorporates appropriate factors pertinent to the plant design (e.g., reactor type and power level, containment size).

d. Environmental Conditions for Equipment Outside Containment

(1) Equipment that is located outside containment and that could be subjected to high-energy pipe breaks as defined in the Standard Review Plan should be qualified to the conditions resulting from the accident for the duration required. The techniques to calculate the environmental conditions should employ a plant-specific model based on good engineering judgment.

(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.

(3) Equipment not served by environmental support systems within the scope of this guide or served by other systems within the scope of this guide 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.

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

a. Equipment items identified in items (2) and (3) of Regulatory Position C.4.d are not required to be qualified by test if they are in a mild environment, i.e., an environment that would at no time be more severe than the environment that would occur during normal power plant operation or during anticipated operational occurrences. Design or purchase specifications that contain a description of the functional requirements and the specific environmental conditions during normal and abnormal conditions and that are supported by a certificate of compliance based on test data and analysis will generally be acceptable. A well-supported surveillance program in conjunction with a good preventive maintenance program should be provided to ensure that such equipment will function for its design life.

b. Equipment located in watertight enclosures should be qualified by testing that demonstrates the adequacy of such protection. Equipment that could be submerged should be identified and demonstrated to be qualified by testing that demonstrates seal integrity and functional operability for the duration required. Shortened test periods and analytical extrapolation should be justified.

c. Equipment located in an area where rapid pressure changes are expected should be qualified by testing that demonstrates that, under the most adverse time-dependent relative humidity conditions (superheated steam followed by saturated steam may be a limiting condition) and the most adverse postulated pressure transient for the equipment location, the equipment seals and vapor barriers will prevent moisture from penetrating into the equipment to the degree necessary to maintain equipment integrity for the length of time the equipment function is required.

d. The temperature to which equipment is being qualified by exposure to a simulated environment should be determined by temperature readings sufficiently close to the equipment to characterize its environment.

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

f. Chemical spray or demineralized water spray should be incorporated during simulated event testing at or near the maximum pressure and temperature conditions that would occur when the spray systems actuate.

g. Expected extremes in power supply voltage and frequency should be applied appropriately during simulated event testing.

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

6. In the absence of plant-specific margins, the suggested values in Section 6.3.1.5, "Margin," of IEEE Std 323-1974 may be used as a guide subject to the following:

a. Quantified margins should be applied to the design parameters discussed in Regulatory Position C.4 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 the specified plant parameters unless those conservatisms can be quantified and shown to contain sufficient margin. The margins should (1) account for uncertainties associated with the use of analytical techniques in deriving environmental parameters when best-estimate methods are used rather than conservative licensing methods, (2) account for uncertainties associated with defining satisfactory performance (e.g., when only a few units are tested), (3) account for variations in the commercial production of the equipment, and (4) account for the inaccuracies in the test equipment to ensure that the calculated parameters have been adequately enveloped.

b. Some equipment may be required by the design to perform its safety function only within a short time period into the event (i.e., less than 10 hours), and, once its function is completed, 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 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. For all other equipment (e.g., post-accident monitoring, recombiners), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std 323-1974 should be used.

7. Section 6.3.3, "Aging," of IEEE Std 323-1974 should be supplemented with the following:

a. Where synergistic effects have been identified (e.g., effects resulting from dose rates in combination with other aging effects and from different sequences of applying qualification test parameters), they should be accounted for in the qualification program.

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. Other aging methods that can be supported by tests will be evaluated on a case-by-case basis.

c. Known material phase changes and reactions should be identified to ensure that no adverse changes occur within the extrapolation limits.

d. 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.

e. Periodic surveillance testing under normal service conditions is not considered an acceptable method for ongoing qualification unless the testing includes provisions for subjecting the equipment to the limiting service and environmental conditions (specified in accordance with proposed paragraph 50.49(c) of 10 CFR Part 50).

f. Humidity effects should be included in accelerated aging unless it can be shown that the effects of relative humidity are negligible.

g. The qualified life of the equipment (or component, as applicable) and the basis for its selection should be defined and documented.

h. 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 the other methods that may reasonably be used. All assumptions should be documented.

i. An ongoing program to review surveillance and maintenance records to identify age-related degradations should be established.

j. A component maintenance and replacement schedule that includes consideration of aging characteristics of the installed components should be established.

