NUREG-0588
Rev. 1

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

Including Staff Responses to Public Comments

Resolution of Generic Technical Activity A-24

A. J. Szukiewicz, Task Manager

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

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Resolution of Generic Technical Activity A-24

Manuscript Completed: November 1980
Date Published: July 1981

A. J. Szukiewicz, Task Manager

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

ABSTRACT

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

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

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

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

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ACKNOWLEDGEMENT

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

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INTRODUCTION

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

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

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

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

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

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

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

Part

Part I

NUREG-0588

For Comment

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

December 1979

NUREG-0588
For Comment

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

Resolution of Generic Technical Activity A-24

Manuscript Completed: August 1979
Date Published: December 1979

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ABSTRACT

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

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ACKNOWLEDGMENT

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

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

INTRODUCTION

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

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

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

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

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

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

DISCUSSION

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

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

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

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

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

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

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

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

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
1. ESTABLISHMENT OF THE QUALIFICATION PARAMETERS FOR DESIGN BASIS EVENTS	1. ESTABLISHMENT OF THE QUALIFICATION PARAMETERS FOR DESIGN BASIS EVENTS
1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)	1.1 Temperature and Pressure Conditions Inside Containment - Loss-of-Coolant Accident (LOCA)
(1) The time-dependent temperature and pressure, established for the design of the containment structure and found acceptable by the staff, may be used for environmental qualification of equipment.	(1) Same as Category I.
(2) Acceptable methods for calculating and establishing the containment pressure and temperature envelopes to which equipment should be qualified are summarized below. Acceptable methods for calculating mass and energy release rates are summarized in Appendix A.	(2) Same as Category I.
Pressurized Water Reactors (PWRs)	Pressurized Water Reactors (PWRs)
Dry Containment - Calculate LOCA containment environment using CONTEMPT-LT or equivalent industry codes. Additional guidance is provided in Standard Review Plan (SRP) Section 6.2.1.1.A, NUREG-75/087.	Dry Containment - Use the same containment models as in Category I. The assumption of partial revaporization will be allowed. Other assumptions that reduce the temperature response of the containment will be evaluated on a case-by-case basis.
Ice Condenser Containment - Calculate LOCA containment environment using LOTIC or equivalent industry codes Additional guidance is provided in SRP Section 6.2 1.1.B, NUREG-75/087.	Ice Condenser Containment - Same as Category I
Boiling Water Reactors (BWRs)	Boiling Water Reactors (BWRs)
Mark I, II and III Containment - Calculate LOCA environment using methods of GESSAR Appendix 3B or equivalent industry codes. Additional guidance is provided in SRP Section 6.2 1.1.C, NUREG-75/087	Same as Category I.
(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser types of plants, the generic envelope shown in Appendix C may be used for qualification testing.	(3) Same as Category I.
(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) Same as Category I

CATEGORY I	CATEGORY II
Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
1.2 Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)	1.2 Temperature and Pressure Conditions Inside Containment - Main Steam Line Break (MSLB)
(1) The environmental parameters used for equipment qualification should be calculated with a plant-specific model reviewed and approved by the staff	(1) Where qualification has not been completed, the environmental parameters used for equipment qualification should be calculated using a plant-specific model based on the staff-approved assumptions discussed in item 1 of Appendix B.
(2) Models that are acceptable for calculating containment parameters are listed in Section 1.1(2).	(2) Other models that are acceptable for calculating containment parameters are listed in Section 1.1(2).
(3) In lieu of using the plant-specific containment temperature and pressure design profiles for BWR and ice condenser plants, the generic envelope shown in Appendix C may be used.	(3) Same as Category I.
(4) The test profiles included in Appendix A to IEEE Std. 323-1974 should not be considered an acceptable alternative in lieu of using plant-specific containment temperature and pressure design profiles unless plant-specific analysis is provided to verify the adequacy of those profiles.	(4) Same as Category I
(5) Where qualification has been completed but only LOCA conditions were considered, it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:	(5) Where qualification has been completed but only LOCA conditions were considered, then it must be demonstrated that the LOCA qualification conditions exceed or are equivalent to the maximum calculated MSLB conditions. The following technique is acceptable:
(a) Calculate the peak temperature envelope from an MSLB using a model based on the staff's approved assumptions defined in Section 1.1(2).	(a) Calculate the peak temperature from an MSLB using a model based on the staff's approved assumptions discussed in item 1 of Appendix B.
(b) Show that the peak surface temperature of the component to be qualified does not exceed the LOCA qualification temperature by the method discussed in item 2 of Appendix B.	(b) Same as Category I Section 1.2(5)(b)
(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) requalification testing be performed with appropriate margins, or (ii) qualified physical protection be provided to assure that the surface temperature will not exceed the actual qualification temperature. For plants that	(c) If the calculated surface temperature exceeds the qualification temperature, the staff requires that (i) additional justification be provided to demonstrate that the equipment can maintain its required functional operability if its surface temperature reaches the calculated value or (ii) requalification testing be performed with appropriate margins, or (iii) qualified physical protec-

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

are currently being reviewed, or will be submitted for an operating license review within six months from issue date of this report, compliance with items (i) or (ii) above may represent a substantial impact. For those plants, the staff will consider additional information submitted by the applicant to demonstrate that the equipment can maintain its functional operability if its surface temperature rises to the value calculated.

tion be provided to assure that the surface temperature will not exceed the actual qualification temperature.

1.3 Effects of Chemical Spray

Category I:

The effects of caustic spray should be addressed for the equipment qualification. The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system. If the chemical composition of the caustic spray can be affected by equipment malfunctions, the most severe caustic spray environment that results from a single failure in the spray system should be assumed. See SRP Section 6.5.2 (NUREG-75/087), paragraph II, item (e) for caustic spray solution guidelines.

Category II:

Same as Category I.

1.4 Radiation Conditions Inside and Outside Containment

Category I:

The radiation environment for qualification of equipment should be based on the normally expected radiation environment over the equipment qualified life, plus that associated with the most severe design basis accident (DBA) during or following which that equipment must remain functional. It should be assumed that the DBA related environmental conditions occur at the end of the equipment qualified life.

The sample calculations in Appendix D and the following positions provide an acceptable approach for establishing radiation limits for qualification. Additional radiation margins identified in Section 6.3.1.5 of IEEE Std. 323-1974 for qualification type testing are not required if these methods are used.

(1) The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For all other non-LOCA design

Category II:

Same as Category I.

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std. 323-1971
basis accident conditions, a source term involving an instantaneous release from the fuel to the atmosphere of 10 percent of the noble gases (except Kr-85 for which a release of 30 percent should be assumed) and 10 percent of the iodines is acceptable. (2) The calculation of the radiation environment associated with design basis accidents should take into account the time-dependent transport of released fission products within various regions of containment and auxiliary structures. (3) The initial distribution of activity within the containment should be based on a mechanistically rational assumption. Hence, for compartmented containments, such as in a BWR, a large portion of the source should be assumed to be initially contained in the drywell. The assumption of uniform distribution of activity throughout the containment at time zero is not appropriate. (4) Effects of ESF systems, such as containment sprays and containment ventilation and filtration systems, which act to remove airborne activity and redistribute activity within containment, should be calculated using the same assumptions used in the calculation of offsite dose. See SRP Section 15.6.5 (NUREG-75/087) and the related sections referenced in the Appendices to that section. (5) Natural deposition (i.e., plate-out) of airborne activity should be determined using a mechanistic model and best estimates for the model parameters. The assumption of 50 percent instantaneous plate-out of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be justified and quantified by analysis or experiment. (6) For unshielded equipment located in the containment, the gamma dose and dose rate should be equal to the dose and dose rate at the centerpoint of the containment plus the contribution from location dependent sources such as the sump water and plate-out, unless it can be shown by analyses that location and	

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

shielding of the equipment reduces the dose and dose rate.

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

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

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

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

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

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

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

CATEGORY I

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

2. QUALIFICATION METHODS

2.1 Selection of Methods

(1) Qualification methods should conform to the requirements defined in IEEE Std. 323-1974.

(2) The choice of the methods selected is largely a matter of technical judgment and availability of information that supports the conclusions reached. Experience has shown that qualification of equipment subjected to an accident environment without test data is not adequate to demonstrate functional operability. In general, the staff will not accept analysis in lieu of test data unless (a) testing of the component is impractical due to size limitations, and (b) partial type test data is provided to support the analytical assumptions and conclusions reached.

(3) The environmental qualification of equipment exposed to DBA environments should conform to the following positions. The bases should be provided for the time interval required for operability of this equipment. The operability and failure criteria should be specified and the safety margins defined.

(a) Equipment that must function in order to mitigate any accident should be qualified by test to demonstrate its operability for the time required in the environmental conditions resulting from that accident.

(b) Any equipment (safety-related or non-safety-related) that need not function in order to mitigate any accident, but that must not fail in a manner detrimental to plant safety should be qualified by test to demonstrate its capability to withstand any accident environment for the time during which it must not fail.

(c) Equipment that need not function in order to mitigate any accident and whose failure in any mode in any accident environment is not detrimental to plant safety need only be qualified for its non-accident service environment.

CATEGORY II

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

2. QUALIFICATION METHODS

2.1 Selection of Methods

(1) Qualification methods should conform to the requirements defined in IEEE Std. 323-1971.

(2) Same as Category I.

(3) Same as Category I

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

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

CATEGORY I Applicable to Equipment Qualified in Accordance with IEEE Std 323-1974	CATEGORY II Applicable to Equipment Qualified in Accordance with IEEE Std 323-1971
	(4) The staff considers that, for vital electrical equipment such as penetrations, connectors, cables, valves and motors, and transmitters located inside containment or exposed to hostile steam environments outside containment, separate effects testing for the most part is not an acceptable qualification method. The testing of such equipment should be conducted in a manner that subjects the same piece of equipment to radiation and the hostile steam environment sequentially.
2.4 Other Qualification Methods Qualification by analysis or operating experience implemented, as described in IEEE Std. 323-1974 and other ancillary standards, may be found acceptable. The adequacy of these methods will be evaluated on the basis of the quality and detail of the information submitted in support of the assumptions made and the specific function and location of the equipment. These methods are most suitable for equipment where testing is precluded by physical size of the equipment being qualified It is required that, when these methods are employed, some partial type tests on vital components of the equipment be provided in support of these methods	2.4 Other Qualification Methods Same as Category I (except that IEEE Std 323-1971 and ancillary standards endorsed at the time the CP SER was issued may be used)
3 MARGINS	3. MARGINS
(1) Quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing These margins should be applied in addition to any margins (conservatism) applied during the derivation of the specified plant parameters	(1) Same as Category I
(2) In lieu of other proposed margins that may be found acceptable, the suggested values indicated in IEEE Std. 323-1974, Section 6 3 1 5, should be used as a guide (Note exceptions stated in Section 1 4)	(2) The margins provided in the design will be evaluated on a case-by-case basis Factors that should be considered in quantifying margins are (a) the environmental stress levels induced during testing, (b) the duration of the stress, (c) the number of items tested and the number of tests performed in the hostile environment, (d) the performance characteristics of the equipment while subjected to the environmental stresses, and (e) the specified function of the equipment.
(3) When the qualification envelope in Appendix C is used, the only required margins are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for uncer-	(3) Same as Category I.

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

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

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

APPENDIX A

METHODS FOR CALCULATING
MASS AND ENERGY RELEASE

APPENDIX A

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

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

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

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

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

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

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

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

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

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

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

APPENDIX B

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

APPENDIX B

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

1. Methodology to Determine the Containment Environmental Response

a. Heat Transfer Coefficient

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

(1) Condensing heat transfer

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

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

(2) Convective heat transfer

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

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

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

b. Heat Sink Condensate Treatment

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

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

where M_{cond} = mass condensation rate
X = mass condensation fraction (0.92)
q = surface heat transfer rate
h_v = enthalphy of the superheated steam
h_L = enthalphy of the liquid condensate entering the sump region (i.e., average enthalpy of the heat sink condensate boundary layer)

c. Heat Sink - Surface Area

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

d. Single Active Failure Evaluation

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

e. Containment Heat Removal System Actuation

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

f. Identification of Most Severe Environment

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

2. Acceptable Methodology for Safety-Related Component Thermal Analysis

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

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

a. Condensing Heat Transfer Rate

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

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

b. Convective Heat Transfer

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

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

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

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

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

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

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

APPENDIX C

QUALIFICATION PROFILES FOR

BWR AND ICE CONDENSER CONTAINMENTS

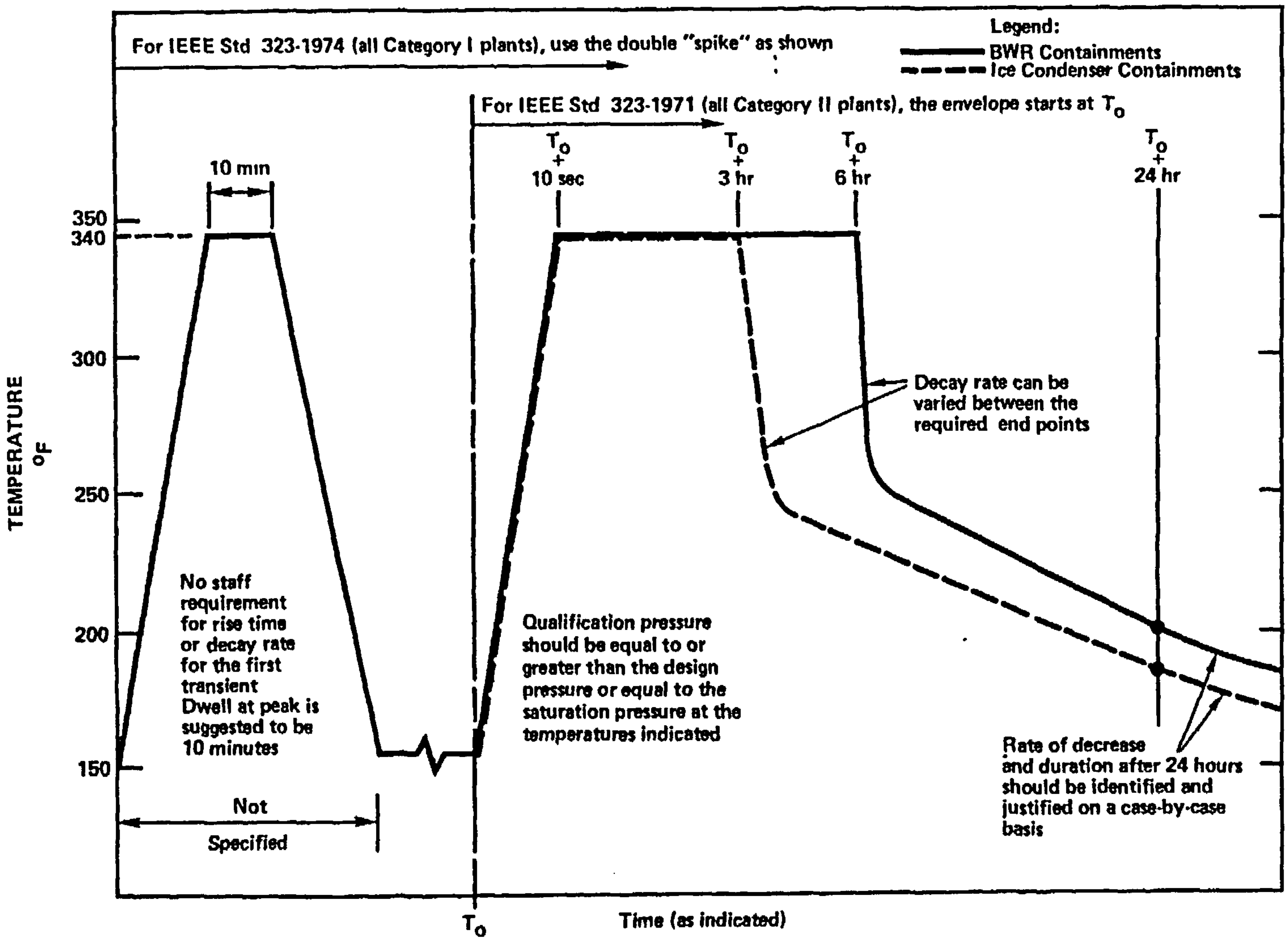

Figure C-1. Qualification Profiles for BWR and Ice Condenser Containments

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

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

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

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

THIRTY-DAY INTEGRATED DOSES

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

1. General Summary of the LOCA Scenario

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

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

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

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

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

2. Basic Assumptions Used in the Analysis

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

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

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

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

3. <u>Analysis of the Concentration of Fission Products in Air</u>

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

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

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

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

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

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

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

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

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

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

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

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

4. Departure from Past Practices

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

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

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

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

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

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

5. Analysis of the Concentration of Fission Products on Surface

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

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

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

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

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

7. Estimates of the Radiation Environment Dose and Dose Rates

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

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

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

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

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

a. <u>Containment Atmosphere Doses and Integrated Dose</u>

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

b. <u>Surface Dose and Dose Rates</u>

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

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

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

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

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

c. Dose Near Sump Water

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

d. Equipment Outside Containment

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

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

8. Comparison of a PWR and a BWR

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

REFERENCES

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

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

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

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

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

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

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

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

APPENDIX E

STANDARD QUESTION ON ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT

APPENDIX E

STANDARD QUESTION ON ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT

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

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

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

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

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

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

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

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

   a. The system safety function requirements.

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

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

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

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

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

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

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

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

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

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

Part II

Staff Responses to Public Comments

Including Appendices A Through D

PART TWO
CONTENTS

LIST OF COMMENTORS

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

T. M. Anderson, Westinghouse Electric Corporation

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

R. H. Buchholz, General Electric Company

N. W. Curtis, Pennsylvania Power and Light Company

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

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

H. W. Pielage, Entor Corporation

D. L. Renberger, Washington Public Power Supply System

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

F. Sillag, Bailey Controls Company

J. H. Taylor, Babcock and Wilcox Company

E. E. VanBrunt, Jr., Arizona Public Service Company

G. E. Wuller, Illinois Power Company

COMMENT NO. 1: (General) — Please don't refer to cable as "equipment." Wire and cable are components which may become part of equipment but are not of themselves "equipment."

Resolution

The term "equipment" as used in this report includes all types of equipment (i.e, components, subassemblies, etc.) essential for plant safety. No attempt is made in this report to differentiate between components, subassemblies, and so forth. Cable--unlike other components such as resistors, capacitors, or wires that are integral parts of other equipment--is a unique and major item that may be qualified independently of any other component and can be treated as a specific piece of equipment.

COMMENT NO. 2A: (General) — In several places in the (Discussion) section, mention is made of valve qualification. We believe this should be "valve actuator qualification." Valve qualification including valve actuators is a recent project by the ASME which has not yet been completed.

COMMENT NO. 2B: P3 Typo - reference to 382 should be for valve actuators, not valves.

Resolution

The staff agrees with the comments.

COMMENT NO. 3: (General) — It is not clear whether the Category I subparagraphs apply to Category II. It is, therefore, recommended that the subparagraphs applicable (if any) to Category II be individually identified.

Resolution

The staff concurs with this comment. If the main section is identified as being applicable to Category II, then all the subsections associated with it are also applicable to Category II unless otherwise noted.

COMMENT NO. 4: (General) — IEEE-323 is entitled "Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." NUREG-0588 is entitled "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," yet is stated to address a method acceptable to the NRC for implementing the requirements of IEEE-323. The NRC needs to explain/define their interpretation of the difference, if any, between "Class IE Equipment" and "Safety-Related Equipment." GE is particularly concerned that "Safety-Related Electrical Equipment" may refer to non-safety grade components assumed in mitigating a transient. If our assumption is correct such an expansion of IEEE-323 is unjustified.

