June 9, 1982

SECY-82-207A

RULEMAKING ISSUE

For: The Commissioners (Affirmation)

From: William J. Dircks
Executive Director for Operations

Subject: FINAL RULE, "ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS"

Purpose: To provide the Commission a final revision of the rule, previously submitted as SECY-82-207, as a result of comments received at the June 1, 1982, Commission meeeting on this subject.

Discussion: Based on the Commission meeting held on June 1, 1982, Enclosure 1, Notice of Final Rulemaking, SECY-82-207, dated May 24, 1982, has been modified as follows:

1. Paragraph 50.49(k) and the statement of considerations (page 4) have been expanded to clarify the requirements for plants currently under review for operating licenses.

2. The definition of "safety-related" has been clarified in paragraph 50.49(c). The statement of considerations (page 6) has been expanded to clarify that certain post-accident monitoring equipment is covered by this rule.

3. Responses to public comments in several areas have been expanded to provide the technical bases for the staff positions.

For immediate reference, all revisions, including some minor revisions not mentioned above, have been indicated by a bar on the right hand side on both Enclosure 1 and Enclosure 2.

CONTACT:
Satish K. Aggarwal, RES
443-5946

Enclosed for your information is a copy of Regulatory Guide 1.89, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," which was issued for public comment in February 1982. The staff is currently resolving public comments, and comments expressed during the recent Commission meeting. The revised guide will be developed on an expedited basis and issued no later than September 30, 1982.

Issuance of the final rule should not be conditioned on issuance of Regulatory Guide 1.89 since the final rule, in conjunction with the statement of considerations is sufficiently explicit.

The Offices of Nuclear Regulatory Research, Nuclear Reactor Regulation, Inspection and Enforcement, and the Chairman, Committee to Review Generic Requirements have concurred in this version of the final rule. The Office of the Executive Legal Director has no legal objection.

Scheduling: Affirmation of this rule by June 24, 1982, will allow publication of the Federal Register Notice prior to the end of June.

William J. Dircks
Executive Director
for Operations

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Enclosures:
1. Federal Register Notice for Final Rulemaking
2. Analysis of Public Comments
3. Value/Impact Statement
4. Regulatory Guide 1.89 (Feb 1982)

Commissioner comments should be provided directly to the Office of the Secretary by c.o.b. Tuesday, June 22, 1982, if the Commission wants to approve this rule in time for publication prior to June 30, 1982.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Thursday, June 17, 1982, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

This paper will be scheduled for Affirmation at an Open Meeting following Commissioner approval.

ENCLOSURE 1

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Environmental Qualification of Safety-Related Electric Equipment for Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: [Proposed] Final rule.

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

EFFECTIVE DATE: [UPON publication in the Federal Register]

[DATES: Comment period expires (60 days after publication in the FederalRegister). Comments received after ---------- will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Written comments and suggestions may be mailed to the Secretary of the Commission, Attention: Docketing and Service Branch,

U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, or hand-delivered to the Commission's Public Document Room at 1717 H Street NW., Washington, D.C., between the hours of 8:30 a.m. and 4:45 p.m. on normal work days.]

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

SUPPLEMENTARY INFORMATION:

Previous Notice

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

Nature and Scope of the Rulemaking

Nuclear power plant equipment important to safety must be able to perform [the] its safety functions throughout its installed life. This requirement is embodied in General Design Criteria 1, 2, 4, and 23 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"; in Criterion III, "Design Control," and Criterion XI, "Test Control," of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel

Reprocessing Plants," to 10 CFR Part 50; and in paragraph 50.55a(h) of 10 CFR Part 50, which incorporates by reference IEEE 279-1971,[1,2] "Criteria for Protection Systems for Nuclear Power Generating Stations." This requirement is applicable to equipment located inside as well as outside the containment.

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

Currently, the Commission has under way a program to reevaluate the qualification of electric equipment in all operating nuclear power plants. As a part of this program, more definitive criteria for environmental qualification of safety-related electric equipment [~~important-to-safety~~] have been developed by the NRC. A document entitled "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines) was issued in November 1979. In addition, the NRC has issued NUREG-0588, "Interim Staff Position on

[1]Incorporation by reference approved by the Director of the Office of Federal Register on January 1, 1981.

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

Environmental Qualification of Safety-Related Electrical Equipment," which contains two sets of criteria: the first for plants originally reviewed in accordance with IEEE 323-1971 and the second for plants reviewed in accordance with IEEE 323-1974.

By its Memorandum and Order CLI-80-21 dated May 23, 1980, the Commission directed the staff to proceed with a rulemaking on environmental qualification of safety-related equipment and to address the question of backfit. The Commission also directed that the DOR Guidelines and NUREG-0588 form the basis for the requirements licensees and applicants must meet until the rulemaking has been completed. This [~~proposed~~] rule is [~~generally~~] based on the requirements of the Division of Operating Reactors (DOR) Guidelines and NUREG-0588. The Commission recognizes the qualification efforts of the industry as a result of CLI-80-21. Therefore, the rule provides relief to operating nuclear power plants (see paragraph k of the final rule.). Requalification of electric equipment in accordance with this rule will not be required for equipment qualified or being qualified in accordance with DOR Guidelines or NUREG-0588 provided the qualification of electric equipment has commenced prior to [insert effective date of this amendment]. Those nuclear power plants that are currently under review and are qualifying safety-related electric equipment in accordance with NUREG-0588 (Category I or II) will satisfy the requirements of this rule. Category I requirements (IEEE 323-1974) apply to nuclear power plants for which the construction permit safety evaluation report was issued after July 1, 1974, and Category II requirements (IEEE 323-1971) apply to nuclear power plants for which the construction permit safety evaluation report was issued prior to July 1, 1974.

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

[The Commission's Memorandum and Order CLI-80-21 directed that the environmental qualification of electric equipment in operating nuclear power plants be completed by June 30, 1982. However, on September 23, 1981, the Commission considered the petition (SECY-81-486) to extend this deadline. The proposed rule covers the same electric equipment as CLI-80-21 and implements SECY-81-486 by incorporating the extension dates recommended by the Chairman in his memorandum dated September 30, 1981. Included in the proposed rule is a requirement that each holder of or each applicant for a license to operate a nuclear power plant identify and qualify the electric equipment needed to complete one path of achieving and maintaining a cold shutdown condition. The Commission specifically requests comment on this proposed additional requirement.]

The scope of the [proposed] final rule does not include all electric equipment important to safety in its various gradations of importance. It [includes] covers that portion of equipment important to safety commonly referred to as "safety-related." [or "Class 1E" equipment in IEEE national standards and some additional non-Class 1E equipment and systems whose failure under extreme environmental conditions could prevent the satisfactory accomplishment of safety functions by accident mitigating equipment.] Safety-related structures, systems and components are those that are relied upon to remain functional during and following design basis events to assure (1) the integrity of the reactor coolant pressure

boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to assure functions (1) through (3) above. Safety-related electric equipment covered by this rule is essentially the same as "Class 1E" equipment defined in IEEE 323-1974. Also covered in the scope of the final rule is certain postaccident monitoring equipment specified as "Category 1 and 2" in Regulatory Guide 1.97," "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" (Revision 2).

Included in the [proposed] final rule are specific technical require ments pertaining to (a) qualification parameters, (b) qualification methods, and (c) documentation. Qualification parameters include temperature, pressure, humidity, radiation, chemicals, and submergence. Qualification methods include (a) testing as the principal means of qualification and (b) analysis and operating experience in lieu of testing. The [proposed] final rule ~~would~~ requires that the qualification program include synergistic effects, aging, margins, radiation, and environmental conditions. Also, a record of qualification must be maintained. Proposed Revision 1 to Regulatory Guide 1.89, [~~is being revised to~~] which has been issued for public comment, will describe methods acceptable to the NRC for meeting the provisions of this [proposed] rule and [to] will include a list of typical equipment covered by it [~~a draft of the proposed~~]. Revision 1

~~[is-being-published-for-public-comment-concurrently with-the-proposed-rule:]~~ to Regulatory Guide 1.89 will be issued after resolution of public comments. Specific guidance on replacement parts will be included in Regulatory Guide 1.89. The Commission expects that licensees and applicants will utilize the replacement process to upgrade the quality of electric equipment to the provisions of this rule.

~~[Also-included-in-the-proposed-rule-is-a-requirement;-which-is-consistent-with-Commission-Memorandum-and-Order;-CLI-80-21;-for-submission---of-an-analysis-by-licensees-to-ensure-that-the-plant-can-be-safely-operated pending-completion-of-the-environmental-qualification-of-electric-equipment:--The-Commission-expects-that;-for-each-of-the-currently-operating power-plants;-this-analysis-and-its-evaluation-by-the-NRC-staff-will-be completed-well-in-advance-of-the-effective-date-of-this-rule:--If-the licensees-of-operating-power-plants-fail-to-provide-these-analyses-in-a timely-manner;-the-Commission-expects-the-NRC-staff-to-take-the-appropriate-steps-to-require-that-the-information-be-provided-and-to-enforce compliance-with-this-requirement:--This-requirement-has-been-included-in this-proposed-rule-to-provide-a-regulatory-basis-for-enforcement:]~~

NRC will generally not accept analysis alone in lieu of testing. Experience has shown that qualification of equipment without test data may not be adequate to demonstrate functional operability during design basis event conditions. Paragraph 50.49(f) provides four methods for qualification. Testing will be preferred. Justification for qualification of any of the remaining three methods must meet NRC approval. To ensure integrity of a testing program, the same piece of equipment must be used throughout the complete test sequence. ~~[Analysis-may-be-acceptable-if-testing-of-the equipment-is-impractical-because-of-size;-or-limitation-due-to-the-state~~

~~of-the-art.--The-proposed-rule-takes-into-consideration-the-prior-qualification-history-of-the-operating-power-plants.--For-example,-the-proposed rule-recognizes-that-for-those-plants-which-are-not-committed-to-either IEEE-323-1971-or-IEEE-323-1974-for-equipment-qualification,-and-have-been tested-only-for-high-temperature-pressure,-and-steam,-some-equipment-may-not need-to-be-tested-again-to-include-other-service-conditions-such-as-radiation and-chemical-sprays.--The-qualification-of-equipment-for-these-service-conditions-may-be-established-by-analysis.]~~

The ~~[proposed]~~ final rule ~~[would]~~ requires that each holder of an operating license provide a list of safety-related electric equipment previously qualified based on testing, analysis, or a combination thereof and a list of equipment that has not been qualified. These lists and the schedule for completion of equipment qualification ~~[would-have-to]~~ must be submitted ~~[written]~~ by [Insert a date 90 days after the effective date of this amendment]. ~~[rule: However,-this-time-period-will-be-adjusted-during-the-final-rule-making-process-to-allow-reasonable-time-for-licensees-to evaluate-NRC's-safety-reviews-that-are-currently-underway.]~~

~~[The-proposed-rule-will-codify-the-Commission's-current-requirements for-the-environmental-qualification-of-electric-equipment.--Upon-publication-of-a-final-rule,-the-DOR-guidelines-and-NUREG-0588-will-be-withdrawn.]~~

The general requirements for seismic and dynamic qualification for electric equipment are contained in the General Design Criteria. Further guidance is provided in Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants" (Revision 1) and NUREG-0800, "Standard Review Plan." ~~[Pending-developments-of-specific-requirements-in this-area,-the-general-requirements-will-continue-to-apply.]~~ NRC is

considering to include [expansion-of-the-scope-of-this-rule-to-include] additional electric equipment important to safety and the requirements for seismic and dynamic qualification of electric equipment [This matter will be the subject of] in future rulemaking.

