AT THE 45TH ANNIVERSARY OF IEEE 323, A PERSPECTIVE

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WARNING

R	RESTRICTED
	UNDER 17 REQUIRES ACCOMPANING PARENT OR GUARDIAN

WOW!

Q1: WHAT YEAR WAS IEEE 323 ISSUED?

A: 1971

B: 1974

1971

USA The Walt Disney World Resort opens in Florida

- The Voting Age in the United States is lowered to 18 years old

World Decimal Day for United Kingdom and Ireland

- Intel releases world's first microprocessor, the 4004.

Costs Dodge Charger $3,579

- Cost of a gallon of Gas 40 cents

IEEE IEEE 323-1971

- A new Career

TOPICS

Formative Years
- Beginning
- Research
- IE Circular 78-08

79-01b Era
- TMI
- Laws
- More Research

For the Ages
- Life Extension
- Status quo

QA
- NQA-1
- CGD

Fukushima
- BDBA
- SA

IEEE No 323
April 1971

IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations

Sponsored by the

Joint Committee on Nuclear Power Standards

of the

IEEE Group on Nuclear Science

and the

IEEE Power Society

Q2: IS DOUBLE PEAK A QUALIFICATION ATTRIBUTE?

A: Yes

B: No

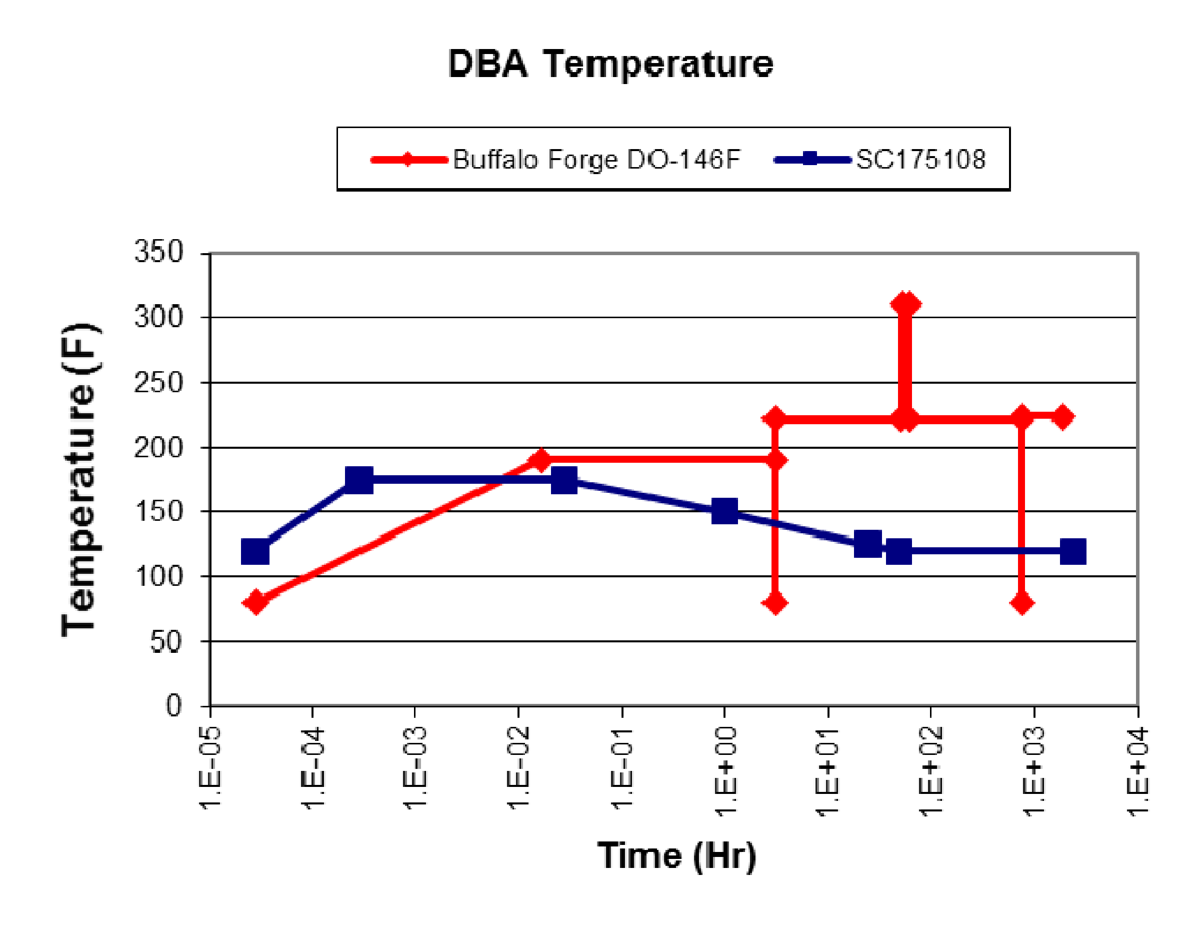
GLSEQ, LLC

Q3: IS DBA TEST CURVE EXAMPLE A QUALIFICATION FAILURE?

A: Yes

B: No

DBA TEST CURVE EXAMPLE



Q4: ARE DBA RAMP RATES REQUIREMENTS?

A: Yes

B: No

Q5: ONE THIRD / TWO THIRDS, A BASIS ?

A: Yes

B: No

Q6: IS THERE AN IEEE 323A-1975?

A: No

B: Yes

Q7: LOW DOSE EFFECTS, A BASIS ?

A: Yes

B: No

Q8: IS COMMERCIAL GRADE DEDICATION EQ?

A: Yes

B: No

Q9: MENISCUS, IMPORTANT?

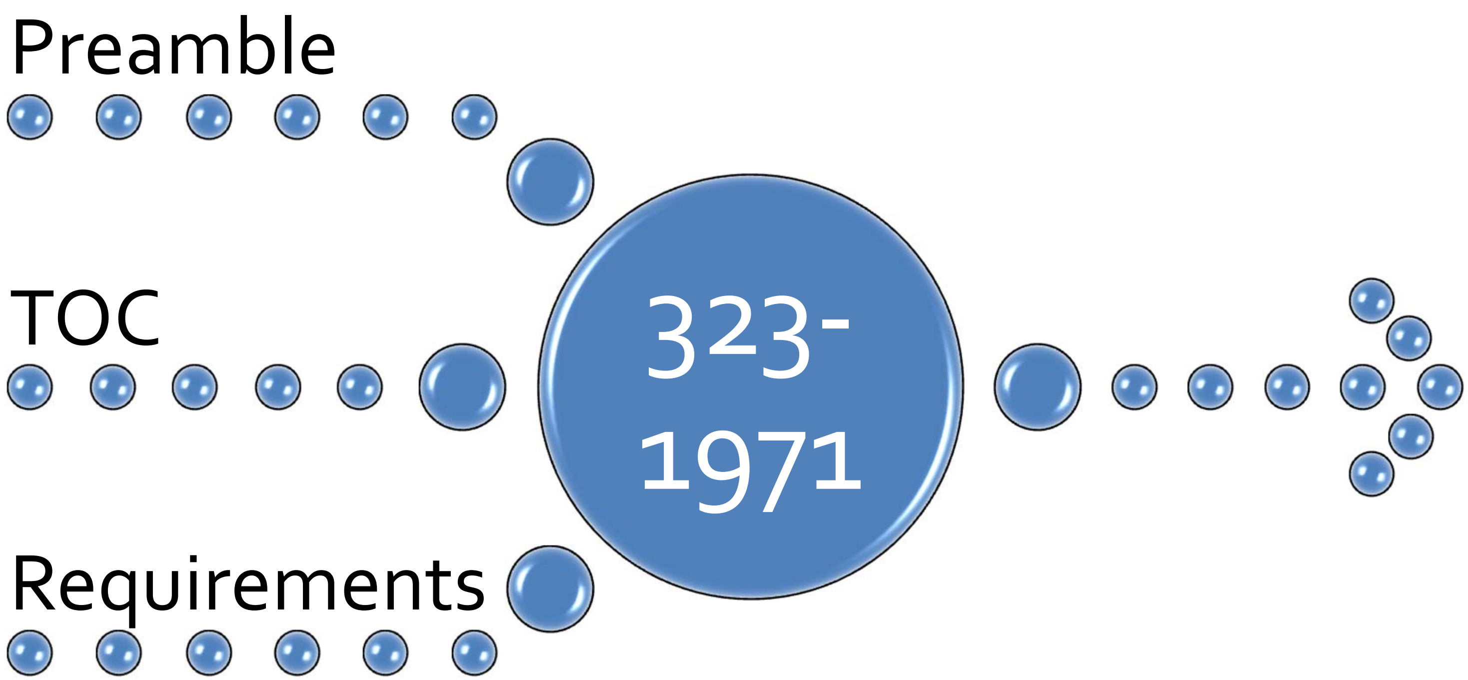
A: Yes

B: No

SEISMIC

IEEE	Rest of world
g's	Scale
RRS	Severity

Q10: 323-1971 HOW MANY PAGES?



Q11: IS THERE A STANDARD OR REQUIREMENT FOR THERMAL LAG?

A: Yes

B: No

Q12: HOW MANY ACTIVATION ENERGIES WERE PROVIDED IN THE ORIGINAL EPRI NP-1558 A REVIEW OF EQUIPMENT AGING THEORY AND TECHNOLOGY?

A: 0

B: >0

IEEE
Std 323A-1975
(Supplement to
IEEE Std 323-1974)

Supplement to the Foreword of IEEE Std 323-1974
IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

(This Supplement is provided to amplify the information in the Foreword; it is not a part of IEEE Std 323-1974)

The concept of aging was addressed explicitly for the first time in IEEE Std 323-1974. The aging guidance therein reflects the requirement of IEEE Std 279-1971 (ANSI N42.7-1972) Section 4.4. It is based on an awareness by the IEEE that the ability of Class 1E equipment to perform its safety related function might be affected by changes due to natural, operational, and environmental phenomena over time (aging). It was not the intent that aging must be applied to all Class 1E equipment, but rather that aging must be considered in the same manner as environmental parameters. The need for aging of particular equipment should be determined based on an evaluation of the specific design and application. If aging is needed a further determination must be made as to whether accelerated aging techniques can be applied to the equipment and yield valid results, that may be correlated to real time, ongoing qualification.

It is acknowledged that the state-of-the-art regarding aging for some Class 1E equipment is more advanced than others. It is expected that known technology will be utilized in any aging program. Optionally (and particularly where the state-of-the-art is limiting), aging as part of the qualification program may be addressed by operating experience, analysis, combined, or ongoing qualification as detailed in Section 5.2, 5.3, 5.4 and 5.5.

Further clarification of aging as it applies to specific types of equipment will be provided in individual IEEE Class 1E equipment qualification documents. For example, an IEEE Standard is being prepared to establish criteria for Class 1E modules. IEEE Standards 334-1974, 382-1972 (ANSI N41.6), and 383-1974 (ANSI N41.10-1975) presently provide guidance for motors, valve operators, and cables respectively.