Resolution

Electrical equipment important to safety (that is, safety-related) is a broad category of equipment and includes the well-defined subset identified in the national standards as Class IE equipment. Equipment important to safety, however, also includes other equipment addressed in the Standard Review Plan Sections 7 and 8 such as equipment required for reactor shutdown and post-accident monitoring. In addition, certain equipment may be required and classified as important to safety because it functions as a supporting system for Class IE equipment, or simply because of its association with Class IE systems. Equipment in this latter category (for example, anticipatory trips) although not essential for accident mitigation, may be considered important to safety if by its association with Class IE equipment may render the Class IE equipment inoperable.

Recognizing that functional requirements differ for different equipment important to safety, the staff is in the process of attempting to establish several categories of safety equipment. However, until these categories are defined, the existing two-category systems (equipment important to safety, and non-safety equipment) will be used.

Mitigation of transients may be considered a safety function for which non-safety grade equipment has been and may be used, as long as it can be shown that failure of that equipment does not significantly impact the mitigation of the transient or adversely affect public health and safety.

COMMENT NO. 5A: (General)

We recognize the intent of the section and, in general, agree with it; however, in the interest of accuracy it should be noted that the IEEE Standards and Regulatory Guides referenced constitute a mixed bag that may not provide the coverage expected.

Not all of them are derived from or contain the basic requirements of IEEE 323-1974. For example R.G. 1.73 endorses IEEE 382-1972 and R.G. 1.40 endorses IEEE 334-1971. Perhaps a better approach would be to state that these older standards in combination with IEEE 323-1974 constitute the bases for an acceptable approach.

COMMENT NO. 5B:

The discussion (page 2, paragraph 5) implies that conformance with the daughter standards and endorsing regulatory guides as specified will provide assurance that the equipment being qualified meets the requirements of NUREG-0588. Not all of the specified standards are related to IEEE 323-1974, such as:

Regulatory Guide 1.73 endorses IEEE 382-1972, which is related to IEEE 323-1971.

Regulatory Guide 1.40 endorses IEEE 334-1971, which is related to IEEE 323-1971.

Furthermore, it is the opinion of Westinghouse that the qualification program recommended by the pre-1974 versions of these standards do not meet the requirements of IEEE 323-1974 and therefore, reference to these Regulatory Guides and Standards should be deleted from NUREG-0588.

Resolution

The staff concurs in part with the comments. It should be recognized, however, that when a standard which has been previously endorsed by the staff is significantly revised to reflect the "state-of-the-art" technology, a revised Regulatory Guide will follow. The staff is or will be in the process of updating and issuing revisions to the above referenced guides.

COMMENT NO. 6:
(General)

We are concerned that qualification of some components may take an extended period of time. Large or heavy components requiring testing may be subject to the restrictions inherent in the very limited number of facilities in which such testing can be performed. Industry, with need for access to such a testing facility, faces a significant extension of time before all components are tested and qualified. Early implementation of the staff philosophy espoused in NUREG-0588 would have a significant impact on the issuance of CPs and OLs for those facilities awaiting component qualification.

Resolution

The staff has been implementing, in part, the positions in NUREG-0588 through its endorsement of related Regulatory Guides and individual positions on a case-by-case basis for quite some time. Therefore, on plants that are currently under review for a construction permit (CP) or operating license (OL) applications, the long lead times for qualification purposes should have been accounted for.

Recognizing that there may be equipment for which qualification may not be completed by the time a plant is ready to start up, it is incumbent on the applicant to provide justification of the adequacy of the existing design on a short-term basis until the qualification program is complete.

Methods such as ongoing qualification may be designed to resolve the long-term qualification programs. Other methods (than those described in NUREG-0588) that are designed to satisfy the requirements of the General Design Criteria 1, 2, 4, and 23 of Appendix A to 10 CFR Part 50 may be proposed, and may be found acceptable on a case-by-case basis.

COMMENT NO. 7A:
(General)
(Category II)

NUREG-0588, particularly as it applies to IEEE 323-1971, is not a reasonable interpretation of the standard. The NUREG, in actuality, extends the standard into new areas rather than interpreting existing criteria.
In three specific areas (aging, margin and qualification by analysis) the NUREG has either added to or deleted from the standard. Neither the words nor the intent of aging and margin have ever before been included as

part of IEEE 323-1971. To include them at this time is to revise the standard nine years after its issuance and to negate the actions of the NRC and the work of the nuclear industry during that period.

COMMENT NO. 7B: (Category II) There are a number of substantially completed plants that will be affected by NUREG-0588, from those with construction activities well advanced to those in the "near term operating license" category. Changes in qualification and documentation requirements have significant cost and schedule impacts on such plants. We strongly question the benefit of across-the-board application of the document in its current form, especially in regard to plants committed to meeting IEEE 323-1971 (Category II). These plants are currently being handled on a case-by-case basis in this area, as is appropriate. Changes in requirements should only be made where there is demonstrable significant additional protection of public health and safety.

Resolution

It is not the intent of the NUREG to interpret IEEE Standard 323-1971 but rather to supplement it and to focus attention and activity on areas where additional improvements and guidance in qualification are deemed essential to satisfy the applicable General Design Criteria of Appendix A to 10 CFR Part 50.

Although the 1971 version of the standard does not uniquely identify aging and margin as parameters that have to be addressed with any defined degree of rigor, aging has been a requirement in the national standards for selected equipment since 1971 and has been incorporated in other ancillary standards since that time. Providing margins during testing (or when analysis is used) has always been considered standard and good engineering practice to ensure that the design conditions under consideration have been enveloped.

Therefore, the staff does not agree with the suggestion that the positions for Category II should be omitted on the basis given. Implementation and the degree of conformance of these positions will be evaluated on a case-by-case basis. On the older plants, backfitting an acceptable degree of conformance to the positions will be made where it is demonstrated that additional assurance is warranted.

COMMENT NO. 8: (General) The introduction to the document states that the staff position developed prior to the TMI-2 event and any additional requirement or modifications will be identified later. This position is unacceptable from the standpoint that the data available today (Reference: "Technical Staff Analysis Report on Alternate Event Squences to President's Commission on the Accident at Three Mile Island," by William R. Stratton, et al., October 1979, Washington, D. C.) demonstrates a significant difference in airborne activity available for release from the containment from those assumed by the staff in a DBA. The Kemeny Report notes that the evaluation of the consequences of reactor accident have, in the past, been dominated by the iodine doses. TMI-2 demonstrates that in this type accident, at least, those estimates have been grossly and conservatively pessimistic. The difference between the design basis LOCA and the Kemeny Report notes that the evaluation of the consequences of reactor accident have, in the past, been dominated by the iodine doses. TMI-2 demonstrates that in this type accident, at least, those estimates have been grossly and conservatively pessimistic. The difference between the design basis LOCA and the Kemeny Report estimates range up to three orders of magnitude. Such a range would have a significant impact on the qualification of components. Prior to the implementation of the staff qualification program, these differences need to be resolved since there appears to be a wide difference between the assumed NRC source terms and the White House approved Kemeny Report estimated values.

Resolution

The staff agrees that its consequence calculations for the site boundary and the low population zone have been dominated by conservative estimates of airborne iodine concentrations. However, for inside containment, the noble gas contribution to the gamma and beta doses is substantially greater and therefore dominates any dose contribution resulting from airborne radioactive idoine. (Refer to NUREG-76-6521 and Sandia Report No. 78-0091 for additional information.)

The NRC staff recently prepared two reports, NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," and NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions." These studies reflect not only the TMI-2 accident experience but also the results of recent research and improved methods of analysis. The findings of these studies will be factored into the rulemaking. In the interim, the source terms for equipment qualification shall remain as defined in position 1.4(1).

COMMENT NO. 9A: (Section 1.2)

In Section 1.2(5) the staff position restricts the use of the calculation model (Appendix B) to deriving the peak surface temperature. This is unnecessarily restrictive in that the item of interest is the temperature of the critical components inside the equipment under test as compared to estimated temperatures under DBE conditions. Westinghouse believes that the method documented in WCAP 8936 continues to be valid and conservative. The following change to Item 1.2(5) is, therefore, recommended:

(a) "Show that the peak internal temperature of the component to be qualified does not exceed the LOCA qualification internal temperature using the method discussed in Item 2 of Appendix 2 as a boundary condition."

(b) "If the calculated internal temperature..."

COMMENT NO. 9B: The many comments on this item (1.2(5)) questioned the wisdom of stressing surface temperature and of the lack of adequate guidance on modeling intrinsic heat capacity in order to obtain a measure of the thermal lag and the resultant effect on the internals of the equipment. A suggested rewording is as follows:

(b) "...or show that the peak internal temperature of the component to be qualified does not exceed the LOCA qualification internal temperature using the method of Appendix B, Item 2 as a boundary condition."

(c) "If the calculated internal temperature (or the calculated surface temperature if internal temperatures are not calculated)..." SC 2 assumes the methodology of Appendix B has an auditable basis.

COMMENT NO. 9C: In Section 1.2(5)(b), the main point of the analysis should be to show that critical internal components do not reach higher temperatures during MSLB than during LOCA. Ideally, the surface temperature is indicative of the internal temperature. This may not apply under the required analysis method, leading to an erroneous conclusion.

COMMENT NO. 9D: This Section (1.2(5)(b) and (c)) indicates that if the calculated surface temperature exceeds the qualification temperature, the component must be requalified or protection must be provided. The qualification temperature should be that which applies in the critical part of the component and not the surface temperature of the component. The peak surface temperature may exceed the required qualification temperature but the component would still function correctly. Furthermore, time-at-temperature is an important consideration which should be factored into any qualification evaluations.

COMMENT NO. 9E: The requirement 1.2(5) should be revised to allow component testing for steam line break environmental parameters as an option to analysis utilizing what are judged to be overconservative heat transfer coefficients given in Appendix B, Item 2.

Resolution

An important consideration in qualifying a piece of equipment is the identification of the various failure modes of the component. This information is necessary prior to the determination of the critical element or elements. For components containing only a few elements with symmetrical geometry the above determination may be achievable. For more complex components, however, all failure modes may not be identifiable. As a result of the difficulty in identifying failure modes, surface temperature was selected as a generic parameter. If additional information can be provided to ensure that the specific failure modes can be identified and justified, then consideration may be given on a case-by-case basis to the use of a temperature other than at the surface of the equipment. Internal component temperature would be considered only on a case-by-case basis for Category II equipment, if supporting justification can be provided. Component testing for qualifying to the steam line break environment certainly can be performed using the actual approved temperature profile. However, in addition to using the correct pressure-temperature profile, the containment turbulence and air content must be properly taken into account. (See also staff response to Comment No. 58.)

<u>COMMENT NO. 10:</u>
<u>(Section 1.2)</u>

In item 1.2(5)(a), Category I requires calculation of the envelope of peak temperature for MSLB while Category II requires only a single point (this is being inferred) peak temperature based on different ground rules.

Why is there a difference in requirements for the categories?

In item 1.2(5)(c) the ordering of the items listed was changed from those given for Category I to those for Category II. Is there any reason or significance to be associated with this?

<u>Resolution</u>

The intent of Section 1.2(5)(a) Category II is to require a calculation for the peak temperature <u>envelope</u> (not a single-point calculation).

With regard to the second point, Section 1.2(5)(c) for Category I requires that testing be the principal qualification method. These plants are in the early stages of design and have the opportunity for such equipment qualification testing at the anticipated bounding design conditions. Category II, which applies to near-term operating license (NTOL) applications and operating reactors, recognizes the vintage of the equipment and allows additional justification to be provided, by analytical means, to demonstrate that the equipment can maintain its required functional operability if the calculated MSLB temperature at or near the surface of the equipment exceeds the LOCA test temperature (for which the equipment has already been qualified). This type of qualification has been and will be applied on older NTOL plants.

<u>COMMENT NO. 11:</u>
<u>(Section 1.1)</u>

Consider referencing CSB BTP 6-1 for one manner in determining LOCA qualification temperature. Consider surface thermocouple measurements as one way for determining LOCA qualification temperature. Consider mentioning that if the component is temperature soaked in a LOCA qualification test for a period of time resulting in justifiable quasi-equilibrium temperature conditions (example: 3 or 4 temperature time constants) then the LOCA qualification temperature would be equal to the test chamber temperature.

Resolution

Other methods may be used to measure surface temperature of the component provided that it can be shown that thermal equilibrium exists in the test chamber and at the equipment under test. Simulating and monitoring the rise time of the temperature transient should not be ignored. (See also staff response to Comments No. 9 and 58.)

COMMENT NO. 12: (Section 1.1) Section 1.2(2) implies that Appendix A contains the only models acceptable for calculating containment environmental parameters. The NRC should clarify that other models are also acceptable if approved by the staff.

Resolution

Sections 1.1(1) and 1.2(1) state that other models approved by the staff may be found acceptable.

COMMENT NO. 13: (Section 1.3) The containment spray system is not the only source of chemicals under high energy line break conditions; boric acid should also be addressed. The following change to Item 1.3 is therefore, recommended:

The sentence:

"The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system."

Should be changed to:

"The concentration of caustics used for qualification should be equivalent to or more severe than those used in the plant containment spray system, during both the initiation and recirculation phases."

Resolution

The staff concurs. This change will be considered in the proposed rulemaking and/or the revision to Regulatory Guide 1.89 to be issued for public comment in December 1981.

COMMENT NO. 14: (Section 1.3) Any chemical change resulting from a malfunction of equipment would be addressed by spraying during testing with the required solution or one that correctly simulates its effects. It would not be necessary to use a different solution as its effects could be different than those of the actual solution.

Resolution

The staff concurs. If a spray solution is used in a simulated test--the effects of which could be different to those provided by the solution in the actual plant--the testing would be considered unacceptable. However, if a more concentrated form of the solution is used during testing as a bounding condition, the adequacy of such testing, if justified, may be found acceptable. No change was proposed as a result of this comment.

COMMENT NO. 15: (Section 1.3)	Add words "where applicable" to this article.

Resolution

The staff does not agree that the position should be modified to include these words. The applicability of all the positions in the text is a function of the design. As indicated in the discussion, alternatives (or exceptions) may be proposed and, if justified, may be found acceptable.

COMMENT NO. 16A: (Section 1.4) (Category II)	The concept of a qualified life is not a requirement for Category II plants. The following change is, therefore, recommended: The words: *"over the equipment qualified life"* should be deleted from the first sentence at Section 1.3 for use in the Category II column.
COMMENT NO. 16B: (Category II)	By statement in Section 1.4, "qualified life" is not applicable to Category II. The statement "over the qualified life" should be deleted from 1.4 for Category II.

Resolution

With the exceptions noted in Section 4(1) of the NUREG, the staff does not require that a qualified life be established for all Category II equipment.

The words "qualified life" may be interpreted as "installed life" for Category II equipment.

COMMENT NO. 17A: (Section 1.4(1))	In item 1.4(1), the LOCA source term utilized should reflect the multi-level term in the proposed revision to Regulatory Guide 1.89, Revision 1, November 1, 1976. For non-LOCA accidents, gap release from 100% of the fuel rods is the principal basis and, in accordance with NUREG/CR-0091, a total release fraction of 1-2% of the noble gases is sufficient.

COMMENT NO. 17B: The staff position in Item 1.4(1) of requiring 100% of the gap activity (approximately 10% LOCA) released to the containment for all other non-LOCA design basis accidents cannot be justified since:

a. Many design basis accidents do not result in a breach of either the primary or secondary systems. Thus, for equipment that is only required to protect against such contained faults, the application of any accident related dose is illogical and unnecessary provided that the equipment can be shown to have no adverse effect under high energy line break conditions, as required by Item 2.1(3)(b) and Appendix E.

b. For equipment that is only required to function following a secondary side break, the application of the dose that would result from the release of 100% of the gap activity is grossly conservative. Westinghouse dose calculations have conservatively assumed 1% clad damage (1% gap activity release) and considering the fraction of the core activity in the RCS as 0.003 Kr-85, 0.001 halogens, and 0.001 of other noble gases. It was also conservatively assumed that all of RCS inventory was instantaneously released into the containment atmosphere at the initiation of the incident. This method is documented in WCAP-8587 Section 6.8.4. The following change to item 1.3(1) is therefore, recommended:

(1) "The source term to be used in determining the radiation environment associated with the design basis LOCA should be taken as an instantaneous release from the fuel to the atmosphere of 100 percent of the noble gases, 50 percent of the iodines, and 1 percent of the remaining fission products. For secondary side break design basis accident conditions, a source term involving an instantaneous release from the fuel to the containment atmosphere of 0.1 percent of the noble gases (except for Kr-85 for which a release of 0.3 percent should be assumed) and 0.1 percent of the iodines is acceptable. For design basis accidents that do not result in a breach of either the primary or secondary systems only the normal operational dose need be considered.

COMMENT NO. 17C: In Item 1.4(1) the position required excessively conservative and unrealistic assumptions in determining the source terms for design basis accidents. For all design basis accidents, any core damage and the subsequent release of radioactive material will not occur instantaneously, but instead will occur over some period of time. Consideration of time dependent release of radioactive material should be permitted in the determination of accident radiation environments.

The use of approximately 10% of the LOCA source terms for all other non-LOCA design basis accidents has no apparent basis and is overly conservative. Equipment which is required to function during or after a non-LOCA design basis accident need only be qualified to the radiation environment resulting from that accident (with adequate margin).

It is recommended that an additional sentence be added to this position as follows:

"The time-dependent release of radioactivity and the use of alternate source terms may be found acceptable when supported by conservative analysis for the specific accident of concern."

COMMENT NO. 17D: Appendix D should reflect the multi-level source term as reflected in the proposed revision to R.G. 1.89, Revision 1, November 1, 1976.

COMMENT NO. 17E: Appendix D provides "Sample Calculations and Type Methodology for Radiation Qualification Dose." In the section on the basic assumptions used in the analysis it is stated that "between 20 and 80% of the fuel rods could experience cladding failure for a PWR and a lesser fraction for a BWR." The current GE licensing basis model calculates that fuel perforations occur only beyond 15,000 to 20,000 MWD/T exposure and then only in the high power bundle. Our best estimate model does not calculcate any fuel perforations for a LOCA. In determining the source term to be used for equipment qualification, the vendor should be able to use as a basis his staff approved Appendix K model in determining the number of rods which are calculated to have failed. This comment also pertains to the statements in Section 1.4(1) of NUREG-0588.

Resolution

An NRC-sponsored research effort is investigating the use of a multi-level and time-dependent release of radioactivity from the fuel following the design basis accident LOCA. Until the results of this effort are available, the staff will continue to use the source terms presented in this interim report. The final rulemaking will factor in the results of any additional findings identified in these ongoing investigations.

The staff maintains the position that in some non-LOCA accidents--in particular those that are power-increasing transients--the inventory in the fuel rod gaps may be larger than predicted by NUREG/CR-0091. Therefore, to be conservative, the value of 10% of the rod inventory in the gaps will be retained. The staff agrees with the comment that the 100% cladding failure assumption may be overly conservative. As part of the proposed rulemaking, the staff is considering using the conservatively calculated estimates of fuel damage for non-LOCA transients instead of using the current assumption.

In response to comment 17E, the source terms in position 1.4(1) are not based on best estimate calculations. The fuel damage estimates in Appendix D were intended to show the amount of conservatism provided by position 1.4(1). The degree of conservatism, however, does not appear to be quite as high as initially envisioned in light of the data of the TMI-2 accident. To avoid misinterpretation of this intent, the discussion of the best estimate models of fuel damage following a DBA has been deleted from Appendix D.

COMMENT NO. 18: (Section 1.4)

In Item 1.4(1), the fission product release assumptions to the containment atmosphere are different from those which have been traditionally used in Regulatory Guide 1.3 for the design basis accident. Furthermore, we have not assumed, nor in the past has the staff assumed, the fission product releases identified in NUREG-0588 for "all other non-LOCA design basis accident conditions." For the fuel drop accident, it appears that the NUREG is inconsistent with the Regulatory Guide 1.25. The document needs to be modified to reflect currently approved fission product transport models.