[Additional views of Commissioner Bradford: Commissioner Bradford believes that the proposed deadline (second refueling outage after March 31, 1982) for qualification is much too relaxed, given the fact that licensees and the NRC have been aware of the problems in this area since 1978. The proposed deadline extends as much as two and one half years beyond the June 30, 1983 date by which the Atomic Industrial Forum concluded that nearly all electrical equipment could be qualified. Given the more generous deadline, he also believes that the rule should have contained requirements for seismic and dynamic qualification. While the general design criteria contain requirements in this area, clarification now would ensure that equipment to be replaced in the near term will not have to be ripped out in a few years because it was not properly seismically qualified.

Commissioner Gilinsky has agreed with these views.]

Comments On The Proposed Rule

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

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

(1) Seismic and Dynamic Qualification - Paragraph 50.49(a)

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

Response: Safety-related electric equipment at operating nuclear power plants was generally qualified for environmental and seismic stresses separately, i.e., by using separate prototypes for environmental and seismic qualification tests. The Commission has decided, after considerable deliberation, to pursue this issue at a future date through the issuance of an advance notice of proposed rulemaking. A future seismic rule may not require retesting for environmental stresses because a single prototype was not used during the original qualification.

(2) Scope - Cold Shutdown Requirement - Paragraph 50.49(c)

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

Response: The Commission agrees that this is a new requirement that may introduce significant costs. The licensing basis of the majority of operating reactors does not require that all electric equipment and systems necessary to bring the reactor from normal operating conditions to cold shutdown be designed to Class 1E standards. Therefore, to require that all plants environmentally qualify the electric equipment and systems needed to complete one path of achieving and maintaining a cold shutdown condition may require the upgrading of a significant amount of equipment and systems that do not currently meet Class 1E

standards for operating reactors. However, electric equipment and systems necessary to shut down the reactor and maintain it in a safe shutdown condition are required to meet Class 1E standards and therefore would be covered by the rule.

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

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

(3) Scope - Equipment in a Mild Environment - Paragraph 50.49(c)

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

Response: The final rule does not cover the electric equipment located in a mild environment. The Commission has concluded that the general quality and surveillance requirements applicable to safety-related electric equipment as a result of other Commission regulations, including 10 CFR Part 50, Appendix B (see for example, Regulatory

Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 3) are sufficient to ensure adequate performance of safety-related electric equipment located in mild environments. Since it has been concluded that no further environmental qualification requirements are needed for such equipment provided they fully satisfy all other applicable regulations, the Commission has determined that no additional requirements are necessary with respect to safety-related equipment located in mild environments in order for licensees to satisfy, with respect to such equipment, existing license conditions or technical specifications calling for qualification of safety-related electric equipment in accordance with DOR Guidelines or NUREG-0588.

(4) Scope - Previous Qualification Efforts - Paragraph 50.49(c)

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

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

(5) Humidity - Paragraph 50.49(e)(2)

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

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

(6) Aging - Paragraph 50.49(e)(5)

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

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

(7) Margins - Paragraph 50.49(e)(8)

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

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

(8) Analysis and partial test data - Paragraph 50.49(f)(4)

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

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

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

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

Response: The Commission agrees. This requirement has been subject to different interpre tations. A record of qualification must be maintained in an "auditable form" but not necessarily in a central file, for the entire period during which the covered item is installed in a nuclear power plant. Recordkeeping requirement of 10 CFR Part 50 Appendix B must be met. Certain records can be kept at the vendors shop.

(10) Justification of continued operation for operating plants.

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

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

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

Effective Date:

This rule is effective upon publication in the Federal Register. The Commission has determined that the final rule should take immediate effect upon publication because it relieves a restriction under subsection (d)(1) of Section 553 of the Administrative Procedure Act. This is so because all operating reactor licensees are currently under a June 30, 1982, deadline to complete environmental qualification of safety-related electric equipment. The final rule's implementation schedule, as explained above, supplants this date and thus gives licensees additional time to

complete environmental qualification of safety-related electric equipment. In addition, the Commission finds that there is good cause--pursuant to subsection (d)(3) of Section 553--to make the rule's requirements effective upon publication. The first licensee actions under the rule are not required until 90 days after the effective date of the rule. This 90-day period is intended to include the statutory 30 days and allow 60 additional days to make the submittal required by Paragraph 50.49(g) of the rule. The overall effect of making the rule effective on publication is to relieve licensees of the June 30, 1982, deadline and to provide a sufficient period after the effective date of the rule for licensees to achieve compliance with the near-term requirements of the rule.

Paperwork Reduction Act

The [proposed] final rule contains recordkeeping requirements that are subject to review by the Office of Management and Budget (OMB). As required by P.L. 96-511, this proposed rule [will be] was submitted to OMB for clearance of the recordkeeping requirements.

Regulatory Flexibility Statement

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule[; if promulgated;] will not have a significant economic impact on a substantial number of small entities. This [proposed] final rule affects the method of qualification of electric equipment by utilities. Utilities do not fall within the definition of a small business found in Section 3 of the Small Business Act, 15 U.S.C. 632. In addition, utilities are required by the Commission's Memorandum and Order CLI-80-21, dated May 23, 1980, to meet the requirements

contained in the DOR "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," (November 1979) and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which form the basis of this [proposed] rule. Consequently, this rule codifies existing requirements and imposes no new costs or obligations on utilities.

List of Subjects in 10 CFR Part 50

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

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, [~~notice is hereby given that adoption of the~~] the following amendment to Title 10, Chapter I, Code of Federal Regulations, Part 50, [~~10 CFR Part 50 is contemplated~~] is published as a document subject to codification.

10 CFR Part 50

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

AUTHORITY: Secs. 103, 104, 161, 182, 183, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2239); secs. 201, 202, 206, 88 Stat. 1243, 1244, 1246 (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C 2234). Sections 50.100-50.102 issued under sec. 186, 68 Stat. 955 (42 U.S.C. 2236).

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

2. A new § 50.49 is added to read as follows:

§ 50.49 Environmental qualification of safety-related electric equipment for nuclear power plants.

(a) Requirements for seismic and dynamic qualification of safety-related electric equipment are not included in this section. Also not included are the requirements for safety-related electric equipment located in a mild environment. A mild environment is an environment that would at no time be significantly more severe than the environment that would occur during normal plant operation including during anticipated operational occurrences.

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

(c) Safety-related electric equipment and systems covered by this section include electric equipment and systems that are [~~essential to emergency reactor shutdown; containment isolation; reactor core cooling; and containment and reactor heat removal or that are otherwise essential in preventing significant release of radioactive material to the environment. Included is equipment (1) that performs the above functions automatically; (2) that is used by the operator to perform these functions~~

~~manually;-and-(3)-whose-failure-can-prevent-the-satisfactory-accomplishment-of-one-or-more-of-the-above-safety-functions.---Also-included-is equipment-needed-to-complete-one-path-of-achieving-and-maintaining-a cold-shutdown-condition.~~] relied upon to remain functional during and following design basis events to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the 10 CFR Part 100 guidelines. Design basis events are defined as conditions of normal operation including anticipated operational occurrences, design basis accidents, external events, and natural phenomena for which the plant must be designed to assure functions (1) through (3) above.

(d) The applicant or licensee shall prepare a list of ~~all~~ safety-related electric equipment covered by this section. [~~and-maintain-it-in an-auditable-form.--This-list-of-equipment-must,-as-a-minimum,-include:~~] In addition, the applicant or licensee shall include the following information for safety-related electric equipment in a qualification file:

(1) The performance specifications [~~and-structural-integrity-requirements~~] under conditions existing [~~during-normal-and-abnormal-operation-and~~] during and following design basis events. [~~and-afterwards-and-the-lengths-of the-periods-during-which-the-integrity-must-be-maintained:~~]

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

(3) The environmental conditions, including temperature, pressure, humidity, radiation, chemicals, and submergence [~~and-the-predicted-varia-~~

~~tions-of-these-environmental-conditions-with-time~~] at the location where the equipment must perform as specified in accordance with paragraphs (d)(1) and (2) of this section.

(e) The electrical equipment qualification program must include the following:

(1) Temperature and Pressure. The time-dependent temperature and pressure at the location of the safety-related equipment must be established for the most [~~limiting~~] severe [~~of-the-applicable-postulated-accidents~~] design basis events during or following which this equipment is required to remain functional. This time-dependent temperature and pressure must be used as the basis for the environmental qualification of safety-related electric equipment.

(2) Humidity. [~~Time-dependent-variations-of-relative~~] Humidity during normal operation and design basis events must be considered.

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

(4) Radiation. The radiation environment must be based on the type of radiation, the total dose [~~and-dose-rate-of-the-radiation-environment~~] expected during normal operation over the installed life of the equipment, [~~plus~~] and the radiation environment associated with the most severe design basis event during or following which the equipment is required to

remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

(5) Aging. Equipment qualified by test must [~~practicable~~] be preconditioned by natural or artificial (accelerated) aging to its installed end-of-life condition. [~~Electromechanical equipment must be operated to the mechanical wear and electrical degradation expected during its installed life.~~] If Where preconditioning to an installed end-of-life condition [~~a qualified life equal to the installed life~~] is not [~~possible~~] practicable and technically meaningful, the equipment may be preconditioned to a shorter [~~qualified~~] designated life. The equipment must be replaced or refurbished at the end of [~~its qualified~~] this designated life unless ongoing qualification [~~of~~] demonstrates [~~prototype equipment naturally aged in plant service shows by artificial aging and type testing~~] that the item has additional [~~qualified~~] life.

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

(7) Synergistic Effects. [~~The preconditioning and testing of equipment must consider known~~] Synergistic effects must be considered when these effects are [~~known~~] believed to have a significant effect on equipment performance.

(8) Margins Margins must be applied to account for production variations and inaccuracies in test instruments. These margins are in addition to [~~margins applied during the derivation of the environmental conditions.~~] any conservatisms applied during the derivation of environmental conditions unless these conservatisms can be quantified and shown to contain appropriate margins.

ANALYSIS OF PUBLIC COMMENTS

ON 10 CFR §50.49 (47 FR 2876, Jan. 20, 1982)

1. §50.49(a) - Seismic Requirements:

A. Comment: Seismic and dynamic qualification is an integral part of environmental qualification. It is therefore inappropriate to codify environmental qualification first and then to codify seismic qualification separately at a later date.

Response: Electric equipment at operating nuclear power plants was generally qualified for environmental and seismic stresses separately; i.e., by using separate prototypes for environmental and seismic qualification tests.

