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Status Report on Equipment Qualification Issues Research and Resolution

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STATUS REPORT
ON
EQUIPMENT QUALIFICATION ISSUES RESEARCH AND RESOLUTION

L. L. Bonzon, F. J. Wyant, L. D. Bustard, and K. T. Gillen

September 1986

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Albuquerque, New Mexico 87185
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Abstract

Since its inception in 1975, the Qualification Testing Evaluation (QTE) Program has produced numerous results pertinent to equipment qualification issues. Many have been incorporated into Regulatory Guides, Rules, and industry practices and standards.

This report summarizes the numerous reports and findings to date. Thirty separate issues are discussed encompassing three generic areas: accident simulation methods, aging simulation methods, and special topics related to equipment qualification. Each issue-specific section contains (1) a brief description of the issue, (2) a summary of the applicable research effort, and (3) a summary of the findings to date.

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Foreword

This report comes about as the result of one order task added to the FY85 QTE Program Brief by the NRC Office of Nuclear Regulatory Research (RES):

Task 20. Provide a report on the status of all SNL and related research to date to resolve outstanding equipment qualification issues in the NRC program plan. Identify and discuss how the research has resolved the issues. Describe how ongoing research will specifically provide for additional issue resolution. Have a peer group of non-SNL experts review and comment on the report. A copy of the comments should be given to the NRC at the same time as SNL receives them. Submit a plan and schedule by January 1985 for accomplishing this task. The January plan submittal should include an outline of the report sections and an abstract of the conclusions for each section. A final draft of the report is desired no later than June 1985.

In early February 1985, NRC and SNL staff agreed to proceed on this major reporting task. Full draft preparation was begun March 1 by seven SNL principal staff. The result was a first full draft, of 850 pages, Federal Expressed to NRC staff for review on March 29. Peer Group members were selected and contracts for expert services completed. The members are:

J. Tanaka, Peer Group Chairman, The University of Connecticut
S. Carfagno, Peer Group Member, Franklin Research Center
J. Gleason, Peer Group Member, Wyle Laboratories
J. Rhoads, Peer Group Member, Washington Public Power Supply System

Based on NRC and SNL staff comments, a first revision was prepared in early April. This version was Federal Expressed to NRC staff and to the Peer Group members on April 15.

The (first) Peer Group meeting was held at SNL during May 13-17 with all members and W. Farmer (NRC) in attendance. Numerous sections were reviewed by the Group and revisions suggested. A full (second) revision was completed in late May based on the Group comments; this version was provided ahead of a planned June 2-3 Peer Group meeting in Baltimore.

Following the (second) Peer Group meeting, additional comments were received from the Group and from W. Farmer. These were incorporated into a "final draft report" in mid-June. This "final draft report" was sent for NRC staff review on June 24 under cover letter to W. Farmer; this review period continued through October.

The final Peer Group meeting on the report was held in Albuquerque on November 7. All technical comments have been resolved.

On January 28, 1986, and again on May 29, 1986, W. Farmer visited SNL to discuss the plans for report completion. He provided several example sections that conformed to NRC staff inputs.

Acknowledgement

This document was authored by the following Sandia National Laboratories staff:

Section	Name
Generic Issues Related to Methods for Simulating Accident Conditions	
1.1 Simultaneous/Sequential Exposures	L. Bustard
1.2 Superheated/Saturated Steam Effects	F. Wyant
1.3 Thermal Shock and Steam Impingement	F. Wyant
1.4 Dose-Rate Effects	F. Wyant
1.5 Beta/Gamma Radiation Effects	F. Wyant/L. Bonzon
1.6 Oxygen Effects	L. Bustard
1.7 Chemical Spray Effects	F. Wyant
1.8 Acceleration of Postaccident Environments	P. Bennett
1.9 Sensitivity of Accident Simulations to Aging Methods	L. Bustard
1.10 Hydrogen Burn Influence on Accident Simulation Methods	E. Richards/L. Bonzon
1.11 Submergence Simulation	E. Richards
Generic Issues Related to Methods for Simulating Aging Conditions	
2.1 Realistic Ambient Environments	F. Wyant/L. Bonzon
2.2 Limitations of the Arrhenius Method	K. Gillen
2.3 Dose-Rate Effects	K. Gillen/R. Clough
2.4 Simultaneous/Sequential Exposures	K. Gillen/L. Bustard
2.5 Mechanical Stress Effects	K. Gillen
2.6 Oxygen Effects	K. Gillen/R. Clough
2.7 Humidity Effects	K. Gillen
2.8 Analytical and Experimental Techniques for Correlation Between Natural and Artificial Aging	R. Clough/K. Gillen
2.9 Comparison of Artificially- and Naturally-Aged Equipment	L. Bustard/L. Bonzon
Special Topics Related to Equipment Qualification	
3.1 Complementary Test Approaches and Considerations	P. Bennett
3.2 TMI-2 Experiences	P. Bennett/ L. Bonzon
3.3 Evaluation of Qualification Procedures for Specific Equipment Types	F. Wyant/L. Bustard
3.4 Regulatory Guide 1.97 Requirements	E. Richards/L. Bonzon
3.5 Advanced Systems Qualification Issues	L. Bonzon
3.6 Realistic Accident Environments and Calculational Models	F. Wyant/L. Bonzon/K. Gillen
3.7 Criteria for Selecting Simulation Methods	L. Bustard/L. Bonzon
3.8 Review of Standards and Guides	L. Bonzon
3.9 Battery Aging Methods	L. Bonzon
3.10 Radiation Damage Thresholds	L. Bonzon

The authors wish to acknowledge and thank O. Steutzer for his contributions to Sections 1.3, 1.7, and 2.5 and D. Hente for his contributions to Sections 2.9 and 3.9.

The authors also wish to acknowledge and thank Carol Schmidt and Della Vigil for their work in typing and correcting this manuscript many times.

Executive Summary

Since its inception in 1975, the Qualification Testing Evaluation (QTE) Program has produced numerous results pertinent to equipment qualification issues. Many have been incorporated into Regulatory Guides, Rules, and industry practices and standards.

The impetus for this report resides in the need to summarize the numerous findings to date. This was also recognized by NRR staff in their comments to the NRC's "Long-Range Research Plan for FY85-FY89," NUREG-1080:

> "...summary reports should be prepared at appropriate times on the research elements of the long-range research plan (LRRP). We recommend that summary reports on equipment qualification...be prepared in 1985...
>
> "Close to one hundred reports and papers have been published from the equipment qualification studies. A summary report should be issued which summarizes and documents what has been learned as of 1985, discusses the regulatory implications of the results, and identifies research needed to resolve any remaining questions or problems."

As a result, RES/DET staff added a specific task to the FY85 Program Brief:

> "TASK 20 Provide a report on the status of all SNL and related research to date to resolve outstanding equipment qualification issues in the NRC program plan. Identify and discuss how the research has resolved the issues. Describe how ongoing research will <u>specifically</u> provide for additional issue resolution. Have a peer group of non-SNL experts review and comment on the report..."

The primary issues involved in meeting the overall QTE Program objectives can be grouped under three major headings:

1. Generic issues related to methods for simulating accident conditions.

2. Generic issues related to methods for simulating aging conditions.

3. Special topics related to equipment qualification.

Each of these groupings contains many specific issues:

1. Generic Issues Related to Methods for Simulating Accident Conditions

1.1 Simultaneous/Sequential Exposures
1.2 Superheated/Saturated Steam Effects
1.3 Thermal Shock and Steam Impingement
1.4 Dose-Rate Effects
1.5 Beta/Gamma Radiation Effects
1.6 Oxygen Effects
1.7 Chemical Spray Effects
1.8 Acceleration of Postaccident Environments
1.9 Sensitivity of Accident Simulations to Aging Methods
1.10 Hydrogen Burn Influence on Accident Simulation Methods
1.11 Submergence Simulation

2. Generic Issues Related to Methods for Simulating Aging Conditions

2.1 Realistic Ambient Environments
2.2 Limitations of the Arrhenius Method
2.3 Dose-Rate Effects
2.4 Simultaneous/Sequential Exposures
2.5 Mechanical Stress Effects
2.6 Oxygen Effects
2.7 Humidity Effects
2.8 Analytical and Experimental Techniques for Correlation Between Natural and Artificial Aging
2.9 Comparison of Artificially- and Naturally-Aged Equipment

3. Special Topics Related to Equipment Qualification

3.1 Complementary Test Approaches and Considerations
3.2 TMI-2 Experiences
3.3 Evaluation of Qualification Procedures for Specific Equipment Types
3.4 Regulatory Guide 1.97 Requirements
3.5 Advanced Systems Qualification Issues
3.6 Realistic Accident Environments and Calculational Models
3.7 Criteria for Selecting Simulation Methods
3.8 Review of Standards and Guides
3.9 Battery Aging Methods
3.10 Radiation Damage Thresholds

Each of the thirty issues is discussed (to the extent practicable) on a stand-alone basis. The reader can then, in one place, obtain all relevant research summaries, key findings, and references. There is a uniform style within each issue presentation:

- First, a brief issue description is presented. It both summarizes salient points covered in previous sections, and highlights specific technical points applicable to the issue.

- Second, the applicable research conducted to date is summarized. The primary reports are discussed and summarized; applicable charts and figures are also provided for clarification. In some cases, subissues are presented within the overall issue context.

- Third, to the extent known, other U.S. and international programs relevant to the issue are addressed; again this is done through a discussion of specific reports.

- Fourth, the key results and conclusions are discussed, this is presented in terms of issue findings.

- Fifth, the list of primary (those that are numbered) report and secondary (those that are lettered) report references are given, including references to other U.S. and international work.

The report documents the QTE Program achievements to date. The QTE Program is a comprehensive and integrated effort and has, and is, addressing the pertinent equipment qualification issues. It also has as a goal the identification and investigation of any relevant new issues. It incorporates industry and NRC experiences and the results of the QTE Program research itself, and monitors the ever-changing standards and regulations.

Results of the Program have been incorporated into Regulatory Guide 1.89 (Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants), 10 CFR 50.49 (Environmental Qualification of Electric Equipment for Nuclear Power Plants), IEEE Std. 323 (IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations), and other relevant national and international Standards and Guides. Research is continuing in the areas of determining acceptable methods of simulating accident environments, of identifying appropriate means of accelerating the aging of equipment for qualification testing, and of investigating the behavior of various materials and components used in safety-related equipment.

We fully expect that the QTE Program will continue to address and meet its objectives:

(1) to obtain data needed for the confirmation of the suitability of current Standards and Regulatory Guides for Class 1 equipment;

(2) to obtain data that will provide improved technical bases for modifications of these Standards and Guides where appropriate;

(3) to establish data-based and standardized test methodologies for equipment qualification programs; and

(4) to support the NRC licensing process with qualification expertise and test capabilities.

I. Introduction

The purpose of the Qualification Testing Evaluation (QTE) Research Program (FIN #A-1051) is to provide technical information and expertise to support NRC's responsibilities for the creation, interpretation and revision of Regulatory Guides and Standards pertaining to equipment qualification and survivability. In addition, this program supports NRC's overall Long-Range Research Plan (LRRP) in its endeavor to assess the "test results reported by applicants and licensees and provide a basis for licensing decisions that ensure protection of the public health and safety." NRC staff must be able to judge whether the results presented are adequate, complete, and technically sound. Specifically then, the objectives of the Qualification Testing Evaluation Program are: (1) to obtain data needed for the confirmation of the suitability of current Standards and Regulatory Guides for Class 1 equipment; (2) to obtain data that will provide improved technical bases for modifications of these Standards and Guides where appropriate; (3) to establish data-based and standardized test methodologies for equipment qualification programs; and (4) to support the NRC licensing process with qualification expertise and test capabilities.

The Qualification Testing Evaluation Program is structured to study the methods for qualifying safety-related electrical equipment to demonstrate the equipment's ability to function both during and following design basis accidents which produce harsh environments, including high radiation, temperature, pressure, humidity, hydrogen burn, and seismic events.

Since its inception in 1975, when the Program was initiated to address a specific question on accident synergism effects, literally hundreds of reports have been produced. The scope of the program has also been broadened to encompass three generic qualification concerns:

- accident simulation methods
- aging simulation methods
- special topics related to equipment qualification

These concerns are being addressed through seven major program tasks:

- provide assessments of accident (environmental) testing methods
- evaluate the adequacy of radiation simulators
- evaluate the aging processes of safety-related equipment and establish accelerated-aging methods
- evaluate the experiences from the TMI-2 accident

- pursue information from international research efforts

- define long-term research and facility needs

- evaluate aging/seismic correlation on selected equipment

Because the Program has produced numerous, and significant research findings, and given the current status of the equipment qualification techniques and effort, it is appropriate to assess and summarize the past results and conclusions. This provides a thorough review of the state-of-the-art, an identification of closed issues, and a focusing of the future efforts on the remaining necessary research.

That then is the goal of this report, to assess the status of Sandia (and related) research on equipment qualification issues, to discuss the relationship between that research and issues resolution, and to describe the research and need for future issues' resolution.

This report is <u>not</u>, however, intended to supplant the referenced documents, rather it should be utilized as a <u>guide</u> to the issues important in equipment qualification and to references that provide further information regarding those issues.

II. Report Purpose and Objectives

The impetus for this report resides in the need to summarize the numerous findings to date from the Qualification Testing Evaluation (QTE) Program. This was also recognized by NRR staff in their comments (Reference II.1) to the NRC's "Long-Range Research Plan for FY85-FY89," NUREG-1080:

> "...summary reports should be prepared at appropriate times on the research elements of the long-range research plan (LRRP). We recommend that summary reports on equipment qualification...be prepared in 1985...
>
> "Close to one hundred reports and papers have been published from the equipment qualification studies. A summary report should be issued which summarizes and documents what has been learned as of 1985, discusses the regulatory implications of the results, and identifies research needed to resolve any remaining questions or problems."

As a result, RES/DET staff added a specific task to the FY85 Program Brief:

> "TASK 20 Provide a report on the status of all SNL and related research to date to resolve outstanding equipment qualification issues in the NRC program plan. Identify and discuss how the research has resolved the issues. Describe how ongoing research will <u>specifically</u> provide for additional issue resolution. Have a peer group of non-SNL experts review and comment on the report..."

Beyond the summarization aspect, this report will be the vehicle for improved technology transfer to NRC and industry staff by providing an assessment of the current status of equipment qualification research. Future documents will provide the specific (recommended) methodologies to be employed for testing of equipment under accident and/or aging conditions. Those applications-oriented "handbooks" may also be published on an issue-by-issue basis. The beginnings of this type of tie between research and application may be found in some of the issues presented here (Issue 3.3, for example, discusses accident tests performed on specific equipment in terms of "how-it-was-done"). The three fundamental objectives of <u>this</u> report, then, are:

1. Briefly discuss the issue.

2. Summarize the research effort to date.

3. Summarize the issue-specific findings.

<u>References</u>

II.1 Letter, H. R. Denton (NRR) to R. B. Minogue (RES), dated April 3, 1984, "Long-Range Research Plan for FY85-FY89," NUREG-1080.

III. Equipment Qualification Process

The legal bases for the environmental qualification of electric equipment is 10 CFR 50.49 (Reference III.1). Regulatory Guide (RG) 1.89 (Reference III.2) describes a method acceptable to the NRC staff for complying with 10 CFR 50.49.

The Regulatory position is that the procedures described by IEEE Std. 323-1974 (Reference III.3) are acceptable to the NRC staff for satisfying the Commission's regulations pertaining to the qualification of electric equipment subject to the provisions of Part C of Regulatory Guide 1.89.

With regard to earlier equipment qualification programs completed by the nuclear power industry, Part D of Regulatory Guide 1.89 states that:

> "In accordance with paragraph 50.49(k), applicants for and holders of operating licenses are not required to requalify electric equipment important to safety (replacement equipment excepted) in accordance with the provision of Section 50.49 and in accordance with this guide if the NRC has previously required qualification of that equipment in accordance with "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines), or NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical equipment." These applicants and licensees may continue to use the criteria in these documents for qualifying electric equipment important to safety in the affected plants, with the exception of replacement equipment."

These three documents (and various IEEE daughter Standards of IEEE 323-1974) also serve as the principal bases for the issues being addressed in the QTE Program. The significance and impact of on-going research is specifically recognized in Section B of Regulatory Guide 1.89:

> "...Paragraph 50.49(e)(5) calls for equipment qualified by test to be preconditioned by natural or artificial (accelerated) aging to its end-of-installed-life condition and further specifies that consideration must be given to all significant types of degradation that can have an effect on the functional capability of the equipment. There are large uncertainties concerning the processes and environmental factors that could result in such degradation. Oxygen diffusion, humidity, and accumulation of deposits are examples of such effects. Because of these uncertainties, state-of-the-art preconditioning techniques are not capable of simulating all significant types of degradation, and natural preaging is difficult and costly. As the state of the art advances and uncertainties are resolved, preconditioning (i.e., aging) techniques may become more definitive. Experience suggests that consideration should be given, for example to a combination of (1) preconditioning of test samples employing the Arrhenius theory and (2) surveillance, testing, and maintenance of selected equipment specifically directed toward detecting those degradation processes that, based on experience, are

not amenable to preconditioning and that could result in common-cause functional failure of the equipment during design basis accidents...

"The regulatory positions delineated in this guide reflect the state of the art. Research programs currently in progress are investigating such concerns as the effects of oxygen in a LOCA environment, the validity of sequential versus simultaneous applications of steam and radiation environments, and fission product releases following accidents. The staff recognizes that the results of research programs may lead to revisions of the regulatory positions."

10 CFR 50.49(a) specifically requires that the electric equipment qualification program must include and be based on the following:

"(1) Temperature and Pressure. The time-dependent temperature and pressure...

"(2) Humidity. Humidity during design basis accidents...

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

"(4) Radiation. The radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident...including dose-rate effects.

"(5) Aging. Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed-life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment...

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

"(7) Synergistic Effects. Synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance.

"(8) Margins. Margins must be applied to account for unquantified uncertainty..."

Each of these areas have uncertainty in terms of the test methods (as discussed above, from Section B of Regulatory Guide 1.89), and research is necessary to provide a sound technical basis for the equipment qualification process.

Industry practices and standards are also sources of methodology issues. A few examples from selected IEEE Standards will serve to illustrate this point.

IEEE 323-1974 (Reference III.3):

> "6.3.2 Test Sequence. The type tests shall be run on the equipment in a specified order. For most equipment and applications, the following constitutes the most severe sequence; however, the sequence used shall be justified as the most severe for the item being tested...
>
> "6.3.4 Radiation...Radiation shall be applied as part of the sequence of environments representative of service conditions. The equipment shall be subjected to the significant type of radiation equivalent to that expected in service. However, if more than one type of radiation is significant, each type can be applied separately...
>
> "6.3.4 Radiation...In determining the total required test radiation equivalent to that of service life, consideration shall be given to oxidation gas-diffusion effects. To facilitate the use of a reasonable test time, an accelerated exposure rate may be necessary. Thus, to allow margin for these effects, a greater total dose than the service lifetime dose should be applied.

IEEE 334-1974 (Reference III.4):

> "6.2 Materials and Component Qualification. Combining radiation with the heat, steam, and chemical ambients in tests of large units is desirable but may not be feasible. In lieu of this, the insulation systems and materials of all the essential components of the motor shall be qualified by testing representative motorettes and individual components using the test methods described herein...
>
> "6.2.1 Combined Environmental Effects. These tests should determine the effect of each significant environmental phenomenon, each significant combination, and each significant sequence of these. Tests should include, when required, (1) temperature, (2) radiation, (3) steam and chemicals, (4) temperature and chemicals, (5) temperature, chemicals, steam, and radiation, and (6) thermal and steam pressure transients..."

IEEE 383-1974 (Reference III.5):

> "2.3.2 Long-Term Physical Aging Properties. Aging data should be submitted to establish long-term performance of the insulation. Data may be evaluated using the Arrhenius technique. A minimum of 3 data points, including 136°C and two or more others at least 10°C apart in temperature, should be used...

"2.3.3.3 The specimens...should be subjected in air to gamma radiation from a source such as ^{60}Co to a dosage of 5×10^7 rd at a rate not greater than 1×10^6 rd per hour...

"2.4.2 Radiation Exposure - Total. Exposure (sic) specimens to the maximum total cumulative radiation dosage expected over the installed life...plus one LOCA exposure to radiation for the particular installation involved as covered in IEEE Std. 323-1974 Appendix A or B. The rate of exposure shall not be greater than 1×10^6 rd per hour. This restriction is removed when simulation of the LOCA profile requires a greater dose rate.

"2.4.3 LOCA Simulation. Test irradiated specimens in a pressure vessel so constructed that the specimens can be operated under rated voltage and load while simultaneously exposed to the pressure, temperature, humidity and chemical spray of a LOCA event."

Finally, one additional point, regarding the need for data-based methods, should be made.

In discussing the qualification process in an issues format, it is easy to lose sight of the purpose of the qualification itself. Qualification is only shown by the resultant sum of its individual components. Each must contribute properly to the process; but each, individually, is not an end unto itself. Thus it is not strictly appropriate to speak of aging qualification, or radiation qualification, or accident qualification. Yet each must be correctly, and properly, conducted to assure "qualification".

It may be true that one event in the sequence dominates the overall qualification process. For example, for most equipment it seems plausible that there is not a strong aging-seismic correlation. On the other hand, the sequence (and/or dose-rate) of the applied thermal-radiation aging cycle is an important parameter. Thus, the only appropriate approach is to concentrate on developing a data-based issues resolution.

With that in mind for example, a proper, data-based, aging method will provide an aged equipment item that can be subjected to its appropriate accident scenerio, be it fire, or seismic, or environmental, or hydrogen burns. And it is not a concern if some equipment, under some accident environments, are not sensitive to the aging method (or even any aging at all). But to know that, means that the data-base had had to be collected somewhere, sometime. And the argument that proper, data-based, methodologies are absolutely essential is only further strengthened.

In summary then, the QTE Program's investigations are founded on the issues requiring data bases delineated (primarily) in 10 CFR 50.49, Regulatory Guide 1.89, and the IEEE Standards.

References

III.1 Code of Federal Regulations, 10 CFR 50.49, Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants, January 17, 1983.

III.2 USNRC Regulatory Guide 1.89, Environmental Qualification of Certain Equipment Important to Safety for Nuclear Power Plants, Revision 1, June 1984.

III.3 IEEE Std. 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, IEEE, New York, New York.

III.4 ANSI/IEEE Std. 334-1974, IEEE Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations, November 24, 1976.

III.5 ANSI/IEEE Std. 383-1974, IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations, April 30, 1975.

IV. Issues Identification

Program Integration

The Qualification Testing Evaluation (QTE) Program has evolved as a comprehensive and integrated effort to address the myriad of issues in equipment qualification. Because it was the first such program, it was forced to envelope the entire qualification process. Hence its efforts are directly related and adaptable to later industry and government programs (e.g., accelerated-aging techniques, naturally-aged equipment evaluation and benchmarking, radiation source term and simulator adequacy, monitoring and surveillance techniques, fragility testing concepts, and the like).

The QTE Program has incorporated industry and NRC experiences. These experiences, the results of the QTE Program research itself, and the ever-changing standards and regulations are the primary bases for the QTE Program as it is now structured. These same forces will assure that the Program also remains viable, with sufficient flexibility to address "new" issues as they are identified.

Unique Test Facilities

An important aspect of this flexibility is the dedicated and unique testing facilities developed under this Program. They are adaptable to the current, and are expected to satisfy any future, needs in equipment qualification research. There are four such major facilities:

- LICA, Low-Intensity Cobalt Array
- HIACA, High-Intensity Adjustable Cobalt Array
- I-Steam
- V-Steam, with transient superheat capability

LICA: The low-intensity cobalt array, shown schematically in Figure IV.1, is designed to provide long-term irradiation, temperature, overgas, and combined environments exposure capabilities for the assessment of accelerated-aging methods on (small-volume) materials and components. It consists of an L-shaped water-filled pool, 18 feet deep, with a water surface area of 240 feet^2. Located on short racks underwater are three Cobalt-60 sources in fixtures which also serve as spacers for specifically locating experimental cans, Figure IV.2 Its principal features are summarized in Table VI.1.

HIACA: The high-intensity adjustable cobalt array is designed for the simultaneous application of radiation (with all other in-containment accident environments), Figure IV.3. HIACA itself contains about 300 kilocuries of Cobalt-60, in 48 pencils about 2 feet long and 5/8 inch in diameter. These in turn are located in a pneumatic-telescoping (32) tube arrangement by which the number of pencils (four at a time for symmetry)

can be positioned to surround the steam test chamber and/or test item. By selecting the number of pencils surrounding the test chamber, the dose rate can be selected for the experiment. (Or, for example, it can be changed during the experiment to observe dose-rate effects.) Its principal features are summarized in Table IV.2.

Table IV.1

LICA Features

Dose Rates:	"0" to 0.7 Mrad/hr
Experimental Locations:	61
Temperature Range:	Ambient to 250°C
Gaseous Environment:	Selectable: as required by experiment Standard: air, nitrogen

Table IV.2

HIACA Features

Irradiation Volumes:	Heights:	24 or 48 inches
	Diameters:	12 to 25 inches in several discrete diameters
Dose Rates:	Maximum Volume:	"0" to 1 Mrad/hr in 9 discrete steps
	Minimum Volume:	"0" to 4 Mrad/hr in 9 discrete steps
Design:	Sized to accept large stainless steel test chamber for simultaneous accident environment testing	

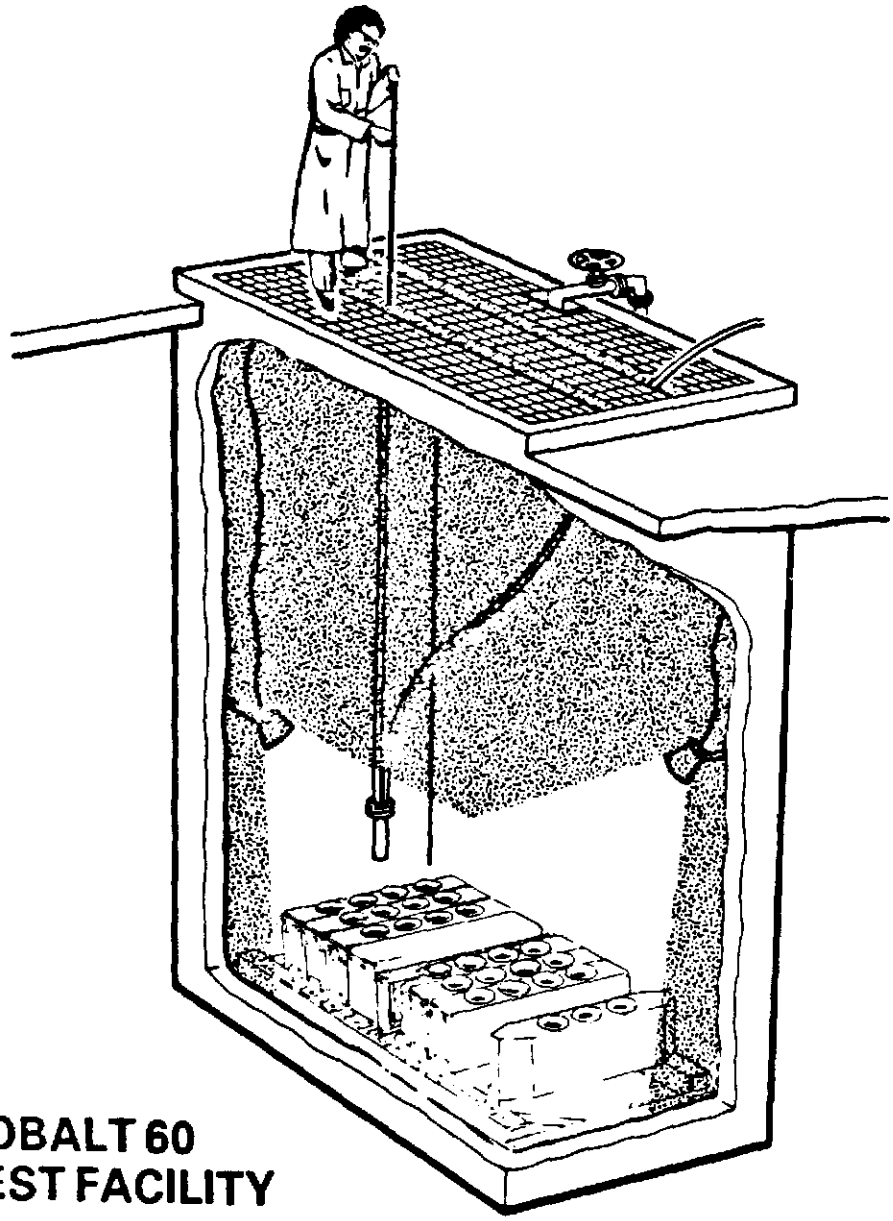


Figure IV.1 General view of LICA (Low-Intensity Cobalt Array)

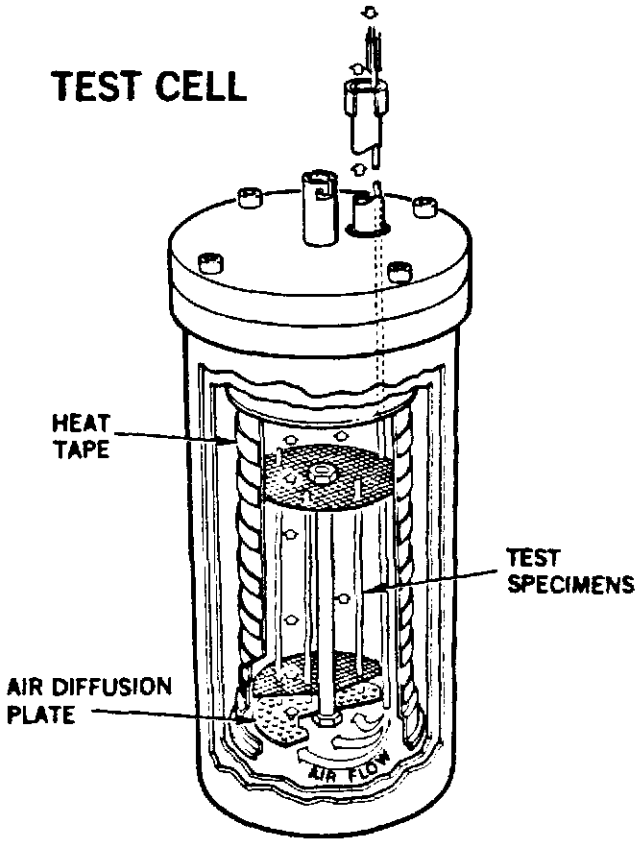


Figure IV.2 Typical LICA underwater radiation test chamber

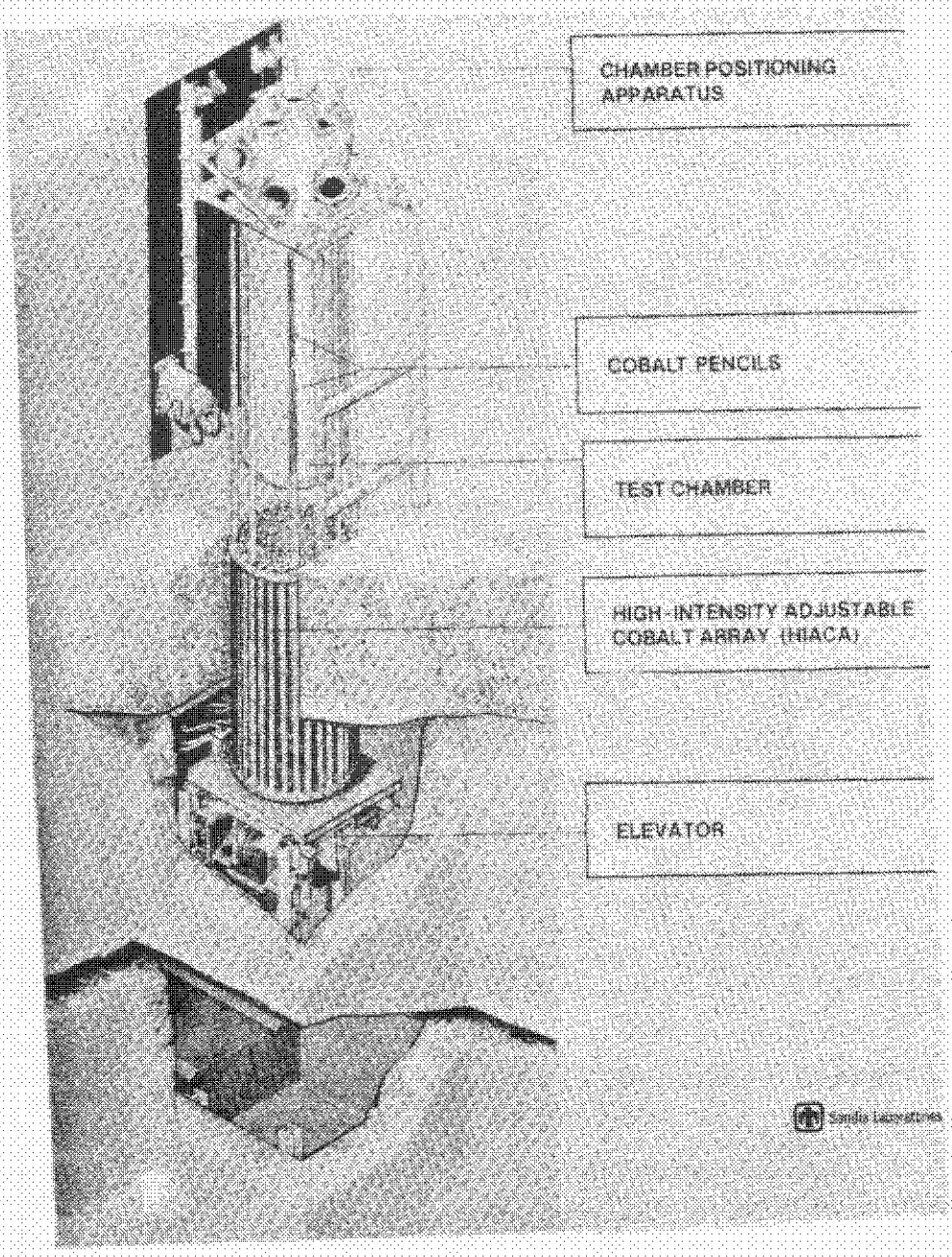


Figure IV.3 HIACA test facility

I-Steam: This steam facility, shown in Figure IV.4, is designed for use in nonradiation or sequential experiments. It has a working pressure of 250 psig, using boilers totaling 9-HP and a 20 ft^3 accumulator steam-storage system. In-line electrical heaters provide significant superheat capability. Four stainless steel test chambers are used: 1-12 ft^3; 2-16 ft^3; 1-40 ft^3 and expandable. Its principal features are summarized in Table IV.3.

V-Steam: This steam facility, shown in Figure IV.5, is specifically designed (and located) for use in simultaneous accident tests in conjunction with the HIACA facility. It has a working pressure of 250 psig, using a boiler totalling 6-HP, and two-35 ft^3 accumulators as steam-storage systems. In-line electric heaters are provided for continuous superheat. A transient superheater, using two-18 ft^3 regenerators each with about 2500 lbs of aluminum oxide in 1-1/4-inch diameter balls for energy storage, has recently been installed. The test chambers described under I-Steam can also be used in this facility; most were specifically sized to accommodate the HIACA source diameter. The V-Steam principal features are summarized in Table IV.4.

Issues Identified

The qualification process and bases and QTE Program integration were discussed previously. The primary issues involved in meeting the QTE Program objectives are grouped under three major headings:

1. Generic issues related to methods for simulating accident conditions.

2. Generic issues related to methods for simulating aging conditions.

3. Special topics related to equipment qualification.

Each of these groupings can be divided further into specific issues which are being addressed by the QTE Program. These are delineated below.

1. Generic issues related to methods for simulating accident conditions.

The nuclear industry typically qualifies electrical equipment to the NRC Guides and through the use of IEEE Std. 323 and its appropriate, specific, daughter Standards. Qualification testing for accidents is normally conducted by exposing the equipment sequentially to radiation levels expected in a LOCA and then to steam at a temperature and pressure history expected for the accident. This overall methodology is currently being investigated by the QTE Program.

In reality, an accident exposure consists of the simultaneous application (to the equipment) of environments including: radiation dose and dose rate, oxygen, chemical spray, superheated or saturated

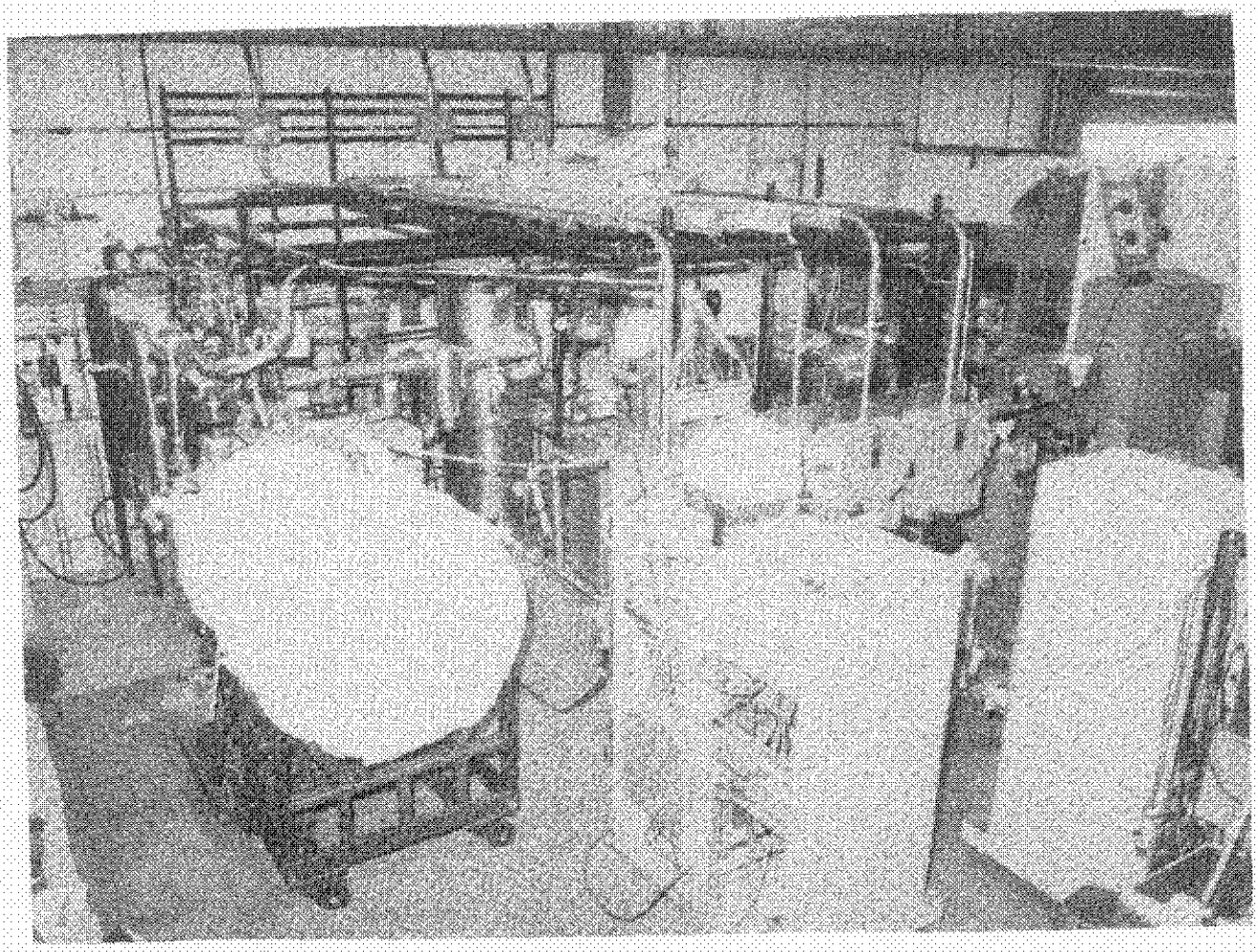


Figure IV.4 I-Steam facility overview, shown with large test chamber in place

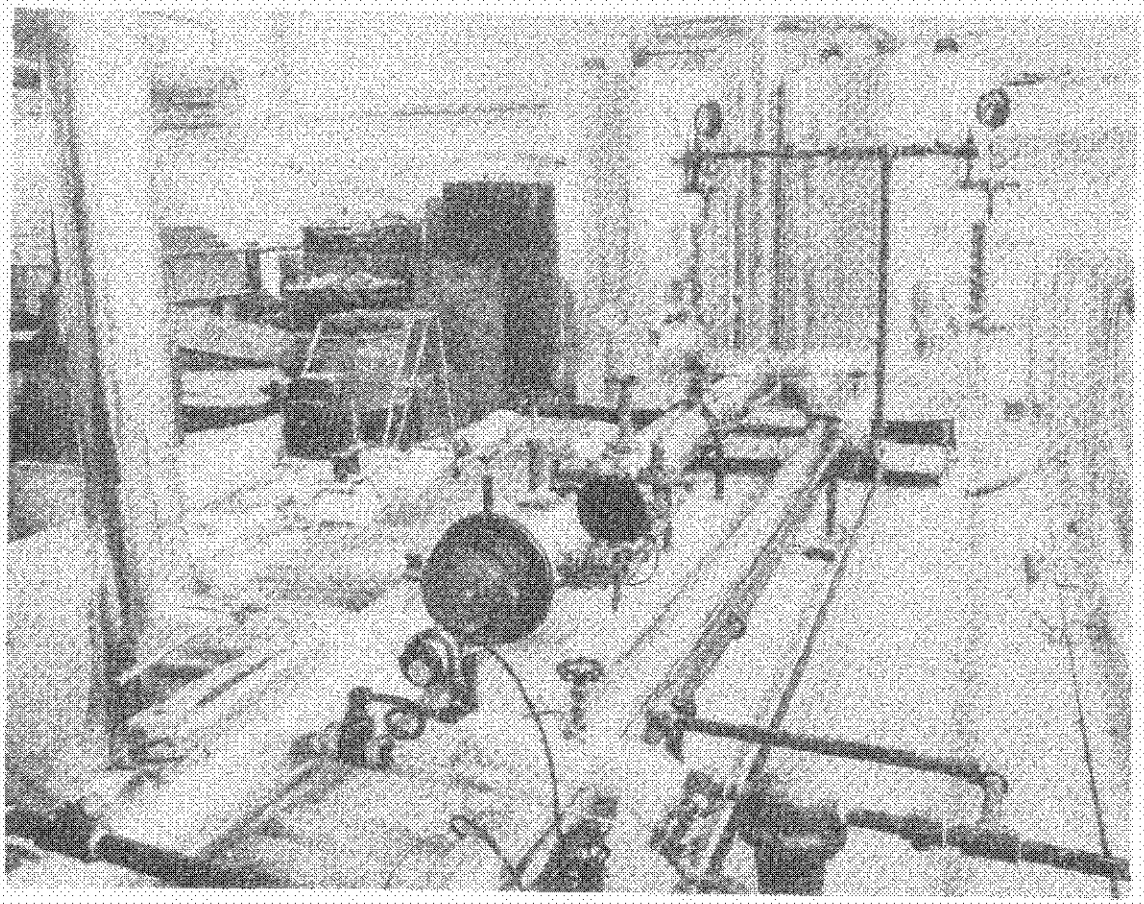


Figure IV.5 V-Steam facility overview

Table IV.3

I-Steam Features

Design		
Pressure	200 psig	
Temperature	750°F	
Saturated-Steam Ramps (8,000 lb/hr peak)	16 ft^3 chamber	40 ft^3 chamber
0 - 70 psig (314°F Sat)	~15 sec	~45 sec
0 - 110 psig (343°F Sat)	~30 sec	~ 2 min
Superheat-Steam Ramps		
Ramp	up to 180°F/hr	
Continuous	up to 750°F	

Table IV.4

V-Steam Features

Design		
Pressure	200 psig	
Temperature	750°F	
Accumulators (each)	35 ft^3	
Total volume	70 ft^3	
Water volume	56 ft^3	
Saturated-Steam Ramps (12,000 lb/hr peak)	16 ft^3 chambers	
	1 chamber	2 chambers
0 - 70 psig (314°F Sat)	10 sec	20 sec
0 - 110 psig (343°F Sat)	15 sec	30 sec
Superheat-Steam Ramps		
Ramp	to 350°F in 1 min.	
Continuous	2000 lb/hr at 750°F for 5 min, then 200 lb/hr at 750°F and 200 psig	

steam, steam impingement, hydrogen burn. A QTE Program goal is to determine criteria which will enable the choice of simulation method which will adequately and/or conservatively replicate the accident exposure. The criteria should allow the choice between sequential and simultaneous testing methods, different preaging methods, different radiation simulation sources, and the like.

The specific issues related to accident simulation being addressed are:

1.1 Is the sequential exposure of equipment to radiation followed by steam an adequate simulation of the accident or are synergistic effects important?

1.2 Are the effects of superheated steam important to duplicate, and what are the appropriate methods?

1.3 Is steam impingement and/or thermal shock important to duplicate, and what are the appropriate methods?

1.4 Are radiation dose-rate effects a factor in equipment degradation/failure and how must they be simulated?

1.5 Can a gamma radiation source be used to simulate the mixed (accident) beta/gamma radiation?

1.6 How does the presence of oxygen affect accident-simulation test results?

1.7 Is the variability of chemical spray composition significant in the accident simulation?

1.8 Can the effects of postaccident environments be satisfactorily accelerated?

1.9 Does the method, or choice, of accelerated-aging affect the behavior of equipment and materials during accident simulation?

1.10 Does the hydrogen burn issue affect the qualification method chosen?

1.11 Is submergence adequately simulated in accident qualification tests?

2. <u>Generic issues related to methods for simulating aging conditions</u>.

The simulation of thermal and radiation aging degradation of materials is used to assure a "conservative" basis for equipment qualification testing. The nuclear industry, to conservatively qualify their equipment, tests it in a state equivalent to the end-of-qualified-life condition.

Since it is not practical to naturally age equipment to the end-of-life condition, faster than real time (i.e., accelerated) techniques are required to artificially change the equipment to a state equivalent to end-of-life. To successfully perform the correct artificial aging program, realistic ambient plant environments must be known, the actual environmental effects of radiation dose and dose rate, mechanical stress, oxygen, humidity and the like must be realistically simulated, and a selection criteria based on an understanding of the correlation between naturally- and artificially-aged material under different test strategies must be identified.

The specific issues related to aging simulation being addressed are:

2.1 What are the realistic plant ambient environments, and how do they impact the choice of accelerated-aging methods in simulation?

2.2 When is the Arrhenius methodology invalid?

2.3 Are dose-rate effects a factor in aging simulation, and how must they be simulated?

2.4 Can sequential exposures be used to simulate simultaneous radiation/thermal environments and does the order of the sequence matter?

2.5 Is mechanical stress an important factor (e.g., in seals/gaskets) in the methods used for accelerated aging?

2.6 What is the effect of oxygen in ambient and accelerated aging?

2.7 Does humidity affect aging, and how can it be accelerated?

2.8 Are there analytical and experimental techniques which can be used to correlate real-time aging and accelerated aging (methods)?

2.9 What aging and degradation experiences are available from actual plant use, and can they be correlated with accelerated-aging methods?

3. <u>Special topics related to equipment qualification</u>.

Besides the generic issues in the previous two groupings, which apply independent of a specific equipment type for example, there are a number of related equipment qualification issues which involve examination of specific events or equipment (e.g., TMI-2 accident experiences, fiberoptics), test methods for specific equipment and in specific Standards, promising new topics (e.g., statistics and fragility, radiation threshold), and the like.

The special topics related to equipment qualification being addressed are:

3.1 Are there test approaches (such as multiple sample, fragility and overstress testing) which can complement qualification type- testing to achieve a broader safety perspective?

3.2 What can we learn from the TMI-2 accident and research regarding simulation of accident conditions?

3.3 Are the procedures given in the IEEE Standards for qualifying specific types of electric equipment adequate?

3.4 Is postaccident monitoring (PAM) equipment adequately qualified to meet the requirements of Regulatory Guide 1.97?

3.5 What are the applicable qualification methods for advanced systems?

3.6 What are the realistic accident environments, and how do they impact the choice of simulation methods?

3.7 Can criteria be established for selecting the appropriate simulation method for equipment and materials?

3.8 Do the Standards and Guides inclusively and adequately encompass the equipment qualification issues?

3.9 What are the appropriate accelerated-aging methodologies for nuclear station batteries?

3.10 Can radiation damage thresholds be established for organic materials, electronics, and equipment generally?

With these issues identified, the balance of this report will concentrate on an issue-by-issue discussion of the results to date and their implications.

V. Format of Issue Status Section

The following section, Section VI, is the substance of this report. But some explanation of its contents is necessary here to guide the reader. Each issue is discussed separately and there is a uniform style to the presentations.

Issue Presentation

The issues are numbered and presented in the order (and style) summarized below:

1. Generic Issues Related to Methods for Simulating Accident Conditions

    1.1 Simultaneous/Sequential Exposures
    1.2 Superheated/Saturated Steam Effects
    1.3 Thermal Shock and Steam Impingement
    1.4 Dose-Rate Effects
    1.5 Beta/Gamma Radiation Effects
    1.6 Oxygen Effects
    1.7 Chemical Spray Effects
    1.8 Acceleration of Postaccident Environments
    1.9 Sensitivity of Accident Simulations to Aging Methods
    1.10 Hydrogen Burn Influence on Accident Simulation Methods
    1.11 Submergence Simulation

2. Generic Issues Related to Methods for Simulating Aging Conditions

    2.1 Realistic Ambient Environments
    2.2 Limitations of the Arrhenius Method
    2.3 Dose-Rate Effects
    2.4 Simultaneous/Sequential Exposures
    2.5 Mechanical Stress Effects
    2.6 Oxygen Effects
    2.7 Humidity Effects
    2.8 Analytical and Experimental Techniques for Correlation Between Natural and Artificial Aging
    2.9 Comparison of Artificially- and Naturally-Aged Equipment

3. Special Topics Related to Equipment Qualification

    3.1 Complementary Test Approaches and Considerations
    3.2 TMI-2 Experiences
    3.3 Evaluation of Qualification Procedures for Specific Equipment Types
    3.4 Regulatory Guide 1.97 Requirements
    3.5 Advanced Systems Qualification Issues
    3.6 Realistic Accident Environments and Calculational Models

3.7 Criteria for Selecting Simulation Methods
3.8 Review of Standards and Guides
3.9 Battery Aging Methods
3.10 Radiation Damage Thresholds

Each issue is presented as a self-contained unit. All page, table, figure, and reference numbering adhere to this Arabic style. There is (essentially) no cross-referencing. The intent is that each issue be displayed as a stand-alone effort; for example, Section 1.6 is discussed in a way that all "Oxygen Effects" activities are presented there. The reader can then, in one place, obtain all relevant research summaries, key findings, current/future effort, and references.

This approach does result in redundancy. However, any quality research program, such as the QTE Program, optimizes its testing effort and hence fosters issue-description redundancy. These general guidelines are followed, to the extent they do not compromise the primary test objectives:

-- combine as many issues into one test as possible; combine equipment, material specimens and/or pieceparts into one test to separate effects and maximize data.

-- several test programs are run in parallel to maximize facility usage and to counter problems in supplier delivery schedules and slippages.

-- the choice of suppliers and models of equipment are based on established criteria, such as:

a. use in safety systems
b. expected sensitivity to imposed environment or test condition
c. experiences in the field or previous test results
d. availability of equipment from supplier
e. general screening for effects

-- sufficiency of testing to demonstrate effect or establish methodologies is not based on exhaustive or inclusive considerations, instead the test results, peer review, and case-by-case judgments dictate sufficiency.

-- test plans, either generic or specific as necessary, are written, reviewed, and approved detailing the choices, rationale, and approaches; this assures consistency, completeness, and agreement.

Thus although on the whole there is redundancy in presentation due to the above guidelines for Program optimization, this approach was consciously chosen to assure an internally complete issue-specific description. The reader may then simply choose the issues of interest and be assured of "completeness" in a global sense.

Issue Format

There is also a uniform style within each issue presentation:

First, a brief issue description is presented. It both summarizes salient points covered in previous sections, and highlights specific technical points applicable to the issue.

Second, the applicable QTE program effort conducted to date is summarized. (Only) the primary reports are discussed and summarized (see fifth item below); applicable charts and figures are also provided for clarification.
In some cases, subissues are presented within the overall issue context.

Third, to the extent known, other U.S. and international programs relevant to the issue are addressed; again this is done through a discussion of specific reports.

Fourth, the key results and conclusions are discussed, this is presented in terms of issue findings.

Fifth, the list of primary (those that are numbered) reports and secondary (those that are alphabetized) reports references are given. This would include references to other U.S. and international work discussed in the third item above.

Appendix

For completeness, the Appendix lists all QTE Program reports, presentations, publications, and relevant correspondence to date. This is done chronologically, by fiscal year. The diligent reader would find that not all appendix references were used in the Section VI issues description. This is mostly due to the original intent of the reference; for example, many of the early formal quarterly and semiannual reports are not used in Section VI.

VI. Issues Status

The following pages contain the issue-specific descriptions. The key to their readability is contained in Section V, and a thorough understanding of the presentation style is necessary prior to any meaningful reading of this section. The issues can be read as a complete QTE Program description, or individual issues of interest can be extracted as complete entities.

VII. Summary

This report has identified the bases for the issues being addressed in the QTE Program and has provided issue-specific status reports.

We conclude that the QTE Program is a comprehensive and integrated effort and has, and is, addressing the pertinent equipment qualification issues. (Yet, it is a vehicle with sufficient flexibility to identify and address any relevant new issues.) It will remain viable by incorporating industry and NRC experiences and the results of the QTE Program research itself, and by monitoring the every-changing Standards and regulations.

Results of the Program have been incorporated into Regulatory Guide 1.89 (Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants), 10 CFR 50.49 (Environmental Qualification of Electric Equipment for Nuclear Power Plants), IEEE Std. 323 (IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations), and other relevant national and international Standards and Guides. Research is continuing in the areas of determining acceptable methods of simulating accident environments, of identifying appropriate means of accelerating the aging of equipment for qualification testing, and of investigating the behavior of various materials and components used in safety-related equipment.

We fully expect that the QTE Program will continue to address and meet its objectives:

(1) to obtain data needed for the confirmation of the suitability of current Standards and Regulatory Guides for Class 1 equipment;

(2) to obtain data that will provide improved technical bases for modifications of these Standards and Guides where appropriate;

(3) to establish data-based and standardized test methodologies for equipment qualification programs; and

(4) to support the NRC licensing process with qualification expertise and test capabilities.

1.1 Simultaneous/Sequential Exposures

Issue Description

Postulated accident environments include simultaneous exposures to radiation, steam, thermal, and possibly chemical spray conditions. Historically, qualification testing has employed a sequential test exposure to simulate the postulated accident environments. Typically, the radiation accident exposure is simulated first. This is followed by a thermodynamic exposure to steam, thermal, and chemical spray conditions. IEEE Std. 323-1974 allows for such an accident simulation sequence. Daughter standards such as IEEE Std. 383-1974 (for qualification of cables) and IEEE Std. 382-1980 (for qualification of safety-related valve actuators) specify such a test sequence.

The influence of synergistic and sequencing effects during accident simulations is not directly addressed by regulatory guidance. For example, 10 CFR 50.49, section e.7 requires synergistic effects to be considered when those effects are believed to have a significant effect on equipment performance. Regulatory Guide 1.89, revision 1 (June 1984) states: "If synergistic effects have been identified prior to initiation of qualification, they should be accounted for in the qualification program." However, Regulatory Guide 1.89 continues: "Synergistic effects known at this time are dose rate effects and the effects resulting from the different sequence of applying radiation and (elevated) temperature." Sequencing effects during accident simulations are not explicitly identified by Regulatory Guide 1.89 as a known synergistic effect.

Numerous safety-related components incorporate polymer type materials into their design. It has been known that the deterioration of polymers is influenced by radiation, thermal, and oxygen environments. In fact, a well published mechanistic pathway for polymer degradation is influenced by all three of these environments. Possible manifestations of the mechanistic pathway include dose-rate effects, synergistic and sequencing effects, and oxygen effects. In this Section, the importance during accident simulation of synergistic and sequencing effects to material and equipment degradation are examined.

In addition to permanent degradation mechanisms, transient failure modes may also be affected by the choice of accident simulation method. Empirically, it is of interest to assess whether equipment failure modes and material properties are susceptible to synergistic effects.

This is the thrust of this effort to address the issue:

> "Is the sequential exposure of equipment to radiation followed by steam an adequate simulation of the accident or are synergistic effects important?"

Program Effort

Reference 1.1.1, NUREG/CR-0275 (SAND78-0067), reported on test results during nine generic LOCA typetests that were conducted on electric cables, cable connector assemblies, and cable field splice assemblies. Testing included both sequential and simultaneous exposures to LOCA radiation, steam and chemical spray conditions. For all tests, saturated steam conditions (without air) were employed. The primary concern was to investigate whether synergisms associated with functional performance of equipment would occur.

Reference 1.1.1 noted that the electric cables showed no obvious synergistic effects in either the electrical or material property. For one cable type, significant enhanced visual degradation was noted for the simultaneously exposed cable compared to the sequentially exposed cable. However, the simultaneously exposed cable experienced substantially more radiation exposure; this was considered an important variable. All cable splice assemblies performed satisfactorily and no evidence of synergistic effects was noted. Most of the cable connector assemblies failed during the saturated steam profile phase of the sequential or simultaneous tests. Hence, no conclusive synergistic effects were observed for these assemblies. There was some indication of a greater failure rate for the sequential test, but the data was sparse.

Reference 1.1.1 also reported on a Franklin Research Laboratories study of their own 1969-1977 historical data to determine whether or not synergistic effects resulted from simultaneous applications of radiation, steam, and chemical spray during qualification testing of Class 1E electrical cable. The study evaluated 43 sequential tests and 6 simultaneous tests. Among the 49 test programs examined, only a single pair of tests (one sequential and one simultaneous) was found to meet certain basic requirements for synergistic-effects comparison; i.e., that the cable specimens and the accident test profiles be essentially the same in both tests. The cables were multiconductor 600 VAC control cables with primary insulations of flame-resistant cross-linked polyethylene, ethylene propylene rubber, and silicone rubber. The outer jackets consisted of flame-resistant neoprene and silicone-saturated asbestos.

All of the cables in the sequential test exposure maintained their electrical load during the 30-day steam and chemical-spray exposure, whereas none of the cables in the simultaneous test maintained their loads beyond 13 days of the combined radiation/steam/chemical-spray (R/S/C) exposure. Two of the three cables used in the simultaneous test program failed before the R/S/C exposure. An analysis of these two tests revealed significant differences in cable handling, thermal aging conditions, and gamma irradiation dose rates. These differences might account for the fact that all three cables in the simultaneous test failed, whereas none of the sequential test cables failed. Thus the comparison could not be regarded as a clear source of information pertaining to synergisms. During the conduct of its test program,

Franklin did not note any obvious correlation between cable performance and whether the test was of the sequential or simultaneous type.

During the conduct of Sandia experimental tests described by Reference 1.1.1, experimental limitations associated with test chamber size, dose rate gradients, etc., were noted. A new unique facility that allowed for exposure to simultaneous accident environments for large-size Class 1E equipment was developed at Sandia. This facility is described by Reference 1.1.2, NUREG/CR-2582 (SAND81-2655). In the facility, a large autoclave test chamber, 88 inches tall by 20-1/2 inches inside diameter, can be surrounded by a 3200-kilocurie cobalt-60 source. The facility is designed to allow for active selection of dose rates during the test.

The facility described by Reference 1.1.2 was employed to examine the effect of LOCA simulation procedures on ethylene propylene rubber (EPR) and cross-linked polyolefin (XLPO) cables' electrical and mechanical properties. The EPR test activity is summarized by Reference 1.1.3, NUREG/CR-3538 (SAND83-1258); the XLPO test activity is summarized by Reference 1.1.4, NUREG/CR-3588 (SAND83-2406).

The test activities described by References 1.1.3 and 1.1.4 examined the issue of simultaneous versus sequential testing by monitoring electrical and mechanical properties of ten commercial cable materials during three simulations of nuclear power plant aging and accident stresses. For all three accident simulations, saturated steam conditions (without air) were employed. Table 1.1.1 summarizes the measured leakage current between conductor and ground at completion of the accident exposure. Prior to performing these measurements, the undisturbed cables and the mandrel on which they were wrapped were inserted into a grounded water bath. The maximum leakage current reading for the detection equipment was 750 mA.

Except for the EPR D cables, leakage currents were comparable for both simultaneous and sequential testing. (For those cables tested using simultaneous techniques only, the leakage currents were comparable to other cables that were tested sequentially.) For multiconductor EPR D cables, Reference 1.1.3 noted a significant leakage current difference between simultaneous and sequential exposures. Surprisingly, electrical properties for EPR D single conductors did not depend on exposure technique. Insulation resistance measurements (Figure 1.1.1) performed periodically throughout the LOCA simulation indicated that electrical degradation for the EPR D multiconductor began several days after the start of the simultaneous LOCA simulation. EPR D insulation also exhibited substantial dimensional swelling; more in fact, than any of the other EPR or XLPO materials tested (Table 1.1.2). EPR D's dimensional swelling was more severe when simultaneous test techniques were employed. EPR F also exhibited substantial moisture absorption during simultaneous exposures but was not tested as a multiconductor.

Reference 1.1.3 suggested that the substantial moisture absorption and dimensional changes produced mechanical damage leading to electrical degradation of EPR D multiconductors. It was hypothesized that

Cable Material	Sequential Test	Simultaneous Test #1	Simultaneous Test #2
Single Conductor:			
EPR A	0.7	0.7	
EPR B	0.8	0.7	
EPR C		0.7	
EPR D	1.0	1.0	0.8
EPR E	0.9	1.2	
EPR F			0.6
EPR G			0.6
XLPO A			0.7
XLPO B			0.6
Multiconductor:			
EPR A	1.0	1.0	
EPR C		1.3	
EPR D	1.2	180-750	150-550
EPR E	1.9	1.8	
XLPO A			1.5
XLPO B	0.7	0.9	0.9
XLPO C			0.9

Table 1.1.1 Measured Leakage Current (mA) at 600 VAC After 1-min Electrification. (From References 1.1.3 and 1.1.4)

Cable Material	Sequential Test	Simultaneous Test #1	Simultaneous Test #2
EPR A	+50	?	
EPR B	+4	-1	
EPR C	+9	+23	
EPR D	+121	+173	+172
EPR E	0	+7	
EPR F			+94
EPR G			
XLPO A			+15
XLPO B			+58
XLPO C			+25

Table 1.1.2 Percentage Weight Increases for EPR and XLPO Tensile Specimens during LOCA simulations. (From References 1.1.3 and 1.1.4)

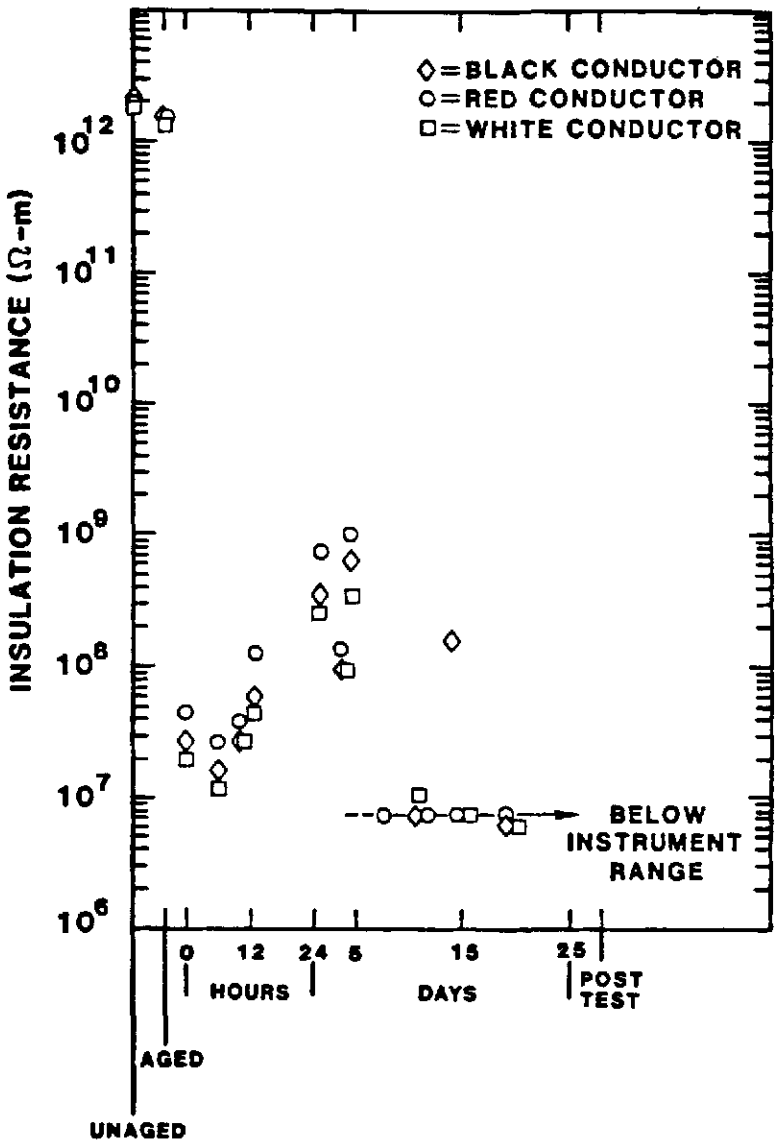


Figure 1.1.1 Insulation resistance values for EPR D multiconductor cable #2 during simultaneous test #2. (From Reference 1.1.3)

dimensional swelling of the insulation caused stress buildup within the multiconductor geometry. When the jacket split to relieve the stress, the sudden release of constrictive force on the insulators may have caused cracking or breakup of the insulation. Alternatively, sections of the insulation which adhered to the jacket during the splitting were pulled away from the conductor. Bare copper conductors (Figure 1.1.2) which were observed at completion of simultaneous testing are suggestive of such a process. Reference 1.1.3 concluded that the importance of simultaneous versus sequential testing for EPR cables would depend on the specific cable material as well as the configuration (multiconductor versus single conductor) in which it is tested.

Reference 1.1.4 examined the implications of the EPR conclusions for three XLPO cables. Substantial moisture absorption and dimensional swelling of XLPO cable insulation during simultaneous accident simulations were not observed. Moreover, ultimate elongation for all three XLPO products at completion of the simultaneous tests were higher than for the EPR D product (Table 1.1.3). Reference 1.1.4 concluded that the lack of XLPO electrical degradation (caused by mechanical degradation) was not surprising. Reference 1.1.4 did not observe for XLPO cables any functional performance differences caused by simultaneous versus sequential testing.

The sequential tests employed by References 1.1.3 and 1.1.4 for cables included sequential aging techniques prior to the sequential accident simulation. Likewise, the simultaneous cable tests included simultaneous aging techniques prior to the simultaneous accident simulation. To differentiate between aging and accident influences on test conclusions, insulation material properties were examined by the efforts described in Reference 1.1.3 for several combinations of aging and accident conditions. Table 1.1.4 illustrates that moisture absorption for several of the EPR materials depended strongly on the accelerated preconditioning applied prior to the simultaneous accident simulation. Those specimens aged to a "40 year equivalent" most severely absorbed moisture during subsequent simultaneous accident simulations. Figure 1.1.3 demonstrates for one of the EPR materials listed in Table 1.1.5 (EPR-1483) that dimensional increases were most severe for simultaneous aging techniques that simulate a "40 year equivalent life". (A simultaneous preconditioning technique to simulate a "40 year equivalent life" had been employed for the EPR D multiconductor cable tests reported in Reference 1.1.3.) Moreover, Figure 1.1.3 suggests that sequential accident simulation techniques may be as severe as simultaneous accident simulation techniques provided appropriate age preconditioning techniques are employed.

The importance of synergistic and sequencing effects during accident simulations was also discussed in Reference 1.1.5, NUREG/CR-4091 (SAND84-2291). This research effort examined synergistic and sequencing effects as part of a comprehensive test program whose experimental variables included: (1) sequential versus simultaneous accelerated aging exposures; (2) the order of the sequential aging exposures; (3) ambient

Product	% Weight Gain	% Outer Diameter Increase	e/eo*	e* (%)
XLPO A	15	6	.16 ± .05	60 ± 20
XLPO B	58	22	.30 ± .12	96 ± 38
XLPO C	25	12	.17 ± .06	56 ± 20
EPR D	173	53	.13 ± .02	31 ± 5

*Measurements performed after moisture desorption had stabilized.

Table 1.1.3 Insulation Specimen Properties at Completion of Simultaneous LOCA Testing. (From Reference 1.1.4)

EPR Material	Accelerated Age	Moisture Absorption (% Weight Increase)
EPR D	Unaged	16
	40-yr equiv*	172
EPR F	Unaged	20
	40-yr equiv*	94
Japanese EOR-5	Unaged	49
	40-yr equiv*	77
EPR-1483	Unaged	17
	5-yr equiv**	22
	40-yr equiv***	67

* A 7d 139°C thermal exposure with simultaneous irradiation for 6 d to 40 Mrd (air-equiv.).

** A 94 d simultaneous exposure to 120°C and 4.9 Mrd (air-equiv.).

*** A 30 d simultaneous exposure to 120°C and 39 Mrd (air-equiv.).

Table 1.1.4 The Effect of "Accelerated Age" on EPR Moisture Absorption During Simultaneous Steam and Irradiation LOCA Simulation. (From Reference 1.1.4)

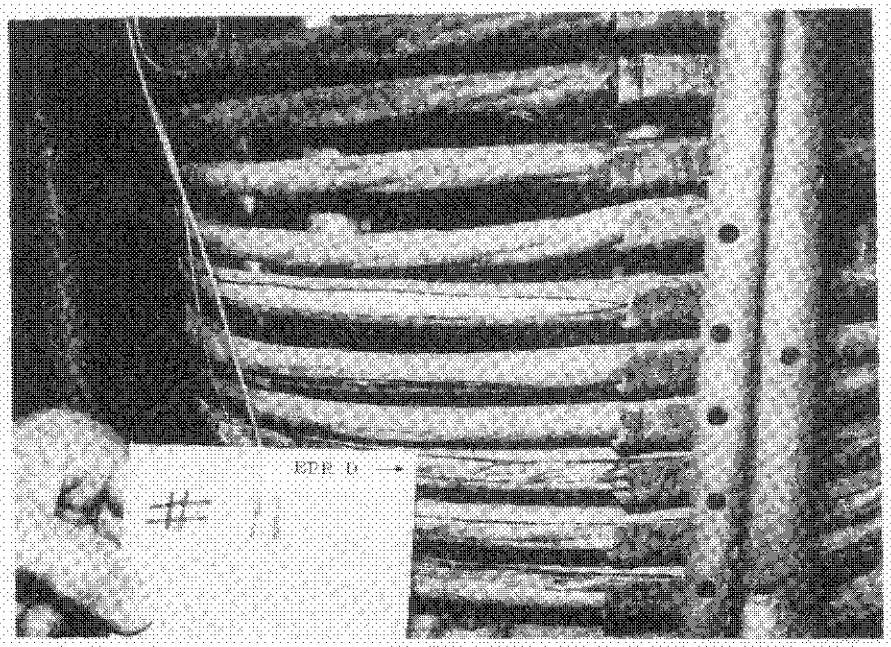


Figure 1.1.2 EPR D cable on day 18 of simultaneous LOCA test #2. (From Reference 1.1.3)

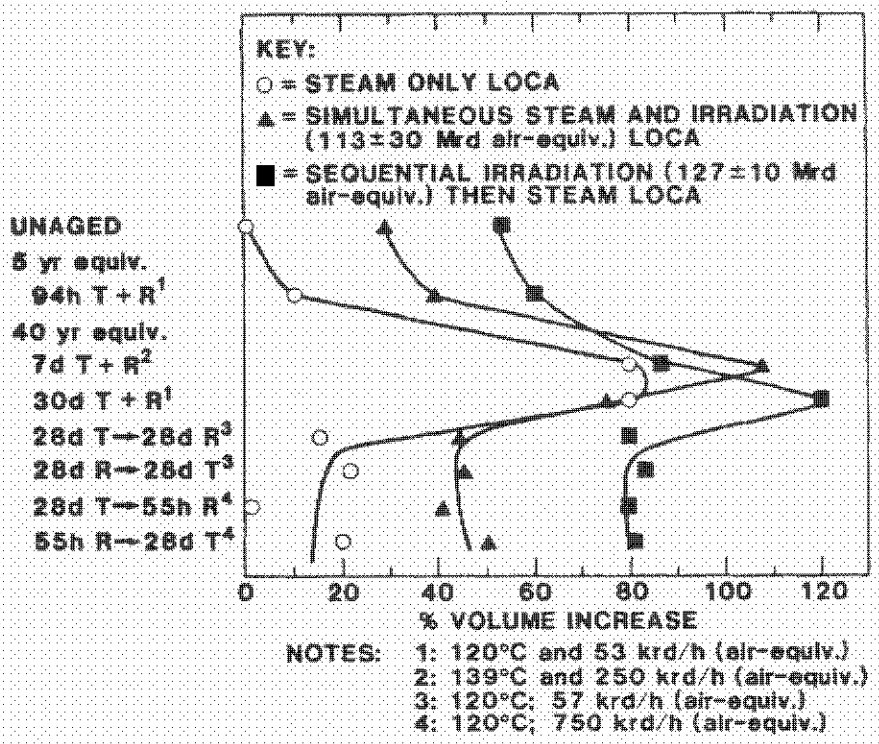


Figure 1.1.3 Effect of aging and accident techniques on % volume change of EPR-1483. (From Reference 1.1.3D)

versus 70°C irradiation temperatures during the sequential aging exposures; (4) presence or absence of oxygen during the accident simulations; (5) sequential versus simultaneous LOCA simulations; and (6) ambient versus 70°C irradiation temperatures during the sequential accident irradiations. Mechanical properties for twenty-three U.S. and French polymer materials were monitored. The overall research goal was to evaluate whether some combinations of alternative aging and accident simulation techniques were better suited for qualification activities than other combinations of simulation techniques.

Reference 1.1.5 noted that perceptions regarding the importance of synergistic effects depended on parameters such as the presence or absence of air during the LOCA simulations. For example, Figures 1.1.4 and 1.1.5 present data for two commercial EPR insulation materials. For both materials, the least degrading accident simulation technique was a simultaneous exposure to radiation and steam conditions without air (i.e., with nitrogen replacing all air in the test chamber). However, one of the more degrading accident simulation techniques was a simultaneous exposure to radiation and steam with air. The sequential accident exposure consisting of a 70°C irradiation followed by a steam exposure with air produced similar material damage as the simultaneous technique.

Reference 1.1.5 data suggests that, for many materials, sequential accident simulations produce similar degradation as do simultaneous simulations, provided appropriate age preconditioning techniques are employed and provided oxygen is included during the steam exposures. Table 1.1.5 identifies the most appropriate sequential simulation procedures for each material included in the Reference 1.1.5 test program. These sequential techniques produced degradation similar to that achieved during simultaneous irradiation and LOCA (with air) accident simulation test exposures except for a fire-retardant French EPDM material (Figure 1.1.6). Reference 1.1.5 considered the simultaneous accident simulation to be the best representation of postulated design basis event accident conditions among those included in the test program.