Resolution

The values for the iodine and noble gas portions of the source term used in NUREG-0588 are identical to the source term identified in TID-14844, which is also the starting point for the source-term assumptions for Regulatory Guides 1.3 and 1.4. The staff is currently evaluating the adequacy of the source terms in light of the TMI-2 event, and will factor in the results of their study in the final rulemaking to be issued to public comment in December 1981. One significant finding of the TMI-2 event is the significant amount of cesium in the coolant (between 40 and 60%). The incorporation of cesium in the source term appears to be a more appropriate treatment of fission products in the coolant (other than iodine) in addition to the current assumption of the 1% solids.

The comments about the source-term assumptions for all other non-LOCA accidents have been addressed in response to Comment No. 17.

COMMENT NO. 19: (Section 1.4 and Appendix D)

Paragraph 2 of Appendix D should be rewritten to state clearly the basic assumption of 100% fuel clad failure. A simple statement that the source term is that given in position C2 of Regulatory Guide 1.89 is appropriate. It is suggested that all wording in the first paragraph after the words "core meltdown" be deleted.

Resolution

Position 1.4(1) and Appendix D describe the current staff positions on source terms for equipment qualification.

See also the response to Comment No. 17 for source terms for non-LOCA accidents.

COMMENT NO. 20: (Section 1.4(1)

A conflict exists between the postulated source term values in NUREG-0588 and NUREG-0578 (TMI Short Term Lessons Learned). The use of NUREG-0578 source terms will result in even higher values than those presently given in NUREG-0588.

Resolution

The "For Comment" version of NUREG-0588 provided the methods for determining the radiation source term when considering LOCA events inside containment (100% noble gases/50% iodine/1% particulates). These methods considered the radiation source term resulting from an event which completely depressurizes the primary system and assumes the release of the source term inventory instantaneously to the containment.

The "For Comment" version of NUREG-0588 also provides the radiation source term to be used for qualifying equipment following non-LOCA events both inside and outside containment (10% noble gases/10% iodine/0% particulates).

NUREG-0578 provided the radiation source term to be used for determining the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment as a result of LOCA. This method considered a LOCA event in which the primary system may not depressurize and the source term inventory remains in the coolant.

The apparent conflict between NUREG-0588 and NUREG-0578 has been resolved and reported as clarification for item II.B.2 in NUREG-0737. The incorporation of all source term assumptions for equipment qualification will be provided in final rulemaking to be issued for public comment in December 1981.

Reduction of the noble gas contribution in the source term (assumed in the reactor coolant system per NUREG-0578) may be warranted for those designs (systems) where the primary coolant system is depressurized before the reactor coolant flow through these designs (systems) is initiated (for example, the residual heat removal system outside containment).

COMMENT NO. 21:
(Section 1.4(1))

(1) In recent requirements imposed by the NRC on the Near Term Operating License Plants, the staff has required a change in the assumptions used for the calculation of post-accident radiation dose for equipment internal to the RCS.

(2) In recent drafts of Regulatory Guide 1.97, the staff is requiring a much lengthened post-accident monitoring time period.

These current staff positions should be included in NUREG-0588 and issued for comment, as part of the NUREG.

Resolution

The staff concurs with item no. 1. Position 1.4(1) contains the current staff requirements for source terms for equipment inside the containment. Regarding equipment internal to the RCS, the source terms to be used have been provided to the Near Term Operating License (NTOL) plants as clarification to Item II.B.2 of NUREG-0737. (See also response to Comment No. 20.)

With regard to item 2, Regulatory Guide 1.97 provides guidance for instrumentation used to assess plant conditions during and following an accident. A limited number of instruments covered by Regulatory Guide 1.97 are to be designed for worst-case conditions, that is, total core meltdown.

The radiation source terms in NUREG-0588 represent a partial core meltdown and should not be used for that limited group. Use of the staff positions in NUREG-0588 for the qualification of the remaining instrumentation covered by the Regulatory Guide depends on an individual functional requirement for each instrument. This determination should consider the type of accident, the function of the instrument during and following that accident, and the portion(s) of that instrument located within a harsh environment caused by the accident. The specific positions for postaccident monitoring are provided in Regulatory Guide 1.97 and are outside the scope of this generic report. Final rulemaking, however, will address the postaccident monitoring requirements.

<u>COMMENT NO</u>. 22: (Section 1.4) Item 1.4.14 on page 10 states that qualification levels given in Appendix D are adequate. However, the Appendix D analysis ignores the normal operation dose which is required in Item 1 on page 7. This should be resolved.

<u>Resolution</u>

Section 1.4 requires that the qualification dose should be the sum of the normal and accident doses. Appendix D addresses only the accident doses. It should be noted that the dose values in Appendix D are provided for illustrative purposes and they may not be appropriate for plant-specific application. Any modifications to position 1.4 (14) will be incorporated in the final rulemaking to be issued for public comment in December 1981.

<u>COMMENT NO</u>. 23: (Section 1.4) In Section 1.4(1) it is stated that 1% of the remaining fission products are released instantaneously to the atmosphere. In contrast, Section 3 of Appendix D ignores these other fission products when determining the airborne sources. Elsewhere in Appendix D, it is stated that these other fission products are released instantly to the sump fluid at T=0. We recommend that this inconsistency be resolved with the other fission products being released to the sump fluid only.

<u>Resolution</u>

The staff agrees with the comment. The intent of the position is that the 1% solids are assumed to be instantaneously released from the fuel to the coolant and are carried by the primary coolant to the containment sump. See also response to comment No. 18.

<u>COMMENT NO</u>. 24: (Section 1.4) In two places of the wording Section 1.4(1) "... instantaneous release from all the fuel..." is suggested for clarity.

<u>Resolution</u> The staff agrees with the comment. The suggested accommodation will be considered in the final rulemaking.

<u>COMMENT NO</u>. 25: (Section 1.4) In Section 1.4(1) the requirement to assume an instantaneous, non-mechanistic release of activity from the fuel is inconsistent with the time-dependent,

mechanistic approach required for radioactivity redistribution analyses in containment and auxiliary building volumes. As briefly discussed in Appendix D to NUREG-0588, any core damage and subsequent release of activity will require a significant amount of time which would depend on the accident scenario. Since this NUREG is establishing more realistic and rational bases for estimated radioactivity levels after release from the fuel the same approach should be applied to fuel releases themselves. This time-dependent fuel release fraction is particularly significant for equipment which is required to function for only a short time following a LOCA/MSLB. Enforcement of this requirement will cause significant equipment replacement for Category II plants. We do not believe enforcement of this position can be defended on a cost/benefit basis.

Resolution

See the staff response to Comment No. 17 regarding time-dependent fuel releases.

Regarding the comments on equipment replacement: Category II plants in NUREG-0588 (applicable to equipment qualified in accordance with IEEE Standard 323-1971) should already have qualified equipment using the source terms previously acceptable to the staff (that is, instantaneous fission product release per Regulatory Guide 1.3 and 1.4). In areas where the "solids" contribution is significant, equipment requalification may be warranted unless appropriate justification is provided to demonstrate the adequacy of previous qualification methods (such as the use of shielding, and so forth).

For plants that did not qualify equipment using the source terms acceptable to the staff, equipment replacement or requalification may also be warranted, unless the adequacy of the qualification methods used is justified on some defined basis.

COMMENT NO. 26: (Section 1.4)

(1) Section 1.4, *"Radiation Conditions Inside* and Outside Containment" does not indicate that Appendix D is a sample calculation for a PWR, but that the general approach is applicable to BWR. Additional wording to this effect would enhance the clarity of the section.

(2) Appendix D itself should indicate very early on that it is a sample calculation for a PWR.

Resolution

The staff concurs with the comment. This recommended change will be considered in the proposed rulemaking and or revision to Reg. Guide 1.89 to be issued for public comment in December 1981. In addition, Appendix D was modified to include the assumptions for modeling radiation environments for BWR (as well as the PWR) containments.

With regard to Item 2, paragraph 3 of the revised Appendix D clearly indicates that the numbers presented are strictly the results of a sample calculation for a PWR which uses the methods and assumptions in the appendix.

COMMENT NO. 27A: (Section 1.4(3)) With respect to 1.4(3) GE does assume uniform distribution of activity throughout the containment at time 0. The mechanistic treatment of fission product transport for the non-mechanistic accident event is thought to be a significant deviation from past staff acceptances of the GE design. The staff needs to provide greater explanation for the need for this change.

COMMENT NO. 27B: In Appendix D(3), the multi-mode mechanistic fission product transport mechanisms should not be considered in the analysis of the BWR. It was thought that by using the conservative real world source terms, such refinements as defined on page D.9 could and should be avoided.

Resolution

Appendix D was modified to provide the appropriate assumptions for distribution of activity within the containment, following the accident, for both PWRs and BWRs. See also response to comment No. 28.

COMMENT NO. 28: (Section 1.4(3)) The last sentence of 1.4(3) appears to be in conflict with Appendix D.8.a which appears to restrict the assumption to PWR while 1.4(3) restricts it in all cases. Which is correct?

Resolution

The intent of Position 1.4(3) is to prevent a situation where an assumed uniform distribution of activity throughout the entire containment could result in a nonconservative estimate of the qualification dose or dose rate. This position by itself does not preclude the use of a uniform distribution when that assumption is appropriate. Section 2.2 and 2.3 of the revised Appendix D provides appropriate assumptions for the initial distribution of activity inside containment.

COMMENT NO. 29: (Section 1.4(6)) GE analysis considers radiation at the centerpoint of any given compartment rather than the NUREG position which specifies radiation at the centerpoint of containment It seems unnecessarily conservative not to account for the presence of internal structures.

Resolution

The staff does not preclude the dose calculation within compartments. Further, the staff position 1.4(6) allows for reduction of the calculated beta and gamma doses if the dose point is such that internal structures or shielding contribute to the reduction of the dose. However, any claims for reduction in the doses due to either internal structures or shielding must be clearly documented and justified.

COMMENT NO. 30: (Section 1.4(7)) With respect to Section 1.4(7) and 1.4(9) in the NUREG, the GE analyses for radiation are based on a semi-infinite medium analagous to Regulatory Guide 1.3 - doses to people. The staff is apparently taking the position that an infinite concept is unacceptable.

Resolution

The use of the infinite cloud assumption is in connection with the airborne beta radiation dose only. The assumed dose point on the containment centerline is surrounded on all sides by the containment atmosphere thus an infinite cloud assumption is appropriate. Also, positions 1.4(7) and 1.4(9) do not preclude the use of a semi-infinite cloud assumption if it can be adequately justified.

COMMENT NO. 31: (Section 1.4(7) Appendix D) Any justification of the assumption in Appendix D Section 7(b), that, "all betas directed toward the coating were assumed to be absorbed in the coating," would be analytically difficult. We feel that it would be more appropriate for the actual beta dose at a designated depth to be evaluated; the 10-mil depth where adhesion occurs would probably be most appropriate.

COMMENT NO. 32: Also in Appendix D, Section 7(b), the method of dose evaluation to be applied to cable insulation layers is vague. Is it intended that the total absorbed energy be distributed throughout the mass of the insulation or that the dose determined for the coating be applied to the entire cable insulation? The first method would underestimate while the second would be an overestimate. Once again, we recommend that it would be more appropriate to determine the actual beta dose at a predetermined critical depth. It should also be noted that item 1.4.9 on page 9 implies that the beta dose from plate-out on cables can be ignored, but this contradicts item 1.4.7 and page 9.

Resolution

The doses calculated using the methods of Appendix D are estimates at the surface of the equipment. The staff does not wish to use the approach of specifying a dose at a predetermined critical depth because the critical depth doses are dependent on the absorbing materials (which is different for different equipment).

The beta dose from plate-out on cables cannot be ignored. The intent of position 1.4(2) is to explicitly require the consideration of all radiation sources when calculating the qualification doses, which includes the beta dose from plate-out sources on cables.

COMMENT NO. 33: (Section 1.4(7) Appendix D) In Section 7(b) it is stated that the gamma dose for coatings due to plate-out is negligible because the absorbed dose in the coatings is small. Since the purpose of the model in Appendix D is to determine the radiation environment to which the coatings should be subjected in qualification tests rather than the absorbed dose in the coatings, the gamma dose in Rads (C) or Rads (air) should be determinted on the basis of the total dose due to both airborne and plated-out sources at the surface of the coatings.

Resolution

The staff concurs. Position 1.4(2) shall be interpreted to include all potential sources when calculating qualification doses (which would include both airborne and plated-out sources).

COMMENT NO. 34: (Section 1.4(9)) The argument given in Section 1.4(9) for reducing, by a factor of at least 2, the beta dose for qualification of cables arranged in trays based on localized or self shielding effects can be extended to other components. Any exposed components will be sufficiently massive to attenuate beta radiation from the containment atmosphere on the opposite side. Hence, the beta dose at the surface of unshielded equipment should, in general, be half the beta dose calculated at the containment center.

Resolution

Implementation of this assumed dose reduction may be warranted for a piece of equipment in a specific location. Sufficient justification should be presented to warrant the above-mentioned reduction in calculated beta doses and should be evaluated on a case-by-case basis. A beta dose reduction due to shielding from large internal structures is acceptable.

COMMENT NO. 35A: (Section 1.4) With respect to Sections 1.4(7), (8), (10), (14), requiring qualification to beta radiation at this date for Category II could be traumatic and needlessly so. We recommend that the requirement be tied to results of the IEEE 323-1974 qualification programs such that material which evidences adverse beta effects be addressed or replaced for Category II.

COMMENT NO. 35B: With respect to item 1.4(7), (8), (9), (10, (14), the qualification for the effects of beta radiation was not a requirement for Category II plants and systematic enforcement of this requirement at this stage will have major impact. A reasonable alternative would be to require that any significant adverse experience gained during qualification testing of equipment, for the effects of beta radiation, to IEEE 323-1974 be considered for applicability to Category II plants.

Resolution

The staff agrees with the comment. Any modification to the positions will be considered during the final rulemaking to be issued for public comment in December 1981.

It is the staff's belief that the qualification dose should account for all types of radiation present at the equipment location. The staff permits the reduction of calculated beta doses to account for localized shielding (that is, component and/or structural shielding) and has provided additional guidance in the DOR guideline document. When a significant beta dose reduction can be justified, the staff expects the equipment qualification dose to equal or exceed the gamma radiation dose calculated using assumptions and models similar to those in Appendix D. For any safety-related component not meeting the calculated qualification values, justification for the adequacy of the design should be provided, or a modification of the design to satisfy the above radiation requirements may be warranted. Replacement equipment or equipment that has not yet been qualified should conform to the Category I requirements.

COMMENT NO. 36: (Section 1.4(10)) Position 1.4(11) discussed the need to consider the exposure received by ECCS equipment located outside containment from sump fluids. This need is understood but the reference to Appendix K to 10 CFR Part 50 is unclear This reference should be made more specific or removed from the position.

Resolution

The staff agrees with the comment. Any modification to the positions will be considered during the final rulemaking.

COMMENT NO. 37A: (Section 1.4(11)) The implication that radiation qualifications must be performed for low level doses (no matter how small) on all equipment (no matter how radiation tolerant) is unfortunate. Surely some guidance can be given of a more practical nature for equipment located in regions of trivial integrated dose (present consensus is that the threshold of triviality occurs at approximately 1×10^4 rads).

COMMENT NO. 37B: The implication that radiation qualification must be performed for low level doses (no matter how small) on all equipment (no matter how radiation tolerant) is unfortunate. Surely some guidance can be given of a more practical nature for equipment located in regions of trivial integrated dose (present consensus is that the threshold of triviality occurs between 5×10^3 and 1×10^4 rads). The guidance might take the form of the tabulation in Appendix C of the NRC's recent "Guidelines for Evaluating Qualification of Class IE Electrical Equipment on Operating Reactors," November 1979.

COMMENT NO. 37C: This position (item 1.4(11)) requires that test data are required to exempt equipment from radiation qualification, even if the integrated dose is less than 10^4 rads. There is no apparent basis for this requirement. Numerous tests have provided data that show that radiation damage thresholds are greater than 10^4 rads. To require radiation testing below 10^4 rads will only add significantly to the cost of testing programs without adding to plant safety. It is recommended that the position be rewritten as follows:

Equipment that shall not be exposed to an integrated dose greater than 10^4 rads may be exempted from radiation qualification.

Changes to this position should be made for specific materials if any, with a radiation damage threshold below 10^4 rads, are identified.

COMMENT NO. 37D: Inclusion of equipment qualification testing for equipment with radiation doses below 10^4 rads would require substantial expenditures of time and money for qualification testing with no corresponding benefit to health and safety. Of the general classes of materials or components (organic compounds, ceramics, metallics, electronic components), only organic compounds and electronic components are susceptible to damage from moderate amounts of gamma or beta radiations. Numerous studies have compiled radiation effects data on all the classes of organic compounds and show that the least radiation resistant compounds have damage thresholds greater than 10^4 rads and would remain functional with exposures substantially above the threshold value. Thus, for organic materials, an exposure level of 10^4 rads is reasonable threshold value below which proper qualification is assured without adding the substantial costs of testing.

For electronic components, studies have shown failures of metal-oxide-semi-conductor devices at 3.5×10^3 rads. Therefore, a lower minimum qualification value should be assigned probably in the range of 1×10^3 rads. This would also provide adequate margin for safety without an unreasonable qualification test requirement.

Resolution

It is not the staff's intent to imply that testing is the only acceptable method for demonstrating qualification adequacy of equipment that will be exposed to low-level radiation. Other methods (such as analysis), if they are supported by test data, literature search (on identical or sufficiently similar material and/or equipment where extrapolation of the data to the actual equipment being qualified is feasible), or operating history (if it is supported by test data) may also be found acceptable.

The staff concurs that there may be information available to indicate that many materials used today have radiation dose and dose rate damage thresholds greater than 10^4 rads. However, there are also components that may be made of materials susceptible to low level radiation dose and dose-rate damage (for example, Telecon TFE and integrated circuits). Therefore, low-level radiation should not be dismissed on a generic basis and should be evaluated on a case-by-case basis.

COMMENT NO. 38A: (Section 1.4(11)) (Category II)

It has long been the industry practice to ignore radiation in zones where the total dose was less than $10^3/10^4$ rads. To impose this requirement without technical basis is arbitrary and without adequate cost/benefit consideration.

COMMENT NO. 38B: (Category II)

Enforcement of this requirement for Category II plants will have a major impact. The systematic addressment of radiation doses below 10^4 rads constitutes addressment of low-level inservice radiation aging effects, which was never a requirement for these plants. Westinghouse believes this effort cannot be justified on a cost/benefit basis and should be limited to consideration of any radiation sensitive materials identified during qualification of equipment to IEEE 323-1974.

Resolution

See staff response to Comment No. 37.

COMMENT NO. 39: (Section 1.5(3))

Where Class IE equipment is served by redundant environmental support systems, such as the main control room, this section should not be interpreted to mean the loss of both redundant support systems.

Resolution

If the redundant systems are separate and independent so that a single failure will not render both systems inoperable, the interpretation of the comment is correct. There may be designs and/or procedures, however, that may shut down both redundant and independent environment support systems during plant outages. For those designs, a loss of both redundant support systems should be assumed.

COMMENT NO. 40: (Section 1.5(2)) (Category II)

For equipment not subject to a design-basis-event accident environment, documentation of environmental qualification to the limits of normal and abnormal environments was not required for plants committed to IEEE 323-1971. Rather, equipment specifications included such environmental limits to be considered in the design and purchase of the equipment. A requirement to document qualification by test or analysis would constitute a major impact for Category II plants.

The following change is, therefore, recommended:

The word "qualified" in both subparagraphs should be changed to "qualified or designed" for use in the Category II column.

Resolution

IEEE Standard 323-1971 states that the service conditions required to be addressed include "Environmental conditions expected as a result of normal operating requirements, expected extremes in operating requirements (i.e., abnormal environments) and postulated conditions appropriate for the design basis events of the station." Therefore, the staff does not agree with the conclusions reached in the comment.