The proposed revision to Regulatory Guide 1.89 (Feb. 1982) specifies "single prototype" testing (sequence testing) as an acceptable method for qualifying safety-related electric equipment. The implementation section of this guide will include NTOL's and future plants and not operating plants. A future seismic rule may not require retesting for environmental stresses because a single prototype was used during the original qualification.

Also refer to response to of comment 1.C.

B. Comment: The proposed rule has introduced a new term, "dynamic qualification" without definition.

Response: "Dynamic Qualification" is outside the scope of this rulemaking. Therefore, no specific definition is required at this time. The term will be specifically defined as part of the future rulemaking.

C. Comment: In the absence of seismic requirements in §50.49, equipment which may be replaced in the near term may have to be ripped out if it fails to meet the backfitting requirements, if any.

Response: For plants operating prior to the effective date of the final rule, replacement parts that were environmentally and seismically qualified by the use of separate prototypes prior to the effective date of this rule may not require "ripping out" because a single prototype was not used. This will be a subject of the seismic rule.

D. Comment: It is appropriate that seismic and dynamic qualification requirements should not be included in §50.49. It must, however, be stated that qualification to IEEE 344-1975 is one acceptable method for seismic qualification.

Response: Regulatory Guide 1.100 already endorses IEEE 344-1975 in this area.

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

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

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

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

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

[~~(i)~~] (j) A record of the qualification including documentation in paragraph (d) of this section must be maintained in [~~a central file~~] an auditable form for the entire period during which the covered item is installed in the nuclear power plant or is stored for future use to permit verification that each item of safety-related electric equipment covered by this section (1) is qualified for its application and

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

(k) Licensees are not required to qualify safety-related electric equipment in accordance with the requirements of this rule provided the following conditions are met:

~~perform-an-analysis-to-ensure-that-the-plant-can-be-safely-operated-pending completion-of-the-environmental-qualification.--The-detailed-analysis-for each-equipment-type-with-appropriate-justification-must-be-submitted-to Director-of-Nuclear-Reactor-Regulatory-by-(insert-the-effective-date-of the-rule)-and-must-include,-where-appropriate,-consideration-of:~~

~~(i)--Accomplishing-the-safety-function-by-some-designated-alternative equipment-that-has-been-adequately-qualified-and-satisfies-the-single failure-criterion-if-the-principal-equipment-has-not-been-demonstrated-to be-fully-qualified;~~

~~(2)--The-validity-of-partial-test-data-in-support-of-the-original qualification;~~

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

~~(4)--Completion-of-the-safety-function-prior-to-exposure-to-the-ensuing-accident-environment-and-the-subsequent-failure-of-the-equipment-does not-degrade-any-safety-function-or-mislead-the-operator;~~

~~(5)--No-significant-degradation-of-any-safety-function-or-misleading of-the-operator-as-a-result-of-failure-of-equipment-under-the-accident environment.~~]

[~~(k)~~] (i) The applicant for an operating license that is granted on or after [insert the effective date of this amendment] but prior to November 30, 1985, [~~must~~] shall perform an analysis to ensure that the plant can be safely operated pending completion of environmental qualification. [~~in-accordance-with-paragraph-(j)-of-this-section-except-that-this-analysis~~] This analysis must be submitted to the Director of Nuclear Reactor Regulation for consideration prior to the granting of an operating license and must include, where appropriate, consideration of:

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

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

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

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

[~~(4) Analysis in lieu of testing in the following cases:~~

~~(i) if type testing is precluded by the physical size of the equipment or by the state of the art;~~]

(4) [~~(ii)~~ By] Analysis in combination with partial type test data [which] that supports the analytical assumptions and conclusions. [~~; if the equipment purchase order was executed prior to May 23, 1980.~~

~~(g) If an item of electric equipment is to be qualified by test--~~

~~(1) The acceptance criteria must be established prior to testing;~~

~~(2) The tests must be designed and conducted to demonstrate that the equipment can perform its required function as specified in accordance with paragraph (d)(1) of this section for all conditions as specified in accordance with paragraphs (d)(2) and (3) of this section; The test profile (e.g., pressure, temperature, radiation vs. time) must include margins as set forth in paragraph (e)(8) of this section;~~

~~(3) The test profile must be either (i) a single profile that envelops the environmental conditions resulting from any design basis~~

~~event-during-any-mode-of-plant-operation-(e.g.,-a-profile-that-envelops the-conditions-produced-by-the-postulated-spectrum-of-main-steamline break-(MSLB)-and-loss-of-coolant-accidents-(LOCA))-or-(ii)-separate-profiles-for-each-type-of-event-(e.g.,-separate-profiles-for-the-MSLB-accidents-and-for-LOCAs).~~

~~(4)--The-same-piece-of-equipment-must-be-used-throughout-the-complete test-sequence-under-any-given-profile.~~]

[~~(h)~~] (g) Each holder of an operating license issued prior to (insert the effective date of this amendment) [~~must,~~] shall, by (insert a date 90 days after the effective date of this amendment), identify the safety-related electric equipment already qualified to [~~the-provisions-of-this-rule~~] and submit a schedule for the qualification to the provisions of this rule [~~testing~~] or replacement of the remaining safety-related electric equipment. This schedule must establish a goal of final environmental qualification by the end of the second refueling outage after March 31, 1982. The Director of Nuclear Reactor Regulation may grant requests for extensions of this deadline to a date no later than November 30, 1985, for specific pieces of equipment if [~~such~~] these requests are filed on a timely basis and demonstrate good cause for the extension, such as procurement lead time, test complications, and installation problems. In exceptional cases, the Commission itself may consider and grant extensions beyond November 30, 1985, for completion of environmental qualification.

[~~(i)~~] (h) Each licensee shall notify the Commission of any significant equipment qualification problem that may require extension of the completion date within [~~30~~] 60 days of its discovery.

~~[(j)--For-the-continued-operation-of-a-nuclear-plant,-each-holder-of an-operating-license-issued-prior-to-the-effective-date-of-this-rule-shall~~

(1) The operating license for the nuclear power plant was issued prior to [Insert effective date of this rule] and has the existing license conditions or technical specifications that require safety-related electric equipment to be qualified according to "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," November 1979, or NUREG-0588 (For Comment version), "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" and

(2) Qualification of safety-related electric equipment commenced prior to [insert effective date of this rule].

Dated at _____ this _____ day of _____, 1982.

For the Nuclear Regulatory Commission.

_____
Samuel J. Chilk
Secretary of the Commission

ENCLOSURE 2

2. §50.49(b) - Establishment of a Qualification Program

Comment: The rule should recognize previous submittals pursuant to the DOR Guidelines and NUREG-0588.

Response: The final rule has been expanded to recognize this concern. Requalification of electric equipment in accordance with this rule will not be required for equipment qualified in accordance with DOR Guidelines or NUREG-0588, provided the qualification of electric equipment has commenced prior to the effective date of the rule. See §50.49(k), which has been added in the final rule.

3. §50.49(c) - Scope of the Rule

A. Comment: This section seems to be much greater in scope as compared to NRC interim requirements.

Response: For clarity §50.49(k) has been added.

B. Comment: The scope of the proposed rule should be limited to Class 1E or safety related equipment.

Response: The scope of §50.49 is limited to "safety-related electric equipment," which is essentially Class 1E. A typical list of equipment and systems covered by this rule has been included in a revision to Regulatory Guide 1.89.

C. Comment: The scope should be reworded (47 FR 2878, Col. 2, Line 3) as: "... shutdown, maintain the integrity of the reactor coolant pressure boundary, containment isolation..."

Response: The language for the scope of the rule has been redrafted in terms of safety-related electric equipment. Safety-related electric equipment is essentially "Class 1E" equipment as defined in national standards.

D. Comment: The proposed rule introduces a new requirement to qualify "equipment needed to complete one path of achieving and maintaining a cold shutdown condition" and this modifies the licensing basis for the majority of operating nuclear power plants. A change of this magnitude, at this advanced stage of industry's qualification effort, most certainly introduces significant new costs and obligations with no demonstrated improvement in safety.

Response: The staff agrees that this is a new requirement that may introduce significant costs. The licensing basis of the majority of operating reactors does not require that all electric equipment and systems necessary to bring the reactor from normal operating conditions to cold shutdown be designed to Class 1E standards. Therefore, to require that all plants environmentally qualify the electric equipment and systems needed to complete one path of achieving and maintaining a cold shutdown condition may require the upgrading of a significant amount of equipment and systems that do not currently meet Class 1E standards for operating reactors.

However, electric equipment and systems necessary to shut down the reactor and maintain it in a safe shutdown condition are required to meet Class 1E standards and therefore, would be covered by the rule.

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

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

E. Comment: The scope includes, "... systems that should be qualified, those systems that could fail in such a way that would make a safety system unable to perform its function." The wording could also imply that qualification encompass systems that could mislead the operator to the extent that the required safety functions would not be accomplished. Qualification of non-safety instrumentation should not

be required where such instrumentation is not the primary source of data used by the operator in controlling events.

Response: The scope has been redrafted in terms of safety-related electric equipment. The interpretation that the qualification of non-safety instrumentation is not required if such instrumentation is not the primary source of data used by the operator in controlling events, is correct.

4. §50.49(d) - List of Equipment

A. Comment: There is no distinction made between equipment located in a harsh or mild environment.

Response: The final rule does not cover equipment located in a mild environment.

B. Comment: Lists of equipment which have been compiled in response to NRC bulletins and letters should be used instead of requiring generation of a new list in another data format. An alternative could be to identify on existing lists the equipment covered by this rule, and to reference other licensing documents, such as FSARs, design calculations, and equipment specifications, where additional information is available.

Response: It has been the experience of the staff that simply referencing other licensing documents as suggested in this comment

can result in uncoordinated and incomplete reviews of the qualification status of equipment. For this reason, a separate list of equipment covered by this rule is required.

See also response to comment 2.

C. Comment: Equipment located in a mild environment should be excluded from the proposed rule since the NRC has indicated that the stresses for this equipment would be less severe than for those in harsh environments.

Response: See response to comment 4.A.

5. §50.49(d)(1) - Performance Characteristics

A. Comment: Environmental qualification should not be limited to design basis events, but should consider Class IX accidents. Also, the rule omits the serious risk to electric equipment caused by internal missiles from pumps, valves, and burst pipes.

Response: Severe accidents (Class IX accidents) are being considered in other rulemakings. Environmental qualification does not include consideration of missiles. Protection against missiles must be provided in order to satisfy the requirements of GDC 4.

B. Comment: Structural integrity requirements should be deleted from the rule.

Response: Staff agrees. As long as the equipment can perform its safety function under postulated accident conditions, structural integrity is not the concern of this rule. Paragraph 50.49(d)(1) has been accordingly modified.

C. Comment: The terms "performance characteristics" and "structural integrity" are open to diverse interpretations. Suggestions have been made to use the terms "safety functional requirement," performance "specifications" or "the safety-related functions" in place of "performance characteristics."

Response: With regard to structural integrity, see response to comment 5.B. The term "performance characteristic" has been changed to "performance specifications".