8. Sections 6.4 and 6.5 of IEEE Std 323-1974 discuss qualification by operating experience and by analysis, respectively. The adequacy of these methods should be evaluated on the basis of the quality and detail of the information available in support of the assumptions made. Operating experience and analysis based on test data may be used where testing is precluded by the physical size of the equipment or the state of the art of testing. When the analysis method is employed because of the physical size of the equipment, tests on vital components of the equipment should be provided.

9. Components that are part of equipment qualified as an assembly (e.g., a motor starter that is part of a motor control center qualified as a whole) may be replaced with components of the same design. If components of the same design are not used for replacement, the replacement component should be designed to meet the performance requirements and should be qualified to meet the service conditions specified for the original components.

10. In addition to the requirements of Section 8, "Documentation," of IEEE Std 323-1974, documentation should address the information identified in Appendix D to this guide. A certificate of conformance by itself is not acceptable unless it is accompanied by information on the qualification program, including test data or comparable test data from equivalent equipment. A record of the qualification should be maintained in a central file to permit verification that each item of electric equipment is qualified for its application and meets its specified performance requirements when subjected to the conditions present when it must perform its safety function up to the end of its qualified 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 proposes an acceptable alternative method for complying with specified portions of the Commission's regultions, 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:

1. Plants that are not committed to either IEEE Std 323-1971 or the November 1974 issue of Regulatory Guide 1.89/IEEE Std 323-1974 and whose equipment has been tested only for a high-temperature, high-pressure, and steam environment may not need to test such equipment again to include other service conditions such as radiation and chemical sprays. The qualification of equipment for these service conditions may be established by analysis.

2. The provision that testing should be performed on a single prototype in the sequence indicated in Section 6 of IEEE Std 323-1974 will be waived for operating power plants.

3. With regard to aging considerations in equipment qualification, plants that are not committed to the November 1974 issue of Regulatory Guide 1.89/IEEE Std 323-1974 need not demonstrate a specific qualified life except in the case of equipment using materials that have been identified as being susceptible to significant degradation due to aging. Component maintenance or replacement schedules should include considerations of the specific aging characteristics of the component materials. Ongoing programs should exist at the plant to review surveillance and maintenance records to ensure that equipment exhibiting

age-related degradation will be identified and replaced as necessary. However, the valve operators and the motors should be preconditioned by aging prior to testing for those plants that are committed to Regulatory Guide 1.73, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants," which endorses IEEE Std 382-1972, and Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," which endorses IEEE Std 334-1971.

4. Replacement components or spare parts used to replace currently installed equipment or components should be qualified according to this guide unless there are sound reasons to the contrary. Unavailability of prototype equipment or the fact that the component to be used as a replacement is in stock or was purchased prior to May 23, 1980, are among the factors to be considered in weighing whether there are sound reasons to the contrary.

APPENDIX A

TYPICAL EQUIPMENT OR FUNCTIONS FOR ACCIDENT MITIGATION

Engineered Safety Feature Actuation

Reactor Protection

Containment Isolation

Steamline Isolation

Main Feedwater Shutdown and Isolation

Emergency Power

Emergency Core Cooling[1]

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

Postaccident Sampling and Monitoring[2]

Radiation Monitoring[2]

Safety-Related Display Instrumentation[2]

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[1]These systems will differ for PWRs and BWRs and for older and newer plants. In each case the system features that allow for transfer to the recirculation cooling mode and establishment of long-term cooling with boron precipitation control are to be considered as part of the system to be evaluated.

[2]More specific identification of these types of equipment can be found in the plant emergency procedures and in Tables 1 and 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," Categories 1 and 2.

APPENDIX B

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 & 2 for General Electric plants. NEDO-20533 dated June 1974 and Supplement 1 dated August 1975 (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 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.)
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 C

SAMPLE CALCULATION AND METHODOLOGY FOR RADIATION QUALIFICATION DOSE

This appendix illustrates the staff model for calculating dose rates and integrated doses for equipment qualification purposes. The doses shown in Figure B-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 here 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, which, for the design basis loss-of-coolant accident (LOCA), may well exceed one year.

The beta and gamma integrated doses presented in Tables B-1 and B-2 and Figure B-1 have been determined using models and assumptions contained in this appendix. This analysis is conservative and 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 B-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 accident at Three Mile Island Unit 2 (TMI-2), 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. Current rulemaking proceedings are reevaluating 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. While the final resolution of the source term assumptions is conditioned on the completion of these rulemaking efforts, the staff believes it is prudent to incorporate the knowledge gained of fission product behavior from the TMI-2 accident in defining source term assumptions for equipment qualification.