The main purpose of qualification is to verify the performance adequacy and/or the capability of a design. A specification alone does not provide this verification. A purchase specification supported by a certificate of compliance which is based test or test and analysis could constitute acceptable qualification documentation. It should also be noted that the position in Section 2.1(4) does not limit the qualification to only test or analysis.

COMMENT NO. 41: (Section 1.5)

Section 1.5(1) requires that equipment located in areas that could be subjected to high energy pipe breaks (HEPB) should be qualified to the condition resulting from the accident for the durations required.

Comment: Only that equipment necessary to mitigate the consequences of the postulated HEPB accident need be qualified to the respective HEPB conditions.

Resolution

The staff concurs in part. If any equipment failure resulting from an HELB will be detrimental to safety (even though this equipment is not necessary for mitigating the consequence of the accident), that equipment should also be qualified to the HELB conditions (see Section 2.1(3)).

COMMENT NO. 42A: (Section 1.5)

There is a significant lack of evidence of consideration of the "systems analysis method" required in the guidelines accompanying IE Bulletin 79-01B. This is especially notable in the treatment of high energy line breaks (HELB) outside containment. The words in the introduction to the NUREG indicate that all equipment is required to meet the worst environments resulting from all events. The NRC Branch Technical Position on HELB outside containment clearly indicates that only that equipment required to mitigate the HELB, that is to achieve safe plant shutdown, is required to be qualified to the HELB environment. The introduction and the body of the NUREG (e.g., paragraphs 1 and 2 of Section 1.5) should be revised accordingly. We suggest the Supplement to IE Bulletin 79-01B provides some clarification in this area.

Along the lines indicated above, analytical approaches to determine HELB environment should be clearly identified. HELB outside containment, in some cases, is calculated in a different manner from HELB inside containment. Longer time frames and multicompartment steam migration can be considered, and accordingly, different computer codes are often used.

COMMENT NO. 42B: The acceptable methods referred to in Appendix A are all used for PBIC analysis of DBA LOCAs with ECCS. These may not be reasonable for equipment qualification purposes, especially outside containment.

Resolution

The staff concurs; the techniques to calculate the environmental parameters should employ plant specific models reviewed and approved by the staff. The reference to Appendix A for outside containment qualification purposes will be modified in the final rulemaking. (See also staff response to Comment No. 41.)

COMMENT NO. 43A: (Section 2.1(2)) The statement in Section 2.1(a) is not strictly true. It should be changed to read as follows:

Second sentence - "Experience...without test data may not be adequate..."

Third sentence - "In general,...size limitations, (b) ..., (c) capability to perform the required function can be readily analyzed (such as mechanical support, simple conductivity, etc.), and (d) especially in aging, where components or devices can be shown not to be limiting to the overall performance of the function."

As written, this section is far too narrow and restrictive. For example, in aging a piece of equipment which contains many materials it is sufficient to eliminate those not affected using analysis (evaluation of activation energies) in order to determine which materials are controlling by being most susceptible to aging effects.

COMMENT NO. 43B: The statement that, "in general the staff will not accept analysis in lieu of test data...," imposes a severe limitation on the industry that is not present in IEEE 323-1971 and 1974. IEEE 323-1971 states that, while type tests are *preferred, other methods may be used "when size or other* practical requirements limit or preclude type tests." At this late date, the Commission, by its action in NUREG-0588 proposes the deletion of a phrase from the standard, thereby invalidating a great deal of work that has been done and accepted up to this point in time.

We believe those sections of the NUREG dealing with aging, margin and qualification by analysis should be revised to reflect the standards as they are now written and have been interpreted since their issuance.

Resolution

See staff response to Comment Nos. 46 and 51. With regard to "other practical requirements" that may limit or preclude type testing, the staff has stated (see "Discussion") that alternatives (or exceptions) to the interim positions may be proposed and, if justified, may be found acceptable. These exceptions should be identified and evaluated on a case-by-case basis.

<u>COMMENT NO</u>. 44: (Section 2.1(3))

It is not clear whether the term "safety margins" is intended to be the same as the term "margin" used in IEEE 323-1974 or a new undefined term. If the former, delete "safety;" if the latter, it should be defined.

<u>Resolution</u>

The staff agrees with the comment. The term "safety margins" is intended to be the same as the term "margin" used in IEEE 273-1974.

<u>COMMENT NO</u>. 45: (Section 2.1(3))

With respect to the last sentence, does the term "operability" mean safety function?

To what factors or conditions should "safety margin," as used in the last sentence, be applied?

<u>Resolution</u>

Operability refers to assuring that the performance characteristics of the equipment that are necessary to perform a safety function are satisfied, including but not limited to accuracy, response time, and so forth. Test margin should be applied as described in Section 3.0 of the document.

<u>COMMENT NO</u>. 46: (Section 2.1(3)

The need to qualify non-safety-related equipment by test is overly stringent. In many cases, analysis is adequate to determine whether or not a failure mechanism exists which can result in reduced plant safety. The wording should be changed to, "...should either be shown by analysis to not fail in a manner detrimental to plant safety or be qualified to demonstrate such capability."

<u>Resolution</u>

The staff concurs in part; clarification of this requirement will be considered in final rulemaking. The staff maintains, however, that for active electrical equipment subjected to a DBA environment, type testing is the preferred qualification method. Other methods may be justified and will be evaluated on a case-by-case basis.

<u>COMMENT NO</u>. 47: (Section 2.1(3))

This section (Section 2.1(3)(c)) deals, in part with qualification of equipment that has been shown and justified to be unrelated to accident mitigation or plant safety. This goes beyond the bounds of IEEE 323-1971 and 1974 which state that they are for the qualification of Class IE equipment not non-safety-related equipment. Therefore, the requirement to qualify non-safety-related equipment should be deleted.

Resolution

Equipment that has been shown and justified to be unrelated to accident mitigation or plant safety is exempt from qualification.

COMMENT NO. 48: (Section 2.1(3)) The inclusion of non-safety-related equipment in this section is incompatible with the scope of this document. Furthermore, the effects and consequences of adverse environments on non-safety-related equipment has been raised as a Category I item on NUREG-0585 "TMI-2 Lessons Learned Task Force Final Report." We strongly recommend that the staff delete this requirement under NUREG-0588 to permit orderly resolution of this generic issue.

Resolution

The position addresses designs where equipment or systems have been incorrectly classified as non-Class IE strictly on the basis that they did not have to perform a specific safety function (such as actuation). The designs may have not factored in the broader function of determining whether or not their failure or improper actuation could also effect safety. It is the staff's intent to assure that this broader functional scope is factored into the design and as such, the qualification of some previously classified non-Class IE equipment may be warranted. It is not the staff's intent to require qualification of all non-safety-related equipment. (See also the staff response to Comment 4 and Comment 47.) Regarding the second point in the comment, the staff does not agree with the recommendation to delete this requirement from NUREG-0588. Resolution and implementation of the generic issue identified in NUREG-0588 are independent of the requirement stated herein.

COMMENT NO. 49: (Section 2.1(3)) This paragraph (Section 2.1(3)(c)) should be clarified to indicate applicability to safety-related equipment only. Non-safety-related equipment is not environmentally qualified unless it falls into Category 2.1(3)(b).

Resolution

The staff concurs. NUREG-0588 is only applicable to safety-related equipment.

COMMENT NO. 50: (Section 2.1(3)) Paragraph 2.1(3)(c) on page 11 seems to suggest "qualification" requirements for non-IE equipment, which is beyond the subject of this staff position document.

COMMENT NO. 50B: This position (Item 2.1(3)(b)) requires the qualification of non-safety-related equipment to show that the equipment will not fail in a manner detrimental to plant safety. This requirement does not fall under the scope of Environmental Qualification of Safety-Related Electrical Equipment. This requirement is being addressed elsewhere under such headings as "Systems Interaction" and "Consequential Failure" and in other documents such as NUREG-0578, and NUREG-0660. The overall program of "System Interaction" is very large and must be part of a long-term program as defined in some of the references mentioned above. The ultimate results of this long-term program may have some impact on the environmental qualification of safety-related electrical equipment but this should not be forced into NUREG-0588. It is recommended that the reference to non-safety-related equipment be deleted from this paragraph since these studies are addressed elsewhere and a parallel review under equipment qualification would detract from the remainder of the qualification program and would not add to plant safety.

COMMENT NO. 50C: The word "qualified" in this section (Section 2.1(3)(c) presents problems. We do not "qualify" non-Class IE equipment. We recommend deletion of both paragraphs of this section as, otherwise, we may have to obtain documentation for items of no safety significance.

COMMENT NO. 50D: It should be clarified that this paragraph (Item 2.1(3)(c)) applies only to safety-related equipment.

Resolution

Refer to the staff response to Comment Nos. 46, 47, 48, and 49.

COMMENT NO. 51A: (Section 2.1(3)) The requirement to qualify such equipment by test only is incompatible with the alternatives recognized under paragraph 2.1(2).

Therefore, delete the words "by test" from Section 2.1(3)(a) and the words contained in brackets in Section 2.1(3)(b).

COMMENT NO. 51B: Methods of qualification other than type testing should be applicable to item 2.1(3)(a)(b) also.

COMMENT NO. 51C: The requirement to demonstrate by test that the equipment will not fail in a manner detrimental to plant safety should be expanded to allow demonstration by analysis as well as test.

COMMENT NO. 51D: The words "qualified by test" should read "qualified by test or analysis." Otherwise, the testing program would expand considerably to no apparent benefit, expecially for non-safety-related materials and equipment.

Resolution

For electrical equipment located inside or outside containment that may be exposed to high energy line breaks (for example, LOCA, MSLB, feedwater line rupture), analysis alone is generally inadequate to demonstrate functional operability such as accuracy or response time, or to verify seal integrity (as in connectors), or even to detect intermittent or spurious failures. Although some analysis may be used when the testing is the principal qualification method, that analysis should be limited to extrapolations of data or to analyzing similarities in equipment or materials. In either case, analytical assumptions should be verifiable or supported by test data.

Recognizing the complex interaction of the environment on materials and equipment (such as aging or simultaneous vs. sequential effects) the staff does not agree that analysis by itself is an acceptable alternative for qualifying equipment required to function in the above-mentioned hostile environments. (See exceptions in Section 2.4.)

<u>COMMENT NO</u>. 52: (1) The implication in this section (Section 2.1(3)(a)) is that equipment that must function at any time during an accident must be shown to be capable of operating for the entire duration of accident conditions. This fails to differentiate between those items that must function throughout the accident and those that must perform some specific task at a given point in time during the accident. This paragraph should be changed to reflect these different classes of items.

<u>Resolution</u>

Refer to Section 3(4) which addresses this issue.

<u>COMMENT NO</u>. 53: <u>(Section 2.1(3))</u> We interpret the equipment referred to in these sections to be that which is subjected to the environment of a LOCA or MSLB.

<u>Resolution</u>

The interpretation is correct in part. Environments caused by other high energy line breaks such as feedwater line rupture should also be considered.

Section 2.1(3) was modified for clarity.

<u>COMMENT NO</u>. 54A: <u>(Section 2.1(3))</u> Throughout Section 2.1, reference is made to "accident" and "DBA." These terms should be defined. Comments pertaining to this section are predicated on the assumption that these terms mean LOCA or MSLB.

<u>COMMENT NO</u>. 54B: The term "event" (in Section 2.1) is not obvious. This should not be clarified (i.e., LOCA, MSLB, etc.).

<u>COMMENT NO</u>. 54C: Introduction of term "DBA" at this point (Section 2.1(4)) is inconsistent as it has not previously been used nor defined and its meaning is not clear.

Resolution

Accidents in this category include the complete spectra of break sizes for loss-of-coolant accidents (LOCAs) and for other high energy line breaks (HELBs) such as main steam line breaks (MSLBs) or feedwater line breaks.

For a listing of accidents required to be analyzed, refer to the Standard Review Plan Section 15.

The term "event" refers to occurrences such as natural phenomena, including but not limited to earthquakes and flooding resulting from other than pipe breaks. "Event" also refers to occurrences such as loss of ventilation which may occur as a result of a single failure.

COMMENT NO. 55A: (Section 2.2(12)) If only a source for the simulation of gamma is to be recommended in Section 2.2(12), and not a source for beta simulation, we recommend deletion of the item to avoid possible misunderstanding that Co-60 is acceptable for simulation of all radiation.

COMMENT NO. 55B: On page 13, item 1.1(12), from a commercial standpoint, it is encouraging to see that NRC considers cobalt simulation of the radiation environment; it has not yet been technically accepted and is in fact being questioned by WG 2.6 of IEEE/NPEC/SC-2. Unless the NRC is privy to some information that is unavailable to the rest of the industry this item should be removed. Furthermore, it implies that acceptable accounts for radiation can be considered by cobalt simulation, which case has just not been demonstrated and is also being pursued by the above mentioned WG 2.6.

COMMENT NO. 55C: In Section 2.2(12) sources other than Cobalt-60, e.g., Cesium-137, should be acceptable as qualification sources.

Resolution

The staff currently has a research effort with Sandia Laboratory to investigate the adequacy of qualifying equipment for both gamma and beta radiation environments by using only a gamma radiation source. While the results are very preliminary, there does not seem to be any significant problem in using only a gamma source to qualify certain types of equipment for a beta/gamma environment provided the gamma dose rate during the qualification tests is consistent with the expected beta and gamma dose rates (energy deposition rates) during the LOCA. It appears therefore that a gamma source (only) may be used for qualification testing, provided an analysis or test data indicates that the dose and dose rate produces damage similar to that which could be produced under accident exposure (i.e., combined gamma and beta environment), or a beta and gamma qualification dose and dose rates may be determined separately and the testing may be performed using both a beta and a gamma test source. The staff notes that the research effort is still continuing and that the preliminary findings may change, but until such time as other evidence is presented, the use of either Co-60 or Cs-137 for equipment qualification would seem appropriate.

COMMENT NO. 56: (Section 2.2(1))

The requirement to establish "failure criteria" would appear to be an unfortunate choice of words in that it imposes an unbounded set. We recommend changing to "acceptance criteria" as being not only more practical but more correct. Note the conflict with Appendix E, item 4.

Resolution

The staff concurs. Any modifications to the positions will be considered during final rulemaking.

COMMENT NO. 57A: (Section 2.2(4))

By stating a preference for an environmental profile that envelopes any design basis event there is a strong implication that equipment is to be designed to withstand more than one event. We know of no such requirement, and suspect that equipment exposed to the environment of an MSLB would not be allowed to be returned to operation without extensive inspection and, perhaps, refurbishing. While such enveloping should certainly be allowed, we fail to see the technical basis for the preference (especially since experience with NRC's "preferences" is that they eventually become requirements). There is an additional problem with the "preferred" approach when one considers the margin.

Current practice of doubling the number of transients would mean that a given equipment would have to be designed to withstand two MSLB events and two LOCAs. This may be excessive. We recommend the following wording: "...located inside containment, a single profile may be used that envelopes the environmental...loss-of-coolant accidents, but in any case, the equipment shall be shown to operate correctly under the environmental conditions of any design basis event for which it must perform a safety function."

COMMENT NO. 57B:

Use of separate profiles for LOCA and SLB should remain an acceptable option. While it may be convenient to test with one profile, the test is unnecessarily more severe than separate profiles.

COMMENT NO. 57C:

Requiring a single profile to envelope the worst case environmental conditions is neither practical nor realistic in terms of previously established acceptable qualification test methods. This procedure implies that one item of equipment could be subjected to both a LOCA and an MSLB, and this is not part of any accident analysis scenario nor is it consistent with previously acceptable practice.

In addition, in order to comply with the margin application requirements of IEEE 323-1974 the question of margin on peak transients is raised with respect to the number and severity if indeed this combined profile is to simulate a LOCA and MSLB case which would result in four transients at elevated temperature.

In order to expect equipment to operate successfully during this type of test it would almost certainly require substantial redesign and retesting to new conditions which would obviate the usefulness of any prior test performed.

This item should carefully be re-thought and substantially revised to allow the continuation of past type-test practice.

Resolution

There are components and equipment inside containment that are important to safety and are required to operate in both a LOCA or an MSLB environment. There may also be equipment or components (such as cable, penetrations, connectors, valves) that may not be required to perform a specific function but whose failure or inadvertent operation in a LOCA or an MSLB environment may be detrimental to safety. For such equipment, although a single bounding profile used to qualify the equipment is prefereable, other envelopes used in testing for a LOCA and an MSLB, either separately or sequentially, may also be used. The staff's preference for a single bounding envelope is to minimize the review effort by reducing the documentation and the analysis that would be required to demonstrate qualification to both environments.

With regard to margins, the staff considers that exposing the same component or equipment to a combined or sequential LOCA and MSLB envelope is sufficiently conservative to justify omitting the additional requirements of doubling the number of transients. These options will be evaluated on a case-by-case basis.

COMMENT NO. 58A: (Section 2.2(6)) We believe this to be an incorrect requirement that appears to miss a fundamental concept of qualification. The equipment is to work in the required environment and the qualification test should so demonstrate. The actual real time temperature of any portion of the surface boundary of the equipment is of no consequence in meeting this requirement (except to the equipment designers) and, may be misleading and non-conservative. We recommend the following wording:

> "The temperature to which equipment is tested to demonstrate qualification shall be measured and recorded throughout the test. The thermal capacity of the environment simulation shall be shown to provide an adequate simulation."

COMMENT NO. 58B: Suggest the NRC should keep away from designing tests by requiring thermocouple readings. It would be sufficient to request that the component temperature be determined by suitable means. The temperature to which equipment is qualified does not have to be defined as the surface temperatures but on the basis of its ability to perform as specified in a bulk ambient environment expected to occur as defined for the design basis. Whether or not the surface temperature ever reaches this value is immaterial, particularly with respect to short-term high peak temperatures, such as could occur for an HELB.

<u>COMMENT NO</u>. 58C: In general, surface temperature is not monitored directly during testing. Instead, the ambient air temperature at various locations within the test chamber is monitored. We assume the implication behind requesting measurements of surface temperature is to ensure that the device has stabilized at the test temperature prior to timing its exposure. If so, revise this section to so state the above. If not, revise Section 1.2(5) to clarify the reason for requesting component surface temperature to be monitored.

<u>Resolution</u>

The staff agrees with the intent of the comment. It should be noted that the objective of the position is to ensure, by independent verification, that the equipment or component was exposed to the bulk temperature equivalent to or more severe than that temperature assumed in the bounding envelope derived from the accident analysis. Temperature sensors (not necessarily limited to thermocouples) located only on the inlet piping of the test chambers may not be indicative of the bulk temperature at the component being tested.

The intent is to ensure that temperature sensors are located as close as practical to the components being qualified.

It may also be prudent to provide temperature sensors that in addition to monitoring bulk temperature would also monitor the surface temperature of the equipment. This would facilitate the comparative studies discussed in Section 1.2(5)(b) of the NUREG. Without these readings, the use of the more conservative comparison to the "bulk" LOCA test temperature would be warranted. See also staff response to Comment No. 9.)

<u>COMMENT NO</u>. 59: (Section 2.2(5)) Full duration testing for extended periods of submergence is impractical and unnecessary. Short duration testing to demonstrate seal integrity plus an addressment of potential corrosion mechanisms by test or analysis are adequate. The following change to item 2.2(5) is therefore, recommended:

The sentence:

"Where equipment could be submerged, it should be identified and demonstrated to be qualified by test."

Should be changed to:

"Where equipment could be submerged, it should be identified and demonstrated to be qualified by test to demonstrate seal integrity. The effects of corrosion mechanisms for the duration required should be addressed by test or analysis."

Resolution

The staff concurs in part. Shorter test periods and analytical extrapolation may be found acceptable if adequately justified. This justification should be included as part of the qualification documentation. Analysis by itself, however, may not be adequate see staff response to Comment No. 51 for additional information.

