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50-348/364
2/19/92

Staff Exh. 27

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

June 6, 1979

IE Bulletin No. 79-01A

SUPPLEMENT NO. 79-01A TO IE BULLETIN 79-01 - ENVIRONMENTAL QUALIFICATION OF CLASS 1E EQUIPMENT (DEFICIENCIES IN THE ENVIRONMENTAL QUALIFICATION OF ASCO SOLENOID VALVES)

Description of Circumstances:

Recently, a noncompliance report under 10 CFR Part 21 was received by the NRC from the Henry Pratt Company, manufacturer of butterfly valves which are installed in the primary containment at the Three Mile Island Unit 2 Nuclear Station. These butterfly valves are used for purge and exhaust purposes and are required to operate during accident conditions. The report discusses the use of an unqualified solenoid valve for a safety-related valve function which requires operation under accident conditions. The solenoid valve in question is Catalogue No. HT-8331A45, manufactured by the Automatic Switch Company (ASCO) of Florham Park, New Jersey. This pilot valve is used to pilot control the pneumatic valve actuators which are installed on the containment ventilation butterfly valves at this facility.

The deficiency in these solenoid valves identified in the Part 21 Report concerns the parts made of acetal plastic material. The acetal disc holder assembly and bottom plug in the pilot valve assembly are stated by ASCO to have a maximum service limit of 400,000 Rad integrated dosage and 200 degrees F temperature. According to ASCO, exposure of these acetal plastic parts to specified maximum environmental conditions may render the solenoid pilot valve inoperable which would cause the associated butterfly valve to malfunction.

Further investigation at ASCO by the NRC staff has revealed that the valve seals in most ASCO solenoid valves contain Buna "N" elastomer material, which reportedly has a maximum service limit of 7,000,000 Rad integrated dosage and 180 degrees F temperature. The investigation further revealed that ASCO has available a line of qualified solenoid operated pilot valves (ASCO Catalogue No. NP-1) which have no plastic parts, utilize ethylene propylene or viton elastomers and have a continuously energized operating life of four years, under normal embient conditions up to 140 degrees F. According to the manufacturer, at the end of this period, the coil, manual operator (optional feature) and all resilient parts must be replaced. These preventive maintenance instructions are specified in the installation and instruction bulletins which are provided to the purchaser with each shipment of solenoid valves.

The final items of concern identified during this investigation deals with the application of Class "A", "B", or "F", solenoid coils which are exposed to an accident environment. In this regard, ASCO representatives stated that the

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In the matter of Alabama Power Company

Staff ✓ IDENTIFIED 12:18 p.m. 2/19/92

Applicant RECEIVED 12:23 p.m. 2/19/92

Intervenor REJECTED

Cont'g Off'r

Contractor DATE 2/19/92

Other Witness

Reporter L Estep

high temperature coils identified as Class "HT" or "HB" are the only coils considered suitable for service under accident conditions; whereas, Class "A", "B", and "F" coils are not.

With respect to the corrective measures to be taken to resolve the above concerns, ASCO recommends the following:

1. The parts of the solenoid valve made of acetal plastic material should be replaced with similar parts made of metal which can be provided by ASCO.

2. The valve seals and gaskets which are made of Buna "N" material should be replaced with either ethylene propylene or viton elastomers, considered by ASCO as suitable for the service intended.

3. Review and determine that the coils of the solenoid valves installed inside containment are Class "HT" or "HB" as required for high temperature environmental conditions.

4. Review and determine that the solenoid enclosures installed inside containment have at least a NEMA 4 enclosure rating.

5. Establish a preventive maintenance program to assure replacement of those valve parts identified above in the time period recommended in the appropriate ASCO valve bulletin.

6. ASCO also stated that all unqualified solenoid valves inside containment be retrofitted to qualified ASCO No. NP-1 valves in lieu of the above.

7. Questions from licensees to ASCO concerning corrective measures should reference both catalogue and serial numbers of each valve in question. These numbers are stamped on the metal nameplate on each solenoid valve.

Action to be Taken by Licensees of all Power Reactor Facilities (except those 11 SEP Plants listed on Enclosure 3) with an Operating License:

1. Determine whether or not ASCO solenoid valves are used or planned for use in safety-related systems at your facility(ies).

2. If such valves are used or planned for use, identify the safety system involved and determine that: (a) valves which could be subjected to a LOCA environment are qualified to that environment. Specifically that no parts made of acetal plastic or Buna "N" materials or Class "A", "B", or "F" solenoid coils are used in such valves; (b) a preventive maintenance program is being conducted such that the solenoid coil, the manual operator (if applicable), and the resilient parts of the valve are being replaced in accordance with the time period established by the manufacturer and documented as the qualified life of the assembled component.

3. All holders of operating licenses of power reactor facilities are obligated to meet the review and reporting requirements established in previously issued IE Bulletin 79-01, regarding environmental qualification of electrical equipment installed in their plants.

