PDR-016

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

November 8, 1982

Diane Curran, Esquire
Harmon & Weiss
1725 I Street, N.W.
Suite 506
Washington, DC 20006

IN RESPONSE REFER
TO FOIA-82-426

Dear Ms. Curran:

This is in further response to your letter dated September 10, 1982 in which you requested, pursuant to the Freedom of Information Act, copies of all documents relied upon by the Commission in promulgating the final rule on environmental qualification and copies of SECY-82-207A, 207B, 207C, and other amendments. This also responds to your letter dated November 2, 1982 in which you requested a releasability determination of 21 documents referred to in my letter to you dated October 14, 1982.

A releasability determination concerning each of those documents has been completed. For ease of accounting, I shall refer to the document numbers as they appeared in the appendix to my October 14, 1982 letter to you. A copy of that appendix is also enclosed.

Documents numbered 28, 35, 41, 45, 53, 66, 70 and 73 are being released in their entirety and have been sent to the NRC Public Document Room (PDR) located at 1717 H Street, N.W., Washington, D.C. where they will be available for pubic inspection and copying. Document number 65 was incorrectly listed. In fact, it is the cover pages for SECY-82-207A, which is already available in the PDR.

Enclosure 1 of document 12, a working copy of the draft proposed rule "Environmental and Seismic Qualification of Electric Equipment for Nuclear Power Plants," and document 29 are being withheld because they contain information which constitutes advice, opinions and recommendations of the staff. This information is being withheld from public disclosure pursuant to Exemption (5) of the Freedom of Information Act (5 U.S.C. 552(b)(5)) and 10 CFR 9.5(a)(5). The persons responsible for the denial of these documents are the undersigned and Mr. Robert B. Minogue, Director, Office of Nuclear Regulatory Research.

In addition, documents numbered 13, 15, 24, 32, 33, 34, 68, 74, 75 and portions of document 23, are also being withheld pursuant to Exemption (5) of the Freedom of Information Act in that they contain predecisional advice, opinions and recommendations of the staff to the Commission. There are no reasonably segregable portions. The person responsible for the denial of these documents, with the exception of document 75, is Mr. Samuel J. Chilk, Secretary of the Commission. Document 75 is being withheld by Mr. Leonard Bickwit, General Counsel of the Commission.

Pursuant to 10 CFR 9.9 and 9.15 of the Commission's regulations, it has been determined that the information withheld is exempt from production or disclosure and that its production or disclosure is contrary to the public interest.

The denials by Mr. Minogue and myself may be appealed to the Commission's Executive Director for Operations within 30 days from the receipt of this letter. As provided in 10 CFR 9.11, any such appeal must be in writing, addressed to the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision". The denials by Mr. Chilk and Mr. Bickwit may be appealed to the Commission and should be addressed to the Secretary of the Commis' 'n.

Sincerely,

J. M. Felton

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Appendix A

*1.	3/2/81	Memo to Mult. Add. from Morrison - Proposed Rulemaking and Associated Regulatory Guide 1.89**8103310008 (70 pages)
*2.	3/17/81	Memo to Morrison from Sniezek - Standard Review Request - Proposed Rulemaking and Associated Regulatory Guide 1.89 - (1 page)
*3.	3/25/81	Memo to Rosztoczy from Watt - Proposed Rulemaking and Associated Regulatory Guide - (4 pages)
*4.	5/4/81	Memo to Knighton from Sullivan - Proposed Rulemaking, "Environmental Qualification of Electric Equipment for Nuclear Power Plants," (RS 025-1) and Regulatory Guide 1.89 (RS 042-2) (1 page)
*5.	5/11/81	Memo to Knighton from Rosztoczy - Proposed Rulemaking and Regulatory Guide - (4 pages)
*6.	6/17/81	Memo to Fraley from Minogue - Proposed Rulemaking and Regulatory Guide - (64 pages)
*7.	6/16/81	Memo to Arlotto from Ross - Package to ACRS on Proposed EQ Rule - (3 pages)
*8.	7/27/81	Memo to Kerr from Fischer - Subcommittee on Electrical Systems Meeting of July 22, 1981 - (7 pages)
*9.	8/7/81	Memo to ACRS Members from Savio - August 7, 5:00-6:00 pm Discussions on the Qualification of Electrical Equipment, Tab 9 - (20 pages)
*10.	---	Schedule for August 7, 1981 Discussion on Environmental and Seismic Qualification of Electrical Equipment Important to Safety, 5:00-6:00 pm - (4 pages)
*11.	8/7/81	Proposed Regulatory Guide 1.89, Rev. 1 - Draft 3 - (53 pages)
12.	8/21/81	Note to Chairman Palladino from Aggarwal - (30 pages)
***13.	9/3/81	Memo to Chilk from Bradford - SECY-81-486 - Petition for Extension of Deadline for Environmental Qualification of Class IE Electrical Equipment - (2 pages)
***14.	9/4/81	Memo to Minogue from Stello - Proposed Rulemaking, "Environmental and Seismic Qualification of Electrical Equipment for Nuclear Power Plants - (1 page)
***15.	9/11/81	Memo to Dircks from Bradford - Seismic Qualification of Electrical Equipment Important to Safety**8111120677 - (1 page)

*16.	9/14/81	Letter to Chairman Palladino from Vandenburgh - Request Commission consideration of NRC Staff's planned program on equipment qualification**8112300009 - (5 pages)
*17.	9/18/81	Memo to Bradford from Dircks - Seismic Qualification of Safety-Grade Electrical Equipment in Diablo Canyon**8111120669 (2 pages)
*18.	9/22/81	Memo to Those on Attached List from Aggarwal - Proposed Rulemaking, "Environmental and Seismic Qualification of Electric Equipment for Nuclear Power Plants**8110080388 4 (28 pages)
*19.	9/23/81	Memo to Case/Stello from Ross - EQ Rule - (2 pages)
*20.	9/30/81	Memo to Aggarwal from Felton - Regulatory Flexibility Statement in Environmental and Seismic Qualification of Electric Equipment for Nuclear Power Plant Proposed Rule - (2 pages)
*21.	10/9/81	Memo to Mult. Add. from Aggarwal - Proposed Rule, "Environmental and Seismic Qualification of Electric Equipment forNuclear Power Plants" - Draft Dated October 8, 1981 - (35 pages)
*22.	10/19/81	Memo to Dircks from Aggarwal - Proposed Rule, "Environmental Qualification of Electric Equipment for Nuclear Power Plants" - (28 pages)
23.	10/20/81	Memo to Dircks from Chilk - Petition for Extension of Deadline for Environmental Qualification of Class 1E Electrical Equipment (SECY 81-486) - (6 pages)
***24.	10/21/81	Chairman Palladino's Response Sheet on SECY-245 - Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations - (This document also requires answers to questions on Equipment Qualification Program Plans) (1 page)
*25.	11/6/81	Memo to Commissioner Ahearne from Aggarwal - SECY 81-603 - (48 pages)
*26.	11/18/81	Memo to Dircks from Chilk - Staff Requirements - Briefing on SECY81-504, Equipment Qualification Program Plan, and SECY 81-603/603A, Proposed Rulemaking, "Environmental Qualification of Electric Equipment for Nuclear Power Plants"**8112100626 (3 pages)
*27.	11/24/81	Memo to Kopeck from Aggarwal - Proposed Rulemaking, "Environmental Qualification of Electric Equipment for Nuclear Power Plants" **8112110041 - (3 pages)
*28.	11/30/81	Memo to Chilk from Roberts - SECY-81-603B - (1 page)
29.	12/2/81	Memo to Ahearne from Dircks - SECY 81-504 AND SECY 81-603 Your Memorandum dated November 17, 1981**8201120008 (6 pages)
*30.	12/9/81	Memo to Johnston from Vollmer - December 18 Briefing for Commissioner Bradford on Seismic Qualification - (1 page)
*31.	12/10/81	Memo to Mult. Add. from Stello - Equipment Qualification Rulemaking - (3 pages)
***32.	12/11/81	Memo to Commission from Bradford - Proposed Rule on Environmental Qualification - SECY 81-603B - (5 pages)