COMMENT NO. 60A: (Section 2.2(9)) There appears to be a conflict between this requirement and that of 2.2(7) in that (7) allows periodic verification of operability and (9) requires continuous verification with justification necessary for periodic. This conflict should be resolved.

COMMENT NO. 60B: The requirement for continuous monitoring seems inconsistent with the requirement for periodic performance verification stated in 2.2(7). The same thing is being required differently. This should be revised.

COMMENT NO. 60C: These paragraphs appear to be requiring in-plant testing for qualification acceptance. The paragraph should be rewritten to clearly identify that this is not the staff's position.

COMMENT NO. 60D: Does continuous monitoring of equipment operability status mean that equipment is to be exercised throughout the test (e.g., coils energized, motors energized...)? If so, the statement is appropriate when actual environmental conditions are simulated. However, if accelerated aging temperatures are being used, the operability should only be checked at discrete intervals with components at anticipated ambient conditions.

Resolution

The intent of Section 2.2(9) is to ensure that intermittent failures in equipment--such as momentary change of state of bistables (that is, contact chatter), a cyclic variation in a transmitter output or a valve position variation--have been accounted for in the qualification testing program. Where intermittent failures in equipment can negate a safety function, the test program should include provisions to monitor selected parameters on a continuous basis in order to detect these failures (if any). It is recognized that certain equipment requires long-term testing (for example, postaccident monitoring equipment) where around the clock monitoring is difficult to accomplish. For this category of equipment, continuous monitoring for spurious or intermittent operation during periodic intervals may be justified.

COMMENT NO. 61: (Section 2.2(8)) The application of spray at the maximum ambient condition is unrealistic because during actual initiation in a plant it is at much lower temperature prior to any recirculation. This condition has generally been simulated during type-testing and has also been shown to produce the same results where spray has been at elevated temperature, which in itself is difficult to attain during type-testing and is not necessary.

Resolution

Chemical spray ingress (if any) is one area of concern addressed by the position. Pressure is considered a driving force influencing ingress into vital components through materials such as seals, jackets, and so forth. It is therefore prudent during testing to ensure that chemical (or demineralized water) sprays are introduced at or as close to the simulated maximum containment peak pressure conditions (if not already introduced before the maximum peak pressure conditions are reached).

COMMENT NO. 62: (Section 2.2(8)) Add words, "where applicable" to this article.

Resolution

The staff does not agree. See staff response to Comment No. 15.

COMMENT NO. 63A: (Section 2.2(10)) This is only applicable when the design range of voltage and frequency is significant. For Class IE devices fed from a guaranteed stabilized power source, such a demonstration is unnecessary. The following change to item 2.2(10) is, therefore, recommended:

> "The aspects of the expected extremes in power supply voltage range and frequency need only be considered during simulated event environmental testing if there is a significant design range for these parameters."

COMMENT NO. 63B: When would it be required to demonstrate performance under expected extremes in operating characteristics? This has been delineated for valve actuation in IEEE 382-1980 and its omission permitted by suitable justification to demonstrate that it does not upgrade the equipment's ability to perform its specified safety function.

COMMENT NO. 63C: If simulated event environment is accelerated, then voltage and frequency ranges should be applied at discrete intervals with components at anticipated ambient conditions.

Resolution

The loss of offsite power is assumed concurrently with a design basis accident (as in the case of a LOCA, MSLB, and so forth). As a result of sequencing the loads onto the diesel generators, power and frequency variations will be sensed on selected equipment such as valves, motors, and relays, and may affect their performance characteristics (for example, response time) and negate their safety function. If equipment can sense these effects, these variations should be accounted for in the test program.

There are, however, designs where the power supplies remain unchanged (because, for example, of the instantaneous availability of backup power supply systems). In those cases, exceptions to this position would be justified.

COMMENT NO. 64: (Section 2.2(10)

This position is too binding and does not allow analysis to be considered to establish most critical input conditions. Also, simulation of under-voltage and/or frequency is applied during seismic testings and is considered more severe.

Resolution

This position does not exclude the use of analysis to establish critical input conditions. See staff response to Comment No. 51.

Regarding the second point, it is not evident that under-voltage and/or under-frequency simulation during seismic testing is always more severe than in other hostile environment conditions. Where this is the case, the basis for excluding such testing under these other hostile environment conditions should be provided and documented in the qualification reports.

COMMENT NO. 65A: (Section 2.2(11))

Rather than addressing "dust" in qualification, it would make more sense to improve cleanliness requirements on plant operations. We recommend deleting this non-quantitative item.

COMMENT NO. 65B:

Dust has not been considered in the environmental specification. This is a potential major change for IEEE 323. Technical justification for its inclusion must be supplied. Rather than addressing "dust" in qualification, it would make more sense to improve cleanliness requirements on plant operations. We recommend deleting this non-quantitative item.

COMMENT NO. 65C:

We disagree that this should be a "service condition" specified in the qualification programs. The dust accumulation is primarily a function of housekeeping. If any special cleaning requirements are necessary in order to ensure operability of the equipment, it should be addressed in the operating and maintenance requirements of the equipment and not the qualification service conditions. It should not be the intent of a qualification program to address all possible service conditions that could occur if normal maintenance is not performed.

COMMENT NO. 65D: The paragraph requires that "dust environments" should be addressed when establishing qualification service conditions. NRC should delete or be more definitive.

Resolution

The staff agrees in concept with the comment.

It is not the staff's intent to require quantitative testing to ensure equipment operability in dusty environments, but rather to highlight a potential failure mechanism. Equipment susceptability to dust should be considered when qualifying safety-related equipment and be accounted for in the interface requirements via, for example, in improved periodic maintenance, or by the use of protective covers. The staff is currently in the process of rulemaking and will consider the recommendations expressed in the above comments, in the "Final" position.

COMMENT NO. 66:
(Section 2.3(1))

The statement that "the test procedures should...accident environment" is in conflict with the recommendation that the test sequence should conform fully to the guidelines of Section 6.3.2 of IEEE 323-1974. The conflict results from Section 6.3.2(3) permitting the operational performance extremes test to be completed on other, essentially similar equipment. It is, therefore, recommended that the identified sentence be deleted as being inaccurate and redundant.

Resolution

The implementation of Section 6.3.2(3) of IEEE 323-1974 (or the staff's position) establishes a data base during normal environments which should provide a comparison of the performance characteristics at the more severe environments. The staff agrees with the statement in the standard that if a data base is available from other tests "on identical or essentially similar equipment," then there is no need to repeat a test to establish a redundant set of performance characteristics at a normal environment. However, caution should be taken in using data from other than identical equipment, so that extrapolation of data is indeed valid. When exposing equipment to hostile environments, the same piece of equipment should be used in sequence see resolution to Comment No. 80). The staff does not agree that Section 2.3(1) is in conflict with Section 6.3.2(3) of IEEE 323-1974 but does recognize that justified exception may also be found acceptable.

COMMENT NO. 67:
(Section 2.3(1))

IEEE Standard 323-1974 permits deviations from the recommended test sequence providing adequate justification can be provided.

Resolution

The staff agrees with the comment. See staff's response to Comment No. 15.

COMMENT NO. 68:
(Section 2.3(2))
(Category II)

This is incompatible with item 2.2(2). The following change is, therefore, recommended:

(2) "The test should simulate as closely as practicable the combination of postulated environments necessary to meet the requirements of subparagraph 2.2(2)."

Resolution

The intent of the above-referenced section is to ensure that all environmental service conditions expected to occur would be enveloped. Any apparent incompatibility will be corrected during the final rulemaking.

COMMENT NO. 69: (Section 2.3(4)) (Category II)

Separate effects testing may have been done on penetrations, etc. It may be very difficult to retest such equipment. This requirement should be revised to a "best efforts" basis.

Resolution

Justification for the adequacy of the sequence used should be established and provided as part of the qualification documentation. Exceptions, if justified, may be established and will be evaluated on a case-by-case basis. If the adequacy of the qualification method can not be justified, retesting or equipment replacement may be warranted.

COMMENT NO. 70A: (Section 3(1))

The paragraph calls for margin on margin. Presumably, the staff requires demonstrable margin with respect to accident parameters which have been established employing a calculation model acceptable to the staff. The following change to item 3(1) is therefore, recommended:

(1) "Qualification margins should be applied to the design parameters discussed in Section 1, which are established employing a calculation model acceptable to the staff, to assure that the postulated accident conditions have been enveloped during testing."

COMMENT NO. 70B:

The application of margin in addition to the margin applied during derivation of the service conditions would be doubly redundant and not necessary if the previous margins have been quantified.

COMMENT NO. 70C:

There is no technical basis for the summary dismissal of margins just because they are part of the plant parameters rather than just of the test parameters. It has always been the intent of IEEE 323-1974 that margin need not be added if it can be shown that adequate margin is already included in the environmental requirements. The position taken in 3(1) is not consistent with the position in 1.4 which states that additional radiation margins are not required if certain procedures are followed.

Resolution

The staff is in agreement that additional margin need not be added if it can be shown that adequate margin (to account for uncertainties identified in IEEE 323) is already included in the environmental requirements. Although claims are made that these margins are included in the calculated envelopes, experience has shown that those margins may not be adequately quantified to facilitate independent verification.

In general qualified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident environmental conditions have been enveloped. The margins should (1) account for uncertainties associated with the use of analytical techniques in deriving environmental parameters; (2) account for uncertainties associated with defining satisfactory performance (e.g., when only a small number of units are tested) (3) account for variations in the commmerical production of the equipment and (4) account for the inaccuracies in the test equipment to assure that the calculated parameters have been enveloped. These margins should be provided in addition to any conservatisms applied during the derivation of the specified plant parameters unless these conservatisms can be quantified and shown to contain sufficient margin. It is the staff's belief that when the temperature and pressure conditions are derived using the methods identified in Section 1.1(2) or the qualification envelope in Appendix C is used, or the radiation methodology described in Appendix D is used the only additional margins to be provided are those accounting for the inaccuracies in the test equipment. Sufficient conservatism has already been included to account for the uncertainties identified in (1) through (3) above.

COMMENT NO. 71A: (Section 3(1),(2)) It appears that three levels of margin are to be employed. The first is that applied during the derivation of plant conditions. The second would be for accident conditions to ensure enveloping postulated accident conditions, and the third would be in accordance with Section 6.3.1.5 of IEEE 323-1974 to account for normal variations in commercial production. Please confirm if the above understanding is correct. There is general concern in the industry regarding regulatory requirements resulting in the cascading of margins. In some instances this leads to unrealistic qualification testing parameters and results.

COMMENT NO. 71B: This position on margin (Section 3(1)) is excessive. The conservatism used in calculating specified plant parameters is a form of margin. Not allowing any credit for this conservatism will only force vendors/engineers to go back and recalculate these parameters in a less conservative manner in order to establish more realistic qualification plans or to validate tests already completed. Such work will do nothing to better show the adequate qualification of electrical equipment.

It is recommended that the second sentence from paragraph 3(1) be dropped. In lieu of this sentence, it is recommended that the following be added to the end of paragraph 3(2):

"Normally, margin is applied to the specified plant parameters; however, credit may be taken for the conservatism applied in calculating specific plant parameters if this conservatism, along with whatever other margin is applied, is shown to provide an overall adequate level of margin."

Resolution

See the staff's response to Comment Nos. 70 and 73.

COMMENT NO. 72A: (Section 3(1)) (Category II)

For Category II, the requirement ("Same as Category I") is in conflict with the requirement for Category II on aging, in Section 4. Furthermore, since IEEE 323-1971 did not require margin, and since many pieces of equipment have operated satisfactorily for a long time with qualification that did not include margin, it may be counter to safety to require such equipment to be replaced because the test data did not include margin.

COMMENT NO. 72B:

On page 14, Category II, item 3(1) -- "Margin was not a requirement for qualification under the guidelines of IEEE 323-1971 but its incorporation is now required by reference to the requirements for Category I applicability. This could have significant adverse impact on the acceptability of the qualification process particularly if it is determined under the present ground rules that the service conditions have to be changed when the original methodology used during the FSAR preparation for the plant licensing basis was considered adequate. Remove from Category II this requirement to implement margin per the IEEE 323-1974 groundrules."

COMMENT NO. 72C:

Item 2.2(1), 3(1) -- Since the application of margin was not a requirement of IEEE 323-1971, it may not be possible to demonstrate margin in all cases. In such cases, lack of documentation demonstrating qualification margin should not constitute unacceptability for Category II plant equipment.

COMMENT NO. 72D:

Paragraph 2.2(2) suggest change "...all service conditions postulated (with margin, see Section 3.0) during..." for clarity.

Resolution

Qualification documentation should clearly show that the environmental parameters (to which the equipment may be exposed) have been adequately enveloped. If no margin can be claimed per Section 3.2 of the NUREG (Category II), the adequacy of the design is considered questionable. (See also response to Comment No. 7.)

COMMENT NO. 73: (Section 3(2))

The Nuclear Power Engineering Committee has taken the following position to clarify the intent of the margin requirements in IEEE 323-1974. We feel it would be of use to the industry if this clarification were included in NUREG-0588.

"IEEE 323 requires margins and suggests considerations increasing the test level, increasing the number of test cycles, or increasing the test duration as methods of assuring margin. It was the general consensus that while IEEE 323 tends to promote that all three be used, there are situations where it could be demonstrated that one higher transient is equivalent to or more conservative than, for example, two lower level transients, etc. The choice is, therefore, up to the user and depends upon the type of equipment, method of test, etc. In any event, the user should justify his method."

Resolution

The staff supports the Nuclear Power Engineering Committee position on margins and considers the comment an amplification of the staff positions identified in the NUREG (specifically, Section 3(2) of Category II).

COMMENT NO. 74: (Section 3(3)) The specific reference to inaccuracies in the test equipment should be provided so that the required application of margin can be correctly implemented in the test program.

Resolution

Margins to account for inaccuracies in the test equipment should factor in the accuracy tolerance of the sensors used to monitor the test conditions (for example, pressure or temperature sensors). These margins should be added to the test profiles to ensure that the calculated environments have been enveloped. For example, if the maximum temperature to be sensed is 300°F and the sensor can be in error by 5°F at that value, then the indicated temperature during the test should not be less than 305°F to ensure that maximum conditions have been simulated.

COMMENT NO. 75: (Section 3(1)) For new designs, quantified margins should be applied to the design parameters discussed in Section 1 to assure that the postulated accident conditions have been enveloped during testing. Where existing designs are being qualified for a new application, margin should be applied as a difference between service conditions and design limits.

Resolution

Margins should be applied to account for test, production, and analytical uncertainties that are identified in IEEE Standard 323 and NUREG-0588, independent of their design chronology.

COMMENT NO. 76A: (Section 3(4)) There is no technical basis for the application of an arbitrary margin of one hour (in Section 3(4)). The Reactor Protection System, for example, is designed to operate in terms of milliseconds. As long as subsequent failure can be shown to not undo the safety action already taken, there is no need to require survival for some totally arbitrary period.

COMMENT NO. 76B: Westinghouse is totally opposed to the arbitrary application (in Section 3(4)) of an additional one-hour time requirement in excess of the calculated worst-case time required to perform the safety function as derived from accident analysis. Implementation of this requirement will negate extensive qualification testing already completed by industry and, furthermore, will severely impact qualification test schedules established for the lead plants committed to IEEE 323-1974.

The staff has indicated that this requirement has arisen from concerns over earlier transmitter tests where failure of some units was noted after a few minutes. Thus, Westinghouse recommends that the sentence:

"Equipment in these categories is required to remain functional in the accident for a period of at least 1 hour in excess of the time assumed in the accident analysis."

Be changed to:

"Equipment in these categories is required to remain operable, in the accident environment, for a period of at least 1 hour in excess of the time assumed in the accident analysis. The equipment performance during this additional 1 hour shall be shown not to negate any prior completed automatic safety functions or, in the case of equipment required for post-accident monitoring, provide misleading information to the operator."

COMMENT NO. 76C: The environmental standard does not imply that equipment should be functional in the accident environment(s) for a period of at least one hour, as required by the staff position. This requirement should be removed. There is no technical basis for the application of an arbitrary margin of 1 hour. The Reactor Protection System, for example, is designed to operate in terms of milliseconds. As long as subsequent failure can be shown to not undo the safety action already taken, there is no need to require survival for some totally arbitrary period.

COMMENT NO. 76D: The "1-hour minimum operability time" following the DBE is a new requirement and will impact present and previous equipment programs. It also is over and above that required by paragraph 2.1(3) and Appendix E, Section 2.

Adding a 1-hour operability requirement to equipment qualification will discourage additional transducer suppliers, whose equipment is designed to function quickly for safety purposes. The "margin" defined in IEEE 323-1974 appears sufficient.

COMMENT NO. 76E: The margin requirements (in Section 3(4)) are excessive for equipment intended to function for less than one hour in an accident environment. A more appropriate margin would be based on a percentage increase above the operability requirements.

COMMENT NO. 76F: This position (in Section 3(4)) states that equipment which is required to only perform its safety function within a short period into the event (i.e., within seconds or minutes) is required to remain functional in the accident environment for a period of at least one hour in excess of the time assumed in the accident analysis. We feel that this qualification requirement is unnecessary for this type of equipment.

COMMENT NO. 76G: The staff should document the concern being expressed to allow industry the opportunity to develop alternative ways to resolve the issue.

COMMENT NO. 76H: This position (in Section 3(4)) will impose harsh requirements on equipment qualification without improving plant safety. Basically this position requires a minimum of one-hour margin on the required operating time for safety-related electrical equipment. This is a harsh requirement on instrumentation which must function early in a LOCA or an MSLB with specific accuracy. Qualifying this equipment with some reasonable margin (including time) is difficult but feasible. Maintaining the required accuracy for one hour beyond the specified time in an (LOCA/MSLB) accident environment will not improve safety. Such a requirement will only invalidate tests, disqualify equipment previously qualified and force users to obtain and qualify equipment (if available) in order to pass a test but not to improve plant safety. It is recommended that this paragraph be deleted or, as a minimum, be replaced with a paragraph that discusses the need to show adequate time margin for equipment with short specified operating times after a DBE.

COMMENT NO. 76I: Implementation of this requirement (Section 3(4)) will negate extensive qualification testing already completed by industry and, furthermore, will severly impact qualification test schedules established for the lead plants committed to IEEE 323-1974. As a minimum, a review of equipment capability to meet these revised requirements will be necessary prior to embarking on an expensive test program and, at worst, an equipment development program may be required to meet this arbitrarily imposed functional requirement. Tests and analyses of Category II equipment in some cases did not include a requirement to remain functional for at least an hour longer than assumed in the accident analysis.

This requirement should be considered on a case-by-case basis, especially for such items as isolation valves.

Resolution

For equipment subjected to hostile environments resulting from pipe breaks, an accepted practice is to qualify that equipment to the most limiting environment (which would envelope the less hostile environments caused by a range of different pipe breaks). Subjecting the equipment to the most severe portion of the hostile environment (maximum pressure, temperature, and radiation) for only a very short time period (seconds or minutes) does not provide adequate assurance that all the environmental service conditions have indeed been enveloped. It is the staff's belief that the additional one hour of demonstrated functional operability for equipment required to operate for only a short period (that is, less than or equal to 10 hours), provides for the most part, the assurance that the equipment will function in any accident environment that can exist during large and small line-break accident scenarios.

There may be some designs where less restrictive margins may be justified and found acceptable on a case-by-case basis (see Category II, Section 3(2)). The staff believes, however, that the general requirement of testing for an additional hour is warranted.

COMMENT NO. 77: (Section 4(1))

The staff position (in Section 4(1)) should be clarified to assure that the requirements are being applied to Class IE equipment only.

Resolution

See staff resolution to Comments Nos. 4 and 48.

COMMENT NO. 78: (Section 4(1)) (Category II)

In Section 4.1, the Category I position far exceeds those established in IEEE 382-1972 or IEEE 334-1971. Compliance to the provisions of these standards should be sufficient for Category II equipments. It is recommended to delete the last sentence of the Category II position 4(1).