D. Comment: The proposed requirement of paragraph (d)(1) is redundant, unnecessary, and arbitary since equipment technical specifications contain design criteria and requirements for safety equipment which is sufficient.

Response: See response to comment 4.B.

E. Comment: The required list of equipment should not include performance characteristics. This will lead to recording of extraneous information, diluting the importance of safety related parameters information.

Response: See response to comment 5.C. Performance characteristics are not extraneous information.

6. §50.49(d)(2) - Electrical Characteristics

A. Comment: Change "can" to "must" on last line of paragraph (d)(2).

Response: The use of word "can" is appropriate since the requirement in paragraph (d)(1) pertains only to the listing of performance specification of the equipment.

B. Comment: Requiring the "range" to be qualified is overly restrictive, unnecessary, and will have a large cost impact on testing. The ranges of the parameters are covered by performance requirements of pertinent national standards.

Response: Staff agrees. The words "The range of" have been deleted from paragraph 50.49(d)(2).

C. Comment: Delete paragraph (d)(2).

Response: See response to comment 6.B.

D. Comment: Testing conducted in the past typically did not consider all possible electrical conditions. Therefore, the requirements of paragraph (d)(2) should be removed from the proposed rule at least for equipment previously evaluated to the DOR Guidelines or NUREG-0588, Category II.

Response: See response to comment 2.

7. §50.49(d)(3) - Environmental Conditions

A. Comment: The term "where applicable" needs to be added after the list of environmental parameters in paragraph (d)(3).

Response: Paragraph (d)(3) states that the environmental conditions apply to the location where the equipment must perform. The staff recognizes that all the environmental parameters listed are not applicable at all equipment locations.

B. Comment: The term "chemical" is too broad and should either be defined or specific chemicals named.

Response: Clarification regarding qualification for chemical spray environments is given in paragraph 50.49(e)(3). Additional guidance is provided by Regulatory Guide 1.89.

C. Comment: Predicted variations in environmental conditions are not necessary if extreme conditions are identified and used in the qualification program.

Response: Extreme environmental conditions cannot be identified for some parameters, e.g., temperature and pressure, until their time-dependent variations have been predicted.

The proposed rule does not state that the use of identified extreme conditions, with appropriate margins, is unacceptable.

See also response to comment 7.E.

D. Comment: It is suggested that paragraph (d)(3) be supplemented with the following: "These environmental conditions may be determined using realistic inputs."

Response: The bases for determinating environmental conditions must be justifiable. Guidance in this area is provided in Regulatory Guide 1.89.

E. Comment: It is recommended that paragraph (d)(3) be deleted because of the phrase "the predicted variations of..."

Response: The requirement of paragraph (d)(1) concerning the predicted variations of environmental conditions with time has been deleted. Requirements in this area are specified for the individual environmental parameters elsewhere in this section.

8. §50.49(e)(1) - Temperature and Pressure

A. Comment: The phrase "most limiting" needs clarification.

Response: For clarity, the phrase "most limiting" is changed to "most severe."

B. Comment: For consistency, "design basis events" should be used in paragraph (e)(1) rather than "postulated accidents."

Response: Staff agrees. The rule has been modified accordingly.

9. §50.49(e)(2) - Humidity

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

Response: Staff agrees. Humidity variations during normal operation are difficult to predict. It has not been demonstrated that the time dependent variation in humidity will produce any differences in degradation of electric equipment. The rule has been modified accordingly.

10. §50.49(e)(3) - Chemical Effects

A. Comment: Since corrosion effects of various chemical components are generally well known, this paragraph should provide latitude to allow analysis that justifies using different chemical spray constituents or less severe concentrations than specified by plant environmental requirements.

Response: Analysis is acceptable if adequately justified.

11. §50.49(e)(4) - Radiation

A. Comment: In general, the aging and accident radiation cannot be combined, i.e., the word "plus" is misleading or incorrect since it implies integrated effects.

Response: Staff agrees. The word "plus" has been changed to "and."

B. Comment: The requirement that the dose rate be as in the power plant is totally impractical. The normal operation dose occurs over a 35 to 40 year period. Obviously dose rate acceleration must be permitted.

Response: The reference to dose rate has been deleted with regard to normal operation.

12. §50.49(e)(5) - Aging

A. Comment: The requirement that on-going qualification be done using "prototype equipment naturally aged" is overly restrictive and is not in harmony with (f). There are other, equally acceptable methods of extending qualified life and it is not appropriate to single out just one of them.

Response: Staff agrees. The rule has been modified.

B. Comment: The specific inclusion of aging requirents for electromechanical equipment is inappropriate in the rulemaking. Such details should be included in the revision to R.G. 1.89.

Response: Staff agrees. Reference to "electromechanical equipment" has been deleted from the rule and will be included in Regulatory Guide 1.89.

C. Comment: Use of accelerated aging to define a qualified life is not technically feasible.

Response: The staff believes that preconditioning by accelerated aging is technically feasible for simple electric equipment for plant life and for complex electric equipment for shorter designated life. Staff recognizes that state-of-the-art technology will be utilized in any aging program. Reference to "qualified life" has been deleted from the final rule.

13. §50.49(e)(7) - Synergistic Effects

Comment: "known synergistic effects...." must be considered. NRC should be more specific.

Response: The word "known" has been deleted from the rule.

14. §50.49(e)(8) - Margins

A. Comment: The proposed rule states that margins are used to account for inaccuracies in test instruments. Test instrument inaccuracies are a QA problem associated with required calibration programs and should not be encompassed under margins.

Response: Staff disagrees. The test instrument errors must be accounted for.

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

Response: Staff agrees. The paragraph on margin has been accordingly modified to recognize conservatisms that can be quantified.

15. §50.49(f) - Methods of Qualification

Comment: Qualification by analysis should not be allowed.

Response: Analysis "alone" is generally inadequate to demonstrate qualification, and type testing is the preferred qualification method. Although some analysis may be used, as identified in the rule, that analysis should be limited to extrapolations of data or to

analyzing similarities in equipment or materials. In any case, analytical assumptions should be verifiable or supported by test data.

16. §50.49(f)(2) - Testing of Similar Items and Analysis

Comment: Paragraph (f)(2) should state that it is acceptable to test a similar item of equipment under similar conditions with a supporting analysis that shows the equipment to be qualified is acceptable.

Response: The staff disagrees. The intent of paragraph (f)(2) is to cover both "similar" and "identical" environments.

17. §50.49(f)(3) - Experience and Analysis

Comment: Experience has proven to be of very limited use in qualification because of the lack of supporting documentation. It is suggested, therefore, that the words "Adequately documented" be inserted at the beginning of paragraph (f)(3).

Response: All information used to demonstrate the qualification of equipment, including test results, analytical assumptions, and experience with identical or similar equipment, must be adequately documented.

18. §50.49(f)(4) - Analysis

Comment: Are subparagraphs (f)(4)(i) and (f)(4)(ii) independent?

Response: Paragraph (f)(4)(i) has been deleted. See paragraph (f)(4) of the final rule.

19. §50.49(f)(4)(ii) - Analysis and Partial Test Data

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

Response: Staff agrees. The rule has been modified (see paragraph (f)(4) of the final rule).

20. §50.49(g) - Testing

A. Comment: This paragraph is written specifically for equipment employed for hostile environment applications and does not recognize alleviations appropriate for equipment located in mild environments.

Response: Environmental testing is not required for equipment located in mild environments. Paragraph (g) of the proposed rule has been deleted since it is too prescriptive. It will be included in Regulatory Guide 1.89. See response to comment 4.A.

B. Comment: Strict application of these requirements will negate testing already completed for earlier plants. The relief must be included in the Reg. Guide 1.89.

Response: See response to comment 2.

C. Comment: Paragraph 50.49g should be deleted as it limits the options available for qualification testing.

Response: This paragraph will be included in Regulatory Guide 1.89. See response to comment 20.A.

D. Comment: As written, this requirement applies to all equipment which has or will undergo qualification testing. This paragraph should not be applied to equipment which predated the requirements of IEEE 323-1974.

Response: See response to comment 2.

E. Comment: This paragraph should also make provisions for acceptance of testing that does not totally envelop all plant environmental conditions by supporting analysis.

Response: Paragraph (f)(2) covers "similar" conditions.

F. Comment: The detailed requirements for qualification by testing should not be contained in a rule, but should instead be discussed in Reg. Guide 1.89.

Response: See response to comment 20.A.

G. Comment: The first sentence should be changed to "If an item of electric equipment is to be qualified by test or analysis..."

Response: Staff disagrees. All of the requirements listed are not appropriate for analysis as a qualification method. See also response to comment 20.A.

21. §50.49(g)(1) - Acceptance Criteria

A. Comment: The requirement for acceptance criteria does not clearly say that they must be relevant. Acceptance criteria are application dependent.

Response: The staff disagrees that acceptance criteria are necessarily plant-dependent. However, this paragraph has been deleted. See response to comment 20.A.

B. Comment: The establishment of acceptance criteria before testing should be deleted. "Failure" is often a plant-specific consideration.

Response: The staff disagrees. Acceptance criteria, whether generic or specific, should be established prior to testing. See also response to comment 20.A.

C. Comment: If the documentation in paragraphs d(1), (2), and (3) is established, a clear record that the equipment provides the performance required will have been established. Therefore, this requirement for acceptance criteria should be eliminated.

Response: Staff disagrees. Paragraphs (d)(1), (d)(2), and (d)(3) refer to performance. Performance and acceptance criteria are not necessarily identical. See also response to comment 20.A.

D. Comment: This paragraph precludes reevaluating test criteria following the actual test. When equipment does not meet the acceptance criteria, system redesign, reconfiguration, and analysis should be allowed in order to verify that the initial acceptance criteria were in fact valid.

Response: The rule specifies the methods for demonstrating successful qualification. Failures during testing due to faulty test equipment or invalid acceptance criteria are outside the scope of the rule. See also response to comment 20.A.

22. §50.49(g)(2) - Demonstration by Test

A. Comment: Delete reference to paragraphs (d)(1), (d)(2), and (d)(3) concerning characteristics, electrical characteristics and environmental conditions, respectively.

Response: See response to of comment 21.C.

B. Comment: Paragraph g(2) requires that a radiation dose rate exposure profile vs. time be established and enveloped by the qualification testing. Testing at qualification dose rates exceeding accident dose rates, and total exposures exceeding the accident and normal exposure, is an overly conservative approach.

Response: See response to comment 20.A.

C. Comment: The radiation vs. time simulation requirement should be deleted from paragraph g(2).

Response: See response to comment 20.A.

D. Comment: The radiation dose rate should be simulated to the best extent possible within the limitations of the test facility and measuring instruments.

Response: See response to comment 20.A.

E. Comment: The rule should state that the accident radiation dose exposure with appropriate margin may be performed as a part of the preconditioning procedure. Also, margin need not be applied if the methods in Appendix D of NUREG-0588 have been employed.

Response: See response to comments 2 and 20.A.

F. Comment: Many utilities have carried out expensive qualification testing to service conditions unique to their plants in accordance with IEEE 323-1971 and demonstrated compliance with previous NRC regulations. New increased margins should not be applied to these existing tests.