***33.	12/14/81	Memo to Chilk from Ahearne - SECY-81-603B - (5 pages)
***34.	12/14/81	Memo to EDO from Ahearne - December 2, 1981 Memorandum: SECY 81-504 And SECY 81-603 - (1 page)
*35.	12/17/81	Memo to Chilk from Gilinsky - SECY 81 603B - (1 page)
*36.	12/18/81	Memo to Kopeck from Aggarwal - Proposed Rulemaking, "Environmental Qualification"**8201150123 - (3 pages)
*37.	12/17/81	Memo to Bradford from Palladino - Proposed Rule on Environmental Qualification - SECY 81-603B**8201060398 - (7 pages)
*38.	12/17/81	Memo to Stello from Arlotto - Equipment Qualification Ruler (5 pages)
*39.	12/21/81	Memo to Commission from Dircks - Proposed Rule on Environme. Qualification of Electric Equipment**8201220033 - (32 pages)
*40.	12/29/81	Memo to Stello from Denton - Equipment Qualification Rulemaking (1 page)
*41.	1/6/82	SECY 81-603B - Proposed Rulemaking - Environmental Qualification of Electric Equipment for Nuclear Power Plants - (18 pages)
*42.	1/8/82	Memo to Felton from Minogue - Request forPublishing Federal Register Notice of Proposed Rule, "Environmental Qualification of Electric Equipment forNuclear Power Plants" - (1 page)
*43.	1/15/82	Letter to Mult. Add. from Minogue - Proposed Rulemaking**8202050446 (2 pages)
*44.	1/18/82	Letter to Thompson from Felton - Proposed Rulemaking - (15 pages)
*45.	1/20/82	Memo to Ahearne from Dircks - December 14, 1981 Memorandum: SECY 81-504 And SECY 81-603**8202180082 - (4 pages)
*46.	1/21/82	Memo to Arlotto from Sullivan - Background Information on Proposed Revision to RG 1.89 - (3 pages)
*47.	2/11/82	Letter to Palladino from Reynolds - Proposed Rulemaking **8204010034 - (3 pages)
*48.	2/19/82	Letter to Mult. Add. from Arlotto - RG 1.89**8203050137 - (1 page)
*49.	Undated	Supporting Statement for 10 CFR 50 "Environmental Qualification of Electric Equipment for Nuclear Power Plants" - (3 pages)
*50.	3/2/82	Letter to Palladino from Reynolds - SECY 81-504, Rev.1, Equipment Qualification Program Plan - (3 pages)
*51.	3/8/82	Letter to Glenn from Palladino - NRC Comments on S 1080 **8204160204 - (7 pages)
*52.	3/8/82	Memo to Scott from Cameron - Request for OMB Clearance of Proposed Recordkeeping Requirement - (6 pages)

*53. 3/15/82 Memo to EDO from Ahearne - Environmental Qualification of Electric Equipment--Justification for Continued Operation**8204130511 (1 page)

*54. 3/16/82 Memo to Dircks from Aggarwal - Proposed Rule, "Environmental Qualification of Electrical Equipment for Nuclear Power Plants " Comment Period - (3 pages)

*55. 3/15/82 Letter to Reynolds from Chilk -Response to 2/11/82 Letter - (3 pages)

*56. 3/17/82 Letter to Steptoe from Aggarwal - Proposed Rulemaking - (1 page)

*57. 4/15/82 Memo to Fraley from Minogue - Final Rule, Section 50.49 of 10 CFR Part 50 "Environmental Qualification of Electric Equipmer for Nuclear Power Plants" - (37 pages)

*58. 4/26/82 Letter to Dale and Mult. Add. from Minogue - Proposed Rulemaki. (9 pages)

***59. 5/3/82 Memo to Aggarwal from Shields - Comments on EQ Rule - (4o pages)

*60. 5/4/82 Note to Tom Rehm from Aggarwal - Chronology Environmental Qualification Rule - (4 pages)

*61. 5/12/82 Letter to Palladino from Shewmon - Rulemaking on Environmental Qualification of Electric Equipment - (2 pages)

*62. 5/14/82 Note to Mult. Add. from Aggarwal - Section 50.49 EQ Rule - (4 pages)

*63. 5/14/82 Memo to Aggarwal from Felton - DRR Review of Final Rule Concerning Environmental Qualification of Electric Equipment for Nuclear Power Plants; 10 CFR Part 50 - (22 pages)

*64. 5/19/82 Memo to Dircks from Stello - Minutes of CRGR Meeting No. 13 - (3 pages)

*65. 6/9/82 Memo to Commissioners from Dircks - Final Rule, "Environmental Qualification of Safety-Related Electric Equipment for Nuclear Power Plants"**8206220059 - (2 pages)

*66. 6/10/82 Memo to Dircks from Chilk - SECY 82-207 - Final Rule - (2 pages)

*67. 6/17/82 Letter to Winkler from Aggarwal - Final Rule - (2 pages)

***68. 6/21/82 Memo to Chilk from Asselstine - Extension of June 30, 1982 Deadline for Environmental Qualification of Safety Related Electric Equipment SECY 82-207B - (1 page)

*69. 6/25/82 Memo to Mult. Add. from Aggarwal - Final Rule - (7 pages)

*70. 6/25/82 Memo to Dircks/Bickwit from Chilk - SECY 82-207/82-207A - Final Rule - (2 pages)

*71. 6/30/82 Federal Register Notice - Environmental Qualification of Electric Equipment (2 pages)

* 72.	7/9/82	Memo to Arlotto from Felton - Review of Draft Final Rule, Dated 6/30/82, on the Environmental Qualification of Electric Equipment - (1 page)
* 73.	7/12/82	Memo to Aggarwal from DeYoung - Response to Mr. Chilk's Memorandum Pertaining to Section 50.49 to 10 CFR Part 50, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Plants - (31 pages)
*** 74.	8/8/82	Memo to Chilk from Palladino - SECY-82-207C - Final Rule, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants - (1 page)
*** 75.	9/2/82	Memo to Commission from Shields - Draft Final Rule "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, SECY 82-207C/207D - (3 pages)
* 76.	9/10/82	Letter to Palladino from Reynolds - NRC Staff's Proposed Rule Regarding Environmental Qualification of Electrical Equipment **8208010040 - (20 pages)

* = Document Placed in the NRC Public Document Room
** = NRC Accession Number
*** = Document WITHHELD per FOIA Exemption 5 - Contains Pre-decisional (advice, opinions and recommendations)

A-12 Release

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

January 19, 1981

ALL LICENSEES OF OPERATING PLANTS AND APPLICANTS FOR OPERATING LICENSES AND HOLDERS OF CONSTRUCTION PERMITS

SUBJECT: INFORMATION REGARDING THE PROGRAM FOR ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT (GENERIC LETTER 81-05)

Reference (a) - Order for Modification of License concerning the Environmental Qualification of Safety-Related Electrical Equipment, October 24, 1980

Reference (b) - Commission Memorandum and Order of May 23, 1980 (80-CLI-21)

Reference (c) - I&E Bulletin 79-01B Supplement No. 3, October 24, 1980

On October 24, 1980, the staff issued Orders [Reference (a)] to all power reactor licensees, which modified their Technical Specifications in accordance with the Commission-ordered Environmental Qualification Requirements [Reference (b)]. The purpose of this letter is to provide information in response to licensee requests, regarding these Orders and the associated staff actions. The specific item to be addressed involve the environmental qualification (E-Q) requirements of the electrical equipment for the following: (1) equipment necessary to achieve a cold shutdown, (2) replacement parts, (3) Three Mile Island (TMI) Action Plan (NUREG-0737) equipment and (4) the June 30, 1982 deadline of Ref. (a).