Resolution

This area is under staff review. Any modifications to the staff positions will be included in the final rulemaking which is planned to be issued to public comment in December 1981. In general the staff does not require, for Category II plants, the same degree of rigor in the proof testing, analysis, and documentation as it does for Category I equipment. Recognizing the limitations in the state of the art in assessing synergistic effects, the position regarding synergisms for Category I is not applicable to Category II plants unless known synergistic effects have been identified on the materials that are in use in these older plants. With the exception noted above (synergisms), the aging positions identified for Category I are applicable for Category II equipment identified in Section 4(1).

COMMENT NO. 79A: (Section 4(2)) (Category II)

Section 4.2 requires an aging evaluation program be conducted and a periodic replacement schedule be established. This is of major impact. For this category of plants, the staff should specifically state what equipment has been shown susceptible to aging effects. One source could be the NPRD program. Trend studies could be conducted that point to equipment aging at an unacceptable rate. Periodic bulletins could alert the utilities and corrective action taken based on good data rather than engineering guesses. Requiring a reevaluation of aging effects in the Category II equipment is well beyond the licensing commitments.

COMMENT NO. 79B:

For Category II equipment, identification of materials susceptible to aging would require a long list compiled from literature of test data. Also, each manufacturer uses his own formulation and may be reluctant to release information. Going back to manufacturers, and particularly their subsuppliers, of equipment delivered several years ago will be extremely time consuming and probably inconclusive. The benefits, in the form of improvement in safety, do not appear to be commensurate with the potential effort required.

Resolution

As stated in the position of Section 4.2, the staff has and will continue to identify materials and or equipment that may be susceptible to deleterious aging effects. It is, however, incumbent on the user of the equipment (that is, the utility) to ensure that the equipment that has been identified by the staff and by others as being susceptible to significant degradation because of aging is properly accounted for. Data banks established by owners groups are one way of maintaining current information of specific equipment in use today. Ongoing programs should exist at the plant to review surveillance and maintenance records to ensure that equipment which is exhibiting age-related degradation will be identified and replaced as necessary.

COMMENT NO. 80A: (Section 4(3))

In Section 4(3), the term "investigation" with regard to synergistic effects is ambiguous. This could mean an experimental program or a literature search. In any case, the state of the art in aging to a single environmental stress is rudimentary at best. Requiring experimental studies of combined effects exceeds the existing technology. We recommend deleting the second sentence. We further recommend deletion of the last sentence which references partial results of questionable test programs until such time as they are completed, reviewed and verified.

COMMENT NO. 80B:

For item 4(3), it has not yet been determined whether or not synergistic effects (not defined in this NUREG) are necessary for consideration during any phase of the qualification program. Reference is made to a previously established position developed by the above-mentioned WG 2.6 and subsequently endorsed in full, by SC-2 and forwarded to NPEC. NRC should seriously take this statement into consideration before requiring the consideration of unknown in NUREG requirements.

Furthermore, the reference given in the NUREG to partial results of questionable test programs should be totally omitted as there is no technically sound basis for assuming that these results are definitive and represent the state of the art.

COMMENT NO. 80C: In item 4(3), the position concerning synergistic effects is contradictory to the state of the art as discussed in paragraph 4 of the introduction. Thus, Westinghouse recommends that this paragraph be deleted.

Consideration of synergistic effects is a new requirement for which there are no specific guidelines to apply to equipment involved in a qualification program, that including NUREG/CR-0276 and NUREG/CR-0401. Equipment that is properly qualified to the intent and requirements of IEEE 323-1074 demonstrates its ability to survive and perform its safety function.

The evaluations of synergisms for Class IE equipment appear to be more in line with an R&D program that introduces new equipment, but not in line with the qualification of equipment to the requirements of IEEE 323-1974. See our response to Reg. Guide 1.131 for additional detail.

COMMENT NO. 80D: In paragraph 4(3), synergistic effects should be considered in the accelerated aging programs where applicable.

COMMENT NO. 80D: In Section 4(3), to date, contractor qualification procedures have not included testing methods which would establish synergistic effects.

COMMENT NO. 80F: For item 4(3), this position on synergistic effects implies that every qualification report must include documentation to show that synergistic effects were investigated or that at least a document search was conducted. This is an artifical requirement. Synergistic effects are not "testing" parameters but are the subject of research projects. Even the existence of synergistic effects is questionable depending on how the data are evaluated in the limited research conducted thus far.

This position should be dropped or at least modified to say that synergistic effects need only be addressed where they have been identified. The following rewrite of this paragraph is recommended:

4(3) Synergistic effects should be considered in accelerated aging. Synergistic effects need only be addressed, however, if known synergistic effects exist for the materials of concern. See NUREG/CR-0276 (SAND 78-0799) and NUREG/CR-0401 (SAND 78-1452), "Qualification Testing Evaluation Quarterly Reports."

Resolution

The staff is aware that some equipment important to safety may contain materials whose aging effects from combined environments (applied either concurrently or sequentially) are more severe than the sum of the effects of each environmental parameter applied separately. Identifying the most limiting combination of environmental parameters in order to establish a qualified life through research programs, however, may be a long-term, on-going process. Therefore, in lieu of research programs, the qualification program should:

(1) Identify potentially significant synergistic effects through a literature search and account for those effects through testing or analysis when establishing a qualified life, or

(2) Establish through a literature search or operating experience the basis for omitting synergistic considerations.

For equipment where, for example, significant radiation and temperature environments may be present (and in lieu of contrary information determined through items 1 or 2), the synergistic effects to these parameters should be considered during the simulated aging portion of the overall test sequence. The testing sequence used to age the equipment (or material) should be justified and the basis documented in the qualification report. For equipment where thermal aging evaluation has been conducted prior to issuance of this document on non-irradiated equipment or materials, the adequacy of the assumptions made and the conclusions reached will be evaluated on a case-by-case basis. Other methods designed to address synergisms (such as ongoing surveillance with additional qualification testing) may also be found acceptable and will be evaluated on a case-by-case basis.

COMMENT NO. 81: (Section 4(4)) Arrhenius is presumably limited to addressment of thermal aging effects. The following change to item 4(4) is, therefore, recommended:

(4) "The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging. Other thermal aging methods tha can be supported by type tests will be evaluated on a case-by-case basis."

Resolution

The staff agrees. Any modifications to the staff position will be included in the final rulemaking which is planned to be issued for public comment in December 1981.

COMMENT NO. 82: (Section 4(4)) If the NRC considers that acceptable methods exist to address aspects of the qualification, then it would be most helpful if they provide examples similar to those given in the Guidelines for Operating Reactor Qualification (Ref. Denton to Stello, dated November 13, 1979).

Resolution

The examples given in the referenced document are compatible with NUREG-0588. Implementation of the examples should be conditioned on their applicability to the vintage plant, and is outside the scope of this document.

COMMENT NO. 83: (Section 4(4))

In Section 4(4), the Arrhenius equation can be linearized by assuming activation energies are independent of temperature. The linear equation can be used to derive an accelerated aging time by inputting an aging temperature, the desired component life, and ambient temperature. The accelerated aging parameters are then used to type test the component. An alternate approach is to cycle material samples at a number of test temperatures until failure occurs. The data are then used to form a linear regression as described in IEEE 101, "IEEE Guide for the Statistical Analysis of Thermal Life Data." The regression line can be extrapolated to determine a life based on an ambient temperature. Do these approaches meet the NRC's intent of using the Arrhenius methodology?

Resolution

The test procedures, and the assumptions used, should be evaluated on a case-by-case basis and may be found acceptable.

COMMENT NO. 84: (Section 4(4))

Paragraph 4(4) on page 16 speaks of "The Arrhenius methodology" regarding aging. It is suggested that a reference be given with a source of information on this methodology.

Resolution

Numerous references can be found in qualification publications. The reports identified in Comment No. 80 or the IEEE Standard 101-1072 referenced in Comment No. 83 also provide information on this methodology.

COMMENT NO. 85: (Section 4(5))

Item 4(5): This position requires that known phase changes of materials should be defined to ensure that no changes occur during accelerated aging. This is certainly a valid means of supporting an aging program. However, there are other means that are equally valid. For example, some equipment has undergone previous tests (such as UL tests) at elevated temperatures. If these test temperatures exceed the temperatures being used for accelerated aging, the previous test will provide sufficient evidence that no known changes will occur. To more clearly allow for such alternate methods, it is recommended that this paragraph be rewritten as follows:

"(5) Effects of temperatures used in accelerated aging and within the extrapolation limits must be considered to evaluate materials phase changes. Relevant phase changes should be shown not to occur by defining known phase changes, referencing previous testing, or providing other supportive evidence."

Resolution

The use of previous testing to support the claims that conservative extrapolation limits have been implemented in the qualification programs is acceptable, provided the materials used in previous tests are identical or sufficiently similar so that a comparison is valid. The position is generallyto allow such specific applications.

COMMENT NO. 86: (Section 4(6)) Endorsement of Arrhenius methodologies should be limited to thermal aging only. We agree that this method should be allowed for want of any better approach. Criteria for selecting conservative activation energies should be included for cases where multiple degradation phenomena are operative or where the activation energy is not known.

Resolution

For cases where equipment is composed of different material components having different activation energies, and testing each component separately is not practical, the testing of the equipment should be conducted using the most limiting (lowest) activation energy of the components.

COMMENT NO. 87: (Section 4(6)) We interpret Section 4(6) as being applicable to post-accident environmental thermal age acceleration also.

Resolution

For equipment that is required to function for an extended period of time in a hostile environment, postaccident environmental aging considerations may be warranted.

COMMENT NO. 88: (Section 4(7)) Section 4(7): The staff appears to be requiring that the plant design include procedures for subjecting the equipment to the limiting service environment conditions. Periodic testing of equipment subjected to the most limiting service environmental conditions would undoubtedly result in more rapid equipment aging. The necessity to perform such testing on components already qualified is questionable. This requirement should be removed.

Resolution

This position applies when the choice of qualification is on-going, in order to extend, verify, or provide a more realistic qualified life. It is the opinion of the staff that component degradation due to aging for the most part may not be readily detectable by visual inspection or testing at only the normal service conditions. However, in the hostile environments this degradation, if significant, should be readily apparent.

COMMENT NO. 89A: (Section 4(8))

For item 4(8), the exemption of humidity from aging considerations for cable should also apply to other insulating materials and particularly for cases where there is significant heat generation within the device to cause humidity reduction. The basis for this exemption should be stated so that other materials can be considered for exemption also. At the present time, there is no known aging mechanism in the electrical materials due to humidity and, certainly, no known method of accelerating this unknown mechanism.

4(8) "The exemption of humidity from aging considerations for cable should apply to any insulating material for which there is adequate justification."

COMMENT NO. 89B:

For item 4(8), the basis for exception to humidity effects on cable should be provided as it is not clear how this can be considered acceptable in a NUREG. There are certainly situations where the humidity effects should be accounted for, particularly if the performance requirements specify the need to demonstrate integrity of the insulation if brittleness could be a factor. This particular parameter should be considered in the same context as all other parameters even for cable.

COMMENT NO. 89C:

In item 4(8), this position is not clear and might be interpreted incorrectly. The effect of relative humidity on aging is discussed in neither IEEE Standard 323-1974 nor Regulatory Guide 1.89 of November 1974. This lack of discussion is valid, however, for relative humidity is not recognized as an aging mechanism. Relative humidity is recognized as an environmental parameter for equipment performance and should be addressed in that manner.

Your position that the "effects of relative humidity need not be considered in the aging of electrical cable insulation," is correct. However, the possible interpretation that relative humidity must be considered for all other electrical equipment is not correct. The following rewrite of the paragraph is recommended:

4(8) "Effects of relative humidity need not be considered in the aging of electrical equipment."

Resolution

The effects of humidity on equipment should be considered in the qualification program. Justification, however, may be established to limit the testing of selected equipment to the range and the duration of humidity environments expected at a plant site. A literature search of the tests conducted on identical or similar equipment (or materials) or operating experience may be used to establish a basis for not including rigorous humidity testing. As an example, the Sandia Laboratory report SAND 78-0344 (October 1978) on "Aging of

Nuclear Power Plant Safety Cables" provides assurance that humidity effects on the cable insulation materials tested is not a significant aging contributor. Therefore, for qualification of equipment using these materials, the aging effects due to humidity may be omitted. The basis for these exemptions, however, should be documented.

An NRC-funded research program is presently investigating aging mechanisms due to humidity and is developing methods to qualitatively assess these efforts on selected materials (reference NUREG/CR-1466, "Predicting Life Expectancy and Simulating Age of Complex Equipment Using Accelerated Aging Techniques," April 1980). The staff has not, however, endorsed any one specific method of accelerating humidity. At this time, various methods of accelerating humidity effects during the aging portion of the test program or humidity conditioning during a test sequence may be found acceptable.

COMMENT NO. 90: (Section 5(2)) (Category II)

The documentation requirement for Category II plants allows for exclusion of the format and content of the Standard question given in Appendix E and it is not clear why this is being done. The documentation requirements of 323-71 and suggestions of the previously mentioned Guidelines for Operating Reactors would seem to indicate the need to address these same items in a consistent form for both operating plants and those in the CP or OL phase. Otherwise, the documentation requirements of 323-74 should be followed exclusively, as they are considered sufficient.

Resolution

The information in Appendix E is applicable equally to Category II as well as to Category I plants. The only difference in the position is that Category II plants should utilize the documentation guidelines identified in IEEE Standard 323-1971, whereas Category I plants should utilize IEEE Standard 323-1974.

COMMENT NO. 91: (Section 5(2))

The requirement in Section 5(2) to require "test data" on each piece of complex and varied equipment, much of which could be qualified by extension to equivalent or identical pieces, would be extremely cumbersome and expensive to manage. Paragraph 3.0 of IEEE 334-1975 illustrates the difficulty that might be involved.

It appears sufficient for the last sentence to read in effect:

"...unless it is accompanied by information on the qualification program, including test data or comparable test data from equivalent equipment."

Resolution

The staff position does not exclude the use of data from tests conducted on similar equipment as long as independent verification of similarity or equivalence can be established. It is incumbent on the applicant to have the necessary documentation to justify the adequacy of using data from similar or equivalent equipment. Any modification to the staff positions will be included in the final rulemaking which is planned to be used for public comment in December 1981.

COMMENT NO. 92: (Section 5(2)) Page 17, item 5(2), Better definition of the level of documentation required to support a certificate of conformance should be provided.

Resolution

See response to Comment No. 91 or the documentation requirement identified in IEEE Standard 323-1971 or IEEE Standard 323-1974. Additional information is also provided in applicable ancillary standards on specific equipment.

COMMENT NO. 93: (Appendix A) Appendix A provides acceptable methods for calculating the mass and energy release both for a LOCA and a main steam line break. For GE, an acceptable reference is stated to be NEDO-10320. This reference is incomplete since it pertains solely to the GE Mark I containment concept. Additions of the appropriate references should be:

a. Mark II containment -- NUREG-0487 (Mark II Interim Acceptance Criteria).

b. Mark III -- NEDO-20533 (GE Mark III Pressure Suppression Containment System Analytical Model) dated June 1974.

Resolution

NUREG-0487 does not contain an acceptable method for calculating the mass and energy release for Mark II containments. The staff has requested that the Mark II applicants perform confirmatory analysis using RELAP 4 to confirm the conservatism in the mass and energy release for Mark II containments as calculated by General Electric.

With respect to NEDO-20533, the staff has accepted the methodology included therein only on a case-by-case basis. The staff does not consider this document fully acceptable on a generic basis. The reference used by the staff is as follows: "Mark III-NEDO-20533 (GE Mark III Pressure Suppression Containment System Analytical Model), dated June 1974, and Supplement 1, dated August 1975."

As a result of this comment, Appendix A was modified to include the "Mark I" reference. The Mark II and III methodology will be evaluated on a case-by-case basis.

COMMENT NO. 94: (Appendix B) The velocity equation used in this section (Appendix B item 2b) is overly conservative. If applied as is, it may yield velocities of several hundred feet per second or more in all areas of the containment. While these high velocities may exist in the region very near the pipe break, they are unreasonably high for remote areas of the containment. This equation does not consider the effects of containment geometry which will affect the convective velocity. Certainly, in the case of PBOC, where the compartment being analyzed is downstream from the break compartment, these velocities are inappropriate. The option should be allowed to calculate velocities for components on a case-by-case basis.

Resolution

The interim position has been developed to conservatively predict the velocities within the containment. It is not the staff's intent to mechanistically arrive at a velocity profile throughout the blowdown, because a mechanistic approach would be complex and would require significant justification for the profiles that were developed. It should be noted that these are interim criteria, and that final resolution will occur upon completion of Task A-21. If more detailed analysis indicates that a less conservative value can be justified, this parameter will be modified. However, at this time, there is no other basis for an alternate approach.

COMMENT NO. 95: (Appendix B)

In BTP CSB 6-1, NRC describes an acceptable heat transfer coefficient for use during LOCA blowdown as a linear ramp from 8 BTU/hr-ft^2 - °F at time = 0 to a peak value of four times the Tagami correlation,

$$h_{max} = 72.5 \left(\frac{Q}{V_{tp}}\right)^{0.62}$$

where h_{max} = 4 x Tagami correlation, BTUs/hr - ft^2 - degrees F

Q = primary coolant energy, BTUs

V = net free containment volume, ft^2

tp = time to end of blowdown, seconds

In Appendix B, item 2 -- acceptable methodology for safety-related component thermal analysis requires the use of the largest of either a condensing heat transfer coefficient based on four times the Uchida correlation, a condensing heat transfer coefficient equal to four times the Tagami correlation, or a convective heat transfer coefficient.

It is unclear whether this requirement is referring to the increasing ramp type Tagami correlation (modified Tagami) used in CSB 6-1 or to the final value of the Tagami correlation, h_{max}. In the latter case it is unclear whether tp is properly defined. In both cases, it is unclear that Q is properly defined.

Please clarify the use of the Tagami correlation for SLB analysis, including all points mentioned above.

Resolution

The intent of the criteria is to maximize the heat transfer coefficient and thereby ensure that the heat flux to components is at a maximum rate. The increasing type Tagami correlation (modified Tagami) as defined in BTP CSB 6-1 is being used. In this Technical Position, "Q" is defined as the energy of the primary system input into the containment at the time that the peak pressure occurs. The time (tp) is defined as the time to the end of blowdown, or that time when approximately 90 percent of the energy is input into the system.

To ensure the selection of a conservative heat transfer coefficient over a range of break sizes that include a turbulent level within the containment, the criteria specify that the maximum heat transfer coefficient should be used by considering 4 x Tagami coefficient or 4 x Uchida coefficient and selecting whichever is greater.

It should be noted that the convective heat transfer should not be used initially, but only as the surface temperature begins to approach the saturation temperature.

COMMENT NO. 96: (Appendix B)

In Section 1.a of Appendix B (Heat Transfer Coefficient), when the containment is superheated and $T_s > T_w$, both convective and condensing heat transfer act simultaneously. The convective heat transfer driving potential is the temperature difference between T_v and T_s and the condensing heat transfer driving potential is the temperature difference between T_s and T_w.

Resolution

Assuming that T_v is equivalent to the bulk temperature, the staff agrees that the convective heat transfer driving potential is the temperature difference between T_v and T_s, and the condensing heat transfer driving potential is the temperature difference between Ts and T_w.

COMMENT NO. 97A: (Appendix C)

We feel and have felt that there is inherent danger in publishing a "universal" environmental profile for use by all in qualifying equipment (Figure C-1). This is the reason IEEE 323-1974 listed numeric values as "typical" and to be used with caution. We recommend the same here.