Nov 21,1975

NRC RESEARCH: BILL FARMER

Sandia: Bonson, Gillen, Clough, & Jacobus

Carfagno, Gleason, & Rhodes

NUREG/CR-6384

GLSEQ, LLC

EPRI RESEARCH: GEORGE SLITER

Carfagno

NP-1558 Aging Theory

Gleason

NP-3326 Seismic - Aging Correlation
NP-5024 Seismic Ruggedness
NP-5000 Sealing Handbook
NP-6408 Shelf Life
NP-6731 Optimized Seal Replacement

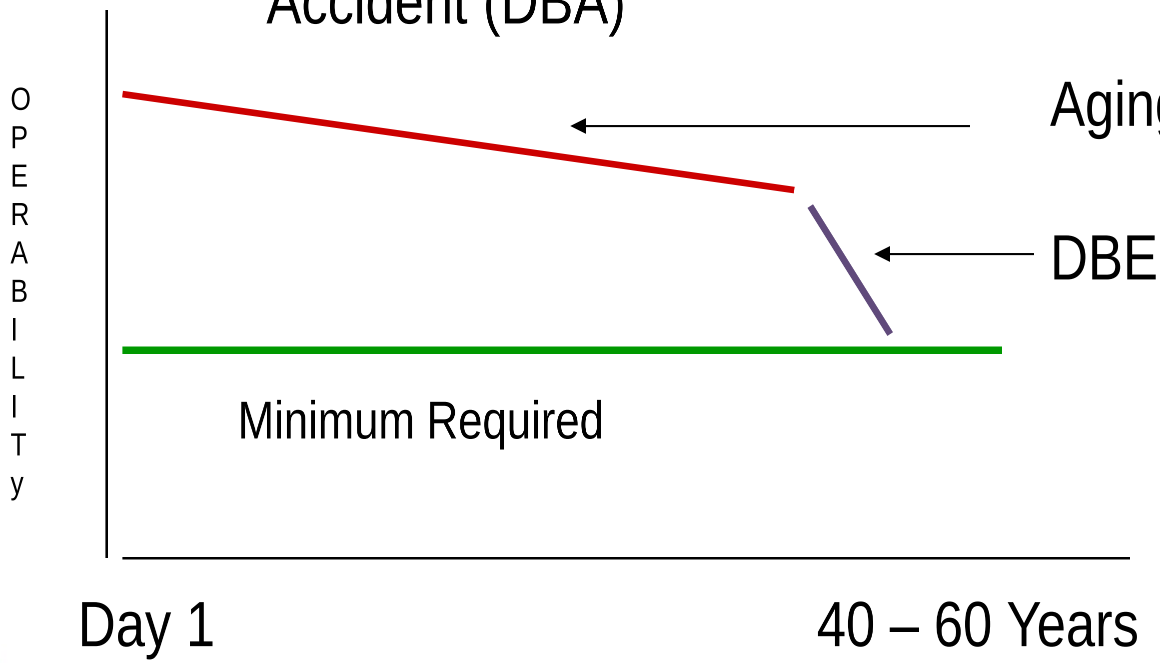
EQ CONCEPTS

DBE – Design Basis Event

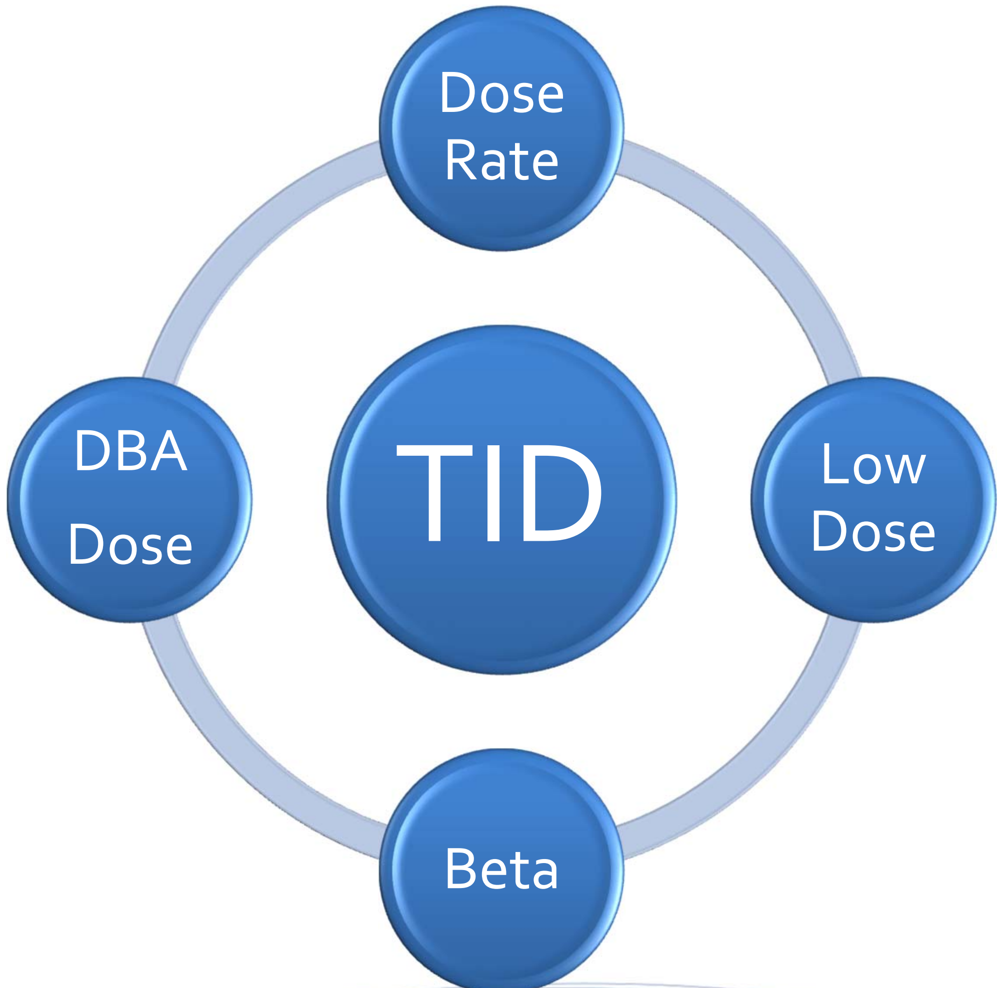
Seismic

Accident (DBA)