Response: See response to comment 2.

23. §50.49(g)(3) - Test Profile

Comment: The option presented in this paragraph is excessive in its limitations. The envelope should not be that which results from any design basis event during any mode of operation but rather the envelope that results during any mode of operation during which the subject equipment must perform its function.

Response: See response to comment 20.A.

24. §50.49(g)(4) - Single Prototype

A. Comment: Does this section apply to aging also? For example, could parts of a component be aged separately, then assembled, then tested as per g(3)?

Response: The intent of paragraph (g)(4) is that the test stresses, e.g., aging and radiation, are not shared among two or more pieces of equipment. See response to comment 20.A.

B. Comment: Paragraph 50.49g(4) requires qualification by sequential test. Without direction on seismic and dynamic requirements, sequential tests cannot be done.

Response: See resolution of comments 1.A, 1.C, and 20.A.

C. Comment: This section may be interpreted as requiring MSLB and LOCA qualification tests of the same device. Testing to either is sufficient, provided the limiting accident is identified.

Response: Testing to the most limiting condition is acceptable. See response to comment 20.A.

D. Comment: Allowance for justifications for deviations from using the same piece of equipment throughout a test sequence should be allowed.

The present 50.49g(4) conflicts with the proposed Revision 1 of Regulatory Guide 1.89.

Response: See response to comments 2.A and 20.A.

25. §50.49(h) - Extension Date and Schedule Submission

A. Comment: The proposed rule's extended deadline for compliance with environmental criteria is unjustified and too liberal.

Response: In developing the position on the extension of the deadline for qualification of electric equipment, the NRC has considered information supplied by equipment vendors, utilities, test laboratories, consultants, and other interested parties. The amount of work, the availability of qualified personnel and equipment, and the impact on overall plant safety were factored into the Commission's decision to extend the deadline. Licensees have submitted information to the NRC showing that plants can be safely operated pending completion of the required environmental qualification.

B. Comment: Mild environment equipment should be excluded from the schedule for equipment testing or replacement to be submitted to the NRC 90 days after the effective date of the rule.

Response: Staff agrees. See response to comment 4.A.

Response: Prior submittals have not satisfied the requirements of paragraph 50.49(h). For example, the schedule for qualification had never before been required.

G. Comment: The words "but prior to November 30, 1985" in 50.49(h) and 50.49(k) should be deleted. As currently written, no recourse is provided for plants receiving operating licenses after November 30, 1985.

Response: Plants licensed after November 30, 1985, will be required to be in compliance with this rule.

H. Comment: The requirement to submit a schedule for the testing or replacement of equipment is not warranted. The date for submitting a schedule for testing and replacement has no safety significance whatsoever.

Response: The achievement of full qualification by the November 30, 1985, deadline depends on the early identification of deficiencies and a commitment to a firm plan for systematic corrective action.

I. Comment: The requirement for submission of schedules for qualification within 90 days of the rule should be revised to allow more time for mild environment equipment.

Response: See resolution of comment 4.A.

C. Comment: Within 90 days of the effective date of the rule, a schedule for "testing or replacement" of unqualified equipment is to be provided to the NRC. The word "testing" should be replaced by "qualification".

Response: Staff agrees. The word "testing" has been replaced by "qualification".

D. Comment: We assume the goal of final environmental qualification is for the second refueling outage starting after March 31, 1982.

Response: Staff agrees.

E. Comment: The requirement for "testing" of equipment identified in the submittal due 90 days after the publication of the final rule is inconsistent with 50.49(f) concerning qualification methods and with the proposed revision to Regulatory Guide 1.89 (Section C.5.9) regarding qualification in mild environments.

Response: Staff agrees. See resolution of comment 4.A and 20.A.

F. Comment: The rule should recognize that previous submittals to the NRC containing equipment identification and schedules for qualification are adequate for fulfilling the requirements in 50.49(h).

J. Comment: The proposed rule appears to require a new round of submittals (90 day letters) covering information that has already been submitted to the NRC. A statement should be included to indicate that this requirement applies only to plants that did not submit a 90-day response.

Response: Although the date for completion of environmental qualification would be extended by §50.49, new schedules for completion of qualification must be submitted. Duplicate submittals are not required.

26. §50.49(i) - Significant Problem Notification

A. Comment: The schedule for notification of the Commission of any significant qualification problem within 30 days of its discovery should be separated from the technical requirements of the rule.

Response: Staff disagrees. The purpose of this requirement is to provide advance notice and basis for possible extensions.

B. Comment: We believe the requirements to notify the Commission of potential problems within 30 days of discovery may be too stringent, particularly if a scheduled completion date is six months or longer from the date of discovery of a potential problem that may require extension.

Response: The staff agrees. See response to comment 26.C.

C. Comment: The notification period of 30 days to allow industry to evaluate minor qualification problems should be extended to 90 days. This would minimize the number of insignificant problems to be addressed by the Commission and industry.

Response: The staff agrees with the general point. The notification period has been extended to 60 days.

27. §50.49(j) - Justification for Continued Operation

A. Comment: The proposed rule requires "analyses" to justify continued operation with unqualified equipment. These analyses are vague and insubstantial and will allow licensees to rationalize the use of unsafe equipment based on its behavior during normal operating conditions.

Response: This paragraph has been deleted from the final rule. The licensees of the operating plants have justified the continued operation of nuclear power plants based on the criteria stated in paragraph (j) of the proposed rule.

B. Comment: The submittal of justification for continued operation should be required 90 or 180 days after the effective date of this rule, not on the effective date, to be consistent with the Supplementary Information section.

Response: See response to comment 27.A.

C. Comment: The provisions of the rule concerning justification for continued operation should be deleted as this information has been previously submitted in response to IE Bulletin 79-01B.

Response: Staff agrees. See resolution of comment 27A.

28. §50.49(j)(1) - Designated Alternative Single Failure Criterion/Partial Test Date

A. Comment: If redundant, qualified, "alternative" equipment is available to perform a safety function in lieu of unqualified equipment, then compliance with the regulation has already been achieved and the unqualified equipment may be exempted from the program. This requirement should be deleted.

Response: Staff disagrees. The terms "alternative" (or alternate) and "principal equipment" are used in the context of section 4.7.4.1 of IEEE 279-1971. Specifically, the alternative and principal equipment are mutually diverse (to protect against common-mode failures.) However, each set of equipment separately should meet the provisions of IEEE 279-1971. In this rule, the terms are not restricted to equipment in the protection systems.

B. Comment: The requirement for satisfaction of the single failure criterion for justification for continued operation is overly restrictive. If this requirement were met, no justification for interim operation would be needed.

Response: Staff agrees. The final rule has been modified. See paragraph 50.49(i)(1) of the final rule.

C. Comment: The phrase "and satisfies the single failure criterion" is unclear as used in this section. Also define the term "adequately qualified."

Response: The word "adequately" has been deleted. See response to comments 28.A and 28.B.

D. Comment: If there is designated alternative equipment which is qualified and satisfies the single failure criterion, the principal equipment need not be classified as safety related and hence need not be qualified.

Response: See response to comment 28.A.

E. Comment: The new rule states that partial test data may be used as justification for continued operation. Both this rule and the current requirements recognize that analysis and partial test data, appropriately applied, constitute qualification.

Response: Partial type test data and analysis, appropriately applied to envelop the predicted environmental conditions, are sufficient for qualification. If the test data are insufficient to demonstrate full qualification, partial test data may be utilized to justify continued operation.

function or misleading of the operator as a result of failure of equipment under the accident environment. Assurance of the above should comply with the Commission's intent.

Response: This section applies to relatively new power plants and assumes that the majority of the equipment is already fully qualified prior to the issuance of an operating license. This provision is intended to justify operation if alternative qualified equipment can compensate for the potential malfunction of relatively few items that may not be "fully" qualified.

31. §50.49(k) - Justification for Continued Operation for Near-Term Operating Licensees

A. Comment: The provision allowing applicants for new licenses (to be granted on or after the effective date of the amendment and prior to November 30, 1985) to submit "analyses" in lieu of test results to demonstrate environmental qualification should not be permitted. Licensees have been under directives to document the qualification of safety equipment since 1977.

Response: See response to comment 30.

B. Comment: Previous submittals by NTOLs pursuant to NUREG-0588 which contain justification for operation should be acknowledged.

Response: This rule does not require duplicate submittals.

29. §50.49(j)(4) - Completion of Safety Function

A. Comment: The proposed rule states that justification for continued operation may be determined if equipment performs its safety function prior to exposure to the accident environment, and subsequent equipment failures do not degrade the safety functions or mislead the operator. This should be sufficient for full qualification.

Response: A demonstration with appropriate margins that equipment fulfills the above requirements can constitute full qualification.

B. Comment: The evaluation of whether the failure of a single piece of equipment will, of itself, mislead the operator is subject to interpretation and engineering judgement. Because redundant equipment would be available, the justification for interim operation should not consider the aspect of unqualified instrumentation misleading the operator.

Response: Licensees should examine on a case-by-case basis the impact of equipment failures on operator actions. The licensees should decide whether the erroneous information subsequent to accomplishment of protection function can mislead the operator.

30. §50.49(j)(5) - [illegible]cant Degradation

Comment: One of the considerations for justification for continued operation is the occurrence of no significant degradation of a safety

32. §50.49(1) - Requirement of a Central File

A. Comment: The requirement to maintain a record identifying that the equipment meets its specific performance requirement exceeds the verification necessary to establish the performance of safety function.

Response: The qualification test by nature is limited to verifying the performance characteristics and not the actual safety function performed by the equipment, e.g., cooldown of a core.

B. Comment: The requirement for a central file should be for equipment located and potentially subject to a harsh environment only.

Response: See response to comment 4.A.

C. Comment: The requirement for a central file should be deleted because some records may be kept in the utility general file.

Response: This paragraph has been revised to require that auditable files permitting verification of qualifications be available. Certain records can be kept at vendor's shop.

D. Comment: The terms "application" and "specific performance requirements" should be changed to state that safety functions will be performed when subjected to the conditions predicted.

Response: See resolution of comment 32.A.

E. Comment: We suggest that it may be difficult, if not impossible, to obtain the record of qualification required, particularly for equipment in older plants, and we suggest that for equipment that has significant successful operating experience this record should not be necessary.

Response: The successful operating experience does not necessarily qualify equipment for accident conditions. See paragraph (f)(3).

F. Comment: The contents of the central file may vary considerably depending on whether the file is a record of qualification to the harsh or mild environment. Recognition of content requirements by reference to any proposed regulatory guide would be appropriate.

Response: See response to comment 32.B.

G. Comment: Central file qualification information should include equipment in a harsh environment only and should only support the equipment's ability to perform its safety function.

Response: See response to comments 4.A, 32.A, and 32.B.

H. Comment: Please clarify as to exactly where the licensee shall maintain qualification records, particularly with respect to files which are proprietary to the NSSS vendor.

Response: It is acceptable to keep the qualification file for NSSS equipment at the NSSS vendor provided the file is maintained in an auditable form for the entire period during which the covered item is installed in the nuclear power plant.