COMMENT NO. 97B:

Delete note on use of "double spike" (Figure C-1) based on our general comments on Section 3. Also, we feel and have felt that there is inherent danger in publishing a "universal" environmental profile for use by all in qualifying equipment. This is the reason IEEE 323-1974 listed numeric values as "typical" and to be used with caution. We recommend the same here.

COMMENT NO. 97C:

Appendix C - Considering the finite number of both operating plants and plants in the license review process and the NRC manpower devoted to this effort in the NUREG it would seem prudent for the NRC to review, or request review from the utilities involved, of the actual DBE environmental conditions so that these profiles could be provided and envelope profiles drawn based on existing plant configurations. By promulgating a single set of service conditions, the NRC gives the appearance, again, of designing or specifying test conditions where their function is clearly that of addressing the adequacy of what is proposed or has been done. By so specifying these conditions, the NRC is exercising an authority that omits the attendant liability incumbent upon such action.

The basic requirement for qualification is to demonstrate performance under specific service conditions and there are numerous acceptable ways of doing this. The guidance in this Appendix does not address this basic qualification element.

Resolution

The bounding qualification profiles in Appendix C have been generated based on a wide spectrum of postulated accidents. In some cases, these profiles can be considered to be overly conservative; however, in the absence of an approved plant-specific profile, this profile may be used and is considered the minimum bounding profile. In general, the profiles may represent 6 hours of superheat conditions followed by 18 hours of saturated conditions. The actual degree of superheat is left as an open parameter for, as a minimum, the test temperature is to be 340°F for the time specified and the test pressure is to be equal to or greater than the containment design pressure. Obviously, the higher the pressure the less superheat that will exist for a fixed temperature.

For this bounding profile the staff requires a rise in temperature and pressure from normal containment conditions to 340°F and a pressure equal to or greater than the containment design in 10 seconds. This rapid increase in pressure and temperature will provide the component with a severe shock representative of the type of conditions which could be found following a major accident. Following the rapid rise in the test chamber temperature and pressure, the chamber should be stabilized for 6 hours at 340°F to envelope the MSLB conditions.

The basis for the temperature of 340°F and time duration of 6 hours follows from work performed by the staff and General Electric. This is the amount of time one assumes to be required following an accident using a normal cooldown rate of 100°F per hour, because steaming is assumed to occur for that length of time. Because alternative criteria do not exist for a faster cooldown rate, the normal cooldown figure is used. In order to approximate and envelope the LOCA conditions, the temperature and pressure should be reduced after 6 hours to approximately 250°F saturated (approximately 30 psia) and held at these conditions for 2 hours to ensure an equilibrium state has been attained. (This reflects the conditions one might expect following a LOCA.) From 8 to 24 hours, the temperature and pressure can be reduced so that the end point conditions are approximately 250°F and atmospheric pressure. This test for 6 hours at superheated conditions and 18 hours at saturated conditions will bound all possible recirculation line breaks and is, therefore, a bounding test. Any testing beyond 24 hours is beyond the scope of this work and should be addressed in conjunction with postaccident monitoring requirements.

One should recognize that the curve in Figure C-1 is provided for those BWR and PWR ice condenser facilities which do not have plant-specific accident profiles available for use in their specific equipment-qualification program. It would be possible for a utility/applicant to provide, on a case-by-case basis, documentation which yields a plant-unique curve. Should a plant owner want to upgrade a currently installed spray system, it may be possible to shorten the time for the superheat qualification test by providing adequate justification that the spray system will indeed actuate and serve to mitigate the accident, thereby yielding a substantially lower environmental profile. Because Sections 1.1(3) and 1.2(3) clearly allow the use of the plant-specific profiles, no modification was proposed as a result of these comments. (See also staff response to Comment No. 57 regarding the use of a "double spike.")

COMMENT NO. 98: (Appendix C) Appendix C provides DBE qualification profiles for BWR and Ice Condenser Containments only.

Is it the staff's intention to provide the profiles for PWR and other containments at a later date?

Resolution

The staff does not intend at this time to provide generic temperature profiles for PWRs, because there are significant differences between the PWR plants. As a result, no basis can be established to provide a single generic profile.

COMMENT NO. 99: (Appendix D) Page 3 of Appendix D references position C.2 of Regulatory Guide 1.89 which appears to be some other source term not discussed in the current Regulatory Guide revision. The NRC cannot reference an apparent revision to a Regulatory Guide which has not yet undergone public, industrial, and ACRS peer review.

Resolution

The position 1.4(1) source terms coupled with the methodology of Appendix D produce a source term consistant with the above position and with current regulatory practice. Appendix D has been modified for clarity when referencing Position C.2 of Regulatory Guide 1.89.

COMMENT NO. 100: (Appendix D) In Appendix D, page 4, paragraphs G and H, refer to methods which can be used for iodine removal for the PWR. This appendix needs to be expanded with the incorporation of an applicable staff approved and GE-reviewed BWR method.

Resolution

The SPIRT program calculations are independent of reactor type. The necessary parameters for the calculation of the spray lambdas are clearly spelled out in NUREG/CR-0009. Heretofore, GE has not used or referenced a spray system design which would result in a large enough value for lambda to result in any appreciable amount of iodine removal. If GE were to supply an appropriate design, credit for iodine removal by action of the sprays may be taken in the calculation of the equipment qualification dose inside containment. No changes to the staff positions are proposed as a result of this comment.

COMMENT NO. 101: (Appendix D)

Since "accurate coupling of the various time sequences is beyond the scope of this analysis," the lengthy discussion on time sequence on pages D-1 through D-5 is unnecessary. Thus, Westinghouse recommends that this discussion be replaced by statement that an instantaneous release and dispersal is conservatively assumed.

Resolution

The staff concurs with the recommendation. Appendix D was modified by deleting the General Summary of the LOCA Scenario Section and incorporating the statement that an instantaneous release is conservatively assumed (see Section 2.1 of the Appendix).

COMMENT NO. 102: (Appendix D)

On page D-6, Section 6, the applicability of the inclusion of solid fission products in the sump water, to radiation dose estimates, is in the long-term dose rates in recirculation equipment.

Resolution

The staff concurs that the solid fission products will affect the long-term dose-rate calculcation for recirculation equipment. The original staff concern, however, is related to the magnitude of an appropriate fission products release of solids. See also response to comments No. 17.

COMMENT NO. 103: (Appendix D)

In Appendix D, Section 7, the handling of daughter products by a simple multiplication factor of 1.3 is not a rigorous approach for a contribution of such magnitude. The emphasis in this improved NUREG has been on mechanistic and analytical treatment in such areas as activity redistribution and spray removal. Therefore, explicit treatment of daughter products should be included.

Resolution

The staff concurs that explicit treatment of the daughter products should be included in any calculation of the qualification dose for equipment. The staff calculations in Appendix D were included only to demonstrate the estimation of qualification doses at a point inside containment, and, therefore a very rigorous approach was not employed.

COMMENT NO. 104: (Appendix D)

The discussion in Appendix D, Section 7A, considers the airborne gamma and beta dose to the containment centerpoint plus the gamma dose to that point from plateout on the containment walls. Why has the gamma and beta dose from plateout on centrally located equipment been ignored? In the past we have found this to be a significant source.

Resolution

The staff agrees that plate-out on centrally located equipment may be significant and should not be ignored. Position 1.4(2) should be interpreted to require that all potential radiation sources be considered when calculating qualification doses (which would include plate-out on centrally located equipment).

As a result of this comment, no modifications are proposed.

COMMENT NO. 105: (Appendix D) For beta radiation, the shielding effects of the humidity in the containment atmosphere (i.e., a density greater than that of dry air) can be significant in reducing doses, particularly during steam release and containment spray periods. Credit for these effects should be explicitly allowed.

Resolution

The staff does not preclude the option of using a different atmospheric density in the containment to calculate the beta and gamma doses provided the assumed values for density are appropriately justified. The adequacy of the justification for the assumptions used (if other than the conservative dry air conditions) will be evaluated on a case-by-case basis.

COMMENT NO. 106A: (Appendix D) The discussion in Appendix D, Section 7B ("Surface Dose and Dose Rates") considers the contribution from airborne beta and gamma sources and plated-out beta sources but it dismisses the plated-out gamma dose contribution as not being significant. The argument given for this is that "the coating is calculated to be relatively permeable to gammas with only about 1% of the plated-out gamms absorbed by the coating." This seems to be a case of misunderstanding of the definition of "dose," viz. Although the amount of energy deposited in a thin layer may be small, the mass of that thin layer is correspondingly small so that (attenuation ignored) the absorbed dose due to a given incident gamma field is independent of the coating thickness. (Note: Microdosimetric considerations such as electron equilibrium are second order effects and have no impact on the above mentioned concerns.)

COMMENT NO. 106B: The first and second sentences of page D-8, Section 7(b), paragraph 3, are inconsistent. The first refers to gamma exposure rate due to airborne activity while the second refers to gamma absorption rate due to plate-out activity. Since the absorption properties are a function of gamma ray energy and not the location of the source, Westinghouse recommends that exposure rate be used for consistency.

Resolution

The staff concurs. A model used in the "For Comment" version of NUREG-0588 incorrectly calculated the gamma doses in the vicinity of the wall. The doses near the containment walls should consider the photon flux from all sources. As a result of the comment, Appendix D was modified to remove the model.

COMMENT NO. 107: (Appendix D) In Appendix D, we assume that all doses calculated are for a dose point material of air. We would recommend normalizing dose to rads-carbon. This should be stated explicitly and thereby indicate the appropriate method of dosimetry to be applied when testing.

Resolution

All doses have been calculated at a dose point in air. The staff calculation seeks only to illustrate the model, and the values are not to be used as actual qualification values. For actual qualification values, the radiation dose values can be specified in units of rads for the absorbing material or by normalizing to rads-carbon, if preferred.

COMMENT NO. 108: (Appendix D)
Our attempts to reproduce the evaluations of Appendix D lead us to believe that gamma buildup factors were not taken into account. We recommend that this consideration be included.

Resolution

The staff concurs that the dose calculations given in Appendix D did not incorporate the use of gamma buildup factors. A modified Appendix D explicitly cites the use of buildup factors for the sump source. The gamma buildup factors in the containment atmosphere were assumed equal to unity.

COMMENT NO. 109: (Appendix D)
A definition of "shielded" as it is used in items 1.4.7 and 1.4.8 on page 9 of Appendix D is needed.

Resolution

As defined in the Radiological Health Handbook, January 1970, a shield is any body of material used to prevent or reduce the passage of particles or radiation.

COMMENT NO. 110: (Appendix D)
The effect on radiation qualification of ECCS equipment leakage is mentioned on page 2 of Appendix D. Was this effect ignored in the Appendix D analysis?

Resolution

Appendix D does not address (in the sample calculations) the dose contributions resulting from ECCS leakage. Position 1.4(2) should be interpreted to require that all potential radiation sources be considered when calculating qualification doses, which should include dose contributions from ECCS leakage where appropriate (for example, for selected areas outside containment).

COMMENT NO. 111: (Appendix D)
In Appendix D more consideration should be given to the accurate use of dosimetric terminology. Rad and R (Roentgen) are used interchangably in the tables of Appendix D where they shouldn't be. In particular (for example), the use of R (Roentgen) to specify beta-dose is inappropriate. The Roentgen is a unit of "exposure" which is a dosimetric concept reserved for the measurement of ionization of air in a gamma or x-ray field. All doses must be given in rads, and for exactness should be given in rad-carbon, since at the high energies experienced post-accident the "Z" of the receiver material will have a significant effect on the absorbed dose from gammas.

Resolution

The staff agrees. As a result of this comment, the tables of Appendix D have been modified to reflect the recommendations.

COMMENT NO. 112: (Appendix D) In Appendix D, Table D-10 gives the dose rates near an ECCS recirculation pipe. To be useful, it is important to know the size of the pipe and the time post-accident for which the dose rates were determined. Integrated doses would be more useful for radiation qualification purposes than are the dose rates.

Resolution

The staff concurs. The calculation of dose rates in recirculation piping should be performed in a plant-specific manner using guidance contained in NUREG-0578.

COMMENT NO. 113: (Appendix D) The values given in the table on page D-1 do not correspond to those in Tables D-5 through D-8. This inconsistency should be resolved.

Resolution

Appendix D has been corrected to resolve this inconsistency.

PART II
APPENDIX A

Rev

REVISION 1

METHODS FOR CALCULATING

MASS AND ENERGY RELEASE

APPENDIX A

METHODS FOR CALCULATING MASS AND ENERGY RELEASE

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

(1) Topical Report WCAP-8312A (Revision 2, August 1975), for Westinghouse plants. Rev

(2) Section 6.2.1 of CESSAR System 80 PSAR through Amendment 36 for Combustion Engineering plants. Rev

(3) Appendix 6A of B-SAR-205 through Amendment 15 for Babcock & Wilcox plants. Rev

(4) NEDO-10320 and Supplements 1 & 2 for General Electric Mark I plants. (For GE Mark II and III containments, the methods will be evaluated on a case-by-case basis.) Rev

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

(1) Appendix 6B of CESSAR System 80 PSAR through Amendment 36 for Combustion Engineering plants. (The analysis should also include single-failure considerations. The justification for the limiting case that is used will be evaluated by the staff on a case-by-case basis.) Rev

(2) Section 15.1.14 of B-SAR-205 through Amendment 15 for Babcock & Wilcox plants. Rev

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

(4) Topical Report WCAP-8822 (Dated September 1976) for Westinghouse plants. (Although this Topical Report is currently under review, the use of this method is acceptable in the interim if no entrainment is assumed. Reanalysis may be required following the NRC staff review of the entrainment model as presently described.) Rev

PART II
APPENDIX B

Rev

REVISION 1

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

APPENDIX B

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

1. Methodology to Determine the Containment Environmental Response

a. Heat Transfer Coefficient

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

(1) Condensing heat transfer

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

where q/A = the surface heat flux

h_{cond} = the condensing heat transfer coefficient

T_s = the steam saturation (dew point) temperature

T_w = surface temperature of the heat sink

(2) Convective heat transfer

$$q/A = h_c \text{ " } (T_v - T_w)$$

where h_c = convective heat transfer coefficient

T_v = the bulk vapor temperature

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

b. Heat Sink Condensate Treatment

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

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

where M_{cond} = mass condensation rate

X = mass condensation fraction (0.92)

q = surface heat transfer rate

h_v = enthalpy of the superheated steam

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

c. Heat Sink - Surface Area

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

d. Single Active Failure Evaluation

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

e. Containment Heat Removal System Actuation

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

f. Identification of Most Severe Environment

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

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

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

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

a. Condensing Heat Transfer Rate

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

where q/A = component surface heat flux

h_{cond} = condensing heat transfer coefficient is equal to the larger of 4x Tagami correlation or 4x Uchida correlation

T_s = saturation temperature (dew point)

T_w = component surface temperature

b. Convective Heat Transfer

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

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

where Nu = Nusselt number

Re = Reynolds number

C,n = empirical constants dependent on geometry and Reynolds number

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

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

where V = velocity in ft/sec

M_{BD} = the blowdown rate in lbs/hr

V_{CONT} = containment volume in ft3

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

Rev

PART II
APPENDIX C

REVISION 1

QUALIFICATION PROFILES FOR

BWR AND ICE CONDENSER CONTAINMENTS

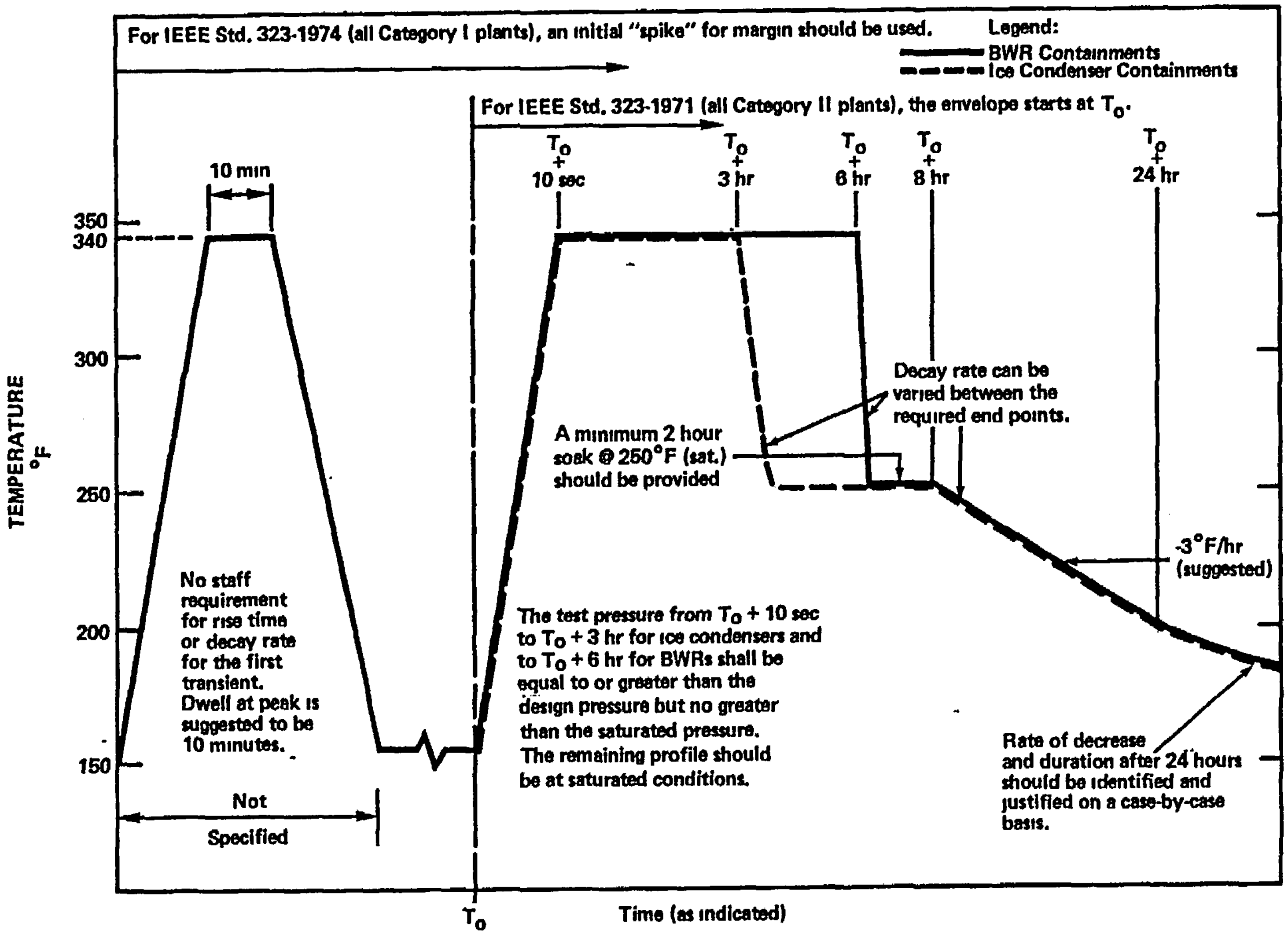

Figure C-1. Qualification Profiles for BWR and Ice Condenser Containments.