No additional written response to this Supplement IE Bulletin is required other than those responses described above. NRC inspectors will continue to monitor the licensees' progress in completing the requested action described above. If additional information is required, contact the Director of the appropriate NRC Regional Office.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

SEP Plants

Plant	Region
Dresden 1	III
Yankee Rowe	I
Big Rock Point	III
San Onofre 1	V
Haddam Neck	I
LaCrosse	III
Oyster Creek	I
R. E. Ginna	I
Dresden 2	III
Millstone	I
Palisades	III

June 4, 1979

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, OIE

SUBJECT: IE BULLETIN NO. 79-01A - ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT (DEFICIENCIES IN THE ENVIRONMENTAL QUALIFICATION OF ASCO SOLENOID VALVES)

The subject IE Bulletin should be dispatched for action on June 6, 1979, to all power reactor facilities with an operating license. The Bulletin should also be dispatched to the 11 SEP Plants (Enclosure 3) and those plants with a Construction Permit for information only. The text of the Bulletin and draft letter to licensees are enclosed for this purpose.

Norman C. Moseley, Director
Division of Reactor Operations
Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letter
2. IE Bulletin No. 79-01A
3. List of SEP Plants (11)

CONTACT: V. D. Thomas, IE
49-28180

PU:SM OFFICE ▶	ROI VDT	~~ROI~~	ROI V.D.T. for	XOOS	ROI V.D.T. for
/1/79 SURNAME ▶	VDThomas	~~[illegible]~~	NCMoseley	DThompson	ELJordan
DATE ▶	6/ /79	~~6/ /79~~	6/ /79	6/ /79	6/ /79

(Draft letter to all power reactor facilities with an operating license or a construction permit)

IE Bulletin No. 79-01A

Gentlemen:

Enclosed is supplement IE Bulletin 79-01A. It requires action by you with regard to power reactor facilities with an operating license except for the 11 SEP plants which are listed in Enclosure 3.

This Bulletin is also being sent for information to the 11 SEP plants and all power reactor facilities with a construction permit. No action or written response is required for construction permit facilities or the 11 SEP plants.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-01A
2. List of IE Bulletins Issued in the past 12 months
3. List of SEP Plants (11)

Jordan

January 10, 1980

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, Office of Inspection and Enforcement

SUBJECT: IE BULLETIN NO. 79-01B - ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

The subject IE Bulletin should be dispatched for action on January 14, 1980, to all power reactor facilities with an Operating License.

The text of the Bulletin and draft letter to licensees are enclosed for this purpose.

/S/

Norman C. Moseley, Director
Division of Reactor Operations Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letter
2. IE Bulletin No, 79-01b and attachments

CONTACT: V. D. Thomas, IE
49-28180

WPU:SM	ROI VDT	ROI	ROI	XOOS
12/31/79	VDThomas	ELJordan	NCMoseley	GCGower
JOB A	1/8/80	1/8/80	1/9/80	1/10/80

(Draft letter to all power reactor facilities with an operating license)

IE Bulletin No. 79-01B

Gentlemen:

Enclosed is IE Bulletin No. 79-01B which requires action by you with regard to your power reactor facility(ies) with an operating license.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-01B
   with Enclosures
2. List of Recently Issued
   IE Bulletins

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

SSINS No.: 6820
Accessions No.:
7910250528

January 14, 1980

IE Bulletin No. 79-01B

ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

Descr' ion of Circumstances:

IE Bulletin No. 79-01 required the licensee to perform a detailed review of the environmental qualification of Class IE electrical equipment to ensure that the equipment will function under (i.e. during and following) postulated accident conditions.

The NRC staff has completed the initial review of licensees' responses to Bulletin No. 79-01. Based on this review, additional information is needed to facilitate completion of the NRC evaluation of the adequacy of environmental qualification of Class IE electrical equipment in the operating facilities. In addition to requesting more detailed information, the scope of this Bulletin is expanded to resolve safety concerns relating to design basis environments and current qualification criteria not addressed in the facilities' FSARS. These include high energy line breaks (HELB) inside and outside primary containment, aging, and submergence.

Enclosure 4, "GUIDELINES FOR EVALUATING ENVIRONMENTAL QUALIFICATION OF CLASS IE ELECTRICAL EQUIPMENT IN OPERATING REACTORS", provides the guidelines and criteria the staff will use in evaluating the adequacy of the licensee's Class IE equipment evaluation in response to this Bulletin.

In general, the orting problems encountered in the original responses and the additional information needed can be grouped into the following areas:

1. All Class IE electrical equipment required to function under the postulated accident conditions, both inside and outside primary containment, was not included in the responses.

2. In many cases, the specific information requested by the Bulletin for each component of Class IE equipment was not reported.

3. Different methods and/or formats were used in providing the written evidence of Class IE electrical equipment qualifications. Some licensees used the System Analysis Method which proved to be the most effective approach. This method includes the following information:

    a. Identification of the protective plant systems required to function under postulated accident conditions. The postulated accident conditions are defined as those environmental conditions resulting from both LOCA and/or HELB inside primary containment and HELB outside the primary containment.

b. Identification of the Class IE electrical equipment items within each of the systems identified in Item -, that are required to function unde the postulated accident conditions.

c. The correlation between the environmental data requirements specified in the FSAR and the environmental qualification test data for each Class IE electrical equipment item identified in Item b above.

4. Additional data not previously addressed in IE Bulletin No. 79-01 are needed to determine the adequacy of the environmental qualification of Class IE electrical equipment. These data address component aging and operability in a submerged condition.

Action To Be Taken By Licensees Of All Power Reactor Facilities With An Operating License (Except those 11 SEP Plants Listed on Enclosure 1)

1. Provide a "master list" of all Engineered Safety Feature Systems (Plant Protection Systems) required to function under postulated accident conditions. Accident conditions are defined as the LOCA/HELB inside containment, and HELB outside containment. For each system within (including cables, EPA's terminal blocks, etc.) the master list identify each Class IE electrical equipment item that is required to function under accident conditions. Pages 1 and 2 of Enclosure 2 are standard formats to be used for the "master list" with typical information included.