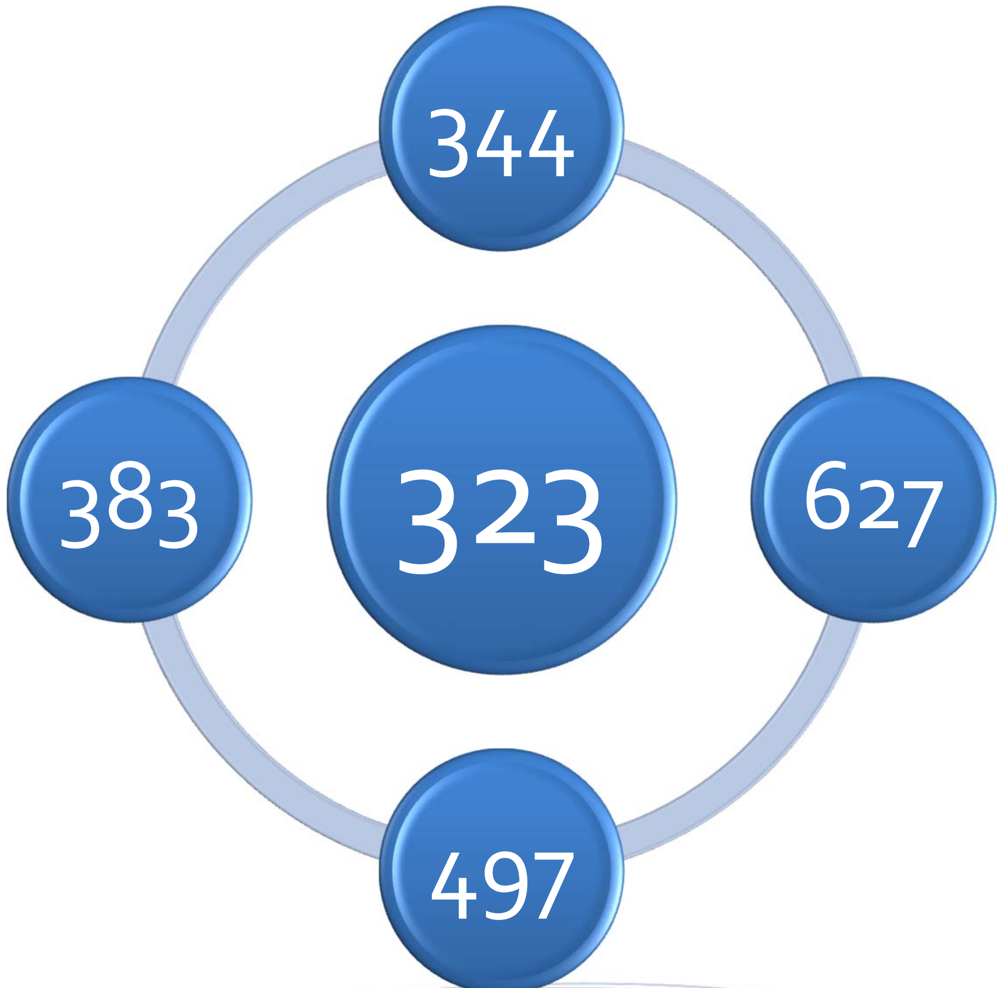
DBE Can Happen Anytime



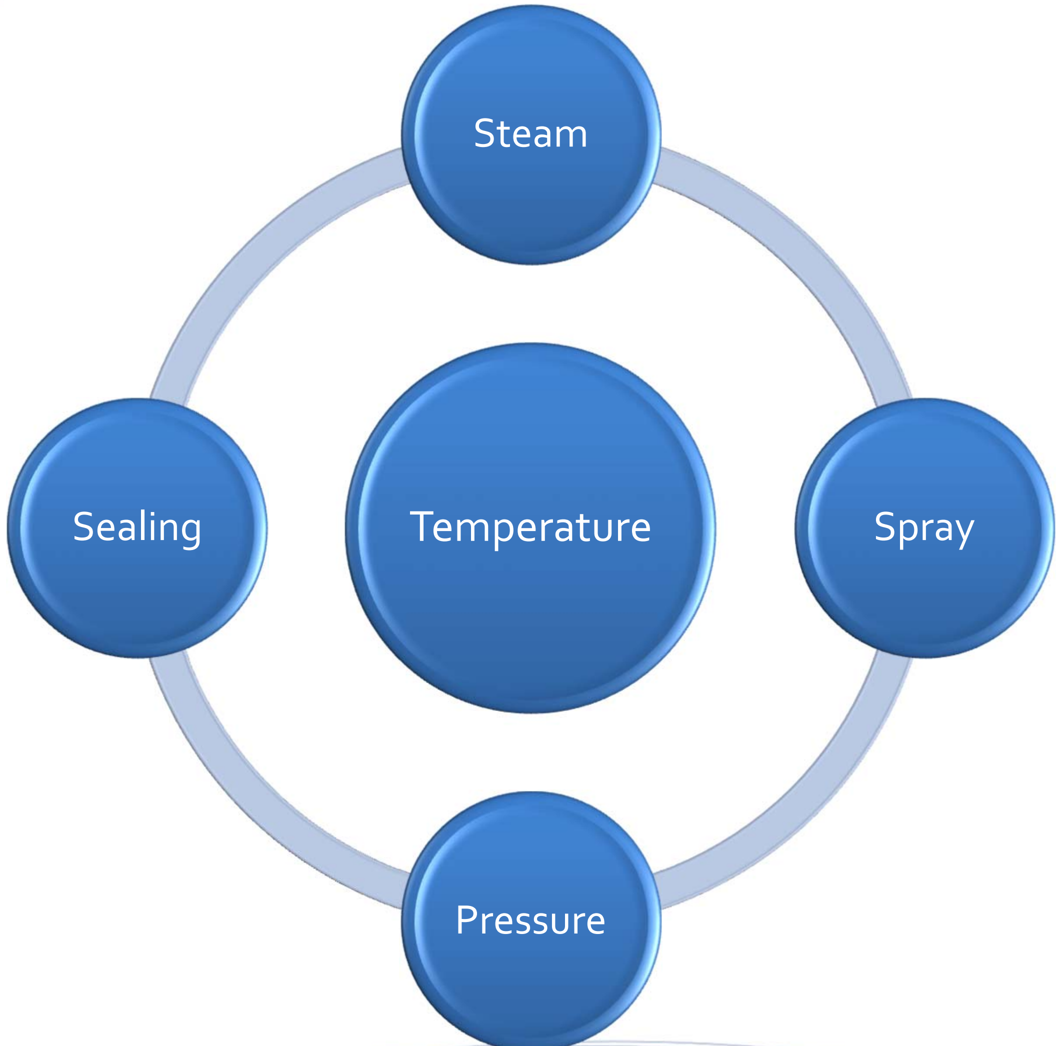
RADIATION



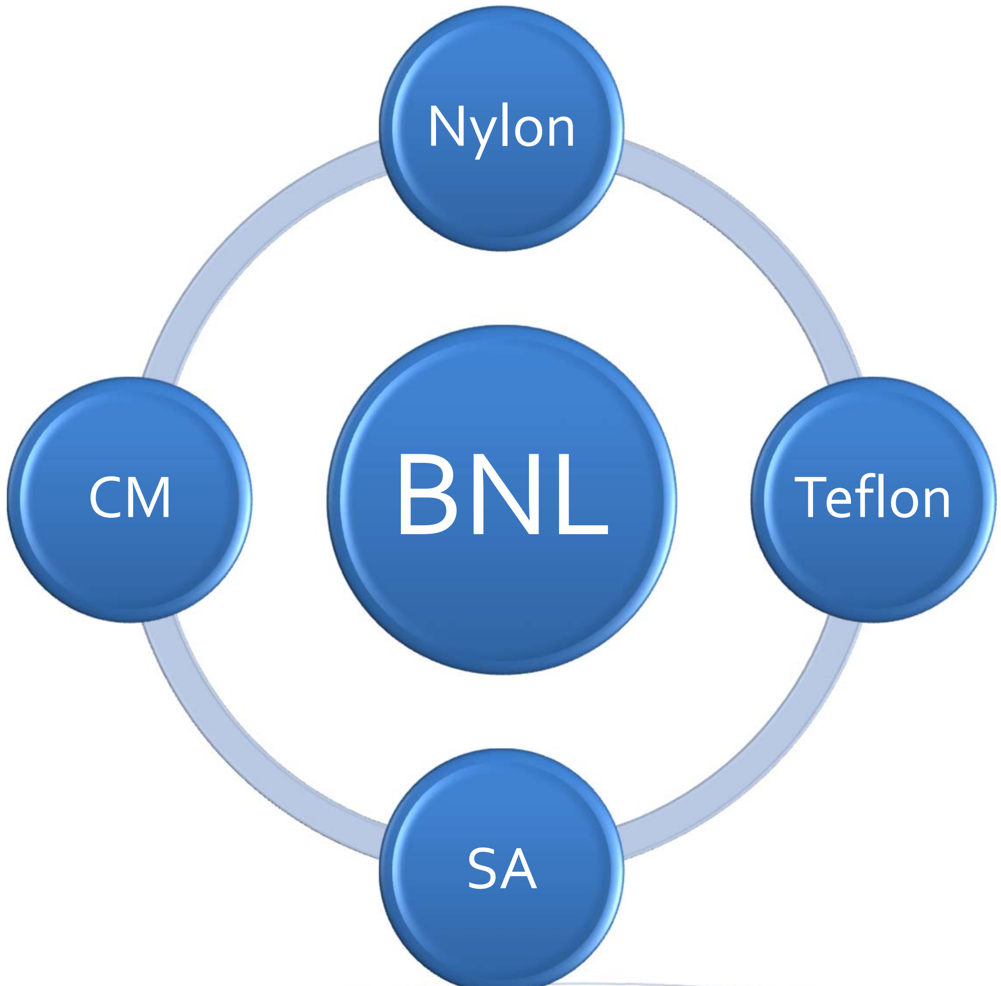
IEEE



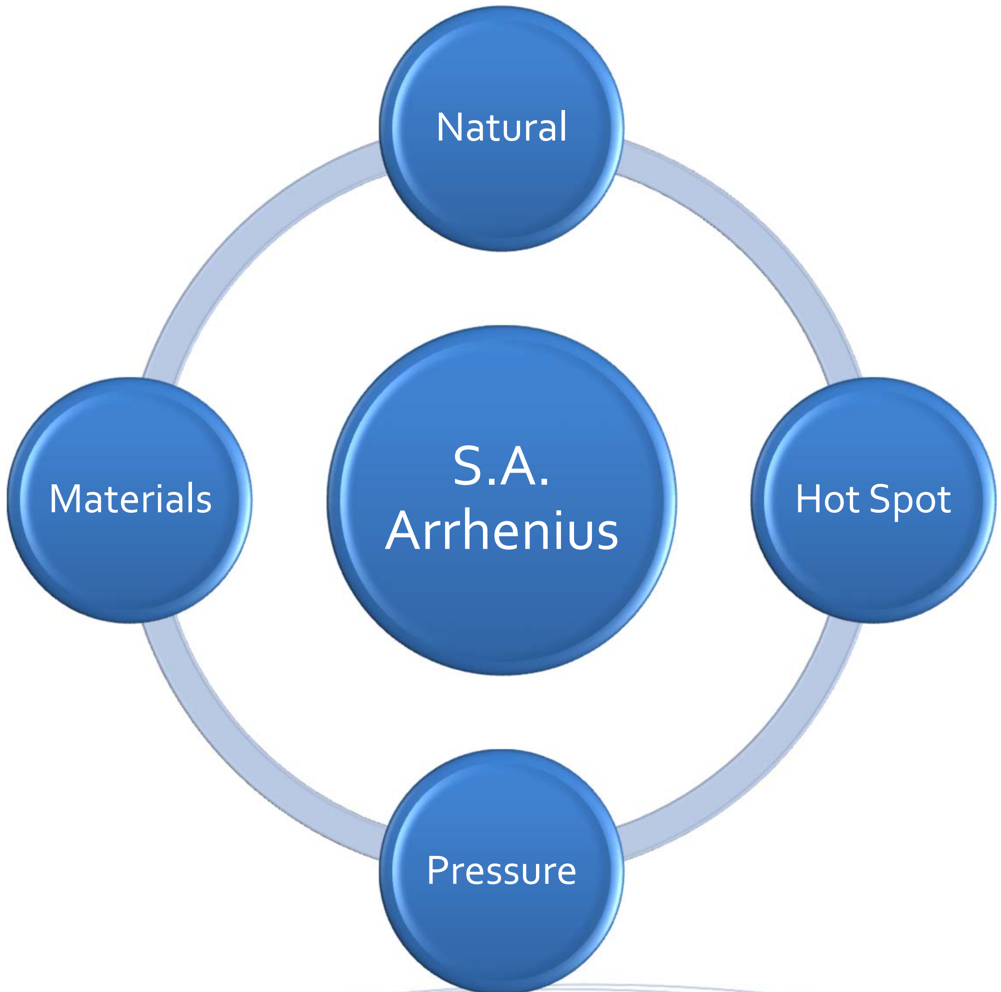
LOCA



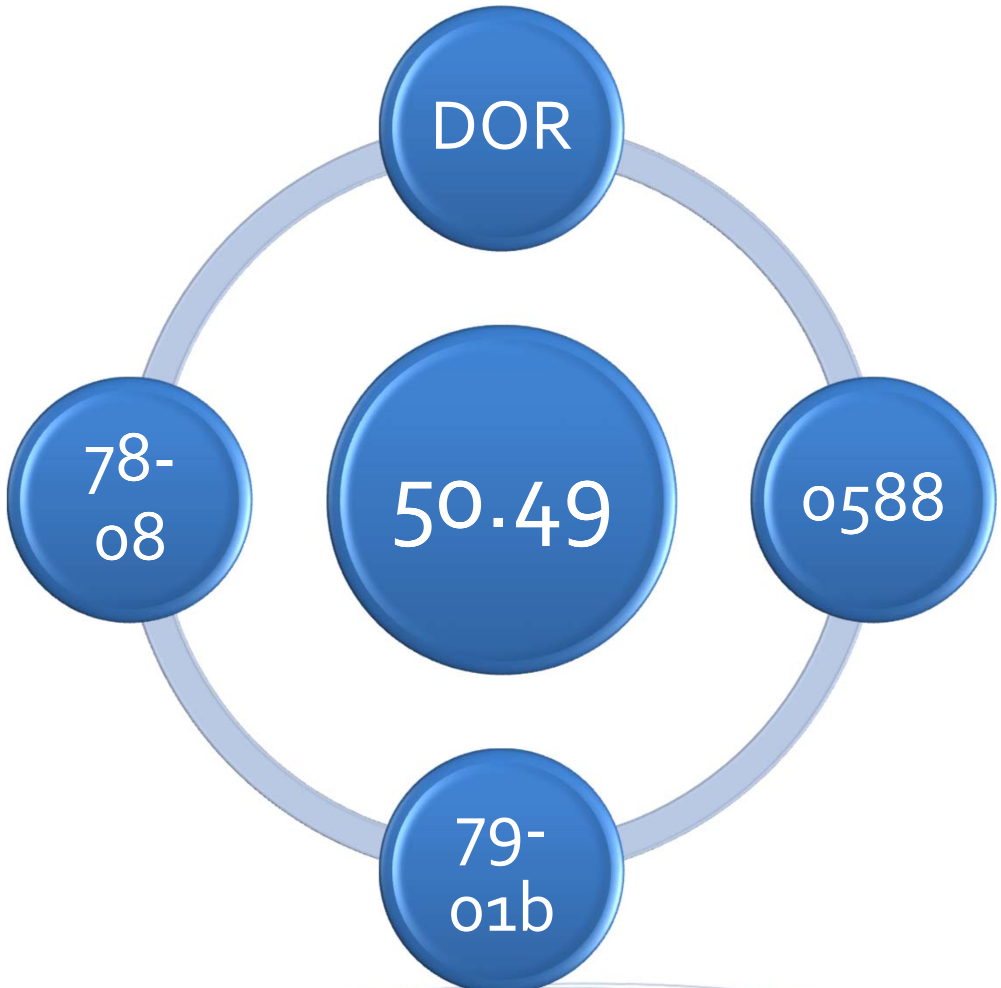
LESSONS LEARNED



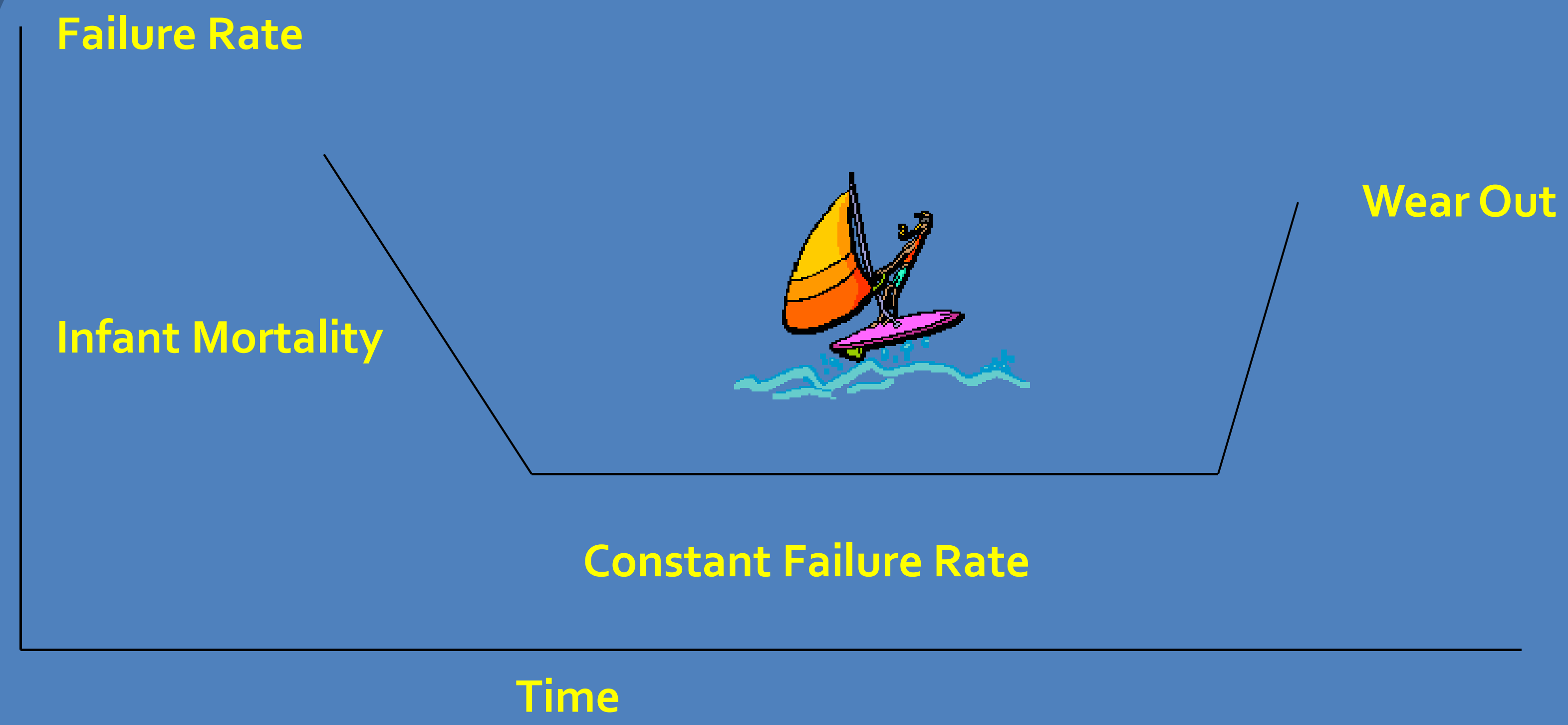
AGING



REGULATIONS



BATHTUB CURVE



Q13: IS CLASS 1E EQUIPMENT DESIGNED AS FAIL SAFE AND THE SAFETY FUNCTION IS THE FAIL SAFE POSITION, NEED TO BE QUALIFIED?

A: Yes

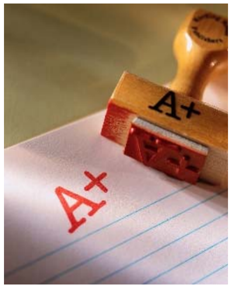
B: No

AMUCK

- a murderous frenzy that occurs chiefly among Malays

AMUCK UNIVERSITY

- Risk Amuck
- Specification Amuck
- Commercial Grade Dedication Amuck
- QA Amuck
- Safety Culture Amuck



IE CIRCULAR NO. 78-08

- Environmental Qualification of Safety- Related Electrical Equipment at Nuclear Power Plants
  - Commission's April 13, 1978 Order in response to a petition from the Union of Concerned Scientists.
  - specific deficiencies were identified
  - Poor installation practices
  - inadequate consideration of subcomponents
  - omission of certain environmental parameters in the design

Bendix Connector test at Wyle

Bulletin 77-05: Electrical Connector Assemblies

BULLETIN 77-05: ELECTRICAL CONNECTOR ASSEMBLIES

- Sandia Laboratories
- electrical connector/cable assemblies
- <u>While electrical connectors of the type tested are not normally used in applications that are required to survive LOCA conditions</u>, it is not possible in the absence of specific information to conclude that such applications do not exist.