International and Other U.S. Programs

Tensile properties for two EPR sheet materials were monitored for simultaneous, sequential, and reverse sequential LOCA tests. Results are presented in Reference 1.1.6. In the sequential method, the EPR sheets were exposed to a PWR LOCA steam and chemical spray profile after a 150 Mrad irradiation. For the reverse sequential method, the steam and chemical spray profile was before the irradiation. For the simultaneous exposure, irradiation, steam, and chemical spray exposures were combined. With regards to ultimate tensile elongation properties for the two EPR materials, there was little difference between the three accident simulation methods. Differences between the three simulation techniques were observed for the ultimate tensile elongation. The sequential method was most destructive while the reverse sequential method had a minor

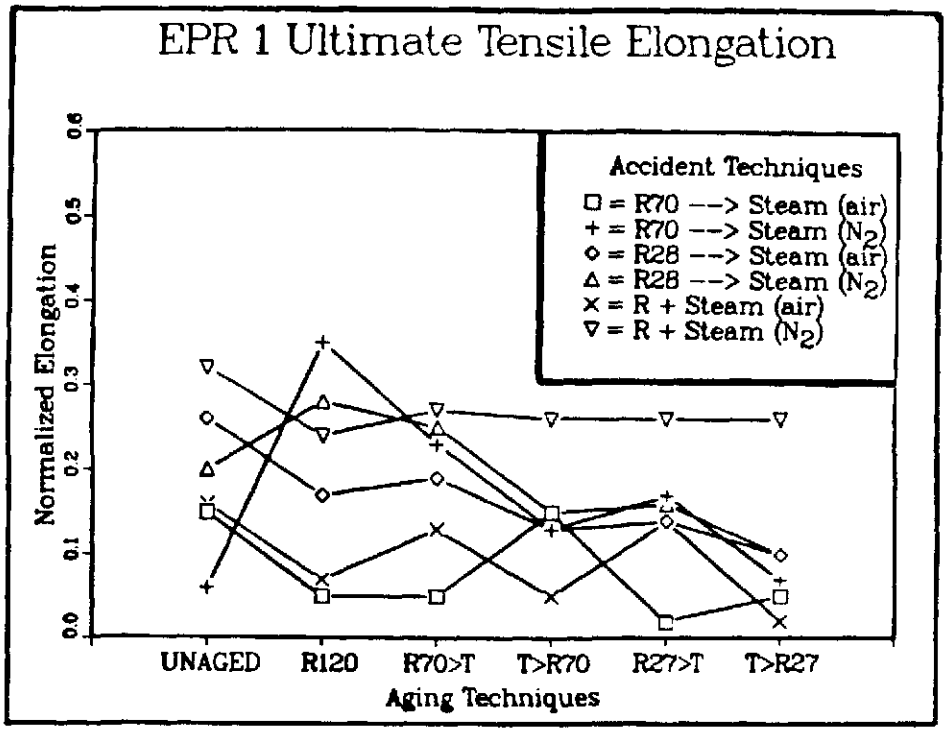


Figure 1.1.4 Ultimate tensile strength of EPR 1 at completion of the accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/h.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/h.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 16-day 120°C and ~ 65 krad/hr exposure (Dose ~ 25 Mrad). R28 and R70 also refer to accident irradiations (dose ~ 60 Mrad; dose rate ~ 300 krad/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.1.5)

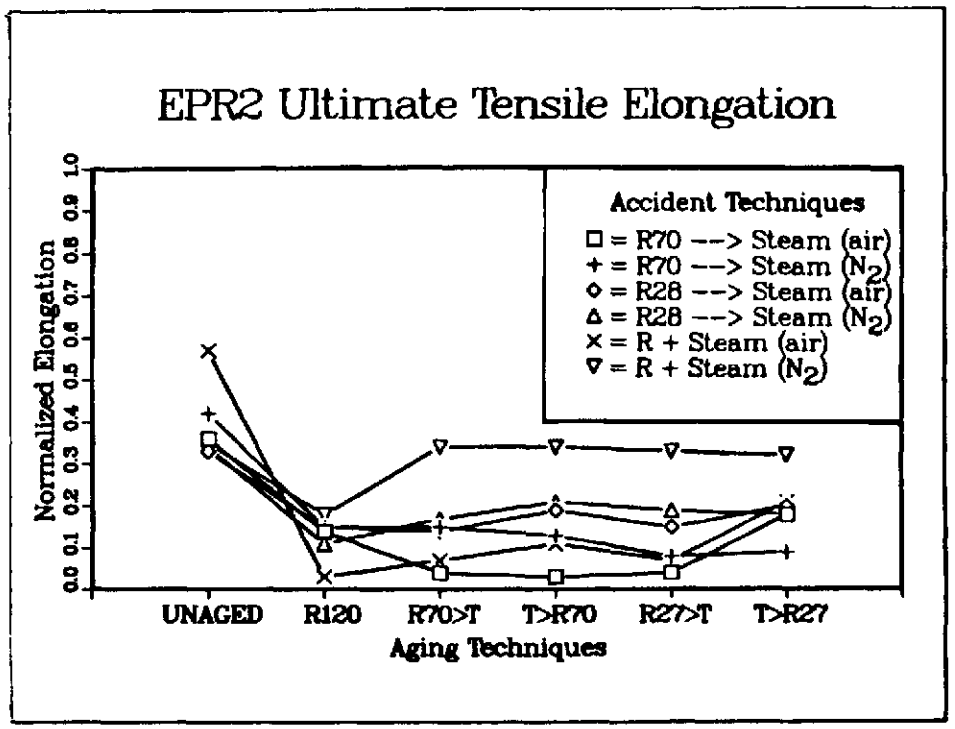


Figure 1.1.5 Ultimate tensile strength of EPR 2 at completion of the accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/h.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/h.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 16-day 120°C and ~ 65 krad/hr exposure (Dose ~ 25 Mrad). R28 and R70 also refer to accident irradiations (dose ~60 Mrad; dose rate ~ 300 krd/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.1.5)

Material	Appropriate Sequential Qualification Procedures
Cross-linked Polyolefins	
XLPO 1	Any sequential simulation
XLPO 2	Any sequential simulation
PRC (82I1)	Any aging simulation followed by the R70→LOCA(air) accident simulation.
Ethylene Propylene Rubbers	
EPDM (82I2)	A R→T aging sequence followed by any accident simulation
EPDM(82I9)	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR (82H4)	Any sequential simulation
EPR 1	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR 2	A R70→T, R27→T, or T→R70 aging sequence followed by R70→LOCA(air) accident simulation
TEFZEL	
TEFZEL 1	Elevated temperature irradiations for sequential aging and accident exposures
TEFZEL 2	R70→T aging sequence followed by an accident simulation
Chlorosulfonated Polyethylene	
HYPALON (82G10)	Any sequential simulation
CSPE	R70→T aging sequence followed by any accident simulation.
O-ring Materials	Any sequential simulation
Connector Materials	
PPS (82H6)	Any sequential simulation
Polydiallylphtalate (82H5)	Any aging simulation followed by the R28→LOCA(air) or R70→LOCA(air) accident simulations
VAMAC	Any sequential simulation
CPE	R70→T aging technique followed by any accident simulation

Table 1.1.5 Appropriate Sequential Procedures to Simulate a Simultaneous Accident Radiation, Steam, and Air Environment. (From Reference 1.1.5)

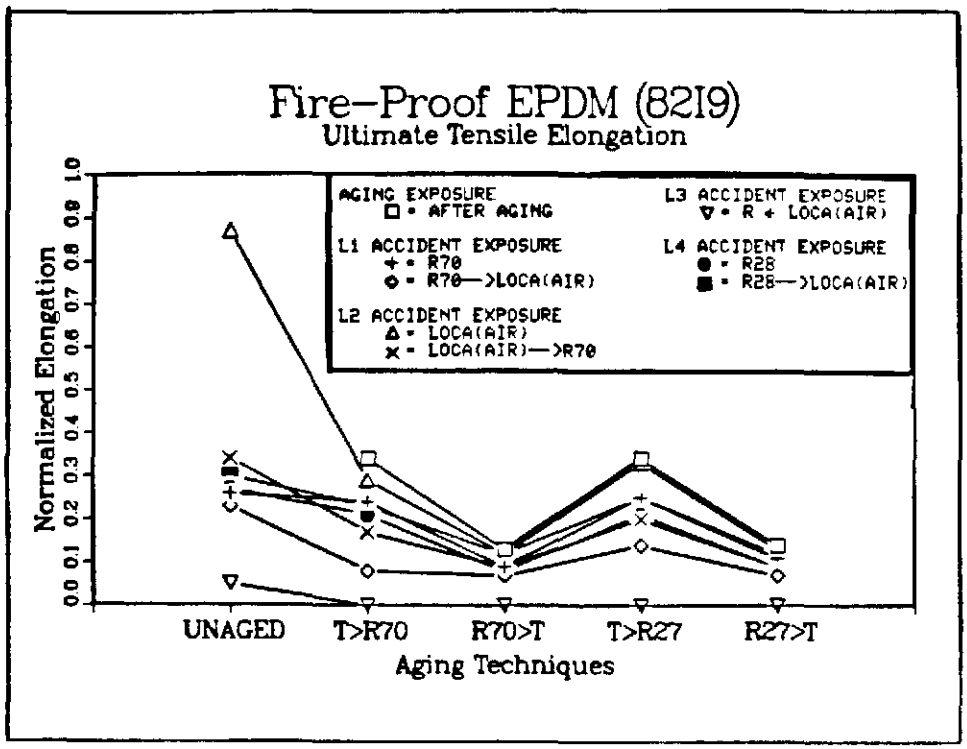


Figure 1.1.6 Ultimate tensile elongation of fireproof EPDM at completion of various phases of the accident exposure, various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation (Dose ≃ 25 Mrad; dose rate ~ 105 krad/h). R70 refers to a 70°C aging irradiation (Dose ≃ 25 Mrad; dose rate ~ 105 krad/h). T indicates a 10-day elevated temperature exposure at 140°C. R28 and R70 also refer to accident irradiations (dose ~60 Mrad; dose rate ~ 300 krd/h). (From Reference 1.1.5)

effect on the ultimate tensile strength. The degree of deterioration caused by the simultaneous technique was intermediate between the two sequential methods.

Reference 1.1.7 compares tensile test results for a flame-resistant EPR insulated, Hypalon jacketed cable that was exposed to a simultaneous test at Franklin Institute Research Laboratories and a sequential test that was performed in Japan. Results are illustrated in Figure 1.1.7. Reference 1.1.7 concludes that synergistic effects were not noted.

Tensile property and volume resistivity data after simultaneous and sequential LOCA testing (with and without air) is presented in Reference 1.1.8. Both PWR and BWR LOCA conditions were simulated. The total accident radiation exposure for PWR conditions was 150 Mrad while the total accident radiation exposure for the BWR conditions was 26 Mrad. Both preconditioned and non-preconditioned specimens were tested. Preconditioning for all specimens was a sequential exposure of thermal aging (7 days at 121°C) followed by irradiation (50 Mrad).

Figures 1.1.8 and 1.1.9 illustrate the comparison of mechanical properties for the test materials after simultaneous and sequential PWR accident simulations without and with air, respectively. Note that when air is not included during the PWR accident simulations that sequential techniques seem appropriate for most of the materials including EPR A and XLPE A. However, when air is included during the PWR accident simulations, simultaneous techniques become more severe. Reference 1.1.8 notes that deterioration of the materials is milder for the BWR simulation than for the PWR simulation. This suggested that the BWR irradiation dose is not sufficient to promote oxidative degradation as was observed during PWR simulations.

Figure 1.1.10 presents Reference 1.1.8 volume resistivity data for several insulating materials at completion of LOCA simulation tests. Simultaneous, sequential, and reverse sequential simulations were employed. Simulations were performed both with and without air in the steam. For the air containing simulations, the simultaneous method was most degrading for the resistivity of EPR, Hypalon, and Neoprene unaged specimens while the reverse sequential method was least degrading. For aged specimens, the simultaneous and sequential simulations produced similar degradation of the resistivity.

Reference 1.1.9 provides additional data comparing simultaneous and sequential LOCA simulations. PWR LOCA simulations were performed for both air and non-air conditions. The authors note that when the LOCA environments were simulated by saturated steam conditions, most materials were slightly more degraded in tensile strength by sequential simulation techniques. However, when air was included in the steam environment, then the simultaneous simulation techniques were slightly more severe.

Reference 1.1.9 also provides degradation behavior data for a Hypalon and an EPR material. It notes that for both materials the ultimate tensile

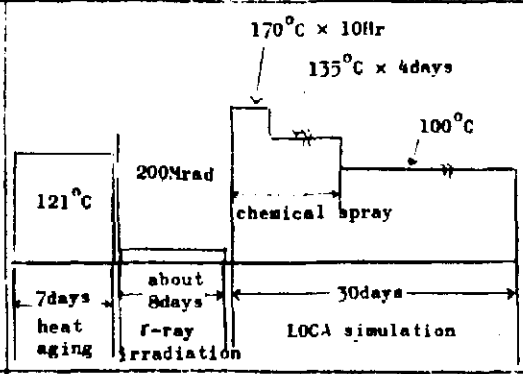
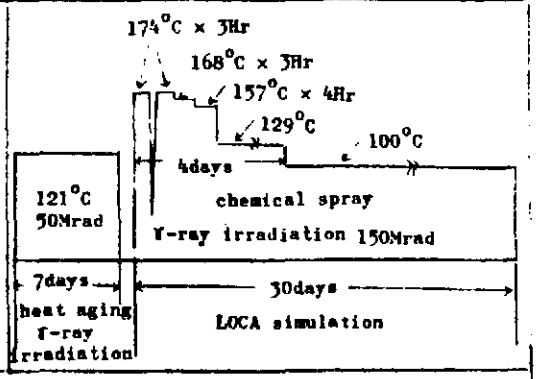
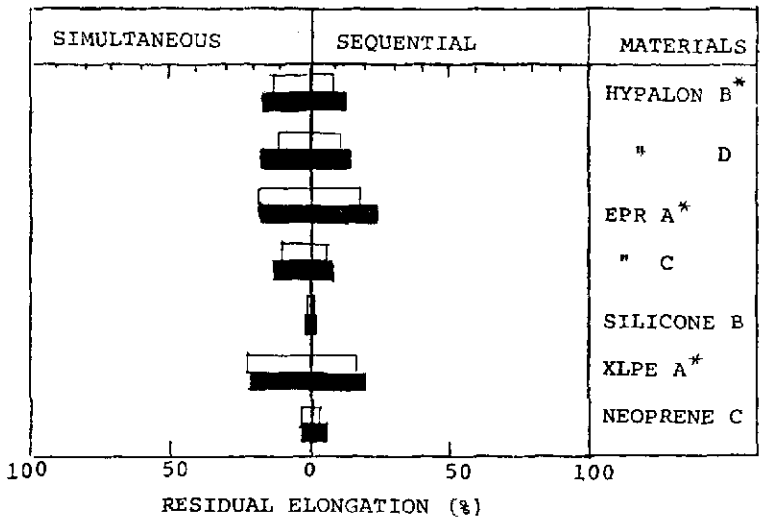
Method	Sequential test			Simultaneous test		
Test Condition						
Specimen	$7^c \times 2mm^2$ Flame resistant EP rubber insulated Hypalon jacketed cable					
Tensile Test after LOCA simulation	Sample	Tensile strength (kg/mm^2)	Elongation (%)	Sample	Tensile strength (kg/mm^2)	Elongation (%)
	Insulation	0.40	140	Insulation	0.38	130
	Jacket	0.90	60	Jacket	1.2	100

Figure 1.1.7 Comparison of tensile test results between sequential test and simultaneous test. (From Reference 1.1.7)



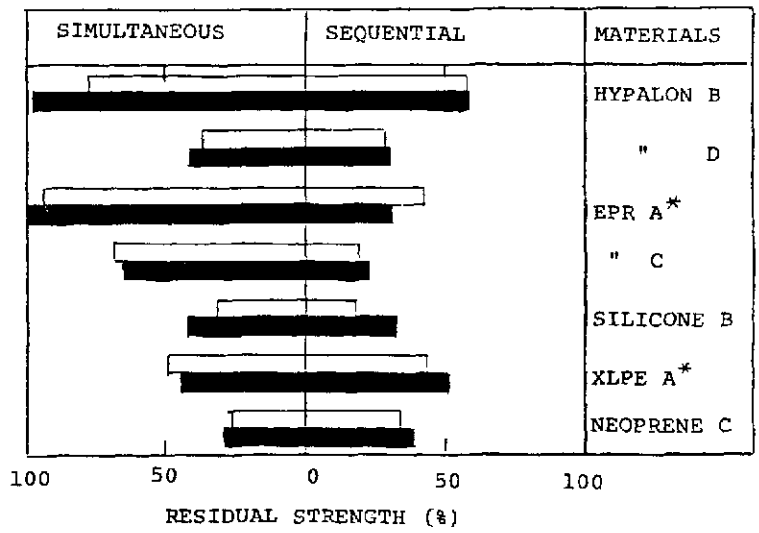


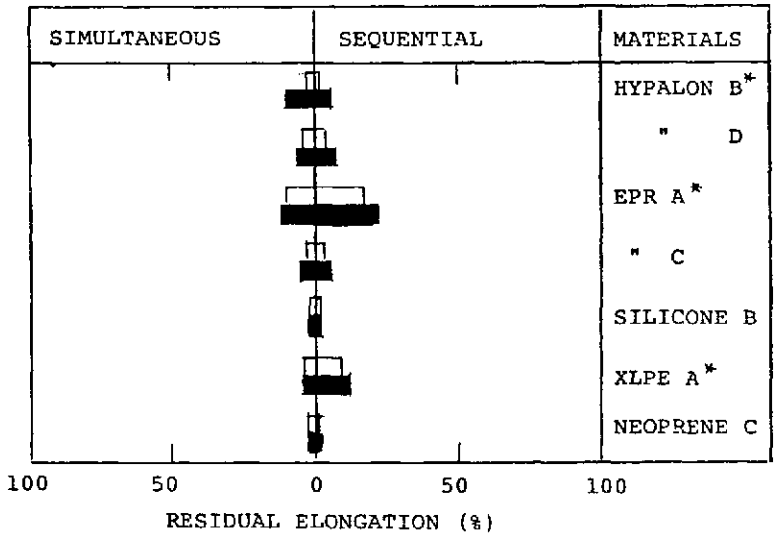
Fig. 12 Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (PWR conditions).

□ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 1.1.8 (From Reference 1.1.8)



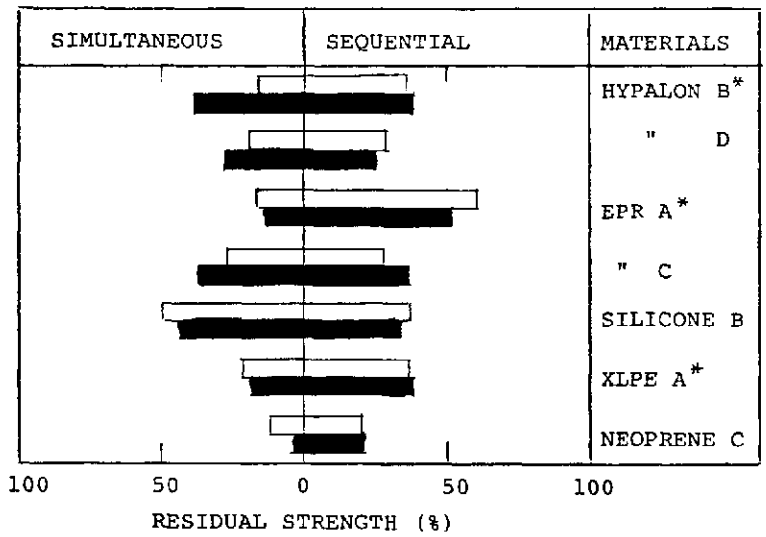
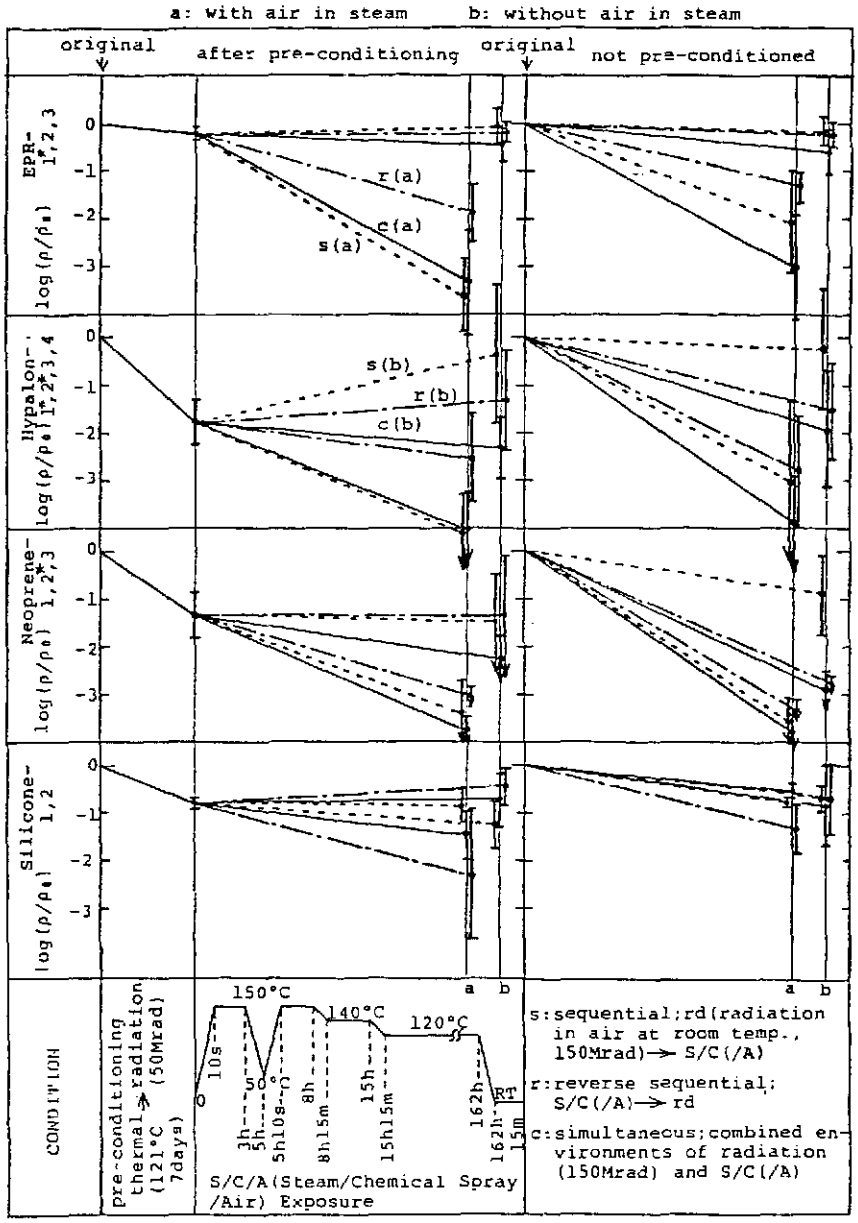


Fig. 13 Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (PWR conditions containing air).

□ ; Pre-conditioned

■ ; Not pre-conditioned

Figure 1.1.9 (From Reference 1.1.8)



Log mean values of volume resistivity for some insulating materials after simultaneous, sequential and reverse sequential LOCA tests.

Figure 1.1.10 (From Reference 1.1.8)

elongation is dominated by the total radiation dose and affected only slightly by the test sequence. However, the ultimate tensile strength for both materials was affected considerably by the test sequence. Simultaneous simulations with air present were the most degrading. Simultaneous simulations without air were least degrading.

Reference 1.1.10 discusses simultaneous and sequential tests performed in France on nine polymer materials. Mechanical and electrical measurements for most of the materials indicated that sequential accident exposures were more severe than simultaneous accident exposures. An exception was Hypalon. For this material, simultaneous testing was most severe. Test materials included two polyamide-imide varnishes used in motors and coils; one epoxy resin, glass fiber insulation, three elastomeric materials, a silicon thermoset, VAMAC and others.

Reference 1.1.11, NUREG/CR-3863 (SAND84-1264), described an experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested; each transmitter was exposed to a unique environment. The environments were (1) simulated LOCA steam/chemical spray conditions alone, (2) temperature alone, (3) radiation alone, (4) simultaneous radiation and LOCA temperature (no steam) conditions, and (5) simultaneous radiation and simulated LOCA steam/chemical spray conditions.

The response of the transmitters showed that temperature was the primary environmental stress affecting the tested transmitters' performance. The design of the transmitter proved to be exceptionally hard to radiation effects and there appeared to be no significant synergistic effects between radiation and accident temperature. For one transmitter, accuracy was monitored several times, while the radiation exposure (applied concurrently with LOCA temperature (no steam) conditions) was discontinued and then restarted. Transmitter output varied by less than one percent during this exercise. Exposure dose rates ranged between 0.4-0.6 Mrad/h during the test.

Issue Findings

Research experiments have given differing results regarding the importance of potential accident synergistic effects for materials degradation and equipment performance depending on the type of material tested. For many materials, synergistic effects during accident simulation were not observed. The majority of materials studied in References 1.1.3, 1.1.4, and 1.1.10 present examples. Scattered throughout the References however are examples where synergistic effects did appear to be important. This includes the U.S. EPR D cable mentioned in Reference 1.1.3, the fire-retardant French EPDM material mentioned in Reference 1.1.5, the French Hypalon material mentioned in Reference

1.1.10 as well as numerous Japanese materials mentioned in References 1.1.8 and 1.1.9.

The presence of oxygen during accident simulations influences perceptions regarding the importance of accident synergistic effects for some materials. This was demonstrated for U.S. EPR materials by Reference 1.1.5 and for Japanese materials by References 1.1.8 and 1.1.9. For LOCA simulations containing air (References 1.1.8 and 1.1.9), the simultaneous accident simulation technique was frequently more severe to materials' properties than were sequential techniques. Early research studies did not include oxygen as an experimental parameter and hence did not consider this as an accident synergistic effect in reaching conclusions. Currently, a realistic accident oxygen partial pressure profile for use in equipment qualification has not been determined. The current oxygen simulation tests are studying how to simulate the time behavior of the oxygen partial pressure during a realistic accident.

The choice of age preconditioning techniques prior to the LOCA simulation also influences the results when assessing how simultaneous and sequential accident simulation techniques affect material properties. References 1.1.3 and 1.1.5 suggest that for many materials, appropriate age-preconditioning techniques assure that degradation observed during sequential accident simulations is similar to that resulting from simultaneous accident simulations.

An industry preconditioning technique is to apply the combined aging and accident irradiation dose prior to the thermal aging exposure. A LOCA steam and chemical spray exposure then follows thermal aging as the last environmental exposure steps in the test sequence. This particular sequence avoids multiple shipments of test specimens to irradiation facilities from qualification test laboratories (thus limiting possible shipping damage to specimens) and has, therefore, become increasingly popular.

While numerous material property studies have been conducted, substantially less data is available comparing equipment functional performance during simultaneous and sequential accident simulations. Cables are the single equipment item that has been studied extensively. Other experimental efforts using connectors, pressure switches, and flexible metal hoses did not provide conclusive information since equipment failure mechanisms unrelated to the simultaneous/sequential issue caused termination of these research tests. (These tests are described in Section 3.3.)

Only one equipment item, a pressure transmitter, has been tested to evaluate whether simultaneous radiation, steam, and chemical spray conditions cause transient failure mechanisms that would not be observed during sequential testing. For the tested pressure transmitter, transient synergistic effects were not noted.

Ultimate tensile properties for cable insulation and jacketing materials were measured and studied in numerous research programs. Reference 1.1.3 suggests that moisture absorption may contribute to mechanical damage of some cables which exhibit large swelling; an example was noted during simultaneous testing.

There are no facilities currently performing qualification tests in the United States using simultaneous radiation/steam/chemical spray. However, including a partial pressure of oxygen during accident simulations and performing appropriate age preconditioning, both of which might mitigate the need for using simultaneous accident simulation, has been noted in recent industry practice. For example, in October 1984, a Class 1E equipment manufacturer informed the NRC that its qualification tests for a new line of equipment would include air in the test chamber during the entire environmental accident simulation.

In summary then, the findings within this issue are:

1. Demonstration that the presence of oxygen during accident simulations can impact the synergistic behavior of some materials.

2. Demonstration that age preconditioning appears to mitigate the importance of synergistic effects during accident simulations for some materials.

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1.2 Superheated/Saturated Steam Effects

Issue Description

In certain combinations of accident conditions and equipment location, some components (requiring qualification) could be subjected to superheated steam environments. The effects of these "dry," high-temperature environments may be a contributing factor in materials and equipment degradation and/or failure. This is especially true of components considered to be "temperature sensitive" under milder conditions. Conversely, "enveloping" tests which attempt to combine superheated and saturated steam profiles, may mask failures of equipment sensitive to moisture condensation.

The requirement to consider superheat conditions and/or effects is not directly addressed in 10 CFR 50.49. Regulatory Guide 1.89, Section 2.a states that "...superheated steam followed by saturated steam may be a limiting condition and should be considered."

Current equipment qualification test standards do not require subjecting specimens to superheat conditions. For example, IEEE 323-1974, Table A2, has a footnote stating that "...it is acceptable...to follow the temperature profile and allow the pressure to conform to saturated conditions..." The justification for allowing this procedure is the assumption that temperature is the more important parameter and the increased pressure will, if anything, increase the severity of the test. However, current industry practice is to perform qualification tests on pressure-sensitive equipment using superheat conditions at the start of the test profile followed by saturated conditions at lower temperatures and pressures.

This is the thrust of this effort to address the issue:

> "Are the effects of superheated steam important to duplicate, and what are the appropriate methods?"

Program Effort

Reference 1.2.1, NUREG/CR-0275 (SAND78-0067), presented the results obtained in a series of tests of Class 1 equipment under postulated LOCA profiles. Among the equipment tested were cable connector assemblies constructed by three suppliers. The connectors were aged (radiation and thermal) prior to the LOCA simulations. The LOCA profiles consisted of simultaneous or sequential applications of radiation and saturated steam/chemical spray environments. Of eleven connector assemblies tested, six failed immediately after 70-psig steam application, two sets failed later during the test, and three did not fail during the test -- but posttest measurements indicated failure.

Reference 1.2.2, NUREG/CR-1191 (SAND79-2311), discusses a qualification verification test of six electrical connectors. Three each of two different types of electrical connectors were tested. Although not

identical to the connectors discussed in Reference 1.2.1, these six connectors were similar in design (male-to-female pin connectors, with screw-type coupling and O-ring seal). The connectors were aged (radiation and thermal), then subjected to an accident test (superheated steam). The connectors were subjected to pressure/ temperature conditions as high as 55°F superheat at 55 psig during the test, with superheat conditions existing throughout the remainder of the test. Posttest inspection revealed little or no degradation of the connectors.

Upon reviewing the test conditions and results presented in References 1.2.1 and 1.2.2, there may be a qualitative argument that (for these component types, at least) superheat environments are less conservative than saturated conditions.

Reference 1.2.3 (RS4445/81/03) describes another verification test made on an instrumentation penetration assembly, similar to those installed in some nuclear power plants. The assemblies were subjected to thermal and radiation aging and LOCA/MSLB steam (superheat and saturated) exposure sequences. Temperatures at the first two LOCA/MSLB profile plateaus were enhanced 122°F and 82°F, respectively, above the 15 psig saturated steam condition. Penetration assembly performance was measured by insulation resistance measurements between conductors and ground.

During LOCA/MSLB accident exposure, several circuits did not maintain a minimum acceptable insulation resistance. These circuit failures became evident only in the saturated steam portion of the test (i.e., they were not detected during the superheat phases). During disassembly, subsequent to the exposure, an abnormal distortion in cable grommets, an elastomeric component in the penetration connectors, was observed.

Circuit failures were attributed to grommet extrusion caused by thermal expansion due to high temperatures reached during aging and accident exposure sequences. Extrusion of the grommet caused cable insulation to be stripped from cable conductors and ultimately resulted in shorts to ground as water condensed during the saturated steam portion of the accident simulation.

Reference 1.2.4, NUREG/CR-4536 (SAND86-0450), discusses the results of a superheat-steam test conducted on three (different) EPR cable products. Unaged and simultaneously aged (282°F for 168 hours and 40 Mrads, 40-year equivalent) cable specimens were tested in a simultaneous radiation and steam environment. The cables were energized at 480V and 0.6A throughout the accident exposure, and IR measurements were made periodically during the exposure.

The report compares the results obtained to those reported in reference 1.2.5 (NUREG/CR-3538, SAND83-1258), in which similar cable products were tested under saturated steam conditions. The cables were exposed to the same (IEEE 323-1974) temperature profiles in both tests (340°F, peak) but the chamber pressure was maintained at 77 psia for the two peaks and first (320°F) plateau during the superheat test. Cable specimens

included single conductors and multiconductors in each test; however, only one cable set was the same for both tests. Despite this, reference 1.2.4 lists the following comparisons of results from the two cable tests:

(1) single conductor cables exhibited high IR values and low leakage currents in both tests; and

(2) both tests showed the following results for the multiconductor cables:

- the jacket had a longitudinal split;
- the exposed gap in the jacket was approximately 0.6cm wide;
- bare conductors were visible;
- the circumference of the jacket increased during the test, although the increase was not large enough to contain the bundle of conductors;
- the cables had large leakage currents after one minute at the following voltages:

saturated steam		superheated steam	
600 V	180-750 mA	600 V	2 mA
		1200 V	5 mA
		1800 V	750 mA

- the IR values followed the same pattern throughout the tests except that in the superheated-steam test the IR readings were below the instrument range at 19-21 days, whereas in the saturated-steam test the readings were below the instrument range at approximately 8, 12, and 18 days into the accident.

Based on the results of both tests, reference 1.2.4 goes on to make the following general conclusions:

(1) Results from single conductor cable tests may not be conservative when trying to qualify multiconductor cables.

(2) Superheat appears to have little effect on cables other than slightly delaying the time to failure.

(3) The differences between the electrical degradation of single and multiconductor cables does not appear to be generic to all cables.

International and Other U.S. Programs

A LOCA-simulation test of terminal blocks was performed under a prior NRC-funded program (Independent Verification Testing Program), and is summarized in Reference 1.2.6, NUREG/CR-1952 (SAND81-0151).

Twenty sliding (or "U") link terminal blocks were subjected to an

80-psig, 10-second saturated steam thermal shock test followed by a 2-hour steady-state environment. Other conditions of a LOCA simulation, such as radiation, chemical spray, or temperature/pressure soak, were not a part of this test series.

The assembled terminal blocks were wired and installed in a NEMA-4 enclosure modified with four 1-inch holes cut into its bottom. The holes allowed cabling/thermocouple access and simulated the "leaky" enclosures typically used in tests.

Rise time (to 80 psig) was about 6 seconds. Rise time (to 156°C) was about 10 seconds. Saturated steam conditions prevailed during the test.

Table 1.2.1 summarizes the response of the six thermocouples. The response of the thermocouples attached directly to the terminal block binding posts is very interesting with very slow response time, even though four large holes were cut into the bottom of the NEMA-4 enclosure.

Table 1.2.1
Thermocouple Response
(From Reference 1.2.6)

Time	Chamber Temperature Average of 4 Thermocouples °C	NT Block °C	ZWM Block °C
0	27	27	26
10 seconds	158	83	79
20	161	102	98
30	161	114	108
40	161	121	114
50	161	123	118
1 minute	162	125	122
2	162	133	132
3	161	138	138
4	161	141	141
5	161	144	144
10	161	152	152
15	161	155	154
20	161	156	156
25	161	157	157
30	161	158	158
40	161	159	159
50	161	160	159
60	161	160	159

We believe that these results may be broadly significant in LOCA-simulation testing. They indicate that protected/enclosed equipment does not respond quickly to transient external environments. Hence, at least qualitatively, many thermal transients (e.g., short-term superheat) may not need to be accurately/actually simulated, with no important impact on the overall qualification program adequacy.

We are not aware of any other significant activities related to this issue.

Issue Findings

By performing transient and steady-state superheated steam tests and comparing the results with tests performed using saturated steam on identical test samples, the importance of a wet environment on the test results can be evaluated. In some cases, primarily because of test item size, shape, or mass or superheat duration, it may be that superheat need not be a critical test parameter.

There has been little research effort on this issue, and the findings below are based on a limited number of examples.

In summary, the findings within this issue are:

1. Sequential superheat followed by saturated-conditions may lead to component degradation as moisture condenses during the postsuperheat periods (e.g., connector sealing systems).

2. Accident superheat effects have not always been considered during qualification tests.

3. Superheat appears to have little effect on some cables other than slightly delaying the time to failure.

References

Program Effort

1.2.1 L. L. Bonzon, An Experimental Investigation of Synergisms in Class 1 Components Subjected to LOCA Typetests, NUREG/CR-0275, SAND78-0067, Sandia National Laboratories, Albuquerque, New Mexico, August 1978.

1.2.2 L. L. Bonzon, W. H. Buckalew, D. W. Dugan, and F. V. Thome, Confirmation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3, NUREG/CR-1191, SAND79-2311, Sandia National Laboratories, Albuquerque, New Mexico, December 1979.

A. L. L. Bonzon, W. H. Buckalew, D. W. Dugan, and F. V. Thome, Confirmation of the Original Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3, SAND79-2311 (Supplement) Sandia National Laboratories, Albuquerque, New Mexico, February 1980).

1.2.3 W. H. Buckalew and F. V. Thome, Equipment Qualification Research Tests of the Duke Power/D. G. O'Brien Type K Instrumentation Penetration Assembly, RS4445/81/03, Sandia Laboratories, Albuquerque, New Mexico, March 1982. Limited Distribution; obtain through W. Farmer, USNRC.

1.2.4 P. R. Bennett, S. D. St. Clair, and T. W. Gilmore, Superheated-Steam Test of Ethylene Propylene Rubber Cables Using A Simultaneous Aging and Accident Environment, NUREG/CR-4536, SAND86-0450, Sandia National Laboratories, Albuquerque, New Mexico. (To be published)

A. P. R. Bennett, "Quick-Look Report on Superheated-Steam Test on Ethylene Propylene Rubber (EPR) Cables: EPR D lot 1, EPR D lot 2, and EPR H," October 16, 1985.

1.2.5 L. D. Bustard, The Effect of LOCA Simulation Procedures on Ethylene Rubber's Mechanical and Electrical Properties, NUREG/CR-3538, SAND83-1258, Sandia National Laboratories, Albuquerque, New Mexico, November 1983.

A. L. D. Bustard, The Effect of LOCA Simulation Procedures on Ethylene Rubber's Mechanical and Electrical Properties, SAND83-0792A. Presented at the ANS Winter Meeting, San Francisco, October 30-November 4, 1983.

B. L. D. Bustard, Cable Electrical Properties During LOCA Simulations, SAND83-1317C. Presented at the ANS Winter Meeting, San Francisco, California, October 30-November 4, 1983.

C. L. D. Bustard, The Effect of Aging on EPR Cable Electrical Performance During LOCA Simulations, SAND83-2018C. Presented at the International Symposium on Aging in Tests of Safety Equipment for Nuclear Power Plants, Paris, France, May 15-16, 1984.

International and Other U.S. Programs

1.2.6 L. L. Bonzon, et al, LOCA-Simulation Thermal Shock Test of Sliding-Link Terminal Blocks, NUREG/CR-1952, SAND81-0151, Sandia National Laboratories, Albuquerque, New Mexico, May 1981. (IVRS Program Results).

1.3 Thermal Shock and Steam Impingement

Issue Description

Steam impingement onto electrical components may cause mechanical stress, due to thermal shock, and/or affect electrical characteristics (surface conductivity, insulation resistance, etc.). It has on occasion caused electrical circuit failure and reactor shutdown (e.g., USNRC, IE Circular #80-15, June 20, 1980).

As electrical parts are generally protected by enclosures or deflectors in individual arrangements (to prevent moisture intrusion), the simulation of steam impingement effects is not straight forward.

The requirement to consider thermal shock and steam impingement conditions and/or effects is not directly addressed in either 10 CFR 50.49 or Regulatory Guide 1.89.

This is the thrust of this effort to address the issue:

> "Is steam impingement and/or thermal shock important to duplicate, and what are the appropriate methods?"

Program Effort

Reference 1.3.1, NUREG/CR-1682 (SAND80-1957), discusses a systematic investigation of terminal block failure. The investigation simulated the TMI-2 accident, which included steam leaks. It was observed, that in the presence of contamination (dust, reactor spray), terminal blocks exhibited leakage currents as soon as the steam reached the block surface. The time of arrival depended on obstacles and location. In Figure 1.3.1 an example of leakage current dependence on steam injection and on voltage application is shown. Terminal blocks were under 480 volts a.c., and leakage current increased by a factor of about 500 on steam arrival. In the case (II) where voltage was applied after steam, the effect was more pronounced. If the leakage currents are high enough (milliampere at 480 volts) and last long enough, complete electrical breakdown of the blocks may occur. The probability of breakdown is shown for one set of test conditions in Figure 1.3.2.

The results of the study with respect to accident simulation are:

1. Whenever steam can reach surfaces of insulating circuit parts, leakage currents may cause circuit deterioration, and after a time ranging from minutes to weeks, complete circuit breakdown may occur.

2. Leakage and breakdown depend on many parameters. Among these are contamination, protection, and location. Mounting on a large structure (wall) acting as a heat sink will affect surface temperature and, may allow water deposition from condensing steam.

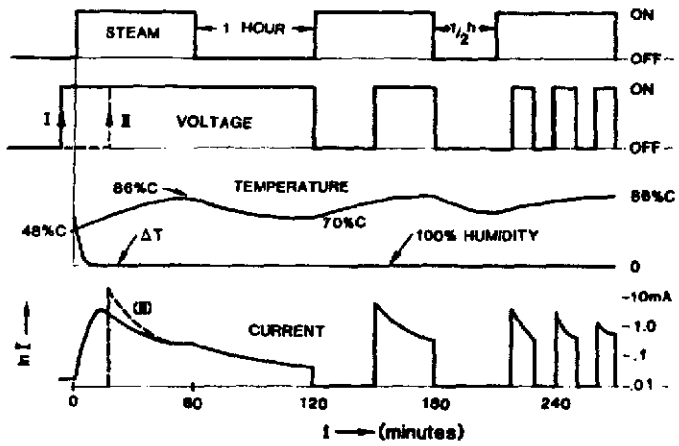


Figure 1.3.1 Steam, voltage, and temperature cycles with representative leakage current. (From Reference 1.3.1)

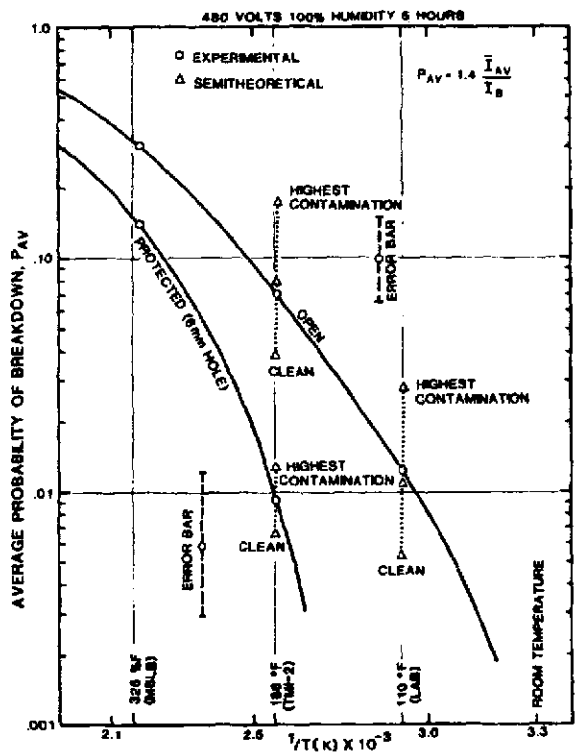


Figure 1.3.2 Breakdown probability of terminal blocks after 5 hours exposure to steam and various contaminants. Upper curve: Open blocks, not directly hit by steam. Lower curve: Blocks in protective (NEMA) boxes with 6mm weepholes. (From Reference 1.3.1)

3. If general simulation of block failure under steam impingement is not possible, arrangements essentially identical to the ones in the containment should be used.

Reference 1.3.2 (RS4445/81/03) describes a verification test made on an instrumentation penetration assembly. The assembly was subjected to thermal and radiation aging and LOCA/MSLB steam (superheat and saturated) exposure sequences. A junction box, inside the steam chamber, protected the connector modules from direct steam impingement.

Temperatures at the first two LOCA/MSLB profile plateaus were enhanced 122°F and 82°F, respectively, above the 15 psig saturated steam condition. The balance of the exposure was made at saturated steam conditions. After a thermal equilibrium of 250°F at 17.5 psig (saturated pressure at Albuquerque elevation) was reached, the penetration assembly was given a thermal shock by spraying it with 39°F water for two minutes. The water spray had little effect on the connector or test chamber temperature. Following thermal shock, the assembly was then sprayed continuously with a borated chemical solution for the remainder of the exposure sequence.

Penetration assembly performance was measured by insulation resistance measurements between conductors and ground. During the latter stages of the LOCA/MSLB accident exposure (i.e., saturated conditions), it was observed that several circuits did not maintain a minimum acceptable insulation resistance. During disassembly, subsequent to the exposure, an abnormal distortion in cable grommets, an elastomeric component in the penetration connectors, was noted.

Circuit failures were attributed to grommet extrusion which was probably caused by thermal expansion under the high temperatures reached during the aging and accident exposure sequences. Extrusion of the grommet caused insulation to be stripped from cable conductors and ultimately resulted in shorts to ground as water condensed during post-LOCA exposure cooling.

International and Other U.S. Programs

Reference 1.3.3 presents the results of a methodology study for qualification testing of safety-related wire and cable. In part, the effect of heating and cooling rates in the first transient part of a LOCA profile on cable materials was studied. The effects of a heating rate of a 1-hour ramp to 150°C was compared to the "standard" IEEE 323-1974 ramp of 10-seconds, and the effects of a rapid cooling rate (150°-50°C in 10 minutes) in the postpeak cooling ramp was compared to the "standard" IEEE 323-1974 cooling rate of 2 hours. Analysis of the effects of the rates on changes in the mechanical properties (elongation and strength) indicated little or no influence on the deterioration of the insulating materials (EPR, XLPE, and Silicone).

This is the only international study on the effects of thermal shock which has come to our attention.

Reference. 1.3.4, NUREG/CR-3863 (SAND84-1264), describes an experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested; each transmitter was exposed to a unique environment. The environments were (T1) simulated loss-of-coolant accident (LOCA) steam/chemical spray conditions alone, (T2) temperature alone, (T3) radiation alone, (T4) simultaneous radiation and LOCA temperature (no steam) conditions, and (T5) simultaneous radiation and simulated LOCA steam/chemical spray conditions.

The response of the transmitters showed that temperature is the primary environmental stress affecting the tested transmitters' performance. Figures 1.3.3 and 1.3.4 show the effect changes in environmental temperature had on transmitter T1: large shifts in transmitter error correspond with the (nearly) step-wise changes in temperature, most notably at the beginning of the test. Transmitters T4 and T5 showed similar behavior to rapid changes in temperature. Transmitter T2, however, exhibited a much smaller error response to the (relatively) slow temperature increase it experienced (see Figures 1.3.5 and 1.3.6).

A LOCA-simulation test of terminal blocks was performed under a prior NRC-funded program (Independent Verification Testing Program), and is summarized in Reference 1.3.5, NUREG/CR-1952 (SAND81-0151). Twenty sliding-link terminal blocks, with known defects, were enclosed in NEMA, Type 4 boxes and subjected to saturated steam (80 psig) conditions -- no chemical spray solution was used. The NEMA-4 enclosure was modified with four 1-inch holes cut into its bottom, thus simulating the "leaky" enclosures typically used in tests.

The purpose of the test was to evaluate the potential for propagation of the defects (cracks) as a result of the LOCA induced thermal shock, specifically to the point of electrical failure.

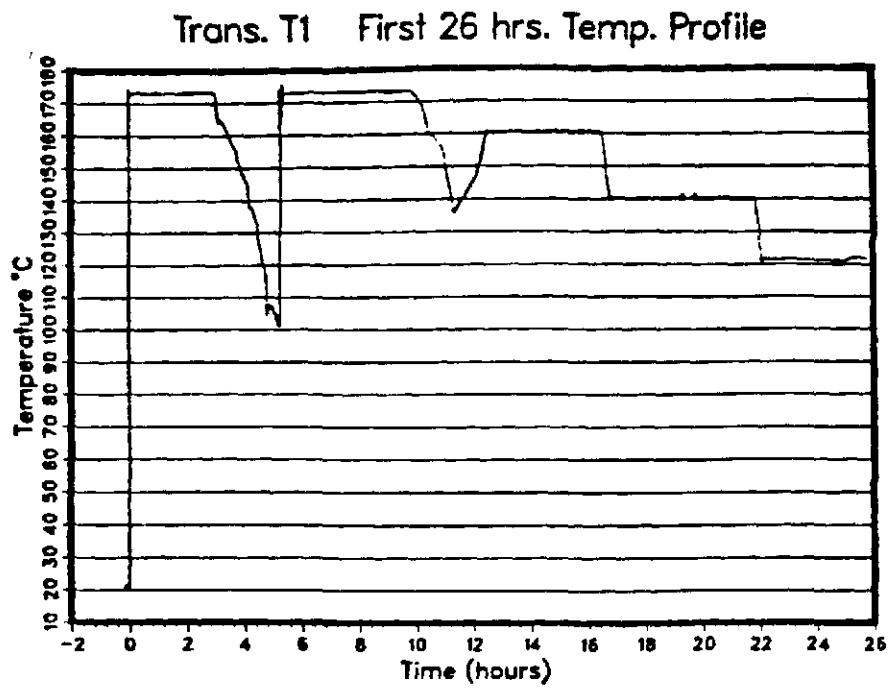
The tests results showed that circuit continuity was not affected by the thermal shock for saturated steam exposure.

Issue Findings

By performing controlled rapid temperature rate tests using thermal shock and/or direct steam impingement on identical test samples, the importance of temperature shock can be evaluated. In some cases, primarily because of test item size or mass, it may be shown that temperature shock need not be a critical test parameter.

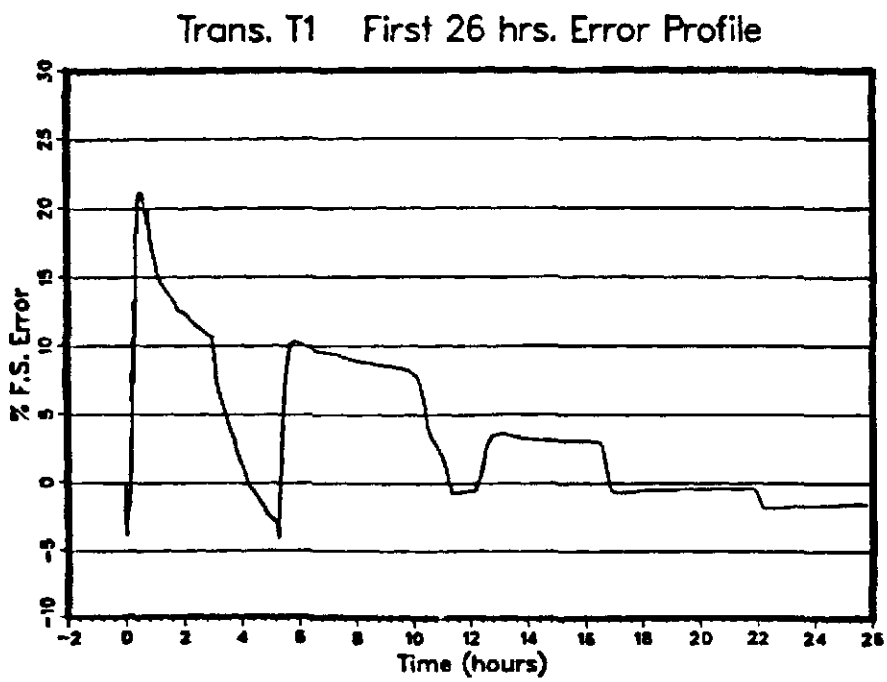
There has been little research effort on this issue, and the findings below are based on a limited number of examples.

Figure 1.3.3



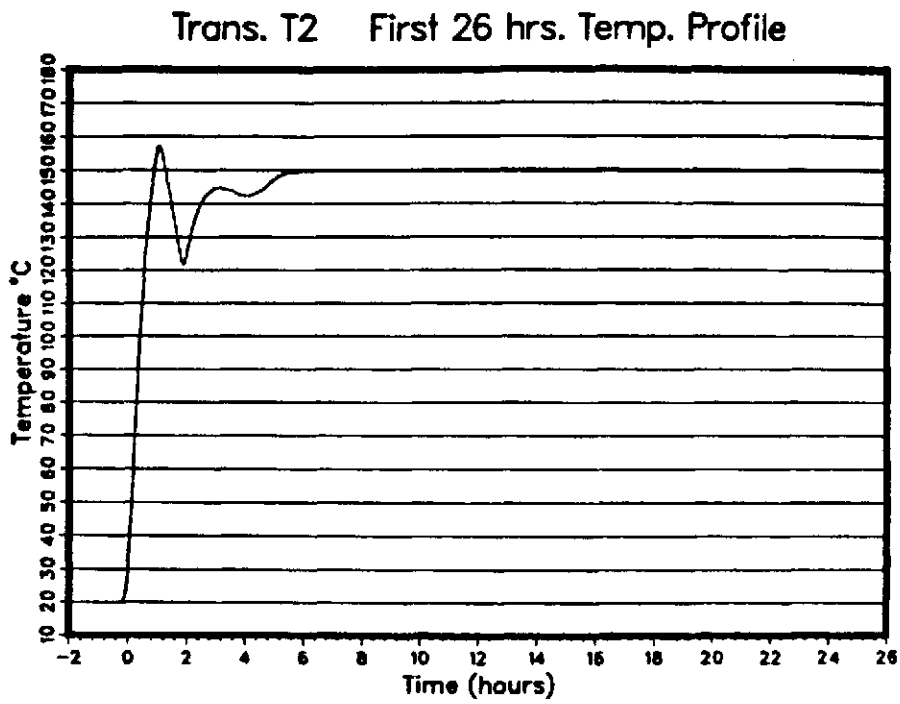
(From Reference 1.3.4)

Figure 1.3.4



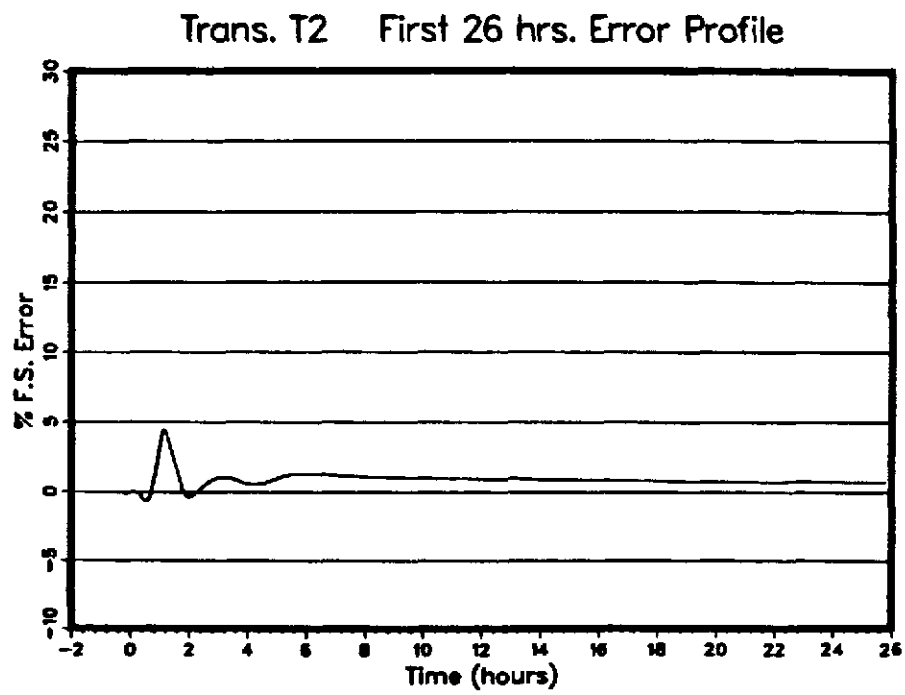
(From Reference 1.3.4)

Figure 1.3.5



(From Reference 1.3.4)

Figure 1.3.6



(From Reference 1.3.4)

In summary then, the findings within this issue are:

1. It is important to consider the degradation and failure of materials and components under conditions of steam impingement and/or thermal shock.

2. Steam impingement simulation is not straight forward and arrangements essentially identical to those existing in containment should be employed in qualification testing.

3. The rate of temperature increase at the start of a LOCA/MSLB simulation can impact the functional performance of some equipment.

References

Program Effort

1.3.1 O. M. Stuetzer, Electrical Insulators in a Reactor Accident Environment, SAND80-1957, NUREG/CR-1682, Sandia Laboratories, Albuquerque, New Mexico, January 1981.

A. O. M. Stuetzer, Preliminary Examination of TMI-2 Equipment: Terminal Blocks, SAND80-2447A. Presented at the USNRC Eighth WRSR Information Meeting, Gaithersburg, Maryland, October 27-31, 1980.

1.3.2 W. H. Buckalew and F. V. Thome, Equipment Qualification Research Tests of the Duke Power/D. G. O'Brien Type K Instrumentation Penetration Assembly, RS4445/81/03, Sandia Laboratories, Albuquerque, New Mexico, March 1983. Limited Distribution; obtain through W. Farmer, USNRC.

International and Other U.S. Programs

1.3.3 Y. Kusama, et al., Methodology Study for Qualification Testing of Wire and Cable at LOCA Conditions, JAERI. Presented at the USNRC Tenth Water Reactor Safety Research Information Meeting, October 12-15, 1982, Gaithersburg, Maryland. Published in NUREG/CP-0041, Vol. 5, pp. 330-360, January 1983.

1.3.4 D. T. Furgal, C. M. Craft, and E. A. Salazar, Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests, NUREG/CR-3863, SAND84-1264, Sandia National Laboratories, March 1985. (CAP Program results)

A. D. T. Furgal and C. M. Craft, Performance Assessment of Class 1E Pressure Transmitters Subjected to Environmental Stresses, SAND84-1655C. Presented at the 12th USNRC Water Reactor Safety Research Information Meeting, October 22-26, 1984, Gaithersburg, Maryland. (CAP Program results)

1.3.5 L. L. Bonzon, et al, LOCA-Simulation Thermal-Shock Test of Sliding-Link Terminal Blocks, SAND81-0151, NUREG/CR-1952, May 1981. (Independent Verification Testing Program results)

1.4 Dose-Rate Effects

Issue Description

One concern in equipment qualification testing under accident conditions is the dose and dose rate(s) required to adequately simulate the radiation environments. Dose-rate effects may be a contributing factor in materials and equipment degradation and/or failure. The requirement for an equipment qualification program to consider dose-rate effects is clearly stated in 10 CFR 50.49, Section e.4 and Regulatory Guide 1.89 (June, 1984), Section 2.c and Appendix D. In the latter for example, the "...dose rate, energy spectrum and particle type..." must be considered.

In general, current testing methods use a sequential application of radiation which is the sum of the expected aging plus accident radiation. Peak rates used in qualification are typically 1 Mrad/hr or less (for example, as recommended in IEEE 383-1974, Section 2.4.2). But much higher peak values have been estimated for the accident condition, as in RG 1.89, Tables D-1 and D-2 which imply peak gamma rates greater than 4 Mrad/hr and peak beta rates greater than 23 Mrad/hr. It should be noted that these peak rates are based upon an instantaneous release (at the start of the accident) assumption, which may not be realistic. Later in the accident, much lower dose rates would be expected, so both high and low dose-rate effects should be addressed. Clearly, accurate (expected) time-dependent accident dose rates should be defined and validated. The effects of such dose rates (peak and low rates, and radiation-induced heating) should then be evaluated with respect to damage, degradation, and functionability of equipment.

This is the thrust of this effort to address the issue:

> "Are radiation dose-rate effects a factor in equipment degradation/failure and how must they be simulated?"

Accident Radiation Dose Rate Definition and Simulation

In the 1977 time frame, Regulatory Guides 1.3, 1.4, and 1.7 (all are siting standards) and proposed drafts of Regulatory Guide 1.89 only implicitly defined the LOCA radiation environment to be simulated in the qualification testing of electrical equipment. However, this information was not directly useable since it was specified in terms of (arbitrary) fission product fractions and distribution assumptions; a translation of these into doses and rates was required.

Reference 1.4.1, NUREG76-6521 (SAND76-0740), was the first report to define these radiation signatures. Notwithstanding the various assumptions (plant specifics, radiological decay, etc.), the results were significant because of the energy spectra and dose and rate magnitudes elucidated. Figures 1.4.1 and 1.4.2 summarize the expected dose and dose rates. The study contained, at least two significant results:

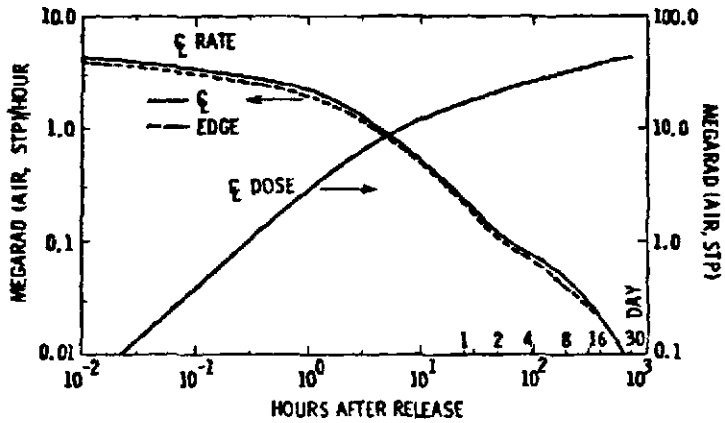


Figure 1.4.1 Rate and dose to air from the combined gamma sources; airbourne source uniformly distributed in the containment volume, plate-out source uniformly distributed on the containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume. (From Reference 1.4.1).

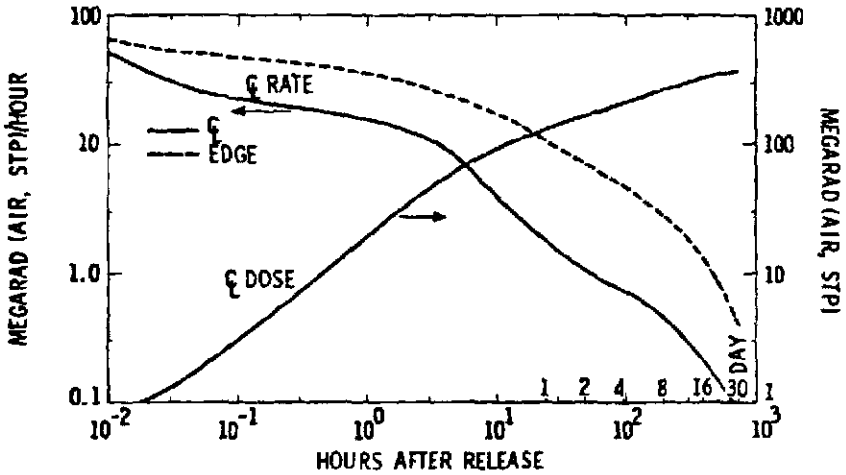


Figure 1.4.2 Rate and dose to air from the combined beta sources; airborne source uniformly distributed in the containment volume, plated-out source uniformly distributed on a containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume. (From Reference 1.4.1).

1. A definition of the expected magnitudes of gamma and beta dose and rates for a typical PWR containment; typical values of initial dose rate are 3.5 and 70 MR/hr and of (30-day) integrated dose are 50 and 300 Mrad, respectively.

2. A formal recognition that the beta dose and rate are significantly greater (about 6 times) then the corresponding gamma values.

Reference 1.4.2, NUREG/CR-1682 (SAND81-2655), characterizes the source strength of a cylindrical Cobalt-60 irradiation array, used in equipment qualification test programs at Sandia. This source array is adjustable in active length (24 or 48 inches) as well as diameter (10-24 inches). Measured dose rates ranged between 0.8 Mrad(air)/hr to 1.5 Mrad(air)/hr for the 22-inch diameter configuration. Higher rates (greater than 5 Mrad/hr) are achievable, but at the sacrifice of irradiation volume size.

Reference 1.4.3, NUREG/CR-3777 (SAND84-0912), discusses a method used to simulate the effects of LOCA electron environments on large-area material and cable samples. It was demonstrated that both electron doses and dose rates -- at the given (time-dependent) average beta spectra energies -- were adequately simulated with large diffuse monoenergetic electron beams.

Equipment Degradation and Failure Modes

Accident dose rates have varied in a number of accident simulation tests (although they were not the primary test parameter).

Reference 1.4.4, NUREG/CR-3538 (SAND83-1258), and Reference 1.4.5, NUREG/CR-3588 (SAND83-2406), discuss the effect of LOCA simulation procedures on two different organic materials. The test series include both sequential and simultaneous aging and LOCA exposures. During these tests, accident radiation (Co-60) exposures in the range of 0.08-0.7 Mrad(air)/hr were utilized.

Material degradation was monitored at two different times during a saturated steam LOCA simulation. Some samples were removed after a 4-day LOCA exposure that included a simultaneous exposure to radiation (~ 65 Mrad at ~ 0.68 Mrad/h). The remainder of the samples were tested after another 12-17 days of exposure that included simultaneous exposures to an additional 44 Mrad at lower dose rates (~ 0.2 and ~ 0.08 Mrad/h). Examination of the two sets of data (Table 1.4.1) allows for a qualitative assessment of the importance of the low dose rate "tail" of the LOCA exposure to material degradation.

Reference 1.4.6 NUREG/CR-3532 (SAND83-2098), reports on investigations of charge breakdown from beta irradiations of EPR insulation material slabs and typical electrical cables. Parameters include air/vacuum, electron energy, and surface termination. Exposure dose rates in the range of 1-40 Mrad(air)/hr were employed. Under certain conditions charge was accumulated and spontaneous breakdown did occur during irradiation in the vacuum environment; however, no evidence of electrical breakdown was

Table 1.4.1
Comparison of Normalized Ultimate Tensile Elongation and Percent Weight Gain for Materials Exposed to Simultaneous LOCA Conditions.
(Based on data presented in References 1.4.4 and 1.4.5)

A. Properties for EPR during Simultaneous Test #1.

Material	4-day LOCA Exposure		21-day LOCA Exposure	
	e/e_o	% wt. gain	e/e_o	% wt. gain
EPR-B	0.17	0	0.14	-1
EPR-C	0.21	9	0.15	23
EPR-D	0.13	120	0.13	173
EPR-E	0.27	NA*	0.19	NA*

B. Properties for EPR and XLPO during Simultaneous Test #2.

Material	4-day LOCA Exposure		21-day LOCA Exposure	
	e/e_o	% wt. gain	e/e_o	% wt. gain
EPR-D (unaged)	0.33	16	0.25	23
(aged)	0.16	144	0.18	172
EPR-F (unaged)	0.28	8	0.10	20
(aged)	0.22	59	0.10	94
XLPO-A (unaged)	0.30	4	0.23	10
(aged)	0.25	9	0.16	15
XLPO-B (unaged)	0.50	15	0.41	32
(aged)	0.42	33	0.30	58
XLPO-C (unaged)	0.47	26	0.30	37
(aged)	0.29	14	0.17	25

* not available

observed during ambient air exposures at approximately 25 Mrad(air)/hr dose rates.

One phenomenon associated with exposure to ionizing radiation is the "self-heating effect". This effect introduces an increase to the applied thermal environment (to the equipment) during accident simulation exposures, and thus may play an important role in the degradation of temperature sensitive components.

As a qualitative example of this phenomenon, Reference 1.4.7 (RS 4445/81/03) notes that during radiation exposure of an electrical penetration assembly, the inner connector temperature reached 64°C, whereas the "free air" equilibrium temperature around the inner connector stabilized at 50°C, and the temperature in the rest of the irradiation cell varied between 32°C and 42°C. After a fan (for forced air circulation) was installed, the "free air" and connector temperatures stabilized at 42°C and 46°C, respectively.

International and Other U.S. Programs

Reference 1.4.8, NUREG/CR-3863 (SAND84-1264), describes an experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested; each transmitter was exposed to a unique environment. The environments were (T1) simulated Loss-of-Coolant Accident (LOCA) steam/chemical spray conditions alone, (T2) temperature alone, (T3) radiation alone, (T4) simultaneous radiation and LOCA temperature (no steam) conditions, and (T5) simultaneous radiation and simulated LOCA steam/chemical spray conditions.

The response of the transmitters showed that temperature is the primary environmental stress affecting the tested transmitters' performance. The design of this transmitter proved to be exceptionally hard to radiation effects and there appeared to be no significant synergistic effects between radiation and temperature.

Exposure dose-rates ranged between 0.4-0.6 Mrad/Hr during this test; however, transmitter T3 response to the raising and lowering of the radiation source was evaluated. By "lowering the source" we mean discontinuing irradiation. During the early hours of exposure (approximately 30 Mrad), lowering the source produced less than -0.2 percent change in transmitter output. At approximately 300 hours (approximately 100 Mrad), the transmitter showed about a -1.2 percent change in output when the source was lowered; while at approximately 800 hours exposure (approximately 300 Mrad), a 0.9 percent sensitivity was observed. Error increased negatively when the source was lowered. The error returned to prior values when irradiation resumed.

Transmitter T4's response when raising and lowering the radiation source was also evaluated. The error increased (became more negative) when the source was lowered. The error returned to prior values when irradiation resumed. For transmitter T4 the change in error profile ranged between -0.2 and -0.6 percent when the source was lowered. Sufficient data was not available to evaluate T5 response.

Reference 1.4.9, NUREG/CP-0058, reports on a Japanese test of electric cable materials conducted to simulate post-LOCA conditions on an accelerated basis. A three-month, 55 krad/hr, test was compared against a one-week, 925 krad/hr test, both with and without oxygen present. Table 1.4.2 summarizes some of the results of this effort. These results should be taken as a qualitative indication of accident dose-rate effects on the materials tested since other factors (such as the presence of oxygen) may also have had an effect.

We are not currently aware of any other significant related activities.

Table 1.4.2

Rate of Decrease in Ultimate Elongation
under LOCA condition (1/100Mrad).
(From Reference 1.4.9)

	Non Air		Air 0.05MPa	
Sample	One Week	Three Months	One Week	Three Months
EPR-A	0.40	0.40	0.83	1.57
EPR-B	0.46	0.46	0.74	1.46
CSPE-A	0.56	1.00	0.77	1.60
CSPE-B	0.80	1.11	1.09	2.14

Decay curves are taken as Maxwellian.

Issue Findings

By performing radiation tests on identical test samples, with dose rate as a parameter, the effect of dose rate on the test results could be evaluated. In accident simulations, one concern is for high-rate effects at the start of the accident followed by an extended low dose-rate exposure.

In summary then, the findings within this subissue are:

Accident Radiation Dose Rate Definition and Simulation

1. Typical (PWR containment) gamma and beta peak accident dose rates are 3.5 Mrad/h(γ) and 70 Mrad/h(β) with integrated (30-day) doses of 50 Mrad(γ) and 300 Mrad(β).

2. Beta doses and rates are significantly greater (about 6 times) than the corresponding gamma values.

3. Early assessment of accident radiation dose rate significance is important in evaluating a radiation qualification program.

4. With a generic data base established, the plant-specific dose and rate values might be accurately established within the bounds of plant-layout assumptions.

References

Accident Radiation Dose Rate Definition and Simulation

1.4.1 L. L. Bonzon, Radiation Signature Following the Hypothesized LOCA, NUREG76-6521, SAND76-0740, Sandia Laboratories, Albuquerque, New Mexico, September 1977.

A. L. L. Bonzon, In-Containment Radiation Environments Following the Hypothetical LOCA (LWR), SAND76-5152A, Transactions of the American Nuclear Society, Vol 23, 1976.

B. L. L. Bonzon, Synergistic Effects and Source Term Considerations Associated with Class 1E LOCA Qualification Testing, SAND77-1713C, USNRC Fifth Water Reactor Safety Research Information Meeting, November 7-11, 1977.

1.4.2 W. H. Buckalew and F. V. Thome, Radiation Capabilities of the Sandia High Intensity Adjustable Cobalt Array, NUREG/CR-1682, SAND81-2655, Sandia Laboratories, Albuquerque, New Mexico, March 1982.

1.4.3 W. H. Buckalew, et al, Capabilities and Diagnostics of the Sandia Pelletron-Raster System, NUREG/CR-3777, SAND84-0912, Sandia National Laboratories, Albuquerque, New Mexico, July 1984.

Equipment Degradation and Failure Modes

1.4.4 L. D. Bustard, The Effect of LOCA Simulation Procedures on Ethylene Propylene Rubber's Mechanical and Electrical Properties, NUREG/CR-3538, SAND83-1258, Sandia National Laboratories, Albuquerque, New Mexico, November, 1983.

1.4.5 L. D. Bustard, The Effect of LOCA Simulation Procedures on Cross-Linked Polyolefin Cable's Performance, NUREG/CR-3588, SAND83-2406, Sandia National Laboratories, Albuquerque, New Mexico, April, 1984.

1.4.6 W. H. Buckalew, F. J. Wyant, and G. J. Lockwood, Response of Rubber Insulation Materials to Monoenergetic Electron Irradiations, NUREG/CR-3532, SAND83-2098, Sandia National Laboratories, Albuquerque, New Mexico, November, 1983.

1.4.7 W. H. Buckalew and F. V. Thome, Equipment Qualification Research Tests of Duke Power/D. G. O'Brien Type K Instrumentation Penetration Assembly, RS4445/81/03, Sandia National Laboratories, Albuquerque, New Mexico, March 1982. Limited Distribution; obtain through W. Farmer, USNRC.

International and Other U.S. Programs

1.4.8 D. T. Furgal, E. A. Salazar, and C. M. Craft, Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests, NUREG/CR-3863, SAND84-1264, Sandia National Laboratories, March 1985. (CAP Program results.)

A. D. T. Furgal, C. M. Craft, Performance Assessment of Class 1E Pressure Transmitters Subjected to Environmental Stresses, SAND84-1655C. Presented at the 12th USNRC Water Reactor Safety Research Information Meeting, October 22-26, 1984, Gaithersburg, Maryland. (CAP Program results.)

1.4.9 M. Ito, et al, Progress on Qualification Testing Methodology Study of Electric Cables, NUREG/CP-0058 Vol. 5, Proceedings of the USNRC Twelfth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 22-26, 1984.

1.5 Beta/Gamma Radiation Effects

Issue Description

To be able to adequately test for equipment qualification under the radiation component of the accident environment, one requires a definition of the source term and an understanding of the adequacy of approximating the radiation source term by current simulation techniques. The on-going NRC review of the source term definition is recognized in Regulatory Guide 1.89 (June 1984) and 10 CFR 50.49 and it centers on the issue of fission product releases and their mechanistic distributions and time histories. In the former for example, the "...dose rate, energy spectrum and particle type..." must be considered.

Current conventional simulator techniques are based on gamma sources with a fixed spectrum, most often Cobalt-60. The ability of these sources to simulate the damage in Class 1E, safety-related, materials and equipment from the expected Beta/Gamma real source term requires assessment. Regulatory Guide 1.89 (June 1984) addresses parts of this subissue:

a) Section C.3.f "Cobalt-60 or Cesium-137 would be acceptable gamma radiation sources for environmental qualification."

b) Section C.2.(c).6 "Shielded components need be qualified only to the gamma radiation environment..."