33. Supplementary Information

A. Comment: The term "important to safety" should be replaced by "Class IE" throughout this rule.

Response: See response to comment 3.B.

B. Comment: The term "safety-related" should be used in place of "important to safety."

Response: "Safe-related" equipment is a subset of equipment "important to safety." The scope of the final rule is limited to safety-related electric equipment. Expansion of the scope of this rule to include additional equipment important to safety will be subject of a future rulemaking.

C. Comment: The scope of the proposed rule should include all electric equipment "important to safety" since that is the same as "safety-related" or "safety-grade" equipment.

Response: See response to comments 3.C and 33.B.

34. Qualification History

A. Comment: It should be noted in the rule that prior to 1971 qualification of electric and electronic equipment was based on [the] use of good engineering practices which included conservative application and design, high quality equipment, and some environmental testing.

Response: Staff agrees with the comment. However, the additional details are inappropriate in the final rule.

35. Basis of Rule

A. Comment: The proposed rule is primarily based on NUREG 0588 Category I. Therefore, it is appropriate that this rule clarifies and recognizes the fact that equipment evaluated in accordance in accordance with [the] DOE guidelines and NUREG-0588 Category II are considered to satisfy the requirements of this rule.

Response: See response to comment 2.

B. Comment: The Federal Register notice states that this rule codifies existing requirements and imposes no new costs or obligations on utilities. We take strong exception to this statement.

Response: The new rule will codify the current requirements in the DOR Guidelines and NUREG-0588.

C. Comment: The rule does not recognize that operating plants have just completed qualification of equipment to the DOR Guidelines or NUREG-0588 Category II.

Response: See response to comment 2.

D. Comment: The Supplementary Information section should also state that the requirements of IE Bulletin 79-01B are being codified.

Response: Reference to DOR Guidelines includes IE Bulletin 79-01B.

36. Replacement Parts

A. Comment: The rule does not address replacement parts.

Response: Specific guidance on replacement parts will be included in Regulatory Guide 1.89.

ENCLOSURE 3

VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

The applicant (licensee) of a nuclear power plant is required by the Commission's regulations to verify that structures, systems, and components important to safety will perform their intended functions in spite of the environments that may result from anticipated operational occurrences or postulated accidents. This verification includes environmental qualification by test, operating experience, and analysis, or a combination of these. The proposed rule sets forth the Commission's requirements for the environmental qualification of safety-related electric equipment by test and analysis.

1.2 Need for Proposed Action

The current general requirements for qualification of electric equipment important to safety are found in General Design Criteria 1, 2, 4, and 23 of Appendix A to Part 50; Sections III and XI of Appendix B to Part 50; and Paragraph 50.55a(h) of Part 50, which incorporates by reference IEEE 279-1971,* "Criteria for Protection Systems for Nuclear Power Generating Stations." The NRC has used several methods to ensure that these general requirements are met for safety-related electric equipment. Prior to 1971, qualification was based on the fact that the electric components were of high industrial quality. For nuclear plants licensed to operate after 1971, qualification was judged on the basis of IEEE 323-1971. However, no regulatory guide was ever issued endorsing IEEE 323-1971, although some of the plants referenced the standard in their licensing submissions to the Commission. For the plants whose safety evaluation reports were issued after July 1, 1974, the Commission has issued Regulatory Guide 1.89, which endorses IEEE 323-1974* subject to supplementary provisions.

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

Currently, the Commission has under way a program to reevaluate the qualification of safety-related electric equipment in all operating reactors. As part of this program, the staff has developed more definitive criteria for the environmental qualification. The Division of Operating Reactors (DOR) issued "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" in November 1979. In addition, for reactors under licensing review, the staff has issued NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

In its Memorandum and Order CLI-80-21 issued on May 23, 1980, the Commission endorsed the staff's actions to use the DOR Guidelines to review operating plants and NUREG-0588 to review plants under licensing review. Further, the Commission ordered that these two documents form the basis for requirements that licensees and applicants must meet in order to satisfy those aspects of Appendix A to 10 CFR Part 50 that relate to the environmental qualification of electric equipment. The Commission also ordered that licensees of operating reactors must comply with these requirements so that the applicable equipment in all operating plants will meet the DOR Guidelines or NUREG-0588.

1.3 Value/Impact of Proposed Action

1.3.1 NRC Operations

Since regulations specifically setting forth requirements for the qualification of safety-related electric equipment in new and operating plants have never been issued, the proposed action should result in more effective effort by the staff in reviewing applications for construction permits and operating licenses and in the backfitting of these requirements to operating plants. The proposed action will codify an NRC position by taking advantage of previous staff effort (1) in completion of a generic activity (A-24), "Qualification of Class 1E Safety-Related Equipment," (2) in the preparation of the DOR Guidelines and NUREG-0588, (3) in IEEE standards committee work, and (4) in the development, funding, and monitoring of related research programs.

There should be little impact on the staff at the time the rule is approved. Approximately two man-years of effort have been spent in preparation of the rule.

1.3.2 Other Government Agencies

Not applicable, unless a government agency is the applicant.

1.3.3 Industry

The licensees and applicants currently must meet the requirements for qualification of safety-related electric equipment in accordance with the Commission's Memorandum and Order CLI-80-21. If the final rule is published as now presented, the rule will not have significant impact on industry because of backfit.

The value of this rule is that the industry will have clearly specified requirements to follow with respect to the qualification of safety-related electric equipment for new and existing plants. This, in turn, should ease the licensing process for industry by eliminating delays resulting from misinterpretation of NRC's requirements.

1.3.4 Public

The proposed action will improve public safety by further ensuring that electric equipment will perform its safety functions in spite of environments that may result from design basis events. These is no perceived impact on the public.

1.4 Decision on Proposed Action

The proposed action has been mandated by the Commission in its Memorandum and Order CLI-80-21 dated May 23, 1980.

2. TECHNICAL APPROACH

The technical approach will be to codify the programs of the DOR Guidelines and NUREG-0588.

3. PROCEDURAL APPROACH

Rulemaking has been mandated by the Commission in its Memorandum and Order cited above.

4. STATUTORY CONSIDERATIONS

4.1 NRC Authority

Authority for this rulemaking is derived from the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended.

4.2 Need for NEPA Assessment

The proposed action does not require an environmental impact statement in accordance with 51.5(d)(3) of 10 CFR Part 51.

5. RELATIONSHIP TO OTHER EXISTING OR PROPOSED REGULATIONS OR POLICIES

No conflicts or overlaps with requirements promulgated by other agencies are foreseen.

6. SUMMARY AND CONCLUSIONS

This rule mandated by the Commission will be effective upon publication, which is expected prior to June 30, 1982.

ENCLOSURE 4

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

February 1982
Division 1
Task EE 042-2

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

PROPOSED REVISION 1 TO REGULATORY GUIDE 1.89

ENVIRONMENTAL QUALIFICATION OF ELECTRIC EQUIPMENT FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

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

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

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

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

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

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

B. DISCUSSION

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

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

It is essential that equipment be qualified to meet its performance requirements under the environmental and operating conditions in which it will be required to function and for the length of time its function is required. The following are examples of considerations to be taken into account when determining the environment for which the equipment is to be qualified: (1) equipment outside containment would generally see a less severe environment than equipment inside containment; (2) equipment whose location is shielded from a radiation source would generally receive a smaller radiation dose than equipment at the same distance from the source but exposed to its direct radiation; (3) equipment required to initiate protective action would generally be required for a shorter period of time than instrumentation required to follow the course of an accident. The specific environment for which individual equipment must be qualified will depend on the installed location, the conditions under which it is required to function, and the length of time (with margin) it is required to operate.

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

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

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

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

Item (12) of Regulatory Position C.4.c addresses qualification of equipment exposed to low-level radiation doses. Numerous studies that have compiled radiation effects data on all classes of organic compounds show that compounds with the least radiation resistance have damage thresholds greater than 10^4 rads and would remain functional with exposures somewhat above the threshold value. Thus, for organic materials, radiation qualification may be readily justified by existing test data or operating experience for radiation exposures below 10^4 rads. However, for electronic components, studies have shown failures in metal oxide semiconductor devices at somewhat lower doses. Therefore, radiation qualification for electronic components may have a lower exposure threshold.

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

C. REGULATORY POSITION

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

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

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

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

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

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

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

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

(a) Methods for calculating mass and energy release rates for LOCAs and MSLBs are summarized in Appendix B to this guide. The calculations

should account for the time dependence and spatial distribution of these variables. For example, superheated steam followed by saturated steam may be a limiting condition and should be considered.

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

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

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

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

b. Effects of Chemicals

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

c. Radiation Conditions Inside and Outside Containment

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

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

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

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

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

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

(2) For all other design basis accidents (e.g., non-LOCA high-energy line breaks or rod ejection or rod drop accidents) the qualification source term calculations should use the percentage of fuel damage assumed in the plant-specific analysis (provided in the FSAR). When only fuel clad perforation is postulated, the nuclide inventory of the fuel elements breached should be calculated at the end of core life, assuming continuous full-power operation. The fuel rod gap inventory should be assumed to be 10% of the total rod activity inventory of iodine and 10% of the total activity inventory of noble gases (except for Kr-85, for which a release of 30% should be assumed). All the gaseous constituents in the gaps of the breached fuel rods should be assumed instantaneously released to the primary coolant. When fuel melting is postulated, the activity inventory of the melted fuel elements should also be calculated at the end of core life assuming full-power operation.

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

(3) For a limited number of accident-monitoring instrumentation channels with instrument ranges that extend well beyond the values the selected variables can attain under limiting conditions as specified in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," the source term should assume an initial release to the containment that considers the fission product release groups associated with grossly melted fuel. Acceptable assumptions for the fractional release for each group are: noble gases, 100%; I, Br, 100%; Cs, Rb, 100%; Te, 100%; Sr, Ba, 11%; Ru, 8%; and La, 1.3% (individual nuclides are listed in Table VI 3-1 of WASH-1400). The effect of natural and mechanical containment fission product removal may be considered on a best-estimate basis to determine the rate of redistribution of the various groups from the containment atmosphere to other locations.

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

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

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

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

plateout of the iodine released from the core should not be made. Removal of iodine from surfaces by steam condensate flow or washoff by the containment spray may be assumed if such effects can be verified and quantified by analysis or experiment.

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

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

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

(9) Shielded components need be qualified only to the gamma radiation dose or dose rate required provided an analysis or test shows that the sensitive portions of the component or equipment are not exposed to significant beta radiation dose rates or that the effects of beta radiation heating and secondary radiation have no deleterious effects on component performance.

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

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

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

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

d. Environmental Conditions for Equipment Outside Containment

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

(2) Equipment located in general plant areas outside containment where equipment is not subjected to a design basis accident environment should be qualified to the normal and abnormal range of environmental conditions postulated to occur at the equipment location.

(3) Equipment not served by environmental support systems within the scope of this guide or served by other systems within the scope of this guide that may be secured during plant operation or shutdown should be qualified to the limiting environmental conditions that are postulated for that location, assuming a loss of the environmental support system.