- Determine whether your facility <u>utilizes or plans to utilize</u> electrical connector assemblies of the type tested by Sandia Laboratories, <u>or any other similar type</u>, in systems that are located inside containment, are subject to a LOCA environment and are required to be operable during a LOCA.

HMMM? WHERE IS THIS FROM?

6.3.3 Aging. The assessment of equipment aging effects is required to determine if aging has a significant effect on operability. The types of aging include thermal, radiation, wear, and vibration. The assessment shall include an analysis of the equipment to determine any significant aging mechanisms for the DBEs under consideration. Where these mechanisms are identified, a suitable aging subprogram shall be included in the type test unless excluded in 6.2.1. When natural aging is used in the qualification program it is not necessary to identify significant aging mechanisms.

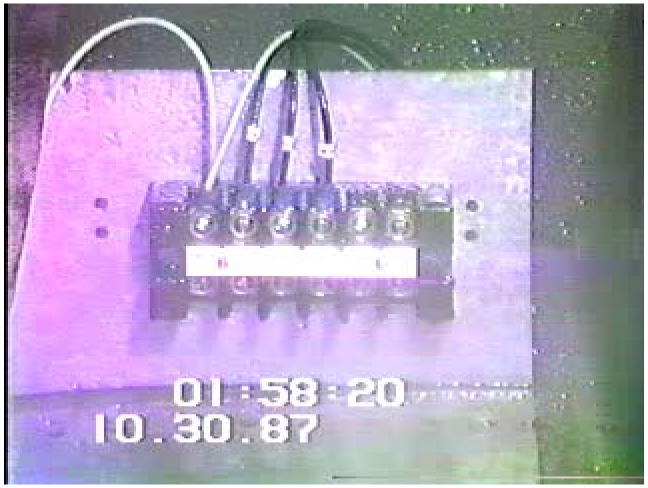
6.3.3.1 Natural Aging. Natural aging is the most technically justified method. Naturally aged equipment may be used for type testing provided that:

(1) The equipment has been aged in an environment at least as severe as the normal one for the intended application

(2) Operating and maintenance/replacement records are available to verify the service conditions

(3) The aged equipment was operated under load at least as severe as that specified for the equipment to be qualified.

MENISCUS



IE BULLETIN 79-01B

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

Description 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 equipme': 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.