The radiation signature is known to be a complex function of time, emission particle type (beta/gamma), and intensity, and is highly dependent upon such factors as the assumed fission product release fractions, and their distribution in containment. However, given that dose rate and dose are scaling factors, to a first approximation simulator adequacy can proceed on the bases of energy and particle-type dependency contained in the radiation signature. Regulatory Guide 1.89 (June 1984) contains a modified version of this approach by specifying three separate source terms (design basis LOCA, all other design basis accidents, Regulatory Guide 1.97 limiting conditions); these are essentially derived from different estimates of core degradation in a "best-estimate" signature.

This is the thrust of this effort to address the issue:

> "Can a gamma radiation source be used to simulate the mixed (accident) beta/gamma radiation?"

Radiation Signature Definition

In the 1977 time frame, Regulatory Guides 1.3, 1.4, and 1.7 (all are siting Standards) and proposed drafts of Regulatory Guide 1.89 only implicitly defined the LOCA radiation signature to be simulated in the qualification testing of electrical equipment. However, this information was not directly useable since it was specified in terms of (arbitrary)

fission product fractions and distribution assumptions; a translation of these into doses and rates was required. The earliest efforts as documented in the following reports used the preceding information and plant specific data to explicitly determine LOCA radiation signatures and to interpret the importance of these results as applied to qualification testing.

Reference 1.5.1, NUREG76-6521 (SAND76-0740), was the first report to define these radiation signatures. Notwithstanding the various assumptions (plant specifics, radiological decay, etc.), the results were significant because of the energy spectra and dose and rate magnitudes elucidated. Figures 1.5.1 - 1.5.4 summarize these spectra and magnitudes. The study contained, at least three significant results:

1. A definition of the expected magnitudes of gamma and beta dose and rates for a typical PWR containment; typical values of initial dose rate are 3.5 and 70 MR/hr and of (30-day) dose are 50 and 300 Mrad, respectively.

2. A formal recognition that the beta dose and rate are significantly greater (about 6 times) than the corresponding gamma values.

3. A preliminary definition of the energy spectras; the gamma and beta spectra exhibit a changing energy spectra with time, not typical of monoor fixed-energy simulators. For both, the

   spectra are initially "hard", then "soften" until a minimum at about 4 days post-LOCA then "reharden."

Reference 1.5.2, (SAND78-0090), incorporated later draft revisions of Regulatory Guide 1.89, best available beta spectra data (not available earlier), and evaluated the sensitivity of key parameters and assumptions in the Guide. Reference 1.5.3, (SAND78-0091), is a more readable summary of the previous effort and highlights the major findings of the study. The report included several hypothesized sources from the regulations and provided a basis for constructing even different sources based on the irradiation history, the treatment of daughter nuclides, and the effects of fissile loading. Generally, this work is significant (1) as the most detailed definition of the hypothesized LOCA-radiation signature to date, and (2) because the parametric evaluation of signature sensitivity indicates relatively little effect from varying most parameters over conservative or expectable ranges (Figures 1.5.5 and 1.5.6).

Perhaps more significant is the impact of this work as a comprehensive data base on LOCA source terms and related parameter studies. In that light, the following conclusions are relevant:

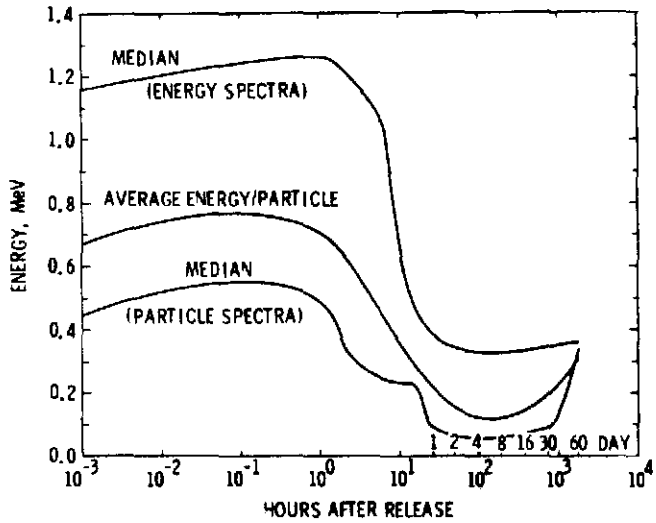


Figure 1.5.1 Measures of the gamma energy spectra for the airborne source; 100% noble gases, 25% iodines. (From Reference 1.5.1)

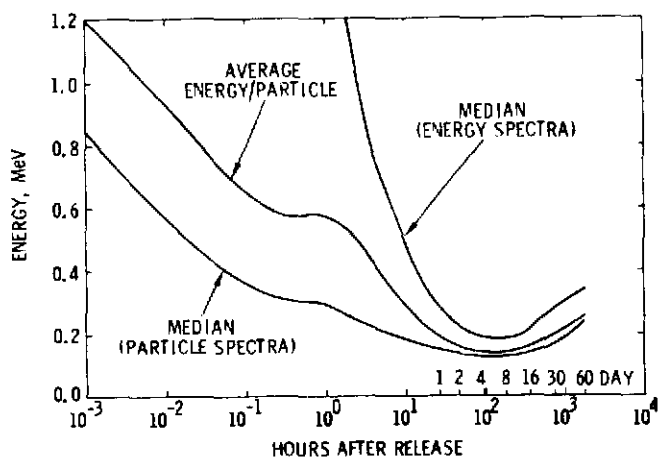


Figure 1.5.2 Measures of the beta energy spectra for the airborne source; 100% noble gases, 25% iodines. (From Reference 1.5.1)

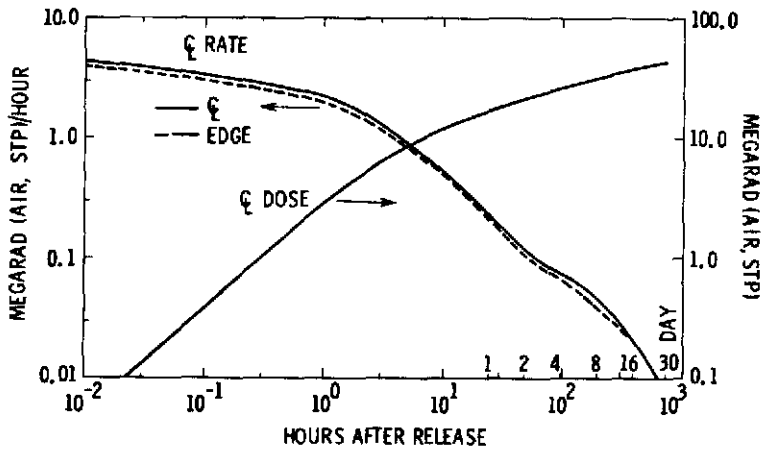


Figure 1.5.3 Rate and dose to air from the combined gamma sources; airborne source uniformly distributed in the containment volume, plate-out source uniformly distributed on the containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume. (From Reference 1.5.1).

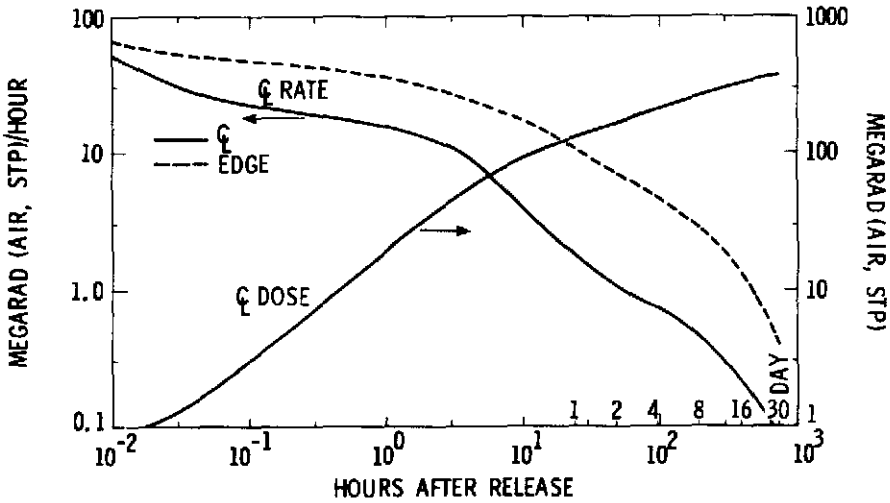


Figure 1.5.4 Rate and dose to air from the combined beta sources; airborne source uniformly distributed in the containment volume, plate-out source uniformly distributed on the containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume. (From Reference 1.5.1).

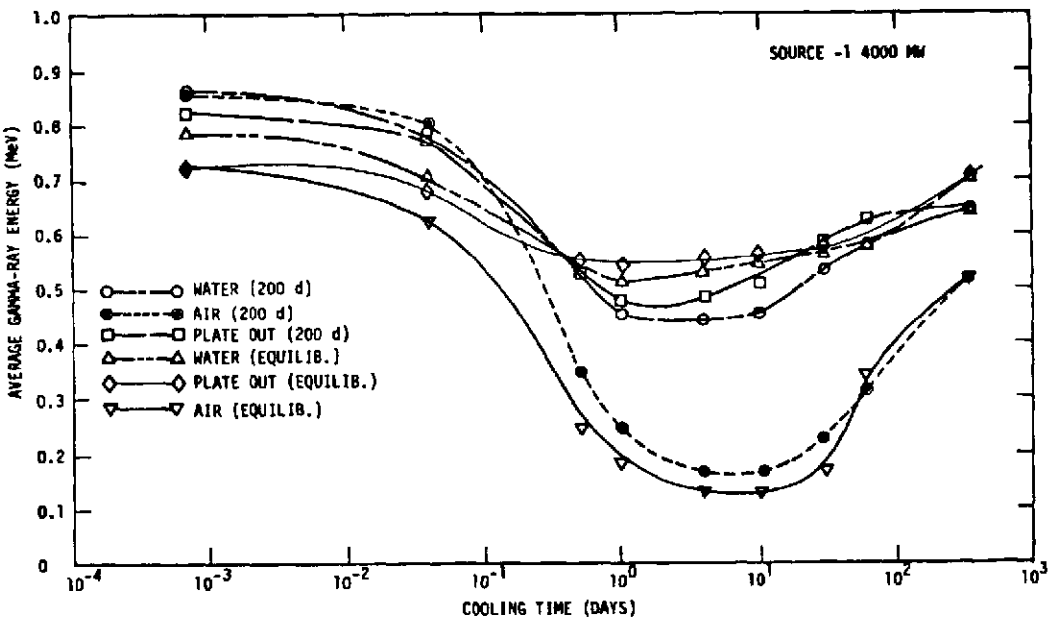


Figure 1.5.5 The average gamma spectrum energy as a function of cooling time for Source 1, 4000 MW, pure U-235, all daughters included (no depletion), equilibrium and 200 days. (From Reference 1.5.3)

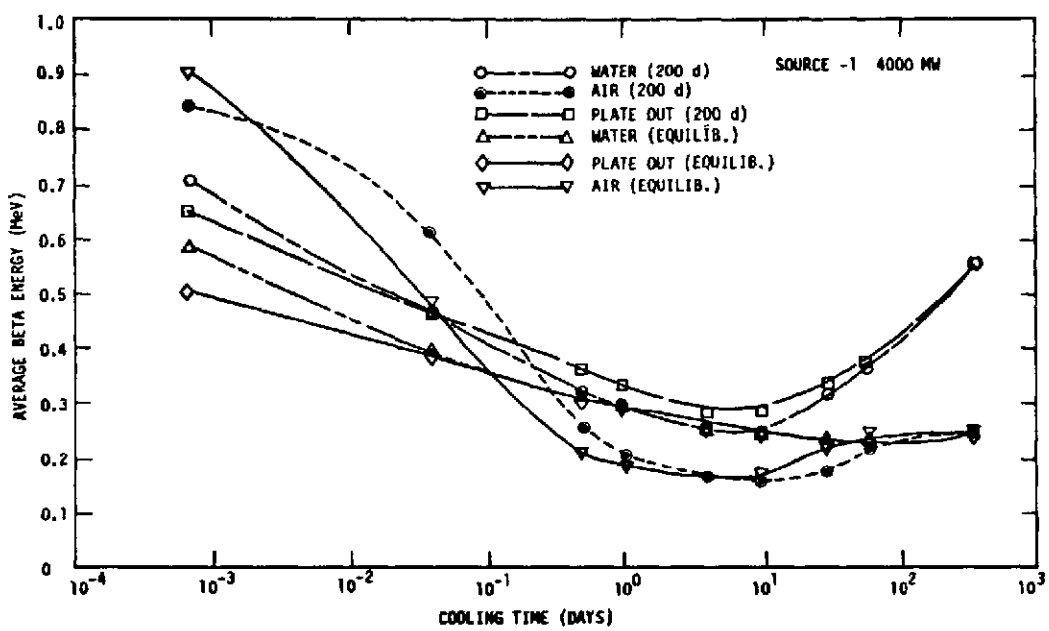


Figure 1.5.6 The average beta spectrum energy as a function of cooling time for Source 1, 4000 MW, pure U-235, all daughters included (no depletion), equilibrium and 200 days. (From Reference 1.5.3)

-- The range of parameters influencing the radiation signature suggests the need for greater preciseness in their specification to avoid variations in the resultant calculations by licensees.

-- It may be prudent to adopt a uniform source signature (from this work), as a replacement for nuclide fractionation specifications, to assure identical generic data base by licensees.

-- The greatest signature-influencing factor appears to be the original nuclide fractionation specifications; it would be useful to support these with experimental data base if possible, to avoid future respecification and subsequent influence on past radiation qualification programs.

-- Early assessment of beta radiation significance is important in evaluating radiation qualification programs.

-- With a generic data base established, the plant-specific dose and rate values can be accurately established within the bounds of plant-layout assumptions.

-- The evaluation of simulatory 'adequacy' can proceed from this generic data base.

-- An even more realistic representation of the LOCA radiation source could be constructed, by techniques similar to those used here, based on a prescribed accident sequencing as suggested in WASH-1400 and supported by ongoing experimental programs. The sequencing would account for the time-dependence and the magnitude of the releases. WASH-1400 provides estimates for the four categories of release (gap, meltdown, vaporization, and steam explosion fractions) and for the time (given as a range) of each release occurrence.

"Best-Estimate" Radiation Signature

The next phase of this effort introduced more realism into the accident-radiation release scenario using the WASH-1400 time and magnitude scenarios. That is to say that the release assumptions are more mechanistic in nature, rather than the earlier hypothesized scenarios based on arbitrary (siting) source terms.

Reference 1.5.4, NUREG/CR-1237 (SAND79-2143), was the first report of its kind to attempt this mechanistic approach. Figure 1.5.7 shows the time sequence assumed for each of the four mechanistic release components employed, and Figures 1.5.8 and 1.5.9 the gamma and beta energy release rates from the various contributors. The effects of the uncertainties in the release fraction as well as variations in the time dependence of the releases were investigated; a wide uncertainty band exists for the "best-estimate" releases. The study contained significant findings and recommendations:

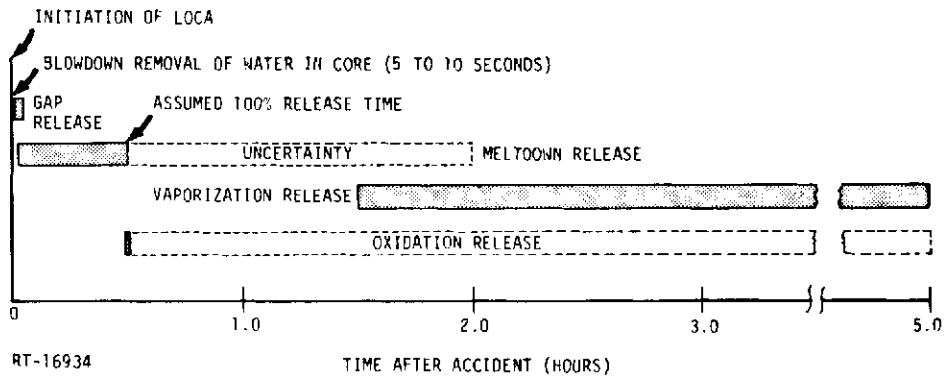


Figure 1.5.7 Time sequence of fission product release for an unterminated LOCA without emergency core cooling. (From Reference 1.5.4).

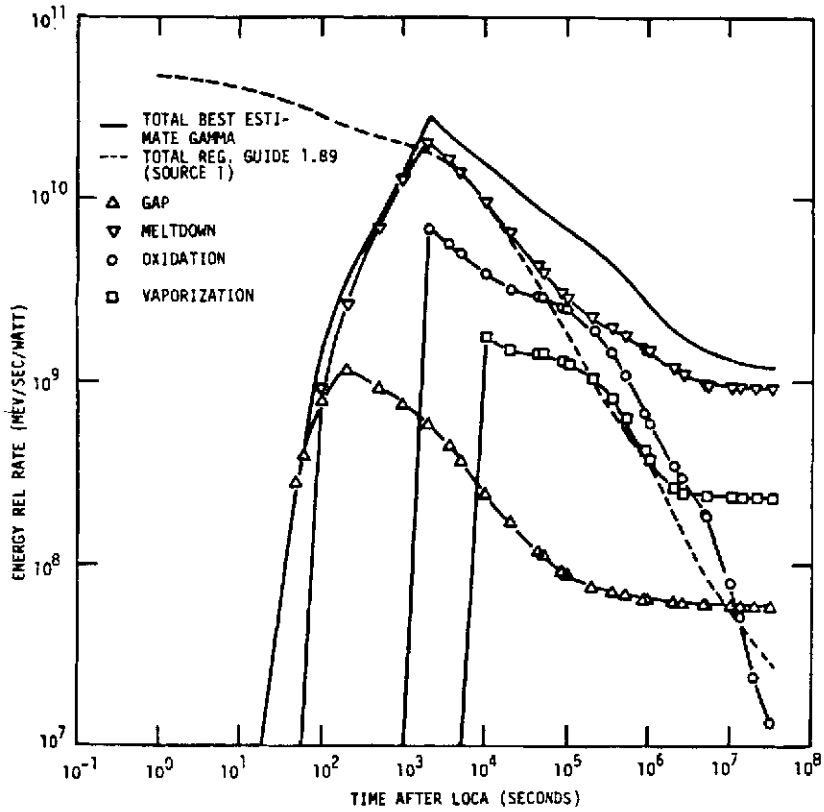


Figure 1.5.8 Gamma-ray energy release rate for "best estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result. (From Reference 1.5.4).

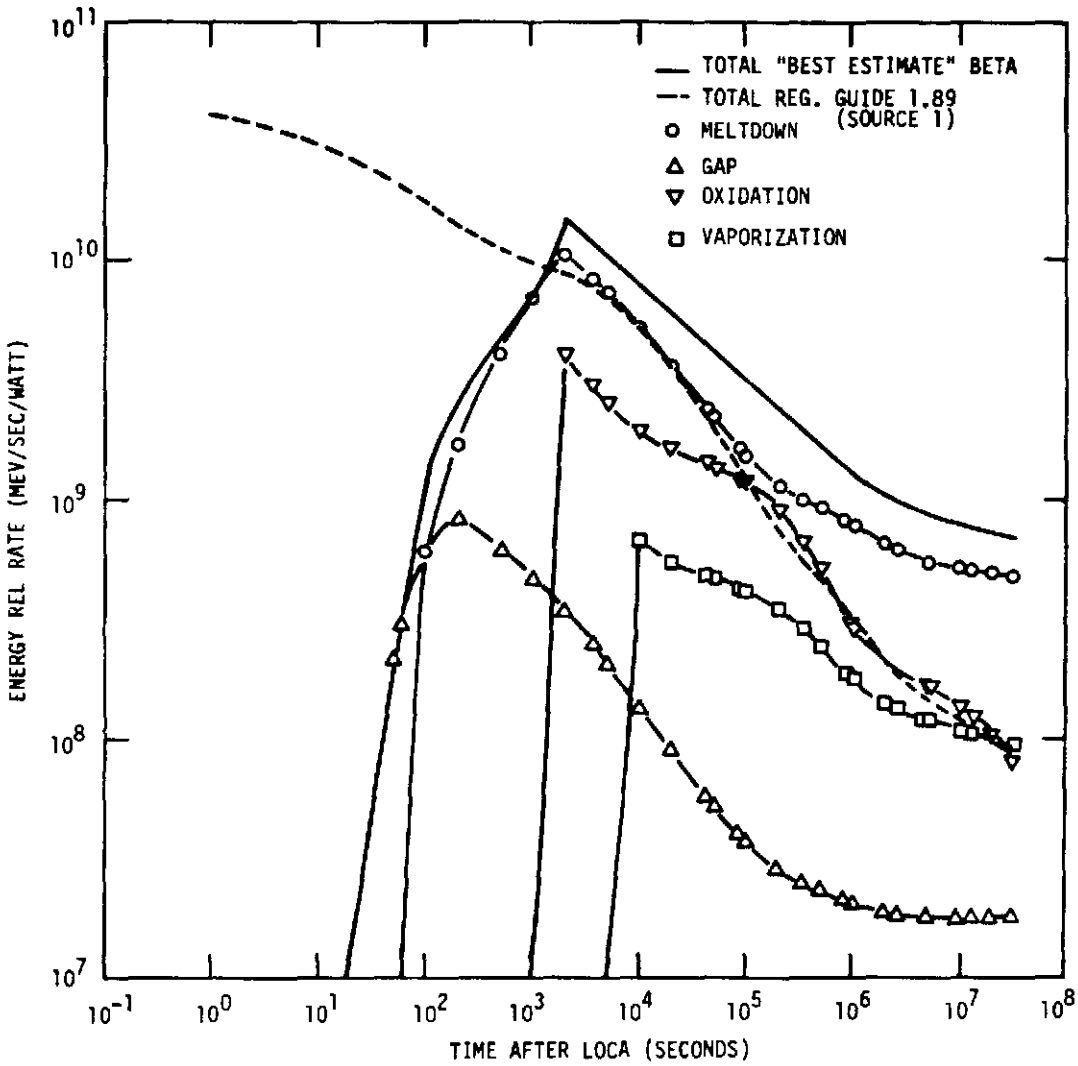


Figure 1.5.9 Beta energy release rate for "best-estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result. (From Reference 1.5.4).

- As far as the energy release rates are concerned, the dominant component is the meltdown term for all times greater than about 60 seconds after an unterminated LOCA without emergency core cooling. Except for the time interval 30-60 seconds, the gap release is the least significant. In an accident that is successfully terminated (i.e., does not result in significant core degradation), the gap release would be the only fission product source.

- Comparisons of the total best-estimate source with the total Regulatory Guide 1.89 source (Source-1) show that the latter source is very much larger and decreasing until about 1000 seconds (about 15 minutes) after the accident. At this time the best-estimate energy release is larger and remains larger for all later times.

- Spectrum calculations for the best-estimate sources were made analogous to the spectra generated for the Regulatory Guide sources. The behavior for the two cases is generally similar, including the behavior of the average particle energy.

Reference 1.5.5, NUREG/CR-2367 (SAND81-7159), was an update of the earlier study intended to incorporate new fission product release research. Of the four release stages (i.e., gap, meltdown, oxidation, vaporization), the only new information was for gap release; this is shown in the following table:

Table 1.5.1 Old Best Estimate vs. New Best Estimate Gap Release Fractions (From Reference 1.5.5)

Fission Product Species	Old Total Gap Release Fraction	New Total Gap Release Fraction
Noble Gases (Xe,Kr)	0.03	0.20
Halogens (I,Br)	0.017	0.30
Alkali Metals (Cs,Rb)	0.05	0.25
Alkaline Earths (Sr,Ba)	0.000001	0.0
Tellurium Group (Te,Se,Sb)	0.0001	0.0

The effect is shown in Figures 1.5.10 and 1.5.11, on the gap release. The report demonstrates that further refinement of the "best-estimate" source term will come primarily from improvements in the meltdown release data.

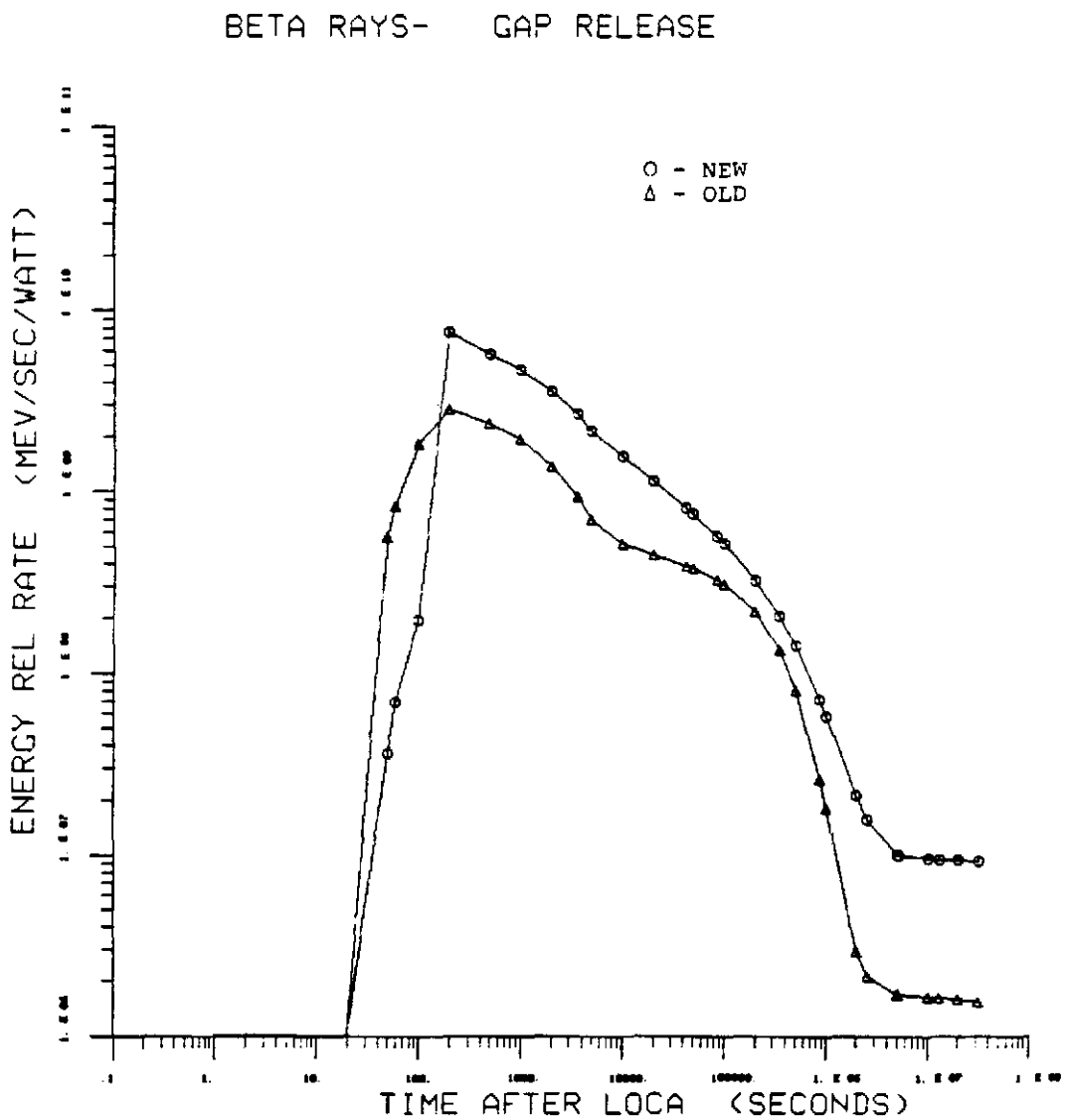


Figure 1.5.10 New total gap energy release rates vs. old, for betas. (From Reference 1.5.5).

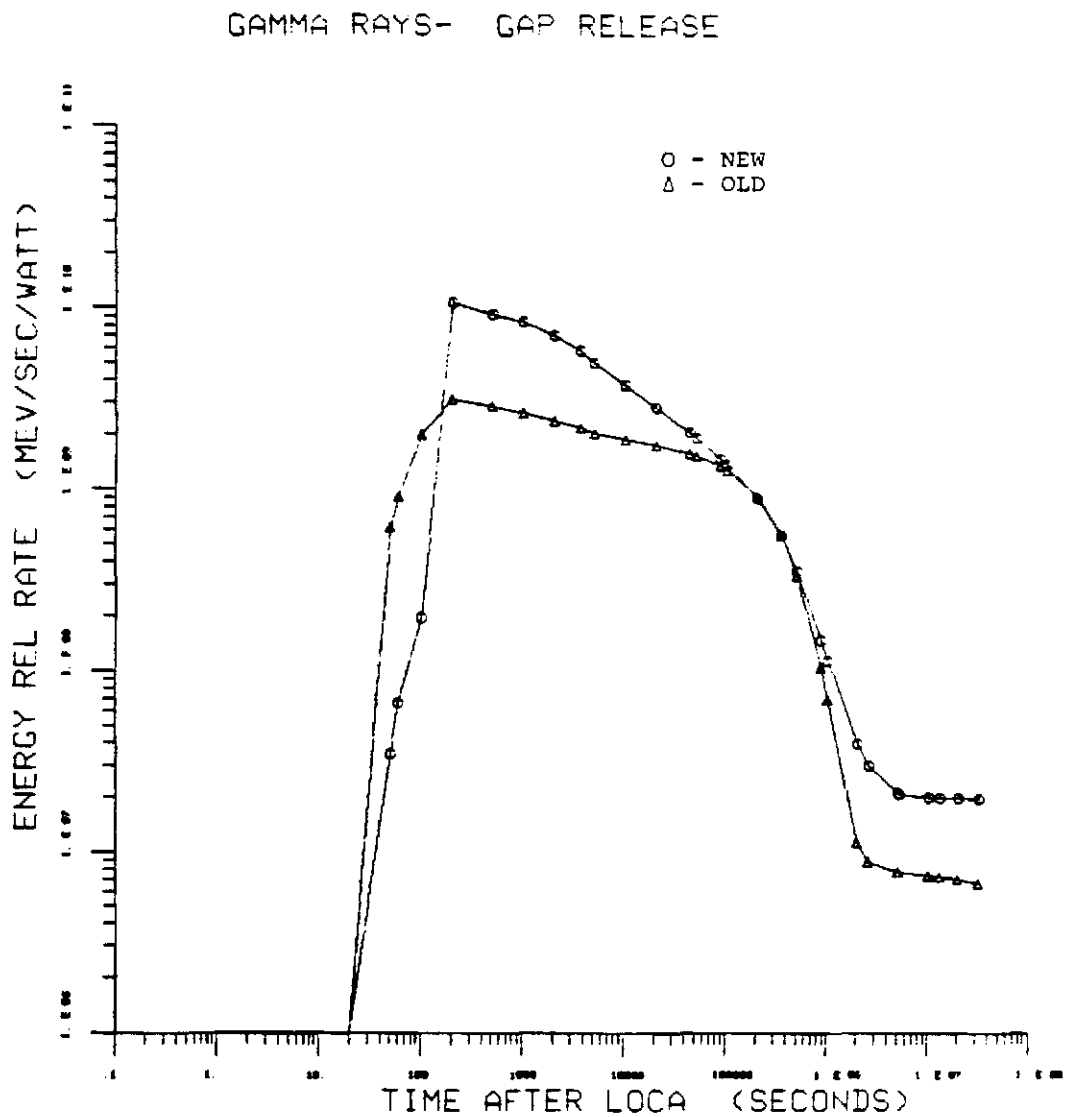


Figure 1.5.11 New total gap energy release rates vs. old, for gammas. (From Reference 1.5.5).

Reference 1.5.6, NUREG/CR-2593 (SAND82-7013), and Reference 1.5.7, NUREG/CR-2594 (SAND82-7014), document the computer codes used in these "best-estimate" studies. RIBD/IRT calculates isotopic concentrations from various power histories and outputs time-dependent inventories, activities, decay powers, and energy releases for 800 fission products. GABAS, gamma and beta spectrum calculates the time-dependent beta and/or gamma spectra from (decaying) fission products. Both codes are available from the Radiation Shielding Information Center (RSIC).

Simulator-Adequacy Investigations

Following on the source definition work, the effort proceeded into the determination of unique failures or degradation of materials and equipment which could not be adequately simulated by standard gamma radiation simulators.

Reference 1.5.8, NUREG/CR-1184 (SAND79-1787), was the first effort in this series, an analytical evaluation of simulator adequacy. The concentration was on polymeric materials, which might typically be exposed to the LOCA radiation. Depth-dose and charge deposition, induced electrical signals, bulk temperature effects, bulk polymer degradation, and differential mechanical stress in insulating materials were examined.

Figures 1.5.12 and 1.5.13 summarize the depth-dose into a typical electrical cable; these illustrate the significance of beta penetration (compared to the gamma simulators). Important results include the following: LOCA radiation-conductivity changes and electrical-field buildup should have negligible effects on electrical cable's function; bulk temperature effects (and surface temperature effects) could be significant with dose rate (Figure 1.5.14), but is mostly independent of the radiation signature (i.e., LOCA or simulator); given equal doses at a point in the material, bulk polymer damage is not a function of the depositing particle (i.e., gamma or beta); differential mechanical stress is a direct result of the penetrability of the beta; i.e., the stress arises from a severely damaged surface adjacent to a relatively undamaged interior section, however, this would conceivably occur only after very severe damage and may not be of practical concern.

Reference 1.5.8, NUREG/CR-1184 (SAND79-1787), makes these conclusions and recommendations:

1. In many cases, the standard gamma-radiation simulators can adequately duplicate the damage mechanisms and damage to materials used in safety-related equipment that result from the postulated nuclear-plant ambient and accident radiation environments.

2. One recommendation is to overstress the equipment/material everywhere to greater total dose than expected from the combined LOCA-radiation signature; dose rates should also approximate the expected (combined) rates. However, other logical data-based techniques (e.g., averaged dose and rates) may also be acceptable.

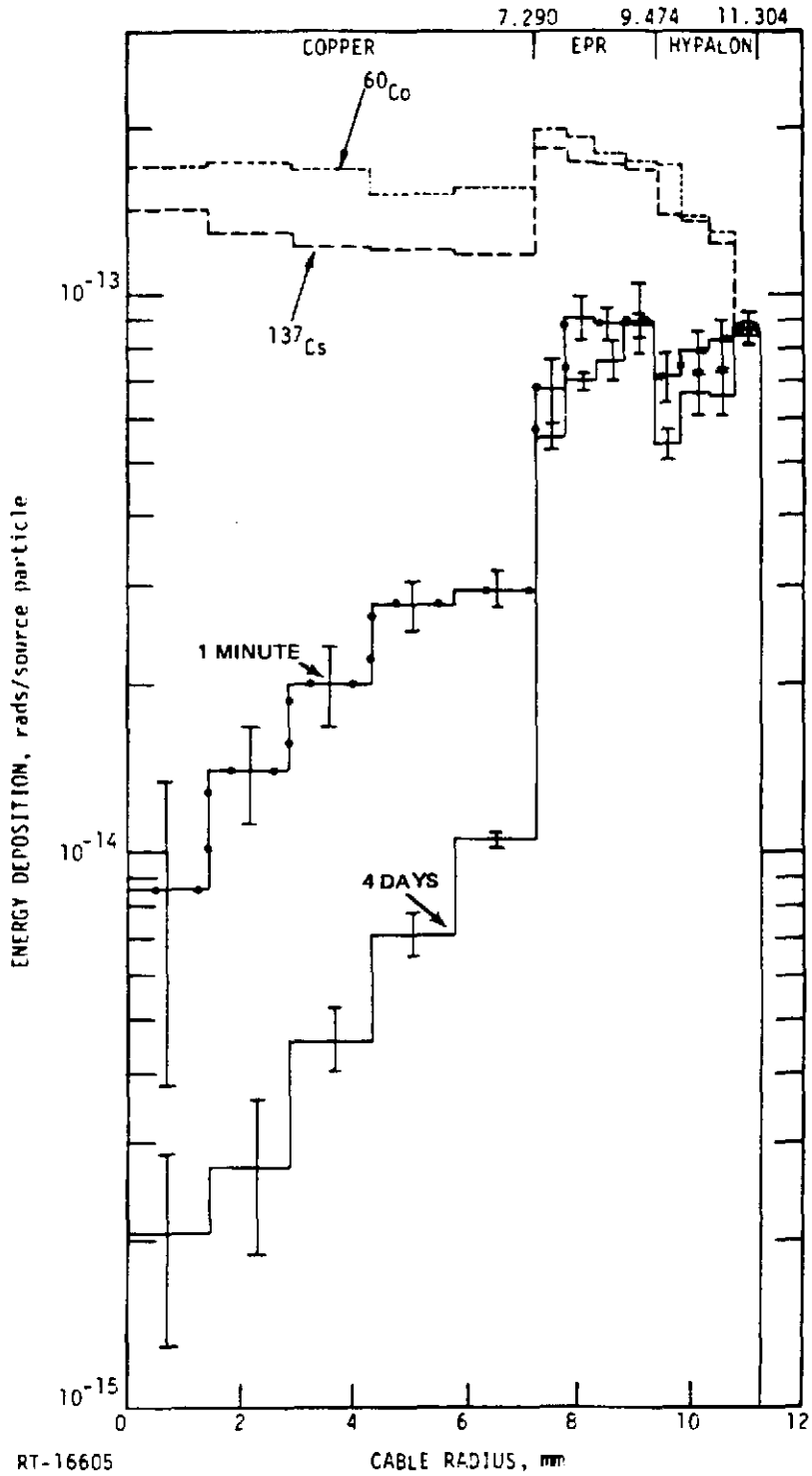


Figure 1.5.12 Dose vs. Penetration Depth - ^{60}Co, ^{137}Cs, and Source 1 (Airborne) Gamma Rays onto Typical Cable Configuration. The isotopic source results have been normalized at the outer zone. (From Reference 1.5.8).

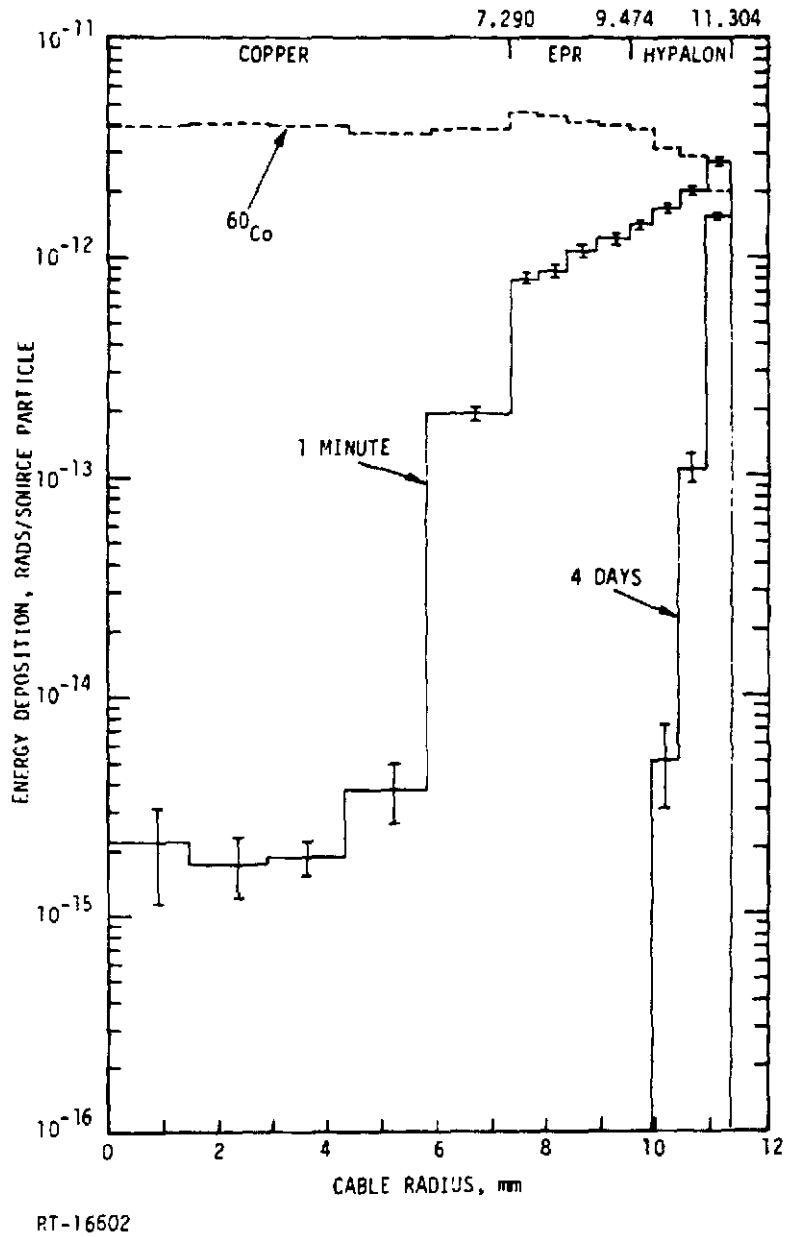


Figure 1.5.13 Dose vs. Penetration Depth - ^{60}Co and Source 1 (Airborne) Betas Onto Typical Cable Configuration. The isotopic source results have been normalized at the outer zone. (From Reference 1.5.8).

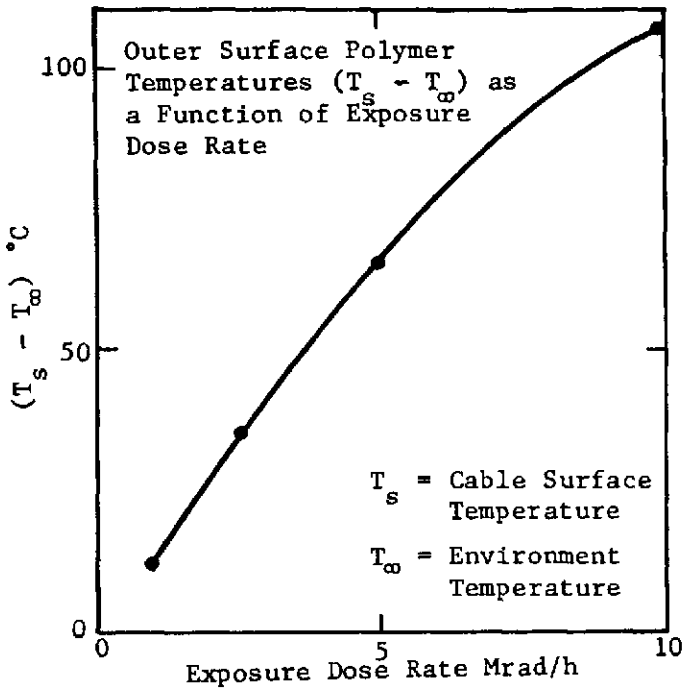


Figure 1.5.14 Outer Surface Polymer Temperature vs. Exposure Dose Rate. (From Reference 1.5.8).

In summary, we have seen no evidence of unique damage mechanisms in exposed organic materials that demand unique radiation-simulation techniques; neither can radiation be arbitrarily applied to the test item without consideration of the complete qualification program.

However, the study stated that not all aspects of simulator adequacy evaluation are complete. The important remaining topics include:

- Equivalence of beta/gamma and or neutron/gamma (for aging environments) ratios in bulk degradation. That is, are peak doses/rates appropriate, or are averaged values acceptable?

- Charge breakdown resulting in transient noise and/or permanent damage should be evaluated for selected equipment.

- Secondary emissions, magnitudes and effects, must be considered.

- Simulator adequacy evaluations should be extended to more complex and specific equipment items.

Reference 1.5.9, NUREG/CR-2581 (SAND82-0449), addresses the effects of electrons slowing down in materials that enclose radiation sensitive components and materials; this contribution of photons (bremsstrahlung) has not generally been considered, but may contribute significantly to the total transmitted (primary and secondary) photon component. A parametric set of calculations were performed with typical results shown in Figure 1.5.15. The conclusion is that secondary emissions need not be considered; they probably only constitute a few percent increase to the total dose to material or equipment from primary radiation passing through shielded enclosures.

Reference 1.5.10, NUREG/CR-3532 (SAND83-2098), reports on investigations of charge breakdown from beta irradiations of EPR insulation material slabs and typical electrical cables. Parameters include air/vacuum, electron energy, and surface termination. Under certain conditions charge was accumulated and spontaneous breakdown did occur during irradiation in the vacuum environment; however, no evidence of breakdown was observed during ambient air exposures. Based on these experiments, it is concluded that electron charge buildup and breakdown is not apt to occur in EPR rubber insulation exposed to electrons from a LOCA radiation environment provided that the insulation is in contact with an ionized medium. With some additional effort, the extent of the generic applicability of this conclusion can be assessed.

Reference 1.5.11, NUREG/CR-3777 (SAND84-0912), discusses a method used to simulate the effects of LOCA electron environments on large-area material and cable samples. It was demonstrated that both electron dose/dose rates and energy spectra were adequately simulated with large diffuse monoenergetic beams.

The QTE program, in part, is concerned with the adequacy of isotopic photon sources (e.g., Co-60) to simulate the combined beta-gamma radiation environment accompanying a LOCA. As part of this study on simulator adequacy, SNL is investigating the dose-damage equivalence of gamma and beta radiations on organic materials. The results of this study may help determine the gamma dose required in a test to adequately simulate the beta radiation effects expected during an accident.

The beta-gamma equivalence study is a two-part effort:

(1) a multi-year joint NRC/French/SNL international cooperative test program, including a study of any synergistic effects associated with a mixed (beta/gamma) radiation field, and

(2) a complementary test program at SNL to screen for major effects and test parameters, followed by specific comprehensive experiments.

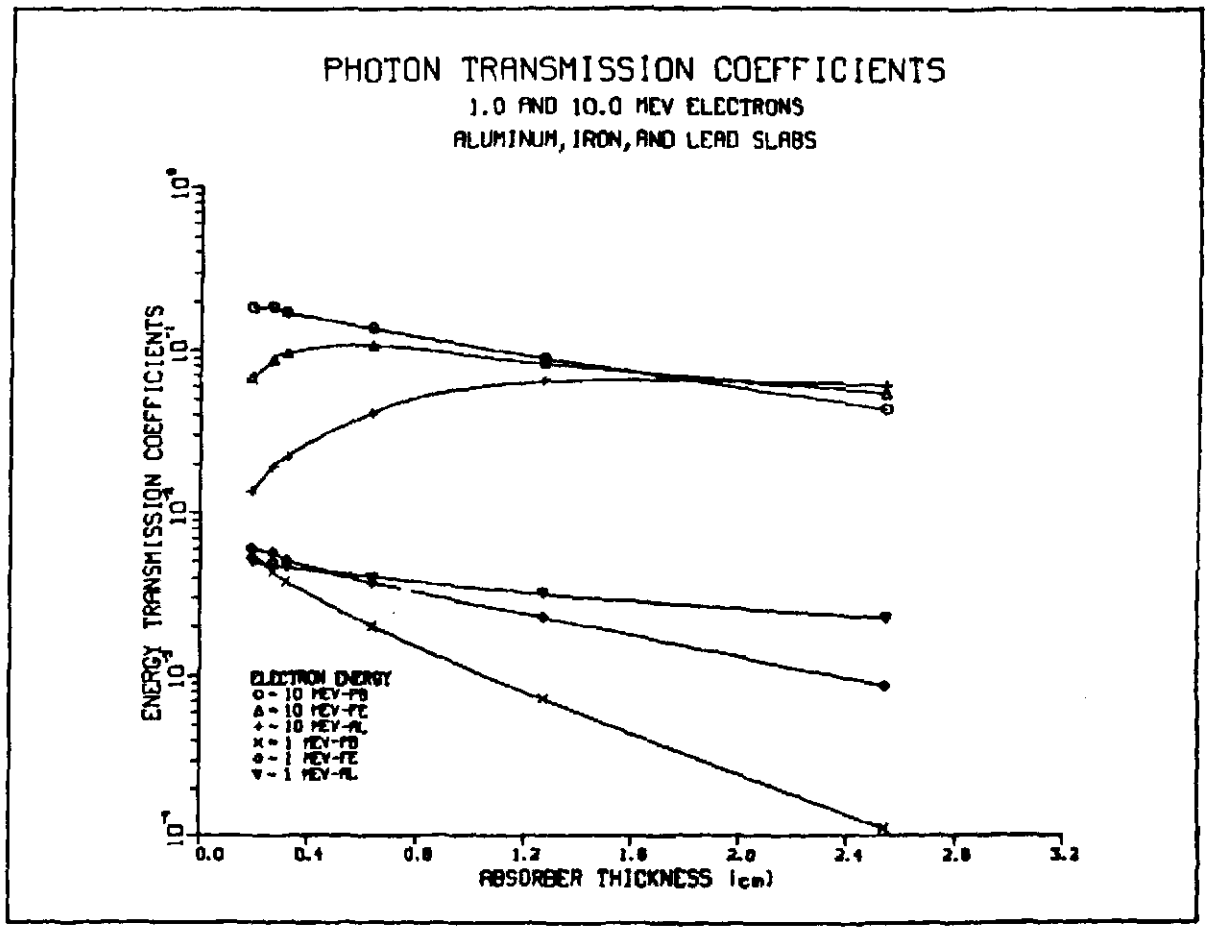


Figure 1.5.15 Photon Transmission Coefficients $\frac{E_{TR}}{E_o}$ for aluminum, iron, and lead slabs of various thicknesses. (From Reference 1.5.9).

Reference 1.5.12, NUREG/CR-4530/1 of 4 (SAND86-0366/1 of 4), discusses the results of the Phase-1 test series completed as part of the U.S./French test program. The purpose of this test series was to cross-correlate the irradiation conditions and postirradiation measurement techniques, and normalize the results obtained by the U.S. research group (SNL) to the French research group (Compagnie ORIS Industrie).

The Phase-1 tests considered one material type (EPR -- one U.S. formulation and one French formulation), one material thickness (1-mm), one electron energy (1.0 MeV), one gamma source (Cobalt-60), one exposure dose (15 Mrad), and one dose rate (1 Mrad/h). The irradiated samples were cut in half and then measured (in both countries) for changes in tensile strength, ultimate elongation, hardness, and density, using standard testing techniques.

In addition, dosimetry samples were exchanged and exposed in each electron and gamma facility. The conditions of exposure were the same for each dosimeter (1.0 MeV electrons or Co-60 gammas, 1 Mrad/h and 2 Mrad total dose). After exposure, the dosimeters were returned to the country from which they originally came and evaluated.

The Phase-1 dosimetry results indicate excellent correlation between the two countries regarding radiation exposure conditions and dosimetry measurement techniques. The Phase-1 EPR test results indicate that consistency is achievable between the two countries for the conditions of material exposure and postirradiation testing techniques employed (see Figure 1.5.16).

The Phase-2 test series of this program, to study the gamma damage equivalence of beta radiation, is currently underway.

Reference 1.5.13, NUREG/CR-4543 (SAND86-0462), discusses the first results obtained by the complementary beta-gamma equivalence test program at SNL. The purpose of these tests was to attempt to identify the degree to which factors, such as

1. differences in energy deposition profiles between electrons and photons,

2. differences in energy deposition for each particle type, and

3. differences in radiation-induced damage mechanisms (cross-linking, charge build-up/break-down, etc.)

might influence the dose-damage equivalence in certain organic materials.

These tests considered one EPR formulation in slabs of three different thicknesses (1, 1.5, and 2-mm), exposed to Co-60 or accelerated electrons at energies spanning the range between 0.235 and 0.85 MeV. In each case, the samples were irradiated at 2 Mrad/h to 10 Mrad. Following exposure

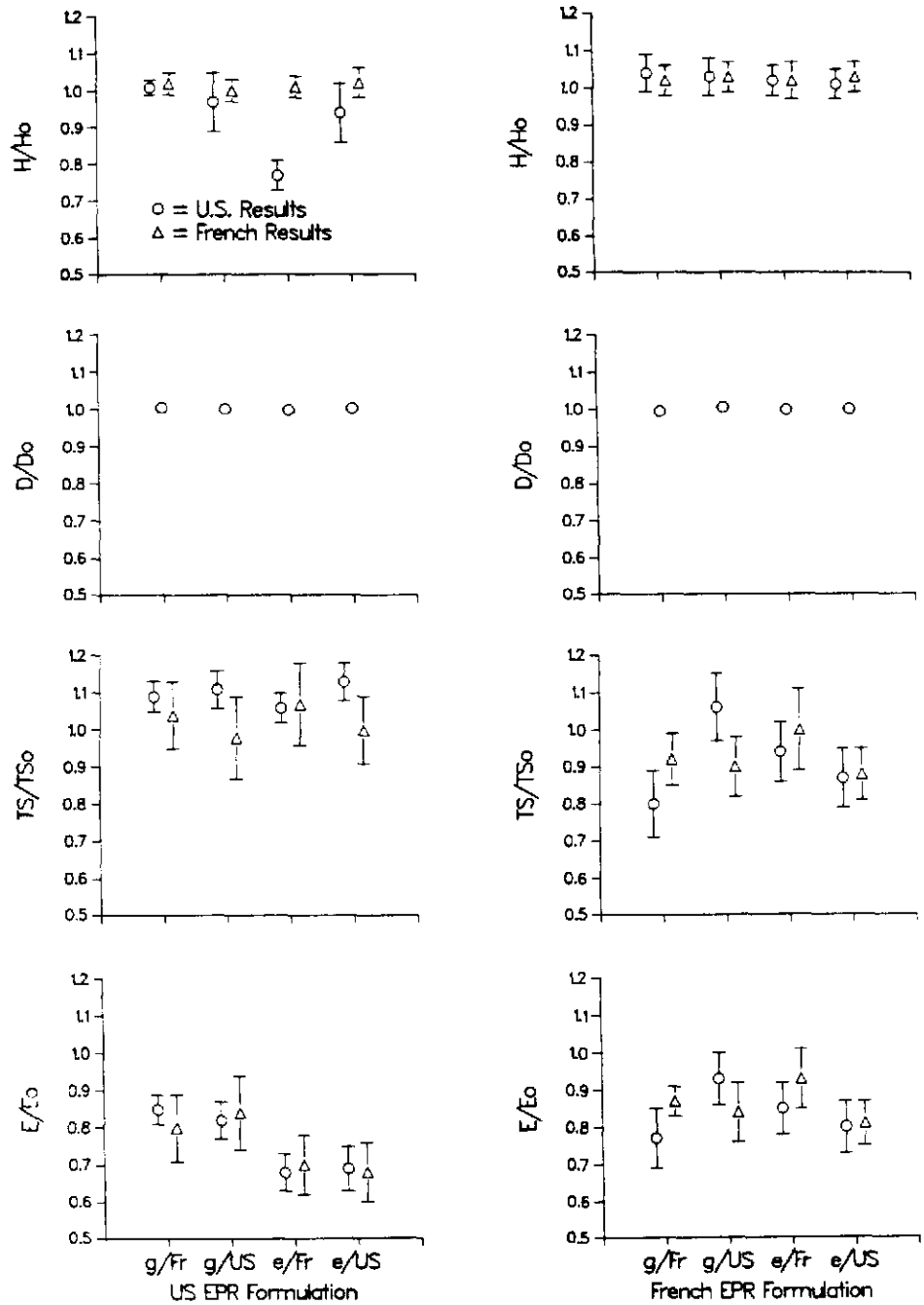


Figure 1.5.16: Comparison of French and U.S. Phase-1 Postirradiation EPR Test Results. The x-axis for each EPR formulation indicates the irradiation particle type (g = Co-60 gammas, e = 1.0 MeV electrons) and the country of exposure. Pairs of points are shown to indicate that each EPR sheet was irradiated as indicated, split in half, and tested separately in both countries. (From Reference 1.5.12).

the samples were cut into tensile specimens, then tested for changes in ultimate elongation and tensile strength.

Using this technique, it was observed that material damage indicators were smoothly varying functions of incident electron average energy, total absorbed energy, and front surface dose. In all instances photon induced material changed tracked with the electron values; this is in agreement with the concept of photon-electron damage equivalence. Combined electron and photon date demonstrated that material damage, as indicated by changes in elongation and tensile strength, was a slowly varying function of particle energy, absorbed energy, and front surface dose. Material thickness data indicated that, for the energies and thicknesses considered, the energy deposition distribution within the sample was not significant; rather, damage was a function of total energy absorbed only. Photon-electron relative effectiveness data (see Figure 1.5.17), derived from the analysis of elongation and tensile strength information, predicted that photon to electron equivalence was a linear function of incident electron energy and that incident particle energy, absorbed energy, and front surface dose were equally dependable estimations of photon and electron equivalence.

Additional tests at other dose rates and electron energies are currently underway.

International and Other U.S. Programs

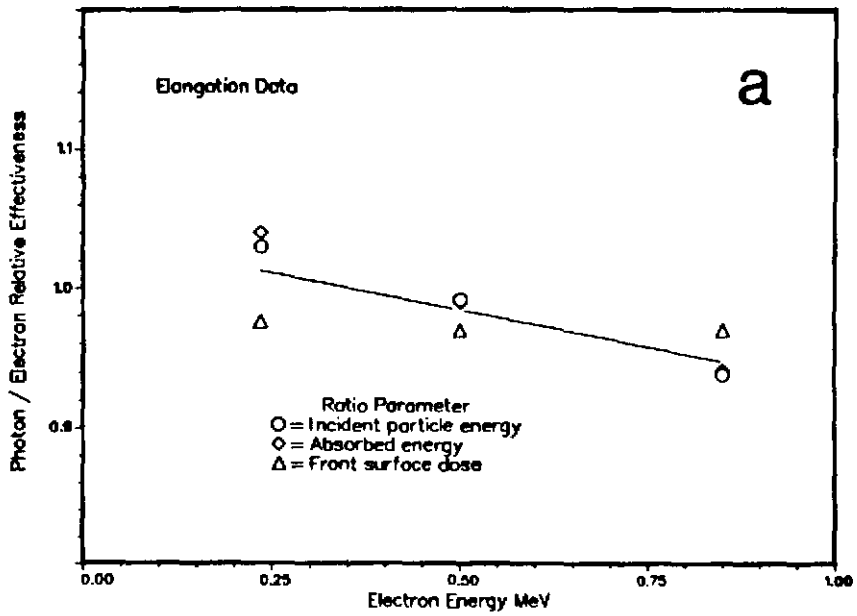
A limited beta-effects research effort was undertaken in France prior to the current U.S./French cooperative test program. Reference 1.5.14 (ORIS/LABRA-036) describes an investigation of the comparative effects of beta and gamma radiation on polymer materials, and the sequencing effects (referred to as "synergistic effects" in the report) of gamma followed by beta radiation. Unaged radiation cross-linked polyethylene and chlorosulfonated polyethylene samples were exposed to gamma, beta, or gamma followed by beta radiation then tested to determine the changes in tensile strength and elongation at break as a function of dose.

The report concludes that gamma and beta radiation appear to have identical effects on the mechanical properties tested, although the gamma radiation shows a slight tendency to be more damaging. Additionally, sequencing effects do not appear to be substantial. The authors recognize the limited nature of this study and recommend additional tests using aged samples as well as other materials.

We are not currently aware of any other significant related activities.

Issue Findings

By performing radiation tests on identical test samples, with radiation type (i.e., betas or gammas) as a parameter, the failure or degradation modes unique to beta and gamma irradiations can be evaluated.



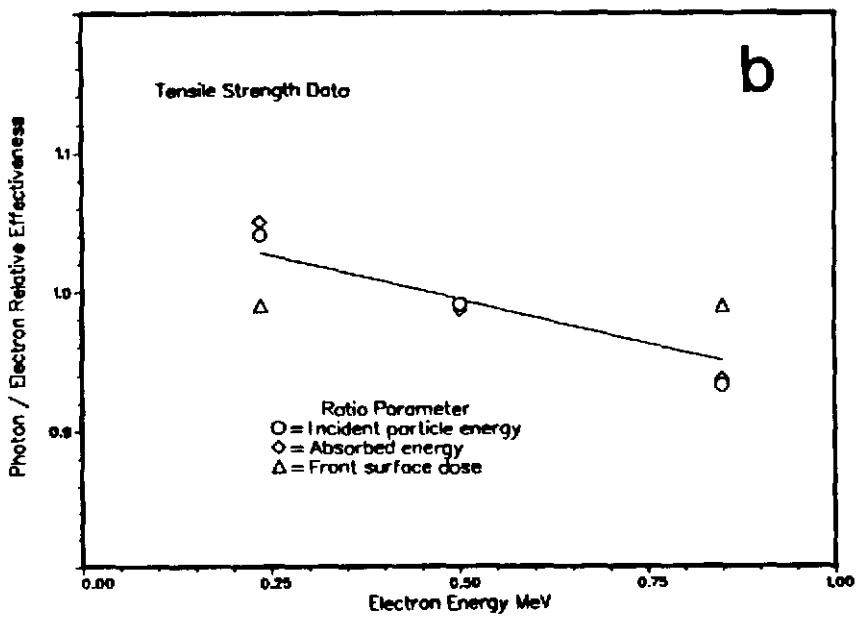


Figure 1.5.17 Photon to Electron Relative Effectiveness Ratios as a Function of Electron Energy, based on (a) Elongation Data and (b) Tensile Strength Data. (From Reference 1.5.13).

Radiation Signature Definition

Certain findings within this subissue have been incorporated directly into industry practice. For example in some cases, beta simulators are being used in equipment qualification test programs and/or betas are being considered in the overall qualification program.

In summary then, the findings within this subissue are:

1. Typical (PWR containment) gamma and beta peak dose rates are 3.5 Mrad/h(γ) and 70 Mrad/h(β) with integrated (30-day) doses of 50 Mrad(γ) and 300 Mrad(β).

2. Beta doses and rates are significantly greater (about 6 times) than the corresponding gamma values.

3. Gamma and beta energy spectra show significant time dependence (not typical of monoor fixed-energy simulators).

4. The range of parameters influencing the radiation signature suggests the need for greater preciseness in their specification.

5. It may be prudent to adopt a uniform source signature as a replacement for nuclide fractionation specifications, to assure the use of an identical generic data base.

6. The greatest signature-influencing factor appears to be the original nuclide fractionation specifications.

7. Assessment of beta radiation significance is important in evaluating radiation qualification programs.

8. With a generic data base established, the plant-specific dose and rate values might be accurately established within the bounds of plant-layout assumptions.

9. A realistic representation of the LOCA radiation source could be constructed, based on a prescribed accident sequencing as suggested in WASH-1400 and supported by ongoing experimental programs.

"Best-Estimate" Radiation Signature

Extensive research continues on defining fission product release fractions (and transport).

In summary then, the findings within this subissue are:

1. As far as the energy release rates are concerned, the dominant component is the meltdown term for all times greater than about 60 seconds after an unterminated LOCA without emergency core cooling. Except for the time interval 30-60 seconds, the gap release is the

least significant. In an accident that is successfully terminated (i.e., does not result in significant core degradation), the gap release would be the only fission product source.

2. Comparisons of the total "best-estimate" source from a previous SNL study with the total Regulatory Guide 1.89 source (Source-1) show that the latter source is very much larger and decreasing until about 1000 seconds (about 15 minutes) after the accident. At this time the "best-estimate" energy release is larger and remains larger for all later times.

3. Spectrum calculations for the "best-estimate" sources were made analogous to the spectra generated for the Regulatory Guide sources. The behavior for the two cases appears to be similar, including the behavior of the average particle energy.

4. Further refinement of the "best-estimate" source term may come primarily from improvements in the meltdown release data.

Simulator-Adequacy Investigations

In summary then, the findings within this subissue are:

1. In many cases, the standard gamma-radiation simulators can adequately duplicate the damage mechanisms and damage to materials used in safety-related equipment that result from the postulated nuclear-plant ambient and accident radiation environments.

2. One recommendation is to overstress the equipment/material everywhere to greater total dose than expected from the combined LOCA-radiation signature; dose rates should also approximate the expected (combined) rates. However, other logical data-based techniques (e.g., averaged dose and rates) may also be acceptable.

3. We have seen no evidence of unique damage mechanisms in exposed organic materials that demand unique radiation-simulation techniques; neither can radiation be arbitrarily applied to the test item without consideration of the complete qualification program.

4. Secondary emissions (from betas penetrating metal enclosures, for example) need not be considered; they probably only constitute a few percent increase to the total dose to material or equipment from primary radiation passing through enclosures.

5. Electron charge buildup and breakdown is not apt to occur in EPR rubber insulation exposed to electrons from a LOCA radiation environment provided that the insulation is in contact with an ionized medium.

6. Particle energy, absorbed energy, and front surface dose may be equally dependable estimations of photon and electron damage equivalence for some organic materials.

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1.6 Oxygen Effects

Issue Description

Oxidation has long been known to be responsible for the deterioration of polymers. Since many polymer materials are important components of safety-related equipment, oxygen effects during qualification accident simulations may be important to equipment degradation. For non-inerted containments and at the start of a "real accident" there will be approximately one atmosphere of air trapped in the containment. Oxygen will therefore be available at the start of and possibly throughout the accident for reaction with oxidation-sensitive materials. (It is currently unclear how the oxygen content inside containment will vary during realistic accidents.) However, LOCA simulations in the U.S. are sometimes done with significant or total oxygen depletion. This is because some steam generators and autoclave apparatus effectively sweep out the oxygen at the initiation of the experiment or remove it in dissolved form in the condensed steam which is withdrawn from the autoclave.

Current equipment qualification test methods do not always specify the need to consider oxygen effects during LOCA simulations. For example, IEEE Std. 323-1974, Table A2, has a footnote stating that " ...it is acceptable...to follow the temperature profile and allow the pressure to conform to saturated conditions..." The justification for allowing this procedure is the assumption that temperature is the more important parameter and the increased pressure will, if anything, increase the severity of the test. However, the saturated steam conditions allowed by this assumption eliminate oxygen presence during the LOCA simulation. Hence this assumption neglects possible oxygen effects during accident simulations.

These issues are not directly addressed by regulatory guidance. For example, 10 CFR 50.49, section e.7 requires synergistic effects to be considered when these effects are believed to have a significant effect on equipment performance. Regulatory Guide 1.89, revision 1 (June, 1984) states: "If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program." However, Regulatory Guide 1.89 continues: "Synergistic effects known at this time are dose rate effects and the effects resulting from the different sequence of applying radiation and (elevated) temperature." Oxygen effects during accident simulations is not explicitly identified by Regulatory Guide 1.89 as a known synergistic effect.

This is the thrust of this effort to address the issue:

> "How does the presence of oxygen affect accident-simulation test results?"

Program Effort

Reference 1.6.1, NUREG/CR-1749 (SAND80-1307), identified the concern that oxygen presence during LOCA simulations may be important for qualification of safety-related equipment. The reference identified the French KALI facility as having the capability to independently provide compressed air to LOCA test chambers during accident simulations. A cooperative U.S.-French test program to assess the importance of oxygen during LOCA simulations was proposed.

Reference 1.6.2, NUREG/CR-2763 (SAND82-1071), presents the results of the cooperative U.S.-French test program proposed by the previous reference. During this test program, nine materials commonly used as electric cable insulations and jackets in nuclear power plant applications were monitored in terms of mechanical properties, weight increases, solubility measurements, and infrared spectroscopy. Three different LOCA simulation tests were employed. The only planned difference among the three tests was the composition of the overpressure gas (gas pressure was constant for the three tests) that was present in addition to the steam environment. The three gas compositions were nitrogen (0% oxygen), 10% oxygen, and air (21% oxygen).

Reference 1.6.2 indicated that the oxygen concentration during a LOCA simulation may have a significant influence on the chemical and physical changes occurring in certain polymeric materials. It noted that the chloroprene material (See Figure 1.6.1) studied appeared to suffer significantly more damage in LOCA simulations containing oxygen, a conclusion which was valid even for samples which were not aged. For aged EPR insulation samples (See Figure 1.6.2), substantially decreased tensile properties resulted when oxygen was included in the LOCA simulations. Solubility, swelling, and Fourier Transform Infrared Spectroscopy results on these EPR materials indicated that increased oxidative scission was responsible for the enhanced degradation. For a PVC material (See Table 1.6.1), the results indicated that the damage became more severe as the oxygen content was lowered (slightly greater mechanical damage, substantially greater swelling). Reference 1.6.2 concludes that it is possible that a LOCA simulation without oxygen may underestimate the changes which occur in some materials, while overestimating the changes which occur in other materials when compared to a LOCA simulation with oxygen.

Accident oxygen effects on polymer materials is also discussed in Reference 1.6.3, NUREG/CR-4091 (SAND84-2291). This research effort examined oxygen effects as part of a comprehensive test program whose experimental variables included: (1) sequential versus simultaneous accelerated aging exposures; (2) the order of the sequential aging exposures; (3) ambient versus 70°C irradiation temperatures during the sequential aging exposures; (4) presence or absence of oxygen during the accident simulations; (5) sequential versus simultaneous accident LOCA simulations; and (6) ambient versus 70°C irradiation temperatures during the sequential accident irradiations. The overall research goal was to

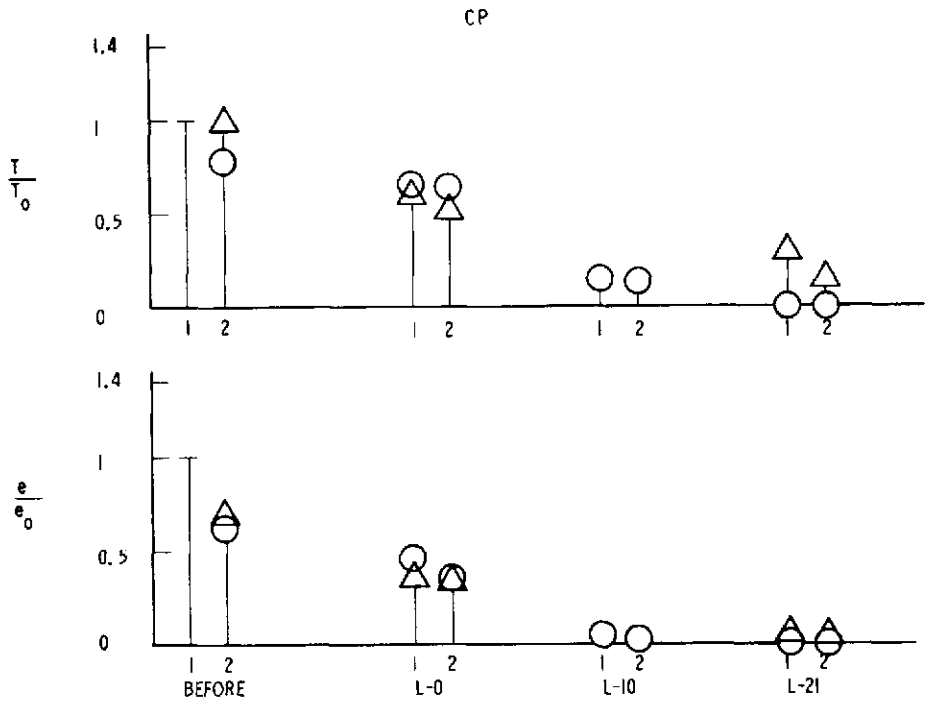


Figure 1.6.1 Tensile property results for the chloroprene (CP) material. The tensile strength, T, divided by its unaged value, T_0, and the ultimate tensile elongation, e, divided by the unaged value, e_0, are plotted after the aging exposure and after the 3 LOCA simulations. The numbers 1 and 2 refer to the two aging conditions: unaged, and 20 Mrad at 830 krad/h, respectively. BEFORE refers to the results before the LOCA simulation, L-0, L-10, and L-21 refer to results after the LOCA simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 1.6.2)

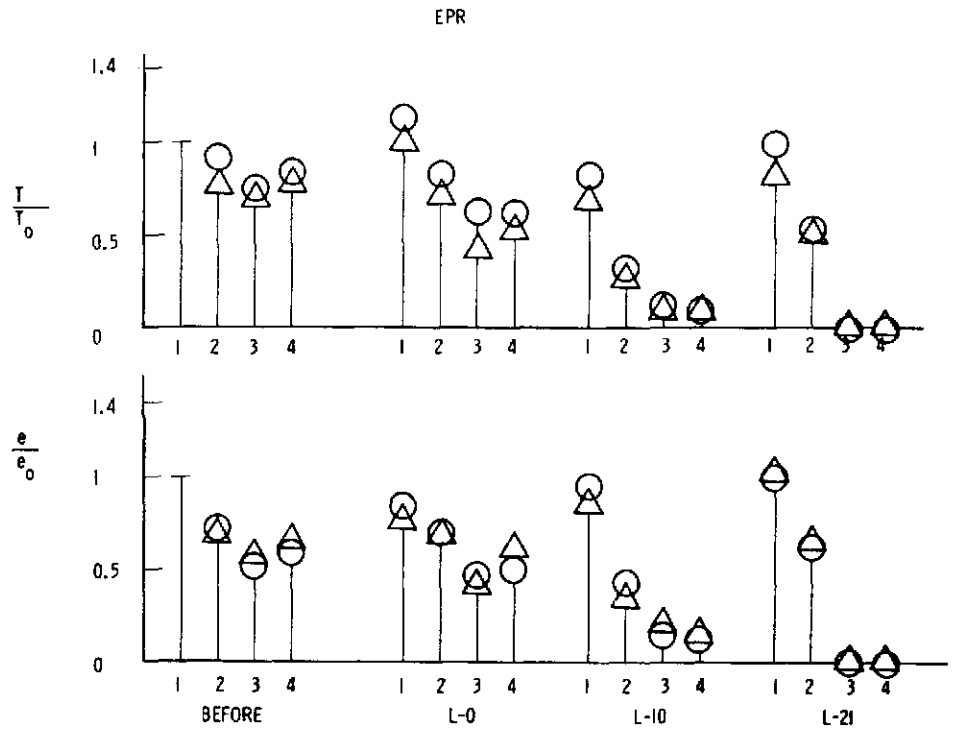


Figure 1.6.2 Tensile property results for the EPR material. The tensile strength, T, divided by its unaged value, T_o, and the ultimate tensile elongation, e, divided by the unaged value, e_o, are plotted after the aging exposure and after the 3 LOCA simulations. The numbers 1, 2, 3, and 4 refer to the four aging conditions: unaged, 22 Mrad at 910 krad/h; 47 Mrad at 930 krad/h; and 26 Mrad at 27 krad/h, respectively. BEFORE refers to the results before the LOCA simulation, L-0, L-10, and L-21 refer to results after the LOCA simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 1.6.2)

MATERIAL	LOCA TEST % O_2	AGING LOT	PERCENT WEIGHT CHANGE AFTER 6 HR	1500 HR	5000 HR
PVC	0	1	+93	+14	-4
		2	+220	+50	-1
		3	+347	+46	-4
		4	+330	+62	-7
	10	1	+53	-2	
		2	+150	-4	
		3	+360	-3	
		4	+230	-4	
	21	1	0	-1	-1
		2	+18	0	-2
		3	+39	+4	-2
		4	+35	+4	-3

Table 1.6.1 Weight Change Due to LOCA Simulations versus ambient drying exposure times at completion of the LOCA Simulation. Aging Lot #1 refers to unaged PVC material. Lot #2 was aged to 20 Mrad at 830 krad/h. Lot #3 was aged to 42 Mrad at 830 krad/h. Lot #4 was aged to 23 Mrad at 24 krad/h. (From Reference 1.6.2)

evaluate whether some combinations of alternative aging and accident simulation techniques were better suited for qualification activities than other combinations of simulation techniques.