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

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

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

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

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

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

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

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

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

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

a. Quantified margins should be applied to the design parameters discussed in Regulatory Position C.4 to ensure that the postulated accident

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

b. Some equipment may be required by the design to perform its safety function only within a short time period into the event (i.e., less than 10 hours), and, once its function is completed, subsequent failures are shown not to be detrimental to plant safety. Other equipment may not be required to perform a safety function but must not fail within a short time period into the event, and subsequent failures are also shown not to be detrimental to plant safety. Equipment in these categories should remain functional in the accident environment for a period of at least 1 hour in excess of the time assumed in the accident analysis. For all other equipment (e.g., post-accident monitoring, recombiners), the 10 percent time margin identified in Section 6.3.1.5 of IEEE Std 323-1974 should be used.

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

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

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

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

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

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

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

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

h. Qualified life should be established on the basis of the severity of the testing performed, the conservatisms employed in the extrapolation of data, the operating history, and the other methods that may reasonably be used. All assumptions should be documented.

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

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

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

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

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

D. IMPLEMENTATION

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

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

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

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

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

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

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

APPENDIX A

TYPICAL EQUIPMENT OR FUNCTIONS FOR ACCIDENT MITIGATION

Engineered Safety Feature Actuation

Reactor Protection

Containment Isolation

Steamline Isolation

Main Feedwater Shutdown and Isolation

Emergency Power

Emergency Core Cooling[1]

Containment Heat Removal

Containment Fission Product Removal

Containment Combustible Gas Control

Auxiliary Feedwater

Containment Ventilation

Containment Radiation Monitoring

Control Room Habitability System (e.g., HVAC, Radiation Filters)

Ventilation for Areas Containing Safety Equipment

Component Cooling

Service Water

Emergency Systems to Achieve Safe Shutdown

Postaccident Sampling and Monitoring[2]

Radiation Monitoring[2]

Safety-Related Display Instrumentation[2]

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

[2]More specific identification of these types of equipment can be found in the plant emergency procedures and in Tables 1 and 2 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," Categories 1 and 2.

APPENDIX B
METHODS FOR CALCULATING MASS AND ENERGY RELEASE

Loss-of-Coolant Accident

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

1. Topical Report WCAP-8312A for Westinghouse plants.
2. Section 6.2.1 of CESSAR System 80 PSAR for Combustion Engineering plants.
3. Appendix 6A of B-SAR-205 for Babcock & Wilcox plants.
4. NEDO-10320 and Supplements 1 & 2 for General Electric plants. NEDO-20533 dated June 1974 and Supplement 1 dated August 1975 (GE Mark III).

Main Steam Line Break

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

1. Topical Report WCAP-8822 for Westinghouse plants. (Although this Topical Report is currently under review, the use of this method is acceptable in the interim if no entrainment is assumed. Reanalysis may be required following the NRC staff review of the entrainment model as presently described.)
2. Appendix 6B of CESSAR System 80 PSAR for Combustion Engineering plants.
3. Section 15.1.14 of B-SAR-205 for Babcock & Wilcox plants.
4. Same as item 4 above for General Electric plants.

APPENDIX C

SAMPLE CALCULATION AND METHODOLOGY FOR RADIATION QUALIFICATION DOSE

This appendix illustrates the staff model for calculating dose rates and integrated doses for equipment qualification purposes. The doses shown in Figure B-1 include contributions from airborne and plateout radiation sources in the containment and cover a period of one year following the postulated fission product release. The dose values shown here are provided for illustration only and may not be appropriate for plant-specific application for equipment qualification levels. The dose levels intended for qualification purposes should be determined using the maximum time the equipment is intended to function, which, for the design basis loss-of-coolant accident (LOCA), may well exceed one year.

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

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

1. Basic Assumptions Used in the Analysis

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

Following the accident at Three Mile Island Unit 2 (TMI-2), the staff concluded that a thorough examination of the source term assumptions for equipment qualification was warranted. It is recognized, however, that the TMI-2 accident represents only one of a number of possible accident sequences leading to a release of fission products and that the mix of fission products released under various core conditions could vary substantially. Current rulemaking proceedings are reevaluating plant siting policy, degraded cores, minimum requirements for engineered safety features, and emergency preparedness. These rulemaking activities also included an examination of fission product releases under degraded core conditions. While the final resolution of the source term assumptions is conditioned on the completion of these rulemaking efforts, the staff believes it is prudent to incorporate the knowledge gained of fission product behavior from the TMI-2 accident in defining source term assumptions for equipment qualification.

Based on release estimates in the Rogovin Report (Ref. 1), the staff assumptions for noble gas and iodine releases still appear to be conservative. However, the report estimates that the TMI-2 release contained between 40 and 60 percent of the Cs-134 and Cs-137 core activity in the primary system water, in the containment sump water, and in auxiliary building tanks. Comparison of the integrated dose from the TMI-2 cesium release to the previous staff assumption of "1% solids" shows that assuming "1% solids" may not result in a conservative estimate for the radiation exposure for equipment required to function for time periods exceeding thirty days. The staff feels that as a first step toward modification of the TID-14844 source term in the direction indicated by the TMI-2 experience, it may be prudent to include a cesium release in addition to the previously assumed "1% solids." As a result, the revised regulatory positions propose a release of 50% of the core cesium activity inventory (see Regulatory Position C.4.c items (1) and (2)). The assumed cesium release implies no substantial departure from, and is consistent with, the degraded core conditions previously implied by the assumed release of 50% of the core iodine activity. This change in assumption would have particular significance for the qualification of equipment in the vicinity of recirculating fluids and for equipment required to function for time periods exceeding 30 days.

The assumption of concurrent release of cesium and iodine also is consistent with the findings of recent source term studies reported in NUREG-0772 (Ref. 2). This report also concluded that the predominant form of the iodine released during accidents is cesium iodide (CsI). Although the CsI form is not specifically addressed in this report, it is evident that either CsI or I_2 and Cs would, in the long term, be located primarily in the reactor water and the containment sump water of a PWR or the suppression pool of a BWR. The staff recognizes that the revised source terms contained in this report are interim values and that the conclusions from the report cited above, as well as further results from current research efforts in the source term area, should ultimately form the basis for any revision of source term assumptions. Any revision of the source term assumptions, such as the inclusion of additional radionuclides, would be incorporated into this guide before it is issued as an active guide.

2. Assumptions Used in Calculating Fission Product Concentrations

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

The staff used a computer program, TACT, to model the time-dependent behavior of iodine and noble gases within a nuclear power plant. The TACT code or other equivalent industry codes would provide an acceptable method for modeling the transfer of activity from one containment region to another and in modeling the reduction of activity due to the action of ESFs. Another staff code, SPIRT (Ref. 3), is used to calculate the removal rates of elemental iodine by plateout and sprays. These codes were used to develop the source term estimates. The assumptions in the following sections were used to calculate the distribution of radioactivity within the containment following a design basis LOCA.

2.1 PWR Dry Containments

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

1. The source terms used in the analysis assumed that 50% of the core iodines and 100% of the core noble gases were released instantaneously to the containment atmosphere and 50% of the core cesium and 1% of the remaining "solid" activity inventory were released from the core and carried with the primary coolant directly to the containment sump.

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

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

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

5. Air exchange between the sprayed and unsprayed region was assumed to be one-half of the design flow rate of ESF fans. Good mixing of the containment activity between the sprayed and unsprayed regions is ensured by natural convection currents and ESF fans.

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

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

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

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

10. The spray removal and plateout processes were modeled as competing iodine removal mechanisms.

11. A spray removal rate constant (λ) for particulate iodine concentration was calculated using the staff's SPIRT program (Ref. 3). The staff calculated a value of $\lambda = 0.43$ hr^{-1} and allowed the removal of particulate iodine to continue until the airborne concentration was reduced by a factor of 10^4. The organic iodine concentration in the containment atmosphere is assumed not to be affected by either the containment spray or plateout removal mechanisms.

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

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

14. The analysis assumed that more than one specie of radioactive iodine is present in a design basis LOCA. The calculation of the post-LOCA environment assumed that, of the 50% of the core inventory of iodine released, 2.5% is associated with airborne particulate materials and 2% formed organic compounds. The remaining 95.5% remains as elemental iodine. For conservatism, this composition was assumed present at time t = 0. (These assumptions concerning the iodine form are consistent with those of Regulatory Guides 1.3,

"Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," when a plateout factor of 2 is assumed for the elemental form.)

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

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

2.2 PWR Ice Condenser Containments

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

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

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

3. Removal of airborne iodine in the upper compartment of the containment by the action of both plateout and spray processes may be assumed provided these removal processes are evaluated using conditions and assumptions consistent with items 8 through 12 in Section 2.1 and plant-specific parameters.

2.3 BWR Containments

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

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

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

3. Removal of airborne iodine in the drywell or reactor building by the action of both plateout and spray processes may be assumed provided the effectiveness of these competing iodine removal processes are evaluated using conditions and assumptions consistent with items 8 through 12 in Section 2.1 and plant-specific parameters.

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

3. Model for Calculating the Dose Rate of Airborne and Plateout Fission Products

The beta and gamma dose rates and integrated doses from the airborne activity within the containment atmosphere were calculated for the midpoint in the containment. The containment was modeled as a cylinder with the height and diameter equal. Containment shielding and internal structures were neglected

because they would involve a degree of complexity beyond the scope of the present work. The calculations of Reference 4 indicate that the specific internal shielding and structure would be expected to reduce the gamma doses and dose rates by factors of two or more depending on the specific location and geometry.

Because of the short range of the betas in air, the airborne beta doses were calculated using an infinite medium approximation. This is shown in Reference 5 to result in only a small error. Beta doses for equipment located on the containment walls or on large internal structures may be calculated using the semi-infinite beta dose model.

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

Finally, all gamma doses were multiplied by a correction factor of 1.3 as suggested in Reference 5 to account for the omission of the contribution from the decay chains of the isotopes.