DOR GUIDELINES

Enclosure 4

GUIDELINES FOR EVALUATING ENVIRONMENTAL QUALIFICATION

OF CLASS IE ELECTRICAL EQUIPMENT

IN OPERATING REACTORS

1.0 Introduction

2.0 Discussion

3.0 Identification of Class IE Equipment

4.0 Service Conditions

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

1. Temperature and Pressure Steam Conditions

2. Radiation

3. Submergence

4. Chemical Sprays

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

DOR GUIDELINES

TABLE

THERMAL AND RADIATION AGING DEGRADATION

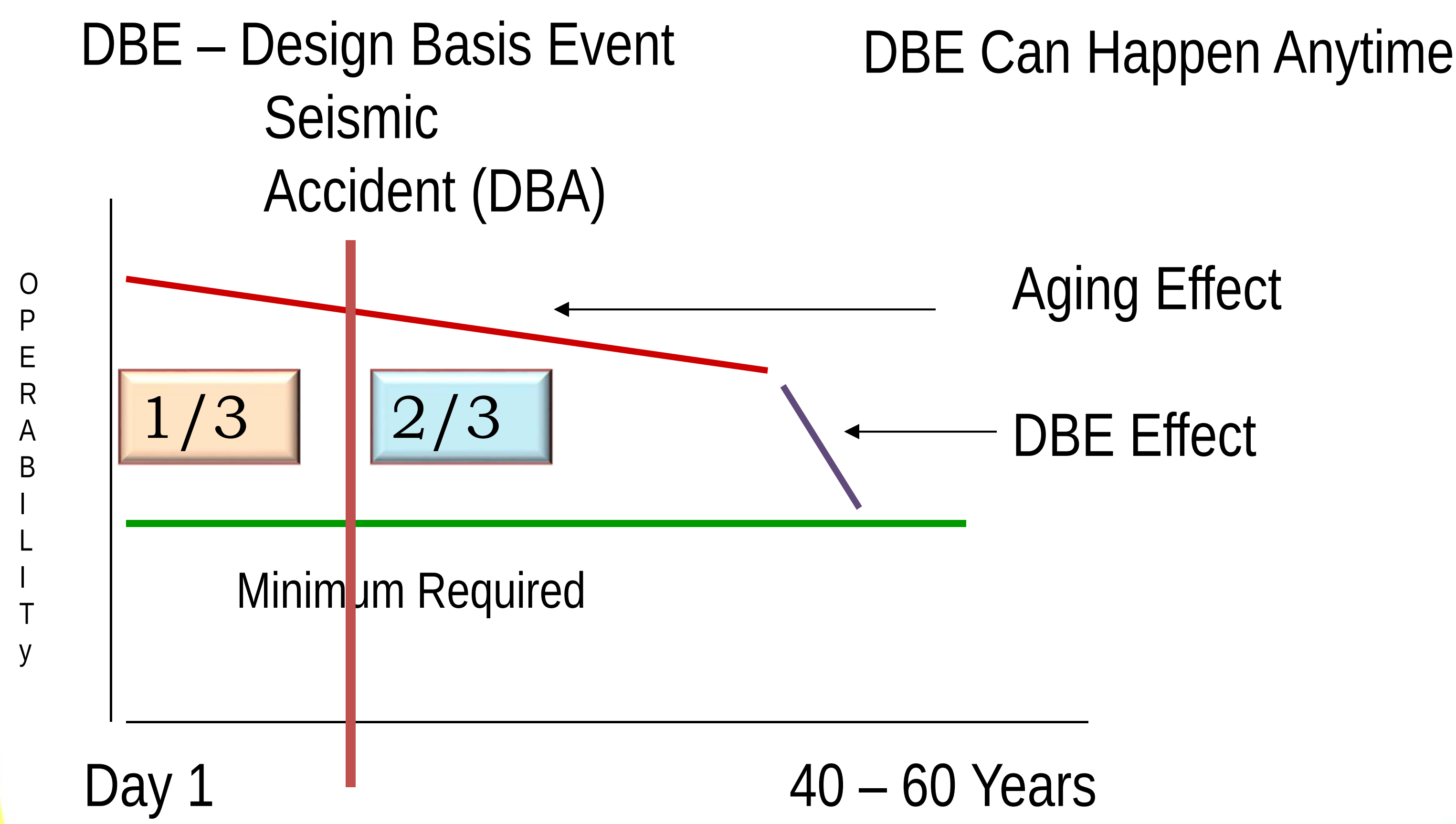
OF SELECTED MATERIALS

MATERIAL	ALSO KNOWN AS	POTENTIAL FOR SIGNIFICANT AGING: 10 YRS	POTENTIAL FOR SIGNIFICANT AGING: 40 YRS	RADIATION SUSCEPTIBILITY: RADS GAMMA	RADIATION SUSCEPTIBILITY: BASIS	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	[illegible]	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.

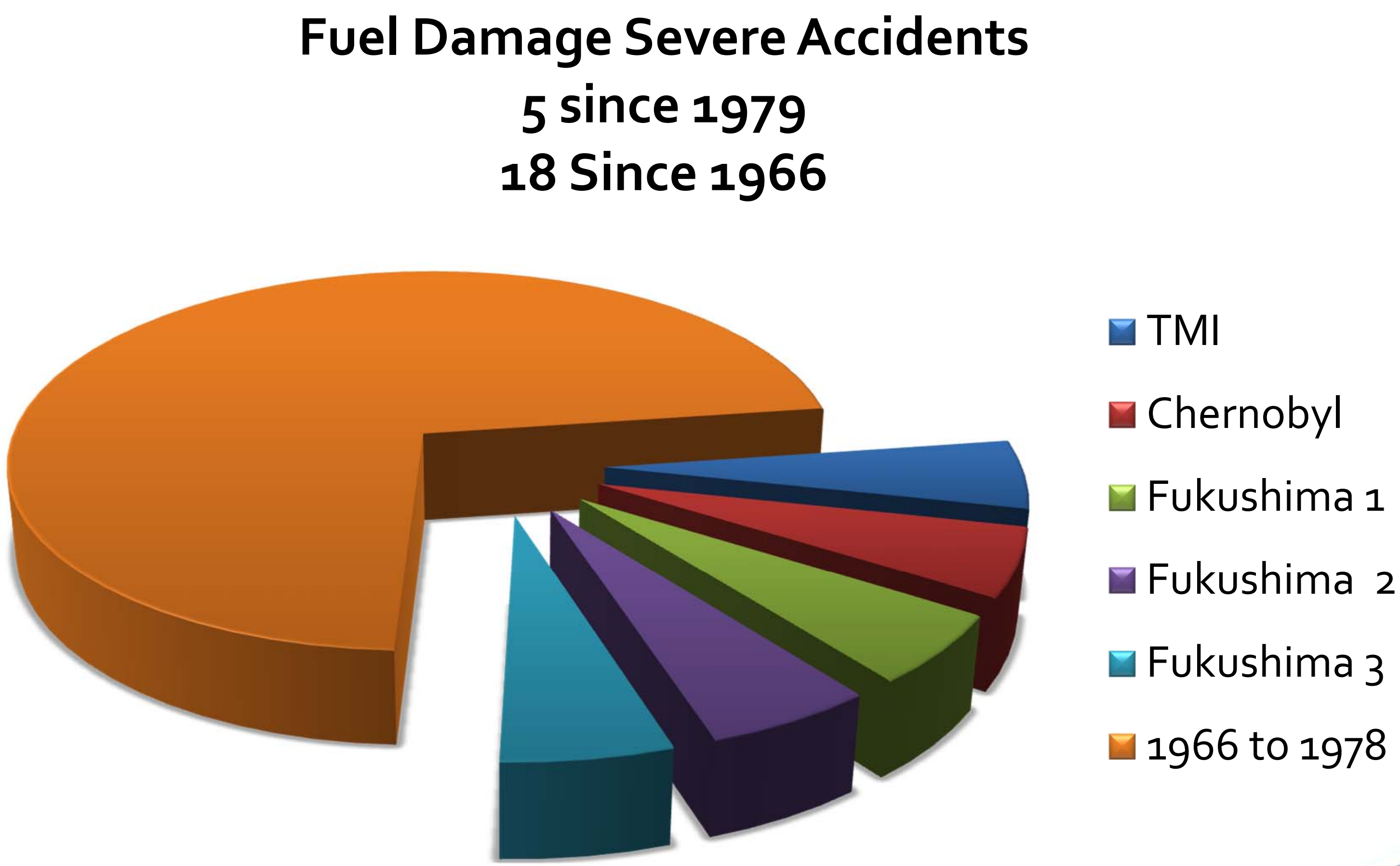
GLSEQ,LLC

EQ CONCEPTS ONE THIRD/TWO THIRDS

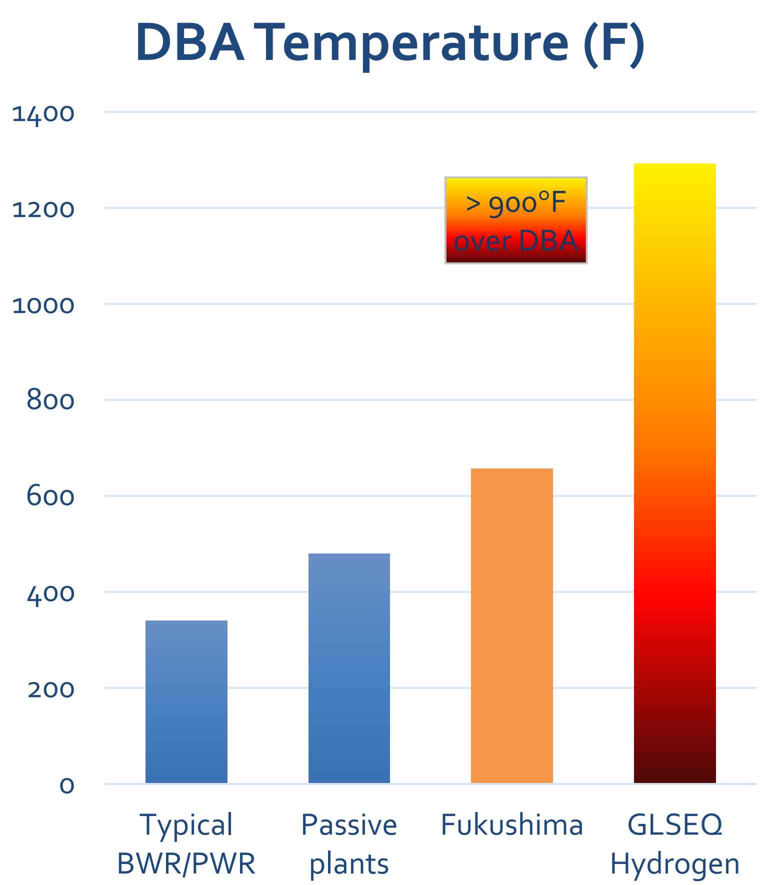


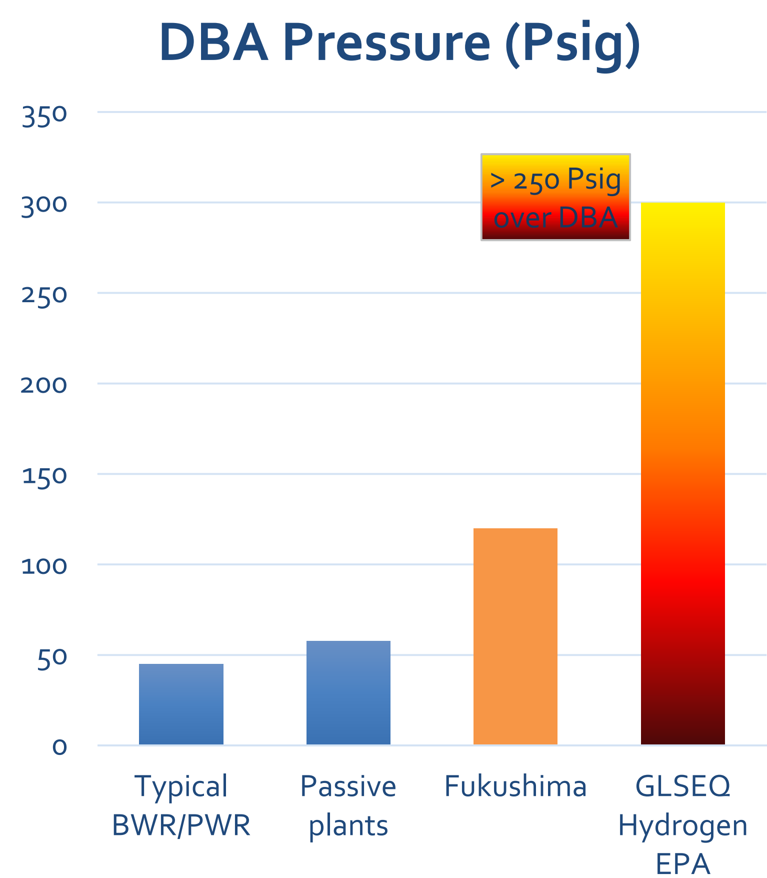
GAME CHANGER

SEVERE ACCIDENTS



Severe Accidents vs Design Basis





IEEE 497-2015

- Qualification Criteria – Clause 7.6 {PROPOSED VERSION}
- Instrument channels shall be type tested to the anticipated severe accident conditions. Type testing may be done sequentially.
- If during testing the required test conditions are not reached, due to test equipment limits, and the tested equipment does not experience a failure, then a survivability analysis shall be performed for the untested anticipated severe accident conditions.
- The survivability analysis shall determine the constraints for the reliable use of the instrument data and these constraints shall be provided to the end user.