(1) Cold Shutdown - Reference (c) requires licensees to submit E-Q information for the equipment necessary to achieve and maintain a cold shutdown condition. This Bulletin requirement was not intended to invoke a change in the licensing basis of the plant. Plants licensed to a hot "safe shutdown" condition are only required by Reference (a) to qualify the equipment necessary to achieve a hot shutdown (i.e., plant specific safety-related equipment). However, the Bulletin (Reference c) does require that the licensee submit the presently available information for one path to achieve the cold shutdown conditions. The Reference (c) position represents an enveloping staff position to be implemented on a case-by-case basis. Regulatory Guide 1.139 contains the implementation plans for the cold shutdown requirements, of which E-Q is a part. Staff reviews are in progress on this issue.

(2) Replacement Parts - We note that this requirement is set forth in reference (b) but not explicitly in the ordering clauses of reference (a). In this regard, the E-Q requirements for replacement parts are

Ref: A-12 8103190387 PDR/LPDR

clearly presented in Supplement No. 2 to IEB 79-01B (September 30, 1980). It is the licensee's responsibility to justify deviation from the Category I column of NUREG-0588 in an auditible manner. "Sound reasons" for such deviation are plant/equipment specific. Examples such as availability or system incompatability, are a matter of degree and will be judged accordingly.

(3) NUREG-0737 Equipment - The qualification requirements for this equipment are described in Appendix B to the NUREG. The schedule for submitting the information to the NRC is contained in Reference (c). Contingencies for equipment unavailability are addressed in the NUREG. If the licensee's position on any of this equipment is that it is not safety-related within the meaning of reference (a), that position should be justified in the submittal. Staff judgments in this regard will be made in a Supplement to the original Safety Evaluation.

(4) June 30, 1982 Deadline - Some licensees have indicated that the new E-Q requirements have resulted in saturation of the test/production capabilities of the industry, and violations of the deadline may occur. Licensees should note that the Ref. (a) orders implemented a Commission imposed deadline [Ref. (b)] and the staff is not authorized to grant relief from this deadline.

Some licensees have submitted either a hearing request or a request for an extension of the time to request a hearing in regard to Ref. (a). Others have submitted letters of concern regarding specific requirements. The former will be addressed on an individual basis. This letter responds to the generic concerns of both groups.

Sincerely,

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

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

NOV 1 3 1979

MEMORANDUM FOR: V. Stello, Director, Office of Inspection and Enforcement

FROM: H.R. Denton, Director, Office of Nuclear Reactor Regulation

SUBJECT: GUIDELINES FOR EVALUATING QUALIFICATION OF CLASS IE ELECTRICAL EQUIPMENT IN OPERATING REACTORS

Enclosed is a copy of the subject guidelines. These guidelines were prepared by NRR, DOR to satisfy its commitment to IE to provide guidelines and criteria for IE to use in its reviews of the licensee responses to IE Bulletin 79-01.

As stated in Section 1.0, Introduction, the objective in preparing the guidelines was to set forth guidelines that should be used to identify Class IE equipment installed in operating reactors whose documentation does not provide reasonable assurance of environmental qualification. Once IE has identified any such equipment it is anticipated that IE would transfer the lead responsibility for the final resolution to NRR, DOR. This is consistent with our plan as outlined at the July 11, 1979, Commission Briefing on IE Bulletin 79-01 and equipment qualification.

Your particular attention is directed to Appendix C, Thermal and Radiation Aging Degradation of Selected Materials. This appendix is provided to support implementation of the staff position stated in Section 7.0, Aging. In summary, the staff position for existing equipment in operating reactors is that a specific qualified life based on thermal and radiation age degradation need only be established for equipment using materials known to exhibit significant degradation from these aging effects. Appendix C is a partial listing of materials which may be found in nuclear power plants along with an indication of the material susceptibility to aging. This listing is based on input from only one of several DOR consultants. Reports from the other DOR consultants are under review. We expect to complete the review by the end of December 1979, and we will supplement Appendix C with additional information at that time.

H R Denton

H.R. Denton, Director
Office of Nuclear Reactor Regulation

Contact:
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X-27900

Ref: A-12 7912190933 PDR

Enclosure:
As stated

cc w/enclosure:
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GUIDELINES FOR EVALUATING ENVIRONMENTAL QUALIFICATION OF CLASS IE ELECTRICAL EQUIPMENT IN OPERATING REACTORS

GUIDELINES FOR EVALUATING ENVIRONMENTAL QUALIFICATION

OF CLASS IE ELECTRICAL EQUIPMENT

IN OPERATING REACTORS

1.0 INTRODUCTION

On February 8, 1979, the NRC Office of Inspection and Enforcement issued IE Bulletin 79-01, entitled, "Environmental Qualification of Class IE Equipment." This bulletin requested that licensees for operating power reactors complete within 120 days their reviews of equipment qualification begun earlier in connection with IE Circular 78-08. The objective of IE Circular 78-08 was to initiate a review by the licensees to determine whether proper documentation existed to verify that all Class IE electrical equipment would function as required in the hostile environment which could result from design basis events.

The licensees' reviews are now essentially complete and the NRC staff has begun to evaluate the results. This document sets forth guidelines for the NRC staff to use in its evaluations of the licensees' responses to IE Bulletin 79-01 and selected associated qualification documentation. The objective of the evaluations using these guidelines is to identify Class IE equipment whose documentation does not provide reasonable assurance of environmental qualification. All such equipment identified will then be subjected to a plant application specific evaluation to determine whether it should be requalified or replaced with a component whose qualification has been adequately verified.

These guidelines are intended to be used by the NRC staff to evaluate the qualification methods used for existing equipment in a particular class of plants, i.e., currently operating reactors including SEP plants.

Equipmont in other classes of plants not yet licensed to operate, or replacement equipment for operating reactors, may be subject to different requirements such as those set forth in NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

In addition to its reviews in connection with IE Bulletin 79-01 the staff is engaged in other generic reviews that include aspects of the equipment qualification issue. TMI-2 lessons learned and the effects of failures of non-Class IE control and indication equipment are examples of these generic reviews. In some cases these guidelines may be applicable, however, this determination will be made as part of that related generic review.

2.0 DISCUSSION

IEEE Std. 323-1974[1] is the current industry standard for environmental qualification of safety-related electrical equipment. This standard was first issued as a trail use standard, IEEE Std. 323-1971, in 1971 and later after substantial revision, the current version was issued in 1974. Both versions of the standard set forth generic requirements for equipment qualification but the 1974 standard includes specific requirements for aging, margins, and maintaining documentation records that were not included in the 1971 trial use standard.

The intent of this document is not to provide guidelines for implementing either version of IEEE Std. 323 for operating reactors. In fact most of the operating reactors are not committed to comply with any particular industry standard for electrical equipment qualification. However, all of the operating reactors are required to comply with the General Design Criteria

[1] IEEE Std. 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations."

specified in Appendix A of 10 CFR 50. General Design Criterion 4 states in part that "structures, systems and components important to safet, shall be designed to accomodate the affects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents." The intent of these guidelines is to provide a basis for judgements required to confirm that operating reactors are in compliance with General Design Criterion 4.

3.0 IDENTIFICATION OF CLASS IE EQUIPMENT

Class IE equipment includes all electrical equipment needed to achieve emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, and prevention of significant release of radioactive material to the environment. Typical systems included in pressurized and boiling water reactor designs to perform these functions for the most severe postulated loss of coolant accident (LOCA) and main steamline break accident (MSLB) are listed in Appendix A.

More detailed descriptions of the Class IE equipment installed at specific plants can be obtained from FSARs, Technical specifications, and emergency procedures. Although variation in nomenclature may exist at the various plants, environmental qualification of those systems which perform the functions identified in Appendix A should be evaluated against the appropriate service conditions (Section 4.0).

The guidelines in this document are applicable to all components necessary for operation of the systems listed in Appendix A including but not limited to valves, motors, cables, connectors, relays, switches, transmitters and valve position indicators.

4.0 SERVICE CONDITIONS

In order to determine the adequacy of the qualification of equipment it is necessary to specify the environment the equipment is exposed to during normal and accident conditions with a requirement to remain functional. These environments are referred to as the "service conditions."

The approved service conditions specified in the FSAR or other licensee submittals are acceptable, unless otherwise noted in the guidelines discussued below.