4. Model for Calculating the Dose Rate of Sump Fission Products

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

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

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

5. Conclusion

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

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

Table C-1
SUMMARY TABLE OF ESTIMATES FOR TOTAL AIRBORNE GAMMA DOSE
CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

Time (Hr)	Airborne Iodine Dose (R)	Airborne Noble Gas Dose (R)	Plateout Iodine Dose (R)	Total Dose (R)
0.00	-	-	-	-
0.03	4.82E+4	7.42E+4	1.69E+3	1.24E+5
0.06	8.57E+4	1.39E+5	3.98E+3	2.29E+5
0.09	1.09E+5	1.98E+5	7.22E+3	3.14E+5
0.12	1.25E+5	2.51E+5	1.10E+4	3.87E+5
0.15	1.38E+5	3.01E+5	1.52E+4	4.54E+5
0.18	1.47E+5	3.48E+5	1.96E+4	5.15E+5
0.21	1.55E+5	3.92E+5	2.41E+4	5.71E+5
0.25	1.64E+5	4.49E+5	3.03E+4	6.43E+5
0.38	1.87E+5	6.19E+5	5.05E+4	8.57E+5
0.50	2.03E+5	7.61E+5	6.90E+4	1.03E+6
0.75	2.36E+5	1.03E+6	1.06E+5	1.37E+6
1.00	2.66E+5	1.26E+6	1.40E+5	1.67E+6
2.00	3.62E+5	2.04E+6	2.61E+5	2.66E+6
5.00	5.50E+5	3.56E+6	5.40E+5	4.65E+6
8.00	6.63E+5	4.38E+6	7.47E+5	5.79E+6
24.0	1.01E+6	6.26E+6	1.45E+6	8.72E+6
60.0	1.31E+6	7.16E+6	2.10E+6	1.06E+7
96.0	1.45E+6	7.56E+6	2.39E+6	1.14E+7
192.	1.68E+6	8.29E+6	2.86E+6	1.28E+7
298.	1.85E+6	8.76E+6	3.19E+6	1.38E+7
394.	1.95E+6	8.85E+6	3.41E+6	1.42E+7
560	2.07E+6	9.06E+6	3.64E+6	1.48E+7
720.	2.13E+6	9.15E+6	3.76E+6	1.50E+7
888.	2.16E+6	9.19E+6	3.83E+6	1.52E+7
1060	2.18E+6	9.21E+6	3.87E+6	1.53E+6
1220	2.19+E6	9.21E+6	3.89E+6	1.53E+7
1390	2.20E+6	9.21E+6	3.90E+6	1.53E+7
1560	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1730	2.20E+6	9.22E+6	3.91E+6	1.53E+7
1900	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2060	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2230	2.20E+6	9.22E+6	3.92E+6	1.53E+7
2950	2.20E+6	9.23E+6	3.92E+6	1.54E+7
3670	2.20E+6	9.24E+6	3.92E+6	1.54E+7
4390	2.20E+6	9.24E+6	3.92E+6	1.54E+7
5110	2.20E+6	9.25E+6	3.92E+6	1.54E+7
5830	2.20E+6	9.25E+6	3.92E+6	1.54E+7
6550	2.20E+6	9.26E+6	3.92E+6	1.54E+7
7270	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8000	2.20E+6	9.27E+6	3.92E+6	1.54E+7
8710	2.20E+6	9.28E+6	3.92E+6	1.54E+7
			TOTAL	1.54E+7

Table C-2
SUMMARY TABLE OF ESTIMATES FOR TOTAL AIRBORNE BETA DOSE CONTRIBUTORS IN CONTAINMENT TO A POINT IN THE CONTAINMENT CENTER

Time (hr)	Airborne Iodine Dose (rads)*	Airborne Noble Gas Dose (rads)*	Total Dose (rads)*
0.00	-	-	-
0.03	1.47E+5	5.48E5	6.95E+5
0.06	2.62E+5	9.86E+5	1.25E+6
0.09	3.33E+5	1.35E+5	1.68E+6
0.12	3.83E+5	1.65E+6	2.03E+6
0.15	4.20E+5	1.91E+6	2.33E+6
0.18	4.49E+5	2.14E+6	2.59E+6
0.21	4.73E+5	2.35E+6	2.82E+6
0.25	5.00E+5	2.60E+6	3.10E+6
0.38	5.67E+5	3.30E+6	3.87E+6
0.50	6.15E+5	3.86E+6	4.48E+6
0.75	7.13E+5	4.89E+6	5.60E+6
1.00	8.00E+5	5.81E+6	6.61E+6
2.00	1.07E+6	9.02E+6	1.01E+7
5.00	1.58E+6	1.65E+7	6.54E+7
8.00	1.88E+6	2.20E+7	2.39E+7
24.0	2.87E+6	4.08E+7	4.37E+7
60.0	3.89E+6	6.15E+7	6.54E+7
96.0	4.37E+6	7.48E+7	7.92E+7
192	5.14E+6	1.00E+8	1.05E+8
298	5.64E+6	1.17E+8	1.23E+8
394	5.99E+6	1.25E+8	1.31E+8
560	6.34E+6	1.34E+8	1.40E+8
720	6.53E+6	1.39E+8	1.46E+8
888	6.63E+6	1.42E+8	1.49E+8
1060	6.69E+6	1.44E+8	1.51E+8
1220	6.73E+6	1.45E+8	1.52E+8
1390	6.75E+6	1.47E+8	1.54E+8
1560	6.76E+6	1.49E+8	1.56E+8
1730	6.76E+6	1.51E+8	1.58E+8
1900	6.76E+6	1.52E+8	1.59E+8
2060	6.76E+6	1.54E+8	1.61E+8
2230	6.77E+6	1.55E+8	1.62E+8
2950	6.77E+6	1.62E+8	1.69E+8
3670	6.77E+6	1.69E+8	1.76E+8
4390	6.77E+6	1.76E+8	1.83E+8
5110	6.77E+6	1.83E+8	1.90E+8
5830	6.77E+6	1.89E+8	1.96E+8
6550	6.77E+6	1.96E+8	2.03E+8
7270	6.77E+6	2.03E+8	2.10E+8
8000	6.77E+6	2.09E+8	2.16E+8
8710	6.77E+6	2.16E+8	2.23E+8
		Total	2.23E+8

*Dose conversion factor is based on absorption by tissue.

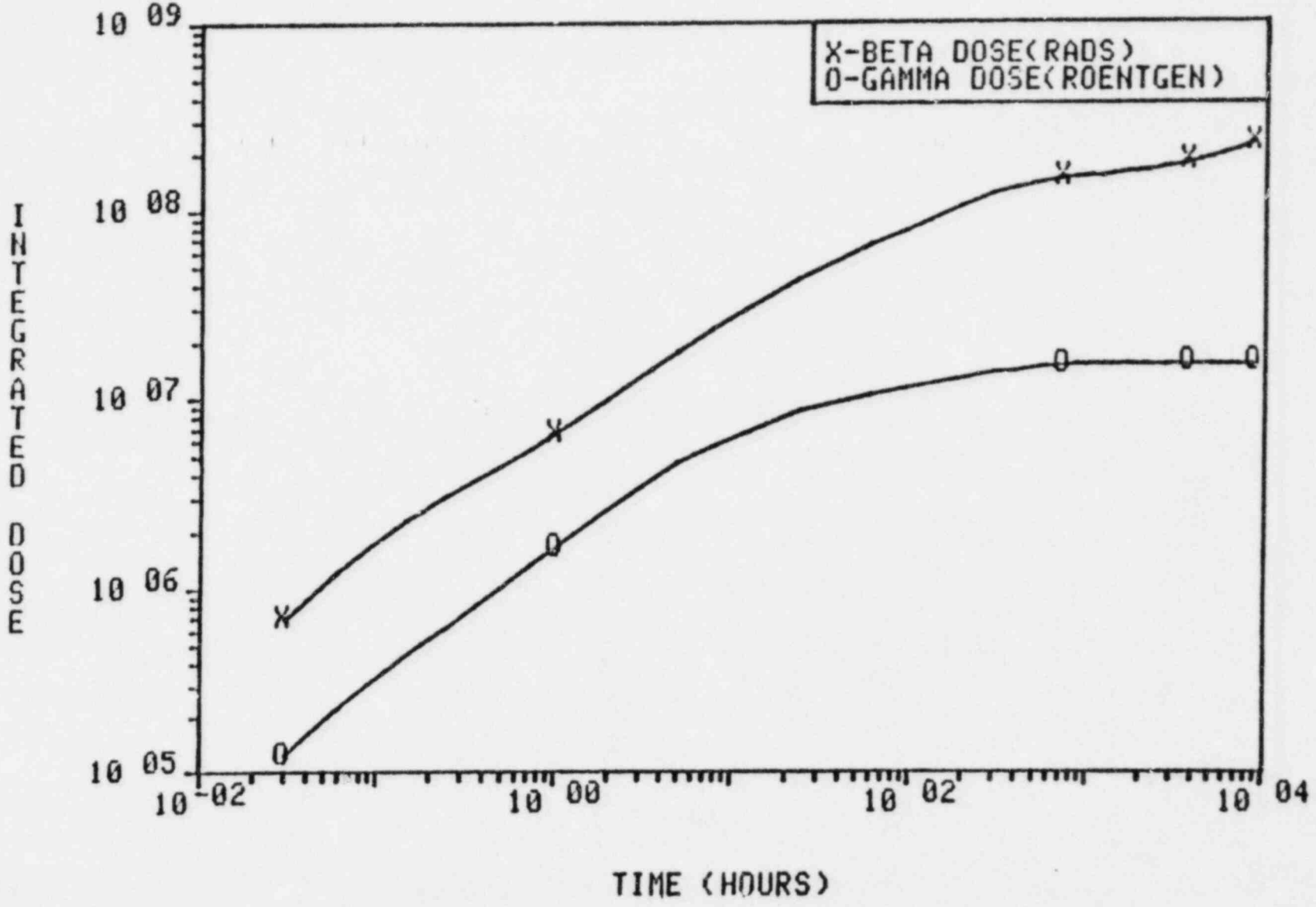

Figure C-1 Sample airborne doses for a dose point on the containment centerline

APPENDIX C

REFERENCES

1. Mitchell Rogovin, George T. Frampton, Jr., et al., "Three Mile Island--A Report to the Commissioners and to the Public," NUREG/CR-1250, Volume II, Part 2, April 5, 1979. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

2. "Technical Basis for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981. Available for purchase from the National Technical Information Service, Springfield, Virginia 22161.

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

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

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

BIBLIOGRAPHY

Kocher, D. C., 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 the National Technical Information Service, Springfield, Virginia 22161.

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

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

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

APPENDIX D
QUALIFICATION DOCUMENTATION FOR ELECTRIC EQUIPMENT

In order to ensure that an environmental qualification program conforms to General Design Criteria 1, 2, 4, and 23 of Appendix A and Sections III and XI of Appendix B, and § 50.49 of 10 CFR Part 50 and to the national standards mentioned in Part II, "Acceptance Criteria," of Standard Review Plan Section 3.11 (which includes IEEE Std 323), the following information on the qualification program is required for electric equipment within the scope of this guide.

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

2. For each item of equipment identified in 1, provide the following:
    a. Type (functional designation)
    b. Manufacturer
    c. Manufacturer's type number and model number

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

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

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

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

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

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

a. The system safety function requirements.

b. An environmental envelope as a function of time that includes all extreme parameters, both maximum and minimum values, expected to occur

during plant shutdown, normal operation, abnormal operation, and any design basis event (including LOCA and MSLB), including postevent conditions.

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

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

5. Provide the qualification test plan, test setup, test procedures, and acceptance criteria for at least one of each group of equipment in item 1 as appropriate to the category identified in item 3. If any method other than type testing was used for qualification (operating experience, analysis, combined qualification, or ongoing qualification), describe the method in sufficient detail to permit evaluation of its adequacy.

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

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

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

DRAFT VALUE/IMPACT STATEMENT

Background

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

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

Substantive Changes and Their Value/Impact

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

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

   b. Regulatory Position C.3, which provides the staff position regarding the various qualification methods (e.g., test, operating experience,

analysis, on-going qualification). Testing should be the primary method. The other methods, when used, should be supported by test data.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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