   Electrical equipment items, which are components of systems listed in Appendix A of Enclosure 4, which are assumed to operate in the FSAR safety analysis and are relied on to mitigate design basis events are considered within the scope of this Bulletin, regardless whether or not they were classified as part of the engineered safety features when the plant was orginally licensed to operate. The necessity for further up grading of nonsafety-related plant systems will be dependent on the outcome of the licensees and the NRC reviews subsequent to TMI/2.

2. For each class IE electrical equipment item entified in Item 1, provide written evidence of its environmental qual .cation to support the capability of the item to function under postulated accident conditions. For those class IE electrical equipment items not having adequate qualification data available, identify your plans for determining qualifications of these items and your schedule for completing this action. Provide this in the format of Enclosure 3.

3. For equipment identifed in Items 1 and 2 provide service condition profiles (i.e., temperature, pressure, etc., as a function of time). These data should be provided for design basis accident conditions and qualification tests performed. This data may be provided in profile or tabular form.

4. Evaluate the qualification of your Class IE electrical equipment against the guidelines provided in Enclosure 4. Enclosure 5, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," provides supplemental information to be used with these guidelines. For the equipment identified as having "Outstanding Items" by Enclosure 3, provide a detailed "Equipment Qualification Plan." Include in this plan specific actions which will be taken to determine equipment qualification and the schedule for completing the actions.

5. Identify the maximum expected flood level inside the primary containment resulting from postulated accidents. Specify this flood level by elevation such as the 620 foot elevation. Provide this information in the format of Enclosure 3.

6. Submit a "Licensee Event Report" (LER) for any Class IE electrical equipment item which has been determined as not being capable of meeting environmental qualification requirements for service intended. Send the LER to the appropriate NRC Regional Office within 24 hours of identification. If plant operation is to continue following identification, provide justification for such operation in the LER. Provide a detailed written report within 14 days of identification to the appropriate NRC Regional Office. Those items which were previously reported to the NRC as not being qualified per IEB-79-01 do not require an LER.

7. Complete the actions specified by this bulletin in accordance with the following schedule:

   (a) Submit a written report required by Items 1, 2, and 3 within 45 days from receipt of this Bulletin.

   (b) Submit a written report required by Items 4 and 5 wit 90 days from receipt of this Bulletin.

This information is requested under the provisions of 10 CFR 50.54(f). Accordingly, you are requested to provide within the time periods specified in Items 7.a and 7.b above, written statements of the above information, signed under oath or affirmation.

Submit the reports to the Director of the app priate NRC Regional Office. Send a copy of your report to the U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072); clearance expires 7/31/80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosure

RECENTLY ISSUED
IE BULLETINS

Bulletin No.	Subject	Date Issued	Issued To
79-28	Possible Malfunction of Namco Model EA 180 Limit Switches at Elevated Temperatures	12/7/79	All power reactor facilities with an OL or a CP
79-27	Loss Of Non-Class-1-E Instrumentation and Control Power System Bus During Operation	11/30/79	All power reactor facilities holding OLs and to those nearing licensing
79-26	Boron Loss From BWR Control Blades	11/20/79	All BWR power reactor facilities with an OL
79-25	Failures of Westinghouse BFD Relays In Safety-Related Systems	11/2/79	All power reactor facilities with an OL or CP
79-17 (Rev. 1)	Pipe Cracks In Stagnant Borated Water System At PWR Plants	10/29/79	All PWR's with an OL and for information to other power reactors
79-24	Frozen Lines	9/27/79	All power reactor facilities which have either OLs or CPs and are in the late stage of construction
79-23	Potential Failure of Emergency Diesel Generator Field Exciter Transformer	9/12/79	All Power Reactor Facilities with an Operating License or a construction permit
79-14 (Supplement 2)	Seismic Analyses For As-Built Safety-Related Piping Systems	9/7/79	All Power Reactor Facilities with an OL or a CP
79-22	Possible Leakage of Tubes of Tritium Gas in Timepieces for Luminosity	9/5/79	To Each Licensee who Receives Tubes of Tritium Gas Used in Timepieces for Luminosity

Enclosure 1

SEP Plants

Plant	Region
Dresden 1	III
Yankee Rowe	I
Big Rock Point	III
San Onofre 1	V
Haddam Neck	I
LaCrosse	III
Oyster Creek	I
R. E. Ginna	I
Dresden 2	III
Millstone	I
Palisades	III

MASTER LIST
(Typical)

(Class IE Electrical Equipment Required to Function Under Postulated Accident Conditions)

I. SYSTEM: RESIDUAL HEAT REMOVAL (RHR)

COMPONENTS			
Plant Identification Number	Generic Name	Location	
		Inside Primary Containment	Outside Primary Containment
1PT 456	PRESSURE TRANSMITTER	x	
1LT 594	LEVEL TRANSMITTER	x	
1LS 210	LIMIT SWITCH	x	

II. SYSTEM: AUTOMATIC DEPRESSURIZATION SYSTEM (ADS)

COMPONENTS			
Plant Identification Number	Generic Name	Location	
		Inside Primary Containment	Outside Primary Containment
B21 R001	VALVE MOTOR OPERATOR	x	
B21-F003	SOLENOID VALVE		x
B21-F010	PRESSURE SWITCH		x