IEEE 497-2015

- Added Type F Variables for severe accident specific instrumentation
- – Selection Criteria - Clause 4.6
- Type F variables are those variables that provide primary information to accident management personnel to indicate fuel damage and the effects of fuel damage. The selection of these variables represents a minimum set of plant variables that provide the most direct indication of the parameters needed to execute the severe accident management guidelines (SAMGs) and / or variables needed to mitigate those accidents postulated in a plants severe accident analysis.
- Selection Criteria for the Variable Type:
- • Indicate fuel Damage
- • Monitor the direct effects (e.g. combustible gasses concentration, radiation, pressure, or temperature) of fuel damage
- Potential Source Documents:
- • Plant SAMGs
- • Plant Severe Accident An
- Severe Accident Analyses

EQ CALL TO ACTIONS

NQA-1

- EQ Engineering should lead not QA

CGD

- Safety Function Performance before Failure Modes
- EQ is the most Critical Characteristic

Severe Accidents

- Much more common than LOCA
- Need SA Qualification

Bulletin 77-05: Electrical Connector Assemblies

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

November 8, 1977

IE Bulletin 77-05

ELECTRICAL CONNECTOR ASSEMBLIES

Description of Circumstances

Recent tests conducted by the Sandia Laboratories of electrical connector/cable assemblies in a simulated post-LOCA containment environment (LWR) demonstrated that the assemblies, failed to perform in an acceptable manner. The connectors are the pin and socket type, with metal shell and screw couplings. The specific test specimens were manufactured by Bendix, ITT Cannon and Gulton Industries using combinations of Anaconda and ITT Surprenant cables. Details of the specific connector/cable combinations, test conditions, test results and other pertinent information are described in the Attachment.

While electrical connectors of the type tested are not normally used in applications that are required to survive LOCA conditions, it is not possible in the absence of specific information to conclude that such applications do not exist. Further, it is unknown whether other manufacturers have supplied similar assemblies, whether such assemblies have been properly qualified for the intended service, or whether these types of assemblies are utilized in applications that must continue to operate reliably in a LOCA environment.

Action To Be Taken By Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit:

1. Determine whether your facility utilizes or plans to utilize electrical connector assemblies of the type tested by Sandia Laboratories, or any other similar type, in systems that are located inside containment, are subject to a LOCA environment and are required to be operable during a LOCA.

2. If any such applications are identified, review the adequacy of qualification testing for the assemblies and submit the documentation for NRC review.

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IE Bulletin 77-05 November 8, 1977

3. If evidence is not available to support a conclusion of adequacy, submit your plans and programs toward qualifying existing equipment or your plans for replacing unqualified assemblies with qualified equipment.

4. Provide your response in writing within 30 days for facilities with an operating license and within 60 days for facilities with a construction permit. Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Division of Reactor Construction 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.

Attachment:
Trip Report by W. R. Rutherford
Electrical Connector Assemblies

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IE Bulletin-77-05
November 8, 1977

TRIP REPORT
by
W. R. Rutherford

ELECTRICAL CONNECTOR ASSEMBLIES

On September 1, 1977 a meeting was held in Alburquerque, New Mexico to investigate the electrical connector assembly malfunctions or failures that occurred during tests under LOCA conditions performed by Sandia Laboratories. The following is a description of the equipment, test scope and results of these tests.

Equipment

The test assemblies of particular interest consisted of three types of connectors: Bendix, ITT Cannon, and Gulton installed on two types of cables; Anaconda and ITT Surprenant.

1. Bendix Connector: A 3 conductor/No. 12 AWG with crimp pin conductors, anodized aluminum shell, silicone rubber insert, rigid back plane, potting, pliable over-potting.

2. ITT Cannon Connector; A 3 conductor/No. 12 AWG with crimp pin conductors, anodized aluminum shell, silicone rubber inserts anodized aluminum back shell, rubber packing boot, mechanical retaining clamp.

3. Gulton Connector: A 3 conductor/No. 12 AWG with crimp pin conductor, stainless steel shell, hard fiber insert, pin back sealed with RTV 112, stainless shell, back plane poured with Sylgard potting, mechanical clamp termination.

4. Anaconda Cable: A 3 conductor/No. 12 AWG, tinned copper conductor, 30 mil ethylene propylene rubber insulation 15 mil Hypalon jacket, cable asbestos tape, 60 mil Hypalon Jacket, rated 600 volts, cable diameter 0.55".

5. ITT Surprenant Cable: A 3 conductor/No. 12 AWG, tinned copper conductor, 30 mil Exane II insulation, silicone glass tape, 65 mil Exane jacket, rated 600 volts, cable diameter 0.455".

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Test Scope

The three tests performed by Sandia were composed of two sequential and one simultaneous exposure to LOCA environments. In each case the equipment was exposed to radiation and thermal aging prior to operating under the simulated LOCA conditions. Figures 1 and 2 describe the test profiles for sequential and simultaneous tests respectively (Sandia tests were designed to study synergistic effects). Each of the tests satisfy the intent of IEEE 323-1974. The assemblies were electrically loaded to 20 amperes and 600 volts at the start of the tests. Insulation resistance and capacitance measurements were recorded during the tests to indicate damage.

The equipment assemblies with respect to the sequential and simultaneous tests performed were as follows:

1. Sequential Tests (Two)

Gulton Connector/ITT Cable	1 Assembly
Gulton Connector/Anaconda Cable	1 Assembly
Bendix Connector/ITT Cable	2 Assemblies
ITT Connector/ITT Cable	1 Assembly

2. Simultaneous Test (One)

ITT Connector/ITT Cable	1 Assembly
Bendix Connector/ITT Cable	1 Assembly
Bendix Connector/Anaconda Cable	2 Assemblies

Test Results

Both ITT Cannon connector assemblies and both Gulton connector assemblies showed almost immediate damage in terms of insulation resistance and capacitance as the 70 psig steam was applied.

The ITT Cannon connector assembly failures appeared to be back plane boot seal leakage failures. The assembly construction did not contain potting compound (by design) to protect the pin backs. Therefore, boot failure leads directly to connector failure.

In the case of the Gulton Assemblies, failures were attributed to both the mating surface interface and the back plane seal. The design uses a rigid insert around the mating pins and the O-ring seals are

bypassed by an alignment key slot. This design may lead to leaks due to non-uniform confinement of the O-ring which could cause arcing between pins. Neutron radiography revealed inadequate amounts of potting compound (voids) and cracking of potting compound. These conditions could account for back plane failures. Neutron radiography performed on untested connectors revealed similar conditions, i.e., voids and cracking, thus indicating an apparent quality control problem at Gulton's facility. Other problems detected were identified as:

1. The shrink tube used over the pin cable interface was split length-wise and had pulled away.

2. The potting material showed virtually no adhesion to, or sealing between, the cable jacket, insulation, and the connector shell.

3. The mechanical clamp had been secured so tightly that it cut the cable jacket.

The Bendix connector assembly was the only type to survive an entire test cycle. One Bendix/Anaconda assembly malfunctioned after about eight days into the 10 psig profile and the Bendix/ITT assembly experienced decreasing resistance and increasing capacitance through the simultaneous tests until both readings were off scale at the end of the 10 psig profile. A second Bendix/Anaconda assembly survived the simultaneous tests. During the sequential tests only Bendix and ITT Cannon assemblies were involved and both assemblies failed. The failures of these assemblies would be difficult to define as either connector or cable failures. The ITT cable exhibited a shrinking characteristic which could have provided a leak path through the sealing medium of the connector.

Page Last Reviewed/Updated Tuesday, September 01, 2015

IE Circular No. 78-08, Environmental Qualification of Safety-Related Electrical Equipment at Nuclear Power Plants

CR78008

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

MAY 31, 1978

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, ROI, IE

SUBJECT: IE CIRCULAR 78-08, ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED ELECTRICAL EQUIPMENT AT NUCLEAR POWER PLANTS

The subject document is transmitted for issuance on May 31, 1978. The Circular should be issued to all holders of Reactor Operating Licenses and Construction Permits. Also enclosed is a draft copy of the transmittal letter.

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

Enclosures:
1. IE Circular 78-08
2. Draft Transmittal Letter

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

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(Transmittal letter for Circular 78-03 to each holder of an NRC Reactor Operating License and Construction Permit.)

IE Circular 78-08

Addressee:

The enclosed Circular 78-08 is forwarded to you for information. If there are any questions related to your understanding of the suggested actions, please contact this office.

Signature
(Regional Director)

Enclosure:
1. IE Circular 78-08
2. List of IE Circulars
Issued in 1978

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

May 31, 1978

IE Circular 78-08

ENVIRONMENTAL QUALIFICATION OF SAFETY -RELATED ELECTRICAL EQUIPMENT AT NUCLEAR POWER PLANTS

The NRC staff initiated a series of actions to confirm the environmental qualification of electrical equipment required to perform a safety function under postulated accident conditions. These actions are summarized in the Commission's April 13, 1978 Order in response to a petition from the Union of Concerned Scientists. Information obtained from recent licensee equipment tests and evaluations have indicated potential problems in qualification of installed equipment. As a result, the NRC expanded these actions to include an environmental review of safety-related electrical equipment at selected older plants.1/ This review did not identify generic qualification deficiencies. However, as a result of IE Bulletins and the aforementioned testing to confirm qualification, specific deficiencies were identified. Poor installation practices, inadequate consideration of subcomponents and omission of certain environmental parameters in the design are examples of such deficiencies. In addition, the documentation of qualification was found to be inadequate in many cases and the initial response to some licensees indicated a lack of detailed knowledge of the quality of installed equipment.

The purpose of this Circular is to bring to your attention such deficiencies and to highlight the important lessons learned. In its April 13, 1978 Order, the Commission indicated that

> "In order to fulfill its regulatory obligations, NRC is dependent upon all of its licensees for accurate and timely information. Since licensees are directly in control of plant design, construction, operation and maintenance, they are the first line of defense to ensure the safety of the public. NRC's role is one primarily of review and audit of licensee activities, recognizing that limited resources preclude 100 percent inspection.
>
> Furthermore, the Commission notes that some of the licensee's initial responses indicate a lack on their part of detailed knowledge of the quality of installed plant equipment. Licensees must have this detailed understanding of their own plants in order to meet their obligations

for public,safety by ensuring a

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sound basis for making assessments of plant safety. The NRC establishes general safety criteria, sets specific requirements for many aspects of reactor design and operation, and ensures compliance with these criteria and requirements by independent audit. While, in the Commission's view, these activities play a vital role in ensuring safe plant operation, they are not a substitute for licensee safety reviews. The licensees must be knowledgeable and vigilant and must take more initiative in ferreting out details of potential plant weaknesses."

As part of this obligation, you should examine installed safety-related electrical equipment, and ensure appropriate documentation of its qualification to function under postulated accident conditions. Specific guidance on the subject of environmental qualification can be found in IEEE 323-1971 and 1974, as augmented by Regulatory Guide 1.89.

Examples of specific deficiencies identified in information provided by licensees are as follows:

1. Connectors: Responses to IE Bulletins 77-05 and 77-05A revealed in certain instances a lack of qualification data for environmental parameters and inadequate design of connectors for postulated accident conditions.2/

2. Penetrations: A failed penetration prompted issuance of IE Bulletin 77-06. Responses to this bulletin showed adequate documentation for the qualification of the penetration assembly was not readily available in some cases.2/ In one instance, the electrical connections of the penetrations were not qualified ih conjunction with the penetration assembly,3/ which demonstrates a lack of consideration for qualification of interfacing components.