For some materials, Reference 1.6.3 noted that oxygen presence during accident simulations enhanced tensile property degradation. For example, Figures 1.6.3 and 1.6.4 present data for two commercial EPR insulation materials. For both materials, the ultimate tensile elongation was more degraded by LOCA simulations that included oxygen than those that did not. This conclusion applies to sequential as well as simultaneous accident exposures. Reference 1.6.3 also noted that oxygen was important to the degradation of Tefzel materials for some choices of aging and accident simulation techniques (Table 1.6.2). For example, simultaneous

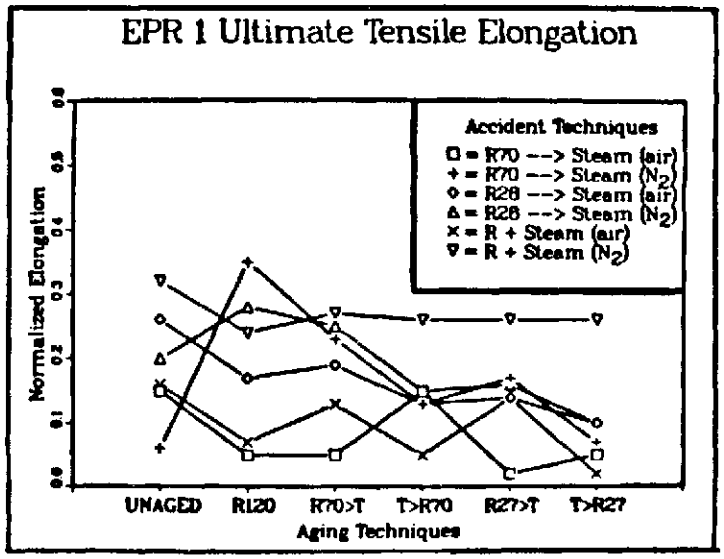


Figure 1.6.3 Ultimate tensile strength of EPR 1 at completion of the accident simulations. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 120°C and ~ 65 krad/hr, 16-day exposure. (Dose ~ 25 Mrad.) R28 and R70 also refer to accident irradiations (dose ~ 60 Mrad; dose rate ~ 300 krad/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.6.3)

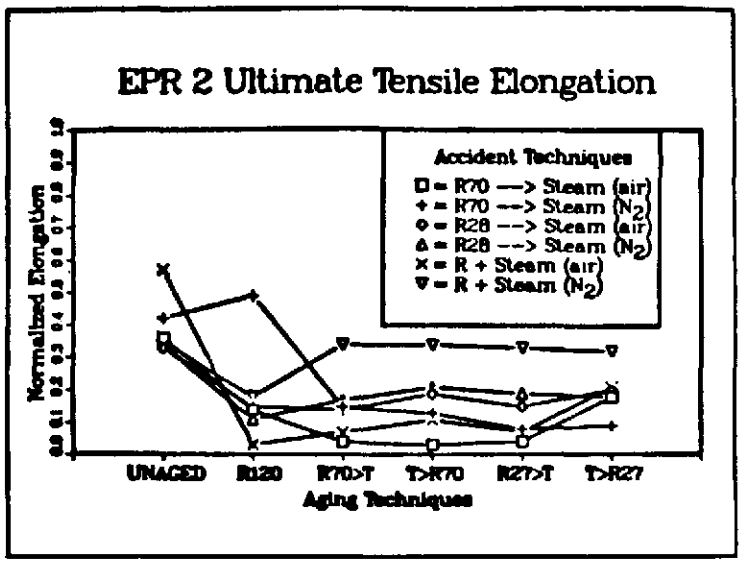


Figure 1.6.4 Ultimate tensile strength of EPR 2 at completion of the accident simulations. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 120°C and ~ 65 krad/hr, 16-day exposure. (Dose ~ 25 Mrad.) R28 and R70 also refer to accident irradiations (dose ~ 60 Mrad; dose rate ~ 300 Krad/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.6.3)

Table 1.6.2 (From Reference 1.6.3)

a. **TEFZEL 1: Bend Radii By Which All Samples Cracked. Table entries are expressed as multiples of the TEFZEL 1 sample radius (i.e., 75 %).**

Aging Technique	ACCIDENT SIMULATIONS					
	R70→ST(AIR)	R70→ST(N_2)	R28→ST(AIR)	R28→ST(N_2)	R + ST(AIR)	R + ST(N_2)
UNAGED	69.00	75.00	11.00	6.00	69.00	
R120	75.00	75.00	75.00		75.00	75.00
R70→120	75.00	75.00	69.00	50.00	75.00	44.00
120→R70	75.00	75.00	44.00	31.00	75.00	69.00
R27→120	75.00	75.00	44.00	75.00	75.00	22.00
120→R27	75.00	75.00	75.00	75.00	75.00	56.00

b. **TEFZEL 2: Radii By Which All Samples Cracked. Table entries are expressed as multiples of the TEFZEL 2 radius.**

Aging Technique	ACCIDENT SIMULATIONS					
	R70→ST(AIR)	R70→ST(N_2)	R28→ST(AIR)	R28→ST(N_2)	R + ST(AIR)	R + ST(N_2)
UNAGED	6.00	6.00			75.00	
R120	75.00	75.00	75.00	75.00	75.00	75.00
R70→120	75.00	75.00	44.00	75.00	75.00	56.00
120→R70	22.00	75.00	75.00	56.00	75.00	75.00
R27→120	11.00	11.00	11.00	11.00	75.00	22.00
120→R27	22.00	11.00	11.00	22.00	75.00	44.00

steam, chemical spray, and radiation accident exposures were typically more degrading if an overpressure of oxygen was present than if an overpressure of nitrogen was present. Oxygen presence during accident simulations were not important to the degradation of chlorosulfonated polyethylene and chlorinated polyethylene samples. For two cross-linked polyolefin materials, Reference 1.6.3 noted that the simultaneous accident simulation containing oxygen was typically the least degrading simulation for the ultimate tensile elongation properties (Figures 1.6.5 and 1.6.6).

Reference 1.6.3 concluded that the presence or absence of air during accident simulations can influence the degree of degradation in some materials.

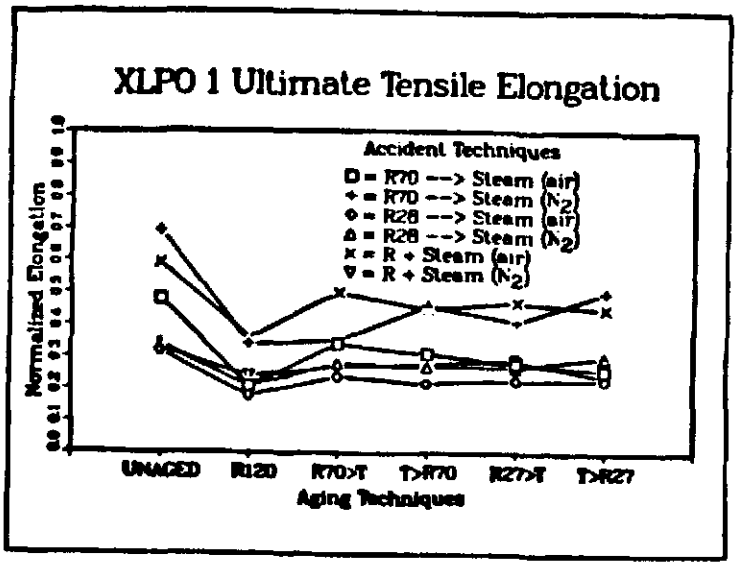


Figure 1.6.5 Ultimate tensile strength of XPLO 1 at completion of the accident simulations. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 120°C and ~ 65 krad/hr, 16-day exposure. (Dose ~ 25 Mrad.) R28 and R70 also refer to accident irradiations (dose ~ 60 Mrad; dose rate ~ 300 krad/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.6.3)

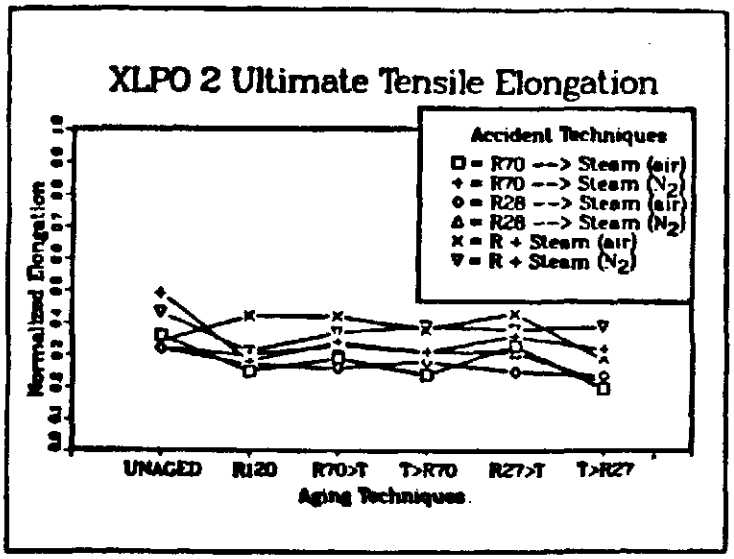


Figure 1.6.6 Ultimate tensile strength of XPLO 2 at completion of the accident simulations. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R27 refers to an ambient temperature aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) R70 refers to a 70°C aging irradiation. (Dose ~ 25 Mrad; dose rate ~ 65 krad/hr.) T indicates an elevated temperature exposure for 16 days at 120°C. R120 refers to a simultaneous 120°C and ~ 65 krad/hr, 16-day exposure. (Dose ~ 25 Mrad.) R28 and R70 also refer to accident irradiations (dose ~ 60 Mrad; dose rate ~ 300 krad/h). Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 1.6.3)

International and Other U.S. Programs

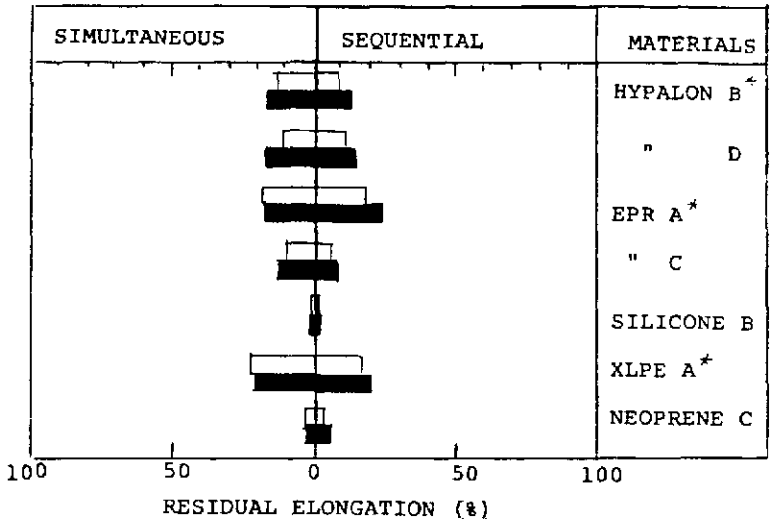
Tensile property and volume resistivity data after LOCA testing (with and without air) is presented in Reference 1.6.4 for several polymer materials. Both PWR and BWR LOCA conditions were simulated. The total accident radiation exposure for PWR conditions was 150 Mrad while the total accident radiation exposure for the BWR conditions was 26 Mrad. Simultaneous and sequential accident simulation techniques were employed. Both preconditioned and nonpreconditioned specimens were tested. Preconditioning for all specimens was a sequential exposure of thermal aging (7 days at 121°C) followed by irradiation (50 Mrad).

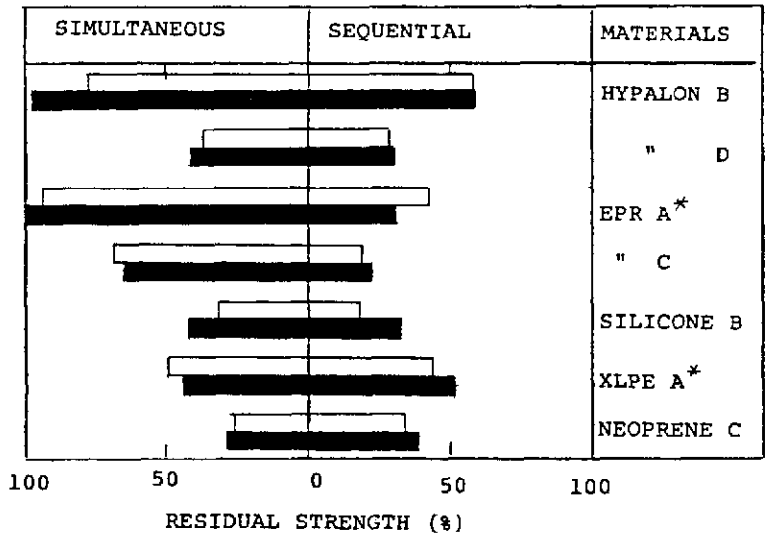
Figures 1.6.7 and 1.6.8 illustrate the comparison of mechanical properties for the test materials after simultaneous and sequential PWR accident simulations without and with air, respectively. Residual elongation of the insulating materials, except for silicone rubber, decreased significantly under air containing conditions in both simultaneous and sequential PWR simulation methods. Figures 1.6.9 and 1.6.10 illustrate the comparison of mechanical properties for the test materials after simultaneous and sequential BWR accident simulations without and with air, respectively. Some oxygen effects were also noted.

Figure 1.6.11 presents Reference 1.6.4 volume resistivity data for several insulating materials at completion of LOCA simulation tests. Simulations were performed both with and without air in the steam. For EPR, HYPALON, and NEOPRENE, the volume resistivity was degraded by two to three orders of magnitude by the air containing simulations but only degraded by zero to two orders of magnitude by the non-air simulations. Silicone volume resistivity was not strongly affected by the presence or absence of air during the LOCA simulation. Reference 1.6.4 noted a relationship between swelling of materials during the LOCA tests and decreases in volume resistivity. It states: "Materials which show a larger drop in volume resistivity swelled easily under LOCA conditions, especially with air. This suggests that radiation-induced oxidation during LOCA tests will result in the formation of hydrophilic part in the materials which results in swelling."

Reference 1.6.5 provides additional data regarding the effects of oxygen during LOCA simulations. PWR LOCA simulations were performed for both air and non-air conditions. Both sequential and simultaneous accident simulation techniques were employed. Tensile strength test results at completion of the accident simulations are presented. The authors note that when the LOCA environments were simulated by saturated steam conditions, most materials were slightly more degraded in tensile strength by sequential simulation techniques. However, when air was included with the steam environment, then the simultaneous simulation techniques were slightly more severe.

Reference 1.6.5 also provides degradation behavior data for a Hypalon and an EPR material. Results are illustrated in Figures 1.6.12 and 1.6.13. For the Hypalon material, elongation and tensile strength were





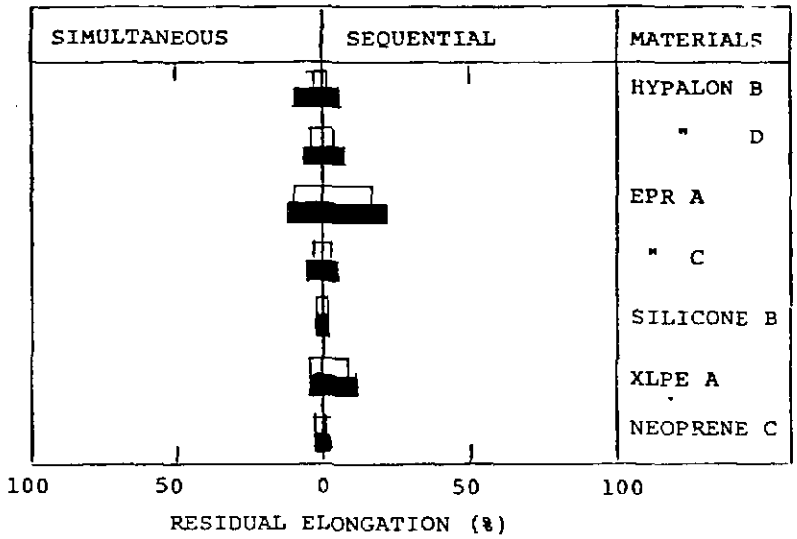
Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (PWR conditions).

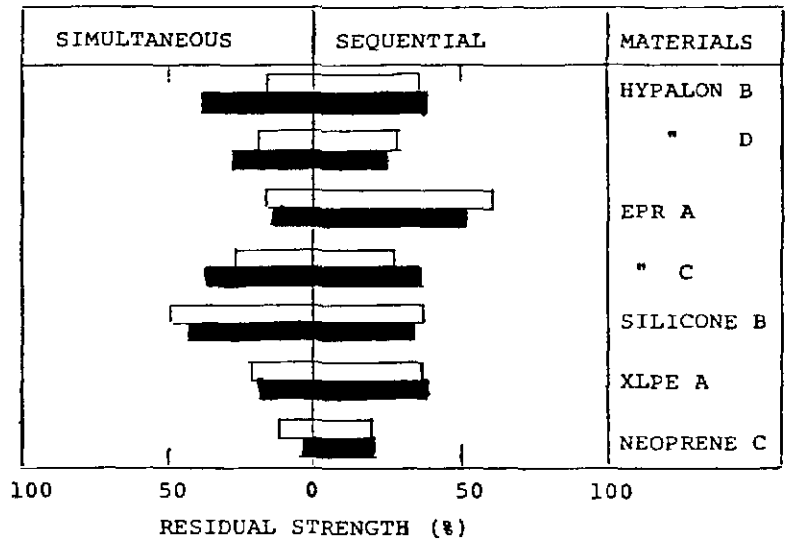
☐ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 1.6.7 (From Reference 1.6.4)



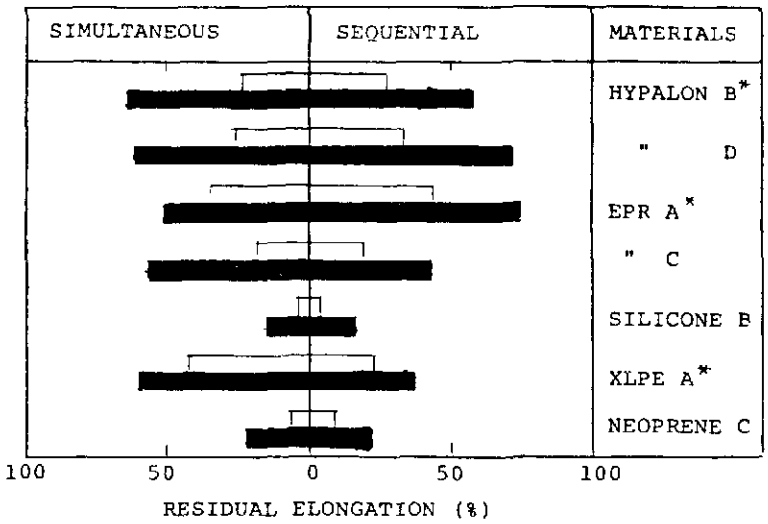


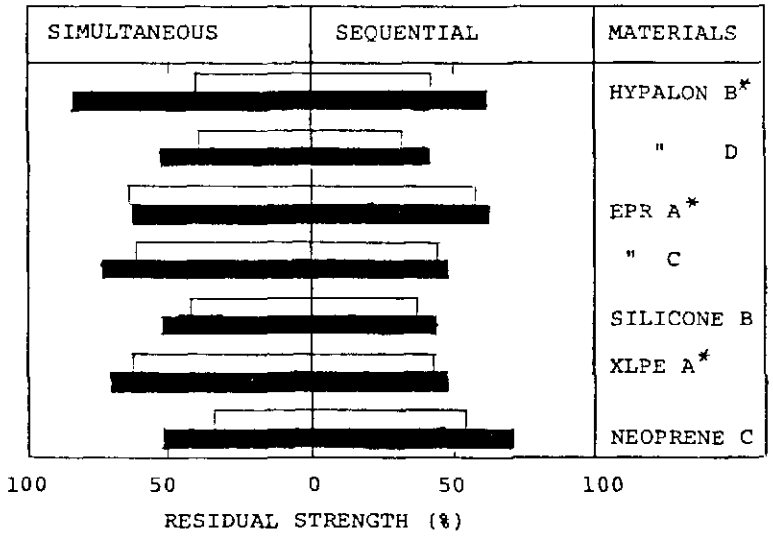
COMPARISON OF MECHANICAL PROPERTIES OF THE MATERIALS AFTER SIMULTANEOUS AND SEQUENTIAL LOCA TESTING (PWR CONDITIONS CONTAINING AIR)

PRE-CONDITIONED

NOT PRE-CONDITIONED

Figure 1.6.8 (From Reference 1.6.4)





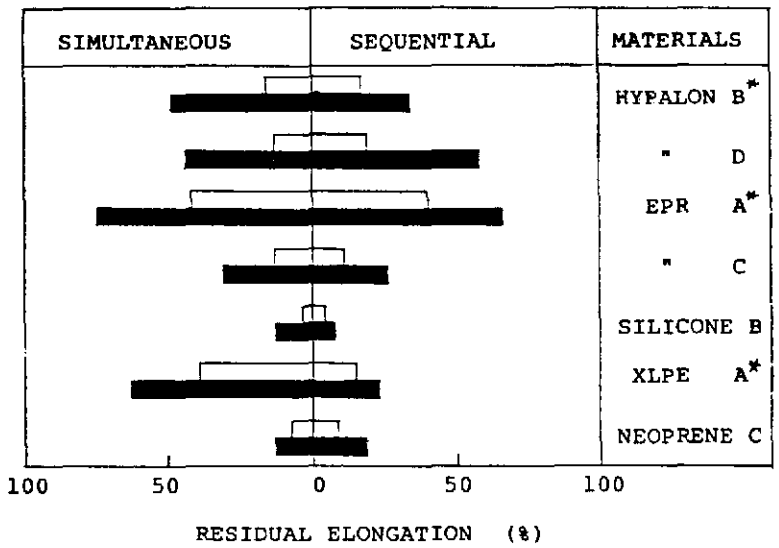
Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions).

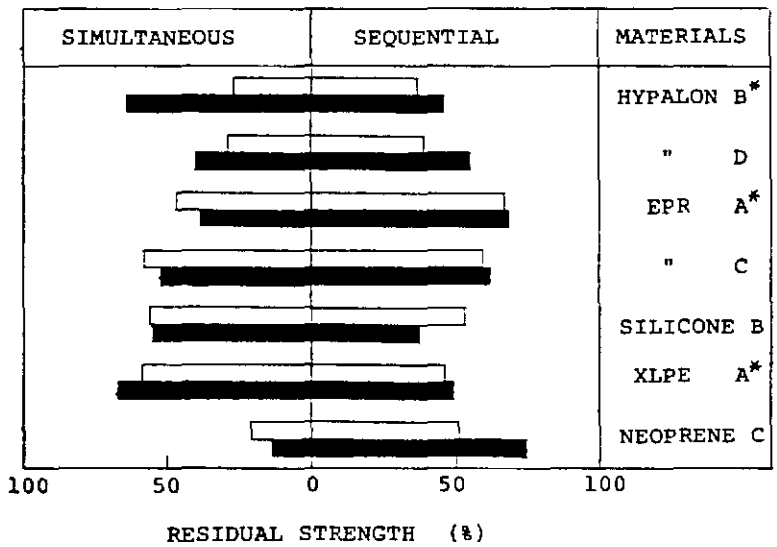
☐ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 1.6.9 (From Reference 1.6.4)



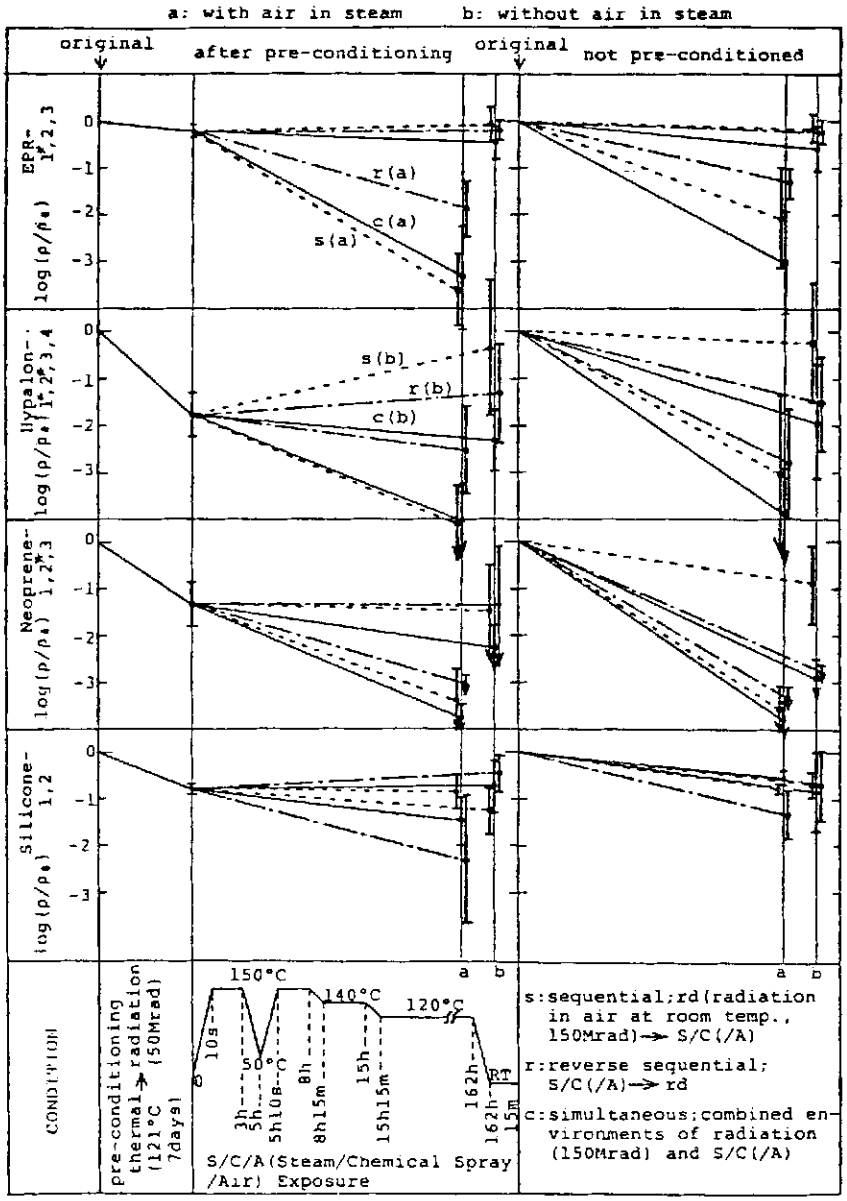


Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions containing air).

☐ ; Pre-conditioned

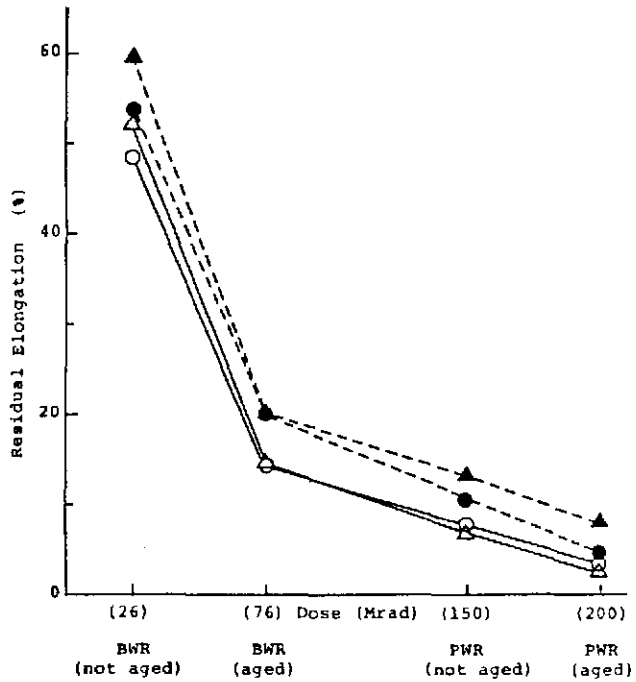
■ ; Not pre-conditioned

Figure 1.6.10 (From Reference 1.6.4)

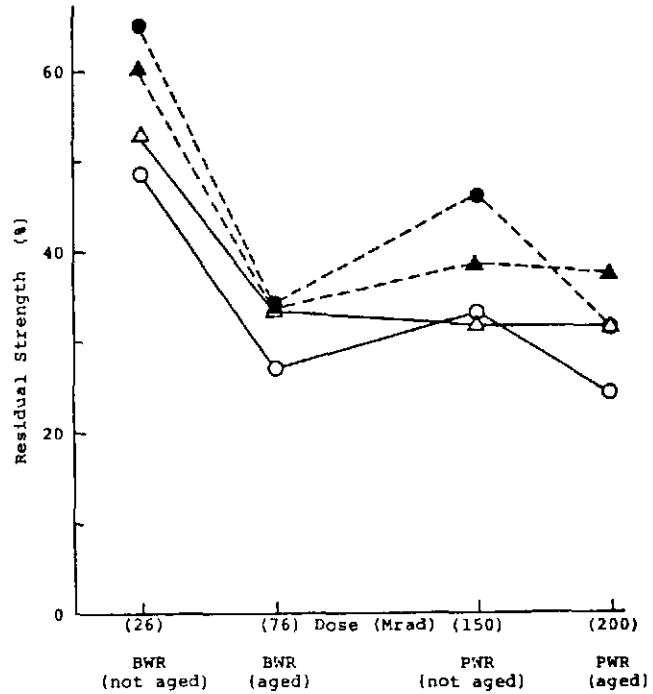


Log mean values of volume resistivity for some insulating materials after simultaneous, sequential and reverse sequential LOCA tests.

Figure 1.6.11 (From Reference 1.6.4)



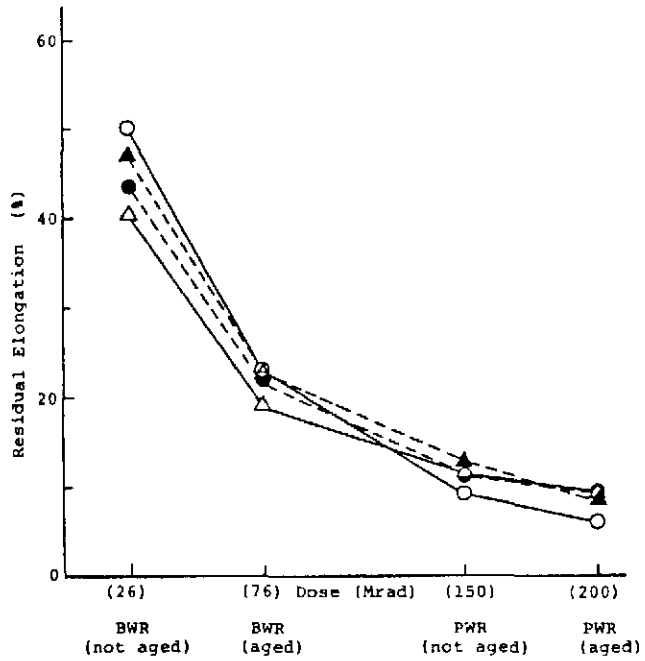
Changes in Elongation of Hypalon under various simulated LOCA conditions.



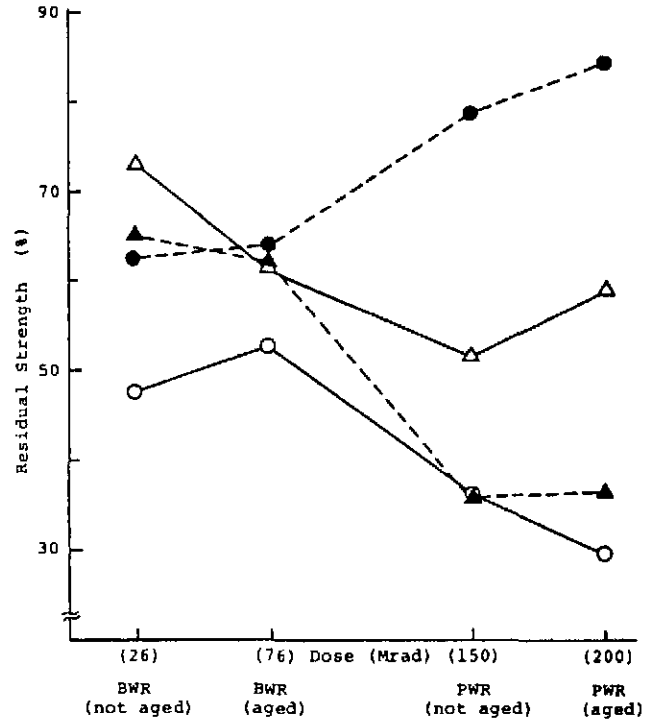
Changes in Tensile Strength of Hypalon under various simulated LOCA conditions.

Symbol	Method	Air
—○—	Simultaneous	Yes
--●--	"	No
—△—	Sequential	Yes
--▲--	"	No

Figure 1.6.12 (From Reference 1.6.5)



Changes in Elongation of EPR under various simulated LOCA conditions.



Changes in Tensile Strength of EPR under various simulated LOCA conditions.

Symbol	Method	Air
—○—	Simultaneous	Yes
--●--	"	No
—△—	Sequential	Yes
--▲--	"	No

Figure 1.6.13 (From Reference 1.6.5)

slightly more degraded by air containing accident simulations. For the EPR material, oxygen effects were not clearly established.

Reference 1.6.6 compares the behavior of cable insulation and jacket materials to the electrical performance of cables. Accident simulations are performed with and without air and for two different time periods (one week and three months). The purpose of varying the accident length was to evaluate the ability to accelerate accident simulations. Air presence was considered an important variable to be considered in this evaluation. For insulation and jacket sheet materials tensile and volume resistivity properties were monitored. For the cable specimens insulation resistance was determined.

For sheet specimens of EPR insulation and Hypalon jacket materials, air presence was important to both the tensile and volume resistivity degradation. In steam with air environments, the volume resistivity and tensile properties were typically more degraded by the three month LOCA exposure than by the one week LOCA exposure. For non-air exposures, volume resistivity and tensile properties did not depend as much on the length of the LOCA exposure (1 week versus 3 months).

For cable specimens, insulation resistance did not exhibit similar large effects. Figures 1.6.14 and 1.6.15 demonstrate that insulation resistance differences between air and non-air LOCA simulations are small. Moreover, Reference 1.6.6 illustrates that the cable insulation material experienced less degradation of the elongation than did the sheet material (Figure 1.6.16). Reference 1.6.6 concludes that the cable construction enhances the performance of the insulation during air containing LOCA simulations.

Issue Findings

References 1.6.2 through 1.6.6 clearly establish that some cable materials' properties are affected by the presence or absence of oxygen during LOCA simulations. Oxygen effects include decreased volume resistivity and tensile properties as well as increased dimensional swelling. Oxygen effects depend on the material studied and sometimes even on the specific composition of material such as EPR which has many industrial formulations. For example, references 1.6.2, 1.6.3, 1.6.4, 1.6.5, and 1.6.6 each provided illustrations of oxygen effects for some EPR materials. However, both references 1.6.2 and 1.6.5 provided examples where certain EPR formulations did not exhibit oxygen effects during accident simulations. Clearly, the chemical composition of EPR is an important variable affecting polymer degradation during LOCA simulations.

References 1.6.2 through 1.6.6 also demonstrated that the absence of oxygen during LOCA simulations may underestimate the changes that occur in some cable materials while overestimating the changes which occur in other materials when compared to accident simulations that include oxygen. EPR materials appear in many of the references as examples of

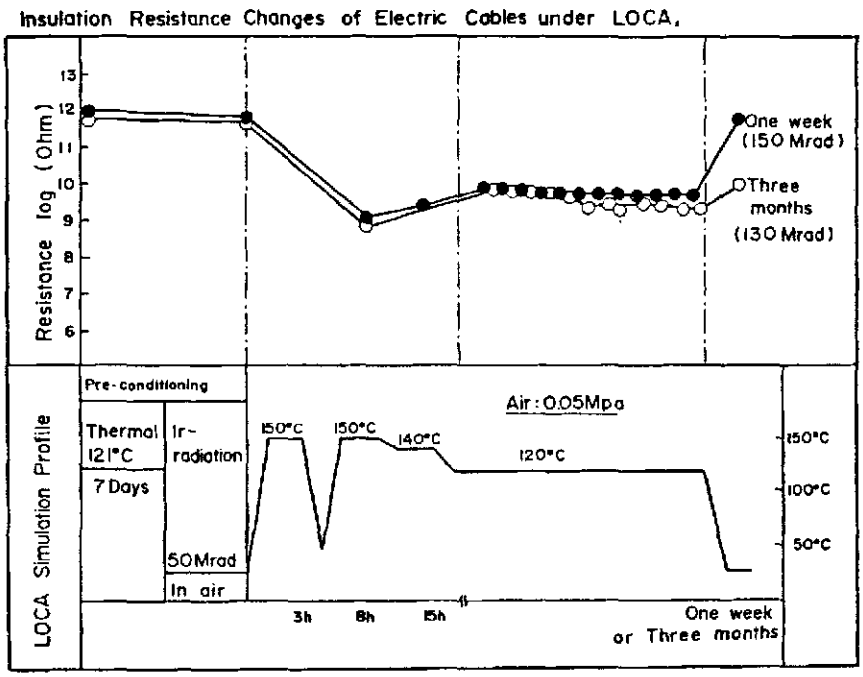


Figure 1.6.14 Electrical cable performance during an air LOCA exposure. Figure 1.6.15 illustrates cable performance during a non-air LOCA exposure. (From Reference 1.6.5)

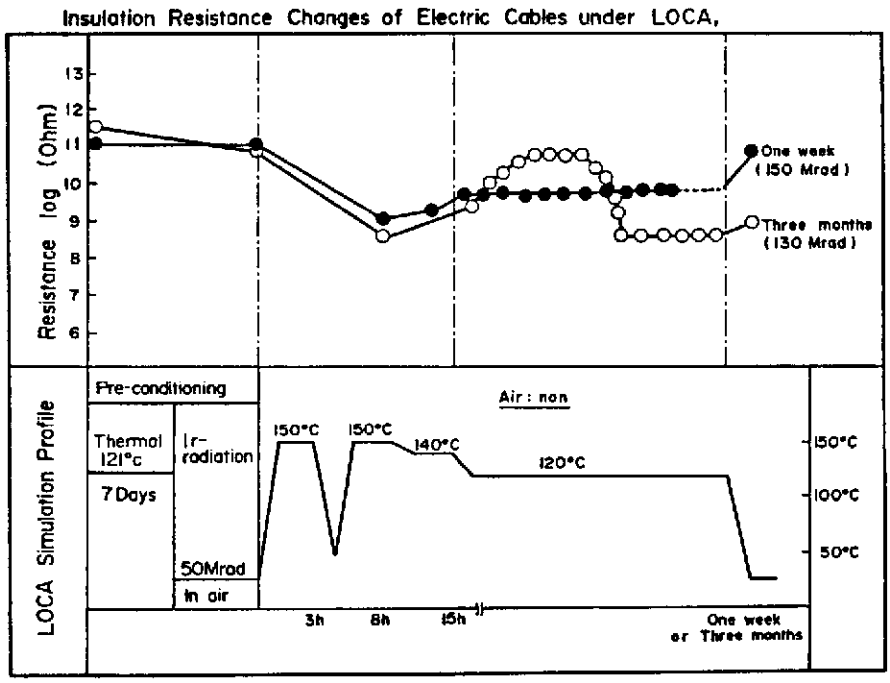


Figure 1.6.15 Electrical cable performance during a non-air LOCA exposure. Figure 1.6.14 illustrates cable performance during an air LOCA exposure (From Reference 1.6.5)

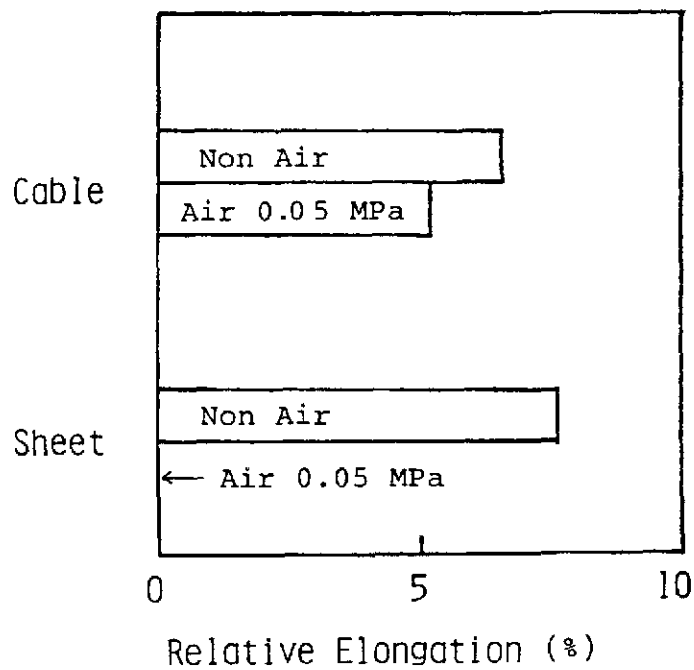


Figure 1.6.16 Comparison between degradation of sheet shape samples and degradation of cables under three months LOCA test. (From Reference 1.6.6)

cable materials whose degradation would be decreased by the absence of oxygen during LOCA simulations. PVC and XLPO cable materials appear in some of the references as examples where degradation may be enhanced by the absence of oxygen during LOCA simulations.

Many of the referenced experiments tested only materials' properties for tensile specimens. Hence the protective effect of the use configuration (i.e., cable jacketing for insulated conductors) was typically not

examined. Reference 1.6.6 addressed this issue for an EPR/Hypalon cable construction and noted that oxygen effects for the cable were less than for exposed EPR insulation sheet material. The cable jacket apparently reduced the oxygen available for interaction with the insulation.

Connection and enclosure equipment which depends on organic seals for functionability have not been extensively tested to determine the importance of oxygen during LOCA simulations.

Research efforts summarized in this Section have investigated whether oxygen presence during a LOCA simulation impacts test results. An oxygen partial pressure profile to simulate realistic accident scenerios has not yet been determined. When this information is obtained, it may be desirable to evaluate whether the current oxygen simulation methods adequately simulate the time behavior of the oxygen partial pressure during a realistic accident.

Findings within this issue have began to be incorporated into industry practices. For example, in October 1984, a Class 1E equipment manufacturer informed the NRC that its qualification tests for a new line of equipment would include air in the test chamber during the entire environmental accident simulation.

In summary then, the findings within this issue are:

1. Oxygen effects during accident simulations may be important to equipment or material degradation.

   a. For some materials degradation is enhanced by the presence of oxygen during LOCA simulations.
   b. For some other materials degradation is enhanced by the absence of oxygen during LOCA simulations.

2. The protective effect of the use configuration (cable jacket) may reduce oxygen effects during accident simulations.

References

Program Effort

1.6.1 L. Bonzon, O. M. Stuetzer, W. H. Buckalew, F. V. Thome, Qualification Testing Evaluation Program: Light-Water Reactor Safety Research Semiannual Report, October 1979-March 1980, NUREG/CR-1749, SAND80-1307, Sandia National Laboratories, Albuquerque, New Mexico, September 1980.

A. L. Bonzon, W. H. Buckalew, R. L. Clough, K. T. Gillen, E. A. Salazar, O. M. Stuetzer, F. V. Thome, Status of the Qualification Testing Evaluation (QTE) Program, SAND80-2111A. Presented at the USNRC Eight Water Reactor Safety Research Information Meeting, October 27-31, 1980.

1.6.2 K. T. Gillen, R. L. Clough, G. Ganouna-Cohen, J. Chenion, G. Delmas, Loss of Coolant Accident (LOCA) Simulation Tests on Polymers: The Importance of Including Oxygen, NUREG/CR-2763, SAND82-1071, Sandia Laboratories, Albuquerque, New Mexico, July 1982.

A. K. T. Gillen, R. L. Clough, G. Ganouna-Cohen, J. Chenion, G. Delmas, The Importance of Oxygen in LOCA Simulation Tests, SAND82-2144J. Published in Nuclear Engineering and Design 74 (1982), 271-285.

B. L. Bonzon, Status of the Qualification Testing Evaluation (QTE) Program, SAND82-1910C. Presented at the USNRC Tenth Water Reactor Safety Research Information Meeting, October 12-15, 1982. A proceedings of the meeting was issued (NUREG/CP-0041, Vol. 5)

C. L. Bonzon, The Qualification Testing Evaluation (QTE) Program at Sandia National Laboratories, SAND82-1032A. Presented at the EPRI/Sandia/NRC Information Exchange Meetings, May 4-5, 1982.

1.6.3 L. D. Bustard, J. Chenion, F. Carlin, C. Alba, G. Gaussens, M. LeMeur, The Effect of Alternative Aging and Accident Simulations on Polymer Properties, NUREG/CR-4091, SAND84-2291, Sandia National Laboratories, Albuquerque, New Mexico, May 1985.

A. L. D. Bustard, et al., The Effect of Alternative Aging and Accident Simulations on Polymer Properties, SAND84-1759C. Presented at the 12th Light Water Reactor Safety Meeting, October 22-26, 1984. Presented at the USNRC Twelfth Light Water Reactor Safety Meeting. A proceedings of the meeting was issued (NUREG/CP-0058, Vol. 5)

B. L. D. Bustard, The Effect of Alternative Aging and Accident Simulations on Polymer Properties, SAND84-2291A. Presented at the IEEE Insulated Conductors Committee Meeting, November 12-14, 1984, St. Petersburg, Florida.

International and Other U.S. Programs

1.6.4 Y. Kusama, S. Okada, M. Yoshikawa, M. Ito, T. Yagi, Y. Nakase, T. Seguchi, and K. Yoshida, Methodology Study for Qualification Testing of Wire and Cable at LOCA Conditions. Presented at the Tenth Water Reactor Safety Research Information Meeting, October 12-15, 1982. A proceedings of the meeting was issued (NUREG/CP-0041, Vol. 5)

1.6.5 K. Yoshida, T. Seguchi, S. Okada, M. Ito, Y. Kusama, T. Yagi, and M. Yoshikawa, Progress on Qualification Testing Methodology Study of Electric Cables. Presented at the Eleventh Water Reactor Safety Research Information Meeting, October 24-28, 1983. Proceedings of the meeting were issued (NUREG/CP-0041, Vol. 5)

1.6.6 M. Ito, Y. Kusama, T. Yagi, S. Okada, M. Yoshikawa, and K. Yoshida, Progress on Qualification Testing Methodology Study of Electric Cables. Presented at the Twelfth Water Reactor Safety Research Information Meeting, October 22-26, 1984. Proceedings of the meeting were issued (NUREG/CP-0058, Vol. 5)

1.7 Chemical Spray Effects

Issue Description

The chemical spray solution commonly used in equipment qualification tests consists of 0.28 molar boric acid, 0.064 molar sodium thiosulfate, and sodium hydroxide added to obtain a pH between 10 and 11 at 77°C (IEEE 323-1974, Table A1). However, this is a guide only, and 10 CFR 50.49, Section e.3, requires that: "The composition of chemicals used must be at least as severe as that resulting from the most limiting mode of plant operation..."

and Regulatory Guide 1.89, Section 2.6:

> "The effects of containment spray system operation should be considered. This consideration should include, as appropriate, the effects of demineralized water spray or chemical spray systems".

Since all surfaces in the containment may be exposed to this spray at elevated temperatures and/or high radiation levels, it is important to examine a range of materials and components for degradation and failure under spray of varying compositions (including deionized water). The simulation technique offering itself consists of exposing identical test samples of materials and components to spray of varying chemical properties, applied under varying conditions such as temperature.

This is the thrust of this effort to address the issue:

> "Is the variability of chemical spray composition significant in the accident simulation?"

Program Effort

Chemical spray, deposited on the surface, affects the properties of electrical insulators. In a reactor containment, insulators possibly exposed to spray are most often connection boards or "terminal blocks."

Reference 1.7.1, NUREG/CR-1682 (SAND80-1957), reports on an investigation into the probability and severity of terminal block damage under steam and spray conditions. More than 600 terminal blocks (made of phenolic) were exposed for an average of five hours to heat, steam, standard spray and contamination variations. The conditions were selected to imitate the TMI-2 accident. In most tests the terminals were protected by a standard NEMA enclosure, into which a pressure equilibrating drainage hole had been cut.

Two electrical phenomena of concern were identified:

- Leakage currents between different circuits on the same terminal block occur under steam and spray exposure.

- Temporary or permanent electrical breakdown by a carbonized electrical path to ground could eliminate the circuit function.

In Figure 1.7.1 leakage currents are shown for two different environment temperatures, 43°C and 86°C. Two groups of experiments are plotted for 86°C. The values around 0.1 mA are for blocks in enclosures equipped with 6 mm "weepholes." The values around 1.0 mA are for open blocks, not directly hit by steam or spray.

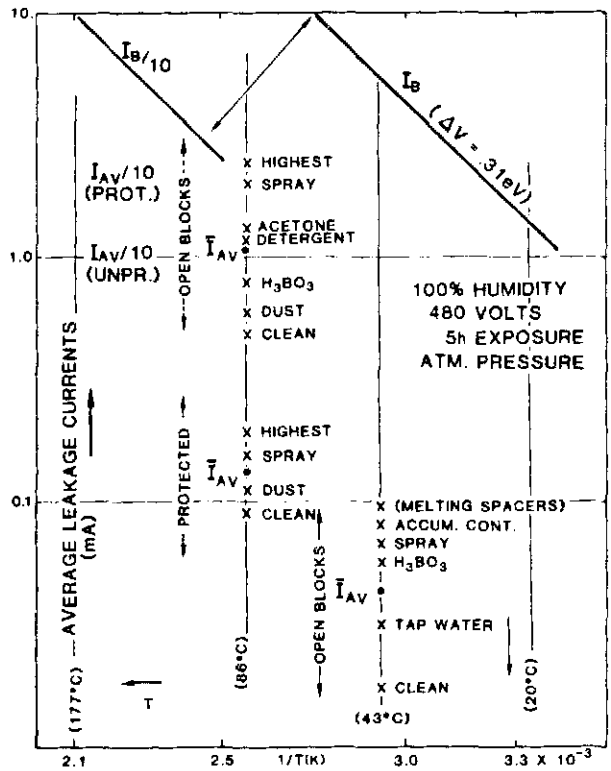


Figure 1.7.1 Leakage Currents for Various Contaminations. (From Reference 1.7.1)

The higher the leakage current is, the higher is the probability that a complete breakdown could occur. Figure 1.7.2 shows breakdown probability values for a 5-hour exposure experiment at 480 volts a.c., again for open (upper curve) and protected blocks (lower curve).

The study reaches the following conclusions:

- Spray -- and other contaminant agents -- generate surface conductivity on insulators leading to leakage currents and possible electrical breakdown.

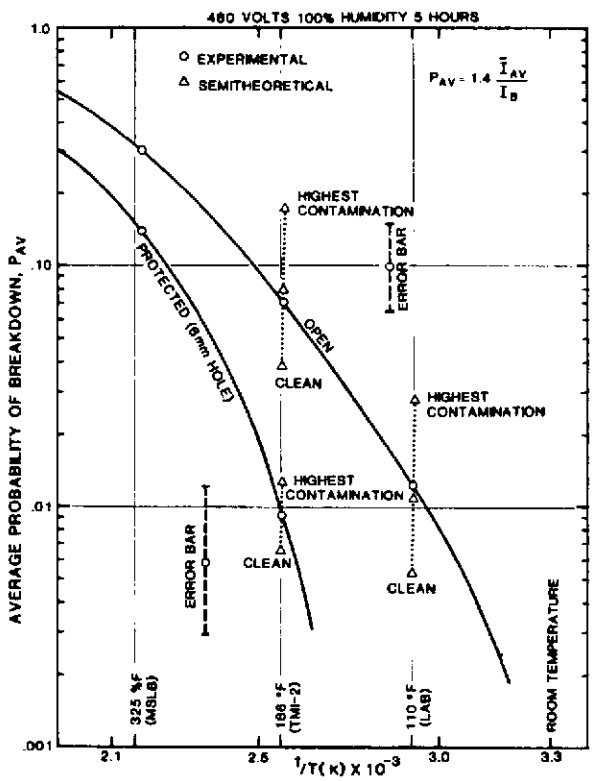


Figure 1.7.2 Average Probability of Breakdown vs. Inverse Temperature. (From Reference 1.7.1)

- Terminal blocks should be cleaned and protected. They should not be mounted in such ways that condensate can collect in corners or screwholes. Blocks or enclosures should not be mounted on a heat sink.

- The particular chemical composition of reactor spray is of lesser influence than the electrical conductivity.

International and Other U.S. Programs

The only international study on the effects of chemical spray (of which we are aware) is presented in Reference 1.7.2 (JAERI-M 83-072). Eight elastomer formulations were aged under various conditions of radiation dose and thermal aging periods, then exposed to boiling spray solutions (water and IEEE 323-1974 chemical spray). A measure of their swelling behavior was then made. Figures 1.7.3 and 1.7.4 summarize the results for two of the materials tested (Hypalon and EPR). In addition, the authors make the following observations:

- All eight elastomers tested showed remarkable swelling with an increase of radiation dose when they are irradiated in air.

- Swelling in boiling water was about twice that as in chemical solution.

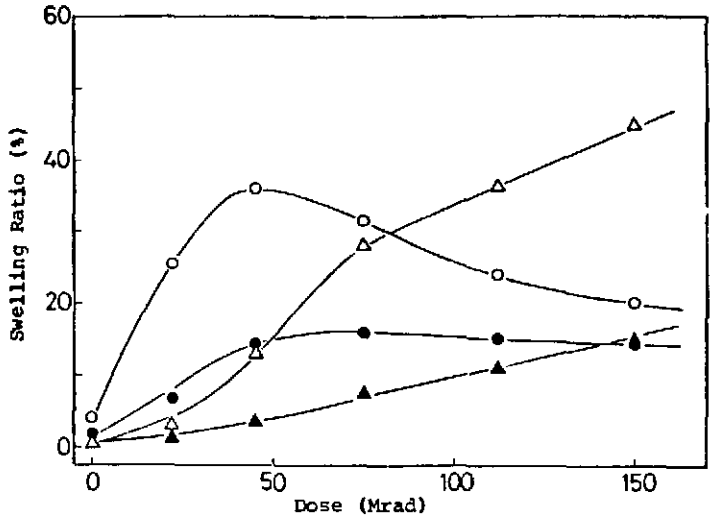
- Some types of Neoprene and Hypalons had an optimum swelling dose where they showed a maxima. Over this dose, the swelling ratio decreased with dose.

Reference 1.7.3, NUREG/CR-3418 (SAND83-1617), describes a two-phase series of simulated LOCA tests on twenty-four terminal blocks. The terminal blocks were housed in NEMA-4 enclosures, and leakage currents were monitored throughout the test. The comparison of spray (phase-2) and no-spray (phase-1) results shows no discernable difference in IRs existed between the periods with and without chemical spray.

Reference 1.7.4, NUREG/CR-3691 (SAND84-0422), is a compendium of terminal blocks used in the nuclear industry, including factors affecting their performance, and their possible impact on plant operations. This document includes a discussion of the theoretical considerations governing film formation and surface conduction on terminal blocks.

Issue Findings

By varying the chemical spray composition (including deionized water) on identical test samples, within the limits of single system failures that can be postulated, the corrosive and conductive effects of spray could be evaluated and its importance in affecting test results determined.



(○); HP−1. Water (●); HP−1. Chemical Solution
(△); HP−2. Water (▲); HP−2. Chemical Solution

Figure 1.7.3 Comparison of Swelling Behavior in Hypalon in Water and in Chemical Solution. (From Reference 1.7.2)

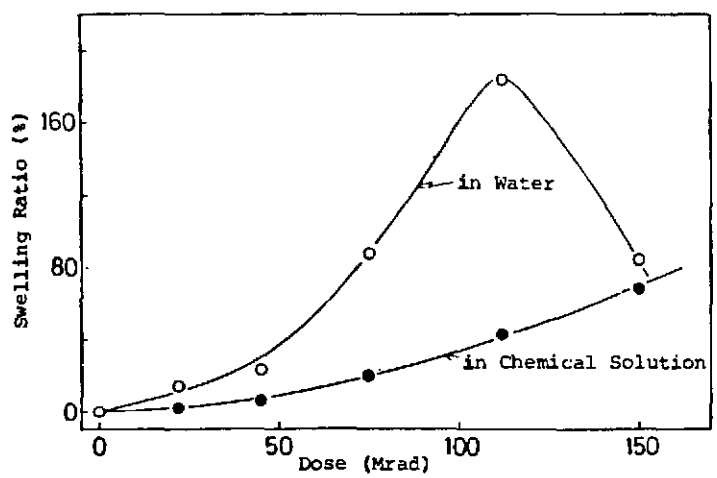


Figure 1.7.4 Comparison of Swelling Behavior of EPR in Water and in Chemical Solution. (From Reference 1.7.2)

Current industry practice is to utilize the chemical spray type and composition which is appropriate to the respective plants (varying from deionized water to the IEEE 323-1974 composition) during LOCA-simulation tests in equipment qualification.

In summary then, the findings within this issue are:

1. Leakage currents between different circuits connected to a terminal block might occur under steam and spray exposure.

2. Temporary or permanent electrical breakdown in terminal blocks, by a carbonized electrical path to ground, may eliminate the circuit function.

3. Spray--and other contaminant agents--might generate surface conductivity on terminal blocks leading to leakage currents and possible electrical breakdown.

4. Terminal blocks might be cleaned and protected, and mounted in such ways that condensate will not collect in corners or screwholes.

5. The particular chemical composition of reactor spray might be of lesser influence than the electrical conductivity.

References

Program Effort

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A. O. M. Stuetzer, Preliminary Examination of TMI-2 Equipment: Terminal Blocks, SAND80-2447A. Presented at the USNRC Eighth WRSR Information Meeting, Gaithersburg, Maryland, October 27-31, 1980.

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1.7.2 T. Yagi, et al, Swelling Behavior of Gamma-ray Irradiated Elastomers in Boiling Spray Solution, JAERI-M 83-072, Department of Development, Takasaki Radiation Chemistry Research Establishment, JAERI, Japan. Received April 14, 1983.

1.7.3 C. M. Craft, Screening Tests of Terminal Block Performance in a Simulated LOCA Environment, NUREG/CR-3418, SAND83-1617, Sandia National Laboratories, Albuquerque, New Mexico, August 1984. (CAP Program results.)

A. C. M. Craft, Performance and Effects of Terminal Blocks in a Loss of Coolant Accident Environment, SAND83-1965C. Presented at the 11th USNRC Water Reactor Safety Research Information Meeting, October 24-28, 1983, Gaithersburg, Maryland. Published in NUREG/CP-0048, Vol. 5, pp. 261-282, January 1984. (CAP Program results.)

1.7.4 C. M. Craft, An Assessment of Terminal Blocks in the Nuclear Power Industry, NUREG/CR-3691, SAND84-0422, Sandia National Laboratories, Albuquerque, New Mexico, September 1984. (CAP Program results.)

1.8 Acceleration of Postaccident Environments

Issue Description

Regulatory Guide 1.89 (June 1984) requires electric equipment important to safety be qualified. Qualification must demonstrate that the equipment is capable of performing its safety function during normal conditions, abnormal conditions, and for the duration of a design basis accident (Section B of Regulatory Guide 1.89). In order to reduce the time and cost of qualification tests, it is standard industry practice to time-compress the postaccident environments simulation.

This is the thrust of this effort to address the issue:

> "Can the effects of postaccident environments be satisfactorily accelerated?"

Program Effort

Reference 1.8.1, NUREG/CR-2763 (SAND82-1071), reports on experiments that were performed to survey the effects on material degradation of the oxygen concentration during a loss-of-coolant accident (LOCA)-simulation. Commercial materials, which are commonly used as electric cable jackets and insulations in nuclear power plants applications, were monitored in terms of weight, mechanical properties, solubility measurements and infrared spectroscopy.

For some of these materials (an EPR insulation, a chloroprene jacket and a PVC jacket), the concentration of oxygen during LOCA simulation was found to be an important parameter. For the first two materials, a higher level of degradation occurred when oxygen was present; for PVC, increased swelling occurred as the oxygen concentration was lowered. These results indicate that realistic LOCA-simulation (and post-DBE simulation) test should include a partial pressure of oxygen gas that is representative of the postulated time-dependent oxygen concentration expected during accident situations.

Reference 1.8.1 also suggests a possible, simple test to screen for LOCA sensitivity. Although applied to oxygen-LOCA environments, it may have post-DBE acceleration potential as well. Table 1.8.1 compares results from the 10% oxygen LOCA simulation (L-10) tests with air oven tests. The air oven tests consisted of 96 hours at 145°C. The temperature range achieved during the LOCA simulations was 171°C near the beginning of the test and dropped off to 121°C after 4-days. Many of the materials exhibited comparable degradation during both the LOCA-10% oxygen exposures and the air-oven exposures which suggests that similar degradation mechanisms may be involved. Hence this data would suggest that Arrhenius techniques may also be applicable for LOCA-oxygen tests; this provides a basis for postaccident acceleration.

The results from Table 1.8.1 should be taken as qualitative, however, it does indicate that the very simple application of a thermal oven to approximate LOCA environments may be useful as a screening device for comparing and choosing materials and formulations for safety component applications.

International and Other U.S. Programs

Reference 1.8.2, NUREG/CR-3863 (SAND84-1264), reports on an experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within, and beyond, the design basis. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested; each transmitter was exposed to a unique environment.

Transmitter T1 was exposed to a LOCA steam and chemical spray environment for 575 hours (24 days) without radiation. (The profile of the exposure generally followed the recommendations of IEEE 323-1974 Appendix A.)

Transmitter T2 was mounted in an air circulating oven and was exposed to a dry, thermal environment. During the experiment, the oven temperature was varied over a range of 105°C to 180°C to observe temperature effects.

Transmitter T3 was exposed to Cobalt-60 radiation to a total integrated dose (TID) of approximately 527 megarads at temperatures of 18 to 35°C.

Transmitter T4 was exposed simultaneously to a Cobalt-60 radiation and a LOCA temperature environment. To achieve the radiation and temperature (no steam) environment, the transmitter was sealed inside a stainless steel enclosure that was in turn placed inside the steam chamber located in the radiation cell. Radiation exposure was approximately 482 Megarads (air) TID. During posttest inspection, it was found that the seal between the transmitter and the stainless steel enclosure had failed, allowing water to enter the enclosure; therefore, the intended objective--to isolate the transmitter from the steam environment--was not maintained throughout the entire test sequence.

Transmitter T5 was exposed simultaneously to a Cobalt-60 radiation, LOCA steam and chemical spray environment.

The results provided both specific equipment performance and general qualification methodology insights. By testing in individual and combined environments, environmental effects on equipment performance were isolated:

(1) The primary environmental stress affecting the Barton model of this transmitter was temperature. The four transmitters exposed to temperatures above 122°C all showed significant shifts in output (between -9% to +27% of full scale).

Table 1.8.1
Sequential Screening Comparisons
(From Reference 1.8.1)

MATERIAL	LOT	e/e_o AIR OVEN	e/e_o L-10
CLPE	1	0.65	0.68
	3	0.05	0.42
	4	0.56	0.53
TEFZEL®	1	0.92	1.02
	4	0	0.04
SILICONE	1	0.93	0.96
	2	0.24	0.38
	4	0.17	0.38
CLPO	1	0.79	0.98
	2	0.33	0.70
	3	0.24	0.49
	4	0.26	0.72
CSPE	1	0.40	0.74
	2	0.27	0.61
	3	0.19	0.48
	4	0.15	0.58
PVC	1	0.57	0.53
	2	0.18	0.09
	4	0.04	0.03
CP	1	0.01	0.05
	2	0.01	0.03
EPR	1	0.77	0.89
	2	0.49	0.38
	3	0.12	0.17
	4	0.14	0.12

Aging: Lot
1 Unaged
2 21 Mrad at 880 krad/hr
3 45 Mrad at 880 krad/hr
4 23 Mrad at 24 krad/hr

(2) The time-at-temperature significantly reduced the magnitude of these output shifts.

(3) Analyses of the potentiometer piece parts indicated that the root cause of the temperature instability was a thermally-activated decrease in the dielectric qualities of the nylon insulating material of the potentiometer.

(4) A possible second, thermally activated, failure mechanism (corrosion due to lubricant) was found, which caused one potentiometer to open, resulting in the transmitter circuit exhibiting an overrange condition.

(5) The transmitters exhibited excellent radiation resistance.

Although this test was not designed to address the question of accelerated postaccident methods, the results indicate that for this particular type of instrument accelerated postaccident radiation may not be of concern (i.e., a large time compression factor may be acceptable), but particular attention should be paid to the accelerated thermal conditions.

Reference 1.8.3, NUREG/CP-0058, reports on a Japanese experiment on electric cable materials conducted to simulate post-LOCA radiation conditions on an accelerated basis. A three-month, 55 krad/hr, test was compared against a one-week, 925 krad/hr test. The results reported were:

1. Mechanical Properties

   In the case of the simulation by saturated steam, the differences of the degradation between the one-week test and the three-months test results were not observed for ethylene-propylene rubber (EPR), but clearly observed for chlorosulfonated polyethylene (CSPE).

   In the case of air-containing steam, the differences between both tests were significant even for EPR.

   Figures 1.8.1 and 1.8.2 show the observed degradation as a function of dose (under LOCA conditions) for two of the materials tested.

2. Volume Resistivity

   A decrease in volume resistivity of less than two orders of magnitude was found for EPR samples after the samples were exposed to LOCA environments by saturated steam. The influence of the exposure length was obscure.

   For LOCA tests simulated by air-containing steam, the influence of the exposure length was remarkable. The value decreased about

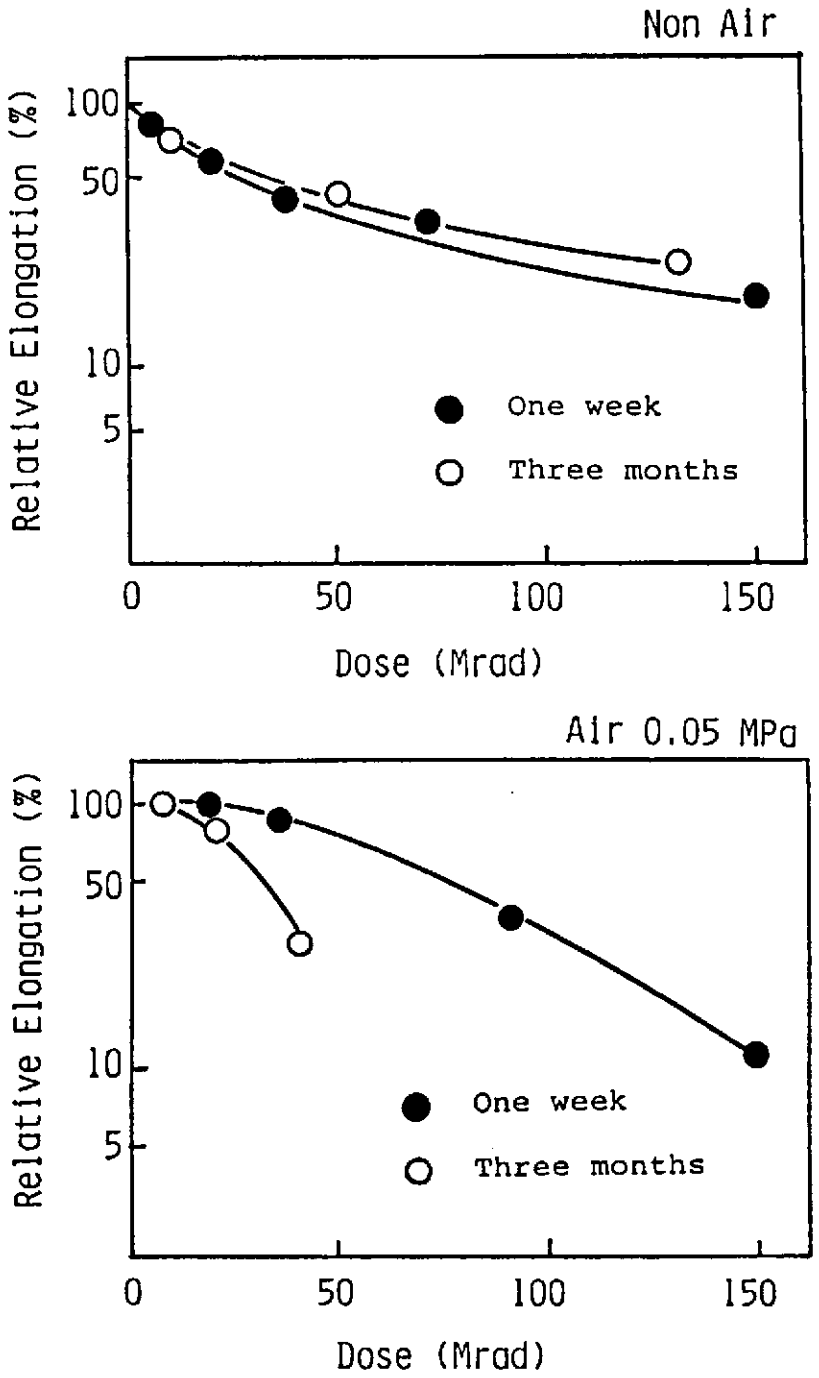

Figure 1.8.1 Ultimate tensile elongation of EPR-A as a function of dose under LOCA condition. (From Reference 1.8.3)

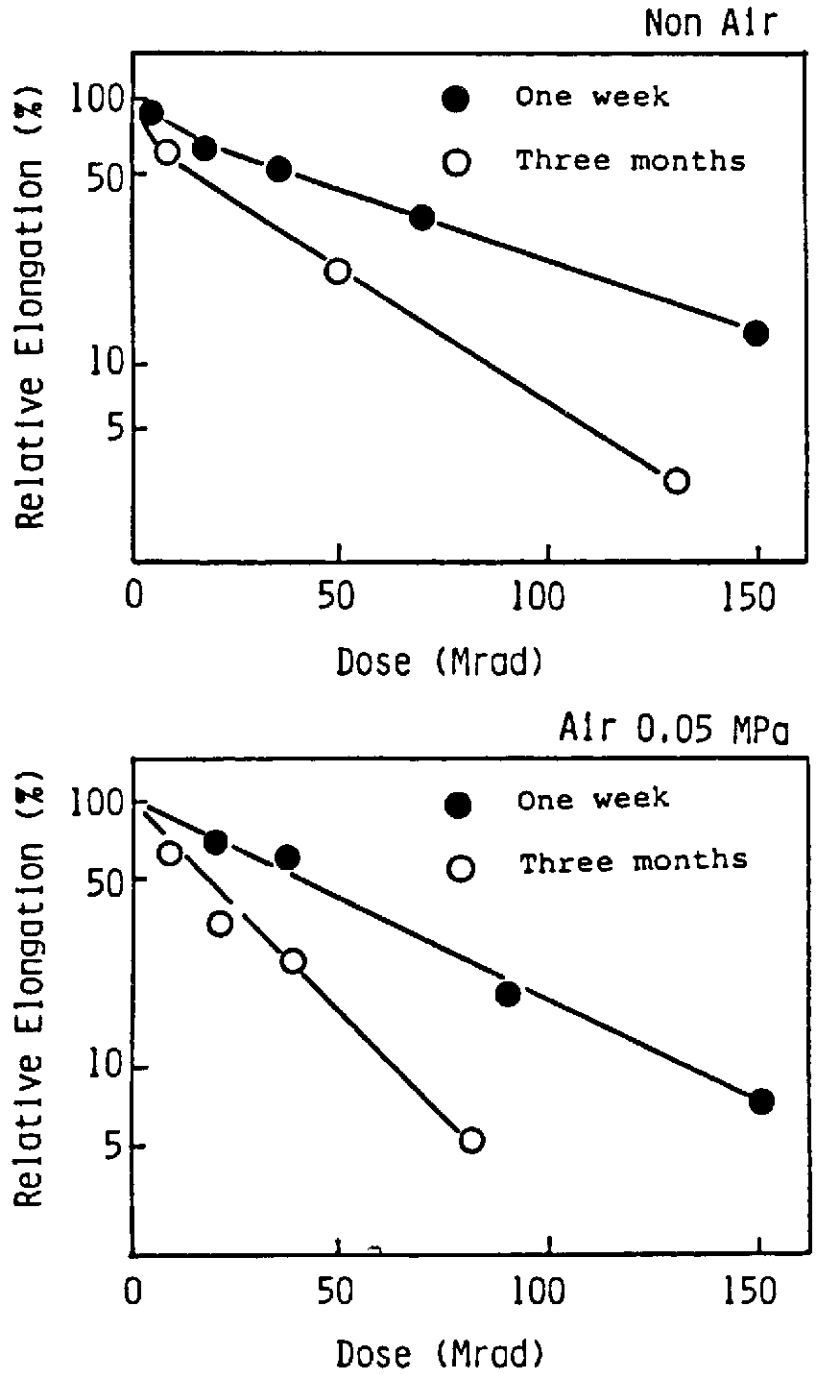

Figure 1.8.2 Ultimate tensile elongation of CSPE-A as a function of dose under LOCA condition. (From Reference 1.8.3)

five orders of magnitude after the three-months test, whereas the drop was only two orders for the one-week test.

These results should be taken as a qualitative measure of effect only, since the other variables associated with these tests (presence/absence of oxygen, and LOCA conditions) may have had additional influence on the results, apart from the dose-rate effect alone.

Issue Findings

The current research available would indicate that accelerated methods for postaccident acceleration may be possible. However, applied research is not currently available.

By performing a series of time acceleration, elevated temperature and humidity tests on identical test samples, an assessment could be made of acceleration methods for extended, postaccident, exposure conditions.

In summary then, the findings within this issue are:

1. Oxygen partial pressure might be included in LOCA/DBE simulations if the test items contain material exhibiting known oxygen sensitivity and are located in an oxygen containing environment.

2. Some material and electrical property differences have been observed during post-DBE acceleration experiments.

3. Arrhenius methods may be useful as a basis for postaccident acceleration of LOCA-oxygen tests.

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Program Effort

1.8.1 K. T. Gillen, R. L. Clough, G. Ganouna-Cohen, J. Chenion, G. Delmas, Loss of Coolant Accident (LOCA) Simulation Tests on Polymers: The Importance of Including Oxygen, SAND82-1071, NUREG/CR-2763, Sandia Laboratories, Albuquerque, New Mexico, July 1982.

A. K. T. Gillen, R. L. Clough, G. Ganouna-Cohen, J. Chenion, G. Delmas, The Importance of Oxygen in LOCA Simulation Tests, SAND82-2144J, Nuclear Engineering and Design, 74 (1982), 271-285.

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1.8.2 D. T. Furgal, E. A. Salazar, and C. M. Craft, Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests, NUREG/CR-3863, SAND84-1264, Sandia National Laboratories, March 1985. (CAP Program results.)

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1.8.3 M. Ito, et al, Progress on Qualification Testing Methodology Study of Electric Cables, NUREG/CP-0058 Vol. 5, Proceedings of the USNRC Twelfth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 22-26, 1984.

1.9 Sensitivity of Accident Simulations to Aging Methods

Issue Description

Safety-related equipment is qualified by tests that simulate aging and postulated accident environments. These simulations rarely reproduce the environmental parameters exactly. In fact, it would be difficult to exactly reproduce aging environments because of the long experimental times (approximately 40 years) that would be required. Other constraints also limit the experimental ability to reproduce accident exposures exactly. Hence a necessary aspect of each qualification program is the choice of appropriate environmental simulation techniques and parameters. This choice should enable the potential equipment user to recognize unique failure or degradation modes that might impact the operation of the equipment during actual use.