III. SYSTEM: RHR EQUIPMENT/COMPONENTS (Typical)

**COMPONENTS

Plant Identification Number*	Generic Name	Location	
		Inside Primary Containment	Outside Primary Containment
16xP455	O-RING GASKET	X	
EPA, Class E, Westinghouse, 100C	ELECTRICAL PENETRATION ASSEMBLY	X	
KULKA No. ET35	TERMINAL BOARD	X	
ONKONITE, 1000V, 3C, Black	POWER CABLE	X	X
X BRAND 10W-40	LUBRICATE OIL		X
15 KB69 (Boston Wire & Cable)	INSTRUMENTATION CABLE	X	X
Cutler Hammer TB No. 6	TERMINAL BOX		X
RAYCHEM XYZ	CABLE SPLICE	X	X
Scotch No. 54	INSULATING TAPE		X
T&B No. 10 INSULATED	TERMINAL LUG		X
Y Brand Epoxy No. 111	SEALANT	X	X

* When a component is not identified by plant identification number, use the manufacturer, model number, serial number, etc.

** Like components may be referenced.

SYSTEM COMPONENT EVALUATION WORK SHEET
INSTRUCTIONS

1. Equipment Description: Provide the specific information requested for each Class IE electrical component. Provide component location, specific information such as the building, access floor elevations, and whether the component is above the flood level elevation. In addition, provide the specified and demonstrated accuracies of all instruments for their trip functions and/or post accident monitoring requirements. Cables, EPA's, terminal blocks, and other items shall be identified as part of the engineered safety features systems.

2. Environment: List values for each environmental parameter indicated. List the "specification values" obtained from postulated accident analysis in the "SPEC" column. List the "qualification values" obtained from test reports, engineering analysis data, etc. in the "Qual" column. Temperature, pressure, etc., as a function of time shall be provided in profile or tabular form. Specify the time period that the component or equipment is required to function and identify the document which provides the basis for this time interval.

   It is expected that some listed parameters were not requested of the licensee at the time of their license issuance. Address each parameter condition during this review. If it is determined that a parameter such as submergence or a service condition such as aging was not previously considered, identify it as an "Outstanding Item."

3. Documentation Reference: Reference the documents from which information was obtained in the "Spec" column. Identify the document, paragraph, etc., that contains the postulated accident environmental specification data. In the "Qual" column identify the document, paragraph, etc., that contains the environmental qualification data.

4. Qualification Method: Identify the method of qualification. To describe the qualification method use words such as simultaneous test, comparison test, sequential test, and/or engineering/mathematical analysis. Words such as "test" and/or "analysis" when used alone do not adequately identify the qualification method.

5. Outstanding Items: Identify parameters for which no qualification data is presently available. Also, identify parameters, service conditions, or environments not previously addressed during FSAR environmental qualification analysis such as submergence, qualified life (aging), or HELB. Identify in the "Notes" section on page 1 of this enclosure the actions planned for determining qualification and the schedule for completing these actions.

Facility:
Unit:
Docket:

SYSTEM COMPONENT EVALUATION WORK SHEET
(Typical)

EQUIPMENT DESCRIPTION	ENVIRONMENT			DOCUMENTATION REF*		QUALIFICATION METHOD	OUTSTANDING ITEMS
	Parameter	Specification	Qualification	Specification	Qualification		
System: RHR Plant ID No. IPT456	Operating Time	15 min.	300 min.	1	5	Simultaneous Test	None
Component: PRESSURE TRANSMITTER	Temperature (°F)	SEE ACCIDENT AND TEST PROFILES PROVIDED		1	5	Simultaneous Test	None
Manufacture: Fischer-Porter Co.	Pressure (PSIA)			1	5	Simultaneous Test	None
Model Number: 50-EN-1071-BCXN-NS	Relative Humidity(%)	100%	100%	1	5	Simultaneous Test	None
Function: Accident Monitoring	Chemical Spray	N_3BO_3/ NAOH		1			See Note 1
Accuracy: Spec: 5% Demon: 4%	Radiation	$4x10^6$ rads	$1.2x10^8$ rads	2	6	Sequential Test	None
Service: RHR Pump 1A Discharge Pressure S/N107 Location: Containment	Aging	40 yrs	40 yrs	3	7, 8	1. Sequential Test 2. Eng. Analysis	None
Flood Level Elev: 620' Above Flood Level: Yes No x	Submergence	Not Required	Not Required				None See Note 2

*Documentation References:

1. FSAR Chapter 3, Paragraph 3.11
2. FSAR Chapter 14, Paragraph 14.2.3
3. Technical Specification 3.4.1, Paragraph A
4. Technical Specification 4.6.5, Paragraph B
5. FIRL Test Report No. 3600 dated November 2, 1972
6. Fischer and Porter Co. Test Report No. 2500-1
7. A. B. DOD Engineering Evaluation Data Report No. 6932
8. Wylie Laboratory Report No. 467

Notes:

1. XYZ Letter No. 237-1, dated November 2, 1979, has been sent to MFG. requesting the qualification information. If qualification not determined acceptable by December 15, 1979, component will be replaced during refueling outage March 1980.

2. In the FSAR submergence was not considered an environmental parameter. ABC Laboratory is to perform submergence test in April 1980.

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.