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PART II
APPENDIX D

REVISION 1

SAMPLE CALCULATION AND TYPE METHODOLOGY

FOR RADIATION QUALIFICATION DOSE

APPENDIX D

SAMPLE CALCULATION AND TYPE METHODOLOGY FOR RADIATION QUALIFICATION DOSE

This appendix illustrates the proposed staff model for calculating dose rates and integrated doses for equipment qualification purposes. The doses shown in Fig. D-1 include contributions from several source locations in the containment and cover a period of one year following the postulated fission product release. The dose values shown here are provided for illustration only and may not be appropriate for plant-specific application. The dose levels intended for qualification purposes should be determined using the maximum time the equipment is intended to function which, for the design basis LOCA event, may well exceed one year. Rev

The beta and gamma integrated doses presented in the tables and in Figure D-1 have been determined using models and assumptions consistent with those of Regulatory Guide 1.7 and 1.89. This analysis is conservative, and factors in the important time-dependent phenomena related to the action of engineered safety features (ESFs) and natural phenomena, such as iodine plate-out, as done in the previous staff analyses. Rev

The doses presented in Figure D-1 were calculated at the midpoint of the containment and are expected to be representive values for PWR plants having a containment free-volume of 2.5 million cubic feet and a power rating of 4100 MWt. Rev

1. Basic Assumptions Used in an Equipment Qualification Analysis Rev

Gamma and beta doses and dose rates should be determined for three types of radioactive source distributions: (1) from activity suspended in the containment atmosphere, (2) from activity plated-out on containment surfaces, and (3) from activity mixed in the containment sump water. Thus, a given piece of 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. Rev

The source term as set forth in the "For Comment" version of NUREG-0588 is consistent with the C.2 position of Regulatory Guide 1.89 (dated November 1974) and represents the staff licensing position for released fission product activity (i.e., a TID 14844 release).

Although the TMI-2 accident represents only one of a number of possible accident sequences leading to a release of fission products, the staff concluded that a thorough examination of the source term assumptions for equipment qualification was warranted. Current rulemaking proceedings are re-evaluating plant siting policy, degraded cores, minimum requirements for engineered safety features and emergency preparedness. These rulemaking activities also included an examination of fission product releases under degraded core conditions. The final resolution of the source term assumptions is conditioned on the completion of these rulemaking efforts. The staff believes it is prudent to factor in the knowledge of fission product behavior gained from the TMI-2 accident in defining source term assumptions for equipment qualification.

Based upon the fission product release estimates in Volume II, Part 2 of the Rogovin Report (Ref. 8), the staff assumptions for noble gas and iodine releases are still conservative. However, the report estimates that the TMI-2 release contained between 40 and 60 percent of the Cs-134 and Cs-137 core activity in the primary system water, in the containment sump water, and in the auxiliary building tanks.

As part of the effort to incorporate TMI-2 data into the licensing process, the Commission directed the staff to initiate an effort which would investigate the adequacy of the current fission product release assumptions, particularly the past staff assumptions (such as Regulatory Guide 1.3 and Regulatory Guide 1.4) concerning fission product (iodine, cesium, solids, and so forth) behavior following a severe accident. The staff findings from this investigation are presented in two reports, NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions" (Ref.9), which discusses the impact of fission product source term assumptions on past licensing practice, present regulations, and possible future licensing actions, and NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents" (Ref. 10), which contains a description of the most recent technical information currently available for estimating the release of fission products during postulated accidents in commercial LWRs. Rev

The staff feels that as a first step toward modification of the TID-14844 source term in the direction indicated by the TMI-2 experience, it may be prudent to factor in a cesium release in addition to the previously assumed "1% solids." This change in assumptions would have particular significance for the qualification of equipment in the vicinity of recirculating fluids and for equipment required to function for time periods exceeding 30 days. The conclusions from the reports cited above will form the basis for any revision of source term assumptions, and any changes of the source terms will be factored into the final rulemaking in equipment qualification.

2. Assumptions Used in Calculating Fission Product Concentrations

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

The staff has developed a computer program TACT (to be published) that models the time-dependent behavior of iodine and noble gases within a nuclear power plant. The TACT code is used routinely by the staff for the calculation of the offsite radiological consequences of a LOCA, and is an acceptable method for modeling the transfer of activity from one containment region to another and in modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 6), is used to calculate the removal rates of elemental iodine by plate-out and sprays. These codes were used to develop the source term estimates. The following assumptions were also used to calculate the distribution of radioactivity within the containment following a design basis LOCA.

2.1 PWR Dry Containments

a. The source terms used in the analysis assumes that 50 percent of the core iodines and 100 percent of the core noble gases were released instantaneously to the containment atmosphere. Also one percent of the remaining core activity inventory of solids should be assumed to be released from the core and carried with the primary coolant to the containment sump. (NOTE: A potential change in this latter assumption is being considered by the staff for the final rulemaking, as noted above.)

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

c. 6.6×10^5 ft^3 of the containment free volume is assumed unsprayed, which includes regions within the main containment space under the containment dome and compartments below the operating floor level.

d. The ESF fans are assumed to have a design flow rate of 220,000 cfm in the post-LOCA environment. Mixing between all major unsprayed regions and compartments and the main spray region is assured.

e. Air exchange between the sprayed and unsprayed region was assumed to be *one-half of the design flow rate of ESF fans. Good mixing of the con*tainment activity between the sprayed and unsprayed regions is assured by natural convection currents and ESF fans. Rev

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

g. Trace levels of hydrazine were assumed added to enhance the removal of iodine.

h. The spray removal rate constant (lambda) was calculated using the staff's SPIRT program (Ref. 6), conservatively assuming only one spray train operation and an elemental iodine instantaneous partition coefficient (H) of 5000. The calculated value of the elemental iodine spray removal constant was 27.2 hr^{-1}.

i. The assumption that 50 percent of the released iodine activity is instantaneously plated out was not used. Plate-out of iodine on containment internal surfaces was modeled as a first-order rate removal process and best estimates for model parameters were assumed. Based on an assumed total surface area within containment of approximately 5.0×10^5 ft^2, the calculated value for the overall elemental iodine plate-out constant was 1.23 hr^{-1}. The assumption that 50 percent of the activity is instantaneously plated-out was not used.

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

k. A spray removal rate constant (λ) for particulate iodine concentration was assumed to have a value of $\lambda = 0.43\ hr^{-1}$ and allowed the removal of particulate iodine to continue until the airborne concentration was reduced by a factor of 10^4. The organic iodine concentration in the containment atmosphere is assumed unaffected by either the containment spray or plate-out removal mechanisms.

l. The spray and plate-out removal processes were assumed to remove elemental iodine until the instantaneous concentration in the sprayed region was reduced by a factor of 200. This is necessary to achieve an equilibrium airborne iodine concentration consistent with previous LOCA analyses.

m. A relatively open (not compartmented) containment was assumed, and the large release was uniformly distributed in the containment. This is an adequate simplification for dose assessment in a PWR containment, and realistic in terms of specifying the time-dependent radiation environment in most areas of the containment.

n. The analysis assumed that more than one species of radioactive iodine is present in a design basis LOCA. The calculation of the post-LOCA environment assumed that 5 percent of the core inventory of the iodine released is associated with airborne particulate materials and 4 percent of the core inventory of the iodine released formed organic compounds. The remaining 91 percent remained as elemental iodine. For conservatism this composition was assumed present at time t=0. Rev

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

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

2.2 PWR Ice Condenser Containments

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

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

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

c. Removal of airborne iodine in the upper compartment of the containment by the action of both plate-out and spray processes may be assumed provided that these removal processes are evaluated using the assumptions consistent with items h through l in Section 2.1 above and plant-specific parameters.

2.3 BWR Containments

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

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

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

c. Removal of airborne iodine in the drywell or reactor building by both the plate-out and the spray process may be assumed provided the effectiveness of these competing iodine removal processes are evaluated using the assumptions consistent with items h through l in Section 2.1 above and plant-specific parameters. Rev

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

3.0 Model for Calculating the Dose and Dose Rate of Airborne and Plate-Out Fission Products

The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere were calculated for a midpoint in the containment. The containment was modeled as a cylinder of equal height and diameter. Containment shielding and internal structures were neglected because this was considered to involve a degree of complexity beyond the scope of the present work. The calculations of both References 4 and 11 indicate that the specific internal shielding and structure would be expected to reduce the gamma doses and dose rates by factors of two or more, depending upon the specific location and geometry.

Because of the short range of the betas in air, the airborne beta doses were calculated using an infinite medium approximation. This is shown in Reference 2 to result in only a small error. For beta dose calculations for equipment located on the containment walls or on large internal structures, the semi-infinite beta dose model may be used.

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

Finally, all gamma doses were multiplied by a correction factor of 1.3, as suggested in Reference 2, to account for the omission of the contribution from the decay chains of the isotopes.

4.0 Model for Calculating the Dose and Dose Rate of Sump Fission Products

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

The dose calculations assumed a time-dependent iodine source. (The difference between the integrated dose assuming 50 percent of the core iodine immediately available in the sump versus a time-dependent sump iodine buildup is not significant.) Rev

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

5.0 Conclusion

The values given in the tables and in Figure D-1 for the center point in the containment provide an estimate of expected radiation qualification values for Rev a 4100 MWt PWR design at that location.

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

TABLE D-1

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

TIME (HRS)	AIRBORNE IODINE DOSE (R)	AIRBORNE NOBLE GAS DOSE (R)	PLATE-OUT IODINE DOSE (R)	TOTAL DOSE (R)	
0.00	-	-	-	-	
0.03	4.82+4*	7.42+4	1.69+3	1.24+5	
0.06	8.57+4	1.39+5	3.98+3	2.29+5	
0.09	1.09+5	1.98+5	7.22+3	3.14+5	
0.12	1.25+5	2.51+5	1.10+4	3.87+5	
0.15	1.38+5	3.01+5	1.52+4	4.54+5	
0.18	1.47+5	3.48+5	1.96+4	5.15+5	
0.21	1.55+5	3.92+5	2.41+4	5.71+5	
0.25	1.64+5	4.49+5	3.03+4	6.43+5	
0.38	1.87+5	6.19+5	5.05+4	8.57+5	
0.50	2.03+5	7.61+5	6.90+4	1.03+6	
0.75	2.36+5	1.03+6	1.06+5	1.37+6	
1.00	2.66+5	1.26+6	1.40+5	1.67+6	Rev
2.00	3.62+5	2.04+6	2.61+5	2.66+6	
5.00	5.50+5	3.56+6	5.40+5	4.65+6	
8.00	6.63+5	4.38+6	7.47+5	5.79+6	
24.0	1.01+6	6.26+6	1.45+6	8.72+6	
60.0	1.31+6	7.16+6	2.10+6	1.06+7	
96.0	1.45+6	7.56+6	2.39+6	1.14+7	
192.	1.68+6	8.29+6	2.86+6	1.28+7	
298.	1.85+6	8.76+6	3.19+6	1.38+7	
394.	1.95+6	8.85+6	3.41+6	1.42+7	
560.	2.07+6	9.06+6	3.64+6	1.48+7	
720.	2.13+6	9.15+6	3.76+6	1.50+7	
888.	2.16+6	9.19+6	3.83+6	1.52+7	
1060	2.18+6	9.21+6	3.87+6	1.53+6	
1220	2.19+6	9.21+6	3.89+6	1.53+7	
1390	2.20+6	9.21+6	3.90+6	1.53+7	
1560	2.20+6	9.22+6	3.91+6	1.53+7	
1730	2.20+6	9.22+6	3.91+6	1.53+7	
1900	2.20+6	9.22+6	3.92+6	1.53+7	
2060	2.20+6	9.22+6	3.92+6	1.53+7	
2230	2.20+6	9.22+6	3.92+6	1.53+7	
2950	2.20+6	9.23+6	3.92+6	1.54+7	
3670	2.20+6	9.24+6	3.92+6	1.54+7	
4390	2.20+6	9.24+6	3.92+6	1.54+7	
5110	2.20+6	9.25+6	3.92+6	1.54+7	
5830	2.20+6	9.25+6	3.92+6	1.54+7	
6550	2.20+6	9.26+6	3.92+6	1.54+7	
7270	2.20+6	9.26+6	3.92+6	1.54+7	
8000	2.20+6	9.27+6	3[illegible]2+6	1.54+7	
8710	2.20+6	9.28+6	3.92+6	1.54+7	
			TOTAL	1.54+7	

*+4=10^4

TABLE D-2*

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

TIME (HRS)	AIRBORNE IODINE DOSE (RADS)	AIRBORNE NOBLE GAS DOSE (RADS)	TOTAL DOSE (RADS)
0.00	-	-	-
0.03	1.47+5	5.48+5	6.95+5
0.06	2.62+5	9.86+5	1.25+6
0.09	3.33+5	1.35+5	1.68+6
0.12	3.83+5	1.65+6	2.03+6
0.15	4.20+5	1.91+6	2.33+6
0.18	4.49+5	2.14+6	2.59+6
0.21	4.73+5	2.35+6	2.82+6
0.25	5.00+5	2.60+6	3.10+6
0.38	5.67+5	3.30+6	3.87+6
0.50	6.15+5	3.86+6	4.48+6
0.75	7.13+5	4.89+6	5.60+6
1.00	8.00+5	5.81+6	6.61+6
2.00	1.07+6	9.02+6	1.01+7
5.00	1.58+6	1.65+7	1.81+7
8.00	1.88+6	2.20+7	2.39+7
24.0	2.87+6	4.08+7	4.37+8
60.0	3.89+6	6.15+7	6.54+7
96.0	4.37+6	7.48+7	7.92+7
192.	5.14+6	1.00+8	1.05+8
298.	5.64+6	1.17+8	1.23+8
394.	5.99+6	1.25+8	1.31+8
560.	6.34+6	1.34+8	1.40+8
720.	6.53+6	1.39+8	1.46+8
888.	6.63+6	1.42+8	1.49+8
1060	6.69+6	1.44+8	1.51+8
1220	6.73+6	1.45+8	1.52+8
1390	6.75+6	1.47+8	1.54+8
1560	6.76+6	1.49+8	1.56+8
1730	6.76+6	1.51+8	1.58+8
1900	6.76+6	1.52+8	1.59+8
2060	6.76+6	1.54+8	1.61+8
2230	6.77+6	1.55+8	1.62+8
2950	6.77+6	1.62+8	1.69+8
3670	6.77+6	1.69+8	1.76+8
4390	6.77+6	1.76+8	1.83+8
5110	6.77+6	1.83+8	1.90+8
5830	6.77+6	1.89+8	1.96+8
6550	6.77+6	1.96+8	2.03+8
7270	6.77+6	2.03+8	2.10+8
8000	6.77+6	2.09+8	2.16+8
8710	6.77+6	2.16+8	2.23+8
		TOTAL	2.23+8

*Tables D-3 through D-9 have been deleted.
+Dose conversion factor is based on absorption to tissue.

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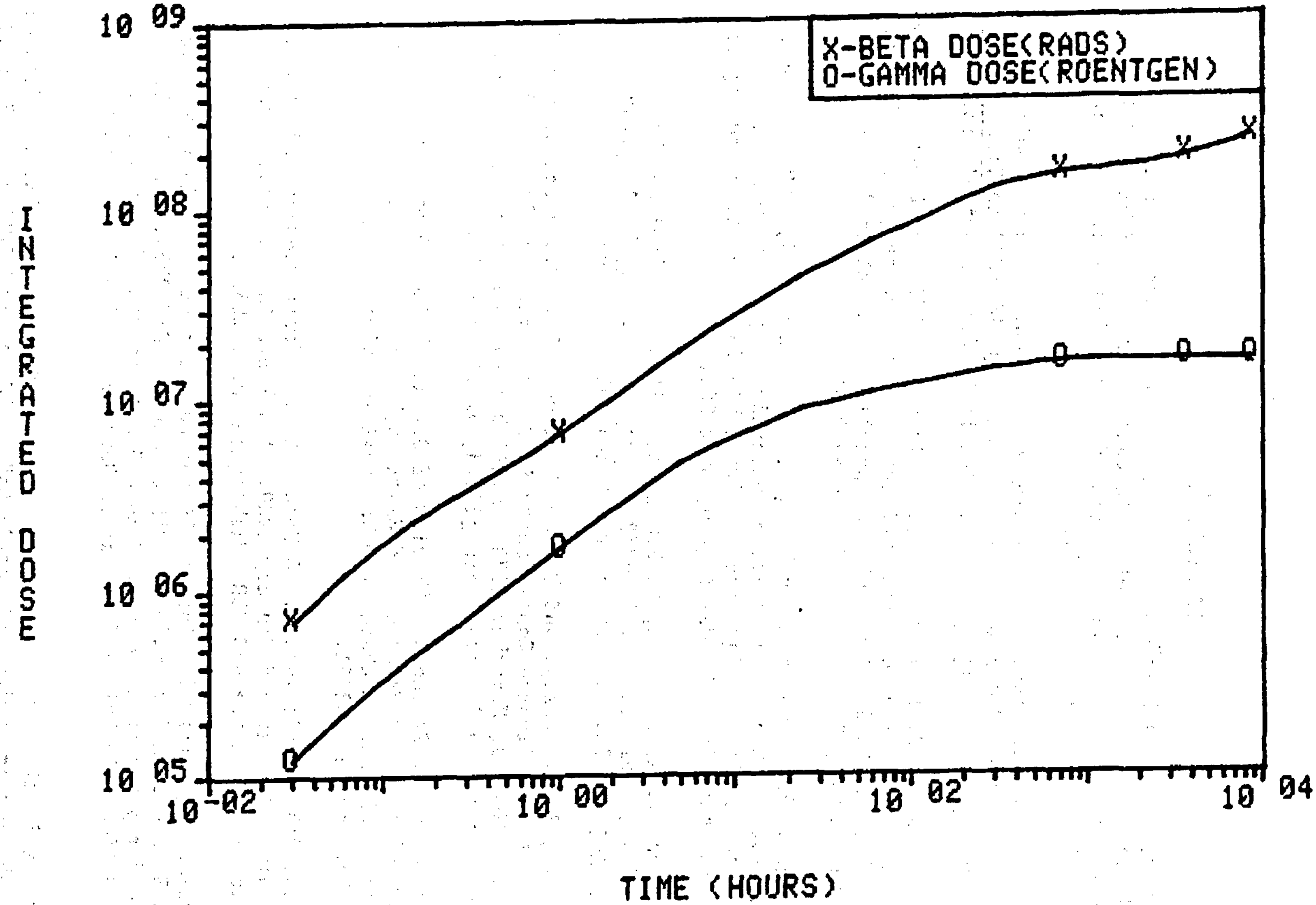

Figure D-1 Sample airborne doses for a dose point on the containment centerline

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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

NUREG-0588
Rev. 1

4 TITLE AND SUBTITLE *(Add Volume No., if appropriate)*

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

Subtitle: Including Staff Responses to Public Comments

2 *(Leave blank)*

3 RECIPIENT'S ACCESSION NO

7 AUTHOR(S)

A. J. Szukiewicz and others

5 DATE REPORT COMPLETED

MONTH	YEAR
November	1980

DATE REPORT ISSUED

MONTH	YEAR
July	1981

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*

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

6 *(Leave blank)*

8 *(Leave blank)*

12 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*

Same as 9 above.

10 PROJECT/TASK/WORK UNIT NO

11. CONTRACT NO

13 TYPE OF REPORT

Regulatory Report

PERIOD COVERED *(Inclusive dates)*

15 SUPPLEMENTARY NOTES

This report documents resolution of Unresolved Safety Issue A-24

14 *(Leave blank)*

16 ABSTRACT *(200 words or less)*

This document provides the NRC staff positions regarding selected areas of environmental qualification of safety-related electrical equipment, in the resolution of Unresolved Safety Issue A-24, "Qualification of Class IE Safety-Related Equipment." The positions herein are applicable to plants that are or will be in the construction permit (CP) or operating license (OL) review process and that are required to satisfy the requirements set forth in either the 1971 or the 1974 version of IEEE-323 standard. Part I of this report contains the original "For Comment" NUREG that was published in December 1979. This "For Comment" issue is now endorsed by the Commission, in the May 23, 1980 Memorandum and Order (CLI-80-21), as the staff's interim positions until the final positions currently being developed in rulemaking, are established. Part II of this reports contains the staff's responses and resolution of the public comments that were solicited and received before May 1, 1980. Appendices A through D identify the additions, modifications and/or corrections that were made as a result of these comments.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b IDENTIFIERS/OPEN-ENDED TERMS

18 AVAILABILITY STATEMENT

Unlimited

19. SECURITY CLASS *(This report)*

Unclassified

20 SECURITY CLASS *(This page)*

Unclassified

21 NO OF PAGES

22 PRICE

$

NRC FORM 335 (7-77)