Parts of this issue are indirectly addressed by regulatory guidance. For example, 10 CFR 50.49, section e(5), states that "equipment qualified by test must be preconditoned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment." 10 CFR 50.49, section e(7), states that "synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance." Combined, these two requirements suggest that the sensitivity of accident simulations to end-of-qualified-life aging methods should be considered in a qualification test program.

A common industry approach towards simplifying qualification tests is to apply each aging and accident environment sequentially. An important question, therefore, is whether some combinations of alternative sequential aging and accident simulation techniques are better suited for qualification activities than other combinations of simulation techniques. This question has many different aspects, some of which are:

1. How important is age preconditioning to the subsequent performance of Class 1E equipment during accident simulations?

2. Can the choice of aging simulation technique be made independently of the choice of accident technique? Alternatively, can the choice of accident technique be made independently of the choice of aging technique?

3. Can the choice of age preconditioning techniques reduce synergistic and sequencing effects during accident simulations and therefore simplify the accident simulation process?

Industry test efforts to demonstrate that seismic capacity of some equipment is not influenced by age is consistent with not including seismic qualification as part of the 10 CFR 50.49 rule for environmental qualification.

This is the thrust of this effort to address the issue:

> "Does the method, or choice, of accelerated aging affect the behavior of equipment and materials during accident simulation?"

Program Effort

Reference 1.9.1 (RS4445/81/03) reports on equipment qualification research tests for an instrumentation penetration assembly. The penetration assembly was subjected to thermal and radiation aging and LOCA/MSLB steam exposure sequences. Performance was determined by insulation resistance measurements between conductors and ground.

During the LOCA/MSLB accident exposure, several circuits did not maintain a minimum acceptable insulation resistance. During disassembly, subsequent to the exposure, an abnormal distortion in the cable grommets, an elastomeric component in the penetration connectors, was observed. Circuit failures were attributed to grommet extrusion caused by thermal expansion due to high temperatures reached during the tests. Extrusion of the grommet caused insulation to be stripped from cable conductors and ultimately resulted in shorts to ground as water condensed during post-LOCA exposure cooling. Reference 1.9.1 estimated the separate effects of thermal aging, radiation exposures, and accident exposures and concluded that the majority of grommet extrusion occurred during thermal aging.

Reference 1.9.2, NUREG/CR-2763 (SAND82-1071), presents results of a cooperative U.S.-French test program. During this test program, nine materials commonly used as electric cable insulations and jackets in nuclear plant applications were monitored in terms of mechanical properties, weight increases, solubility measurements, and infrared spectroscopy. Three different LOCA simulations and four different aging simulations were employed. For the LOCA simulations, the only planned difference among the three tests was the composition of the overpressure gas. The three gas compositions were nitrogen (0% oxygen), 10% oxygen, and air (21% oxygen). The aging simulations included (1) unaged; (2) 21 Mrad at 880 krad/h; (3) 45 Mrad at 880 krad/h; and (4) 23 Mrad at 24 krad/h.

Reference 1.9.2 noted that aging conditions are a parameter that helps define material response during subsequent LOCA simulations. Figures 1.9.1 and 1.9.2 illustrate test results for EPR and CLPO respectively. For both materials ultimate tensile elongation properties depended on the aging technique (total dose and dose rate) employed prior to the LOCA simulation. Table 1.9.1 provides weight change data for a PVC material. The moisture absorbed during the LOCA simulation depended strongly on the aging technique (total dose and dose rate).

Reference 1.9.3, NUREG/CR-3538 (SAND83-1258), reports on the effects of various combinations of aging and accident simulations for two EPR materials. Each material was preconditioned by eight different aging techniques and then exposed to three different LOCA simulations.

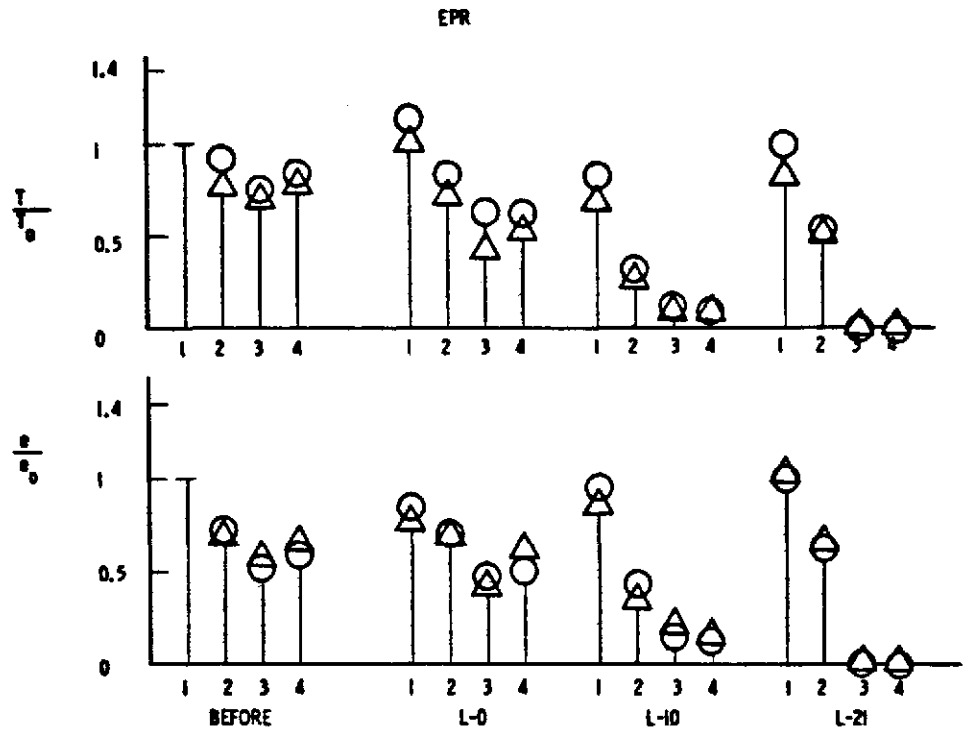


Figure 1.9.1 Tensile property results for the EPR material. The tensile strength, T, divided by its unaged value, T_0, and the ultimate tensile elongation, e, divided by the unaged value, e_0, are plotted after the aging exposure and after the 3 LOCA simulations. The numbers 1, 2, 3, and 4 refer to the four aging conditions: unaged, 22 Mrad at 910 krad/hr; 47 Mrad at 930 krad/h; and 26 Mrad at 27 krad/h, respectively. BEFORE refers to the results before the LOCA simulation, L-0, L-10, and L-21 refer to results after the LOCA simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 1.9.2)

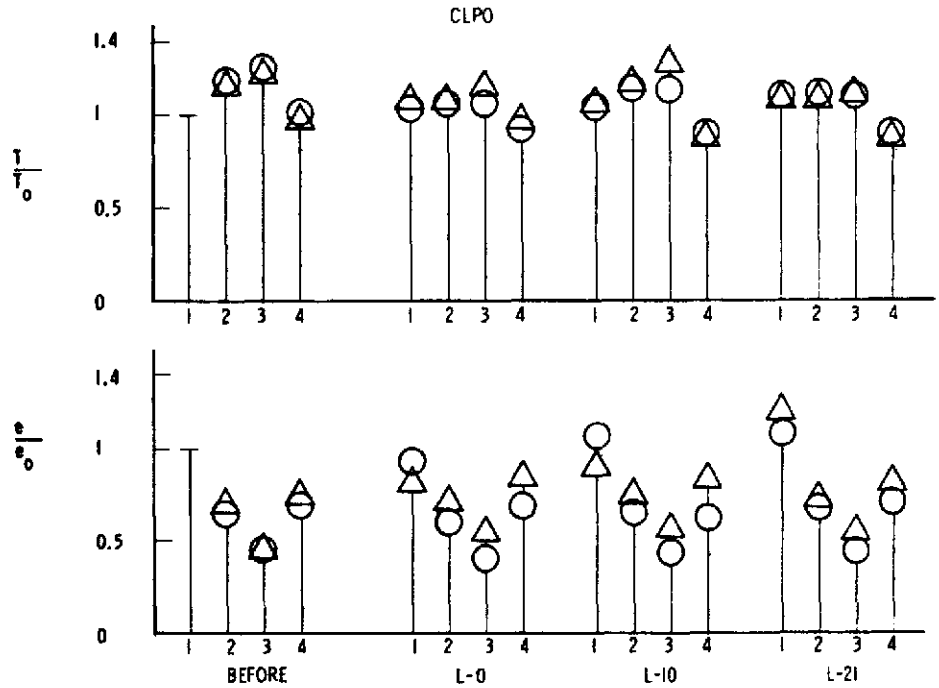


Figure 1.9.2 Tensile property results for the CLPO material. The tensile strength, T, divided by its unaged value, T_0, and the ultimate tensile elongation, e, divided by the unaged value, e_0, are plotted after the aging exposure and after the 3 LOCA simulations. The numbers 1, 2, 3, and 4 refer to the four aging conditions: unaged, 22 Mrad at 910 krad/hr; 46 Mrad at 910 krad/h; and 23 Mrad at 24 krad/h, respectively. BEFORE refers to the results before the LOCA simulation, L-0, L-10, and L-21 refer to results after the LOCA simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 1.9.2)

Table 1.9.1

Weight change due to LOCA simulation versus time after LOCA simulation. Aging Lot #1 refers to unaged PVC material. Lot #2 was aged to 20 Mrad at 830 krad/h. Lot #3 was aged to 42 Mrad at 830 krad/h. Lot #4 was aged to 23 Mrad at 24 krad/h. (From Reference 1.9.2)

MATERIAL	LOCA TEST % O_2	AGING LOT	PERCENT WEIGHT CHANGE AFTER 6 HR	1500 HR	5000 HR
PVC	0	1	+93	+14	-4
		2	+220	+50	-1
		3	+347	+46	-4
		4	+330	+62	-7
	10	1	+53	-2	
		2	+150	-4	
		3	+360	-3	
		4	+230	-4	
	21	1	0	-1	-1
		2	+18	0	-2
		3	+39	+4	-2
		4	+35	+4	-3

Variability among aging techniques included dose rate, aging sequence, sequential versus simultaneous aging, and the duration of the simultaneous aging exposures. The three LOCA simulations were a steam exposure, an irradiation followed by steam exposure, and a simultaneous irradiation and steam exposure. One of the materials, EPR A, experienced complete reversion and loss of original form in response to some combinations of aging and accident conditions. Hence, monitoring weight,

dimensional, and tensile property changes was impossible for some combinations of aging and accident environments. Table 1.9.2 summarizes the visual appearance of the sample groups after completion of a 4-day LOCA exposure. The aging and accident simulation technique combination was important to EPR A's degradation. For example, radiation followed by thermal aging sequences tended to cause more degradation during subsequent LOCA simulations than did thermal followed by radiation aging sequences.

During the efforts described by Reference 1.9.3, EPR-1483 was also exposed to the eight aging techniques followed by the three accident simulations. At completion of the 21-day LOCA exposures, tensile properties, weight changes, and dimensional changes were measured. Results are illustrated by Figures 1.9.3 through 1.9.5. For EPR-1483, there was not a large variation in materials properties caused by alternative sequential aging techniques nor dose rate effects. Rather differences between the simultaneous aging techniques and sequential aging techniques were more important. The simultaneous aging techniques caused more degradation during the accident simulations than did the sequential aging techniques. Moreover, the tensile strength degradation caused by simultaneous aging was amplified by subsequent LOCA exposures. The data also illustrates the importance of aging to total material degradation. For example, specimens unaged prior to the LOCA exposures tended to have lower weight gains and volume increases than did those that were aged.

In addition to EPR A and EPR-1483, Reference 1.9.3 reports on tensile, weight, and dimensional changes for several other EPR materials. However, these other materials were not exposed to the complete set of aging and accident combinations reported for EPR A and EPR-1483. Three other materials, EPR D, EPR F, and EPR 5, were noted to absorb substantial amounts of moisture during simultaneous accident simulations. To differentiate between aging and accident simulation influences on the moisture absorption, both unaged and simultaneously aged specimens were exposed to the simultaneous accident simulations. Results are summarized in Table 1.9.3. The "accelerated age" was clearly important to subsequent moisture absorption during simultaneous steam and irradiation LOCA simulations. Unaged specimens absorbed less moisture than their artificially aged counterparts.

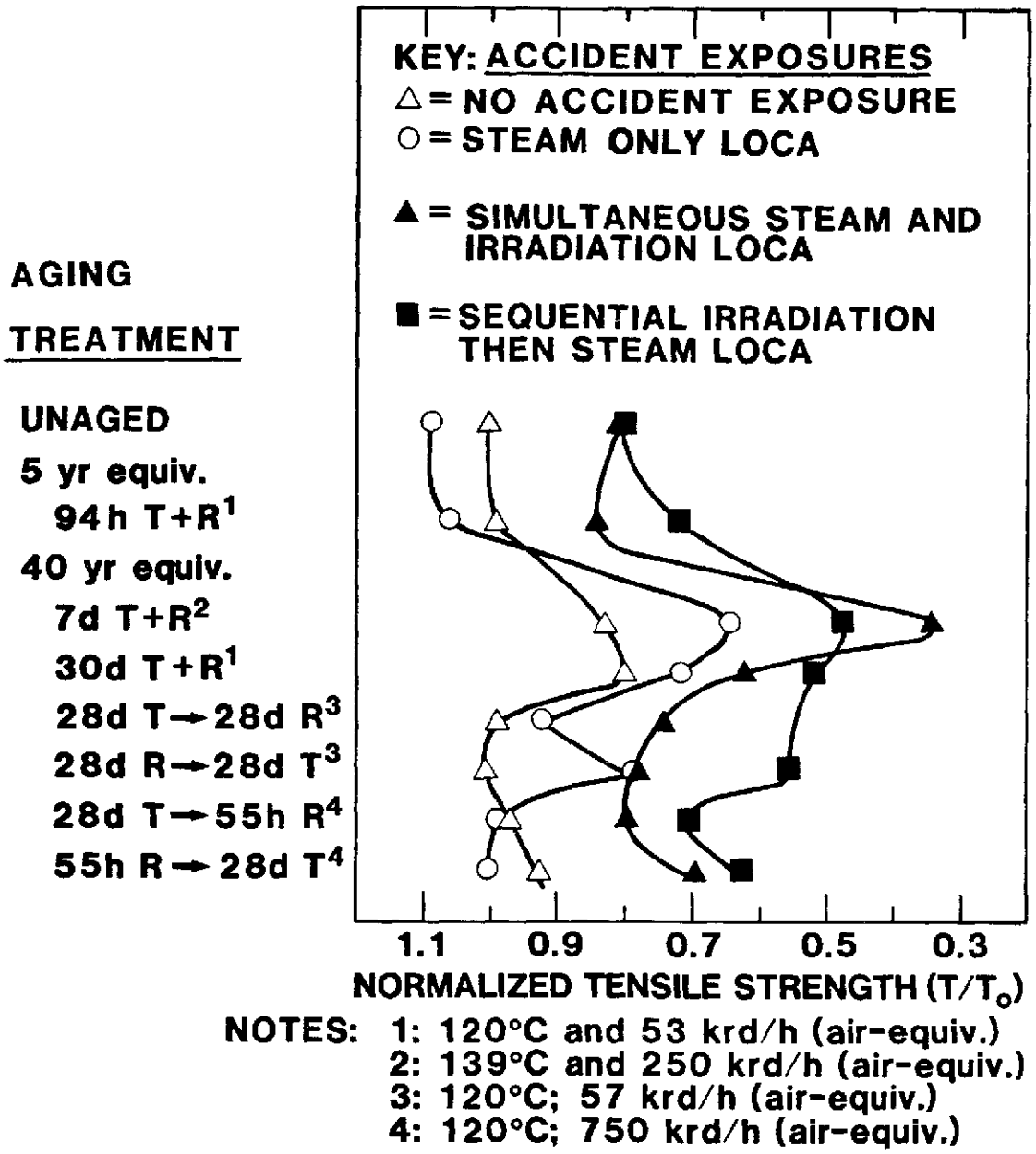
Reference 1.9.4, NUREG/CR-3588 (SAND83-2406), also reports on weight and dimensional increases and tensile property degradation for both unaged and aged cross-linked polyolefin specimens that were subsequently exposed to a simultaneous accident simulation. None of the cross-linked polyolefin products exhibited the amount of degradation reported by Reference 1.9.3 for some of the EPR products. Two of the products, XLPO A and XLPO B, experienced the same effect of aging as did the EPR products, namely, there were larger weight and dimensional increases for the aged specimens compared to the unaged specimens. Unaged XLPO C, in contrast, absorbed more moisture than did the aged XLPO C. For XLPO C, the ultimate tensile strength was enhanced by aging; suggesting

Table 1.9.2

EPR A visual appearance for various combinations of aging and accident simulation techniques. (From Reference 1.9.3)

Aging Method*	Simultaneous Radiation & Steam	Sequential Radiation Then Steam	Steam Only
Unaged	Silverish regions	Relaxation of stresses induced by stripping of wire	--
4 d T + R [1]	Silverish regions; dried out	Blooming and permanent set	Slight drying out
30 d T + R [1]	Slight reversion	Surface cracking Considerable blooming Silverish regions	Complete reversion
28 d T + 28 d R [3]	Blooming Small permanent set	Severe blooming Mild permanent set	Relaxation of stresses induced by stripping of wire
28 d R + 28 D T [3]	Complete reversion	Surface cracking Some reversion near ends Loss of ingredients	Complete reversion
28 d T + 55 h R [4]	Dried out Small amount of permanent set	Blooming Slight reversion	Some permanent set Some stress relaxation
55 h R + 28 d T [4]	Considerable permanent set Partial reversion but rentention of form	Severe blooming Some reversion but retention of form	Almost total reversion
7 d T + R [2]	Range of reversion from minor to severe	Surface cracking Dry surface Silverish regions	Severe surface cracking Substantial reversion but some retention of form

NOTES:
1: 120°C and 53 krd/h (air-equiv.)
2: 139°C and 250 krd/h (air-equiv.)
3: 120°C; 57 krd/h (air-equiv.)
4: 120°C; 750 krd/h (air-equiv.)



NOTES: 1: 120°C and 53 krd/h (air-equiv.)
2: 139°C and 250 krd/h (air-equiv.)
3: 120°C; 57 krd/h (air-equiv.)
4: 120°C; 750 krd/h (air-equiv.)

Figure 1.9.3 Effect of aging and accident techniques on normalized tensile strength of EPR-1483. R refers to aging irradiation. T refers to elevated temperature exposure. Unaged specimens were included in each of the accident simulations, the results are presented as the top row of data points in the figure. The effects of aging alone are plotted with the Δ symbol. The unaged Δ data point has a T/T_o value of one, as expected for specimens not exposed to either aging nor accident exposures. (Based on Reference 1.9.3)

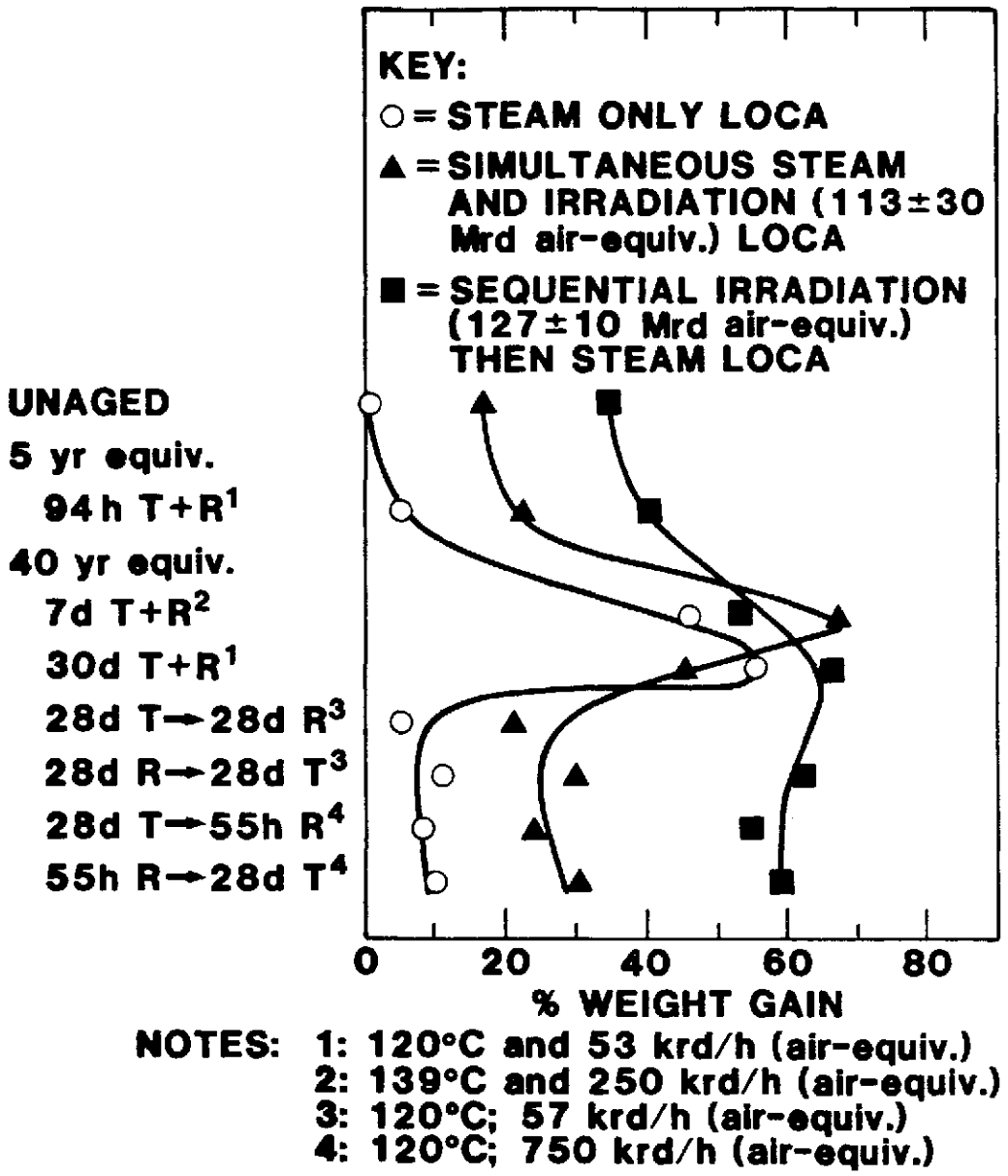


Figure 1.9.4 Effect of aging and accident techniques on % weight gain of EPR-1483.
R refers to aging irradiation
T refers to elevated temperature exposure
(From Reference 1.9.3)

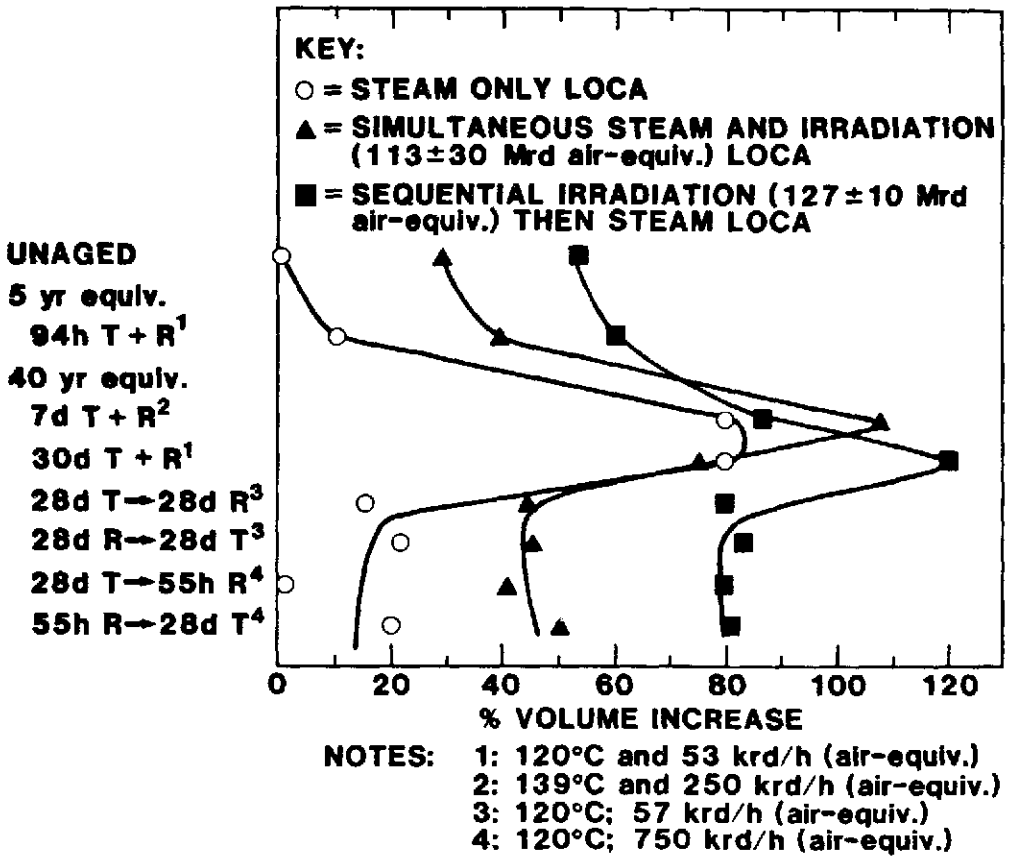


Figure 1.9.5 Effect of aging and accident techniques on % volume change of EPR-1483
R refers to aging irradiation
T refers to elevated temperature exposure
(From Reference 1.9.3)

Table 1.9.3

The effect of "accelerated age" on EPR moisture absorption during simultaneous steam and irradiation LOCA simulation. (From Reference 1.9.3.D)

EPR Material	Accelerated Age	Moisture Absorption (% Weight Increase)
EPR D	Unaged	16
	40-yr equiv*	172
EPR F	Unaged	20
	40-yr equiv*	94
Japanese EOR-5	Unaged	49
	40-yr equiv*	77
EPR-1483	Unaged	17
	5-yr equiv**	22
	40-yr equiv***	67

* A 7d 139°C thermal exposure with simultaneous irradiation for 6 d to 40 Mrd (air-equiv.).

** A 94 d simultaneous exposure to 120°C and 4.9 Mrd (air-equiv.).

*** A 30 d simultaneous exposure to 120°C and 39 Mrd (air-equiv.).

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additional crosslinking of the polymer matrix. This might explain the reduced moisture absorption for the aged specimen compared to the unaged specimen.

The sensitivity of accident simulations to aging methods is examined by Reference 1.9.5, NUREG/CR-4091 (SAND84-2291). This research effort examined the sensitivity of accident simulations to aging methods as part of a comprehensive test program whose experimental variables included:

(1) sequential versus simultaneous aging exposures; (2) the order of the sequential aging exposures; (3) ambient versus 70°C irradiation temperatures during the sequential aging exposures; (4) the presence or absence of oxygen during the accident simulations; and (5) sequential versus simultaneous LOCA simulations; and (6) ambient versus 70°C irradiation temperatures during the sequential accident irradiations. Mechanical properties for twenty-three U.S. and French polymer materials were monitored. The overall research goal was to evaluate whether some combinations of alternative aging and accident simulation techniques were better suited for qualification activities than other combinations of simulation techniques.

Reference 1.9.5 noted that for some materials the effects of the accident simulation technique were sensitive to the aging method. Figures 1.9.6 - 1.9.9 illustrate some examples. Reference 1.9.5 also noted that for certain materials, the aging method had very little impact on material properties during the subsequent accident simulation. Some examples are illustrated in Figures 1.9.10 and 1.9.11.

Reference 1.9.5 also examined whether age preconditioning techniques might reduce synergistic and sequencing effects during accident simulations. Table 1.9.4 identifies the appropriate sequential simulation procedures for each material included in the Reference 1.9.5 test program. These sequential techniques produced degradation similar to that achieved during simultaneous irradiation and LOCA (with air) accident simulation test exposures except for a fire-retardant French EPDM material (Figure 1.9.8). Reference 1.9.5 considered the simultaneous accident simulation to be the best representation of postulated design basis event accident conditions among those included in the test program. For several materials, an irradiation followed by thermal exposure is the most appropriate age preconditioning technique.

Reference 1.9.6, NUREG/CR-4147 (SAND85-0209), discusses the effect of environmental stress on Sylgard 170 silicone elastomer. It notes that Sylgard 170's response to accelerated single stress thermal aging is not particularly age dependent. However, the elastomer response to an applied compressive force was strongly dependent on the environmental temperature and the degree of material confinement. Figure 1.9.12 illustrates the compressive force versus elapsed aging time for the first 70 days of aging. The 150°C aging exposure was more degrading than was the ambient or 43°C exposure.

After 135 days, the 150°C test specimen had decayed to a background stress of 50 lbs or less. After 187 days, the 43°C test specimen had a remaining compressive stress of approximately 500 lbs. The 43°C aging exposure was then terminated. These two specimens and an unaged specimen were then exposed to a LOCA temperature profile (no steam or chemical spray) and the compressive force was monitored. The unaged and 43°C exposed disc behaved very similarly. Results are shown in Figure 1.9.13. Compressive forces during the early stages of the accident temperature simulation reached 2500 lbs. For the specimen aged

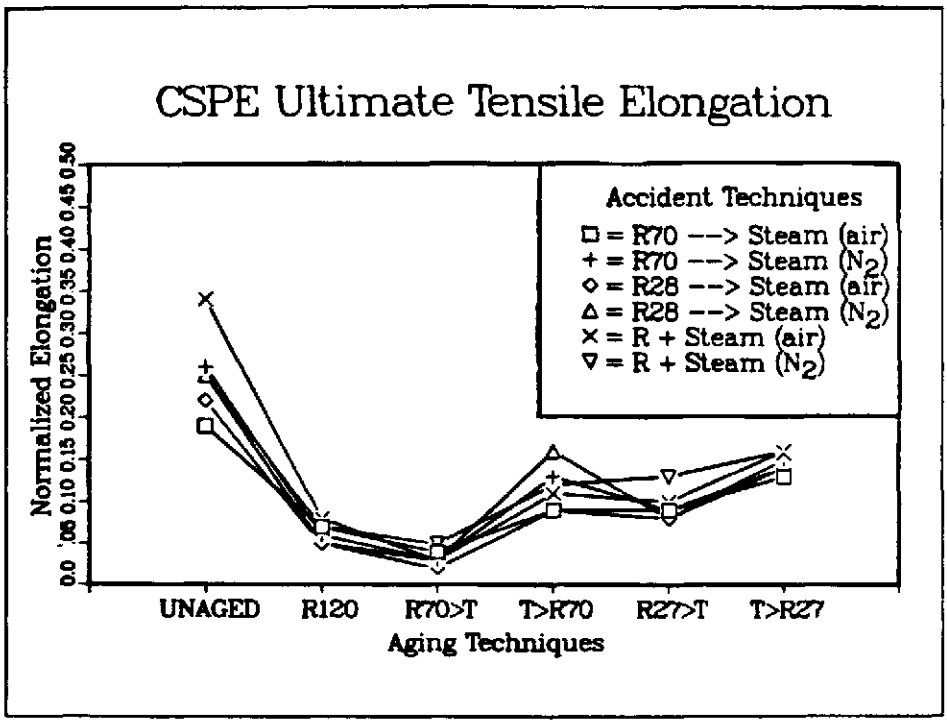


Figure 1.9.6 Ultimate tensile elongation of CSPE at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. R120 refers to a 120°C irradiation. T indicates an elevated temperature aging exposure (16 days at 120°C). The aging irradiation dose was ~25 Mrd; the accident irradiation dose was ~60 Mrd. (From Reference 1.9.5)

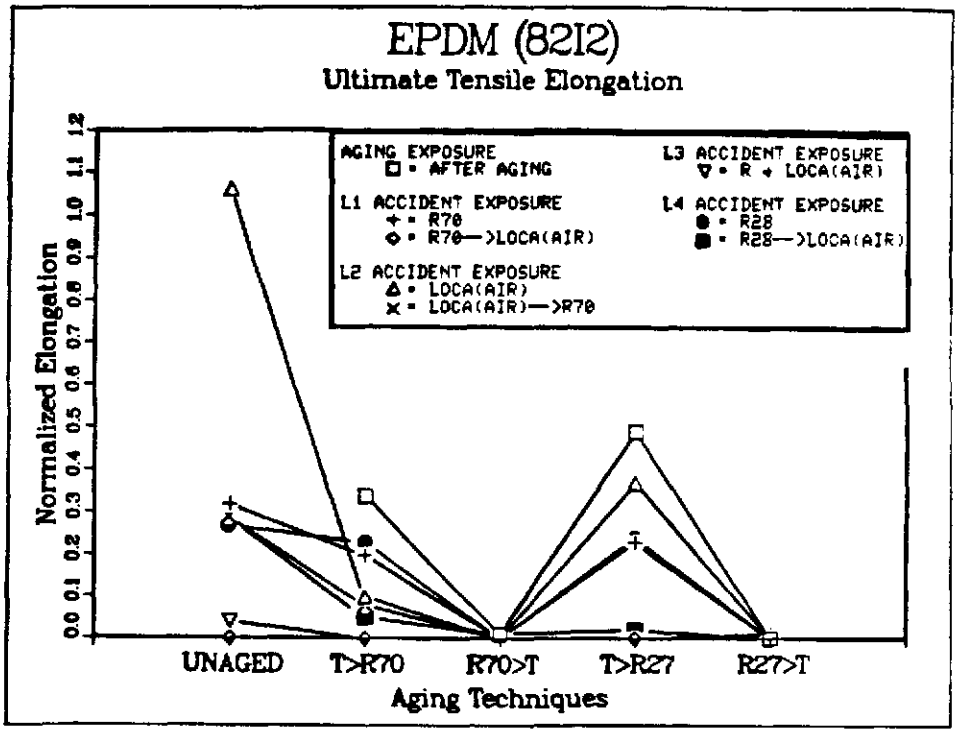


Figure 1.9.7 Ultimate tensile elongation of EPDM (82I2) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure (10 days at 140°C). The aging irradiation dose was ~25 Mrad; the accident irradiation dose was ~60 Mrad. (From Reference 1.9.5)

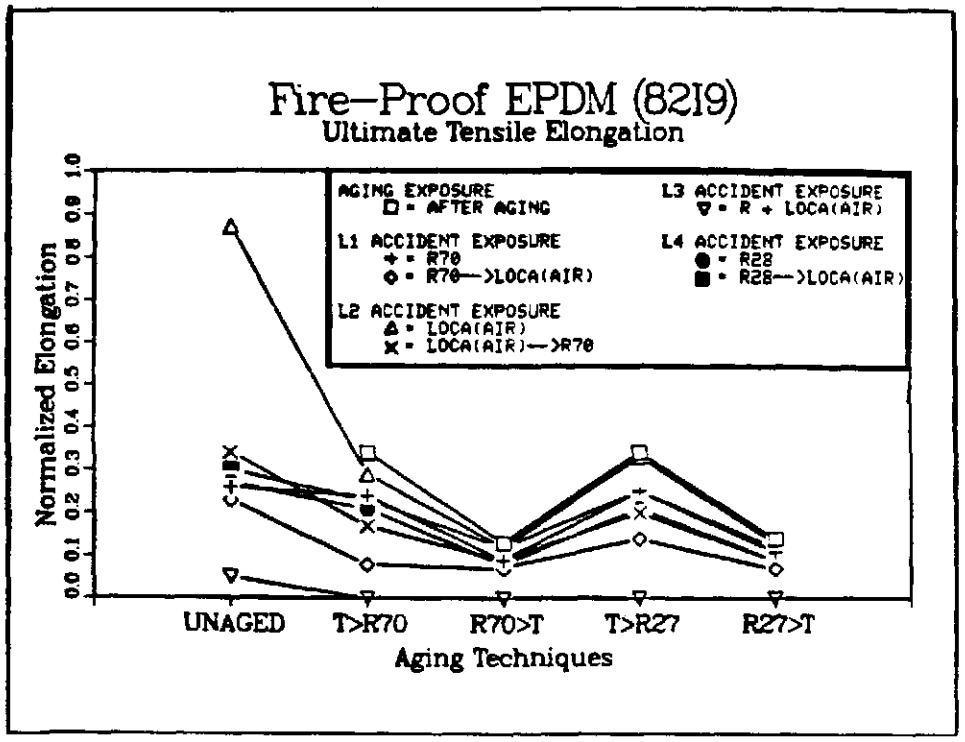


Figure 1.9.8 Ultimate tensile elongation of Fire-Proof EPDM (82I9) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure (10 days at 140°C). The aging irradiation dose was ~25 Mrad; the accident irradiation dose was ~60 Mrad. (From Reference 1.9.5)

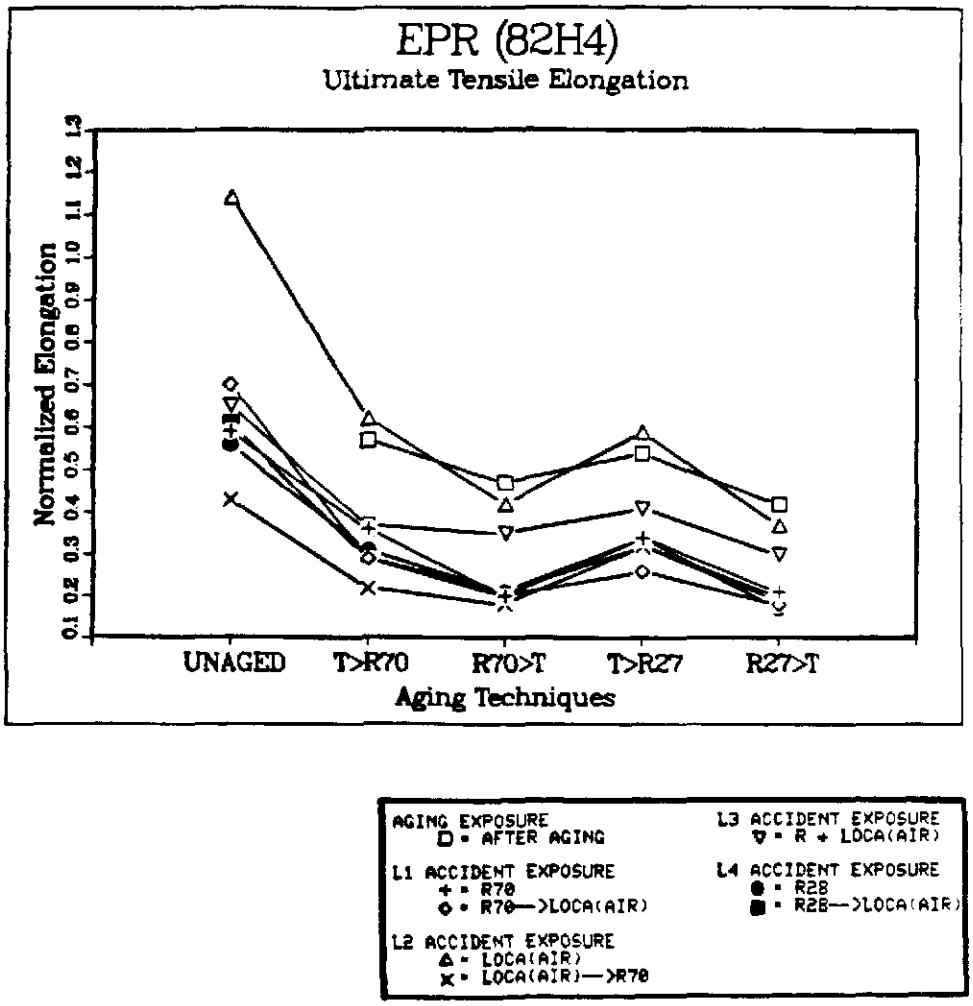


Figure 1.9.9 Ultimate tensile elongation of EPR (82H4) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure (10 days at 140°C). The aging irradiation dose was ~25 Mrad; the accident irradiation dose was ~60 Mrad. (From Reference 1.9.5)

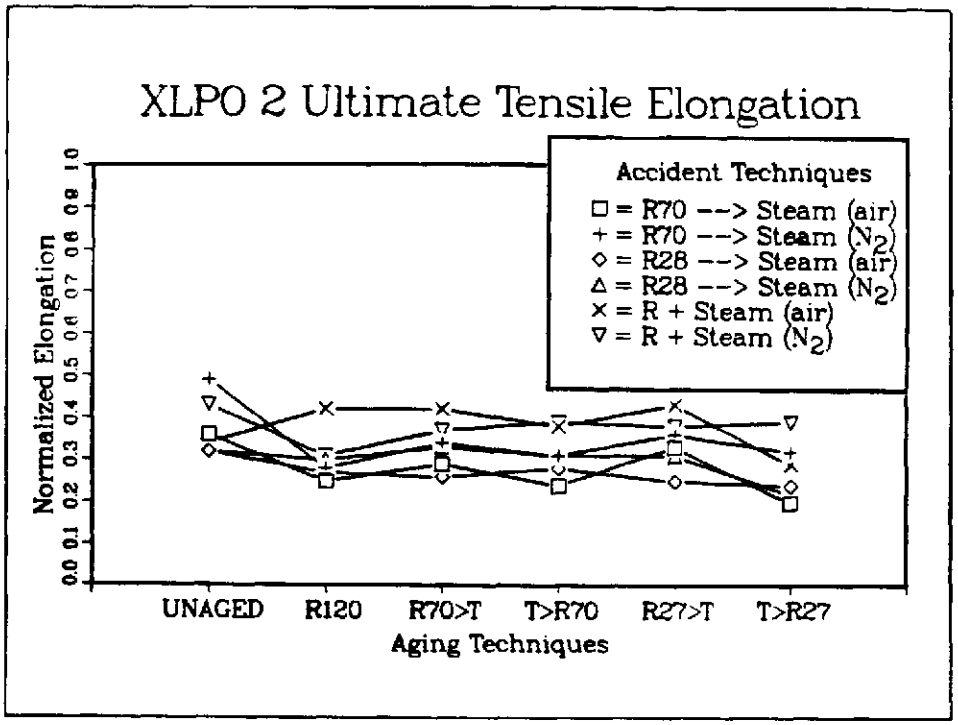


Figure 1.9.10 Ultimate tensile elongation of XLPO 2 at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. R120 refers to a 120°C irradiation. T indicates an elevated temperature aging exposure (16 days at 120°C). The aging irradiation dose was ~25 Mrad; the accident irradiation dose was ~60 Mrad. (From Reference 1.9.5)

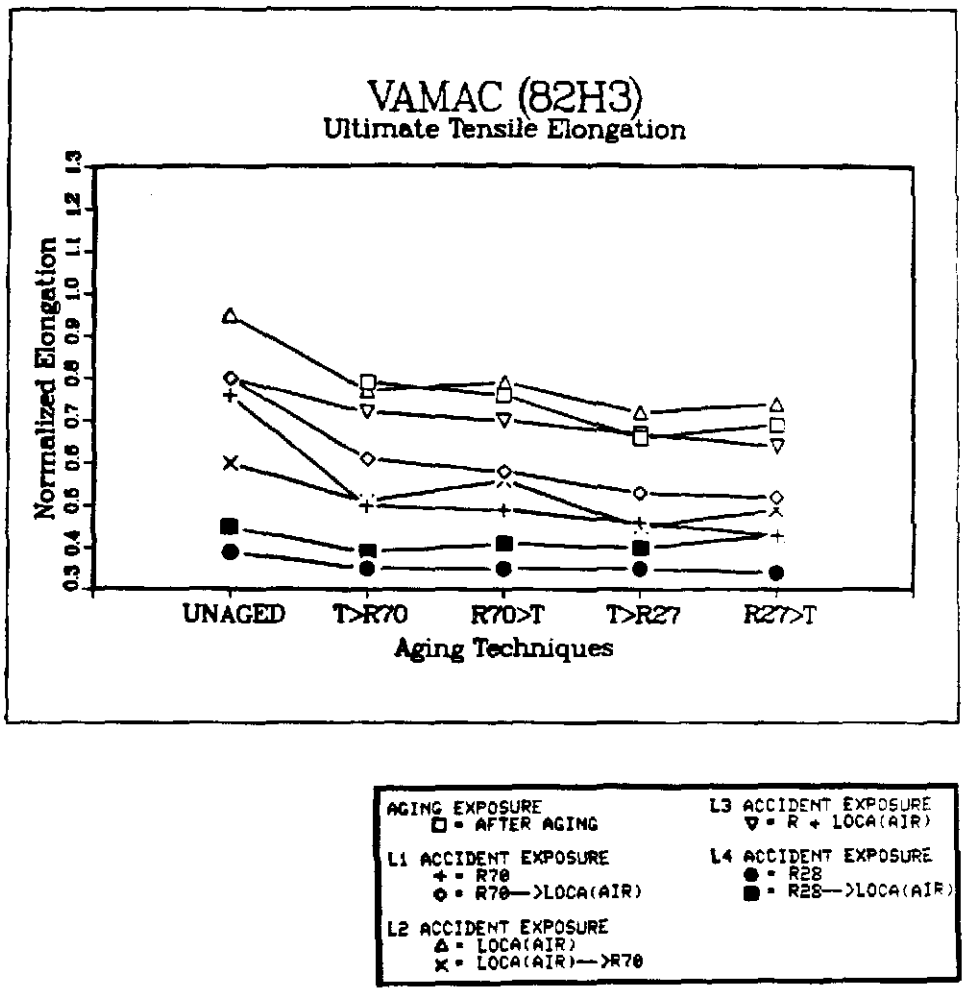


Figure 1.9.11 Ultimate tensile elongation of VAMAC (82H3) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure (10 days at 120°C). The aging irradiation dose was ~25 Mrad; the accident irradiation dose was ~60 Mrad. (From Reference 1.9.5)

Table 1.9.4

Appropriate sequential qualification procedures to simulate an aging exposure followed by a simultaneous accident radiation, steam, and air environment. (From Reference 1.9.5)

Material	Appropriate Sequential Qualification Procedures
Cross-linked Polyolefins	
XLPO 1	Any sequential simulation
XLPO 2	Any sequential simulation
PRC (82I1)	Any aging simulation followed by the R70→LOCA(air) accident simulation.
Ethylene Propylene Rubbers	
EPDM (82I2)	A R→T aging sequence followed by any accident simulation
EPDM(82I9)	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR (82H4)	Any sequential simulation
EPR 1	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR 2	A R70→T, R27→T, or T→R70 aging sequence followed by R70→LOCA(air) accident simulation
TEFZEL	
TEFZEL 1	Elevated temperature irradiations for sequential aging and accident exposures
TEFZEL 2	R70→T aging sequence followed by an accident simulation
Chlorosulfonated Polyethylene	
HYPALON (82G10)	Any sequential simulation
CSPE	R70→T aging sequence followed by any accident simulation.
O-ring Materials	Any sequential simulation
Connector Materials	
PPS (82H6)	Any sequential simulation
Polydiallylphtalate (82H5)	Any aging simulation followed by the R28→LOCA(air) or R70→LOCA(air) accident simulations
VAMAC	Any sequential simulation
CPE	R70→T aging technique followed by any accident simulation

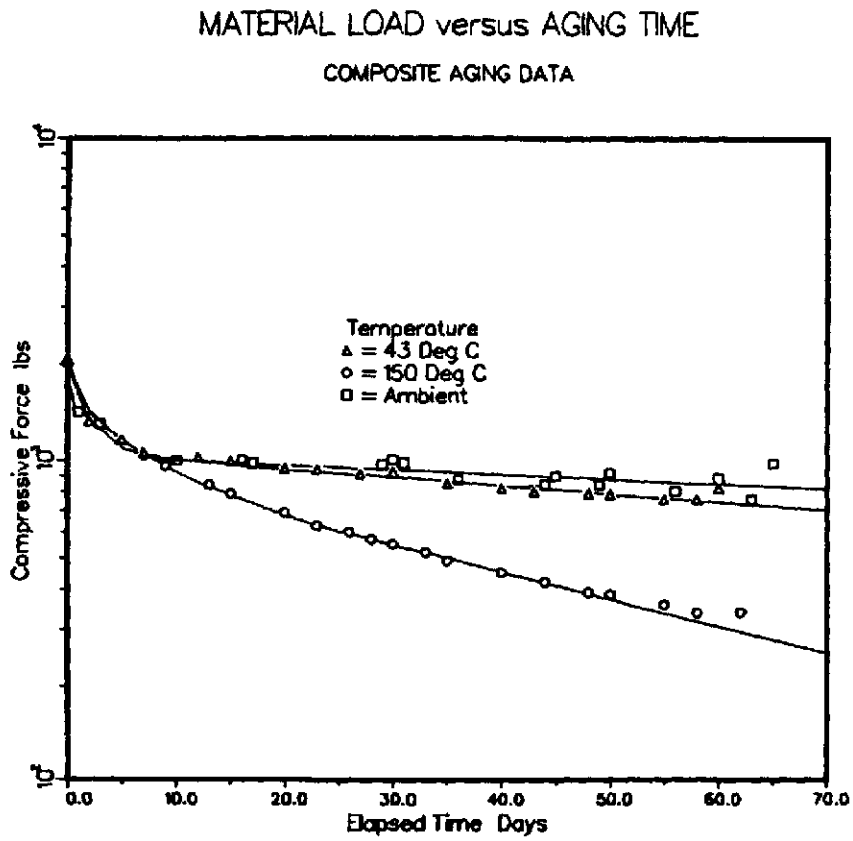


Figure 1.9.12 Compressive force versus elapsed aging time for Sylgard® 170 elastomer confined in a 12-hole fixture. (From Reference 1.9.6)

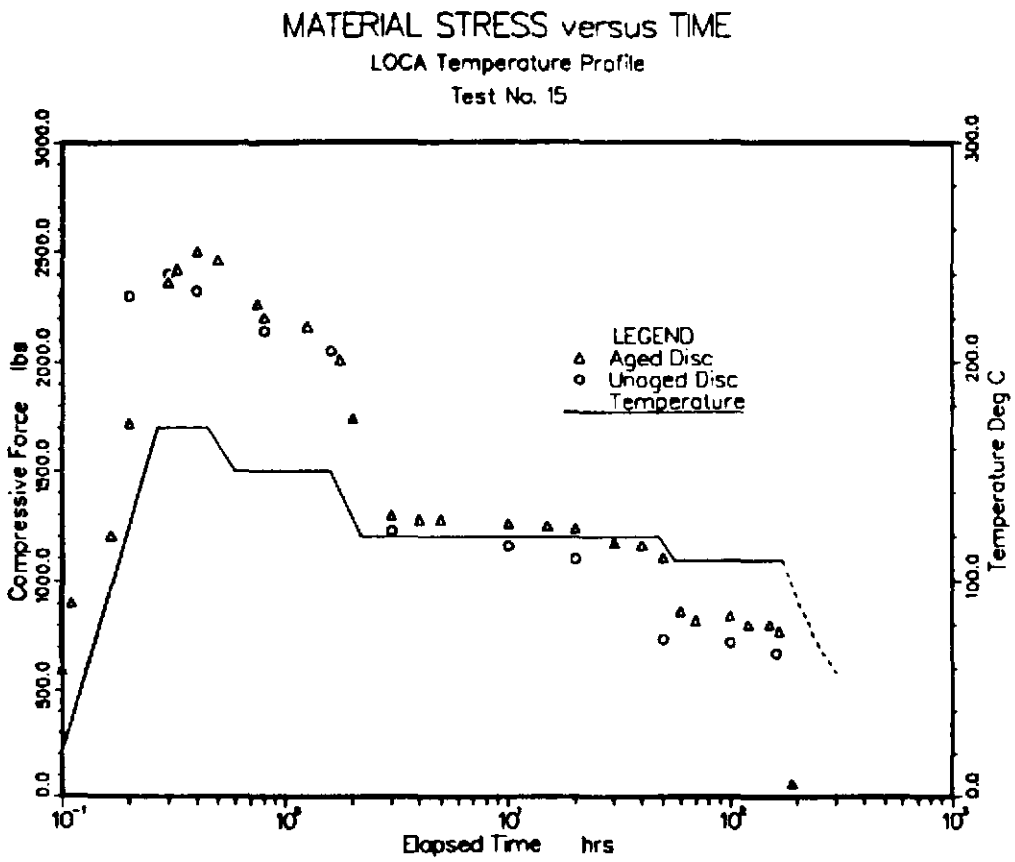


Figure 1.9.13 Compressive force versus elapsed LOCA temperature exposure time for Sylgard 170 elastomer confined in 12-hole fixtures. Aged disc was aged for 187 days at 43°C. Compressive force for aged disc at prior to the LOCA exposure was approximately 500 lbs. The unaged disc was torqued to 2000 lb. compressive force prior to the LOCA exposure. (From Reference 1.9.6)

at 150°C, compressive forces did not exceed 250 lbs during the LOCA temperature simulation (Figure 1.9.14). Thus the method and/or extent of aging (i.e., amount of residual compressive stress at completion of aging) strongly affects the material's response during a subsequent accident simulation.

Reference 1.9.7, NUREG/CR-4536 (SAND86-0450), reports on the effect of a superheated-steam accident test which included insulation specimens of two EPR types. The insulation specimens were preconditioned to a 40-year equivalent life by five different techniques and then exposed to the accident environment. Variability among the aging techniques included dose rate, temperature, and duration of the simultaneous aging exposure. The physical properties and tensile properties, measured after the accident simulation, showed similar behavior at the same dose levels--even though aging occurred at different dose rates and temperatures.

International and Other U.S. Programs

References 1.9.8 and 1.9.9, NUREG/CR-2932, Vol 1 and 2 (SAND81-2027, Vol 1 and 2), provide tensile data for two chemically cross-linked polyolefin insulations. Both materials were exposed to three different aging exposures:

1. A high dose rate (865 krad/h) irradiation to a total dose of 42.5 Mrad followed by thermal aging for 240 hours at 150°C.

2. Thermal aging for 240 hours at 150°C followed by a high dose rate (865 krad/h) irradiation to a total dose of 42.5 Mrad.

3. A low dose rate (48 krad/h) irradiation to total dose of 42.5 Mrad followed by thermal aging for 240 hours at 150°C.

At completion of the aging exposures, tensile properties for some of the specimens were measured. The remaining specimens were irradiated at 865 krd/h, in increments of 50 Mrad, until a total aging and accident dose of 200 Mrad was achieved. At each 50 Mrad increment, tensile properties were monitored. Results for one of the materials are presented in Figures 1.9.15 and 1.9.16. The effects of alternative aging techniques is clearly evident at completion of the aging exposures. However, by completion of the accident irradiation there was very little life remaining and all three aged sample groups were approaching the same level of mechanical degradation. The second material exhibited similar behavior but with much larger scatter in the data.

Tensile property and volume resistivity data after LOCA testing (with and without air) is presented in Reference 1.9.10 for several polymer materials. Both preconditioned and nonpreconditioned specimens were tested. Preconditioning for all specimens was a sequential exposure of thermal aging followed by irradiation (50 Mrad). Both PWR and BWR LOCA conditions were simulated. The total accident radiation exposure for PWR conditions was 150 Mrad while the total accident radiation exposure for

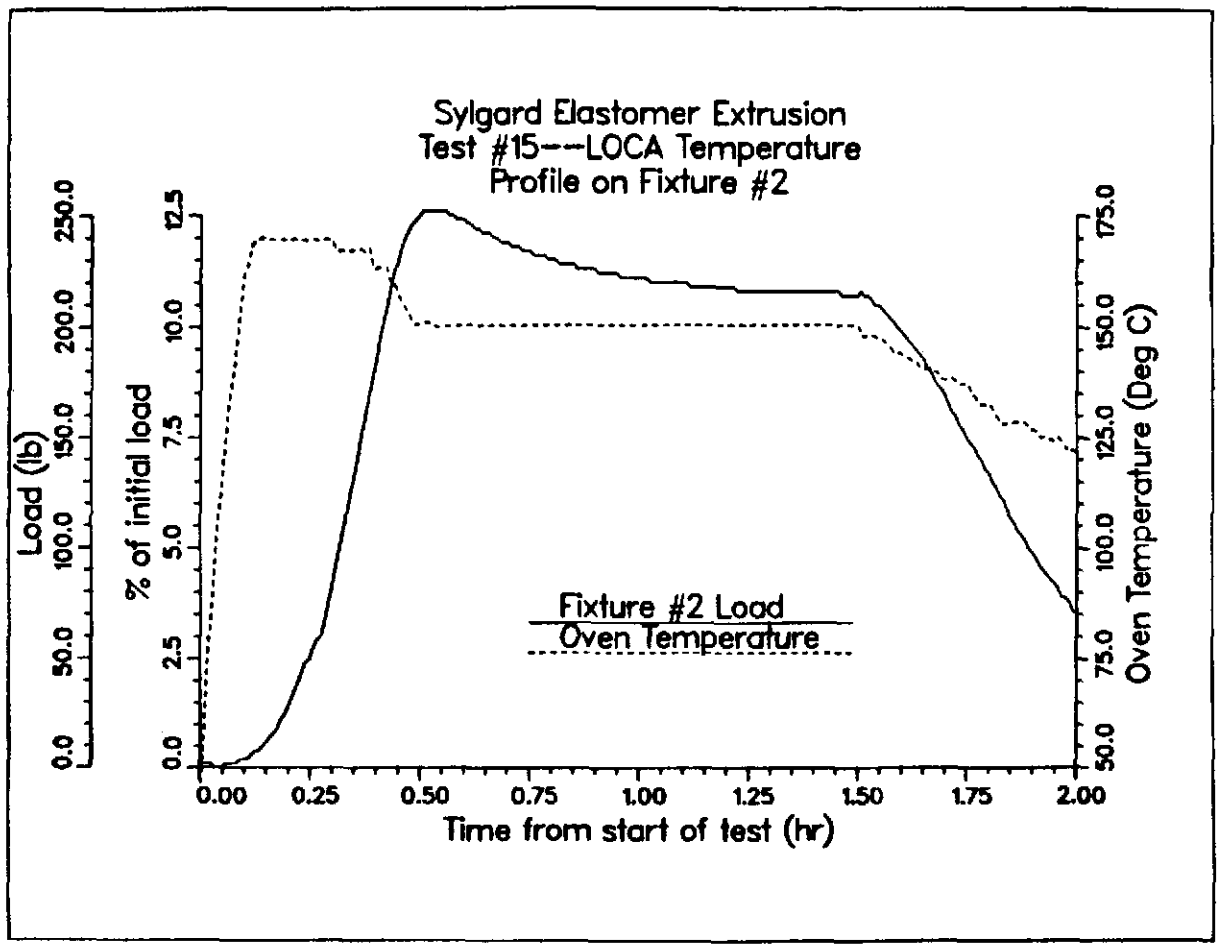


Figure 1.9.14 Compressive force versus elapsed LOCA temperature exposure time for Sylgard 170 elastomer confined in 12-hole fixtures. Aged disc was aged for 135 days at 150°C. After the aging exposure and prior to the LOCA temperature exposure, the compressive load on the disc had decayed to less than 50 lbs. (From Reference 1.9.6)

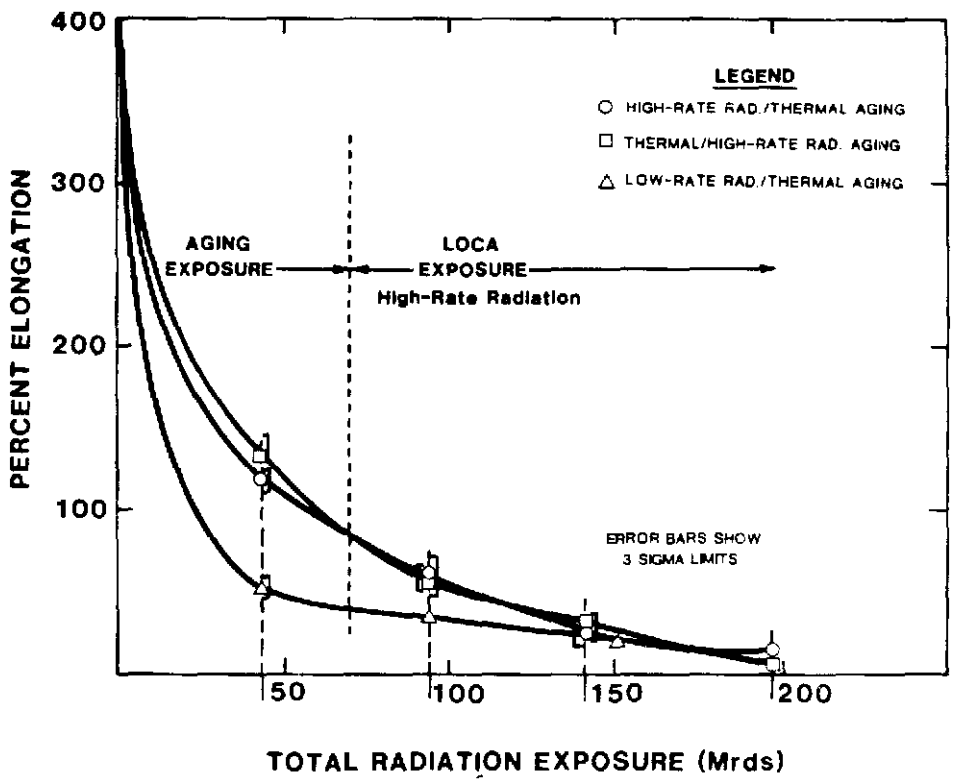


Figure 1.9.15 Ultimate tensile elongation for a chemically crosslinked polyolefin insulation after various phases of aging and accident simulations. (From Reference 1.9.9)

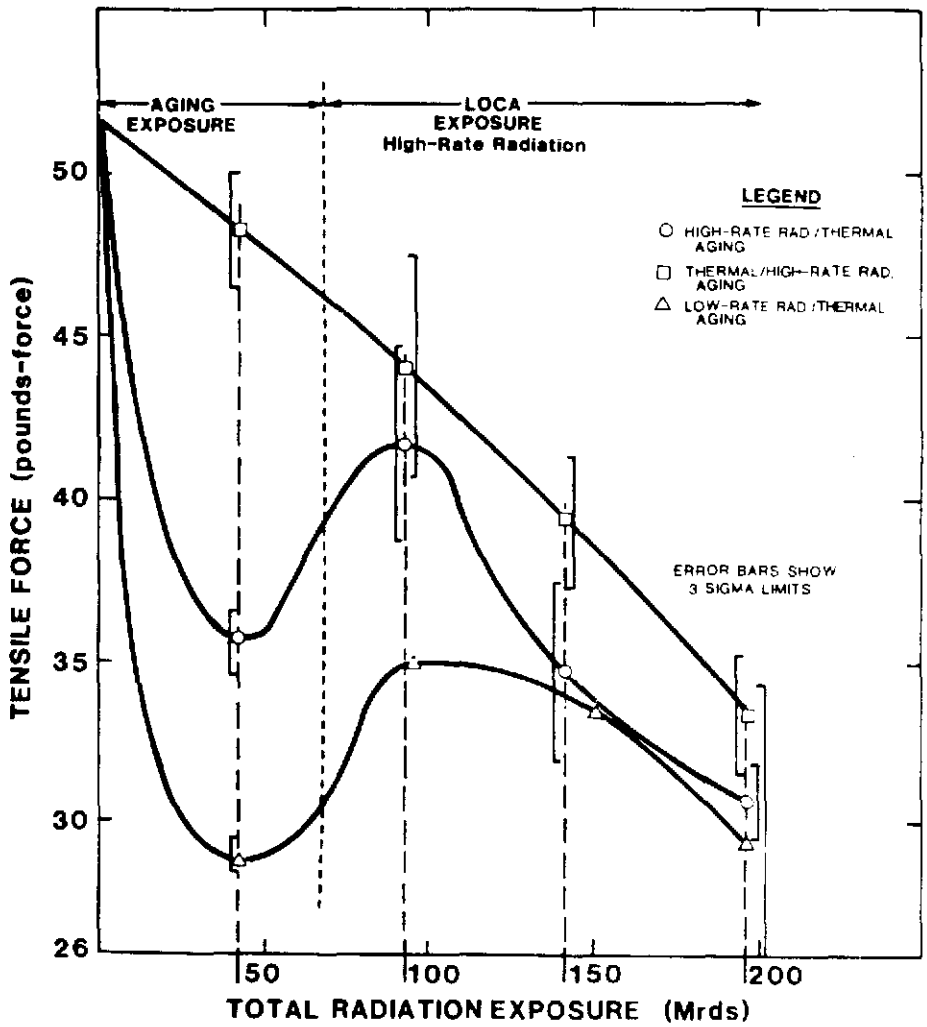


Figure 1.9.16 Ultimate tensile force for a chemically crosslinked polyolefin insulation after various phases of aging and accident simulations. (From Reference 1.9.9)

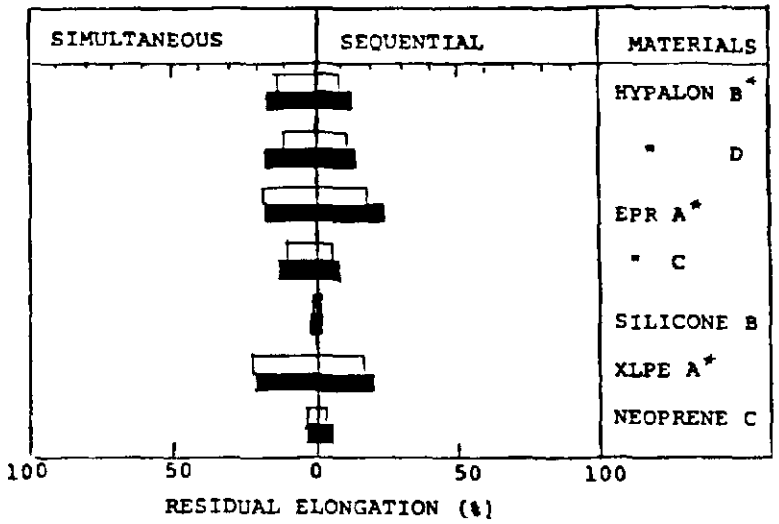
the BWR conditions was 26 Mrad. Simultaneous and sequential accident simulation techniques were employed.

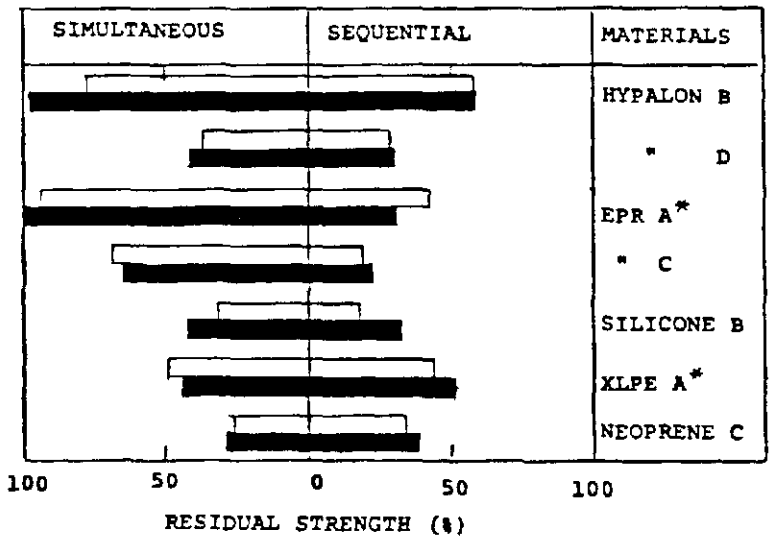
Figures 1.9.17 through 1.9.20 illustrate the comparison of mechanical properties for the test materials. The effect of aging is more noticeable for the BWR simulations. This might be expected since the BWR accident irradiation is approximately half that of the aging irradiation while the PWR accident irradiation is three times the aging irradiation.

Figure 1.9.21 presents Reference 1.9.10 volume resistivity data for several insulating materials at completion of LOCA simulation tests. Simulations were performed both with and with out air in the steam and for preconditioned and nonpreconditioned samples. For EPR, Hypalon, and Neoprene, the volume resistivity was degraded by two to three orders of magnitude by the air containing simulations but only degraded by zero to two orders of magnitude by the non-air simulations. Reference 1.9.10 notes that for the air-containing LOCA environment, the simultaneous method was most degrading for the resistivity of the unaged samples while the reverse sequential was least degrading. For the preconditioned samples, however, the sequential method produced volume resistivity degradation equal to that produced by the simultaneous method. Hence, perceptions regarding which accident simulation methods are most severe are influenced by preconditioning.

The performance of Class 1E pressure transmitters subjected to environmental stresses is reported by Reference 1.9.11, NUREG/CR-3863 (SAND84-1264). Five unaged ITT Barton Model 763 pressure transmitters were exposed to five separate environments. The five environments were (1) simulated LOCA conditions alone; (2) temperature alone; (3) radiation alone; (4) simultaneous radiation and LOCA temperature (no steam conditions); and (5) simultaneous radiation and simulated LOCA steam/chemical spray conditions. Temperature was the primary environmental stress affecting transmitter performance. The four transmitters exposed to a thermal environment all experienced temperature-related and time-at-temperature-related effects. The temperature effects appeared as distinct shifts in transmitter output at exposure temperature transitions above 122°C, and as decreases in these shifts as the time-at-temperature increased.

Reference 1.9.11 identified a potential common failure mode mechanism associated with the zero and span potentiometers. The primary cause of the thermal instability problem was leakage current from the zero and span potentiometers to the transmitter housing. Reference 1.9.11 noted that the only electrical isolation between transmitter circuit elements and the potentiometer case (and hence the transmitter housing) is a nylon rotor assembly. Since nylon has a hydrogen bonded molecular structure, it has a strong affinity to absorb moisture (possibly prior to transmitter assembly). As the temperature was increased, Reference 1.9.11 suggests that the mobility of the hydrogen ions formed from the absorbed moisture increased, causing the dielectric qualities of the nylon to degrade. As time-at-temperature increased, some of the absorbed





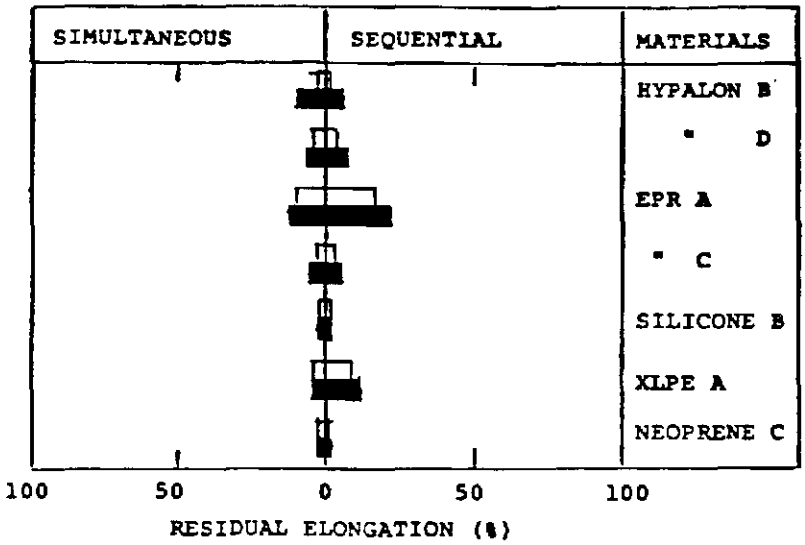
Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (PWR conditions).

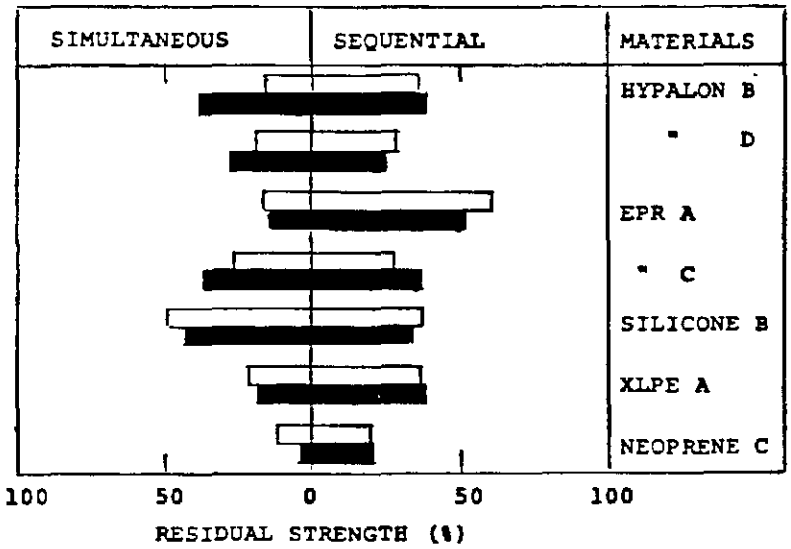
□ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 1.9.17 (From Reference 1.9.10)



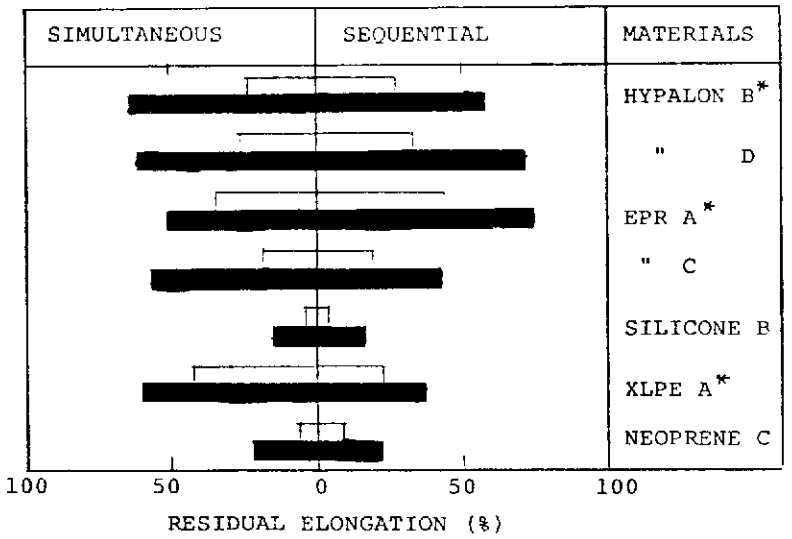


COMPARISON OF MECHANICAL PROPERTIES OF THE MATERIALS AFTER SIMULTANEOUS AND SEQUENTIAL LOCA TESTING (PWR CONDITIONS CONTAINING AIR)

PRE-CONDITIONED

NOT PRE-CONDITIONED

Figure 1.9.18 (From Reference 1.9.10)



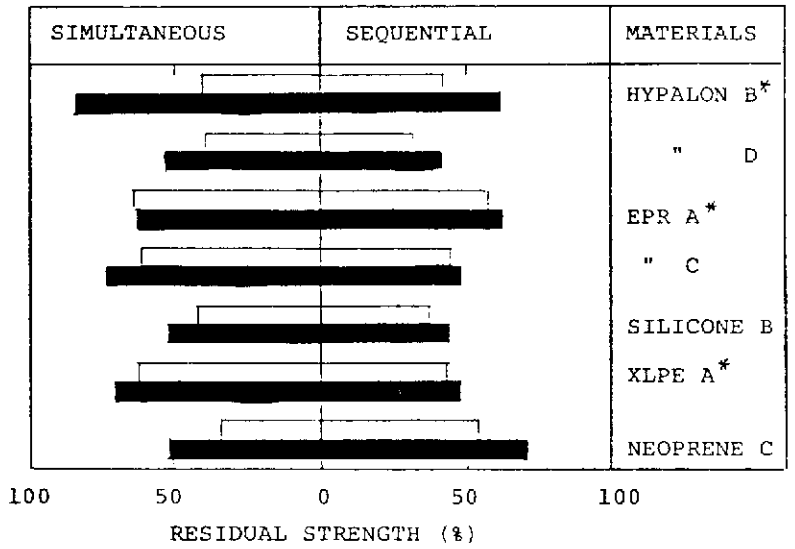


Fig. 14 Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions).