Based on release estimates in the Rogovin Report (Ref. 1), the staff assumptions for noble gas and iodine releases still appear to be 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 auxiliary building tanks. Comparison of the integrated dose from the TMI-2 cesium release to the previous staff assumption of "1% solids" shows that assuming "1% solids" may not result in a conservative estimate for the radiation exposure for equipment required to function for time periods exceeding thirty days. 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 include a cesium release in addition to the previously assumed "1% solids." As a result, the revised regulatory positions propose a release of 50% of the core cesium activity inventory (see Regulatory Position C.4.c items (1) and (2)). The assumed cesium release implies no substantial departure from, and is consistent with, the degraded core conditions previously implied by the assumed release of 50% of the core iodine activity. This change in assumption 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 assumption of concurrent release of cesium and iodine also is consistent with the findings of recent source term studies reported in NUREG-0772 (Ref. 2). This report also concluded that the predominant form of the iodine released during accidents is cesium iodide (CsI). Although the CsI form is not specifically addressed in this report, it is evident that either CsI or I_2 and Cs would, in the long term, be located primarily in the reactor water and the containment sump water of a PWR or the suppression pool of a BWR. The staff recognizes that the revised source terms contained in this report are interim values and that the conclusions from the report cited above, as well as further results from current research efforts in the source term area, should ultimately form the basis for any revision of source term assumptions. Any revision of the source term assumptions, such as the inclusion of additional radionuclides, would be incorporated into this guide before it is issued as an active 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 noble gas and iodine activity airborne 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 in modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 3), 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. The source terms used in the analysis assumed that 50% of the core iodines and 100% of the core noble gases were released instantaneously to the containment atmosphere and 50% of the core cesium and 1% of the remaining "solid" activity inventory were released from the core and carried with the primary coolant directly to the containment sump.

2. The containment free volume was taken as 2.52×10^6 ft^3. Of this volume, 74% or 1.86×10^6 ft^3 was assumed to be directly covered by the containment sprays. (Plants with different containment free volumes should use plant-specific values.)

3. It was assumed that 6.6×10^5 ft^3 of the containment free volume is unsprayed; this includes regions within the main containment space under the containment dome and compartments below the operating floor level.

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. 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 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 to enhance the removal of iodine.

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. 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 5.0×10^5 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.

11. A spray removal rate constant (λ) for particulate iodine concentration was calculated using the staff's SPIRT program (Ref. 3). The staff calculated 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 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. 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 is realistic in terms of specifying the time-dependent radiation environment in most areas of the containment.

14. The analysis assumed that more than one specie 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, 2.5% is associated with airborne particulate materials and 2% formed organic compounds. The remaining 95.5% remains as elemental iodine. For conservatism, this composition was assumed present at time t = 0. (These assumptions concerning the iodine form are consistent with those of 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.)

15. For all containments, no leakage from the containment building to the environment was assumed.

16. 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 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 8 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 8 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 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 on 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 5 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 semi-infinite beta dose model.

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 5 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 constituting 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 assuming 50% 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 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 rates should be calculated including an estimate of the effects of buildup.

5. Conclusion

The values given in Tables C-1 and C-2 and Figure C-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 C-1
SUMMARY TABLE OF 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+6
1220	2.19+E6	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 C-2
SUMMARY TABLE OF 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	5.48E5	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	6.54E+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.