Equipment 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

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[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 safety 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 equipm..nt 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 m..t 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 pene rations), 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

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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 reactors 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 2×10^6 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 Selection of Qualification Method

The cho ce 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 Qualification by Type Testing

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

1. Simulated Service Conditions and Test Duration - 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 C.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 agin (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 de ines 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 Fai e 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.

6.0 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 procedure 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:

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

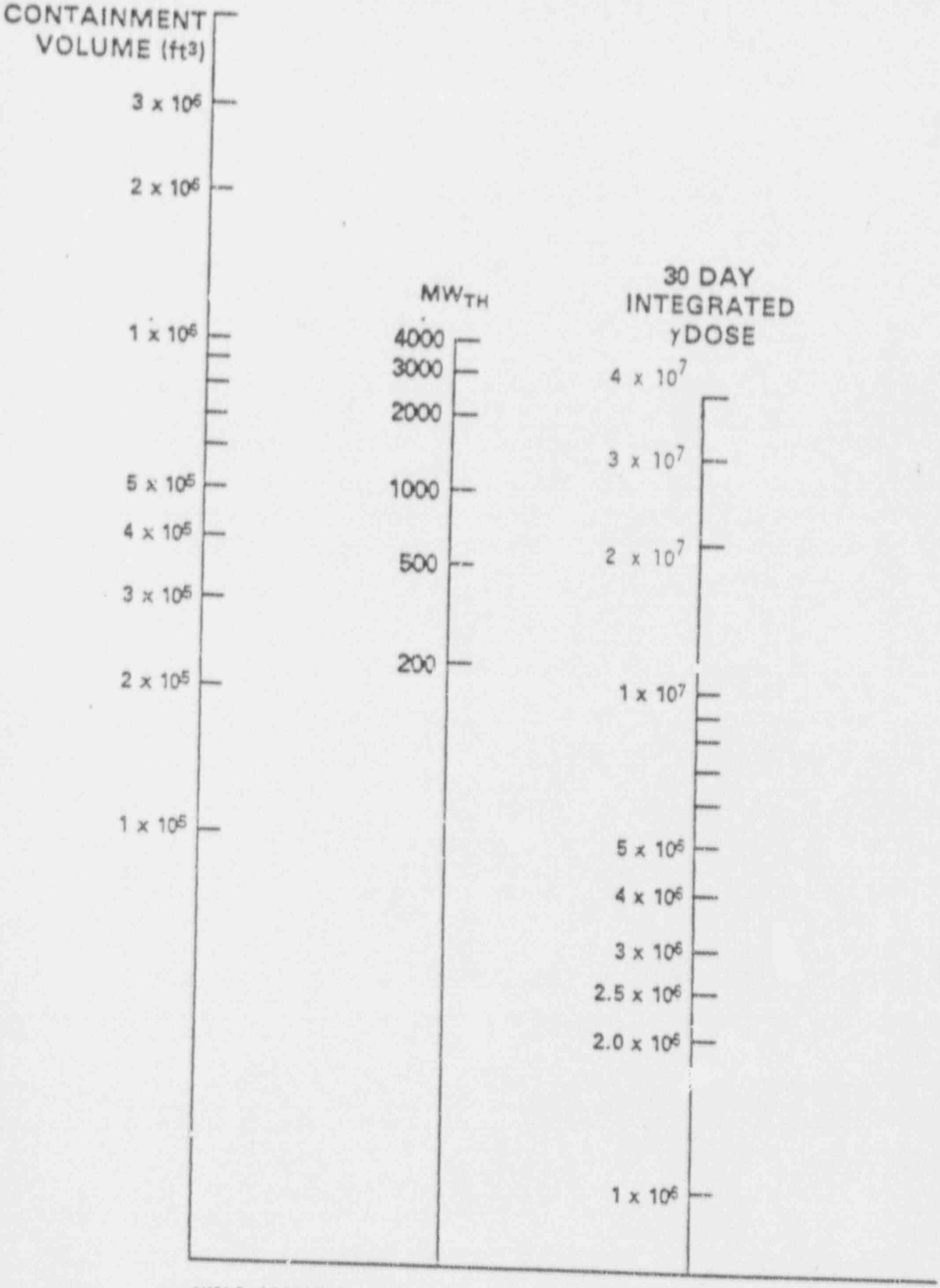
Step 4

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.