☐ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 1.9.19 (From Reference 1.9.10)

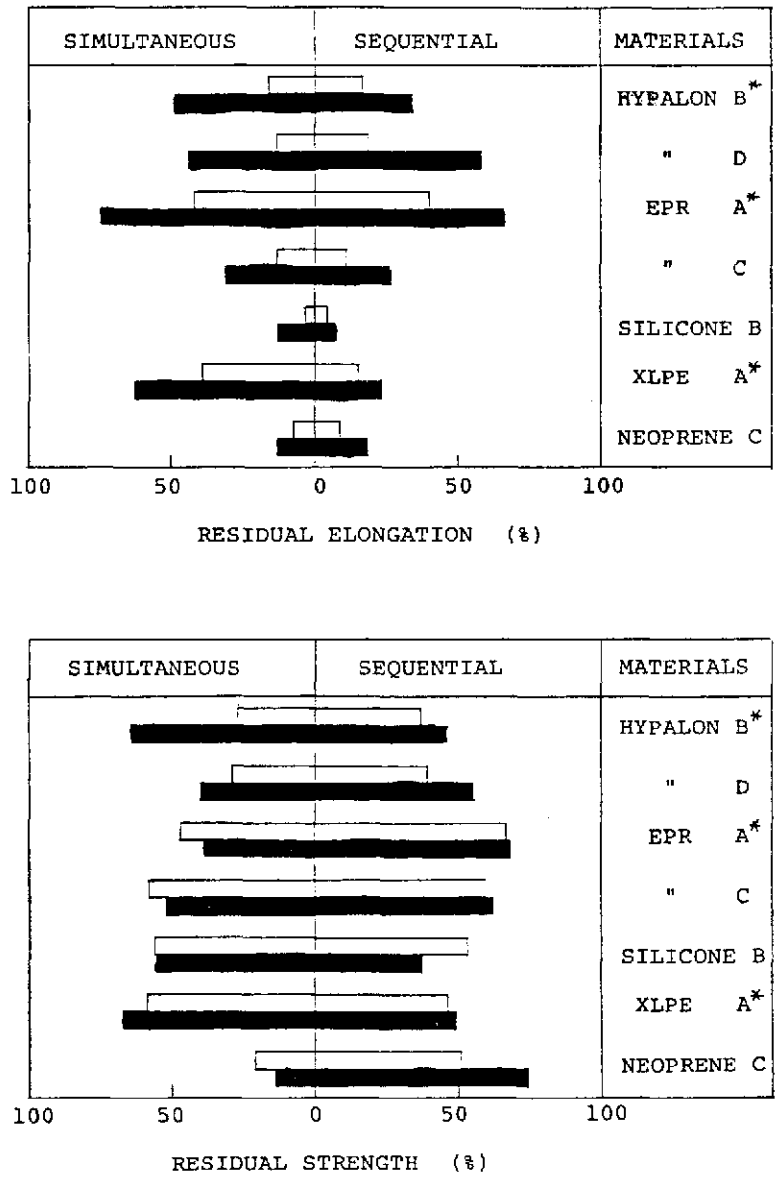
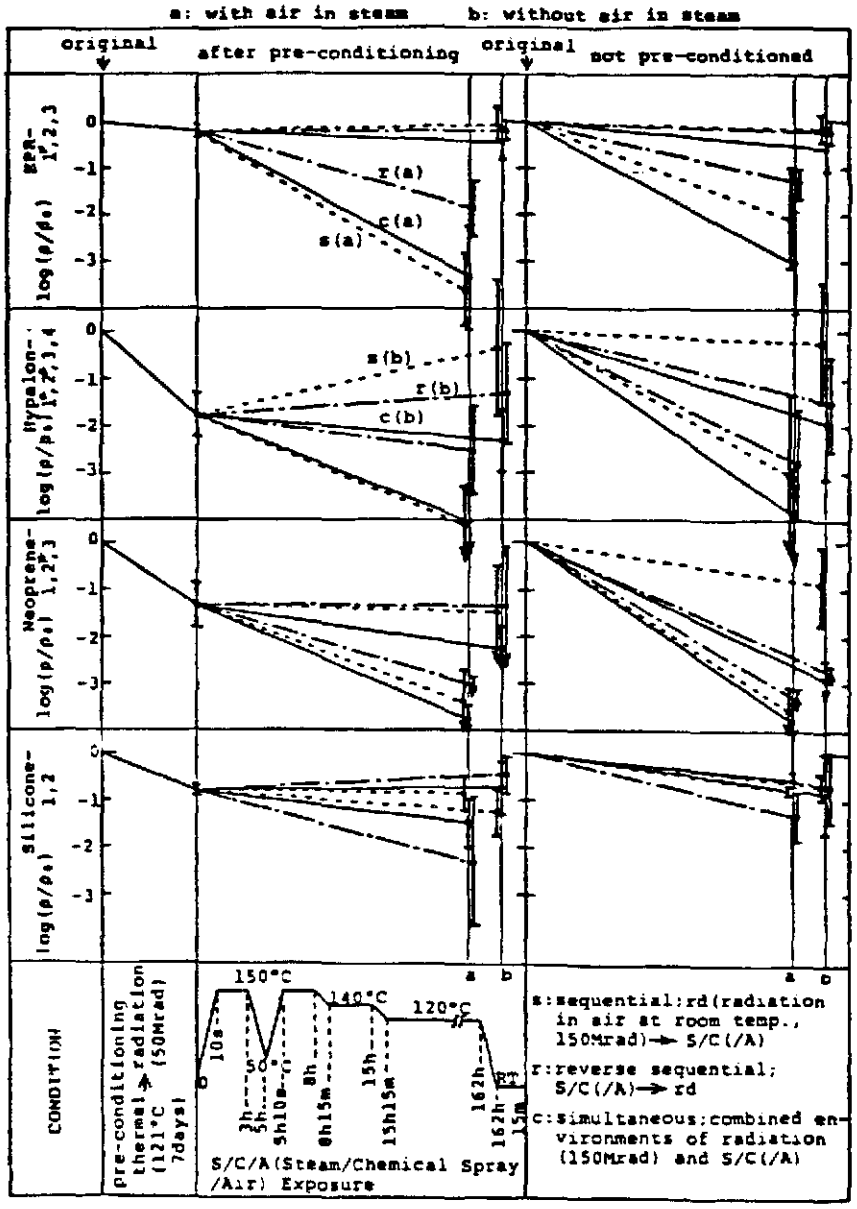


Fig. 15 Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions containing air).

□ ; Pre-conditioned

■ ; Not pre-conditioned

Figure 1.9.20 (From Reference 1.9.10)



Log mean values of volume resistivity for some insulating materials after simultaneous, sequential and reverse sequential LOCA tests.

Figure 1.9.21 (From Reference 1.9.10)

moisture was driven off which partially restored the dielectric qualities of the nylon and improved the transmitter performance.

The observed thermal behavior of the transmitters coupled with the above hypothesized mechanism causing this behavior, lead Reference 1.9.11 to raise the question about whether accelerated thermal aging produces the most vulnerable operational state for this transmitter. Since the magnitude of the transmitter error decreased with time-at-temperature, Reference 1.9.11 postulated that the thermal aging exposure of a qualification sequence may mask or diminish the errors observed during a subsequent LOCA exposure. Thus, thermal aging may not place this model of transmitter in its most vulnerable state prior to the LOCA exposure. The conservative approach is to age, but this is an example of anomalous behavior due to aging.

Such anomalous aging-related behavior is recognized in IEEE 381-1977 where it recommends that an understanding of equipment failure modes is essential to the qualification process. Moreover, obtaining this understanding may dictate that more than one piece of equipment or component may have to be tested such that samples are aged to different degrees of advanced life and then analyzed/tested to establish limiting cases.

Reference 1.9.12 reports on a study of aging effects on the operation of switching devices. An experimental study was conducted to determine whether equipment aging affects the vulnerability of electric switching devices to malfunction caused by vibratory stresses in the range of seismic frequencies and acceleration amplitudes. The study included a vibration test before and after a program of accelerated aging designed to simulate forty years of service in areas outside the containment of a nuclear power generating station. Gamma irradiation, thermal aging, electrical/mechanical cycling and simulation of operating basis earthquakes were included in the program of accelerated aging. Malfunction was defined as spurious opening or closing of contacts for times in excess of one millisecond during the vibration tests.

Reference 1.9.12 reports the fragility level was approximately the same before and after aging for most devices. In some cases the fragility level increased and in others it decreased, but the changes were usually not significantly different from the difference of fragility levels observed for duplicate specimens under identical test conditions. Because of the absence of a clear correlation between accelerated aging and the vulnerability to vibratory stresses, Reference 1.9.12 concludes that the results of the study do not support the requirements that seismic qualification be conducted with aged specimens.

Reference 1.9.13 (NP-3326) discusses the correlation between aging and seismic qualification for nuclear plant electrical components. Seismic tests on new and artificially aged components mounted in a typical electrical cabinet were performed. The report notes that deterioration due to aging does not significantly affect the ability of resistors,

diodes, integrated circuits, transistors, optical couplers, capacitors, and terminal blocks and their interfacing hardware to function during and after a seismic event. The study was inconclusive regarding whether aging degrades the seismic performance of relays.

Reference 1.9.14, NUREG/CR-3424, compares the performance of naturally and artificially aged solenoid valves during accident simulations. No significant differences in performance were noted during seismic tests, but some differences were noted during MSLB/LOCA simulations. During this latter simulation, the naturally aged valves were subjected to less severe aging than the artificially aged valves, but they failed earlier then the artificially aged valves in the MSLB/LOCA test. Only a valve with metal seats performed satisfactorily throughout the program. All other valves tested (both naturally and artificially aged) experienced failure or diminished functional capability as a consequence of degradation of the EPDM components (seals, seats, diaphragms). Degradation manifested itself in the form of severe compressive set, embrittlement, and adherence of tacky material to bounding metal parts. There were also two coil failures.

Reference 1.9.14 concludes that some elements of the artificial aging simulation were nonconservative and other elements were conservative compared to natural aging. One nonconservative element was the use of nitrogen as the process fluid during thermal aging instead of compressed air as in nuclear plant applications. This had the effect of suspending the oxygen-dependent degradation. The exhaust path materials of a valve that were exposed to air during natural aging showed a much higher level of degradation than the artificially aged parts. Cycling the artificially aged valves during thermal aging was considered an overconservative element of the test.

Issue Findings

Several references noted that aging conditions are an important parameter that helps define materials response during subsequent LOCA simulations. References 1.9.1, 1.9.2, 1.9.3, 1.9.6, and 1.9.10 provided examples where the extent of aging affected material properties during subsequent accident simulations. For example, Table 1.9.3 illustrates the importance of the "qualified life" towards the moisture absorption experienced by several EPR materials during LOCA simulations.

Several references provide data indicating whether aging synergistic, sequencing, or dose rate effects that were noted at completion of aging were also observed at completion of subsequent accident simulations. For example, references 1.9.8 and 1.9.9 report the tensile properties for two XLPO products at completion of aging and at completion of accident irradiation exposures. Dose-rate effects observed at completion of the aging exposure were not noted at completion of a subsequent accident irradiation. (Figures 1.9.15 and 1.9.16) Figure 1.9.7 (From Reference 1.9.5) also illustrates that sequencing effects observed for an EPDM material at completion of aging were not observed at completion of

several accident simulations that include exposure to both steam and irradiation. In contrast, other data indicates that aging synergistic, sequencing, or dose-rate effects that were observed at completion of aging continue to be observed at completion of accident simulations. Figures 1.9.3, 1.9.6, 1.9.8, and 1.9.9 provide EPR and CSPE examples where aging sequencing effects were observed at completion of the accident exposures. References 1.9.2 and 1.9.3 provide PVC and EPR examples where aging dose-rate effects were observed at completion of the accident exposures, but Reference 1.9.7 reports that no such aging dose-rate effects were found for two EPR types following a superheated steam accident exposure.

Reference 1.9.5 examined whether age preconditioning techniques might reduce synergistic and sequencing effects during accident simulations. Examples were highlighted where sequential simulation techniques produced similar material degradation as simultaneous techniques. Results are summarized in Table 1.9.4.

The research findings suggest that for equipment items, performance during accident simulations is sensitive to aging and aging methods. For example, Reference 1.9.1's discussion of tests on electrical penetrations concluded that the majority of seal grommet extrusion (which contributed to electrical degradation during the accident simulation) occurred during thermal aging rather than during the accident simulation. Reference 1.9.6 continued the examination of the elastomer employed in these electrical penetrations. It noted that the method and/or extent of elastomer aging (i.e., the amount of residual compressive stress at completion of aging) appeared to affect the material's response during a subsequent accident simulation temperature excursion. Reference 1.9.14 reports that some elements of an artificial aging simulation were nonconservative and others were conservative compared to natural aging of valve actuators. Reference 1.9.14 suggests that these nonconservatisms and conservatisms might have contributed to valve actuator performance during an accident simulation. Reference 1.9.11 suggested that for a pressure transmitter, artificial aging might reduce the effect of a potential common failure mode during an accident simulation.

References 1.9.12 and 1.9.13 discussed the sensitivity of seismic simulations to prior equipment aging. For switching devices, resistors, diodes, integrated circuits, terminal blocks, and their interfacing hardware, these two references demonstrated that aging does not degrade the seismic capacity.

Overall, these results demonstrate that equipment response during MSLB/LOCA accidents may be sensitive to the method of artificial aging. Aging methods may also be an important variable when selecting a sequential aging and accident simulation technique for qualification testing.

While aging appears to be important to accident performance, there are several types of equipment where the sensitivity of accident simulations

to aging methods has not been extensively investigated. These include functional performance of cables, RTDs, pressure switches, sensors, motors, motor operators, radiation monitors and other extended-operating time instrumentation, connection and sealing systems, and possibly additional transmitters and solenoid valves.

In summary, the findings within this issue are:

Demonstration that equipment response during accident simulations may be sensitive to the method of age preconditioning used.

References

Program Effort

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1.10 Hydrogen Burn Influence on Accident Simulation Methods

Issue Description

After the accident at Three Mile Island - Unit 2, the issue of hydrogen produced by the chemical reaction between steam and fuel cladding material received much attention. The original concern was that a hydrogen deflagration or detonation would produce pressure spikes that could rupture the containment. For some reactor designs, it was deemed that insufficient evidence was available for a hydrogen burn rulemaking. Therefore, regulatory action on these designs was deferred "pending completion of NRC- and industry-sponsored research." For other reactor designs, including Mark III BWRs and ice condenser PWRs, hydrogen was considered to be a concern, and the commission issued a rule to address that concern.

The hydrogen control rule was approved on January 25, 1985, and is an amendment to 10 CFR 50. The approved rule requires Mark III BWR and ice condenser PWR facilities to:

1. provide hydrogen control systems (e.g., ignitors) that can handle large amounts of hydrogen

2. demonstrate the survivability/qualification of containment and safety systems during and following a hydrogen burn

3. perform and submit analyses concerning hydrogen control and survivability/qualification of containment and safety systems.

To determine the necessity of qualifying equipment for hydrogen burns, the hydrogen burn environment (i.e., temperature, pressure) to which the equipment should be qualified must first be defined. Second, the effect of the hydrogen burn environment on safety equipment must be determined. Then, from this information, a decision can be made regarding the necessity of modifying qualification methods to include hydrogen burn environments. Here, for example, the effects of aging or test sequence or synergisms would be considerations. And finally, if modifications are required, suitable simulation methods must be developed for demonstrating the qualification of equipment for hydrogen burn environments.

This is the thrust of this effort to address the issue:

"Does the hydrogen burn issue affect the qualification method chosen?"

Program Effort

The effort to date of the QTE Program has been only to monitor the results of other national and international programs. These results can be grouped into:

1. definition of the hydrogen burn environment (i.e., pressure, temperature, detonations, standing flames)

2. effect on equipment survivability

3. miscellaneous activities.

A few representative references are provided in the following discussions to illustrate these ongoing activities.

International and Other U.S. Programs - Hydrogen Burn Environment

The hydrogen burn environment is different from a "typical" IEEE-323 test profile in terms of its transient effects, i.e., the equipment is exposed to high temperatures for short times. The typical LOCA environmental test profile has a 10-second ramp to perhaps 340°F which is maintained for several hours. A hydrogen burn environment is likely to have temperature increases from LOCA temperatures up to 1500°K with a ramp time on the order of 10 seconds. The temperature is not maintained and drops off relatively rapidly (depending on many factors).

The hydrogen burn environment is very dependent on the specific reactor and accident sequence. Containment size and geometry as well as how much hydrogen is generated are very important. Another aspect is how the hydrogen is burned--all in one relatively high-concentration burn or in multiple burns with lower hydrogen concentrations. Standing flames may also be a consideration. References 1.10.1 and 1.10.2 are representative of the research on hydrogen burn environments.

Reference 1.10.1, NUREG/CR-1561 (SAND80-1495), is a compilation of information concerning the possible behavior of hydrogen during hypothetical accidents in light water reactors. The report discusses hydrogen sources and generation, solubility, detection, combustion and containment damage, and recombiners.

Reference 1.10.2, NUREG/CR-2726 (SAND82-1137), is a manual concerning the behavior of hydrogen during both normal operations and accident situations. The specific topics discussed include hydrogen generation, transport and mixing, detection, combustion, and mitigation.

Reference 1.10.3, NUREG/CR-3913 (SAND84-1522), describes HECTR, a computer code designed and used to calculate hydrogen burn environments. It is included here because HECTR was used extensively to calculate the environments used in the studies of equipment response to hydrogen burns. HECTR is a relatively fast-running, lumped-volume containment analysis computer program that is useful for performing parametric studies. The main purpose of HECTR is to analyze nuclear reactor accidents involving the transport and combustion of hydrogen, but HECTR can also function as an experiment analysis tool and can solve a limited set of other types of containment problems.

Reference 1.10.4, NUREG/CR-3912 (SAND83-0501), discusses the predicted pressure-temperature response of an ice-condenser containment for a number of hypothesized severe accident sequences. The base-case accident scenarios were analyzed with the MARCH code to provide steam and hydrogen source terms for HECTR. Both degraded-core and core-meltdown accident scenarios are examined.

Analysis of the hydrogen burn at TMI-2 and the results of EPRI's large scale tests at NTS (discussed below) demonstrated that local variations in the hydrogen burn environment are very important. Reference 1.10.5 (SAND85-0417) contains an analysis of the polar crane pendant cable from TMI-2. The pendant cable was suspended from the polar crane in containment during the accident and was exposed to the hydrogen burn. Although the cable was not safety-related equipment, analysis of the char pattern provided much useful information concerning both the hydrogen burn environment and equipment response to that environment. Data from this study demonstrated that shielding plays a major role, and that the survivability of equipment is very dependant on its specific location within the containment. Equipment located near the containment walls or other massive structures may be protected. Equipment located in large open volumes may be exposed to more severe conditions and could receive higher heat fluxes from the hot gas environment.

Another possible variation in the hydrogen burn environment is the diffusion flame: a standing flame located at a point of entry of hydrogen into the containment (e.g., above the suppression pool in a Mark III BWR). Diffusion flames sometimes occur after an initial global burn is started by the deliberate ignition system and burns hydrogen as it is produced and enters the containment. After the initial transient global burn, the environment resulting from diffusion flames would have slower transients, producing a more gradual rise in containment temperature and pressure. Reference 1.10.6, NUREG/CR-3638 (SAND84-0060), describes Sandia's work, which is continuing, in this area. Although the containment temperature and pressure rise may be gradual, diffusion flames can cause severe environments for equipment because of their long duration (e.g., ~1800s).

Simulating the hydrogen burn environment is difficult; in addition to the surface-area-to-volume ratios being orders of magnitude different (between containments and test simulation volumes), the dominant heat transfer mechanisms--convection and thermal radiation--depend on geometrical factors such as path length and spatial dimension. Reference 1.10.7, NUREG/CR-2730 (SAND82-1150), discusses these scaling considerations. Reference 1.10.8, NUREG/CR-3273 (SAND83-1022), reports on the results of over one hundred hydrogen burn tests conducted in a 5 m^3 cylindrical pressure vessel. The combustion tests were divided into eleven test series, and one test series provides data on the behavior of simulated equipment in hydrogen burn environments.

To test equipment, the hydrogen burn environments may be simulated by means other than actually burning hydrogen in a test volume. Reference

1.10.9, NUREG/CR-3776 (SAND83-1960), describes using Sandia's Central Receiver Test Facility (CRTF), the solar power tower, to provide transient heat fluxes to simulate the thermal environment of a hydrogen burn. This simulation test method has been used for cable, solenoid valves, and pressure transmitters, as discussed in references 1.10.10 and 1.10.11.

International and Other U.S. Programs - Effect on Equipment

Reference 1.10.7, NUREG/CR-2730 (SAND82-1150), provides a general overview of the early results and conclusions from the Hydrogen Burn Survival (HBS) program. Each piece of equipment will respond differently to hydrogen burns because of differences in thermal mass, geometry, and location of the equipment; shielding effects; and the hydrogen burn itself. For a given heat flux environment, a more massive device, such as a pressure transmitter, will have a smaller temperature rise than a thin-walled piece of equipment like a junction box. Equipment located in the middle of a large open volume will receive much larger heat fluxes than identical equipment located near a concrete wall or other thermally massive structure. Cable installed in conduit will experience significantly smaller temperature rises than exposed cable. Figure 1.10.1 shows some typical results from these early studies. A later finding, described below, was that components exposed to multiple burns may see a "ratchet" effect in their temperature response; that is, there may be insufficient time between burns for the component to return to its original temperature.

Reference 1.10.12, NUREG/CR-2864 (SAND82-1684), contains the method used for identifying and selecting the specific safety-related components in ice condenser PWR containments to be studied in the HBS program. The list of equipment is also given; it includes pressure transmitters, valves, flow sensors, EPAs, switches, cables, connectors, seals, terminal blocks, and RTDs and thermocouples.

Reference 1.10.13, NUREG/CR-3521 (SAND83-1715), describes a series of tests conducted at Sandia's Fully Instrumented Test Site (FITS), a hydrogen-burning test facility with a medium-scale volume. The tests provided useful data concerning the hydrogen-burn thermal environment as it relates to equipment survival and about equipment response to hydrogen burns. Findings include:

(1) The temperature rises are not likely to be uniform throughout a piece of equipment (which complicates the survivability analysis). For example, orientation of a particular piece of equipment may play an important role if subcomponents are vulnerable to temperature. Mounting the equipment such that the vulnerable portion is near a wall or other protective structures may be enough to assure survival.

(2) Thick-walled (thermally massive) components respond significantly differently from thin-walled components.

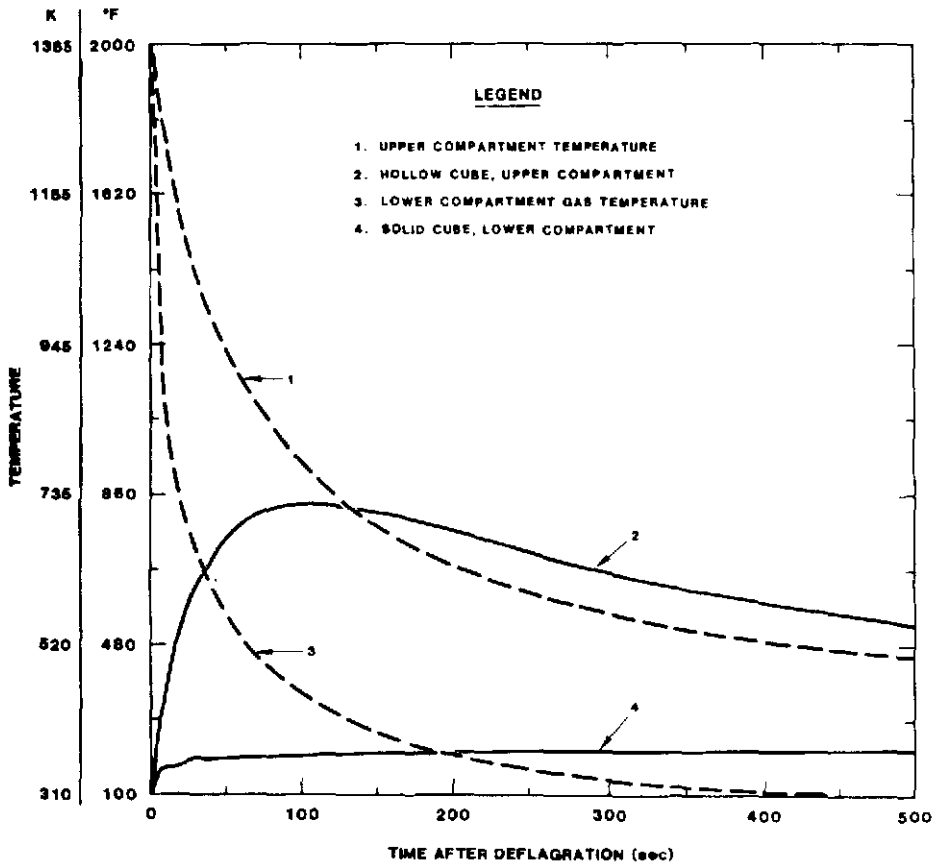

Figure 1.10.1 Predicted Sequoyah gas and hypothetical component profiles resulting from a 10% hydrogen burn in the Sequoyah containment. (From Reference 1.10.7)

(3) Even in this relatively small volume, there were significant local variations in the environment.

(4) It is postulated that ultimate failure of equipment exposed to hydrogen burns is usually due to moisture. The thermal environment may weaken or damage seals and insulation, leaving a path for water to create shorts or grounds.

Reference 1.10.9, NUREG/CR-3776 (SAND83-1960), also describes a series of tests conducted at the CRTF (solar power tower). Using a heat flux profile developed from calculations performed by HECTR, the CRTF simulated the multiple burn thermal environment that HECTR predicted would result from the deliberate ignition of hydrogen generated by a specific accident. Functioning specimens of reactor monitoring and safety system equipment were exposed to this environment. A major finding was that there is insufficient time between burns for the equipment to cool to its original preburn temperature. In general, however, the equipment survived the tests.

Reference 1.10.14 documents some results from the EPRI-sponsored large-scale hydrogen burn tests at NTS. The tests provided data concerning the characteristics of hydrogen burns in large volumes.

Reference 1.10.15, NUREG/CR-3779 (Vol. 1) (SAND84-0160), and Reference 1.10.16, NUREG/CR-3779 (Vol. 2) (SAND83-2579), describe HYBER, a computer program developed as a major portion of the HBS program. HYBER models combustion processes in nuclear reactor containments and estimates the thermal response of safety-related equipment subjected to combustion and postcombustion environments. HYBER was developed for the NRC to use in making licensing decisions regarding hydrogen burns, and it is used extensively to predict equipment response to hydrogen burns in the research effort.

Although HYBER is a useful tool, predicting equipment failure is difficult. In the first place, it is nearly impossible to accurately define a hydrogen burn environment. Once the environmental parameters (such as heat flux) are calculated and the local surroundings (shielding, heat sinks) are defined, established heat transfer computer codes (including HYBER, among others) can be used to predict temperature responses of equipment, but reliable computer models of the equipment must be available. And finally, the heat transfer codes will merely calculate the thermal response of the equipment, not whether it fails or survives.

International and Other U.S. Programs - Miscellaneous Activities

Reference 1.10.17 (EPRI NP-4354) describes the EPRI-sponsored series of large-scale hydrogen burn tests conducted at NTS. The series had two parts, premixed global burns and continuous injection burns (diffusion flames). Several components were tested, including 21 equipment types and 24 cable types. Tests to determine equipment functional performance

were conducted during and after the burns (most cables passed), but the cables were not operated during the burns. Other major findings included confirmation that local variations in the environment are very significant.

Reference 1.10.18 describes the progress of Tennessee Valley Authority's (TVA) research program on hydrogen burn and control. The major elements of the research program include:

- tests of thermal igniters
- experiments and analyses on basic hydrogen combustion phenomena including the effects of steam, turbulence, and flame propagation between compartments
- experiments on hydrogen control methods including water sprays and fogging
- measurements and analyses of hydrogen mixing and distribution in a large compartmentalized volume

The status of each program element varied, thus the results were not comprehensive at the time of publication (September 1981).

Other work is relevant, for example:

1. Reference 1.10.19 (HEDL-TME 82-7, UC-78) presents the results of a literature survey to identify the key issues related to hydrogen formation and behavior under accident conditions.

2. Reference 1.10.20 (HEDL-TME 82-8, UC-78) presents the results of a literature survey to identify the key issues related to hydrogen control systems.

3. Reference 1.10.21 (OH5R8.RK-2) describes a series of tests to take place in a 1/4-scale model of a Mark III BWR. The tests are designed to assess the hydrogen burn environment for this type of containment, and the results will be used in a study of equipment response (although no safety equipment will actually be tested).

There is also considerable international interest in the area of hydrogen burn survival. However, we are not currently aware of any publications concerning equipment survival in hydrogen burns.

Issue Findings

One approach to equipment qualification for hydrogen burns would be to apply the techniques discussed in the references provided, and include sufficient margin to account for uncertainties. Then, if equipment temperatures in a deflagration type burn are shown to be less than the LOCA qualification temperatures to which the equipment has already been

qualified, the equipment could perhaps be considered qualified for that particular hydrogen burn environment. For this case, no further qualification tests beyond the LOCA tests might be required.

However, if the expected equipment temperatures are shown to be higher than the LOCA temperatures, the equipment should be qualified for the more severe hydrogen burn environment, or the installation of the equipment in containment might be modified to protect the equipment from the burn (e.g., install cable in conduit). That is, if the hydrogen burn environment is more severe than the LOCA qualification environment then: (1) the equipment should be qualified to the more severe environment, or (2) the installation of the equipment in containment might be modified, thus reducing the severity of the environment to a level for which the equipment is qualified.

For cases that require equipment qualification for hydrogen burn environments, national and international programs might consider incorporating the burn environment into the sequence of simulation tests (if appropriate) or include the effect of the burn environment by "adjustments" to current simulation methods (e.g., by additional heat, pressure, preaging, or the like).

In summary, the findings within this issue are:

Hydrogen Burn Environment

The work summarized in this section suggests that hydrogen burns in containment may be significant. The hydrogen burn environment is complex, and the survival of safety systems cannot be treated simply.

1. The hydrogen burn environment is dependent on many factors, including accident sequence, volume and geometry of containment, dominant heat transfer mechanisms, and specific location within containment. Global burns can produce gas temperatures as high as 1300K, with pressures up to 400 kPa.

2. Heat flux is one parameter important for equipment survival. The heat flux results from the "cloud" of hot gases from the burn. The heat flux is dependent on the heat transfer mechanisms as well as the gas temperature.

3. The environment is difficult to scale; small volume test facilities have entirely different heat transfer characteristics than large volumes, and geometry also has a major effect on the heat transfer mechanisms.

4. Diffusion flames can cause severe environments for equipment because of their long duration (on the order of 2000s).

Effect on Equipment

Specific methods for demonstrating equipment survival in hydrogen burns have not been established.

1. The thermal response of equipment is dependent on the equipment's thermal mass, its location relative to the burn and heat sinks, its orientation, and local shielding. Equipment surface temperature rises have been calculated to be as low as a few degrees for massive equipment in a relatively mild hydrogen burn environment (i.e., small volume, shielding, and large heat sinks) to as high as 700K for a thin-walled component in a relatively severe hydrogen burn environment (i.e., center of large volume, no local shielding or heat sinks).

2. Components exposed to multiple burns may see a ratchet effect in their temperature response; that is, there may be insufficient time between burns for the equipment to return to its preburn temperature.

3. Predicting equipment failure via analysis is difficult; computer codes can calculate thermal response of equipment (given an accurate environment) but not equipment failures.

References

Program Effort

none

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1.11 Submergence Simulation

Issue Description

Some safety equipment may be located in areas where it may be submerged during its service life in a nuclear plant. Regulatory Guide 1.89 (June 1984) states that "Electric equipment that could be submerged should be identified and qualified by testing in a submerged condition to demonstrate operability for the duration required... ." 10 CFR 50.49 states that the "...qualification program must include and be based on the following: ...(6) submergence (if subject to being submerged)."

Very little guidance exists for conducting submergence tests. Possible questions that could arise include

1. Is a submergence test necessary?

2. What should the submerging fluid be?

3. Are temperature and pressure of submerging fluid important?

4. Is a condensing steam environment a conservative test environment? If so, can condensing steam be used to simulate submergence?

5. Can a submergence test be accelerated?

This is the thrust of this effort to address the issue:

"Is submergence adequately simulated in accident qualification tests?"

Program Effort

Reference 1.11.1 summarizes a brief study on the vulnerability of sealing systems. It suggested that sealing systems are a peculiar source of problems for several reasons:

1. They are ubiquitous; every equipment item depends on a sealing system of some kind.

2. They are generally composed of degradable components (e.g., polymeric and epoxy materials) which are sensitive to aging and accident environments.

3. They have high potential for common-mode sensitivities; their design and construction are generally identical in like pieces of equipment.

4. They can be (largely by virtue of their design) sensitive to maintenance activities; this usually does not directly involve the

seal, but involves removal/replacement/damage of the seal while maintaining other internal components.

5. The "quality" of the seal is not easily detectable, by nondestructive means, especially in situ; hence, the age-degrading functionability of the system cannot be easily verified.

For these reasons, sealing systems may be vulnerable to submergence particularly and moisture intrusion and ingress in general.

International and Other U.S. Programs

Reference 1.11.2, NUREG/CR-3418 (SAND83-1617), a study on terminal blocks, addressed submergence from the following point of view: Would submergence in a chemical spray solution result in permanent contamination of the terminal block surface and cause diminished performance after it was no longer submerged? In the terminal block tests, substantial leakage currents were observed during the submergence test, but recovered to values comparable to nonsubmerged condition values. It was concluded that the films remaining on the terminal blocks that had been submerged were only slightly more conductive than the film introduced to the nonsubmerged blocks by the steam environment only.

The accident at Three Mile Island - Unit 2 caused some equipment to become submerged. About 600 thousand gallons of liquid flooded onto the containment basement (282 ft. elevation) as a result of the accident, containment spray actuation, reactor coolant system degasification, and post accident system leakage. This corresponded to a highly radioactive water level eight feet above the containment basement floor. Some submerged pump motors were determined to be inoperable: sump pump motors, reactor coolant drain pump motors, steam generator secondary side drain pump motors, and leakage transfer pump motors. Non-nuclear instrumentation located on instrumentation racks on the 282 ft. elevation also failed, including pressurizer level, steam generator level, and reactor coolant system pressure instrumentation (from Reference 1.11.3).

Reference 1.11.4 (GEND-INF-056) discusses the results from in situ testing of various electrical circuits at TMI-2. One piece of equipment studied is penetration #R607, which is located at the 282-ft elevation. The water level from the accident was such that it partially submerged the penetration. Fifty-two cables from penetration R607 were tested; of these 48 exhibited anomalous behavior, and 32 were evaluated as being inoperable. Many of the anomalies observed were characteristic of water damage. Crosstalk between circuits was observed, and indications of several broken wires and corroded contacts were found.

Reference 1.11.5 (AEOD/C402) is a study by USNRC staff with several pertinent findings:

1. Corrosion failures of electrical contacts in safety-related switches caused by moisture collection within the switch housing have been a recurring problem at nuclear power plants.

2. Safety-related equipment failures caused by water accumulation within electrical equipment enclosures (e.g., switch housings, terminal blocks, junction boxes, or cabinets) have occurred as a result of water/moisture conveyed within connected conduits and/or entering externally through conduit-to-enclosure air gaps.

3. Failures of safety-related electrical equipment caused by corrosion and/or water collection within protective enclosures have occurred predominantly in mild environments involving high humidity, steam, water leaks or sprays.

4. The more recently reported failures, caused by moisture effects of safety-related electrical devices located outside primary containment structures have occurred mostly at BWR facilities.

5. Failures have involved devices installed in mild environments for which environmental qualification information sheets would indicate they are qualified for service.

6. Failures of safety-related electrical devices (provided with environmental protection) as a result of moisture intrusion, often can be attributed to improper reassembly of enclosures following maintenance or surveillance activities.

7. In situations where cabling ends in open conduits at elevated locations within the reactor building, a potential exists for the conduit to collect moisture at equipment connected to the conduit at lower elevations if the elevated open end is located in a water vapor (or steam) condensation environment or subject to water sprays or leaks.

8. Operating experience shows that failures of safety-related electrical devices, where they have occurred, can often be traced to gaps in the "as-installed" composite assembly of individual components.

Reference 1.11.5 further "... concluded that despite their existing environmental qualification, safety-related equipment fails due to moisture intrusion from various sources..."

Reference 1.11.6 also discusses sealing systems from a moisture intrusion standpoint. Additional work, sponsored by EPRI, is underway.

Issue Findings

It is conceivable that some safety-related components may be subjected to submergence sometime during their service life. Reference 1.11.1

suggests that sealing systems (as part of such components) may be vulnerable to submerged conditions. Two industry studies, References 1.11.3 and 1.11.4, found that some equipment became inoperable as a result of submergence during the accident at TMI-2 while other submerged equipment did not; the equipment types studied were motors, instruments, and electrical circuits. In addition, the potential for equipment failure as a result of moisture intrusion into protective enclosures is discussed in an AEOD (NRC) report (Reference 1.11.5).

In summary, the findings within this issue are:

1. Submerging terminal blocks in a chemical spray/condensate solution did not result in worse leakage currents subsequent to submergence as compared to leakage currents of terminal blocks exposed only to steam condensate; large leakage currents do exist during submergence.

2. Sealing systems may be particularly susceptible to moisture intrusion.

3. Equipment failures due to submergence were reported at TMI-2.

4. A USNRC study "...concluded that despite their existing environmental qualification, safety-related equipment fails due to moisture intrusion from various sources..."

References

Program Effort

1.11.1 Letter L. Bonzon (SNLA) to W. Farmer (NRC), dated September 21, 1984, Sealing Systems Vulnerability Evaluations: Historical Experiences and Program Bases.

International and Other U.S. Programs

1.11.2 C. M. Craft, Screening Tests of Terminal Block Performance in a Simulated LOCA Environment, NUREG/CR-3418, SAND83-1617, Sandia National Laboratories, August 1984. (CAP Program results)

A. C. M. Craft, Performance and Effects of Terminal Blocks in a Loss of Coolant Accident Environment, SAND83-1965C. Presented at the 11th USNRC Water Reactor Safety Research Information Meeting, October 24-28, 1983, Gaithersburg, Maryland. Published in NUREG/CP-0048, Vol. 5, pp. 261-282, January 1984. (CAP Program results)

1.11.3 TMI-Unit 2 Planning Study for Containment Entry and Decontamination, Bechtel Power Corp., July 2, 1979, pp 3-8.

1.11.4 H. J. Helbert, et al, TMI-2 Cable Connections Program - FY84 Status Report, GEND-INF-056, September 1984.

1.11.5 M. El-Zeftawy, Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors, AEOD/C402, USNRC, June 1984.

1.11.6 Sealing of In-Containment Electrical Equipment, Research Project RP1707-12, Private Communication from G. Sliter (EPRI), March 7, 1985.

2.1 Realistic Ambient Environments

Issue Description

To accomplish realistic accelerated aging of a safety-related component, two elements must be known: all actual ambient environments (and their magnitudes), and a reliable acceleration model(s). Generally, industry accelerated aging programs have relied on general, global estimates of these environments rather than equipment location specific estimates. However, this may result in overestimates (i.e., too conservative), or underestimates (i.e., local "hot" spots may significantly affect equipment life); the equipment itself may even produce a local environment, such as self-heating from continuous operation. This is also contained in Regulatory Guide 1.89 (June, 1984), Section C.5.b, "The expected operating temperature of the equipment under service conditions should be accounted for in thermal aging".

Similarly, 10 CFR 50.49(e)(5) states that:

> "Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment..."

This is the thrust of this effort to address the issue:

> "What are the realistic plant ambient environments, and how do they impact the choice of accelerated-aging methods in simulation?"

Program Effort

In the 1977 time frame, effort was directed toward defining research to test a general approach to accelerated aging under combined stress environments. A bases for that work was the study documented in Reference 2.1.1, NUREG/CR-0237 (SAND76-0715). Relative to in-containment plant environments, Westinghouse, Combustion Engineering, and G.E. reports were reviewed; these were the findings:

PWRs: 49°C
0.1 - 200 rad/hr (gamma plus neutron)

BWRs: 65°C
40 - 60% relative humidity
0.3 - 160 rad (carbon)/hr (gamma)
0.1 - 50 rads (ethylene)/hr (neutron)

Reference 2.1.2, NUREG/CR-2877 (SAND81-2613), reports on cable samples taken from the Savannah River K-reactor (not a commercial power

reactor). Here dose rates were measured in the 5-25 rad/hr range at an average temperature of 43°C.

Reference 2.1.3, NUREG/CR-3156 (SAND82-2559), is a survey of the state-of-the-art in aging of electronics. It includes a section on expected in-containment environments:

Temperature:	1. 24°C - 66°C, over an operating cycle. Generally 32° - 38°C 2. 49°C - 54°C, control rod drive area 3. to 94°C, pressurizer shed.
Humidity:	10 - 100% relative humidity.
Vibration:	dependent on positioning of equipment.
Radiaton:	gamma rates: 4×10^{-3} - 7.4×10^{2} rad (tissue)/hr. (10^{3} - 10^{8} rads over 40 years) neutron rates: 4×10^{-6} - 0.54 rad (tissue)/hr.

The values are obviously very location dependent.

Reference 2.1.4 reports on the expected service temperature for devices mounted on hot piping, in this case, RTDs. In a test rig where RTDs were attached to a 200 psig, 387°F saturated steam reservoir, temperatures in the RTD head were measured as a function of insulation, conduit length, and the like. By extrapolation, the service temperature of the RTDs attached to a 600°F source could be 200°F or more, compared to the 125° - 140°F typically assumed.

International and Other U.S. Programs

The most notable attempt to accumulate nuclear plant operating environment data was made by EPRI in a study by Torrey Pines Technology, Reference 2.1.5 (RP 1707-5). The objective was to determine the feasibility of using existing data or future measurements to assess the conservation in current industry practice for specifying the ambient operating environments in various nuclear plant locations. Of the thirty-three nuclear utilities surveyed, eight utilities indicated that no actual measured data was available, and six utilities returned the survey with some measured data. But the study concluded that not enough information was available to reach any generic conclusions and no actual data is presented in the draft report.

A rather dated French report, Reference 2.1.6, purports to provide in-containment radiation values, see Table 2.1.1. We are also aware that the French, and Japanese, have done in-containment environment measurements, but details are not available.

Reference 2.1.7, NUREG/CR-3424, also reports on the concern for service temperature, in this case for solenoid valves. Here the energized coil

Table 2.1.1

Pressurized Water Reactor
Ionizing Radiations in Normal Service

Location	Gamma rays: Dose rate, rad/h	Gamma rays: Integrated over 40 years, rad	Neutrons: Equivalent flux density, rad/h	Neutrons: Integrated over 40 years, rad
Upper base of a primary pump (5)	50	1.7×10^7	0.2 (7)	0.6×10^5 (7)
Control mechanism for the clusters (5)	15	5.2×10^6		
Above the reactor vessel head (1)				
– with a circle of 1 m radius	10	3.5×10^6	0.3 (6)	1×10^5 (6)
– between 1 m and 2.5 m radius	approx. 1	approx. 3.5×10^5	16 (6)	5.6×10^6 (6)
– around the outlet of a hot branch	5	1.7×10^6	80 (6)	2.8×10^7 (6)
Above the reactor vessel outside the shielded port of the primary containment (2)	5	1.7×10^6	40 (6)	1.4×10^7 (6)
Below the reactor vessel, 2 m from the bottom and 3 m from the axis (3)	70	2.4×10^7	600 (6)	2.1×10^8 (6)
Platform at the +20 m level (5)	3	1×10^6		
– environment			0.2 (7)	0.40×10^5 (7
– direct view of the reactor well			10 (7)	3.3×10^6 (7
– RRA motor	7	2.4×10^6 (7)	0.15 (7)	0.5×10^5 (7

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DATA ON THE STANDARD ENVIRONMENTAL FACTORS AND IONIZING RADIATIONS IN THE BUILDINGS OF PRESSURIZED WATER REACTORS -- TRANSLATION

BY: J. ROUBAULT AND M. BARBET

TRANSLATED FROM: ELECTRICITE DE FRANCE, NOV. 30, 1976
REPORT NO. E-SE/EL 75-120 B

(From Reference 2.1.C)

is self-heating, and the resultant temperature must be considered in determining the "ambient" service temperature. Values of coil hot spot temperatures were estimated at 105°C above ambient.

Another often quoted reference for an aging dose value is IEEE 383-1974, Section 2.3.3.3, which states that the specimens "...should be subjected in air to gamma radiation from a source such as Co-60 to a dosage of 5 x 10^7 rad at a rate not greater than 1 x 10^6 rad per hour."

Still another reference for "standardized ambient data" is being developed as Standard ANSI/ANS-56.9, "Environmental Envelopes for Light Water Reactor Nuclear Power Plants." It is currently in draft, but purports to have collected data from eleven AE firms and seven nuclear utilities.

Reference 2.1.8 reports on an ongoing cable monitoring effort at the Oconee Nuclear Station. Various types of cables were installed in life-evaluation circuits in the reactor building. Cable conditions after five and ten years of operation are reported. Average radiation exposures for these two time periods were estimated to be 1.8 x 10^4 Rem and 3.5 x 10^4 Rem, respectively. These calculations were based on estimated operation times. The average radiation exposure rate when the unit was in operation was 650 mrem/h, while the average radiation exposure rate when the unit is off line was 120 mrem/h. The reference notes that the actual exposure level that each individual cable received is considered to have varied considerably over the length of the sample dependent upon the exact location of the cable within the reactor building. Hence, it concluded that the actual exposure level cannot be precisely determined. Average ambient temperatures for the various life-evaluation cables were also estimated. Temperatures ranged from 110°F to 130°F.

Most recently, a new EPRI study being conducted by the University of Connecticut will include (selected) measured plant data, Reference 2.1.9.

Indications of local hot spots can also be implied from several USNRC Inspection and Enforcement Bulletins, Notices, and Circulars issued over the last several years. Damage, particularly cable insulation damage due to higher than expected temperatures, have been noted.

Issue Findings

Certain findings within this issue have been incorporated directly into industry practice. For example, service temperatures must be based on the component in its energized and/or spatial-location state. By evaluating existing plant environment data and performing direct measurements at selected plants and selected plant areas, the expected type and range of ambient environments could be determined. The result would provide for more realistic accelerated aging by focusing on the actual environments and thus selecting the applicable method.

In summary then, the findings within this issue are:

1. The ambient conditions are highly variable within the containment. The peak values may be as high as: 95°C; 100% relative humidity; 200 rad/hour.

2. Measured data seems to be sparse suggesting that actual plant measurements might be made.

References

Program Effort

2.1.1 K. T. Gillen, E. A. Salazar, and C. W. Frank, Proposed Research on Class 1 Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments, NUREG-0237, SAND76-0715, Sandia National Laboratories, Albuquerque, New Mexico, April 1977.

2.1.2 K. T. Gillen, R. L. Clough, L. H. Jones, Investigation of Cable Deterioration in the Containment Building of the Savannah River Nuclear Reactor, NUREG/CR-2877, SAND81-2613, Sandia National Laboratories, Albuquerque, New Mexico, August 1982.

A. R. L. Clough and K. T. Gillen, Investigation of Cable Deterioration Inside Reactor Containment, Nuclear Technology, Vol. 59, November 1982, pp. 344-354.

2.1.3 R. T. Johnson, F. V. Thome, and C. M. Craft, A Survey of the State-of-the-Art in Aging of Electronics with Application to Nuclear Power Plant Instrumentation, NUREG/CR-3156, SAND82-2559, Sandia National Laboratories, Albuquerque, New Mexico, April 1983.

A. R. T. Johnson, F. V. Thome, and C. M. Craft, A Survey of Aging of Electronics with Application to Nuclear Power Plant Instrumentation, SAND83-0658C. Presented at the IEEE Conference on Nuclear and Space Radiation Effects, Gatlinburg, Tennessee, July 18-21, 1983.

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C. F. V. Thome, R. T. Johnson, and C. M. Craft, Aging of Electronics in Nuclear Power Plant Instrumentation: A Perspective on Needed Studies, SAND83-1324C. Presented at the ANS Winter Meeting, San Francisco, California, October 30-November 4, 1983.

2.1.4 F. V. Thome, Quick-Look Report, Tests to Determine Typical Service Temperature Inside RTD Connection Heads, March 25, 1983.

International and Other U. S. Programs

2.1.5 J. T. Bauer, et al, Nuclear Plant Operating Environment Study, EPRI RP 1707-5. Draft dated March, 1982. Private communication from G. Sliter, March 24, 1982.

2.1.6 J. Roubault and M. Barbet, Data on the Standard Environmental Factors and Ionizing Radiations in the Buildings of Pressurized Water Reactors. Translated from Electricite de France, November 30, 1976; Report No. E-SE-EL 75-120 B.

2.1.7 D. V. Paulson, et al, Equipment Qualification Research - Test Program and Failure Analysis of Class 1E Solenoid Valves, NUREG/CR-3424, November 1983.

2.1.8 T. J. Al-Hussaini and J. E. Stoner, Jr., OnGoing Qualification of Cable in a Pressurized Water Reactor Environment; presented at the Nuclear Science Symposium, Orlando, Florida, November 1, 1984.

2.1.9 G. T. Sliter, Status Report to IEEE SC-2, April 15-16, 1985. Private communication.

2.2 Limitations of the Arrhenius Method

Issue Description

The Arrhenius methodology for simulating thermal aging effects has been extensively applied to materials and components for many years, with reasonable success. Because of this, it has become the most commonly accepted method for simulating (thermal aging) lifetimes. IEEE Std. 383-1974, covering type tests for Class 1E electric cables in nuclear power plants, suggests, for example, subjecting coils of insulated conductors "to circulating air oven aging at a temperature and time developed by plotting data using the Arrhenius technique or other methods of proven validity to simulate installed life." Similarly, Regulatory Guide 1.89 states that the aging section of IEEE Std. 323-1974 should be supplemented with the following: "The expected operating temperature of the equipment under service conditions should be accounted for in thermal aging. The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging within the limitation of state-of-the-art technology. Other aging methods should be evaluated on a case-by-case basis."

However, 10 CFR 50.49, Section e.5, does not directly endorse any particular aging methodology.

Because of the acceptance and extensive usage of the Arrhenius approach, numerous reviews have been written about the approximations made and the potential limitations in applying this methodology.

This is the thrust of this effort to address the issue:

"When is the Arrhenius methodology invalid?"

Program Effort

It must be realized that whenever one tries to extrapolate data beyond the range of the available data, there will always be some uncertainty. Only when real-time data exists at the extrapolated conditions can one be completely sure of the validity of an Arrhenius (or any other) simulation. However, in the absence of such direct validations, the confidence level of any extrapolation will depend upon the extent of knowledge of the degradation mechanisms underlying the aging and the quantity and quality of the aging data. The confidence level in the validity of an Arrhenius extrapolation can therefore be increased by improving experimental aging procedures and selecting subsidiary techniques which probe the questions most relevant to determining and following the degradation mechanisms. As shown below, this general approach has been followed in attacking the issue.

Reference 2.2.1, NUREG-0237 (SAND76-0715),

1. discussed typical environments encountered by Class 1E components and aging techniques used to qualify this equipment,

2. reviewed radiation chemistry of polymers and general accelerated aging techniques, and

3. proposed an experimental program on electrical cable material.

Although the thrust of the proposed work was aimed at radiation dose-rate and synergistic effects, some work was suggested in the areas of humidity and thermal aging effects. In order to minimize the chance for changes in reaction mechanisms to occur when thermal aging extrapolations are made, long-term thermal aging studies were proposed.

Reference 2.2.2, NUREG/CR-1466 (SAND79-1561), reviewed and suggested some new approaches towards predicting life expectancy and simulating age of complex equipment. A large part of this document was devoted to thermal aging methodologies in the context of the underlying kinetic mechanisms. In particular it

1. described some of the data analysis techniques necessary to apply the Arrhenius methodology to heat aging studies, and

2. discussed some of the major uncertainties of this methodology, including the potential problems caused by competing reactions, material transitions, oxygen-diffusion effects and sorption effects.

Since these uncertainties can lead to changes in activation energy, the paper recommended aging studies that

1. included long-term exposures so that extrapolation distances are minimized, thereby minimizing any chances for significant changes in slope, and

2. covered as large a temperature range as practical, so that any non-Arrhenius effects might be more easily ascertained.

The suggested data handling techniques involved confirming both that superposition is achieved when thermal aging data is time-temperature superposed and that the functional form of the requisite shift factor has Arrhenius behavior. This superposition concept for shifted data, shown schematically in Figures 2.2.1 and 2.2.2, is equivalent to saying that the Arrhenius activation energy is independent of the extent of material damage, an implicit, but often ignored assumption of the Arrhenius approach.

An alternative but equivalent approach may be used to demonstrate that the Arrhenius activation energy is independent of the extent of material damage. For each degradation endpoint (i.e., 80%, 60%, 40%, 20%, etc. of

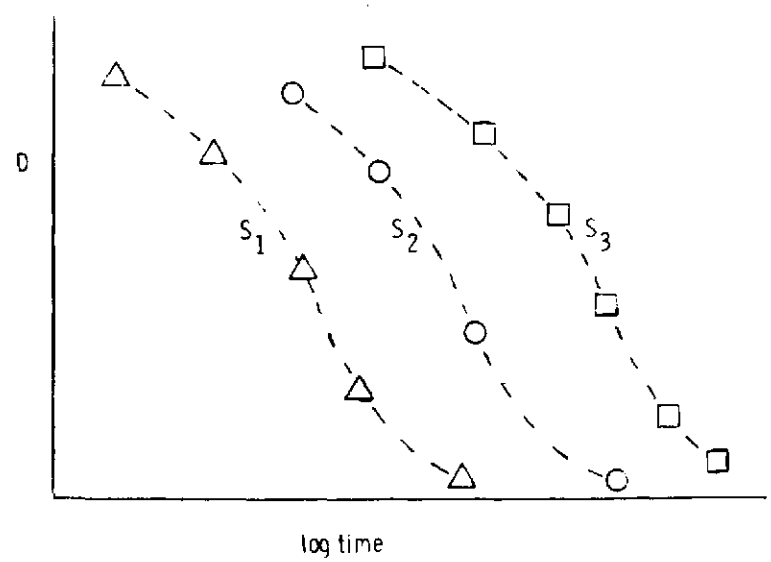

Figure 2.2.1 Hypothetical accelerated aging data, where the damage parameter, D, is followed *vs.* time at 3 different stress levels, S_1, S_2, and S_3. (From Reference 2.2.2)

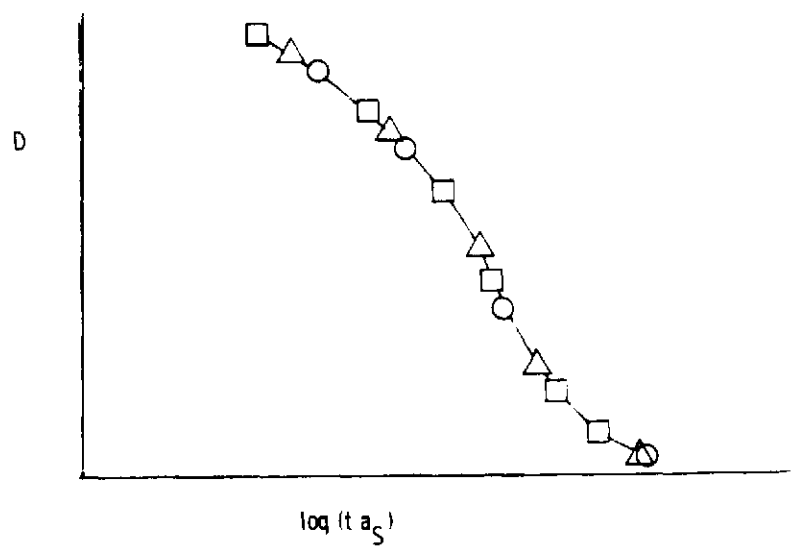

Figure 2.2.2 The data of Figure 2.2.1 after time-stress superposition. All data has been transposed to a reference stress level (usually a reference temperature) by use of appropriate shift factors, a_S. When the different stress levels reflect different temperatures, the shift factor, a_S, is an Arrhenius expression based on an appropriate activation energy. (From Reference 2.2.2)

the original performance) an Arrhenius plot is developed. If all the Arrhenius plots are straight and parallel, then the activation energy is independent of the extent of material damage over the temperature range for which the data was obtained. These two approaches (time-temperature superposition and straight and parallel Arrhenius plots) are equivalent and may be used interchangeably.

Using the concepts outlined above, careful thermal aging studies were conducted on a number of cable materials. Data for a neoprene (chloroprene) material which could be described adequately by the Arrhenius methodology was shown in Reference 2.2.3, NUREG/CR-0696 (SAND78-2254). Figure 2.2.3 shows ultimate tensile elongation data for this material taken at six temperatures ranging from 90 to 140°C. When this data is analyzed using Arrhenius plots, excellent linearity of the data results and the Arrhenius activation energies calculated from the slopes of the lines are independent of material degradation (see Figure 2.2.4). Later data at 70 and 80°C further confirmed the Arrhenius behavior and the excellent superposition for this material.

Data for a Hypalon material which was also Arrhenius over an extended temperature range is shown in Figure 2.2.5 from Reference 2.2.4 (SAND79-2035A). Given (1) the extent of this data, (2) the excellent superposition implied by the data and (3) the small extrapolation required to make predictions under typical nuclear power plant aging conditions of ~50°C (i.e., experimental data is obtained at temperatures as low as 90°C), the confidence level in such predictions should be high.

A more complicated case occurs for an ethylene propylene rubber cable insulation material. Arrhenius plots for ultimate tensile elongation data taken at temperatures ranging from 100 to 170°C are shown in Figure 2.2.6 from Reference 2.2.5 (SAND83-1367C). Clear evidence of non-Arrhenius behavior (i.e., non-straight lines over the experimental temperature range) exists. It is interesting to note that this non-Arrhenius character evidences itself by a slow, subtle change in activation energy occurring over a fairly large temperature range. More limited temperature studies covering a shorter temperature range would have probably concluded that Arrhenius behavior was found.

Understanding and screening for the mechanisms that can lead to non-Arrhenius behavior is an important aspect of determining confidence levels in extrapolations derived from Arrhenius modelling. For the EPR material, the non-Arrhenius behavior was due to the presence of two degradation mechanisms having different activation energies, normal thermal oxidation and copper-catalyzed oxidation. The latter mechanism (which has a higher activation energy) led to greatly enhanced degradation near the inside of the insulation, where copper poisoning from the copper conductors had occurred. The various profiling techniques, developed for understanding dose-rate effects (see Issues 2.3 and 2.8) are extremely helpful in sorting out such mechanisms.

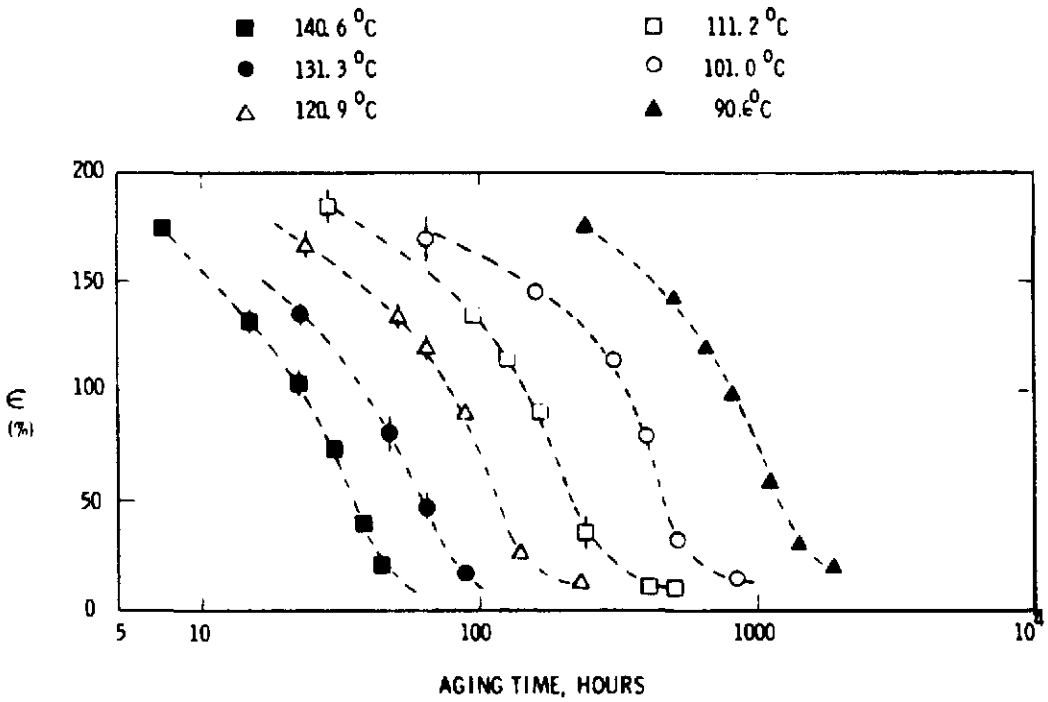

Figure 2.2.3 Ultimate tensile elongation (e) vs aging time for chloroprene material at the six indicated temperatures. (From Reference 2.2.3)

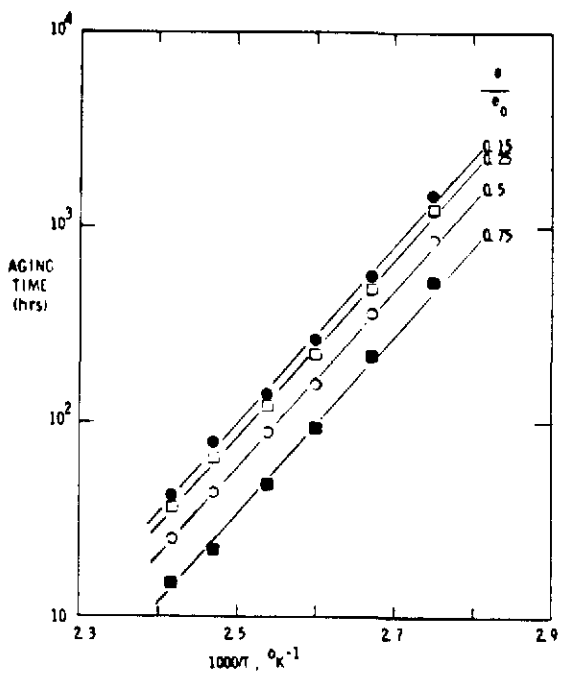

Figure 2.2.4 Arrhenius plot of chloroprene data at the four values of e/e_o indicated. (From Reference 2.2.3)

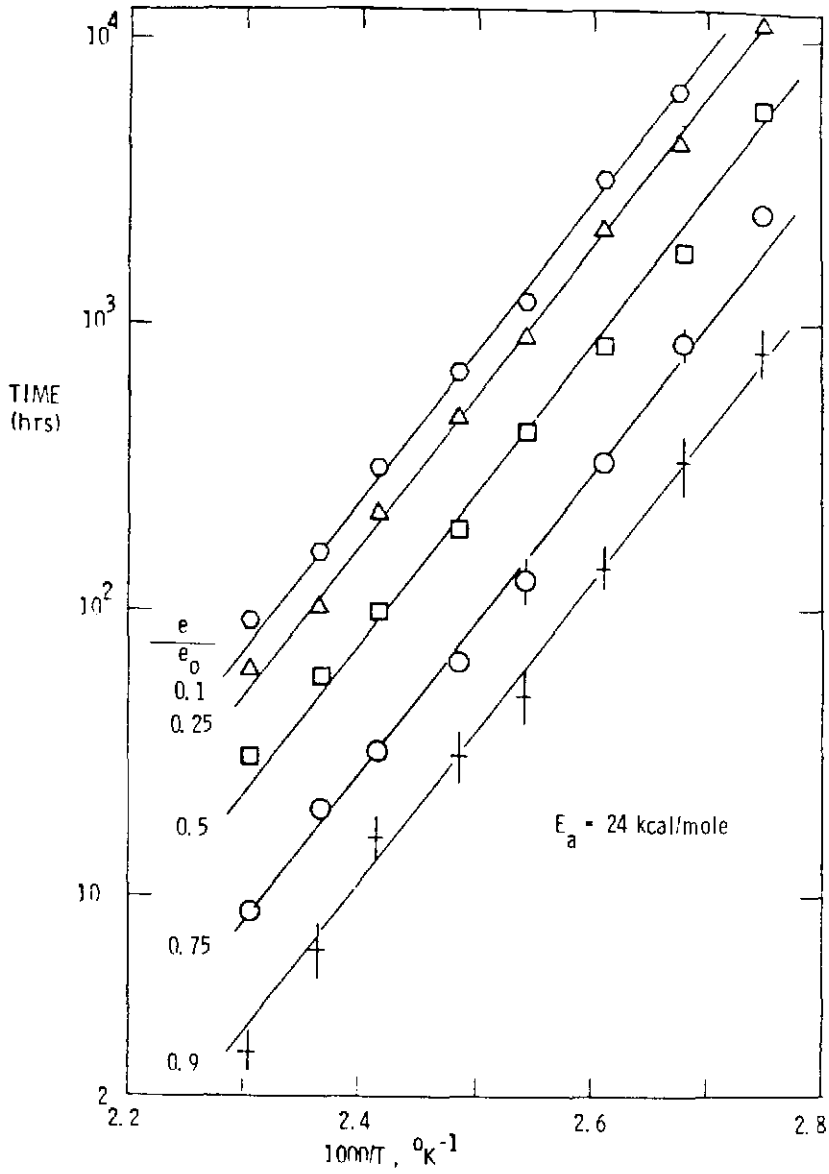

Figure 2.2.5 Arrhenius plot of Hypalon material. (From Reference 2.2.4)

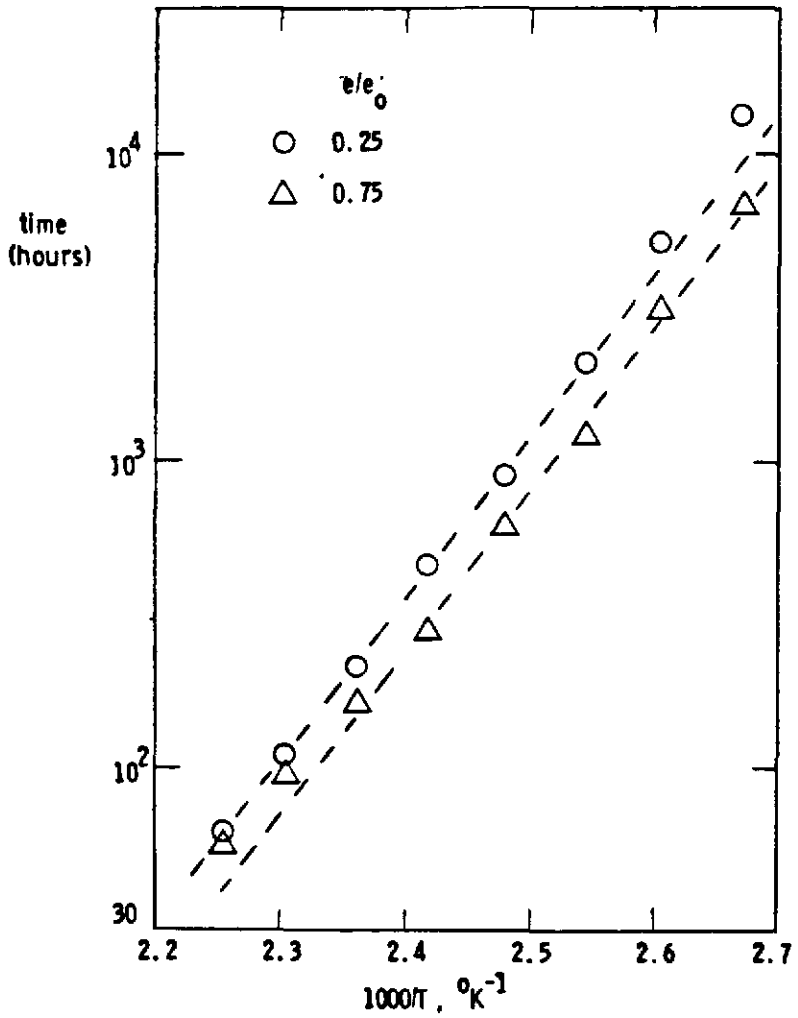

Figure 2.2.6 Arrhenius plot of thermal aging data for EPR. The log of the time required for the ultimate tensile elongation to decrease to 75% and to 25% of original is plotted vs inverse absolute temperature. (From Reference 2.2.5)

Figure 2.2.7 from Reference 2.2.6, NUREG/CR-4358 (SAND85-1557), shows density profiling data for EPR samples which had been aged at 100°C. Clear evidence for greatly enhanced oxidation near the inside of the insulation is apparent, even before measurable changes in mechanical properties are observable (see Figure 2.2.8).

A second mechanism which often leads to non-Arrhenius behavior is the presence of important diffusion-limited oxidation effects. The presence of this mechanism is easily recognized through the use of profiling techniques. Figure 2.2.9 from Reference 2.2.6 shows some density profiling data for a buna-N seal and gasket material aged at 130 and 150°C. Diffusion-limited oxidation effects are evident at both temperatures. For the longer exposure times (i.e., associated with the 130°C aging of Figure 2.2.9, the diffusion effects are less important. For the long times appropriate to real-time power plant aging at say 50°C, uniform oxidation should occur throughout the material. Therefore, the diffusion effects observed in Figure 2.2.9 would have to be eliminated or quantitatively understood before confident predictions from an Arrhenius approach could be made.

Modulus profiling techniques described in Reference 2.2.7 can also be used to monitor diffusion-limited oxidation and other non-Arrhenius effects. Figure 2.2.10, from Reference 2.2.7, shows some representative results for 1.4mm thick nitrile rubber samples aged in air for various times at 120°C. The unaged profile is not quite flat, dropping slightly at one edge. A similar drop near one edge was observed from density profile measurements on the same material. These results are not particularly surprising and indicate that slight thermal gradients may have been present during cure of the compression-molded sheets. At early aging times only a slight hint of diffusion-limited oxidation is evident from the modulus profiling results, yet, at later times, diffusion effects become extremely important. These results are consistent with the often-observed phenomenon that thermal oxidation exhibits an induction period, after which the oxygen consumption rapidly increases.

From the above it should be clear that it is easier to ascertain the presence of diffusion effects using profiling techniques than to discover the often-subtle and slow changes in activation energy from mechanical property data. Thus the use of profiling techniques to screen heat aging data for diffusion effects should both increase the confidence in Arrhenius extrapolations and lead to a decreased number of heat-aging experiments. Some side benefits of such a screening procedure include the sensitivity of the profiling techniques to early degradation and their utility for discovering the presence of competing chemical mechanisms, as illustrated above for the EPR material.

In conclusion, increased confidence in Arrhenius extrapolations can be achieved by

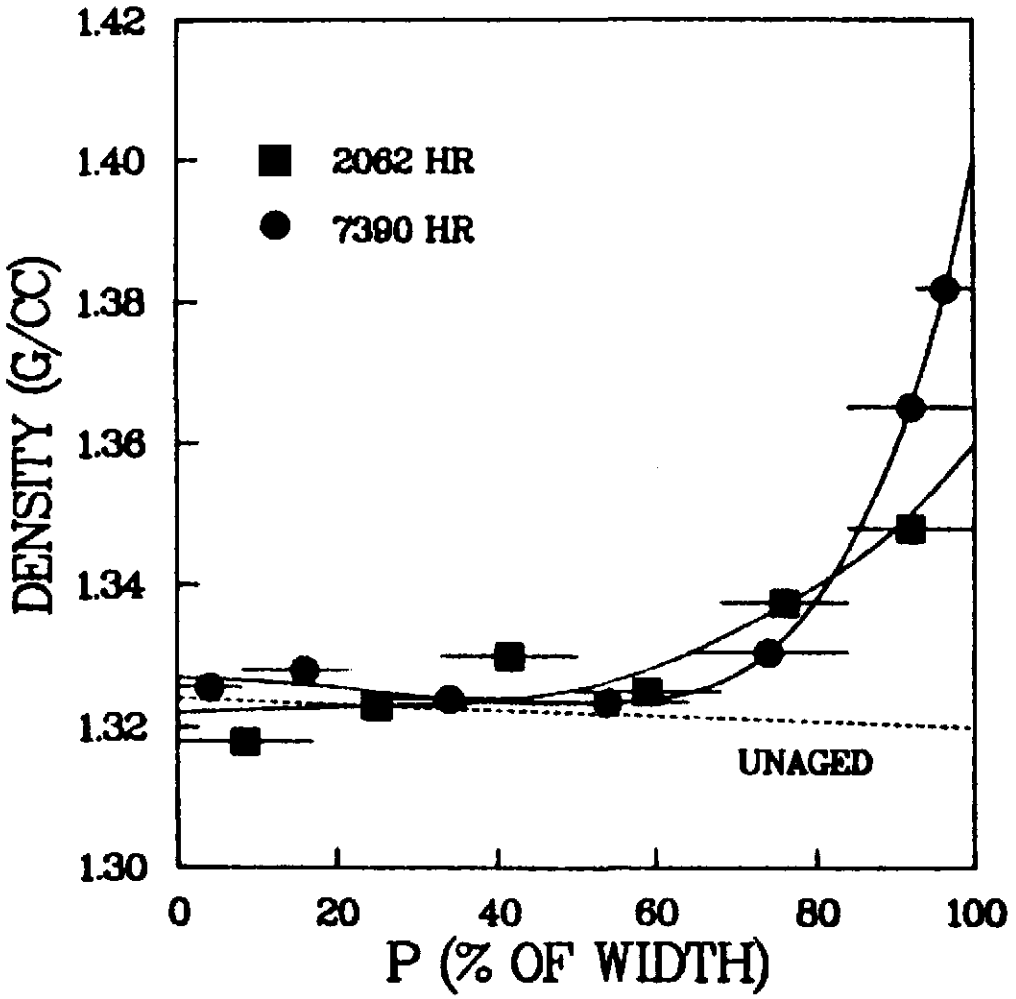

Figure 2.2.7 Density profiles for the EPR cable insulation material which had been heat aged in air at 100°C. (From Reference 2.2.6)

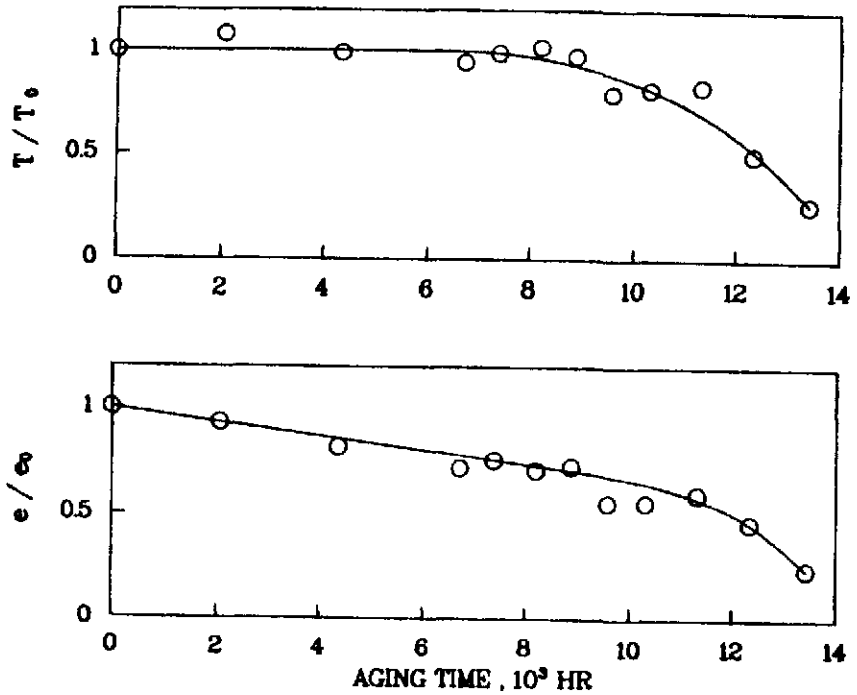

Figure 2.2.8 Mechanical property heat aging results at 100°C for the EPR cable insulation material. The tensile strength after aging divided by the tensile strength before aging (T/T_0) and the tensile elongation after aging divided by the tensile elongation before aging (e/e_0) are plotted against the aging time. (From Reference 2.2.6)

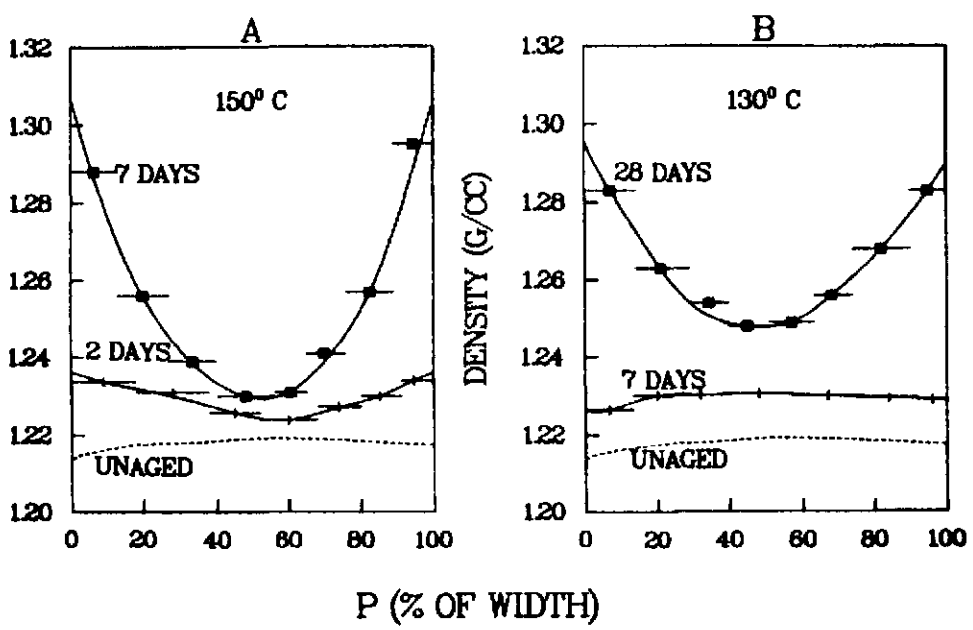

Figure 2.2.9 Density profiles for the nitrile rubber material heat aged in air. A: aging at 150°C. B: aging at 130°C. (From Reference 2.2.6)

1. the use of profiling techniques to screen thermal aging studies for possible diffusion-limited oxidation effects, and

2. the use of long-term aging experiments in order to minimize extrapolation distances.