3. Terminal blocks: Because of unprotected terminal blocks in penetration areas inside containment of Haddam Neck, Bulletin 78-02 was issued. These unprotected blocks were replaced with blocks designed to function in the LOCA and main steam line break environments.4/ Responses to the Bulletin revealed two other facilities, Yankee Rowe and Ginna, with such unprotected blocks.5/,6/ Other terminal blocks were found to be inadequately qualified due to poor design or installation practices, even though they were in enclosures.7/,8/,9/ and 10/

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4. Limit switches: While examining the documentation for the qualification of all safety-related equipment installed inside containment, a vendor identified limit switches mounted on otherwise qualified valves at certain facilities. Preliminary review by the staff of responses to IE Bulletin 78-04 indicates such switches are installed in similar applications at other facilities. Corrective action is presently in progress.

5. Cable splices: Electrical cable splices associated with electrical penetration assemblies were determined to be unqualified by licensees during their search for qualification documentation.11/

6. Other potential problems for specific components currently under staff review include:

   -radiation and temperature effects on electrical cables10/

   -adequacy of qualification testing of components by separate effects versus sequential testing of environmental parameter10/

   -temperature limitations on nylon components of solenoid valves12/

   -qualification of electrical transmitters 13/, 14/

The review of these issues may result in the need for other followup or corrective actions.

No written response to this Circular itself is required. Each licensee should determine the applicability of the qualification items identified above for its facility. Appropriate corrective action should be taken for any problem identified by the licensee as a result of its review. NRC inspectors will review these matters with licensees in future inspections. If further information is required, contact the Director of the appropriate Regional Office.

3 of 3

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REFERENCES

1. "Short Term Safety Assessment on the Environmental Qualifications of Safety-Related Electrical Equipment of SEP Operating Reactors," May 1978, enclosure to staff memorandum to Commission, dated May 12, 1978 and issued as NUREG Report 0458.

2. "NRC Staff Report on Union of Concerned Scientists' Petition for Emergency and Remedial Action," December 15, 1977, enclosure to staff memorandum to Commission, dated December 15, 1977.

3. Letter from Consumers Power Company to NRC dated April 6, 1978, including, "Summary of Qualifications of Electrical Penetration Assembly Connectors for the Palisades Plant," Docket No. 50-255.

4. NRC Summary of January 29, 1978 meeting on "Environmental Qualification of Terminal Blocks and Replacement of Terminal Blocks, Haddam Neck Plant," Docket No. 50-213, dated January 30, 1978.

5. NRC Summary of February 1, 1978 meeting, Yankee Rowe Nuclear Power Station (terminal blocks), Docket No. 50-29, dated February 3, 1978.

6. NRC Summary of February 1, 1978 meeting on "Environmental Qualification of Terminal Blocks and Replacement of Terminal Blocks," R. E. Ginna Nuclear Plant, Docket No. 50-244 dated February 2, 1978.

7. Letter from Connecticut Yankee Atomic Power Company to NRC, dated March 29, 1978, including "Haddam Neck Plant Summary of Environmental Qualification Test Program, Terminal Block/Box Combinations," Docket No. 50-213.

8. Letter from Consumers Power Company to NRC, dated April 12, 1978, including information on terminal blocks at Palisades, Docket No. 50-255.

9. Letter from Indiana & Michigan Power Company to NRC, dated March 22, 1978 regarding terminal blocks at D. C. Cook Unit No. 2, Docket No. 50-316.

10. Letter from Indiana & Michigan Power Company to NRC, dated April 21, 1978, regarding terminations at D. C. Cook Unit Nos. 1 and 2, Docket Nos. 50-315 and 50-316.

11. Staff memorandum, "Status of Monticello Electrical Splice Upgrade," dated May 10, 1978, Docket No. 50-263.

12. Letter from Consumers Power Company to NRC, "'Environmental Qualification for Big Rock Point," dated February 24, 1978, Docket No. 50-155.

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13. Letter from Consumers Power Company to NRC, "Environmental Qualification for Palisades," dated February 24, 1978, Docket No. 50-255.

14. Letter from Westinghouse to E. G. Case, dated April 26, 1978, regarding environmental qualification status for D. C. Cook Unit 2, Docket No. 50-316.

Page Last Reviewed/Updated Tuesday, September 01, 2015

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

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January 14, 1980

IE Bulletin No. 79-01B

ENVIRONMENTAL QUALIFICATION OF CLASS IE EQUIPMENT

Description 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 equipme': 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 reporting 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 a, that are required to function under 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 identified in Item 1, provide written evidence of its environmental qualification 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 within 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 appropriate 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.

IE Bulletin No. 79-01B
January 14, 1980

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 Time-pieces 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

Facility: XYZ
Docket No.: 50-XXX

Enclosure 2

MASTER LIST
(Typical)

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

I. SYSTEM: RESIDUAL HEAT REMOVAL (RHR)

COMPONENTS			
		Location	
Plant Identification Number	Generic Name	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			
		Location	
Plant Identification Number	Generic Name	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	Location: 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 K369 (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 EVAL· ·N WORK SHEET
(Typic

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 Function: Accident Monitoring	Relative Humidity(%)	100%	100%	1	5	Simultaneous Test	None
Accuracy: Spec: 5% Demon: 4%	Chemical Spray	N_3BO_3/ NAOH		1			See Note 1
Service: RHR Pump 1A Discharge Pressure S/N107	Radiation	4×10^6 rads	1.2×10^8 rads	2	6	Sequential Test	None
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 14.2.3.1
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.

Enclosure 4

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

[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.

<u>Gamma Radiation Doses</u> - 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.

<u>Beta Radiation Doses</u> - 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

[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 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 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 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 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.^ 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 d· 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 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 old-r and newer plants. In each case the system features which allow f-· 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:

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

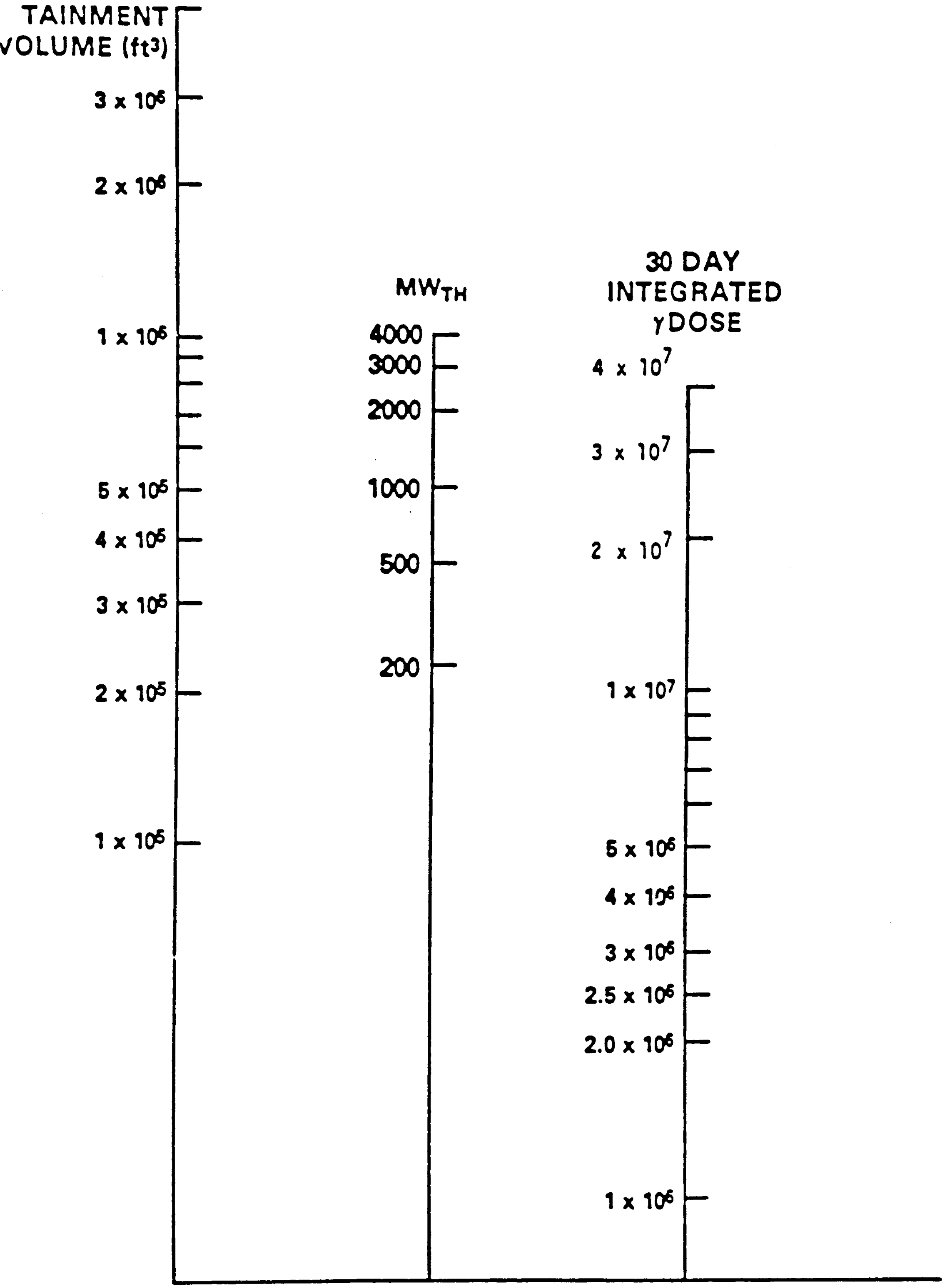
Step 4

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)