4.1 Service Conditions Inside Containment for a Loss of Coolant Accident (LOCA)

1. Temperature and Pressure Steam Conditions - In general, the containment temperature and pressure conditions as a function of time should be based on the analyses in the FSAR. In the specific case of pressure suppression type containments, the following minimum high tempeature conditions should be used: (1) BWR Drywells - 340°F for 6 hours; and (2) PWR Ice Condenser Lower Compartments - 340°F for 3 hours.
2. Radiation - When specifying radiation service conditions for equipment exposed to radiation during normal operating and accident conditions, the normal operating dose should be added to the dose received during the course of an accident. Guidelines for evaluating beta and gamma radiation service conditions for general areas inside containment are provided below. Radiation service conditions for equipment located directly above the containment sump, in the vicinity of filters, or submerged in contaminated liquids must be evaluated on a case by case basis. Guidelines for these evaluations are not provided in this document.

Gamma Radiation Doses - A total gamma dose radiation service condition of 2×10^7 RADS is acceptable for Class IE equipment located in general areas inside containment for PWRs with dry type containments. Where a dose less than this value has been specified, an application specific evaluation must be performed to determine if the dose specified is acceptable. Procedures for evaluating radiation service conditions in such cases are provided in Appendix B. The procedures in Appendix B are based on the calculation for a typical PWR reported in Appendix D of NUREG-0588[1].

Gamma dose radiation service conditions for BWRs and PWRs with ice condenser containments must be evaluated on a case by case basis. Since the procedures in Appendix B are based on a calculation for a typical PWR with a dry type containment, they are not directly applicable to BWRs and other containment types. However, doses for these other plant configurations may be evaluated using similar procedures with conservative dose assumptions and adjustment factors developed on a case by case basis.

Beta Radiation Doses - Beta radiation doses generally are less significant than gamma radiation doses for equipment qualification. This is due to the low penetrating power of beta particles in comparison to gamma rays of equivalent energy. Of the general classes of electrical equipment in a plant (e.g., cables, instrument transmitters, valve operators, containment penetrations), electrical cable is considered the most

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[1]NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

vulnerable to damage from beta radiation. Assuming a TID 14844 source term, the average maximum beta energy and isotopic abundance will vary as a function of time following an accident. If these parameters are considered in a detailed calculation, the conservative beta surface dose of 1.40 x x 10^8 RADS reported in Appendix D of NUREG 0588 would be reduced by approximately a factor of ten within 30 mils of the surface of electrical cable insulation of unit density. An additional 40 mils of insulation (total of 70 mils) results in another factor of 10 reduction in dose. Any structures or other equipment in the vicinity of the equipment of interest would act as shielding to further reduce beta doses. If it can be shown, by assuming a conservative unshielded surface beta dose of 2.0 x 10^8 RADS and considering the shielding factors discussed here, that the beta dose to radiation sensitive equipment internals would be less than or equal to 10% of the total gamma dose to which an item of equipment has been qualified, then that equipment may be considered qualified for the total radiation environment (gamma plus beta). If this criterion is not satisfied the radiation service condition should be determined by the sum of the gamma and beta doses.

3. Submergence - The preferred method of protection against the effects of submergency is to locate equipment above the water flooding level. Specifying saturated steam as a service condition during type testing of equipment that will become flooded in service is not an acceptable alternative for actually flooding the equipment during the test.

4. Containment Sprays - Equipment exposed to chemical sprays should be qualified for the most severe chemical environment (acidic or basic) which could exist. Demineralized water sprays should not be exempt from consideration as a potentially adverse service condition.

4.2 Service Conditions for a PWR Main Steam Line Break (MSLB) Inside Containment

Equipment required to function in a steam line break environment must be qualified for the high temperature and pressure that could result. In some cases the environmental stress on exposed equipment may be higher than that resulting from a LOCA, in others it may be no more severe than for a LOCA due to the automatic operation of a containment spray system.

1. Temperature and Pressure Steam Conditions - Equipment qualified for a LOCA environment is considered qualified for a MSLB accident environment in plants with automatic spray systems not subject to disabling single component failures. This position is based on the "Best Estimate" calculation of a typical plant peak temperature and pressure and a thermal analysis of typical components inside containment.[1/] The final acceptability of this approach, i.e., use of the "Best Estimate", is pending the completion of Task Action Plan A-21, Main Steamline Break Inside Containment.

   Class IE equipment installed in plants without automatic spray systems or plants with spray systems subject to disabling single failures or delayed initiation should be qualified for a MSLB accident environment determined by a plant specific analysis. Acceptable methods

[1] See NUREG 0458, Short Term Safety Assessment on the Environmental Qualification of Safety-Related Electrical Equipment of SEP Operating Reactors, for a more detailed discussion of the best estimate calculation.

for performing such an analysis for operating reac'ors are provided in Section 1.2 for Category II plants in NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment.

2. Radiation - Same as Section 4.1 above except that a conservative gamma dose of 2 x 10^6 RADS is acceptable.
3. Submergence - Same as Section 4.1 above.
4. Chemical Sprays - Same as Section 4.1 above.

4.3 Service Conditions Outside of Containment

4.3.1 Areas Subject to a Severe Environment as a Result of a High Energy Line Break (HELB)

Service conditions for areas outside containment exposed to a HELB were evaluated on a plant by plant basis as part of a program initiated by the staff in December, 1972 to evaluate the effects of a HELB. The equipment required to mitigate the event was also identified. This equipment should be qualified for the service conditions reviewed and approved in the HELB Safety Evaluation Report for each specific plant.

4.3.2 Areas Where Fluids are Recirculated from Inside Containment to Accomplish Long-Term Core Cooling Following a LOCA

1. Temperature and Relative Humidity - One hundred percent relative humidity should be established as a service condition in confined spaces. The temperature and pressure as a function of time should be based on the plant unique analysis reported in the FSAR.

2. Radiation - Due to differences in equipment arrangement within these areas and the significant effect of this factor on doses, radiation service conditions must be evaluated on a case by case basis. In general, a dose of at least 4×10^6 RADS would be expected.
3. Submergence - Not applicable.
4. Chemical Sprays - Not applicable.

4.3.3 Areas Normally Maintained at Room Conditions

Class IE equipment located in these areas does not experience significant stress due to a change in service conditions during a design basis event. This equipment was designed and installed using standard engineering practices and industry codes and standards (e.g., ANSI, NEMA, National Electric Code). Based on these factors, failures of equipment in these areas during a design basis event are expected to be random except to the extent that they may be due to aging or failures of air conditioning or ventilation systems. Therefore, no special consideration need be given to the environmental qualification of Class IE equipment in these areas provided the aging requirements discussed in Section 7.0 below are satisfied and the areas are maintained at room conditions by redundant air conditioning or ventilation systems served by the onsite emergency electrical power system. Equipment located in areas not served by redundant systems powered from onsite emergency sources should be qualified for the environmental extremes which could result from a failure of the systems as determined from a plant specific analysis.

5.0 QUALIFICATION METHODS

5.1 <u>Selection of Qualification Method</u>

The choice of qualification method employed for a particular application of equipment is largely a matter of technical judgement based on such factors as: (1) the severity of the service conditions; (2) the structural and material complexity of the equipment; and (3) the degree of certainty required in the qualification procedure (i.e., the safety importance of the equipment function). Based on these considerations, type testing is the preferred method of qualification for electrical equipment located inside containment required to mitigate the consequences of design basis events, i.e., Class IE equipment (see Section 3.0 above). As a minimum, the qualification for severe temperature, pressure, and steam service conditions for Class IE equipment should be based on type testing. Qualification for other service conditions such as radiation and chemical sprays may be by analysis (evaluation) supported by test data (see Section 5.3 below). Exceptions to these general guidelines must be justified on a case by case basis.

5.2 <u>Qualification by Type Testing</u>

The evaluation of test plans and results should include consideration of the following factors:

1. <u>Simulated Service Conditions and Test Duration</u> - The environment in the test chamber should be established and maintained so that it envelopes the service conditions defined in accordance with Section 4.0 above. The time duration of the test should be at least as long as the period from the initiation of the accident until the temperature and pressure service conditions return to essentially the same levels that existed before the postulated accident. A shorter test duration may be acceptable

if specific analyses are provided to demonstrate that the materials involved will not experience significant accelerated thermal aging during the period not tested.