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

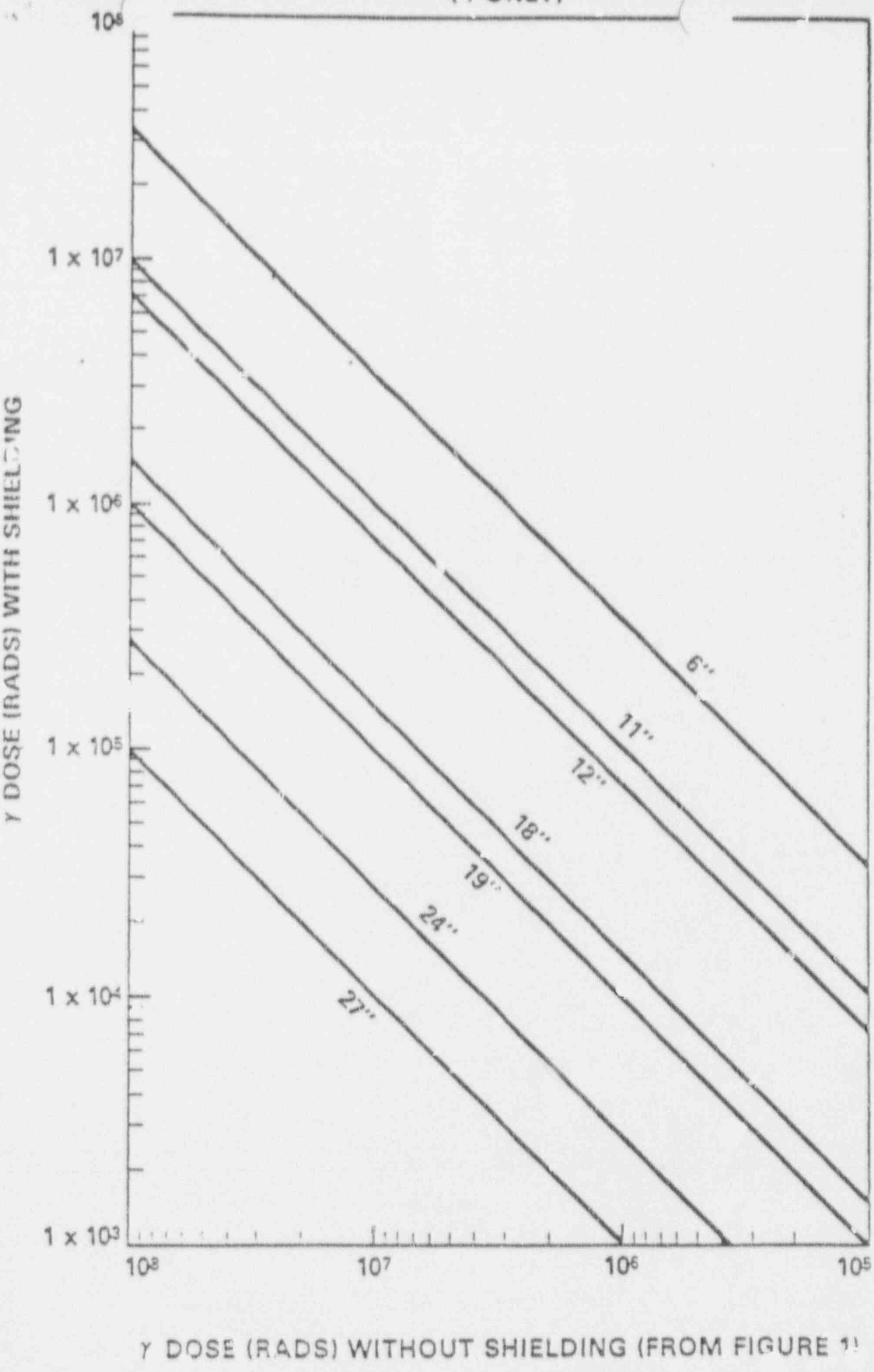
In this particular example the service condition of 2×10^6 RADS specified is conservative with respect to the estimated dose of 3×10^5 RADS calculated in steps 1 through 4 and is, therefore, acceptable.

NOMOGRAM FOR CONTAINMENT VOLUME AND REACTOR POWER
LOCA DOSE CORRECTIONS*



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

(γ ONLY)



10^8
1×10^7
1×10^6
1×10^5
1×10^4
1×10^3
γ DOSE (RADS) WITH SHIELDING
6"
11"
12"
18"
19"
24"
27"
10^8
10^7
10^6
10^5
γ DOSE (RADS) WITHOUT SHIELDING (FROM FIGURE 1)