International and Other U. S. Programs

There are numerous programs throughout the world dealing with accelerated thermal aging and the uses and limitations of the Arrhenius approach. A few studies of particular interest will be mentioned since they concentrate on nuclear power plant components and materials. One is a completed study, prepared by the Franklin Research Center for EPRI. A report (Reference 2.2.8) titled "A Review of Equipment Aging Theory and Technology," was released in 1980. The report gives an excellent account of the Arrhenius approach, highlighting the theoretical aspects underlying the method and the potential problems inherent in its implementation and reviewing the results of its use in various component and material aging studies. An on-going program at the Japanese Atomic Energy Research Institute (Reference 2.2.9) is investigating thermal aging effects on nuclear power plant materials. The effort is concentrating on (1) the use of an oxygen pressure apparatus to study oxygen concentration effects on thermal aging, and (2) the development of empirical kinetic models.

Presentations at two nuclear power plant aging conferences (A Workshop on Nuclear Power Plant Aging, Bethesda, Maryland, August 4-5, 1982 and an International Symposium on Aging in Tests of Safety Equipment for Nuclear Power Plants, Paris, France, May 15-16, 1984) have included discussions of the Arrhenius technique. Presentation summaries are contained in References 2.2.10, NUREG/CP-0036 (SAND82-2264C) and 2.2.11.

Finally, a Naval Research Laboratory study (Reference 2.2.12) has investigated whether hydrolytic degradation of polyimide (Kapton®) can be described by an Arrhenius model. An activation energy for hydrolytic degradation of Kapton of 0.6 eV was determined.

Issue Findings

Some limited thermal aging studies are underway for a number of commercial seal and gasket materials. This data could be used together with earlier data generated on cable materials to help understand the uses and limitations of the Arrhenius approach. The bulk of this effort would be directed towards improving the confidence level in Arrhenius extrapolations. As pointed out above, a key to this objective is the availability of simple, inexpensive profiling techniques. These techniques allow us to screen the thermal aging data for diffusion anomalies and competing reaction problems, as well as reduce the number of thermal aging experiments that would normally be performed. The development of these profiling techniques for dose-rate effects studies

(see Issue 2.3 for a more complete description of results) therefore also offer benefits for improved understanding of the Arrhenius methodology.

Another corrolary result from the dose-rate effects program comes from the modelling studies of diffusion-limited oxidation shapes (see Issue 2.3). The modelling results could allow the conditions necessary to eliminate diffusion effects in thermal aging exposures to be estimated during the planning stage of experiments, thereby eliminating unnecessary experiments before they are run.

A small number of Thermogravimetric Analyzer (TGA) experiments are also being run in order to see if this rapid technique can yield useful information. The temperature dependences of weight loss for very thin slices of material aged in air will be compared to the temperature dependences of the non-diffusion affected long-term mechanical property results to see if activation energy estimates can be made using the TGA.

By performing thermal aging tests on identical test samples over a broad range of temperatures and material degradation, the applicability of Arrhenius methods and range of extrapolation can be determined.

In summary then, the findings within this issue are:

1. Some materials exhibit non-Arrhenius behavior due to changes in activation energy over a large temperature range.

2. Diffusion-limited oxidation effects should be considered before applying the Arrhenius methodology.

3. Long-term material aging experiments might be conducted in order to minimize extrapolation distances.

4. Profiling techniques are extremely useful for

   a. increasing confidence in Arrhenius extrapolations
   b. discovering and/or monitoring non-Arrhenius effects
   c. minimizing the number of experiments which must be run by allowing selection of the appropriate temperature range for conducting experiments.

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Program Effort

2.2.1 K. T. Gillen, E. A. Salazar, and C. W. Frank, Proposed Research on Class 1 Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments, NUREG-0237, SAND76-0715, Sandia Laboratories, Albuquerque, New Mexico, April 1977.

2.2.2 K. T. Gillen and K. E. Mead, Predicting Life Expectancy and Simulating Age of Complex Equipment Using Accelerated Aging Techniques, NUREG/CR-1466, SAND79-1561,Sandia National Laboratories, Albuquerque, New Mexico, January 1980.

2.2.3 L. L. Bonzon, K. T. Gillen and D. W. Dugan, Qualification Testing Evaluation Program, Quarterly Report, July-September, 1978, NUREG/CR-0696, SAND78-2254,Sandia National Laboratories, Albuquerque, New Mexico, March 1979.

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2.2.4 K. T. Gillen, R. L. Clough, and E. A. Salazar, Results From Combined and Single Environment Accelerated Aging Studies, SAND79-2035A, Sandia National Laboratories, Albuquerque, New Mexico. Presented at USNRC Seventh Water Reactor Safety Research Information Meeting, November 5-9, 1979.

2.2.5 K. T. Gillen and R. L. Clough, Inhomogeneous Oxidation Effects Studied Using Density Profiling and Metallographic Polishing Techniques, SAND83-1367C, October, 1983. Presented at the 11th Annual DOE Compatibility Meeting, Livermore, California, October 18-20, 1983.

2.2.6 K. T. Gillen, R. L. Clough and N. J. Dhooge, Applications of Density Profiling to Equipment Qualification Issues, NUREG/CR-4358, SAND85-1557, Sandia National Laboratories, Albuquerque, New Mexico, September 1985.

A. K. T. Gillen, R. L. Clough and N. J. Dhooge, Density Profiling of Polymers, Polymer 27, 225 (1986).

2.2.7 K. T. Gillen, R. L. Clough and C. A. Quintana, Modulus Profiling of Polymers, Polym. Degrad. and Stab. (submitted).

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2.2.8 S. P. Cargagno and R. J. Gibson, A Review of Equipment Aging Theory and Technology, EPRI NP-1558, Electric Power Research Institute, Palo Alto, California (Prepared by Franklin Research Center, Philadelphia, Pennsylvania) September 1980.

2.2.9 T. Seguchi, T. Yagi, and K. Yoshida, Japanese Atomic Energy Research Institute; unpublished data.

2.2.10 B. E. Bader and L. A. Hanchey, Proceedings of the Workshop on Nuclear Power Plant Aging Bethesda, Maryland, August 4-5, 1982, NUREG/CP-0036, SAND82-2264C, Sandia National Laboratories, Albuquerque, New Mexico, November 1982.

2.2.11 Transactions of the International Symposium on Aging in Tests of Safety Equipment for Nuclear Power Plants, Paris, France, May 15-16, 1984. Societe Francaise D'Energie Nucleaire.

2.2.12 F. J. Campbell, Temperature Dependence of Hydrolyses of Polyimide Wire Insulation, NRC Memorandum Report 5158, Naval Research Laboratory, Washington, D.C., August 4, 1983.

2.3 Dose-Rate Effects

Issue Description

To be able to adequately meet the aging requirement for equipment qualification in a radiation environment, the time scales of the aging simulation usually need to be compressed. This necessitates carrying out the simulated radiation exposure at much higher dose rates than those seen by the equipment during ambient aging. The possibility of the existence of dose-rate effects caused by the accelerated dose rates could be addressed. For materials and components which have dose-rate effects, acceptable methodologies which assure conservative aging simulations can be developed. The continuing need to consider and account for dose-rate (and related synergistic) effects is recognized in Regulatory Guide 1.89, Section 5.a, "If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program. Synergistic effects known at this time are dose rate effects and effects resulting from the different sequence of applying radiation and (elevated) temperature."

This is the thrust of this effort to address the issue:

> "Are dose-rate effects a factor in aging simulation, and how must they be simulated?"

Organic Materials - Evidence of Dose-Rate Effects

When this effort was initiated in 1976, the guidance for industry attempting to qualify components for nuclear power plant safety applications came mainly from IEEE Standards. IEEE Std 323-1974 was a general document which described basic procedures for qualifying Class 1E equipment. Certain specific components were discussed in daughter documents. For example, qualification procedures for electric cables, field splices and connections were given in IEEE Std 383-1974. An important aspect of qualification is aging the components to a condition equivalent to the end-of-life condition. It was recognized in these and similar documents that little information existed on how to properly perform accelerated aging simulations, especially in humidity and radiation environments. Although concerns about radiation dose-rate effects during accelerated radiation-aging simulations were not specifically mentioned in these documents, IEEE-323 did state that "consideration shall be given to oxidation gas-diffusion effects" and referred to formulas given in IEEE Std 278-1967 as a means of estimating the test dose equivalent to service. For the radiation aging part of the aging sequence, IEEE-383 suggested that specimens be subjected in air to gamma radiation at a dose rate not greater than 1 Mrad/h.

Given the concerns about aging procedures and the possibility that radiation dose-rate effects and synergistic effects in combined environment aging situations could adversely affect component qualification, we were asked to review aging methodologies and suggest

needed research areas. This request resulted in Reference 2.3.1, NUREG-0237 (SAND76-0715). This report:

1. discussed Class 1E equipment used in nuclear power plants, typical environments encountered by this equipment and the aging techniques used to qualify this equipment.

2. reviewed radiation chemistry of polymers and showed why dose-rate and synergistic effects in air environments should be a potential concern.

3. extensively reviewed accelerated aging techniques.

4. proposed an experimental program which was based on the concerns about possible dose-rate effects and the inadequacies of the existing aging techniques for Class 1E equipment.

Among the main purposes of the proposed work was to obtain relevant data in two areas of particular concern - the effect of radiation dose rate on polymer degradation and the importance of synergism for combined thermal and radiation environments. Reference 2.3.1 also proposed a new model for combined environment accelerated aging and suggested what experimental data was necessary to test the model. It was proposed that real commercial materials be studied instead of the pure polymers on which most of the literature studies had been based. Pure polymers bear little resemblance to commercial materials, since the latter contain various fillers, antioxidants, antirads, pigments, plasticizers and processing impurities, all of which could effect the radiation aging.

Reference 2.3.2, SAND77-0511A, gave:

1. A detailed description of a proposed model for accelerated aging in combined radiation and thermal stress environments. In order for the model to be applicable, single stress radiation exposures at low temperatures could not exhibit dose rate effects. Also, single stress elevated-temperature data must be Arrhenius. Hence it was desirable to evaluate typical nuclear polymer materials for their susceptibility to dose rate effects and non-Arrhenius behavior.

2. Successfully tested the model on literature data for the thermoradiation sterilization of a biological material for which significant synergistic effects were found for combined thermal and radiation environments.

In order to carry out the goals of the experimental program, an extremely versatile dedicated radiation-aging facility was constructed, as documented in Reference 2.3.3, NUREG/CR-2877 (SAND81-2613). Although the facility is continuing to be modified for improved performance, some earlier sketches of the facility shown in Figures 2.3.1-2.3.4 serve to highlight its salient features. Cobalt-60 is permanently positioned in

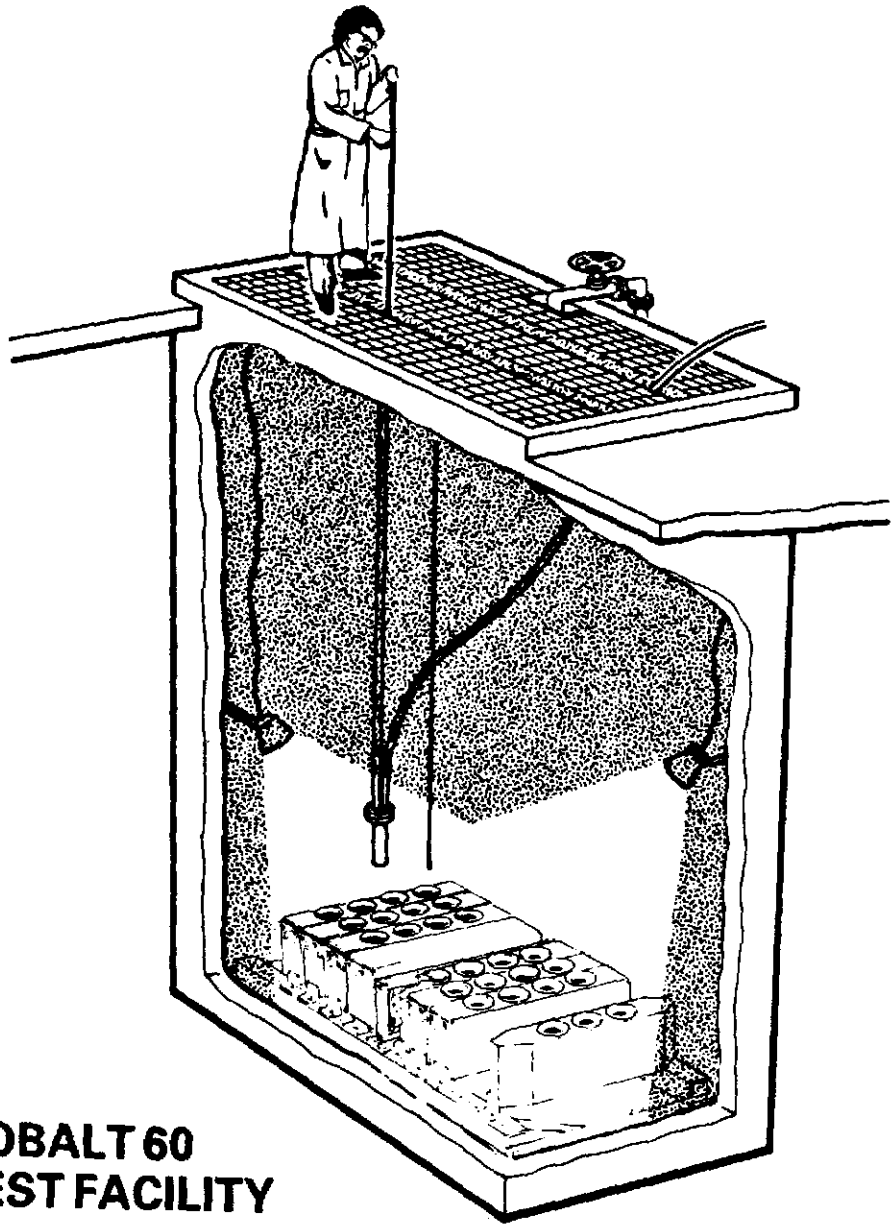
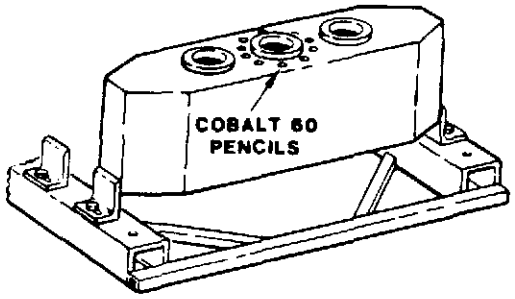
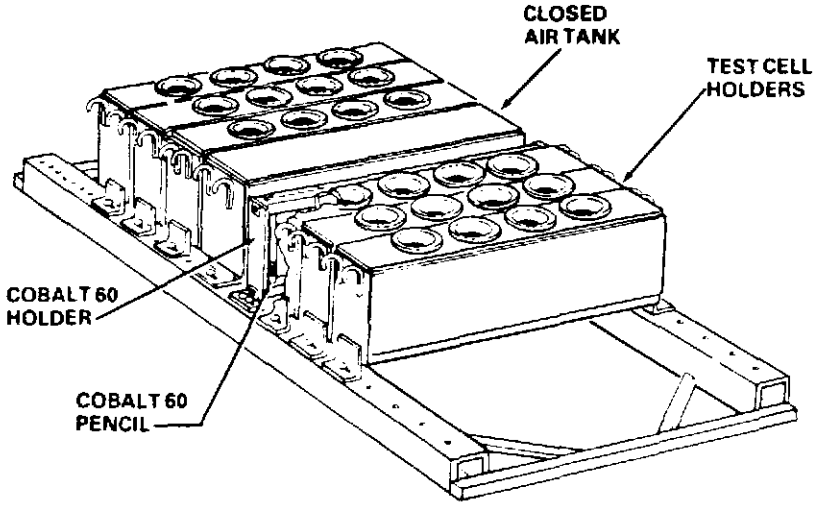


Figure 2.3.1 General view of radiation aging facility. (From Reference 2.3.3)



CIRCULAR ARRAY RADIATION SOURCE

Figure 2.3.2 Diagram of the high dose-rate portion of the aging facility. (From Reference 2.3.3)



LINEAR ARRAY RADIATION SOURCE

Figure 2.3.3 Diagram of the low dose-rate portion of the aging facility. (From Reference 2.3.3)

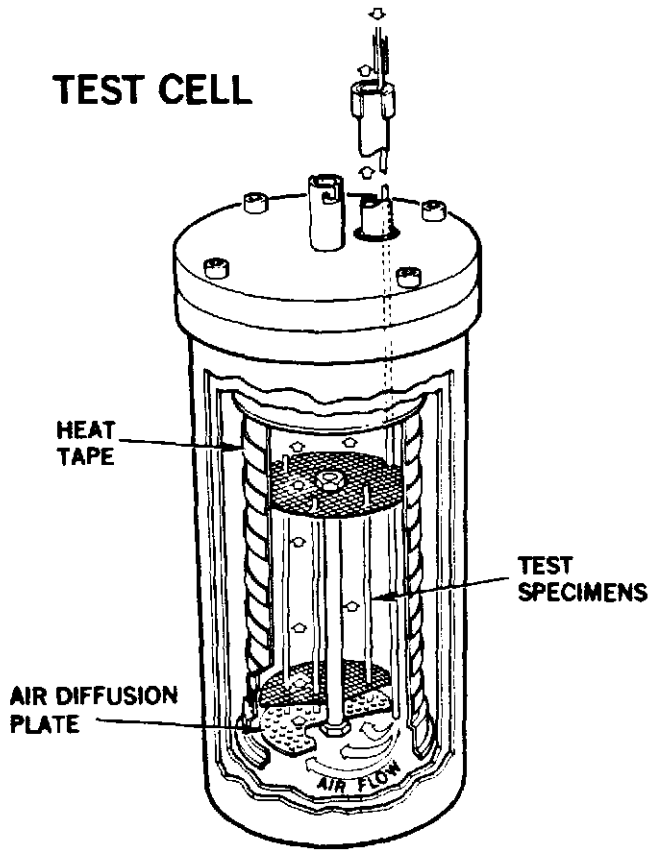


Figure 2.3.4 Underwater radiation test chamber. (From Reference 2.3.3)

various arrays at the bottom of a water-filled stainless-steel lined concrete tank. Radiation aging is carried out in temperature-controllable, water-tight test cells by lowering the cells to the bottom of the tank. The dose rate is controlled by choosing the position of the can and the aging atmosphere is determined by the choice of circulating gas. This facility therefore allows us to carry out long-term aging in selectable combinations of radiation dose rate, temperature and surrounding atmosphere (e.g., air or nitrogen).

Early experimental results on commercial cable materials, taken before the longer-term, low dose rate data was available, were described in Reference 2.3.4, NUREG/CR-0813 (SAND79-0761). Good evidence for dose-rate effects were observed for cross-linked polyolefin, Tefzel and ethylene propylene rubber insulation materials (see for example Figure 2.3.5). No noticeable dose-rate effects were noted for a chloroprene rubber (neoprene) material, a silicone material and two chlorosulfonated polyethylene (hypalon) materials (see for example Figures 2.3.6 and 2.3.7). At most of the radiation dose rates used for this study, each material was exposed under both low (0%) relative humidity and high (~70%) relative humidity conditions. No noticeable effect of humidity on mechanical properties was observed.

Reference 2.3.5, NUREG/CR-0696 (SAND78-2254) showed that:

1. Heat aging data for the neoprene rubber material was Arrhenius (Figure 2.3.8) and

2. When combined environment data for neoprene were analyzed according to the assumptions inherent in the model, the resulting superposition in two-dimensional radiation dose rate-inverse temperature space offered strong evidence in support of the proposed model. Some representative results showing the excellent superposition are shown in Figure 2.3.9 from Reference 2.3.5 Attempted application of the model was feasible since the early radiation dose-rate data at low temperatures indicated that dose-rate effects were small (a necessary requirement).

After sufficient data at lower dose rates had been collected for many of the materials mentioned earlier as well as a number of other materials, it became clear:

1. That dose-rate effects are a very common occurrence for radiation aging of polymeric materials.

2. That dose-rate effects can range from very large to insignificant, depending upon such factors as polymer type, aging conditions, sample geometry and degradation parameter being monitored.

Some examples from References 2.3.6, NUREG/CR-2157 (SAND80-1796) and 2.3.7, NUREG/CR-2156 (SAND80-2149) are shown in Figures 2.3.10 to 2.3.12. The reduced elongation results for PVC (Figure 2.3.10) are an

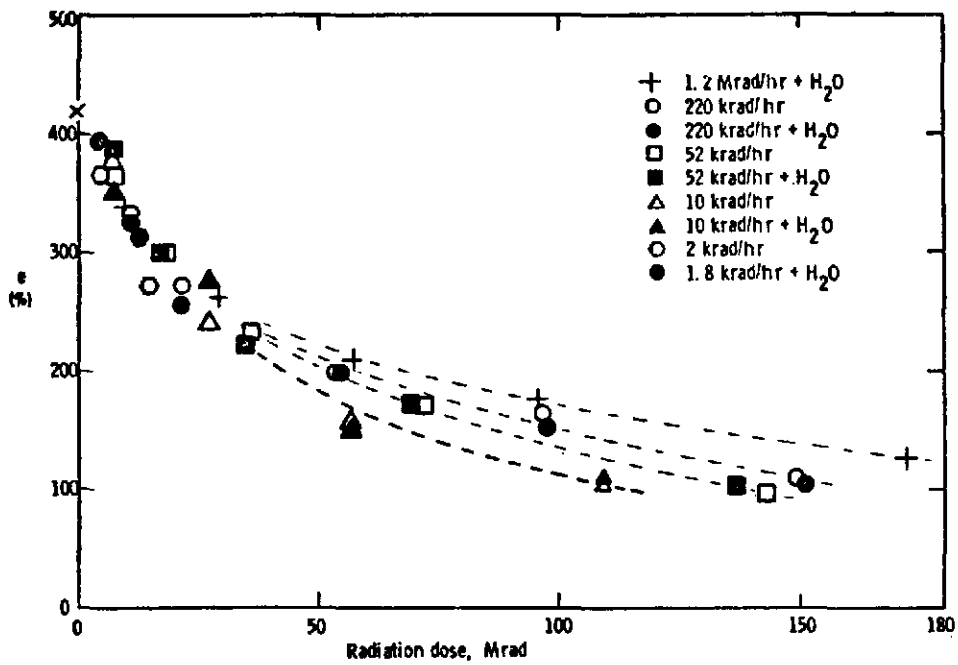


Figure 2.3.5 Ultimate tensile elongation (e) vs. total radiation dose for ethylene propylene rubber insulation. (From Reference 2.3.4)

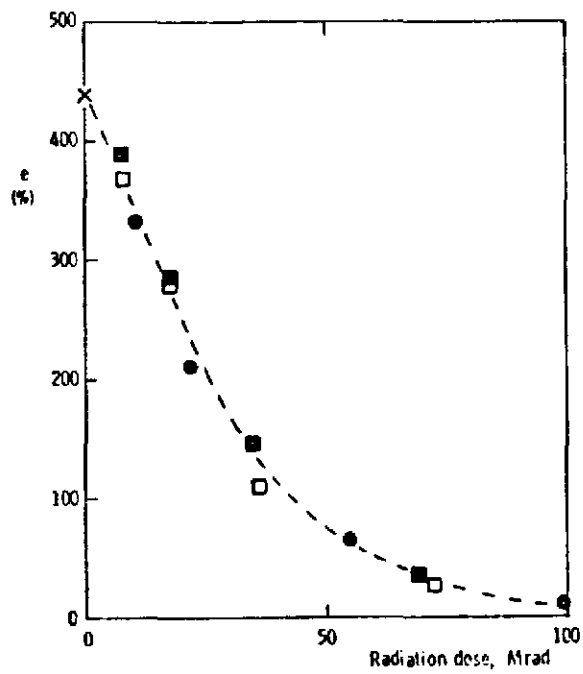


Figure 2.3.6 Elongation vs. total radiation dose for silicone insulation. See Figure 2.3.5 for meaning of symbols. (From Reference 2.3.4)

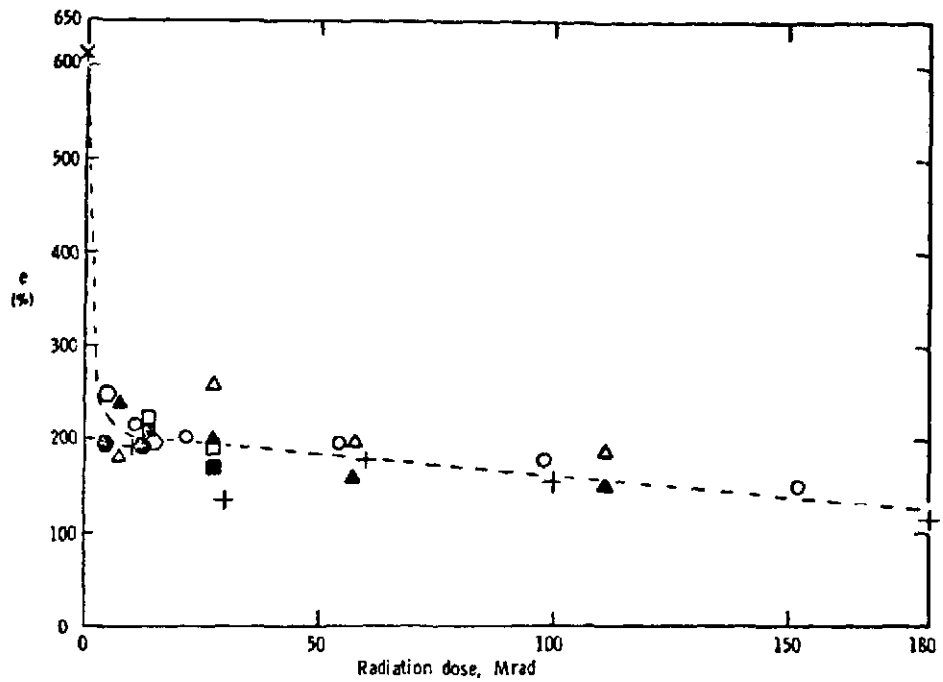


Figure 2.3.7 Ultimate tensile elongation (e) vs. total radiation dose for chlorosulfonated polyethylene insulation material. See Figure 2.3.5 for meaning of symbols. (From Reference 2.3.4)

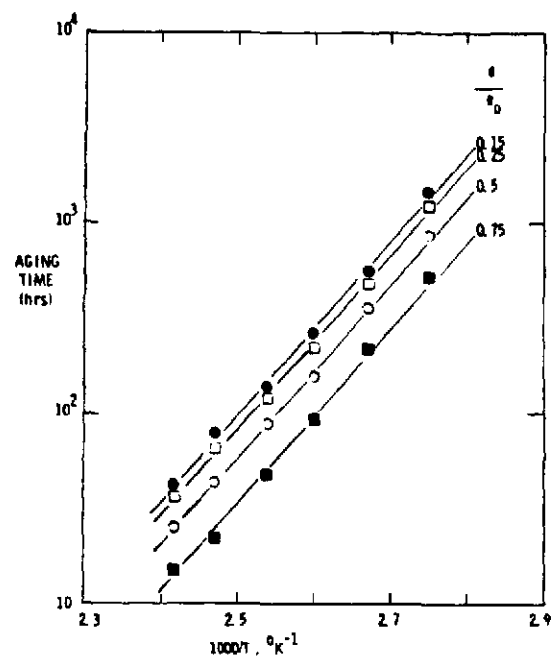


Figure 2.3.8 Arrhenius plot of chloroprene data at the four values of e/e_0 indicated. (From Reference 2.3.5)

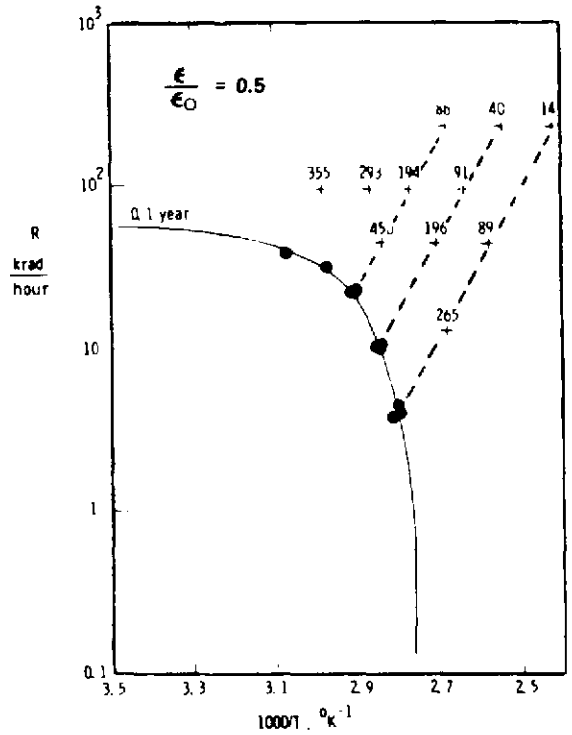


Figure 2.3.9 Analysis of chloroprene data using proposed method. The crosses represent the experimental radiation and temperature coordinates. The numbers above them represent the time (hrs) required for e/e_0 to reach 0.5. The filled circles are the prediction for the 0.1 year contour using the model. (From Reference 2.3.5)

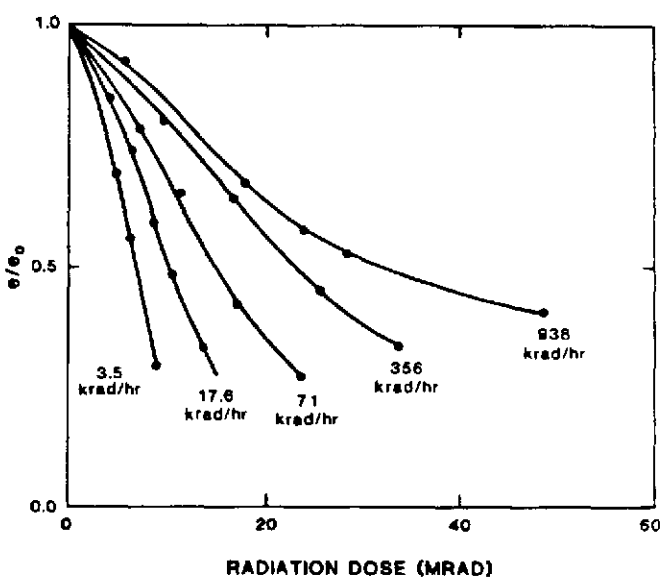


Figure 2.3.10 Aging of PVC at 60°C at a series of radiation dose rates. Sample tensile elongation divided by initial (unaged) elongation is plotted versus total absorbed radiation dosage. (From Reference 2.3.7)

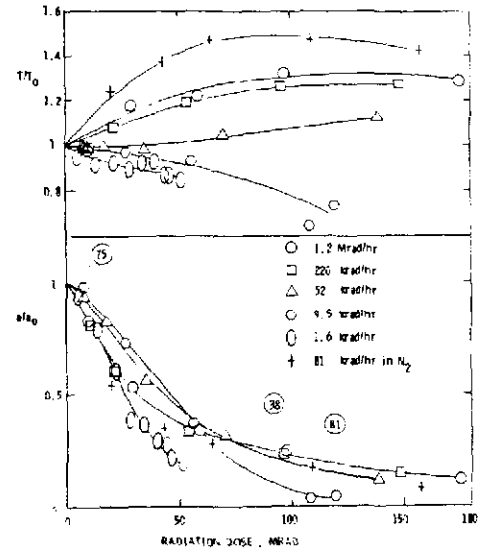


Figure 2.3.11 Aging of cross-linked polyolefin insulation. The tensile strength after aging divided by the tensile strength before aging (T/T_0) and the tensile elongation after aging divided by the tensile elongation before aging (e/e_0) plotted against the total integrated radiation dose at the various indicated dose rates. (From Reference 2.3.6)

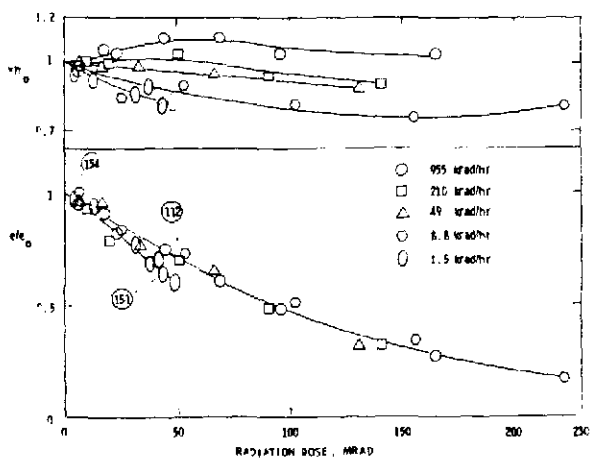


Figure 2.3.12 Aging of chlorosulfonated polyethylene (Hypalon) jacket. Explanation of figure is identical to Figure 2.3.11 (From Reference 2.3.6)

example of very large dose-rate effects. Figure 2.3.11 gives reduced tensile strength and reduced tensile elongation for a cross-linked polyolefin material. The tensile strength results indicate the existence of important dose-rate effects; the elongation results are more subtle and quite complicated. For the Hypalon® material shown in Figure 2.3.12, dose-rate effects for elongation are barely noticeable although they are more apparent for the tensile strength data.

Since most materials were found to have dose-rate effects and the dose-rate effects usually tended to indicate more mechanical degradation for a given total dose as the dose rate was lowered, understanding the mechanisms behind these dose-rate effects was clearly an important aspect of developing aging methodologies. Comparisons of aging results in air versus nitrogen environments immediately implicated oxidation processes in the dose-rate effects (see for example Figure 2.3.11). In fact no evidence exists for dose-rate effects in a nitrogen environment. Reference 2.3.6 described a number of techniques which indicated that oxidative scission becomes more important relative to cross-linking as the dose-rate is lowered. Results from one of these techniques, infrared spectroscopy, is shown in Figure 2.3.13 for the ethylene propylene rubber sample of Figure 2.3.5. The much larger carbonyl peak (at about 1700 cm^{-1}) produced under low dose rate conditions is consistent with the expected increase in the extent of the oxidative reactions.

The almost universal observation of dose-rate effects implies that the combined environment model described earlier cannot be applied to these

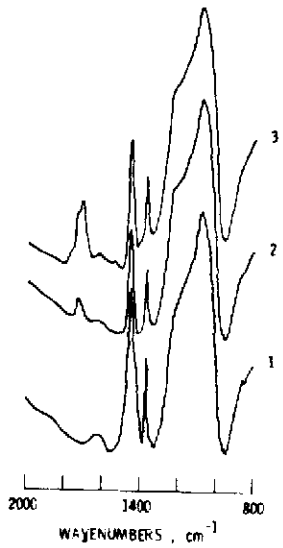


Figure 2.3.13 Infrared spectra of ethylene propylene rubber insulation. Curve 1 - unaged material. Curve 2 - aged in air at 1.2 Mrad/hr. Curve 3 - aged in air at 9.3 krad/hr. (From Reference 2.3.6)

materials until either the dose-rate effects are eliminated or understood. This is due to the fact that one of the necessary assumptions of the model is the lack of dose-rate effects for low-temperature radiation exposures. The model worked nicely for the chloroprene rubber material (Figure 2.3.9), perhaps due to the relatively small dose-rate effects found for its elongation data.

At this point our attention turned towards understanding and modelling the observed dose-rate effects so they could either be accounted for by aging methodologies or be eliminated by the manufacturer through material improvements and/or modifications. The most obvious potential cause of dose-rate effects is oxygen-diffusion-limited degradation, a mechanism which has been observed for many years in various air-aging environments. It occurs when the oxidation processes in a material use up dissolved oxygen faster than it can be replenished from the atmosphere surrounding the material by diffusion processes. This leads to a more heavily oxidized material near the sample surfaces and reduced or depleted oxidation in the sample interior. As the dose rate or other environmental stress is reduced, oxidation of the sample increases due to the longer times available for the diffusion processes. Reference 2.3.8, NUREG/CR-1466 (SAND79-1561), describes how this effect can lead to problems with the Arrhenius methodology for heat aging situations. In radiation environments, diffusion effects can clearly be eliminated by going to low enough dose rates. Using thinner samples will also reduce these effects, but this can create some problems with commercial samples or components.

The other possible explanation for dose-rate effects involves chemical effects, although prior to our studies, no evidence for chemical dose-rate effects existed for any polymer. Chemical dose-rate effects occur whenever some chemical step in the kinetics underlying degradation occurs on a time scale comparable to the sample exposure time. The unfortunate aspect of chemical dose-rate effects is that one cannot guarantee their disappearance by going to low enough dose rates. Aging techniques should either appropriately model the chemical mechanism or polymer formulations should be modified to reduce or eliminate its importance.

Our first discovery of the existence of an important chemical dose-rate effect came from studies on a PVC cable jacketing material and a low density polyethylene cable insulation material identical to materials found to be mechanically degraded in the Savannah River K nuclear plant. The reactor materials had seen an estimated 12 years of operating conditions equal to approximately 25 rad/h and 43°C, which gives a moderate total dose of approximately 2 Mrad. For these conditions the amount of degradation was quite severe, especially for the polyethylene (Reference 2.3.9). Very large dose-rate effects were observed for both materials (Figures 2.3.10 and 2.3.14). Striking synergistic effects of radiation plus elevated temperature environments, which are mechanistically related to dose-rate effects, were also observed as shown for example in Figure 2.3.15. An example of the differences in aging

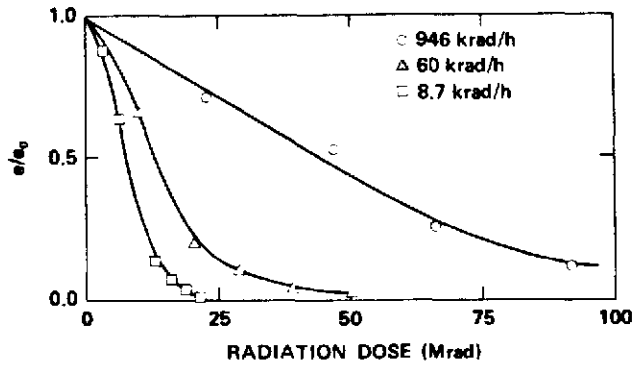


Figure 2.3.14 Tensile elongation results for PE insulation as a function of total absorbed dose at 43°C using three different dose rates. (From Reference 2.3.9)

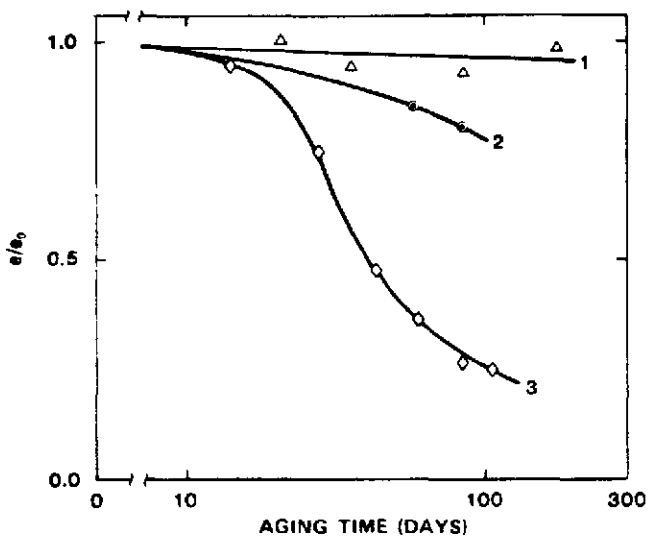


Figure 2.3.15 Tensile elongation results for PVC jacketing material as a function of aging time in three different laboratory environments: (1) elevated temperature, 80°C; (2) radiation at room temperature, 4.4 krad/h at 25°C; and (3) radiation at elevated temperature, 4.4 krad/h at 80°C. (From Reference 2.3.9)

observed for experiments run in the presence and absence of oxygen is shown in Figure 2.3.16. Results such as these again implicated oxidation chemistry as the cause of the dose-rate and synergistic effects.

Unravelling the qualitative mechanism responsible for the dose-rate effects observed in the PVC and polyethylene materials was accomplished with a series of experiments, detailed in References 2.3.3, 2.3.7, 2.3.9 and 2.3.10 (SAND80-1699J). The capabilities of our dedicated radiation-aging facility allowed us to do a number of sequential aging experiments including: 1) radiation in nitrogen (or air) followed by elevated temperature in nitrogen (or air); and 2) elevated temperature in nitrogen (or air) followed by radiation in nitrogen (or air). These results, coupled with the simultaneous aging results and infrared spectra, strongly suggested that the observed dose-rate and synergistic effects were caused by a mechanism involving the rate-determining thermally-induced breakdown of peroxides initially formed by the radiation (Figure 2.3.17). To further confirm this mechanism, a new chemical technique was developed. This technique uses phosphine gas treatment of intact polymer samples to test for the importance of peroxides in the chemical pathway that leads to changes in macroscopic properties. The results, both for mechanical properties and from infrared spectra (Figure 2.3.18), further confirmed the hydroperoxide breakdown mechanism as a major cause of the observed dose-rate and synergistic effects. Infrared spectra were also used to show that the oxidative products generated in the laboratory (accelerated) aged samples were similar to those found in the samples removed from the reactor, as seen in Figure 2.3.19. Thus this series of papers (References 2.3.3, 2.3.7, 2.3.9 and 2.3.10) led to a number of important results:

1. Very large dose-rate effects can exist and involve oxidation mechanisms.

2. Important synergisms of low-temperature radiation and elevated thermal environments and important ordering effects in sequential aging experiments can occur and these are mechanistically related to radiation dose-rate effects.

3. The first evidence for a chemical dose-rate effect (hydroperoxide breakdown mechanism) was presented.

4. A qualitative description for this mechanism was presented.

5. A new chemical technique (phosphine gas treatment) for confirming the importance of this mechanism was developed.

Reference 2.3.11, NUREG/CR-3151 (SAND82-1414),

1. showed that other commercial PVC materials had behaviors in radiation environments similar to those found for the material removed from Savannah River.

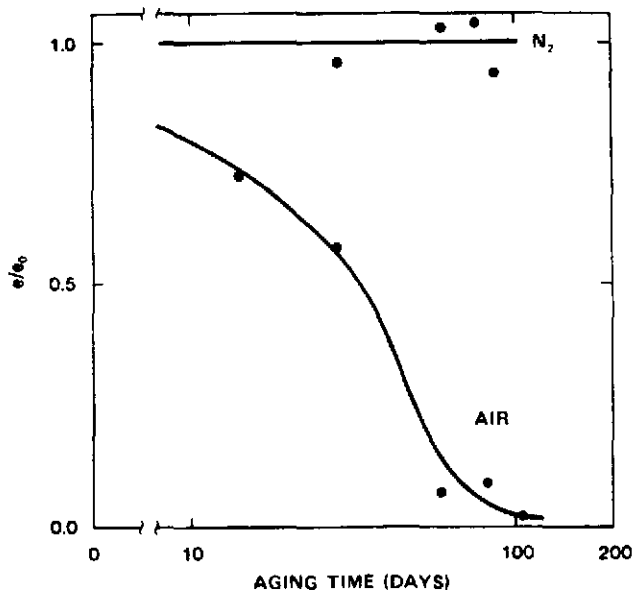


Figure 2.3.16 Tensile elongation results for PE insulation as a function of aging time at 5 krad/h, 80°C in atmospheres of air and of N_2. (From Reference 2.3.9)

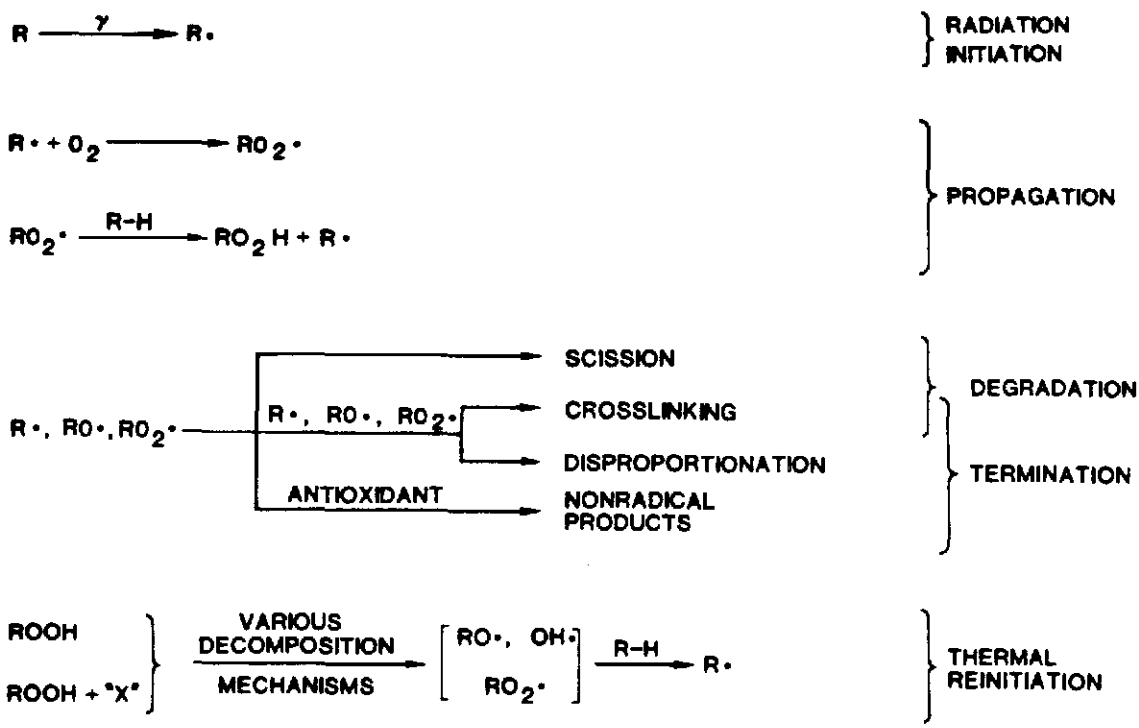


Figure 2.3.17 Chemical mechanism for radiation-induced oxidation. (From Reference 2.3.3)

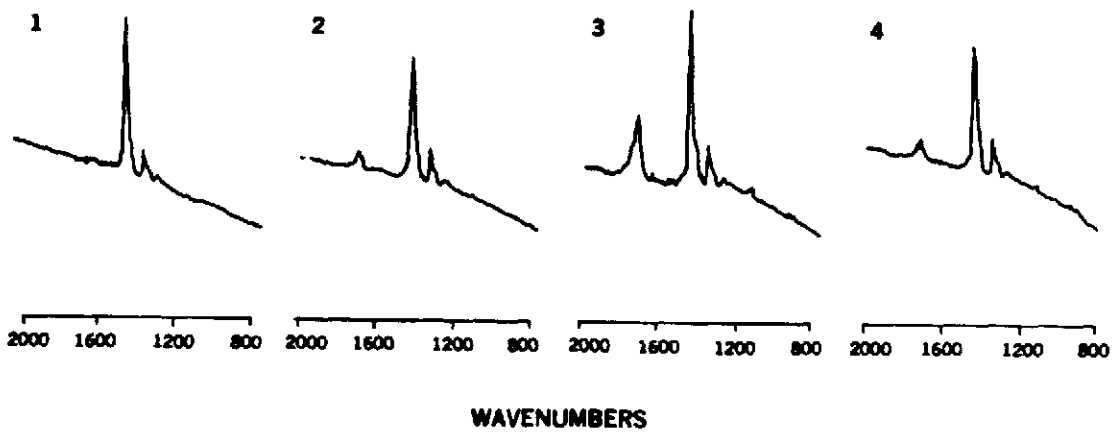


Figure 2.3.18 Fourier transform infrared spectra (FTIR) on sequentially-aged PE samples with and without PH_3 treatment: 1) unaged material; 2) radiation aging only; 3) radiation aging followed by elevated temperature aging; 4) radiation aging followed by PH_3 treatment followed by elevated temperature aging. In each experiment, radiation was 4.4 krad/hr at 25°C (total dose = 10 Mrad). Thermal treatment was 80°C for 25 days. PH_3 treatment was 1.4×10^6 Pa (200 psi) at 25°C for 1 day. (From Reference 2.3.3)

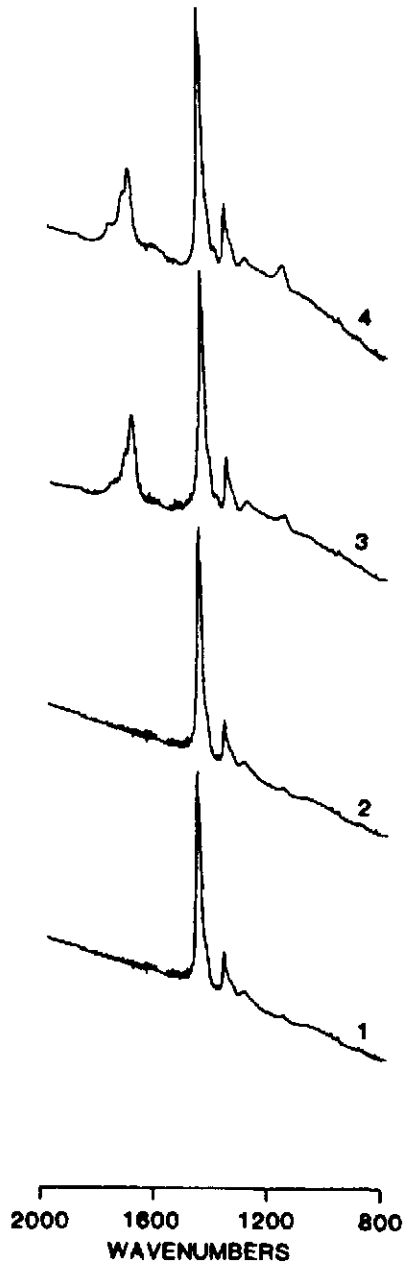


Figure 2.3.19 Fourier transform infrared spectra of PE insulation. Curve 1: unaged laboratory sample. Curve 2: sample removed from containment position which exhibited no apparent deterioration. Curve 3: sample removed from containment position which exhibited substantial embrittlement. Curve 4: laboratory aged sample (42 krad/hr at 90°C to 17.9 Mrad total dose). Note the presence of the carbonyl bands near 1700 cm^{-1} for the aged samples. (From Reference 2.3.3)

2. outlined some of the dangers of an "overstress" concept for aging, whereby the total stress (such as total dose) applied to a material in an accelerated test is chosen to be larger than that expected for the material application environment during the expected lifetime.

The margin approach has been used either 1) to "conservatively qualify" the material by attempting to stress it more severely in the test than what is actually expected during the operating lifetime, or 2) to attempt to compensate for any possible time-dependent effects (such as dose-rate effects). Typical aging data for PVC shown in Figure 2.3.20 exhibit a drop in tensile strength to a minimum, followed by a large increase. Such behavior can clearly lead to problems when attempting to rationalize an overstress approach.

Even more compelling evidence against the use of the overstress concept comes from dose-rate data on a commercial Viton material used for seal and gasket applications. As shown in Figure 2.3.21 from Reference 2.3.12 SAND83-2085C, differences in degradation mechanisms at different dose rates are so profound that the material is actually observed to degrade in opposite ways at low and high dose rates. At the highest dose rate (550 krad/h), the elongation drops steadily, while the tensile strength remains relatively high. The macroscopic observation at high dose rates is that the material becomes hard and brittle as it degrades. At the lowest dose rate (13 krad/h), the tensile strength drops sharply while the elongation remains virtually unaffected. Macroscopically at low dose rates the material becomes soft and stretchable as it degrades. Clearly, for this material, any attempt to model macroscopic property changes at low dose rate, by performing radiation experiments at high dose rates, is futile. Use of "margin" in the high dose rate experiment can only make matters worse. Failure modes and failure criteria in testing a material becoming progressively embrittled as a function of absorbed dose should be very different from failure modes and failure criteria for a material becoming progressively softer as a function of absorbed dose.

Organic Materials - Modelling

The above work had clearly answered the first part of this issue: Dose-rate effects are a definite factor in aging simulations. The work had discovered an important chemical dose-rate mechanism (hydroperoxide breakdown) in addition to the well-known physical mechanism involving oxygen-diffusion-limited degradation. The work had also shown that these dose-rate effects cannot be simply simulated by using the concept of margin. Thus in order to address the second part of the issue, "how must dose-rate effects be simulated", the program turned towards the development of more quantitative techniques and models. It was felt that if dose-rate effects and synergistic effects could be quantitatively understood and described, a rigorous aging methodology would be possible.

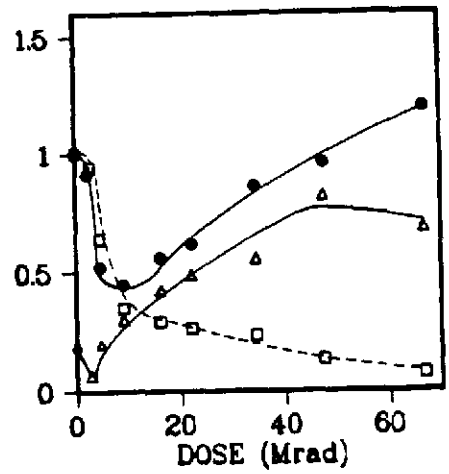


Figure 2.3.20 Radiation aging of PVC-1 at 38 krad/hr, 95°C. (From Reference 2.3.11)

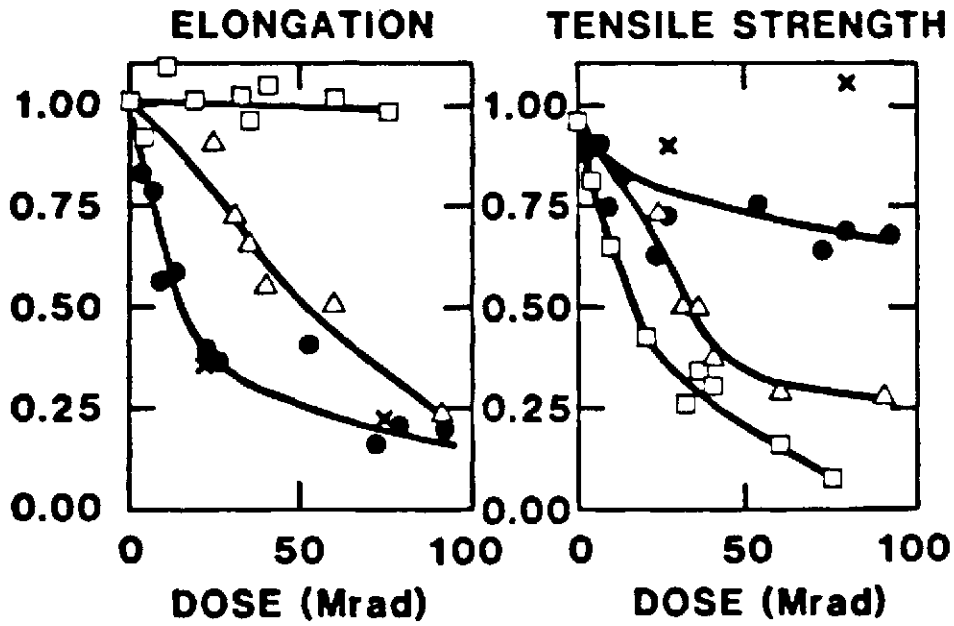


Figure 2.3.21 Change in ultimate tensile properties for a Viton® O-ring material irradiated at 70°C using three different dose rates. • = 5.5 x 10^5 rad/h (in air), Δ = 9.2 x 10^4 rad/h (in air), □ = 1.3 x 10^4 rad/h (in air), X = 5.5 x 10^5 rad/h (in inert atmosphere). Left: reduced elongation (e/e_o); Right: reduced tensile strength (T/T_o). (From Reference 2.3.12)

Since oxygen-diffusion-limited degradation is typically unimportant for flexible polymeric materials under the long-term, low-level environmental stresses appropriate to real-time aging (homogeneous oxidation occurs throughout the sample), the first requirement for quantitatively understanding dose-rate effects was the necessity of having general techniques to determine when and how important the diffusion mechanism is under accelerated aging conditions. Such techniques would then allow accelerated aging conditions to be chosen which assured homogeneous oxidation throughout the material, in agreement with the result anticipated for real-time aging.

Diffusion-limited oxidation leads to a heterogeneously oxidized sample so profiling techniques sensitive enough to detect this heterogeneity were needed. Since the thickness of polymers used in commercial applications are typically on the order of one mm or less, the profiling techniques needed to be capable of resolution down to 0.1 mm or better for commercial materials. Since no existing general techniques which met these criteria could be found, we proceeded to develop a number of versatile and inexpensive profiling techniques.

Reference 2.3.13, SAND82-1413J, describes the first profiling technique developed, density profiling. This technique (details given in Issue 2.8) owes its sensitivity to the large density increases that are found to occur in polymers undergoing oxidation. Figure 2.3.22 shows whole sample density data for an EPR insulation material aged under various dose-rate conditions both in air and in nitrogen. It is immediately clear that this density data is more sensitive to dose-rate effects than the mechanical property data for the same material, which was shown in Figure 2.3.5. The samples irradiated in nitrogen show a very slight increase in density with dose; those irradiated in air show much larger increases, with the amount of increase growing as the dose rate is lowered. Cutting very thin slices across the cross sections of the samples allowed density profiles to be obtained. Figures 2.3.23 and 2.3.24 show representative density profile results for this EPR material. At high dose rates, oxygen diffusion-limited degradation is evidenced by a density profile which shows increased density near the air exposed surfaces of the insulation but little change in density in the interior of the sample. As the dose rate is lowered, this mechanism disappears as anticipated, but a second mechanism responsible for greatly enhanced oxidation at the inside of the insulation (adjacent to the copper conductor) appears. This second mechanism involves copper-catalyzed oxidation which is often found to be significant in elevated temperature aging studies. Density profiling allowed us to determine that this mechanism is also significant under room-temperature, radiation-aging conditions. Thus strong evidence exists for a second (in addition to hydroperoxide breakdown) chemical dose-rate mechanism. This mechanism is the reason for the continuation of dose-rate effects for this EPR material after diffusion effects are no longer important. Reference 2.3.13 therefore led to a number of important results and conclusions:

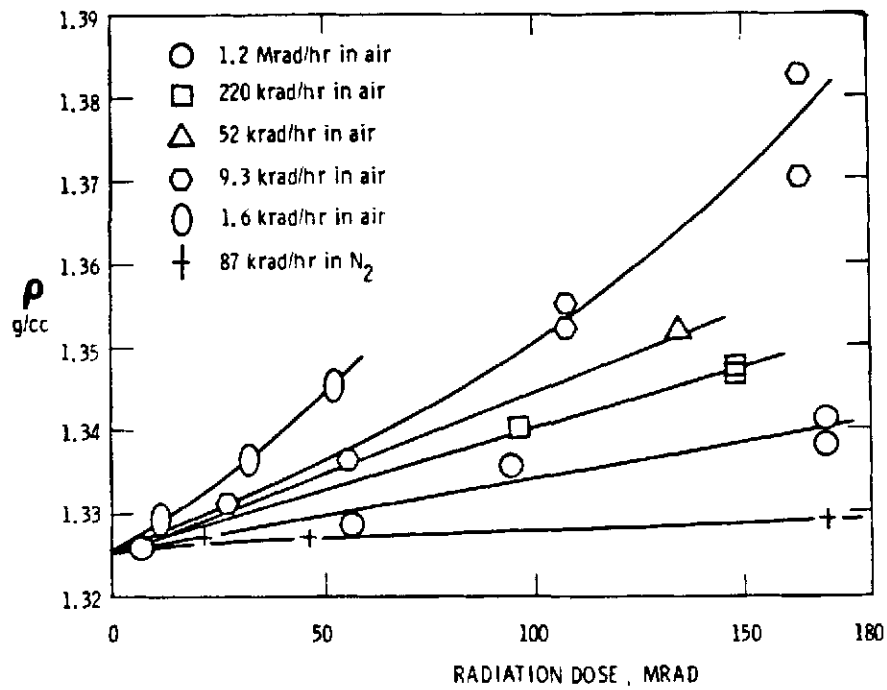


Figure 2.3.22 Overall density of EPR plotted against the total integrated radiation dose at the various indicated dose rates. (From Reference 2.3.13)

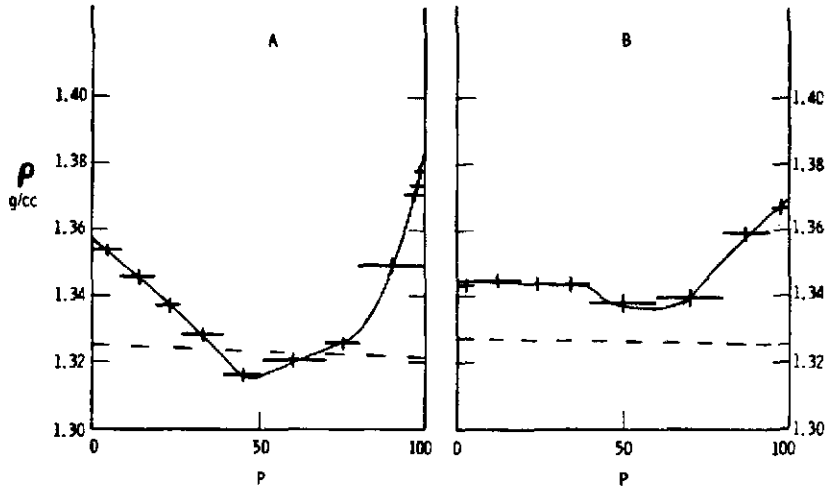


Figure 2.3.23 A: Density profile results for unaged EPR insulation (dashed line) and for a sample aged to 172 Mrad at 1.2 Mrad/hr in air (solid curve). B: Density profile results for a sample aged to 117 Mrad at 87 krad/hr in nitrogen (dashed line) and for a sample aged to 150 Mrad at 220 krad/hr in air (solid curve). The percentage of the distance from the outside to the inside of the sample, P, is plotted as the abscissa. (From Reference 2.3.13)

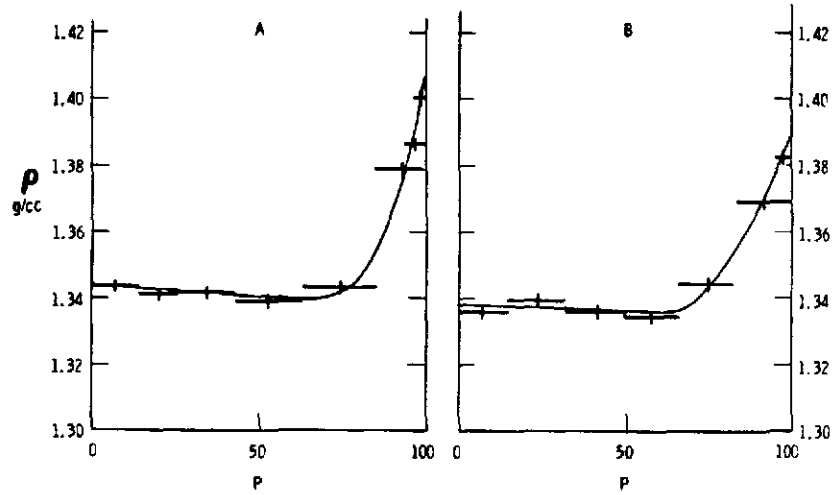


Figure 2.3.24 A: Density profile for EPR insulation aged to 135 Mrad at 52 krad/hr in air. B: Density profile for EPR insulation aged to 53 Mrad at 1.6 krad/hr in air. The percentage of the distance from the outside to the inside of the sample, P, is plotted as the abscissa. (From Reference 2.3.13)

1. It discovered a second chemical mechanism (copper catalyzed oxidation) which can lead to dose-rate effects under low temperature radiation-aging conditions.

2. It developed a new analytical technique, density profiling, which allows one to monitor and understand both physical (oxygen diffusion-limited) and chemical dose-rate effects.

3. It pointed out the potential complexities inherent in material degradation studies on non-realistic material configurations (e.g., compression-molded sheets for material screening purposes), since such studies may overlook either important interaction effects between materials or the possibility of material "poisoning" during processing of the real material.

Reference 2.3.14, NUREG/CR-3643 (SAND83-2493), describes two other profiling techniques which we developed and shows the kinds of information available from these and the density profiling technique. The metallographic polishing technique, which is qualitative but extremely fast and inexpensive, estimates the oxidation depth in oxidized samples by optical examination of metallographically polished cross-sectioned samples: oxidized and non-oxidized regions are distinguished by differences in surface reflectivity. An example of the capabilities of this technique for monitoring dose-rate effects is shown in Figure 2.3.25, where oxygen diffusion-limited effects are clearly important at high dose rates in air but become insignificant under lower dose rate conditions. The other new technique, hardness profiling, determines the shapes of degradation gradients by performing a series of sensitive measurements of relative hardness changes across the surface of cross-sectioned, polished samples. Hardness profiling results for the Viton material which had unusual mechanical property dose-rate behavior (Figure 2.3.21) are shown in Figure 2.3.26. These profiling results indicate that oxygen diffusion-limited degradation is responsible for the dramatic dose-rate effects for the mechanical properties. For radiation exposures under vacuum or at high dose rates in air, the material

undergoes primarily cross-linking to give a hard brittle material (penetration decreases substantially compared to the unirradiated sample). As the dose rate is lowered in air, oxidative scission progresses further and further into the sample, resulting eventually in a soft, stretchable material.

Reference 2.3.14 also calculated the total dose required to cause reaction of all the dissolved oxygen in various polymers. Table 2.3.1 indicates that total doses less than 0.1 Mrad would deplete all the initially dissolved oxygen in the interior of the polymers. Hence, unless the oxygen was replenished via diffusion at a rate faster than it was consumed by radiation, for the remainder of the aging irradiation the interior of the polymer would be starved of oxygen and oxidative degradation would be reduced. Reference 2.3.14 also summarized data on the depth of the oxidized regions for a collection of commercial polymer materials irradiated at several different dose rates and temperatures.

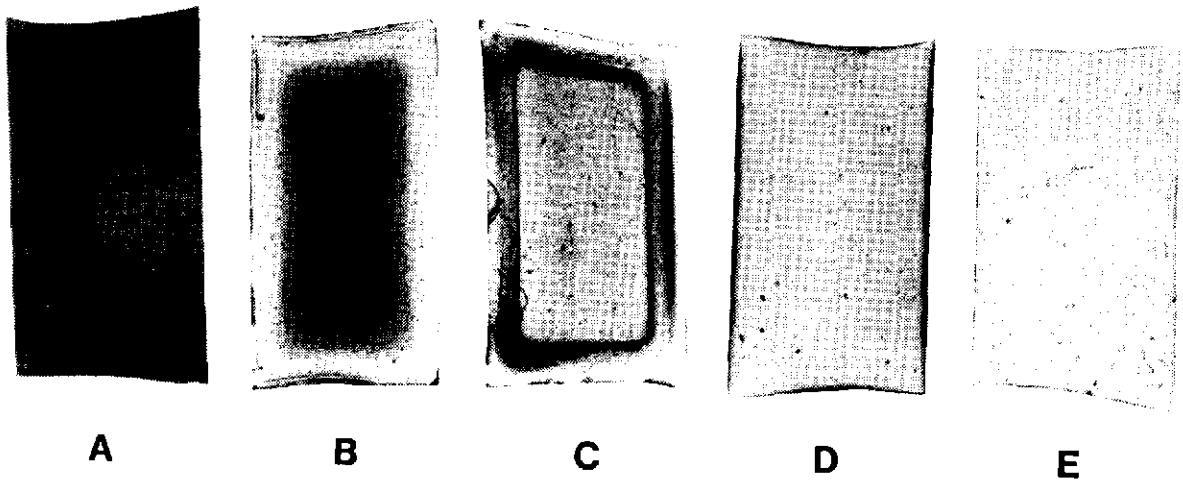


Figure 2.3.25 Cross-sectioned, polished samples of gamma irradiated EPR.
A: Unirradiated material
B: 6.7×10^5 rad/h (in air) to 165 Mrad
C: 6.7×10^5 rad/h (in air) to 297 Mrad
D: 1.1×10^5 rad/h (in air) to 175 Mrad
E: 1.1×10^6 rad/h (in vacuum) to 253 Mrad
All irradiations carried out at 70°C. Actual sample thickness = 3.13 mm. (From Reference 2.3.14)

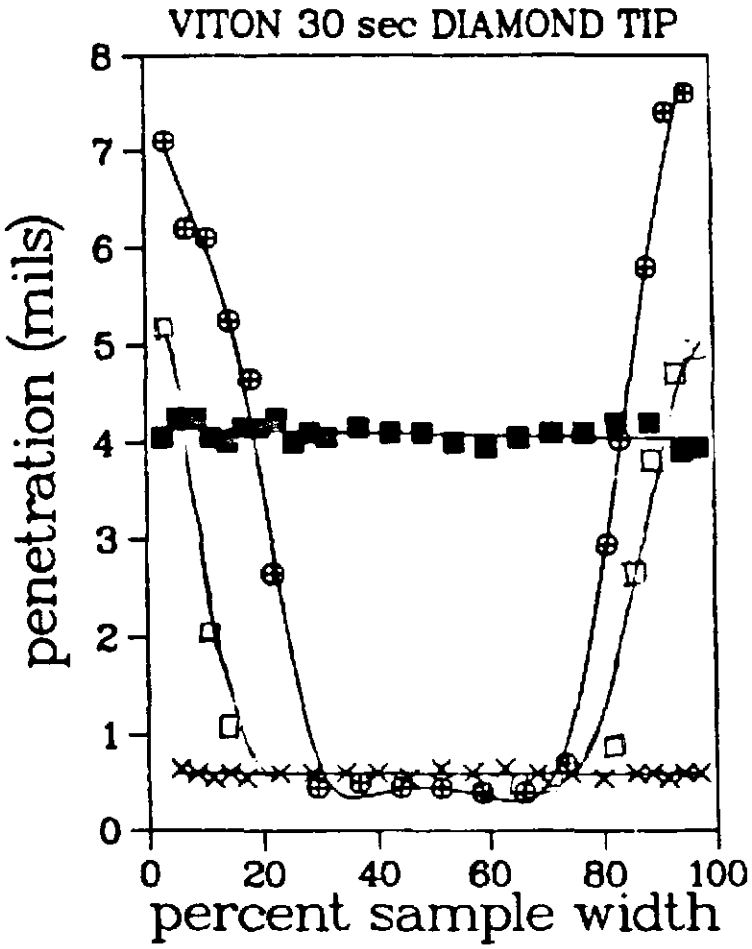


Figure 2.3.26 Hardness profiles for irradiated Viton® samples.
■ = unirradiated material
□ = 5.5 x 10^5 rad/h to 178 Mrad
⊕ = 1.8 x 10^5 rad/h to 186 Mrad
X = 9.0 x 10^5 rad/h to 191 Mrad (under vacuum)
Experimental load was 3 g. (From Reference 2.3.14)

Table 2.3.1
(From Reference 2.3.14)

Literature Values for Solubility and $G(-O_2)$ Yields, and Calculated Equivalent Dose Required to Cause Reaction of Dissolved Oxygen in Equilibrium With Air (.2 Atm O_2 pressure)

Polymer	S^* (ref)	$G(-O_2)$ (ref)	Equivalent Dose (Mrad)
PE	2.2×10^{-6} (18)	10 (23) 22 (24)	.04 .02
PVC	7.5×10^{-6} (22)	12 (25) 30 (25) 57 (26)	.15 .05 .03
EPR	5.4×10^{-6} (22)	20 (27) 40 (27) 60 (27)	.05 .03 .02
CLPO	3.1×10^{-6} (22)	10 (28)	.06

* Expressed in mol/g·atm

Reference 2.3.15, SAND83-2171C, gives a short review of the profiling techniques developed for this program and emphasizes their uses for understanding and monitoring dose-rate effects. By using profiling techniques to separate out the physical diffusion effects in radiation aging situations, we are able to more easily address any chemical dose-rate effects which may be present. In fact the profiling techniques often provide insights regarding chemical dose-rate mechanisms, as described earlier for the copper-catalyzed mechanism in the EPR cable insulating material.