2. Test Specimen - The test specimen should be the same model as the equipment being qualified. The type test should only be considered valid for equipment identical in design and material construction to the test specimen. Any deviations should be evaluated as part of the qualification documentation (see also Section 8.0 below).

3. Test Sequence - The component being tested should be exposed to a steam/air environment at elevated temperature, and pressure in the sequence defined for its service conditions. Where radiation is a service condition which is to be considered as part of a type test, it may be applied at any time during the test sequence provided the component does not contain any materials which are known to be susceptible to significant radiation damage at the service condition levels or materials whose susceptibility to radiation damage is not known (see Appendix C). If the component contains any such materials, the radiation dose should be applied prior to or concurrent with exposure to the elevated temperature and pressure steam/air environment. The same test specimen should be used throughout the test sequence for all service conditions the equipment is to be qualified for by type testing. The type test should only be considered valid for the service conditions applied to the same test specimen in the appropriate sequence.

4. Test Specimen Aging - Tests which were successful using test specimens which had not been preaged may be considered acceptable provided the component does not contain materials which are known to be susceptible

to significant degradation due to thermal and radiation agir (see Section 7.0). If the component contains such materials a qualified life for the component must be established on a case by case basis. Arrhenius techniques are generally considered acceptable for thermal aging.

5. Functional Testing and Failure Criteria - Operational modes tested should be representative of the actual application requirements (e.g., components which operate normally energized in the plant should be normally energized during the tests, motor and electrical cable loading during the test should be representative of actual operating conditions). Failure criteria should include instrument accuracy requirements based on the maximum error assumed in the plant safety analyses. If a component fails at any time during the test, even in a so called "fail safe" mode, the test should be considered inconclusive with regard to demonstrating the ability of the component to function for the entire period prior to the failure.

6. Installation Interfaces - The equipment mounting and electrical or mechanical seals used during the type test should be representative of the actual installation for the test to be considered conclusive. The equipment qualification program should include an as-built inspection in the field to verify that equipment was installed as it was tested. Particular emphasis should be placed on common problems such as protective enclosures installed upside down with drain holes at the top and penetrations in equipment housings for electrical connections being left unsealed or susceptible to moisture incursion through stranded conductors.

5.3 Qualification by a Combination of Methods (Test, Evaluation, Analysis

As discussed in Section 5.1 above, an item of Class IE equipment may be shown to be qualified for a complete spectrum of service conditions even though it was only type tested for high temperature, pressure and steam. The qualification for service conditions such as radiation and chemical sprays may be demonstrated by analysis (evaluation). In such cases the overall qualification is said to be by a combination of methods. Following are two specific examples of procedures that are considered acceptable. Other similar procedures may also be reviewed and found acceptable on a case by case basis.

1. Radiation Qualification - Some of the earlier type tests performed for operating reactors did not include radiation as a service condition. In these cases the equipment may be shown to be radiation qualified by performing a calculation of the dose expected, taking into account the time the equipment is required to remain functional and its location using the methods described in Appendix B, and analyzing the effect of the calculated dose on the materials used in the equipment (see Appendix C). As a general rule, the time required to remain functional assumed for dose calculations should be at least 1 hour.

2. Chemical Spray Qualification - Components enclosed entirely in corrosion resistant cases (e.g., stainless steel) may be shown to be qualified for a chemical environment by an analysis of the effects of the particular chemicals on the particular enclosure materials. The effects of chemical sprays on the pressure integrity of any gaskets or seals present should be considered in the analysis.

6.0 Margin

IEEE Std. 323-1974 defines margin as the difference between the most severe specified service conditions of the plant and the conditions used in type testing to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance. Section 6.3.1.5 of the standard provides suggested factors to be applied to the service conditions to assure adequate margins. The factor applied to the time equipment is required to remain functional is the most significant in terms of the additional confidence in qualification that is achieved by adding margins to service conditions when establishing test environments. For this reason, special consideration was given to the time required to remain functional when the guidelines for Functional Testing and Failure Criteria in Section 5.2 above were established. In addition, all of the guidelines in Section 4.0 for establishing service conditions include conservatisms which assure margins between the service conditions specified and the actual conditions which could realistically be expected in a design basis event. Therefore, if the guidelines in Section 4.0 and 5.2 are satisfied, no separate margin factors are required to be added to the service conditions when specifying test conditions.

7.0 Aging

Implicit in the staff position in Regulatory Guide 1.89 with regard to backfitting IEEE Std. 323-1974 is the staff's conclusion that the incremental improvement in safety from arbitrarily requiring that a specific qualified life be demonstrated for all Class IE equipment is not sufficient to justify the expense for plants already constructed and operating. This position does not, however, exclude equipment using materials that have been identified as being susceptible to significant degradation due to thermal and radiation 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 assure that equipment which is exhibiting age related degradation will be identified and replaced as necessary. Appendix C contains a listing of materials which may be found in nuclear power plants along with an indication of the material susceptability to thermal and radiation aging.

8.0 Documentation

Complete and auditable records must be available for qualification by any of the methods described in Section 5.0 above to be considered valid. These records should describe the qualification method in sufficient detail to verify that all of the guidelines have been satisfied. A simple vendor certification of compliance with a design specification should not be considered adequate.

APPENDIX A

TYPICAL EQUIPMENT/FUNCTIONS NEEDED FOR MITIGATION OF A LOCA OR MSLB ACCIDENT

Engineered Safeguards 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 Systems (e.g., HVAC, Radiation Filters)

Ventilation for Areas Containing Safety Equipment

Component Cooling

Service Water

Emergency Shutdown[2]

Post Accident Sampling and Monitoring[3]

Radiation Monitoring[3]

Safety Related Display Instrumentation[3]

[1] These systems will differ for PWRs and BWRs, and for older and newer plants. In each case the system features which allow for transfer to 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] Emergency shutdown systems include those systems used to bring the plant to a cold shutdown condition following accidents which do not result in a breach of the reactor coolant pressure boundary together with a rapid depressurization of the reactor coolant system. Examples of such systems and equipment are the RHR system, PORVs, RCIC, pressurizer sprays, chemical and volume control system, and steam dump systems.

[3] More specific identification of these types of equipment can be found in the plant emergency procedures.

APPENDIX B

PROCEDURES FOR EVALUATING GAMMA RADIATION SERVICE CONDITIONS

Introduction and Discussion

The adequacy of gamma radiation service conditions specified for inside containment during a LOCA or MSLB accident can be verified by assuming a conservative dose at the containment centerline and adjusting the dose according the plant specific parameters. The purpose of this appendix is to identify those parameters whose effect on the total gamma dose is easy to quantify with a high degree of confidence and describe procedures which may be used to take these effects into consideration.

The bases for the procedures and restrictions for their use are as follows:

(1) A conservative dose at the containment centerline of 2×10^7 RADS for a LOCA and 2×10^6 RADS for a MSLB accident has been assumed. This assumption and all the dose rates used in the procedure outlined below are based on the methods and sample calculation described in Appendix D of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." Therefore, all the limitations listed in Appendix D of NUREG-0588 apply to these procedures.

(2) The sample calculation in Appendix D of NUREG-0588 is for a 4,000 MWth pressurized water reactor housed in a 2.52×10^6 ft^3 containment with an iodine scrubbing spray system. A similar calculation without iodine scrubbing sprays would increase the dose to equipment approximately 15%. The conservative dose of 2×10^7 RADS assumed

in the procedure below includes sufficient conservatism to account for this factor. Therefore, the pro .dure is also applicable to plants without an iodine scrubbing spray system.

(3) Shielding calculations are based on an average gamma energy of 1 MEV derived from TID 14844.

(4) These procedures are not applicable to equipment located directly above the containment sump, submerged in contaminated liquids, or near filters. Doses specified for equipment located in these areas must be evaluated on a case by case basis.