FIGURE 3
DOSE CORRECTION FACTOR FOR COMPARTMENT VOLUME

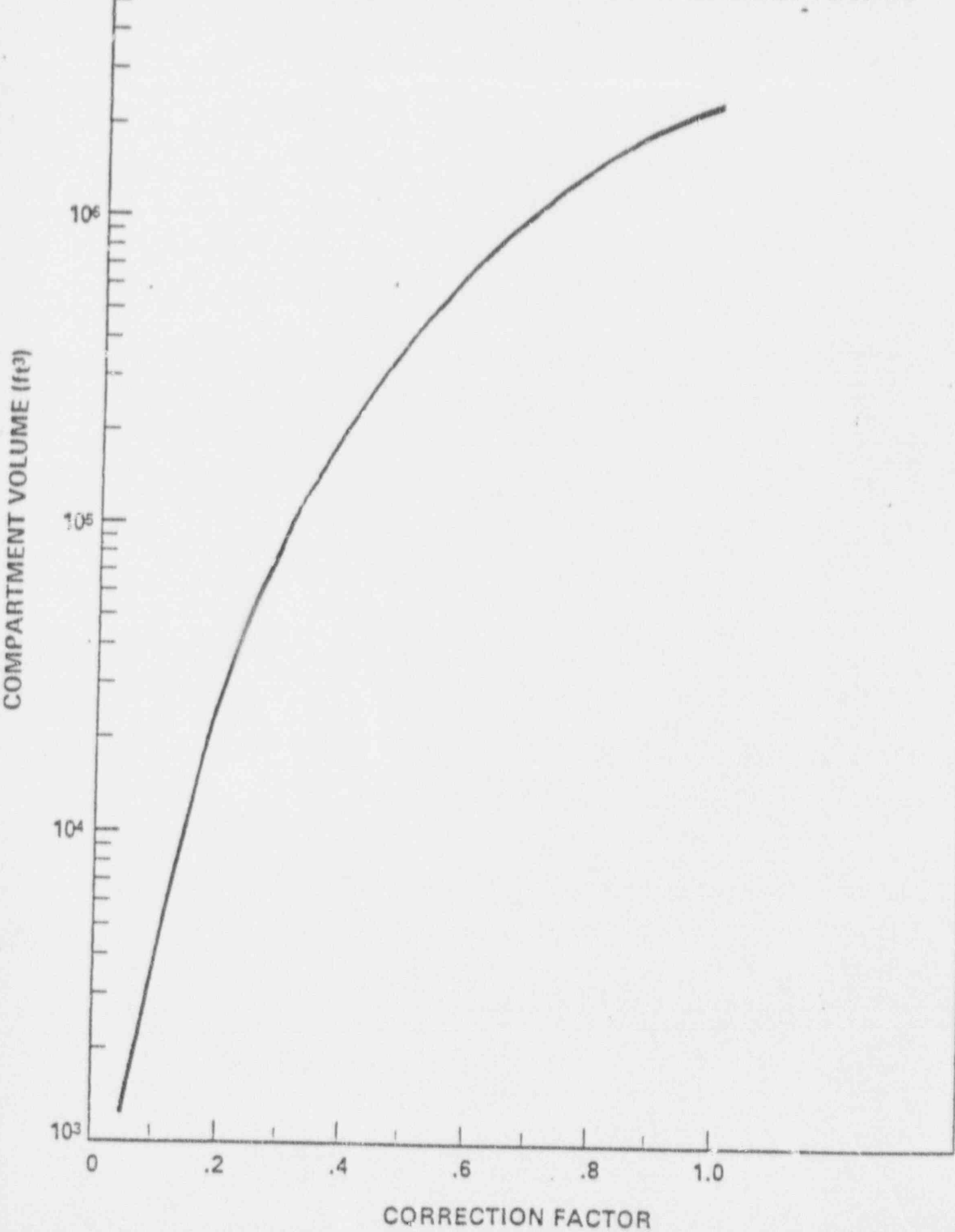


FIGURE 4

DOSE CORRECTION FOR TIME REQUIRED TO REMAIN FUNCTIONAL

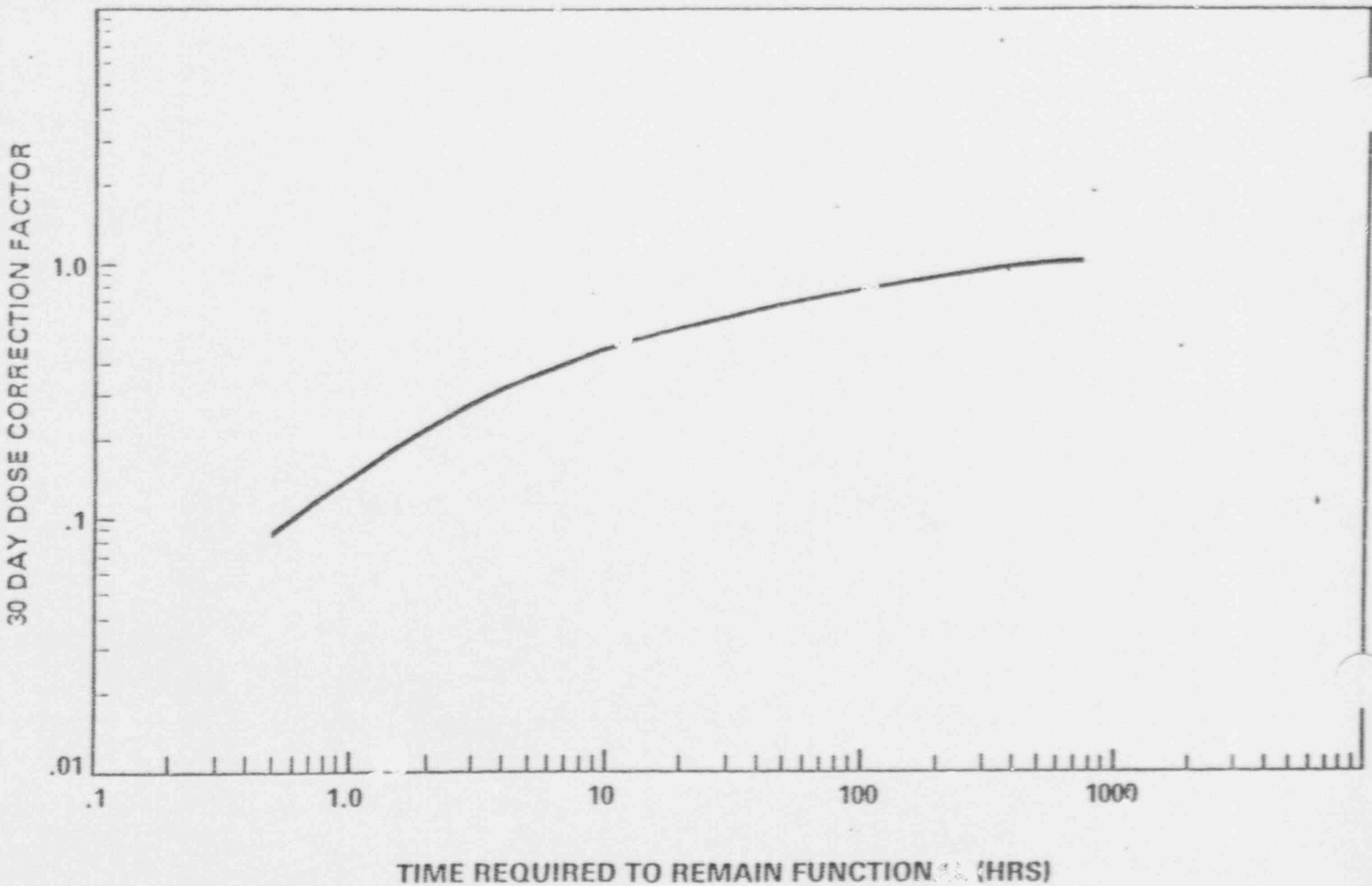


TABLE C-1

THERMAL AND RADIATION AGING DEGRADATION OF SELECTED MATERIALS

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING: 10 YRS	POTENTIAL FOR SIGNIFICANT AGING: 40 YRS	RADIATION SUSCEPTIBILITY: RADS GAMMA	RADIATION SUSCEPTIBILITY: BASIS	TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND): 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	SBR/Nitrile Rubber	*	*	10^6	Threshold				X		X					X		X			X	
Integrated Circuits (IC) TTL				10^6	-				X	X			X				X	X		X	X	
Diallyl Pht.	DAP			10^6	-																	
Silicone R				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.

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.

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING: 10 YRS	40 YRS	RADIATION SUSCEPTIBILITY: RADS GAMMA	BASIS	TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND): CABLE	CONNECTORS	ELECTRICAL PENETRATIONS	HEATERS	INSTRUMENT RACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENERATOR CONTROL EQUIPMENT	FANS	SOLID EQUIPMENT	MCC, LIGHTING CENTERS	SWITCHGEAR
SBR Rubber			*	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 Circuit's (IC) ECL				10^7	-				X	X							X			X		
Formvar				10^7	-	X			X								X	X	X		X	X
NEMA Polyester Glass Laminates, Gr?) 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				10^3	Threshold	X			X									X	X		X	
Cross-Linked Polyethylene			*	10^7	Allowable																	
Capacitors - Mylar				10^8	-				X	X							X	X		X	X	

11/14/79

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING 10 YRS	POTENTIAL FOR SIGNIFICANT AGING 40 YRS	RADIATION SUSCEPTIBILITY RADS GAMMA	RADIATION SUSCEPTIBILITY BASIS	CABLE	CONNECTORS	ELECTRICAL PENETRATIONS	HEATERS	INSTRUMENT RACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENER. & CONTROL EQUIPMENT	FANS	LOGIC EQUIPMENT	MTR. CONTROL CENTERS	SWITCHGEAR
Polysulfone				10^7	Allowable										X							
Nomex	Aramide			10^8	24% Loss of Elongation				X	X		X					X	X	X		X	