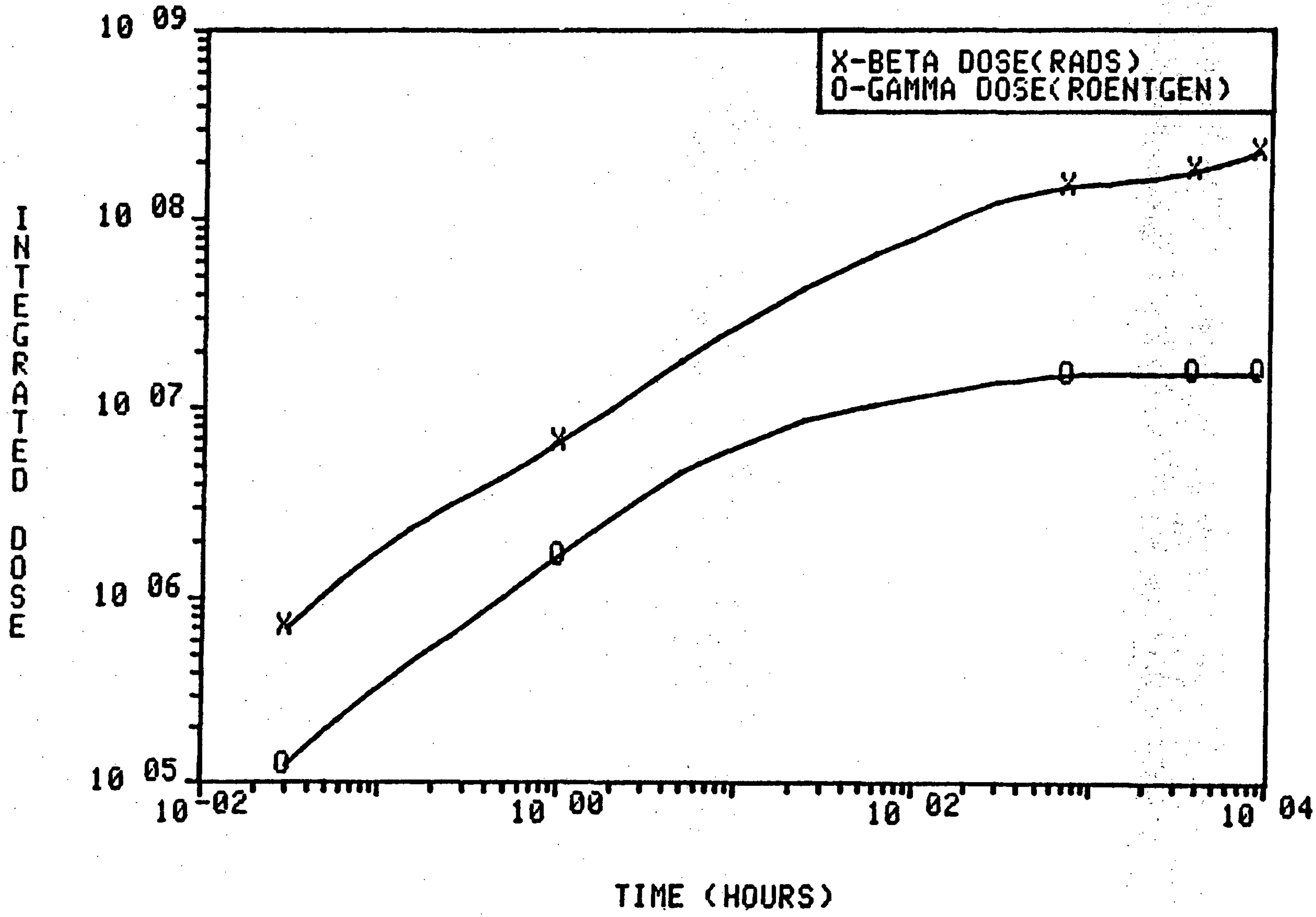

Figure C-1 Sample airborne doses for a dose point on the containment centerline

APPENDIX C

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APPENDIX D
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 and Sections III and XI of Appendix B, and § 50.49 of 10 CFR Part 50 and to the national standards mentioned in Part II, "Acceptance Criteria," of Standard Review Plan Section 3.11 (which includes IEEE Std 323), the following information on the qualification program is required for electric equipment within the scope of this guide.

1. Identify all electric equipment within the scope of this guide including the following, as applicable:
    a. Switchgear
    b. Motor control centers
    c. Valve operators
    d. Motors
    e. Logic equipment
    f. Cable
    g. Connectors
    h. Diesel generator control equipment
    i. Sensors (pressure, pressure differential, temperature, neutron, and other radiation)
    j. Limit switches
    k. Heaters
    l. Fans
    m. Control boards
    n. Instrument racks and panels
    o. Electric penetrations
    p. Splices
    q. 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

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 for which it must function to mitigate such 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 such 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 such 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 normal 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 normal service environment. This equipment would normally be located outside the reactor containment.

4. For each item of equipment in the categories of equipment listed in item 3, 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 postevent conditions.

c. The time required to fulfill its safety function when subjected to any of the extremes of the environment envelope 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. Provide the qualification test plan, test setup, test procedures, and acceptance criteria for at least one of each group of equipment in item 1 as appropriate to the category identified in item 3. 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.

6. For each category of equipment identified in item 3, 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.

7. Provide 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.

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

DRAFT VALUE/IMPACT STATEMENT

Background

Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," issued in November 1974, is being revised to reflect the current staff position on equipment qualification.

NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electric Equipment," was issued for public comment in December 1979. Subsequent to its issuance for comment, the Commissioners (see Memorandum and Order CLI-80-21 dated May 23, 1980) directed the staff to use NUREG-0588 along with a document entitled "DOR Guidelines for Evaluating Qualification of Class 1E Electrical Equipment in Operating Reactors" as requirements licensees and applicants must meet in order to satisfy the equipment qualification requirements of 10 CFR Part 50. Additionally, the Commissioners directed the staff to develop a rule for electric equipment qualification. The rule will be based principally on NUREG-0588 and the DOR guidelines. The proposed revision to Regulatory Guide 1.89 will provide guidelines for meeting the Commission's equipment qualification rule and is essentially equivalent to the staff position and guidance contained in the proposed revised version of NUREG-0588, which is based on consideration of public comments and lessons learned from TMI-2 in source term definition.

Substantive Changes and Their Value/Impact

1. Regulatory Position C.2, which provided radiological source terms for equipment qualification tests, was deleted and the following positions were added:

a. Regulatory Position C.1, which adds to the systems that should be qualified those systems that could fail in some way that would make a safety system unable to perform its function (for example, the associated circuits defined in Regulatory Guide 1.75, "Physical Independence of Electric Systems").

b. Regulatory Position C.3, which provides the staff position regarding the various qualification methods (e.g., test, operating experience, analysis, on-going qualification). Testing should be the primary method. The other methods, when used, should be supported by test data.

c. Regulatory Position C.4, which provides the staff position pertaining to establishing performance and environmental requirements for equipment qualification. Methods for establishing temperature and pressure profiles for the loss-of-coolant accident and main steam line break are provided, and radiological source terms are given.

d. Regulatory Position C.5, which provides the staff position pertaining to test procedures. Mild environment was described and a provision that testing for a mild environment is not required was added.

e. Regulatory Position C.6, which provides the staff position regarding establishing margin in testing requirements.

f. Regulatory Position C.7, which provides the staff position regarding accelerated aging of equipment as part of the testing procedure.

g. Regulatory Position C.8, which provides the staff position regarding the use of operating experience and analysis as qualification methods.

h. Regulatory Position C.9, which provides the staff position on the use of and qualification of replacement components.

i. Regulatory Position C.10, which provides the staff position on the adequacy of the documentation of equipment qualification procedures and results.

<u>Value</u> - All these positions, with the exception of Regulatory Position C.1, provide the staff's position on individual sections of IEEE Std 323-1974. This provides guidance to licensees and applicants using the standard as to what is an acceptable interpretation of the standard's requirements. These positions should enhance the licensing process.

Impact - With the possible exception of Regulatory Position C.1, the impact should be minimal since the scope has not been changed from current practice. The positions merely take established NRC provisions and relate them to appropriate sections of an endorsed voluntary consensus standard. Regulatory Position C.1 will help to ensure that a common-cause failure that results in a safety function not being performed is being addressed insofar as qualification of equipment can prevent such a failure. The impact on each individual licensee will depend on the quality of equipment currently in use or intended for use. The impact could be minimal since plant controls are a vital part of keeping the plant in operation during plant electric power generation.

2. Regulatory Position C.4.d(3), which is not part of NUREG-0588 but which provides a source term for use in the qualification of certain accident-monitoring instrumentation specified in Regulatory Guide 1.97, was added. This instrumentation is for the measurement of designated variables whose maximum value extends beyond the values predicted in the design basis accident analysis.

Value - The source term provided will standardize the radiation value for use in the qualification of the high-level instrumentation specified in Regulatory Guide 1.97 and will eliminate the necessity of determining source terms on a case-by-case basis. This will enhance the licensing process.

Impact - There is no impact. The source term of Regulatory Position C.4.d(3) merely provides an acceptable term for meeting the need expressed in Regulatory Guide 1.97 for a source term.

3. The Implementation Section was modified to be consistent with the implementation of NUREG-0588 and the DOR Guidelines.

Value - The modified implementation is consistent with current requirements as imposed by the Commission's Memorandum and Order CLI-80-21 dated May 23, 1980.

Impact - The impact should be minimal since, with the exception of Regulatory Position C.1, no new requirements are imposed. The impact of Regulatory Position C.1 on each individual licensee will depend on the quality of equipment currently in use or intended for use. The impact could be minimal since plant controls are a vital part of keeping the plant in operation during electric power generation.

119406002001 1 QPS11SXR
US NRC REGION I
R J BORES
631 PARK AVENUE
REGION I
KING OF PRUSSIA PA 19406