(5) Since the dose adjustment factors used in these procedures are based on a calculation for a typical pressurized water reactor with a dry type containment, they are not directly applicable to boiling water reactors or other containment types. However, doses for these other plant configurations may be evaluated using similar procedures with conservative dose assumptions and adjustment factors developed on a case by case basis.

Procedure

Figures 1 through 4 provide factors to be applied to the conservative dose to correct the dose for the following plant specific parameters: (1) reactor power level; (2) containment volume; (3) shielding; (4) compartment volume; and (5) time equipment is required to remain functional.

The procedure for using the figures is best illustrated by an example. Consider the following case. The radiation service condition for a particular item of equipment has been specified as 2×10^6 RADS. The application specific parameters are:

Reactor power level - 3,000 MWth

Containment volume - 2.5×10^6 ft^3

Compartment Volume - 8,000 ft^3

Thickness of compartment shield wall (concrete) - 24"

Time equipment is required to remain functional - 1 hr.

The problem is to make a reasonable estimate of the dose that the equipment could be expected to receive in order to evaluate the adequacy of the radiation service condition specification.

Step 1

Enter the nomogram in Figure 1 at 3,000 MWth reactor power level and 2.5×10^6 ft^3 containment volume and read a 30-day integrated dose of 1.5×10^7 RADS.

Step 2

Enter Figure 2 at a dose of 1.5×10^7 RADS and 24" of concrete shielding for the compartment the equipment is located in and read 4.5×10^4 RADS. This is the dose the equipment receives from sources outside the compartment. To this must be added the dose from sources inside the compartment (Step 3).

Step 3

Enter Figure 3 at 8,000 ft^3 and read a correction factor of 0.13. The dose due to sources inside the compartment would then be $0.13\ (1.5 \times 10^7)$ $= 1.95 \times 10^6$ RADS. The sums of the doses from steps 2 and 3 equals:

$$4.5 \times 10^4 \text{ RADS} + 0.13\ (1.5 \times 10^7) \text{ RADS} = 2.0 \times 10^6 \text{ RADS}$$

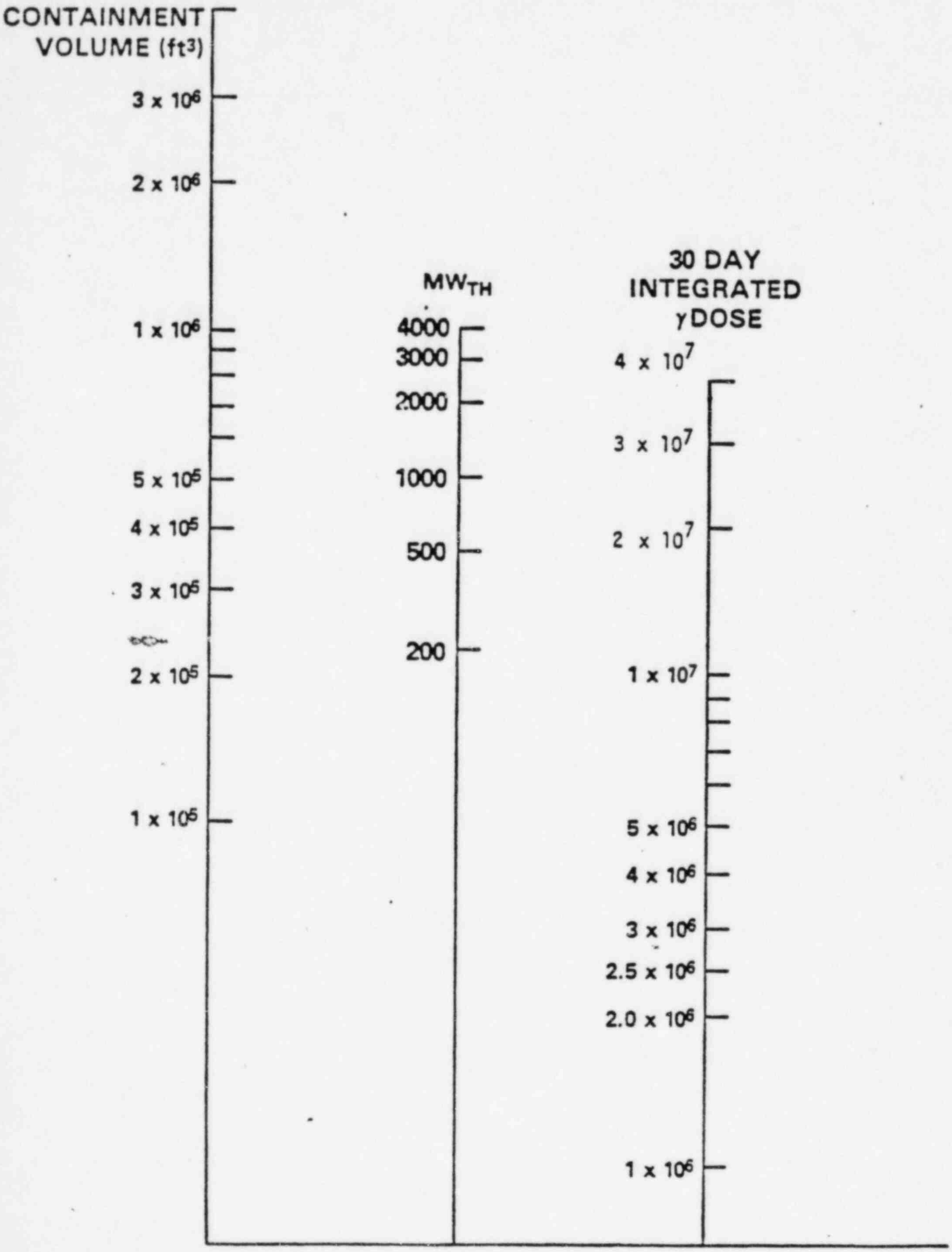
Step 4

Enter Figure 4 at 1 hour and read a correction factor of 0.15. Apply this factor to the sum of the doses determined from steps 2 and 3 to correct the 30 day total dose to the equipment inside the compartment to 1 hour.