Resistors - Wire-Wound				10^9	Threshold				X	X			X				X	X		X	X	
Resistors - Carbon Composition				10^9	-				X	X			X				X	X		X	X	
Capacitors - Ceramic				10^9	Allowable				X	X							X	X		X	X	
Capacitors - Glass				10^9	-				X	X							X	X		X	X	
Capacitors - Mica				10^9	-				X	X							X	X		X	X	
NEMA Thermosetting Laminates, Grade XXXPC				10^9	-					X							X					
NEMA Thermosetting Laminates, Grade XXXP				10^9	-					X							X					
NEMA Thermosetting Laminates, Grade XPX				10^9	-					X							X					
NEMA Thermosetting Laminates, Grade XPC				10^9	-					X							X					
NEMA Thermosetting Laminates, Grade XX				10^9	-					X	X						X	X			X	X
NEMA Thermosetting Laminates, Grade XXP				10^9	-					X	X						X	X			X	X
NEMA Thermosetting Laminates, Grade XXX				10^9	-					X	X						X	X			X	X
NEMA Thermosetting Laminates, Grade CE				10^9	-																	X
NEMA Thermosetting Laminates, Grade C				10^9	-																	X

TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND)

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING: 10 YRS	POTENTIAL FOR SIGNIFICANT AGING: 40 YRS	RADIATION SUSCEPTIBILITY: RADS GAMMA	RADIATION SUSCEPTIBILITY: BASIS	CABLE	CONNECTORS	ELECTRICAL PENETRATIONS	HEATERS	INSTRUMENT RACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENER. AUX CONTROL EQUIPMENT	FANS	STATIC INVERTERS	MTR. CONTROL CENTERS	SWITCHGEAR
NEMA Thermosetting Laminates, Grade L				10^9	Threshold																	X
NEMA Thermosetting Laminates, Grade LE				10^9	"																	X
NEMA Thermosetting Laminates, Grade FR-2		*	*	10^9	"					X												X
NEMA Thermosetting Laminates, Grade FR-3				10^9	"					X												
NEMA Thermosetting Laminates, Grade FR-4				10^9	"					X												X
NEMA Thermosettig Laminates, Grade FR-10				10^9	"				X	X							X	X		X	X	
NEMA Thermosetting Laminates, Grade A				10^9	"					X												
NEMA Thermosetting Laminates, Grade AA				10^9	"					X												
NEMA Thermosetting Laminates, Grado G-3				10^9	"					X												
NEMA Thermosetting Laminates, Grade G-11				10^9	"					X												
NEMA Thermosetting Laminates, Grade FR-5				10^9	"					X												
Resistors - Film				10^{10}	"				X	X			X				X	X		X	X	