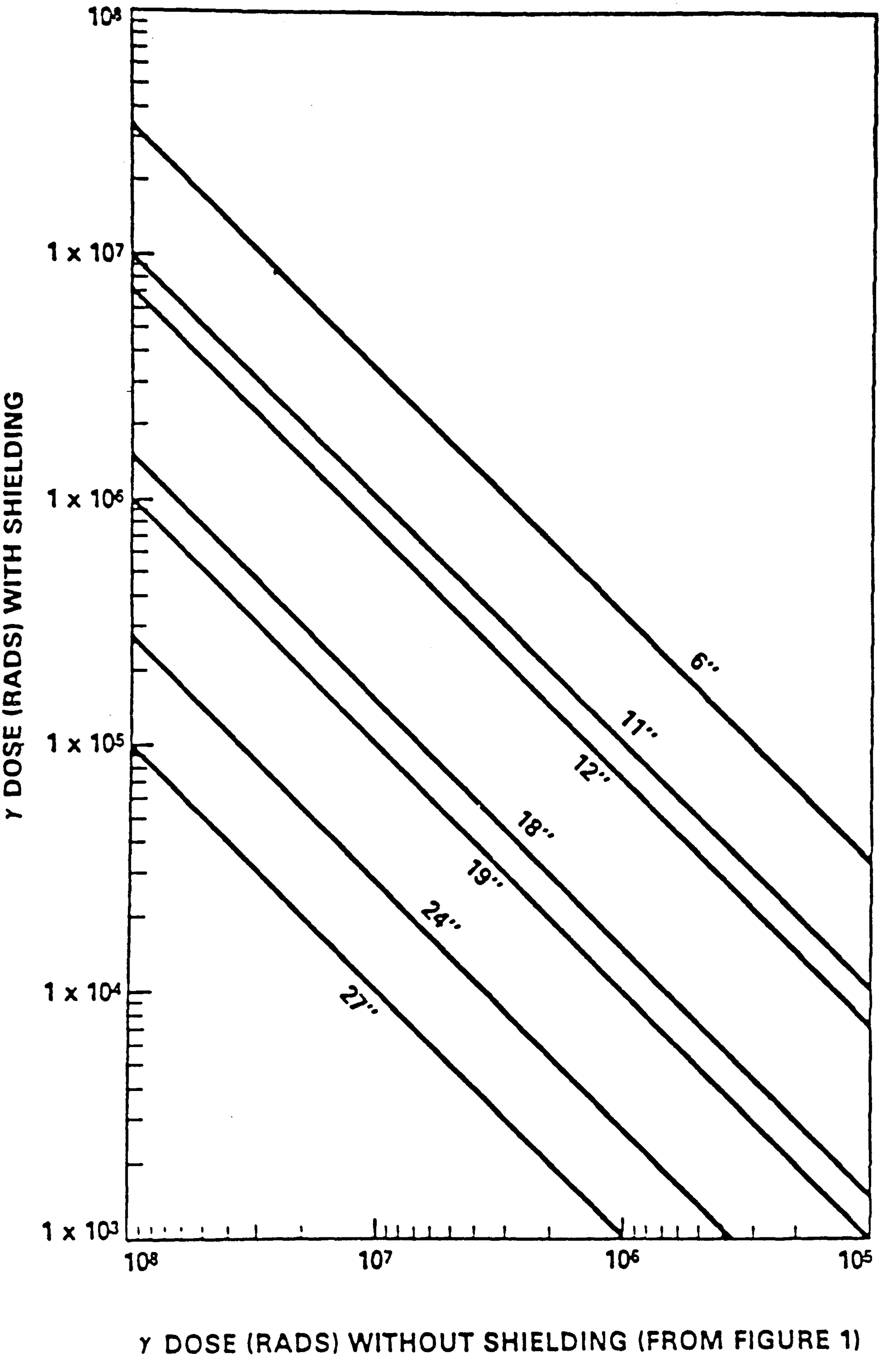


FIGURE 3
DOSE CORRECTION FACTOR FOR COMPARTMENT VOLUME

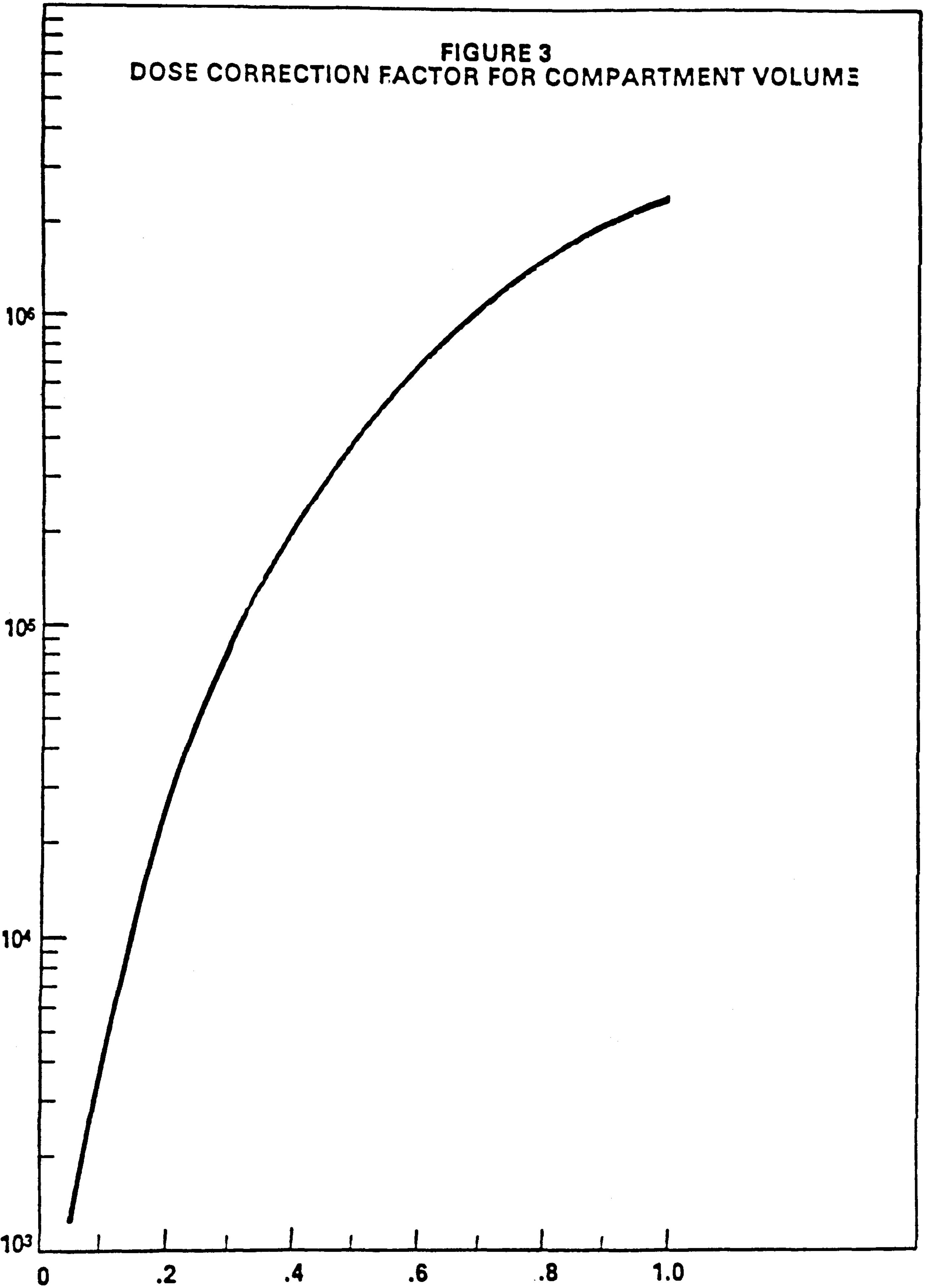
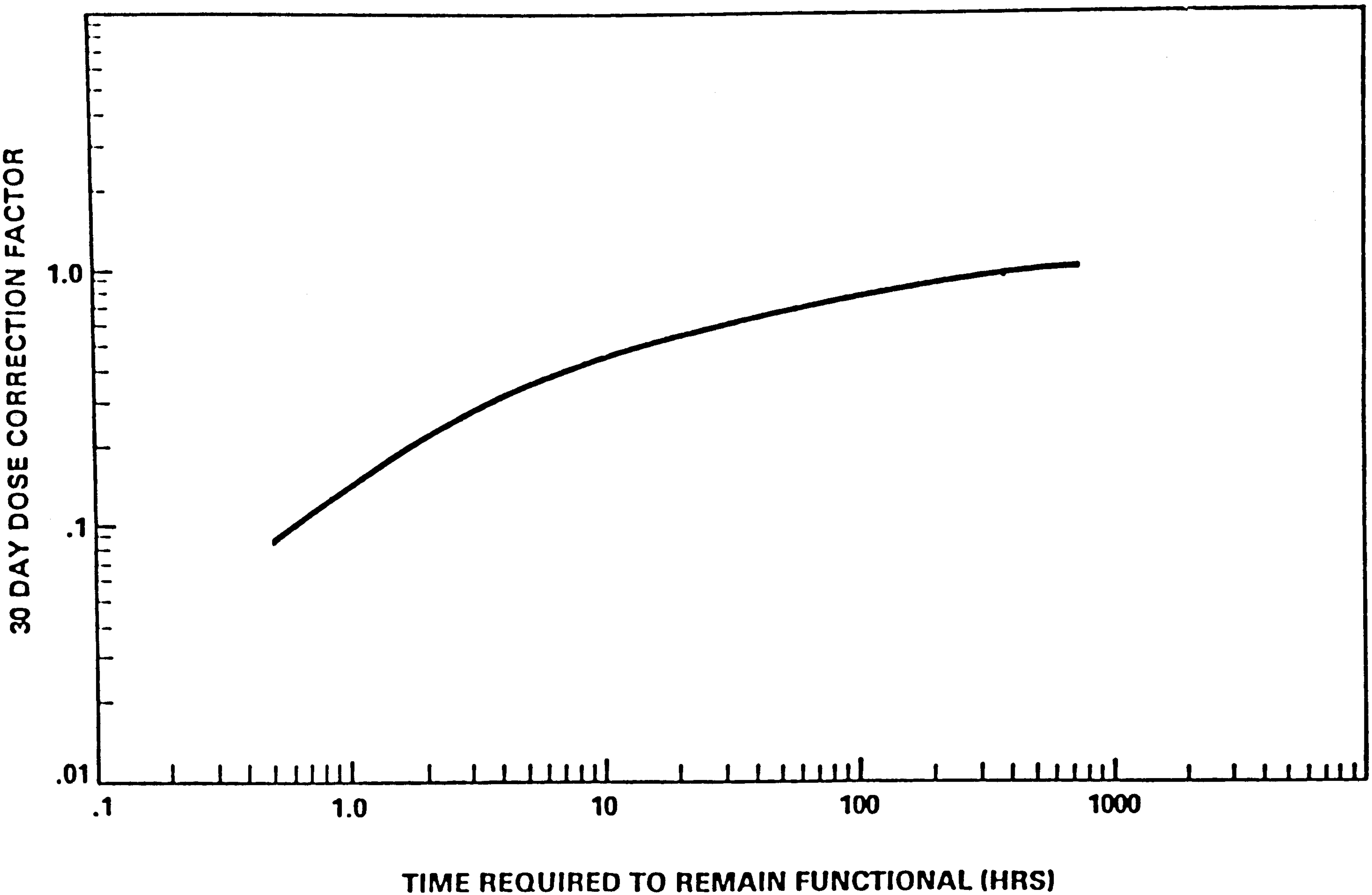


FIGURE 4

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

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	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 State Equipment	Motor 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 RACKS & PANELS	LIMIT SWITCHES	MOTORS	SENSORS	SPLICES	TERMINAL BLOCKS	VALVE OPERATORS	CONTROL BOARDS	DIESEL GENERATOR CONTROL EQUIPMENT	FANS	[illegible] EQUIPMENT	MTR. CONTROL 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 Circu[illegible] (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	

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	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 XXXC				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 CX				10^9	"																	X
NEMA Thermosetting Laminates, Grade C				10^9	"																	X

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 GENERATOR CONTROL EQUIPMENT	FANS	HVAC EQUIPMENT	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, Grade 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	

TYPES OF EQUIPMENT (WITHIN WHICH MATERIAL MAY BE FOUND)