$$0.15\ (2.0 \times 10^6) = 3 \times 10^5 \text{ RADS}$$

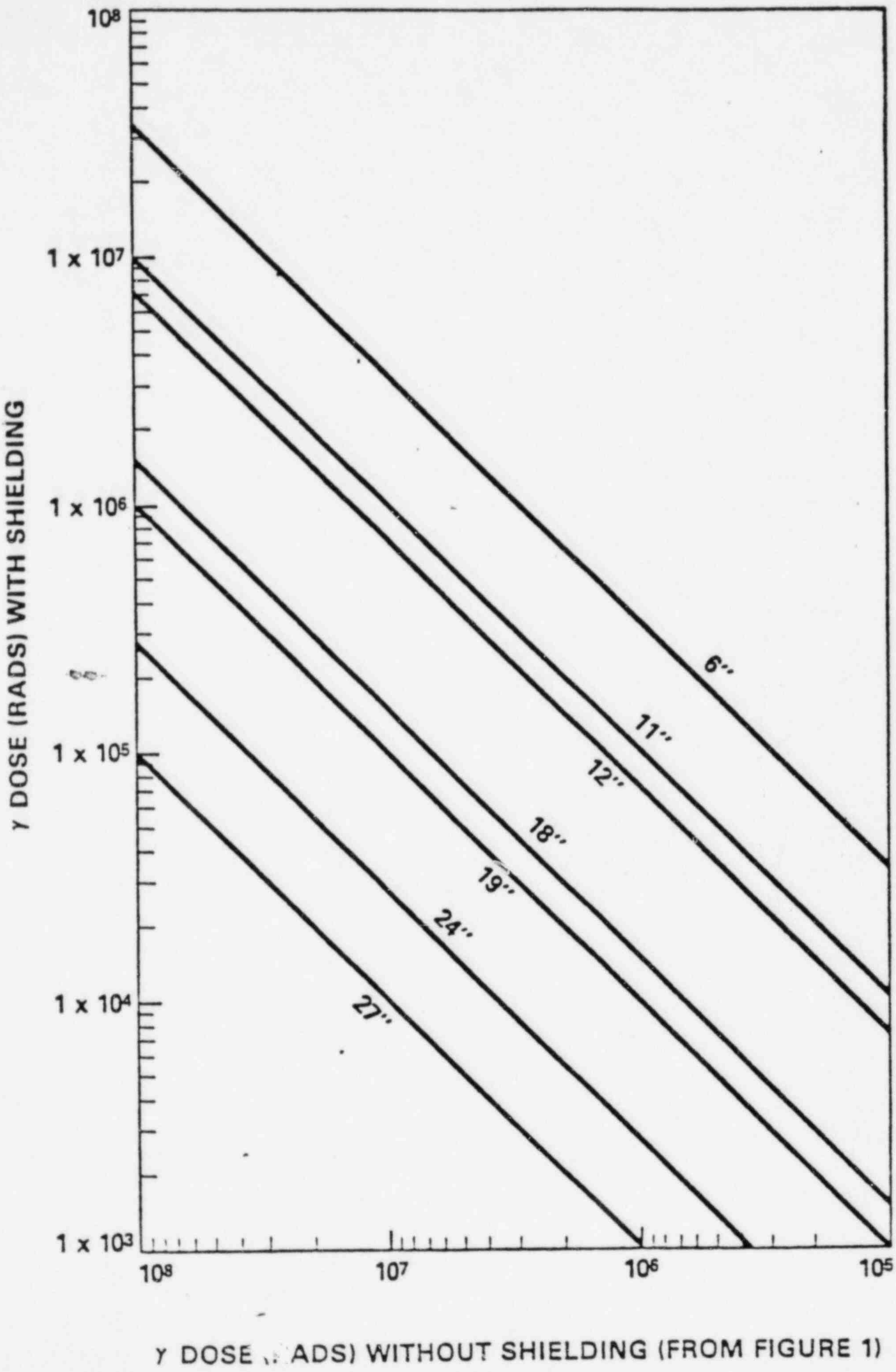
In this particular example the service condition of 2×10^6 RADS specified is conservative with respect to the estimated dose of 3×10^5 RADS calculated in steps 1 through 4 and is, therefore, acceptable.

FIGURE 1
NOMOGRAM FOR CONTAINMENT VOLUME AND REACTOR POWER
LOCA DOSE CORRECTIONS*



*MSLB ACCIDENT DOSES SHOULD BE READ AS A FACTOR OF 10 LESS

FIGURE 2
DOSE CORRECTION FACTOR FOR CONCRETE SHIELDING
(γ ONLY)



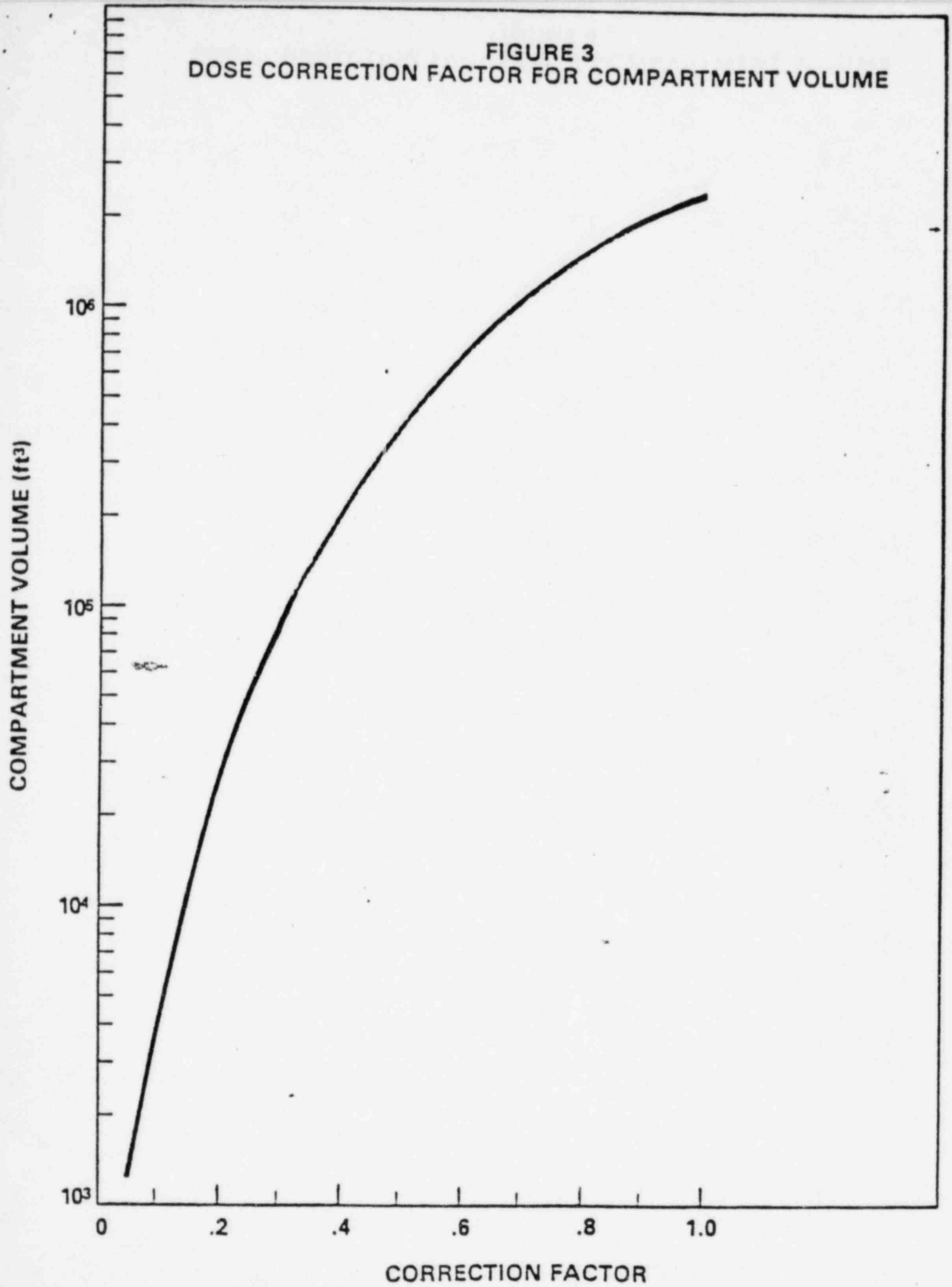
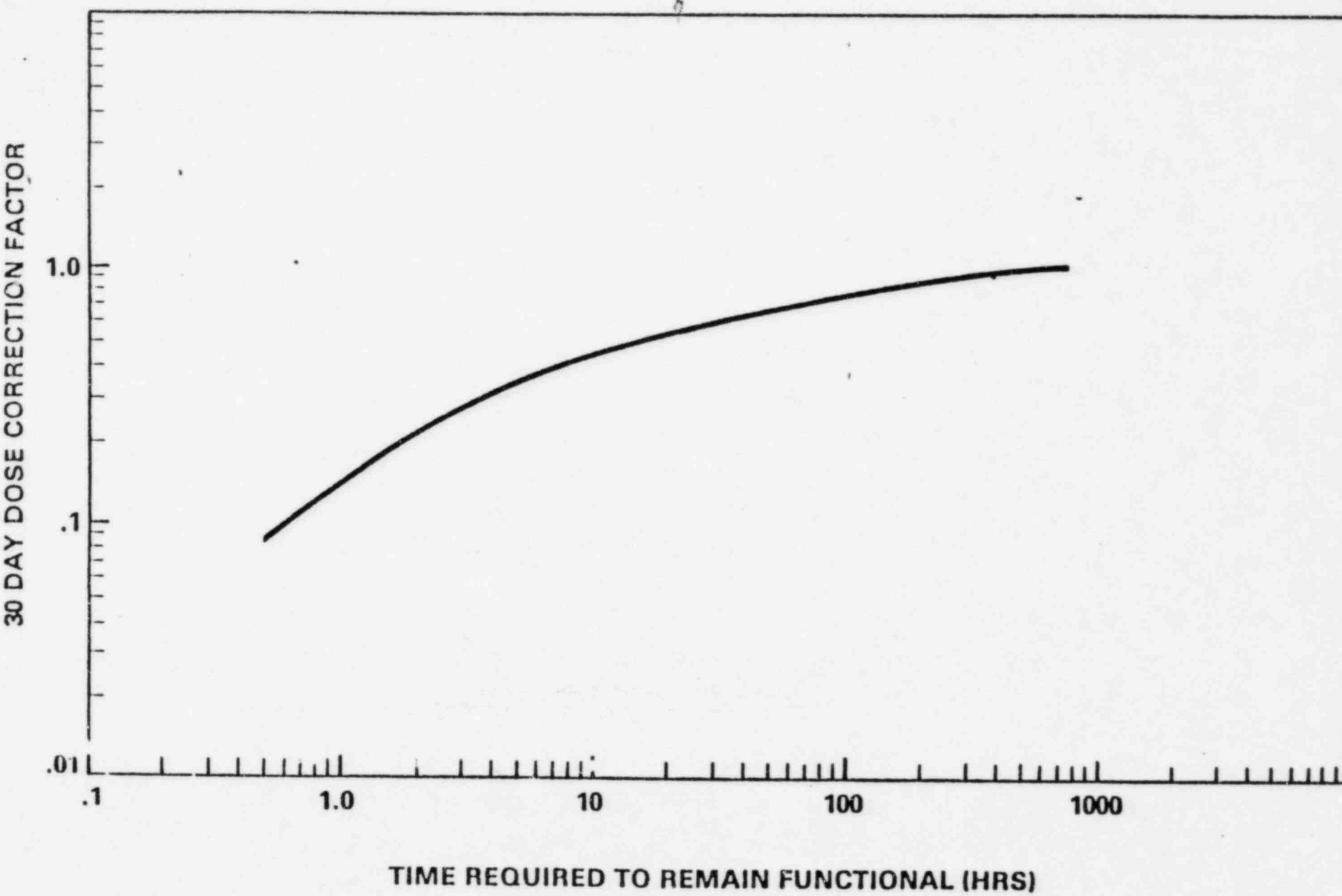


FIGURE 3
DOSE CORRECTION FACTOR FOR COMPARTMENT VOLUME
COMPARTMENT VOLUME (ft³)
10⁶
10⁵
10⁴
10³
0
.2
.4
.6
.8
1.0
CORRECTION FACTOR

FIGURE 4
DOSE CORRECTION FOR TIME REQUIRED TO REMAIN FUNCTIONAL



APPENDIX C

THERMAL AND RADIATION AGING DEGRADATION OF SELECTED MATERIALS

Table C-1 is a partial list of materials which may be found in a nuclear power plant along with an indication of the material susceptibility to radiation and thermal aging.

Susceptibility to significant thermal aging in a 45°C environment and normal atmosphere for 10 or 40 years is indicated by an (*) in the appropriate column. Significant aging degradation is defined as that amount of degradation that would place in substantial doubt the ability of typical equipment using these materials to function in a hostile environment.

Susceptibility to radiation damage is indicated by the dose level and the observed effect identified in the column headed BASIS. The meaning of the terms used to characterize the dose effect is as follows:

- Threshold - Refers to damage threshold, which is the radiation exposure required to change at least one physical property of the material.
- Percent Change of Property - Refers to the radiation exposure required to change the physical property noted by the percent.
- Allowable - Refers to the radiation which can be absorbed before serious degradation occurs.

The information in this appendix is based on a literature search of sources including the National Technical Information Service (NTIS), the National Aeronautics and Space Administration's Scientific and Technical Aerospace Report (STAR), NTIS Government Report Announcements and Index (GRA), and various manufacturers data reports. The materials list is not to be considered all inclusive neither is it to be used as a basis for specifying materials to be used for specific applications within a nuclear plant. The list is solely intended for use by the NRC staff in making judgements as to the possibility of a particular material in a particular application being susceptible to significant degradation due to radiation or thermal aging.

The data base for thermal and radiation aging in engineering materials is rapidly expanding at this time. As additional information becomes available Table C-1 will be updated accordingly.

TABLE C-1

THERMAL AND RADIATION AGING DEGRADATION OF SELECTED MATERIALS

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING		RADIATION SUSCEPTIBILITY		TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND)																
		10 YRS	40 YRS	RADS GAMMA	BASIS	CABLE	CONNECTORS	ELECTRICAL PENETRATIONS	HEATERS	INSTRUMENT RACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENERATOR CONTROL EQUIPMENT	FANS	HVAC EQUIPMENT	MTR. CONTROL CENTERS	SWITCHGEAR
Integrated Circuits (IC) N-MOS				10^3	Threshold					X							X			X		
Integrated Circuits (IC) C-MOS				10^4	"				X								X	X		X	X	
Transistors				10^4	"				X	X			X				X	X		X	X	
Diodes				10^4	"				X	X			X				X	X		X	X	
Silicon-Controlled Rectifiers				10^4	"				X	X			X				X	X		X	X	
Integrated Circuits (IC) Analog				10^4	"				X	X			X				X			X		
Vulcanized Fiber		*	*	10^5	"				X							X	X		X	X		
Fish Paper				10^5	"				X	X	X	X					X	X	X	X	X	X
Polyester (unfilled)		*	*	10^5	"	X		X	X		X	X		X			X	X		X	X	X
Nylon	Polyamide	*	*	10^5	"	X	X	X	X	X	X		X		X	X	X	X	X	X	X	X
Polycarbonate			*	10^6	"				X			X	X				X	X	X	X	X	
Polyimide				10^6	"				X			X					X	X	X		X	
Chlorosulfonated Polyethylene	Hypalon		*	10^7	Allowable	X			X				X					X	X		X	
Buna-N	NBR/Nitrile Rubber	*	*	10^6	Threshold				X		X					X		X			X	
Integrated Circuits (IC) TTL				10^6	"				X	X			X				X	X		X	X	
Diallyl Phthalate	DAP			10^6	"																	
Silicone Rubber				10^6	"	X			X													

*Indicates that there is data available which shows a potential for significant thermal aging of the materials when exposed to normal operating conditions for either 10 or 40 years as indicated.

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING		RADIATION SUSCEPTIBILITY		TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND)																
		10 YRS	40 YRS	RADS GAMMA	BASIS	CABLE	CONNECTORS	ELECTRICAL PENETRATIONS	HEATERS	INSTRUMENT PACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENERATOR CONTROL EQUIPMENT	FANS	LOGIC EQUIPMENT	MTR. CONTROL CENTERS	SWITCHGEAR
SBR Solder			•	10^6	-	X			X													
Capacitors - Tantalum				10^6	Allowable				X	X			X				X	X		X	X	
Delrin		•	•	10^5	Threshold				X			X						X	X		X	X
Tefzel				10^6	16% Loss of Elongation	X			X									X				
G.P. Phenolic P-4050				10^6	Threshold		X		X	X	X		X		X		X	X	X	X	X	X
RTV Silicone				10^7	Allowable								X				X	X			X	X
Cycolac (ABS)			•	10^7	Threshold												X				X	
Integrated Circuits (IC) ECL				10^7	-				X	X							X			X		
Formvar				10^7	-	X			X								X	X	X		X	X
NEMA Polyester Glass Laminates, Grade GPO-2				10^7	-		X				X											X
NEMA Polyester Glass Laminates, Grade GPO-3				10^7	-		X			X	X											X
Polyethylene		•	•	10^7	Allowable	X								X			X	X			X	
Neoprene		•	•	10^7	"	X	X	X	X			X	X		X	X	X	X	X	X	X	
EPR	Ethylene-Propylene Rubber	•	•	10^7	"	X										X						
Polythermaleze				10^8	Threshold	X			X									X	X		X	
Cross-Linked Polyethylene			•	10^7	Allowable																	
Capacitors - Mylar				10^8	-				X	X							X	X		X	X	