An example of a case where the profiling techniques allowed us to quantitatively understand and model a chemical dose-rate effect is the PVC cable jacketing material from Savannah River. We had earlier concluded that the hydroperoxide breakdown chemical mechanism was an important contributor to the dose-rate and synergistic effects found for this material, but the complicated dose-rate and temperature behavior made it difficult to separate this mechanism from the diffusion dose-rate mechanism. Representative combined environment data are shown in Figure 2.3.27 from Reference 2.3.16, NUREG/CR-4008 (SAND84-1984). This figure plots the dose to equivalent damage (DED), where damage represents the lowering of the tensile elongation to 40% of original, as a function of the combined environment conditions (dose rate and temperature). Very complex dose rate and temperature effects are apparent. The profiling techniques as well as oxidation kinetics calculations allowed us to

determine the regions influenced by heterogeneous diffusion-limited oxidation effects. We could then concentrate on the homogeneous aging region at lower dose rates (see Figure 2.3.27), where the dose-rate and synergistic effects were due to the hydroperoxide breakdown mechanism. In this region a general kinetic model was derived. It was based on (1) a number of consensus reactions appropriate to oxidation chemistry, (2) unimolecular termination kinetics,and (3) rate-determining hydroperoxide-mediated branching reactions. Experimental evidence for these kinetics was presented. The model is generally applicable to hydroperoxide-mediated oxidations, a mechanism of wide-spread importance in both radiation and thermal aging situations. Figure 2.3.28 shows model predictions for dose-rate effects due to the hydroperoxide-breakdown mechanism as a function of a parameter R. At high dose rates, where the aging times are too short for significant hydroperoxide breakdown to occur, all curves approach horizontal behavior (no dose-rate effects). As the dose rate is lowered, dose-rate effects appear. When R, which represents the ratio of two kinetic rate constants, is greater than one, the model predicts that the dose-rate effects will eventually disappear as the dose-rate is lowered. This corresponds to the hydroperoxide concentration eventually reaching steady-state conditions. For values of R less than one, the hydroperoxide concentration continues to build up as the dose rate is lowered, resulting in never-ending dose-rate effects. Kinetic analysis of sequential (radiation followed by elevated temperature exposures) aging experiments gave an activation energy for the time-temperature component of the degradation mechanism (e.g., hydroperoxide breakdown). This allowed a time-temperature-dose-rate shifting procedure to be developed. We used this shifting procedure to shift the homogeneous and the borderline (nearly homogeneous) data of Figure 2.3.27 to a reference temperature of 43°C. The results are shown in Figure 2.3.29; the superposition for data covering a temperature range from 43°C to 110°C offers excellent evidence for the assumptions underlying the shift procedure. By utilizing the shift procedure, we have extended the 43°C data down to an equivalent dose rate of approximately 10 rad/h (0.1 Gy/h) which is representative of ambient nuclear power plant aging conditions. The solid curves in this figure represent the theoretical model fits to the shifted data; the excellent agreement between theory and experiment offers additional evidence in support of our approach. Further confirmation occurred when the resulting model predictions for the PVC material were shown to be quantitatively consistent with 12-year real-time aging results for identical material removed from a nuclear aging environment.

Reference 2.3.16 also showed how the results of this approach can be used to predict and simulate aging conditions. For example Figure 2.3.30 gives predicted mechanical property aging results for up to 40 year exposures at 10 rad/h (0.1 Gy/h) plus 45°C.

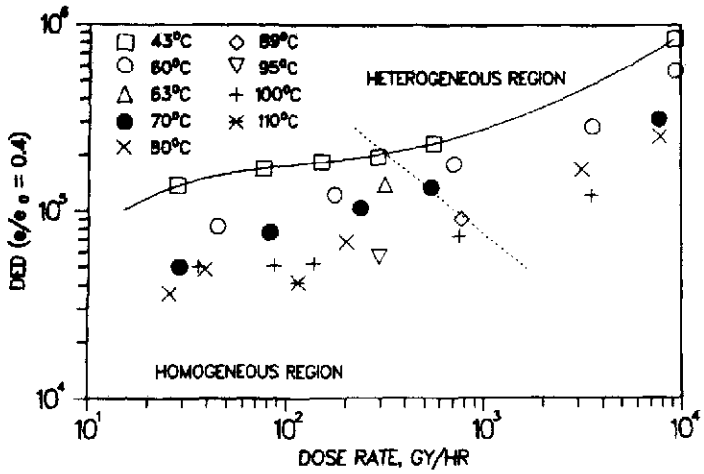


Figure 2.3.27 Radiation dose required for the elongation of PVC to decay to 40 percent of its unaged value at various dose rates and temperatures. To highlight the complicated dose-rate effects, a solid curve is drawn through the 43°C data. The dashed curve separates the homogeneous aging region from the heterogeneous aging region. (From Reference 2.3.16)

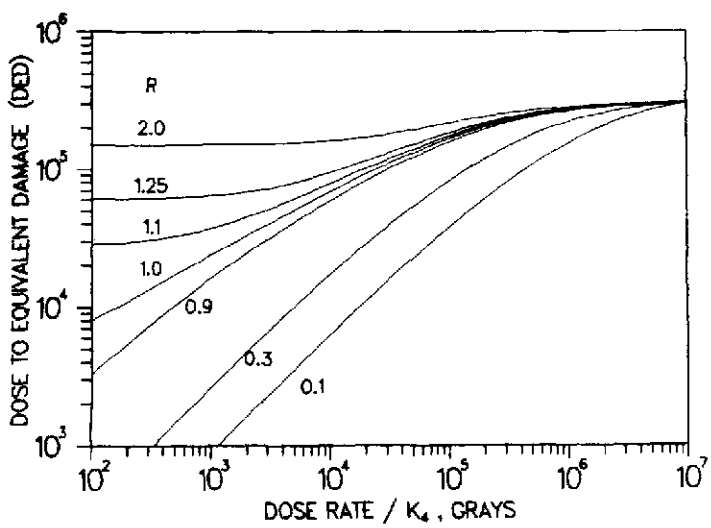


Figure 2.3.28 Theoretical model predictions for the relationship between the dose to equivalent damage (DED), and the dose rate divided by the rate constant for hydroperoxide breakdown. R is a positive constant which depends upon termination and propagation rate constants. (From Reference 2.3.16)

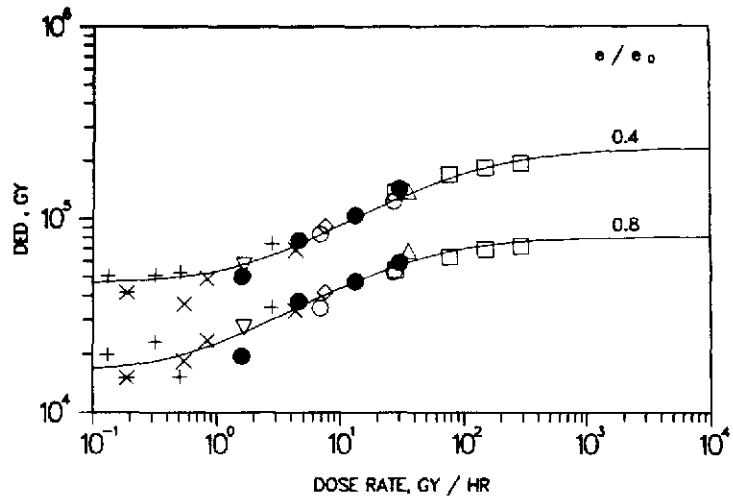


Figure 2.3.29 Upper data represents the homogeneous PVC data of Figure 2.3.31. (e/e_0 = 0.4) shifted to a reference temperature of 43°C. Symbols have same meaning as in Figure 2.3.31. Solid curve represents theoretical fit to the shifted data (R = 1.25, $k_4 = 5.6 \times 10^{-4} h^{-1}$). Lower data and curve are for e/e_0 = 0.8 data. (From Reference 2.3.16)

A number of significant finding resulted from Reference 2.3.16.

1. A quantitative general kinetic model for chemical dose-rate effects caused by the hydroperoxide breakdown mechanism was derived.

2. A method for using the model to make long-term predictions under nuclear power plant aging conditions was developed and successfully tested using experimental PVC data.

3. Predicted 12 year results from the modelled experimental data were confirmed using 12 year real-time aging data.

4. The approaches towards simulating dose-rate effects described in this paper represent a first attempt at a general methodology for handling dose rate effects.

This approach, involving the use of profiling techniques to address dose rate effects caused by diffusion-limited oxidation effects and kinetic modelling to address dose-rate effects caused by chemical mechanisms, therefore represents a significant step towards resolution of the second

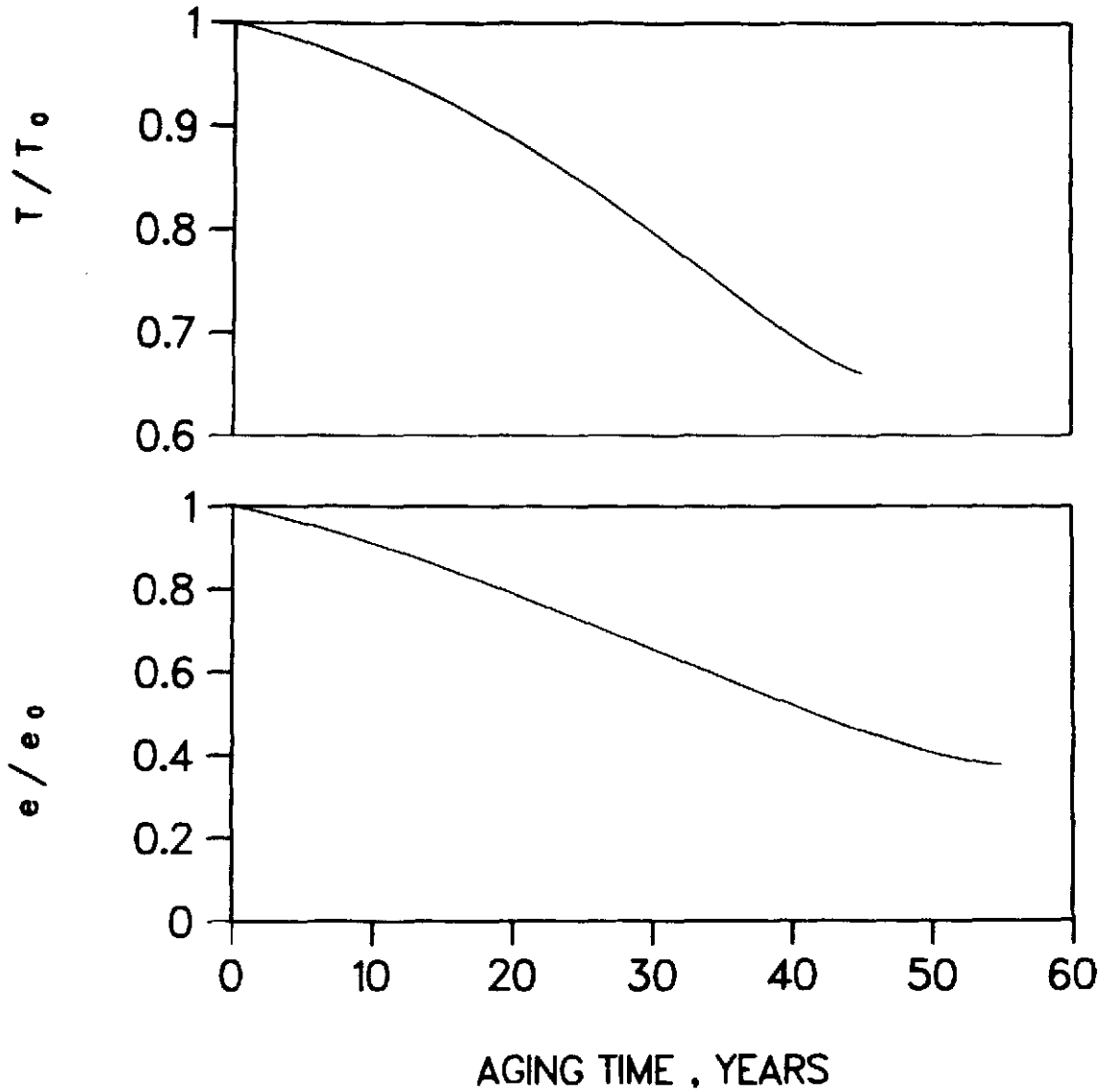


Figure 2.3.30 Mechanical property results predicted for PVC at 45°C plus 0.1 Gy/h. (From Reference 2.3.16)

part of this issue, "How must dose-rate effects be simulated?" As discussed in Reference 2.3.16, quite a few experiments are necessary for this PVC material to carry out these procedures due to the added complication of the chemical dose-rate mechanism. If, however, one is attempting to qualify a "complicated" material, the process can by necessity be complicated. For materials in which neither the hydroperoxide breakdown mechanism nor other chemical dose-rate mechanisms are important, the qualification process can become much easier since the equal damage-equal dose assumption may hold in the homogeneous region. In such a case, one would only have to insure (by means of profiling techniques for example) that the accelerated test was carried out at a dose rate sufficiently low that oxidation is essentially homogeneous, and that under homogeneous aging conditions, material damage is negligible. In other words, if dose-rate and synergistic effects for mildly elevated temperatures are not significant in the homogeneous aging region, accelerated predictions and simulations will be considerably simpler. More extensively documented qualification procedures will only be necessary for materials found to have more complicated (e.g., chemical) dose-rate effects.

Reference 2.3.17 (SAND83-0798J) was an article which included an extensive review on the evidence for dose-rate effects and approaches being developed for simulating such effects. In addition to the Sandia data and approaches, this article reviewed the results and approaches of a number of other research groups from around the world.

A shorter but more up-to-date review of some of our work is given in Reference 2.3.18 (SAND84-2115C). This report reviews some of the aging complications (including dose-rate effects) which occur in radiation environments and points out their effects on designing accelerated aging tests.

Reference 2.3.19, NUREG/CR-4358 (SAND85-1557), reviews in detail the density profiling technique and some of its important applications. Detailed descriptions are given of the experimental procedures as well as the use of the technique to study and monitor oxygen diffusion-limited degradation in radiation, heat aging and mechanical stress environments. An example of the use of the technique for discovering material incompatibility effects and examples of its use for recognizing and easily discovering the existence (or absence) of chemical dose-rate effects are also described. For example density profile data for a low density polyethylene material is given in Figure 2.3.31. The change in density per Mrad dose is profiled at three widely different dose rates. It is immediately clear from the results that significant oxygen diffusion-limited effects occur at the highest dose rate and that these effects essentially disappear by the time the lowest dose rate is reached. In addition the observation that the change in edge density per dose increases dramatically as the dose rate is lowered indicates the presence of important chemical dose-rate effects.

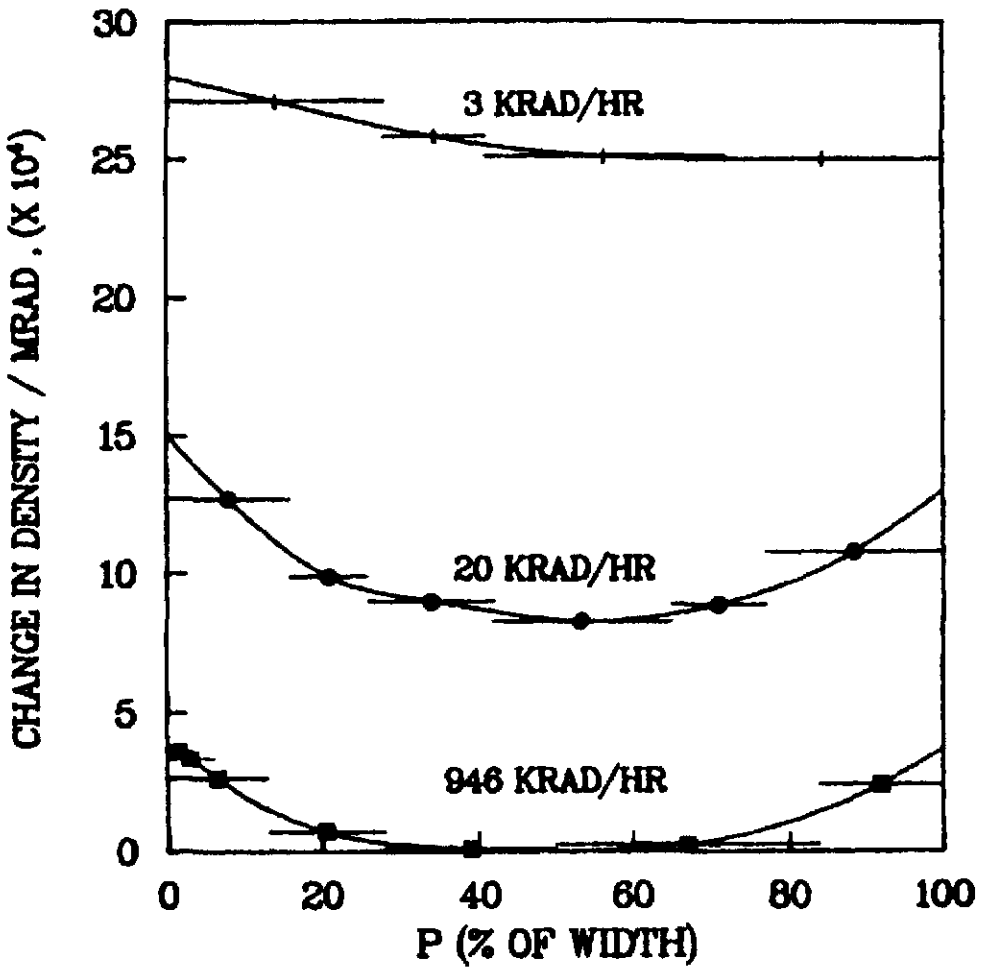


Figure 2.3.31 Profiles of the change in density per Mrad of radiation dose for three samples of the low density polyethylene material, which were aged in air at the three indicated dose rates. P represents the percentage of the distance from the outside edge to the inside edge of the sample. This distance averaged approximately 0.8 mm for this material. (From Reference 2.3.19)

Reference 2.3.20 reviews in detail the modulus profiling technique. The technique utilizes a new analytical apparatus and yields extremely useful information on mechanical property heterogeneities in polymers. The apparatus, which is based on extensive but relatively simple modifications of a commercial thermomechanical analyzer, is capable of obtaining greater than 20 quantitative tensile compliance measurements per mm of sample cross section in less than an hour. Since the inverse of the tensile compliance is closely related to the more-commonly measured tensile modulus, we refer to the technique as modulus profiling. A major application of the modulus profiling apparatus involves studies of radiation dose-rate effects in polymers and a number of representative examples of its utility for such studies are described. For example Figure 2.2.32 shows modulus profiles for a 1.5mm thick viton material which had been aged at 48°C in air to approximately 220 kGy total dose at the three indicated dose rates while under ~ 20% tensile strain. Diffusion-limited effects are very important at the highest dose rate, but are still present at dose rate conditions 20 times lower, even though the 0.33 kGy/h experiment represents a four week radiation exposure. Diffusion effects for this material can only be eliminated through the use of still lower dose rates or by going to a thinner sample. The results of the sample edges offer evidence that the results for D^{-1} at the surface are independent of dose rate, especially at the lower dose rates. Since diffusion-limited effects are absent at the surfaces, this preliminary data implies that chemical dose-rate effects are absent for this material over the range of dose-rate conditions studied.

Electronics

Reference 2.3.21, NUREG/CR-3156 (SAND82-2559), attempted a survey of the state-of-the-art in aging of electronics. The study contained these findings:

1. Semiconductor and integrated circuits are the most sensitive and vulnerable parts in an electronic circuit.

2. Typical tolerances for present electronics are about 10^4 rad (Si) gamma and about 10^{13} n/cm^2 for commercial hybrids and integrated circuits.

3. For specially fabricated or selected radiation-hardened devices, tolerances of about 10^6 rad (Si) gamma and about 10^{14} n/cm^2 are typical.

4. In most testing to date, the interaction of operational temperature and long-term irradiation have not been investigated.

5. To understand the nuclear plant radiation environment effects on semiconductor devices, more work is needed on low dose-rate, long-term radiation effects.

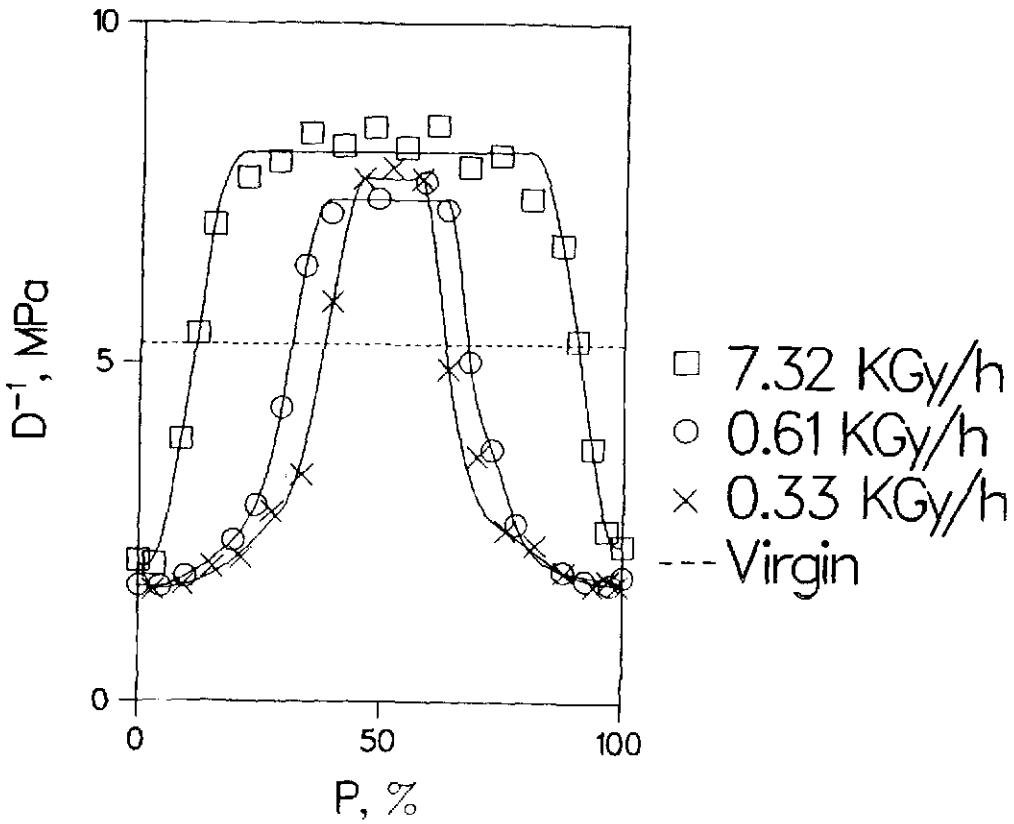


Figure 2.3.32 Modulus profiles of 1.5 mm thick viton aged in air at 48°C and the indicated dose rates. The dashed line represents the results for unaged material. P gives the percentage of the distance from one air-exposed surface to the opposite air-exposed surface. (From Reference 2.3.20)

6. Because the amount of damage sustained by a semiconductor is influenced by its circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

7. Both discrete devices and integrated circuits (op amps) are being used in some instrumentation in containment.

8. Combined neutron and gamma radiation environments should be examined in simulating radiation effects.

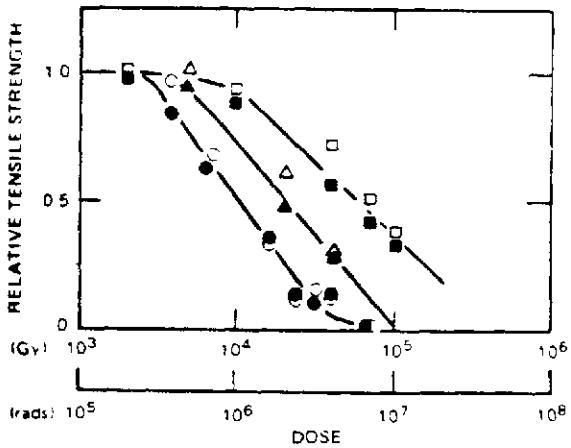
9. More work is needed to understand synergistic reactions and on predicting the effects of a combination of two or more stresses (e.g., radiation, temperature, humidity, chemical contamination, electrical stress, mechanical stress and other environmental factors).

International and Other U.S. Programs

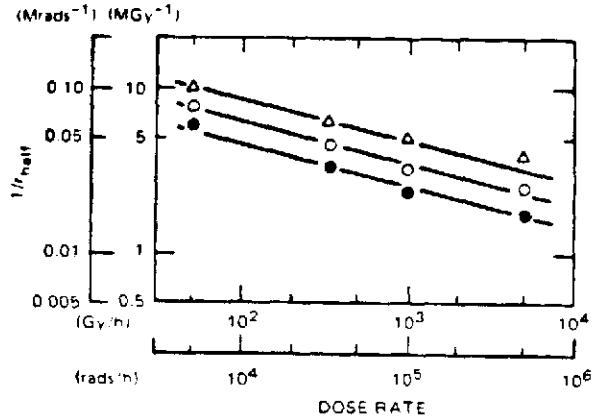
Organic Materials

With respect to the first part of this issue, are dose-rate effects an important consideration in aging simulations, there are numerous laboratories around the world that have generated data on polymers confirming the importance of these effects. Reference 2.3.17 reviews some of this data. Figure 2.3.33 provides some of the international dose-rate data that was presented in Reference 2.3.17. As far as attempting to develop methods/models for simulating dose-rate effects, publications describing such efforts have been made by researchers at Sandia, at the Japanese Atomic Energy Research Institute (JAERI) in Takasaki, at AERE Harwell in the United Kingdom, and at the European Organization for Nuclear Research (CERN).

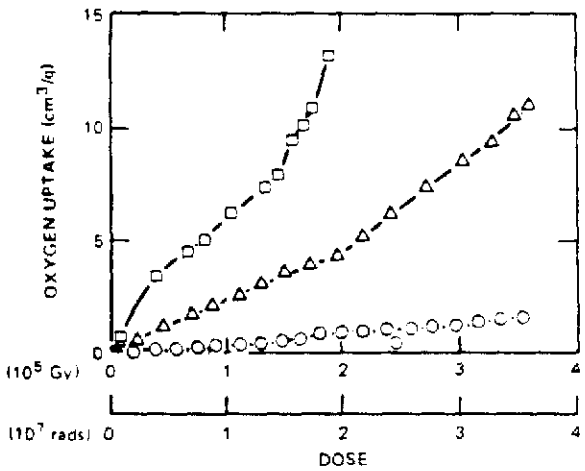
The emphasis at JAERI is on (1) material modifications to eliminate/reduce any dose-rate effects (Reference 2.3.22), (2) development of an oxygen over-pressure technique to allow reduction/elimination of oxygen diffusion-limited dose-rate effects at higher accelerated dose rates (Reference 2.3.23) and (3) modelling development to support the above efforts. Figure 2.3.34 illustrates tensile properties for an EPR material irradiated under different oxygen pressure conditions. Elongation properties are similar for 1 Mrad/h exposures with ambient air or vacuum conditions. Degradation is more substantial for a 5 krad/h exposure with ambient air pressures. By increasing the oxygen pressure to 5 atmospheres, .1 and .5 Mrad/h exposures also produced the more severe degradation. Reference 2.3.23 concluded that degradation for this EPR material depended only on the thickness of the oxidation layer during irradiation. The paper noted that previous JAERI efforts (Reference 2.2.24) had determined a relationship between the oxidation layer (ℓ), the oxygen pressure (P) during irradiation, the dose rate (I), the diffusion (D) and solubility



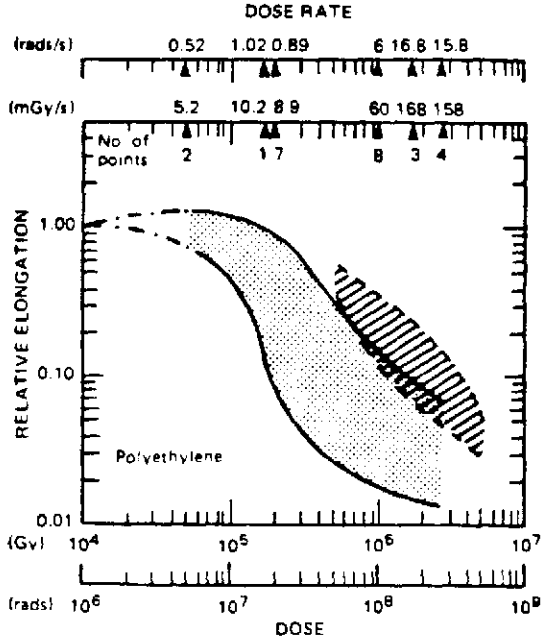
Relative tensile strength at break for polypropylene filaments irradiated in air at three dose rates: □ = 2 × 10^4 Gy/h (2 × 10^6 rads/h); △ = 4.3 × 10^2 Gy/h (4.3 × 10^4 rads/h); ○ = 45 Gy/h (4.5 × 10^3 rads/h). Solid symbols represent samples having an initial strength of 507 N/mm². Open symbols represent samples having an initial strength of 397 N/mm². Based on H. Wilski, Das Verhalten von Polypropylen bei der Einwirkung ionisierender Strahlen in Luft, *Prog. Colloid Polym. Sci.*, 58: 77 (1975).



Dose-rate effects as seen by relative gel fraction on polyethylene (PE) (△) and ethylene propylene rubber (EPR) (○), and by elongation-to-break (E_b) for polyethylene (●). The term r_{half} equals the half-dose required to bring about a given change in the material property of interest. Based on I. Kuriyama, N. Hayakawa, Y. Nakase, J. Ogura, H. Yagyu, and K. Kasai, Effect of Dose Rate on Degradation Behavior of Insulating Polymer Materials, *IEEE Trans. Electr. Insul.*, EI-14(5): 272-277 (October 1979).



Oxygen uptakes as a function of dose for chlorosulfonated polyethylene at 80°C at three different dose rates: ○ = 10^3 Gy/h (10^5 rads/h); △ = 10^2 Gy/h (10^4 rads/h); □ = 14 Gy/h (1400 rads/h). Based on J. Chenion, J. Berthet, G. Gaussens, and M. LeMuer, *Nuclear Qualification Results Obtained at the CEA and NRC-CEA Cooperative Research Program*, presented at the Tenth Water Reactor Safety Meeting, Gaithersburg, Md., Oct. 12-15, 1982.



Elongation at break normalized to the value before irradiation for 11 commercial polyethylene cable-insulating materials as a function of absorbed dose. The measured points fall within the shaded regions for different dose rates: the dotted area for low-to-intermediate dose rates, as shown on the upper scale, and the hatched area for irradiation at high dose rates of 50 Gy/s (1.8 × 10^4 Gy/h; 1.8 × 10^6 rads/h). The data clearly show a pronounced dose-rate effect. Based on P. Maier and A. Stolarz, *Accelerated and Long-Term Radiation Effects on Commercial Cable Insulating Materials*, European Organization for Nuclear Materials (Conseil Europeen de Recherche Nucleaire), Geneva,

Figure 2.3.33 Dose rate data illustrating some of the international effort. (From Reference 2.3.17)

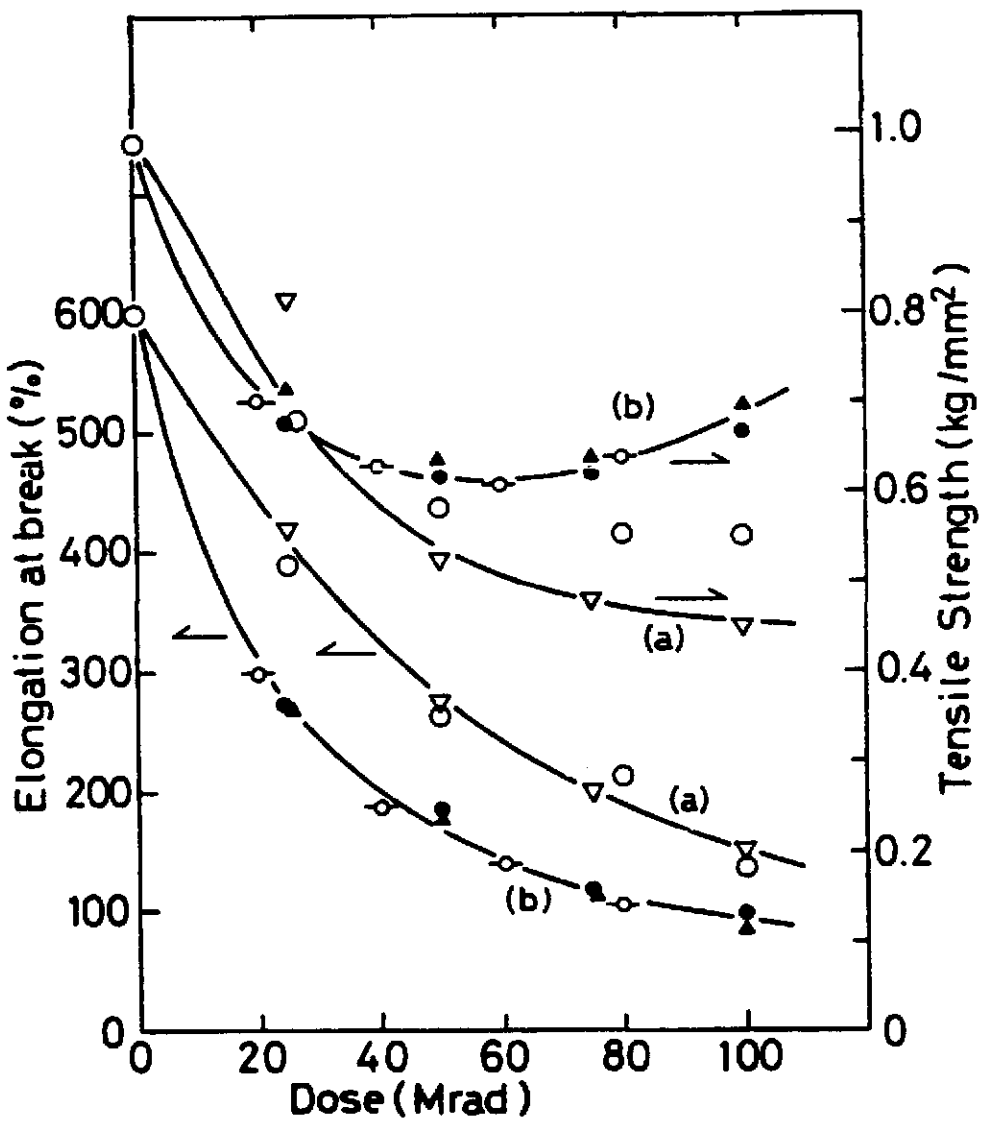


Figure 2.3.34 Elongation (E_b) and tensile strength (TS) at break against dose for formulated EPR irradiated in various conditions at room temperature. ▽ in vacuum with 1 Mrad/h ○ in air with 1 Mrad/h, -○- in air with 5 krad/h, ● in O_2 5 atm with 0.5 Mrad/h, ▲ in O_2 5 atm with 0.1 Mrad/h. (From Reference 2.3.22)

coefficients (S) of oxygen into the polymer material, and the oxygen consumption factor by radiation oxidation (Ψ).

$$\ell = K\,(P/I)^{1/2},\ K = (2DS/\Psi)^{1/2}$$

Though additional experimental data is not presented, Reference 2.3.23 noted that similar behavior as observed for the EPR was noted for tensile properties of polyethylene (PE), Neoprene, and Hypalon.

Efforts at AERE Harwell (Reference 2.3.25) in the United Kingdom are currently focused on time-temperature superposition techniques that might be applicable to accelerated aging of sealing materials during radiation exposures. For compression set measurements on a polyurethane elastomer, Reference 2.3.25 notes that oxygen diffusion produced dose-rate effects. It notes that the master degradation curves at each dose rate can also be superimposed by defining a shift factor proportional to the radiation dose rate to the 3/4 power.

Reference 2.3.26 (CERN 83-08) compares long-term irradiation results obtained at the CERN Super Proton Synchrotron (SPS) accelerator to short term irradiation results employed to predict cable insulation and jacket behavior. The report concludes that for some types of low-density polyethylene (PE) the reduction of elongation at break at the same dose may be up to a factor of 10 higher when irradiated at the CERN accelerator at low dose rates compared with irradiation at high dose rates. The same applies for the tensile strength of PVC. For other types of PE and PVC, and also for EPR, the difference in measured values at the same dose was considerably smaller or even negligible. To allow for extrapolation of short term irradiation results to long term conditions, Reference 2.3.26 discusses available models of dose-rate effects. It concludes that based on short-term irradiation tests only, safety factors between 1 and 10 may be warranted depending on the end-point criteria, the range of dose rates, and the materials studied.

Electronics

The French are known to have a program devoted to radiation effects on sensitive electronics, but complete details are lacking. References 2.3.27 (CEA Report #175) and 2.3.28 (CEA Report #176) are listed merely as examples.

Issue Findings

Organic Materials

The first part of this issue has been demonstrated by an extraordinary amount of data generated in the past few years--Dose-rate effects are an important factor which must be addressed in aging simulations. A large fraction of polymeric materials when aged at high radiation dose rates are found to have physical dose-rate effects caused by oxygen

diffusion-limited degradation. A much smaller fraction have chemical dose-rate effects.

The second part of the issue, the question of how dose-rate effects should be simulated, is probably the most important, yet complex, problem relevant to nuclear power plant aging simulations. Significant progress towards achieving this objective has been made. In fact the skeleton of an idealized approach based on our work is already being considered in industry standards (PAR 775- "Guidelines for Multistress Aging of Electrical Insulation in a Radiation Environment"). This approach involves the use of simple, inexpensive profiling techniques both to identify the importance of the physical dose-rate mechanism (e.g., oxygen diffusion-limited effects) and to determine how low a dose rate is necessary to eliminate these effects. If chemical dose-rate effects are unimportant, aging simulations can then be carried out by going to a dose rate low enough to eliminate the physical mechanism and then using the equal dose-equal damage assumption. When chemical dose-rate effects are identified, two viable approaches are possible. The material or component can be modified by changing material formulation to eliminate the chemical mechanism or a second material without chemical dose-rate effects can be substituted. Alternatively, in analogy with the hydroperoxide breakdown modelling described earlier, the chemical dose-rate mechanism can be modelled sufficiently to calculate conditions necessary to simulate ambient aging.

We have worked on a number of aspects of this idealized approach to refine it and make its implementation easier, thus making it more readily acceptable for inclusion in the appropriate standards and regulations and shifting to the applications-oriented aspects of this effort. These include:

(1) Development of inexpensive, simple profiling techniques. These techniques not only allow one to determine the importance of physical dose-rate effects (oxygen diffusion-limited degradation) and how low a dose rate is necessary to eliminate them, but are also useful for easily discovering/confirming the presence or absence of chemical dose-rate effects. Examples of how density profiling accomplished this were shown in Figures 2.3.23, 2.3.24 and 2.3.31. Another part of this effort was on the development of the micro-modulus profiling technique described above. It profiles the tensile modulus, a well-known and useful mechanical property; the results are therefore complementary to the oxygen-uptake results obtained from density measurements. In the instances when density profiling is impossible to apply, it is imperative to have other profiling methods available. These profiling methods will also have significant impact on validation of heat aging methodologies, as was discussed in Issue 2.2.

(2) Modelling of experimental profile shapes. Understanding how oxidation profile shapes depend on the material parameters

offers tremendous shortcuts to applying the methodology for dose-rate effect simulations. Instead of having to carry out a series of aging exposures at different dose rates and profiling the resulting samples to determine when oxygen diffusion-limited degradation is no longer significant, estimates of the proper dose-rate conditions can be calculated (and confirmed by profiling). This cuts down substantially the number of experiments required to eliminate diffusion effects. In addition, whenever the material configuration changes (e.g., thickness), a quick calculation of the effects on the diffusion mechanism eliminates the need for a new series of empirical experiments. It is also possible to use some simple material parameters to assess the potential for diffusion effects without having to do any experiments.

(3) Work on seal and gasket material to see what effect, if any, dose-rate effects have on materials which are mechanically stressed in their application environments. This is further discussed in Issue 2.5.

(4) Work on hard thermoset type plastics (e.g., epoxies). Dependent on sample thickness and oxygen permeation rates, diffusion effects for these materials may enter both for accelerated aging simulations and under ambient aging conditions.

In addition, a series of tests proposed in a joint program involving Sandia and JAERI could significantly impact this issue. The proposal includes experiments to test the possible application of the oxygen overpressure apparatus located at JAERI to develop methods for simulating dose-rate effects. In particular we are interested in whether oxygen overpressure can be used to eliminate diffusion effects at higher dose rates than air-aging allows and what effect this technique has on any chemical dose-rate effects which may be present. The profile shape modelling described above will also be useful for understanding these experiments.

In summary, we feel that we are on the verge of making significant contributions to predicting/simulating life in radiation environments. Using estimates of basic material parameters from literature studies, one should be able to apply modelling calculations to determine how low a dose rate is necessary to eliminate the oxygen diffusion dose-rate mechanism. To do this, data could be presented in a nomogram type format, since the same experimental variables (oxygen permeation and consumption rates plus sample thickness) are required for all materials. Simple, inexpensive profiling techniques can then be used both to confirm that diffusion effects are absent and to eliminate concern about possible chemical dose-rate effects. If both effects are absent, the simple equal damage-equal dose assumption for accelerated radiation aging will be acceptable. Only when chemical dose-rate effects are observed (a small percentage of materials have these effects) will more complicated accelerated aging procedures be necessary.

For materials and components which have dose-rate effects, acceptable methodologies which assure conservative aging simulation should be developed.

Organic Materials

In summary then, the findings within this subissue are:

1. Dose-rate effects can range from very large to insignificant, depending upon such factors as polymer type, aging conditions, sample geometry and degradation parameter being monitored.

2. Dose-rate effects always involve oxidation mechanisms. In inert environments, concern over radiation dose-rate effects is nonexistent.

3. Two general classes of dose-rate effects exist--physical, involving diffusion-limited oxidation, and chemical, involving a rate-determining chemical reaction.

4. Physical dose-rate effects depend upon sample thickness and oxygen consumption and oxygen permeation rates. When oxidation processes dominate the degradation of a material, conditions can always exist (e.g., high enough dose rates, thick enough samples) where these physical dose-rate effects will be important. If degradation is not limited by these diffusion effects under the long-time low dose-rate conditions appropriate to real-time aging, these effects sould be eliminated from accelerated aging simulations by using a sufficiently low dose rate.

5. Chemical dose-rate effects are not common. Two mechanisms have been identified and studies--the hydroperoxide-mediated mechanism and the copper-catalyzed oxidation mechanism. When chemical dose-rate effects are present, accelerated aging simulations become much more complex. Ideally the mechanisms should be understood in enough detail such that confident predictions can be made based on kinetic modelling or a similar approach.

6. When chemical dose-rate effects are identified, this usually implies that both synergistic effects of low-temperature radiation plus elevated thermal environments and sequential ordering effects will exist. This is due to the fact that all three effects are mechanistically dependent upon the same chemical reactions.

7. A general kinetic model has been developed for treating chemical dose-rate effects, synergistic effects, and ordering effects arising from peroxide breakdown. The model has been tested and shown to match very well with actual long-term data (12 years) from a nuclear plant.

8. The kinetic model predicts the existence of two types of limits for these dose-rate effects. Dependent upon kinetic rate constants, the dose-rate effects may either disappear on going to progressively lower dose rates or remain very strong.

9. Three new techniques have been developed for identifying heterogeneous oxidation, which results from the physical diffusion-limited dose-rate effect. These are: density profiling, relative hardness profiling and cross-sectional polishing. Chemical dose-rate effects can also be determined using these techniques.

10. Careful consideration should be given to the use of dose overstress (increased total dose) in equipment qualification to account for dose-rate effects, in that some materials tend to degrade in opposite ways at low and high dose rates.

Electronics

Numerous reports suggest that electronics are sensitive to radiation damage. Several worldwide research programs may produce useful data in this area.

In summary then, the findings within this subissue are:

1. Both discrete devices and integrated circuits (op amps) are used in some instrumentation in containment.

2. In most testing to date, the interaction of operational temperature and long-term irradiation have not been investigated. Because the amount of damage sustained by a semiconductor is influenced by its circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

3. Combined neutron and gamma radiation environments might be considered in simulating radiation effects.

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2.4 Simultaneous/Sequential Exposures

Issue Description

Real-time aging of components in nuclear power plant environments expose the polymeric materials in the components to a simultaneous combination of low level stress environments. For example a component may experience many years of aging under a combined environment of 50°C plus 10 rad/h radiation. In order to simulate this ambient aging using accelerated aging exposures, several different approaches are possible. If some type of interactions ("synergisms") occur between the two environments, the best approach would be the use of appropriate combined environment accelerated simulations. On the other hand, if synergistic effects were not important, sequential exposure to the two environments might adequately simulate the ambient aging conditions. The possibility also exists that sequential exposure to the two environments might adequately simulate cases where synergistic effects are important, thereby eliminating the necessity for more complex and expensive combined environment exposures. IEEE Std. 323-1974, the general standard for qualifying Class 1E equipment for nuclear power generating stations, states that "radiation shall be applied as a part of the sequence of environments representative of service conditions". The IEEE standard for Class 1E electric cables, field splices and connections, IEEE Std. 383-1974, allows the use of sequential thermal followed by radiation exposures as part of the method for demonstrating the operability of the cable after aging. This procedure, using sequential thermal followed by radiation exposures to simulate aging in a combined radiation-thermal environment, is widely applied.

This is the thrust of this effort to address the issue:

> "Can sequential exposures be used to simulate simultaneous radiation/thermal environments and does the order of the sequence matter?"

Organic Materials

Reference 2.4.1, NUREG-0237 (SAND76-0715), the first document in the aging program

1. concluded from an extensive review of literature results on radiation chemistry in polymers that significant evidence existed for the possible presence of important synergistic effects in combined radiation-thermal environments and

2. suggested an experimental program designed to generate data to determine the importance of dose-rate and synergistic effects.

If a material was determined to have important synergistic effects in combined radiation-thermal environments, it was probable that sequential aging simulations would give different material degradation results.

Reference 2.4.2, NUREG/CR-2877 (SAND81-2613), describes the versatile radiation-aging facility built for this program which allowed us to carry out the simultaneous and sequential aging experiments necessary to address this issue. A detailed description of this facility was presented in Issue 2.3.

Since we especially questioned the applicability of sequential aging exposures for materials with significant synergistic effects in combined radiation-thermal environments, our first sequential experiments were conducted on a PVC and a polyethylene material found to have significant synergistic effects (Figures 2.4.1 and 2.4.2). It is seen for both materials that simultaneous application of radiation and elevated temperature results in a striking increase in degradation over that expected based on the results for radiation at low temperature and for the elevated-temperature exposure. Reference 2.4.3, NUREG/CR-2156 (SAND80-2149), describes the sequential experiments conducted on these materials; some of these results are shown in Table 2.4.1 and Figures 2.4.3 and 2.4.4.

Table 2.4.1 Sequential Aging Experiments: Tensile Elongation Data (From Reference 2.4.3)

EXPERIMENT*	e/e_o**	
	PVC	PE
Unaged Material	1.0 ± 0.05	1.0 ± 0.1
γ; no subsequent T	0.80 ± 0.04	0.68 ± 0.09
T (in air); γ (in air)	0.68 ± 0.04	0.72 ± 0.07
γ (in air); T (in air)	0.32 ± 0.02	0.17 ± 0.04
simultaneous γ & T (in air)	0.26 ± 0.03	0.08 ± 0.03

* γ = gamma radiation - 4.5 krad/hr (for PE), 4 krad/hr (for PVC)- both at 25°C for 83 days
T = thermal exposure of 80°C for 83 days

** Sample tensile elongation divided by initial (unaged) elongation. e_o = 310% for PVC; e_o = 540% for PE

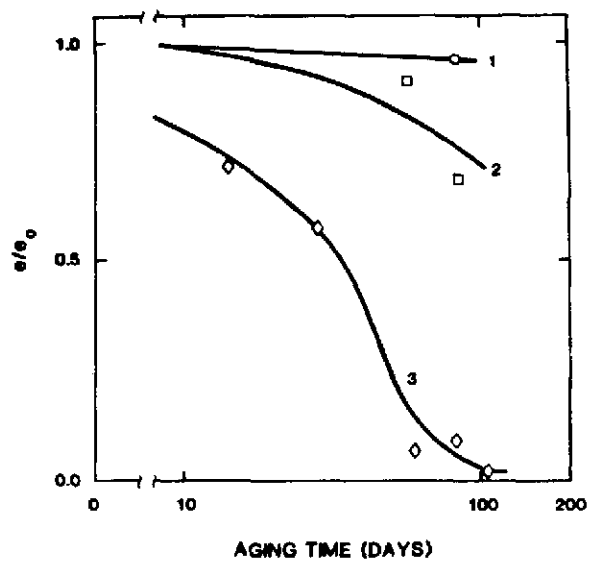

Figure 2.4.1 Tensile elongation results for PE insulation as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 5 krad/hr at 25°C; 3) radiation at elevated temperature, 5 krad/hr at 80°C. (From Reference 2.4.3)

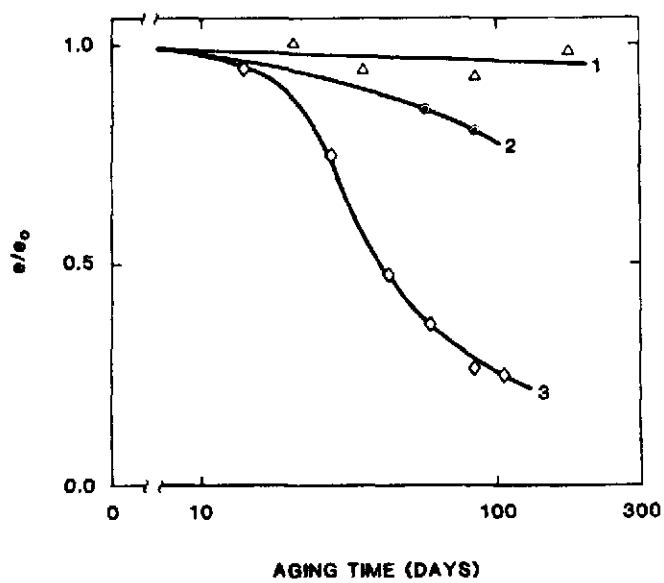

Figure 2.4.2 Tensile elongation results for PVC jacketing material as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 4.4 krad/hr at 25°C; 3) radiation at elevated temperature, 4.4 krad/hr at 80°C. (From Reference 2.4.3)

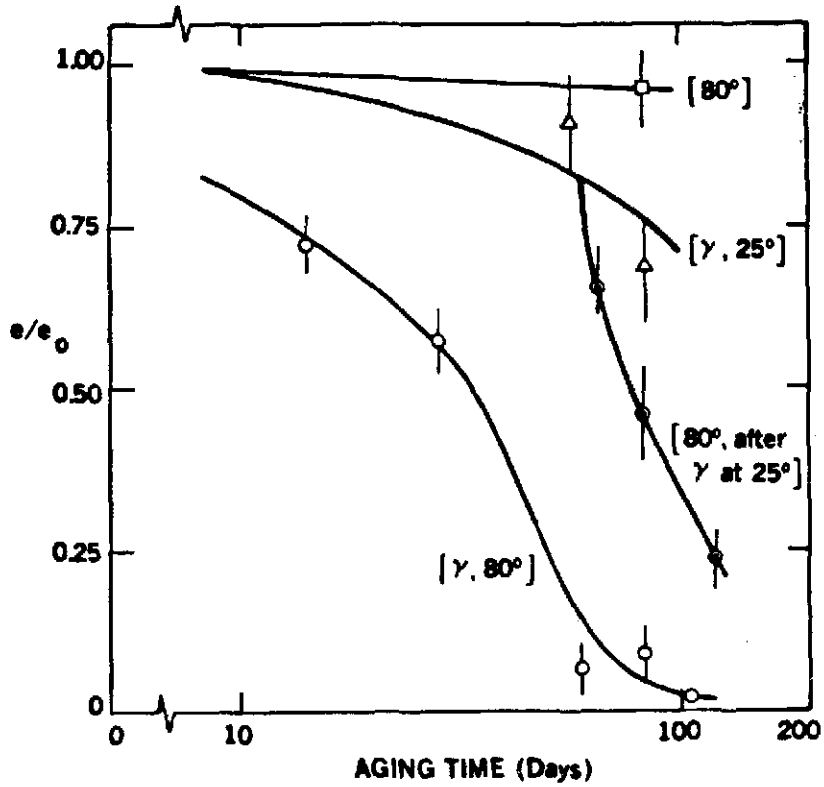

Figure 2.4.3 Aging of PE in various environments. Sample tensile elongation divided by initial (unaged) elongation is plotted versus aging time. (From Reference 2.4.3)

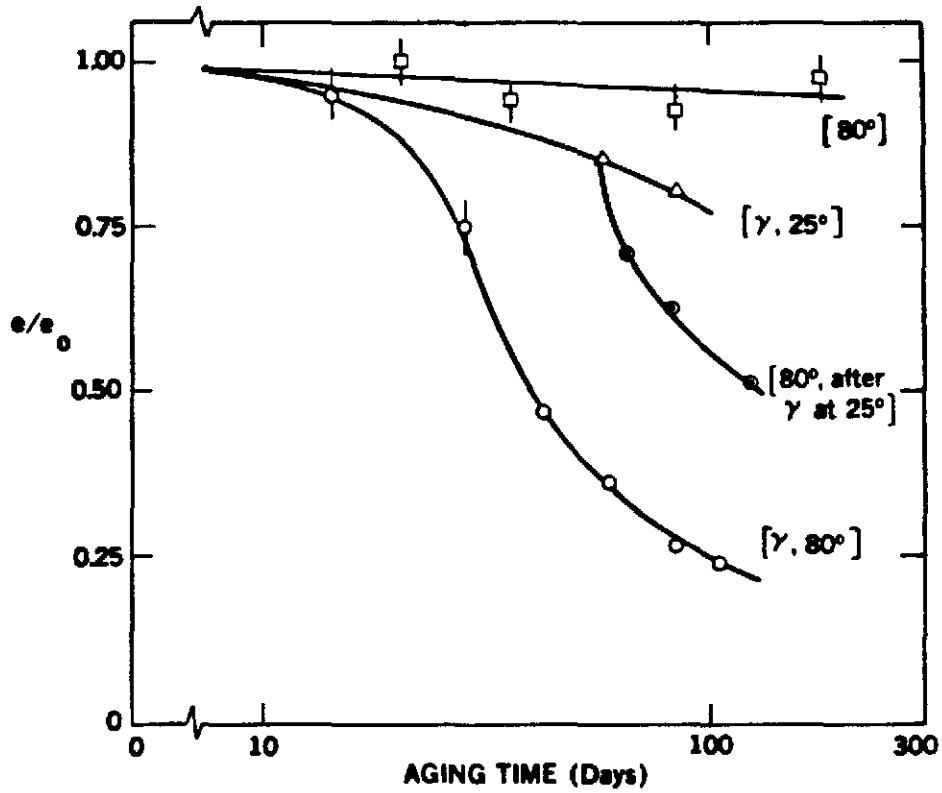

Figure 2.4.4 Aging of PVC in various environments. Sample tensile elongation divided by initial (unaged) elongation is plotted versus aging time. (From Reference 2.4.3)

It is clear that:

1. the order in which the two environments are administered is crucial: for these materials ambient-temperature radiation followed by elevated temperature is substantially more damaging than elevated temperature followed by ambient-temperature radiation,

2. the samples aged in the simultaneous environment are more severely damaged than those sequentially aged in radiation followed by temperature, and

3. evidence for the large synergistic effects found in combined environments is virtually absent from the temperature followed by radiation sequential exposures.

These results led to an important conclusion--the normal sequential exposure of thermal aging followed by ambient-temperature radiation-aging can underestimate material damage when strong synergistic effects of radiation and temperature exist.

Detailed experiments described in References 2.4.2 and 2.4.3 showed that these sequential ordering effects as well as the important dose-rate and synergistic effects can all be qualitatively explained in terms of the same degradation mechanism, involving the thermally-induced breakdown of relatively-stable, intermediate peroxides. This meant that the question of whether and how sequential aging could be used to simulate ambient aging might be answerable when appropriate methodologies for addressing dose-rate effects were developed.

We therefore set out to quantitatively understand and model the hydroperoxide breakdown mechanism. A detailed description of these studies is given in Issue 2.3 on the simulation of dose-rate effects. These studies resulted in a quantitative general kinetic model for hydroperoxide-mediated dose-rate effects, detailed in Reference 2.4.4, NUREG/CR-4008 (SAND84-1948). In addition to quantitatively explaining dose-rate and synergistic effects, the model allowed quantitative predictions to be made for sequential aging experiments, as shown in Figure 2.4.5 from Reference 2.4.4. This means that one could select sequential aging conditions which would exactly simulate the material damage expected during ambient combined environment exposures. This could allow one to avoid the more costly and difficult combined environment simulation.

Reference 2.4.5, NUREG/CR-2553 (SAND82-0346), addressed the effects of sequential aging on the important commercial material, ethylene propylene rubber (EPR). A number of commercial EPR materials were exposed to both sequential and simultaneous radiation-thermal aging stresses. Although some of the materials were insensitive to the aging method, others were sensitive to the order of the sequential application of stresses. Results for one material are summarized in Table 2.4.2.

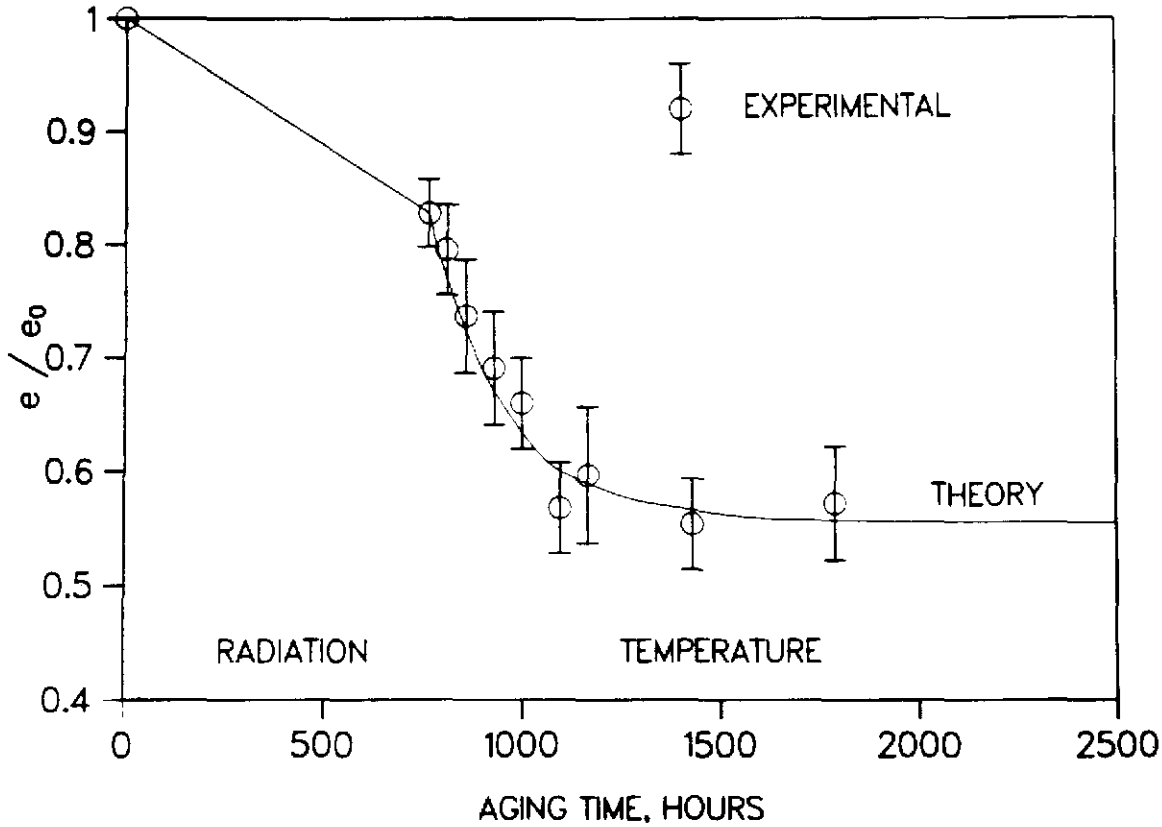

Figure 2.4.5 Relationship between tensile elongation and aging time for a sequential aging experiment. The experimental data (circles) is compared to the theoretical predictions (solid curve). (From Reference 2.4.4)

Table 2.4.2
Relative Tensile Properties of EPR A After Aging
(From Reference 2.4.5)

Aging Method	Center of Chamber Dose Rate in EPR (krd/hr)	Center of Chamber Total Dose in EPR (Mrd)	Ultimate Tensile Strength T/T_o	Ultimate Tensile Elongation e/e_o
1. Unaged	0	0	1.00 ± .03 (8.7 ± 0.3 MPa)	1.00 ± .08 (360 ± 30%)
2. Simultaneous 30 day radiation and thermal exposures	60 ± 4	43 ± 3	~0.2*	< .03*
3. Sequential 28 day thermal then radiation exposures	65 ± 5	44 ± 3	0.85 ± .03	0.33 ± .04
4. Sequential 28 day radiation then thermal exposures	65 ± 5	44 ± 3	0.26 ± .07*	< .03*
5. Sequential 28 day thermal then 55 hour radiation exposures	850 ± 60	47 ± 3	0.99 ± .21	0.31 ± .04
6. Sequential 55 hour radiation then 28 day thermal exposures	850 ± 60	47 ± 3	0.21 ± .02	0.06 ± .03
7. Simultaneous 7 day radiation and thermal exposures	290 ± 20	49 ± 3	0.26 ± .02	0.03 ± .03

NOTES: (1) Errors reflect one standard deviation of three measurements.
(2) Insulation thickness is nominally 0.8 mm.
* Samples were extremely brittle and sometimes cracked in the pneumatic jaws used for the tensile measurements.

The effects of sequential versus simultaneous aging and of the ordering during sequential exposures for a number of other important commercial materials was addressed in Reference 2.4.6, NUREG/CR-3629 (SAND83-2651).

The study concluded

1. For several materials, tensile properties at completion of aging were only slightly affected by both the irradiation temperature and the order of the sequential aging environmental exposures.

2. In general, the choice of irradiation temperature was secondary to the choice of aging sequence in its effect on polymer properties.

3. If sequential ordering of irradiation and thermal exposures was important to the aging degradation of tensile properties, usually the irradiation followed by thermal exposure was most severe.

Figure 2.4.6 from Reference 2.4.6 shows data for an EPR material in which the ordering of the sequential exposure has little effect on the degradation of the tensile elongation. Figure 2.4.7 shows data on a chlorosulfonated polyethylene (hypalon) material in which the radiation aging followed by thermal exposure sequence is more severe than the reverse sequence.

The studies described in References 2.4.5 and 2.4.6 were mainly concerned with empirical comparisons of aging procedures used prior to LOCA simulations. Thus, little attempt was made to probe the chemical and physical mechanism responsible for the observed ordering effects. It is likely that a combination of physical diffusion-limited oxidation and perhaps such chemical effects as peroxide breakdown or copper-catalyzed oxidation were significant contributing factors (see Issue 2.3 for more detailed descriptions of these mechanisms).

Electronics

Reference 2.4.7, NUREG/CR-3156 (SAND82-2559), attempted a survey of the state-of-the-art in aging of electronics. The study contained these findings:

1. Semiconductor and integrated circuits are the most sensitive and vulnerable parts in an electronic circuit.

2. Typical tolerances for present electronics are about 10^4 rad (Si) gamma and about 10^{13}n/cm^2 for commercial hybrids and integrated circuits.

3. For specially fabricated or selected radiation-hardened devices, tolerances of about 10^6 rad (Si) gamma and about 10^{14}n/cm^2 are typical.

4. In most testing to date, the interaction of operational temperature and long-term irradiation have not been investigated.

5. To understand the nuclear plant radiation environment effects on semiconductor devices, more work is needed on low dose-rate, long-term radiation effects.

6. Because the amount of damage sustained by a semiconductor is influenced by its circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

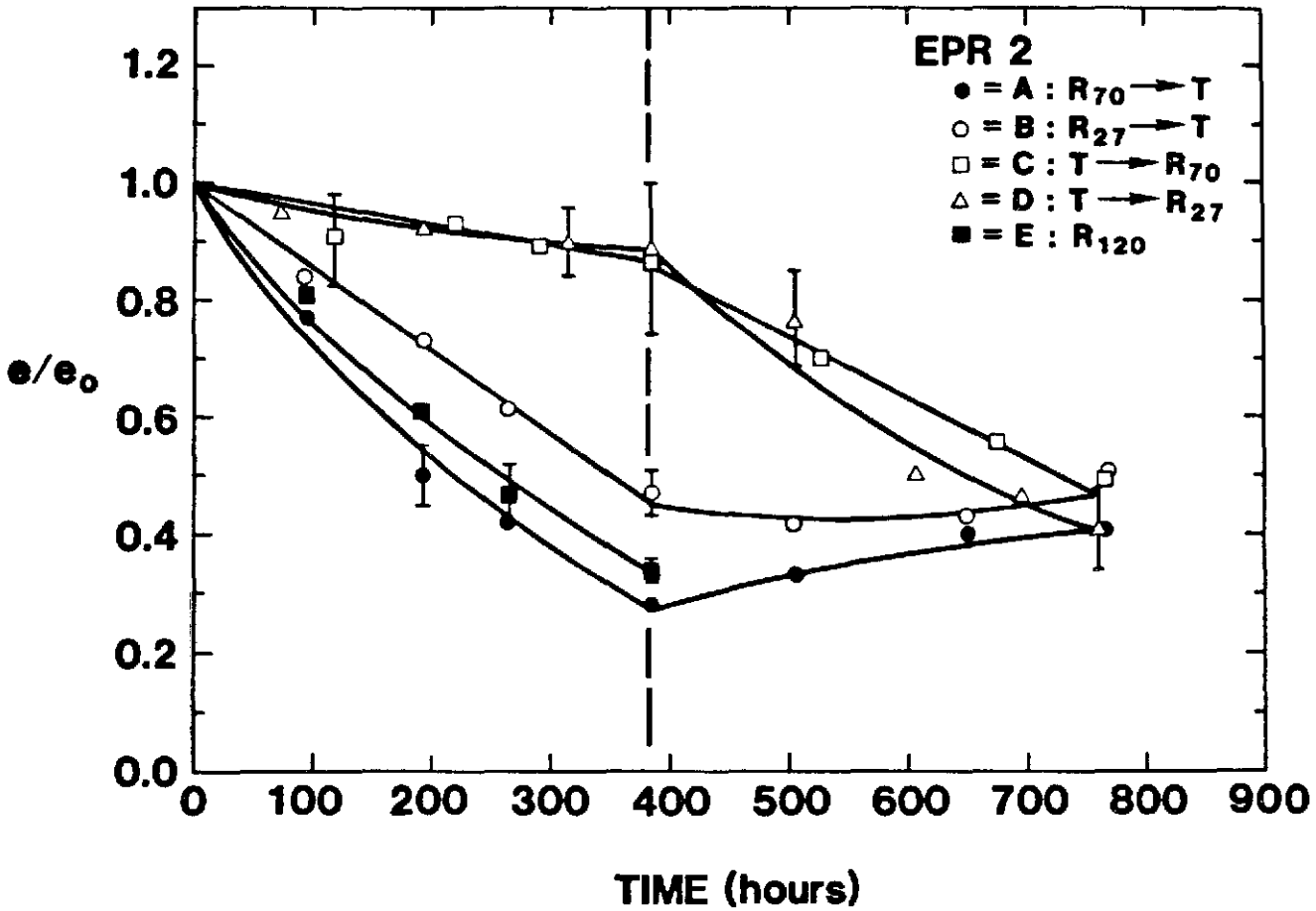

Figure 2.4.6 Ultimate Tensile Elongation of EPR 2 in Various Environments. Sample tensile elongation divided by initial (unaged) elongation is plotted versus aging time. Each portion of the sequential exposures lasted ~380 hours. (From Reference 2.4.6)

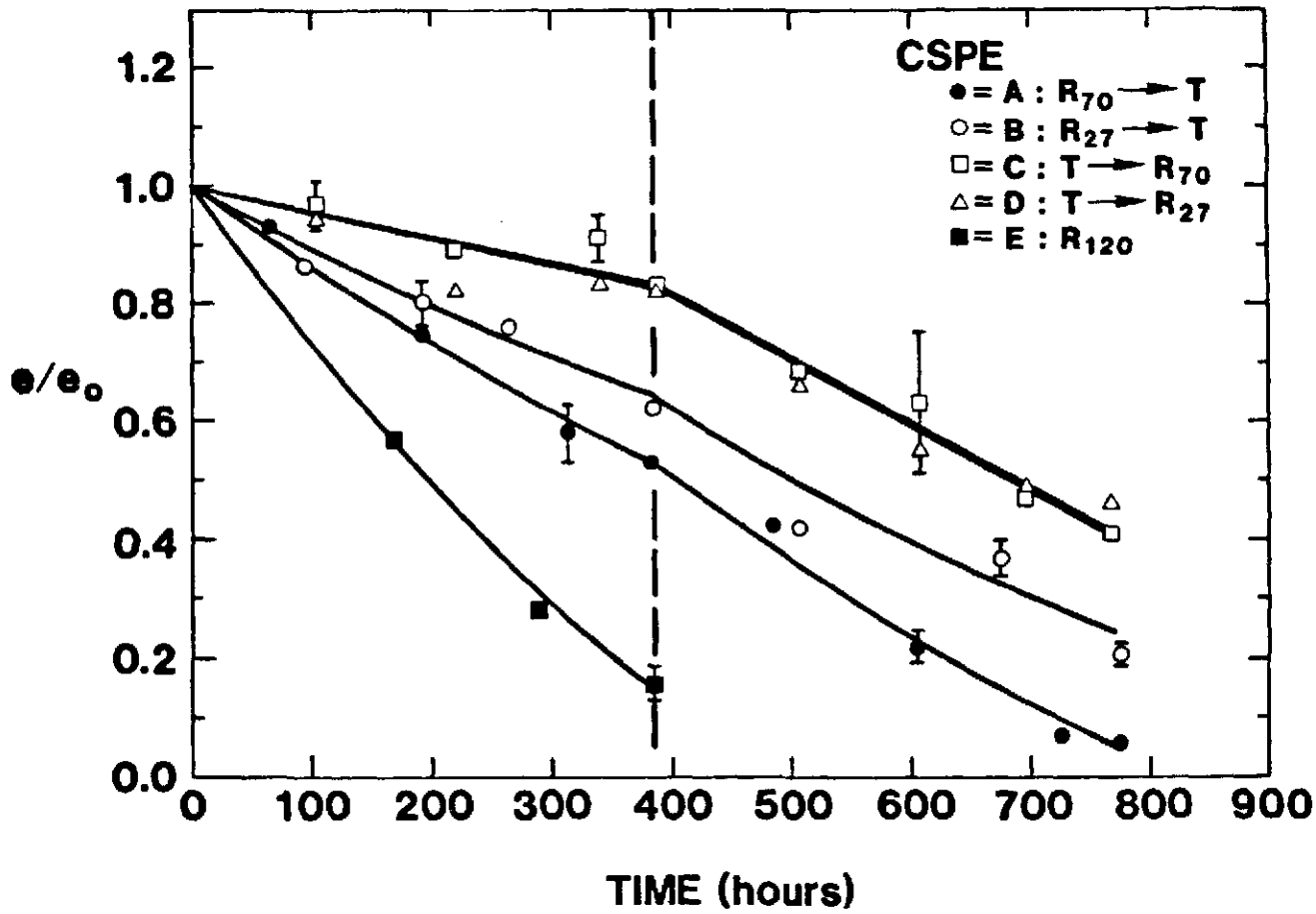

Figure 2.4.7 Ultimate Tensile Elongation of CSPE in Various Environments. Sample tensile elongation divided by initial (unaged) elongation is plotted versus aging time. Each portion of the sequential exposures lasted ~380 hours. (From Reference 2.4.6)

7. Both discrete devices and integrated circuits (op amps) are being used in some instrumentation in containment.

8. Combined neutron and gamma radiation environments should be examined in simulating radiation effects.

9. More work is needed to understand synergistic reactions and on predicting the effects of a combination of two or more stresses (e.g., radiation, temperature, humidity, chemical contamination, electrical stress, mechanical stress and other environmental factors).

International and Other U. S. Programs

Organic Materials

An early study regarding the effect of sequential versus simultaneous exposures to thermal and gamma-radiation environments was performed by Campbell (Reference 2.4.8) and reviewed by Carfagno (Reference 2.4.9). Table 2.4.3 summarizes some of Campbell's work on magnetic wire insulation. Reference 2.4.9 notes that "adding irradiation to thermal aging in sequence had a dramatic effect only on the polytetrafluoroethylene enamel/silicone varnish system; in this case the life was reduced to less than 0.1% of the thermal life. However, with one exception, when the heating and radiation exposures were combined, the life determined by the combined-environment experiments was always greater than the life obtained from sequential-exposure tests." Reference 2.4.9 notes that the combined-environment tests were conducted in closed test cells with no free circulation of air and, therefore, oxygen depletion may have occurred.

E. Oda, et al, (Reference 2.4.10), did observe for a fire-retardant EPR insulation material that the sequential elevated temperature exposure followed by irradiation did cause more ultimate tensile elongation degradation than the reverse order exposures.

Reference 2.4.11 presents test results for an EPR and Hypalon material. Simultaneous and sequential exposures to radiation and elevated temperature were employed. The radiation exposure followed by an elevated temperature exposure degraded the materials more seriously than did the reverse order exposure. Results are illustrated in Figures 2.4.8 and 2.4.9. T. Seguchi of the Japanese Atomic Energy Research Institute is currently trying to model simultaneous and sequential aging experiments through the use of empirical equations.

Electronics

We are not aware of any significant activities related to this subissue.

Table 2.4.3 Effects of Gamma Radiation on Thermal Life of Magnetic Wire Insulations. (From Reference 2.4.9)

INSULATION MATERIAL		(A) THERMAL AGING NO IRRADIATION	(B) THERMAL AGING FOLLOWING IRRADIATION		(C) COMBINED THERMAL AGING AND IRRADIATION	
Enamel	Varnish	Average Life (Hours)	Average Life (Hours)	Percent of Thermal Life	Average Life (Hours)	Ratio of *Combined* Life to *Sequential* Life
Modified Polyester	Oil-modified Phenolic	660	720	110	5125	7.1
Polyester	Oil-modified Phenolic	3160	2640	84	1260	0.5
Polyvinyl formal	None	280	240	86	560	2.3
Silicone	Silicone	350	320	90	570	1.8
Polytetra-fluoro-ethylene	Silicone	>10,000	12	<0.1	60	5.0

Issue Findings

Organic Materials

The aging response of many materials to sequential exposures (radiation followed by thermal and thermal followed by radiation) is similar to their response to a combined radiation plus thermal exposure. In the materials for which significant differences do occur, the radiation followed by thermal sequence is usually more severe than the thermal followed by radiation sequence and the former sequence more closely matches simultaneous exposure conditions. This indicates that the most conservative approach to sequential aging simulations should use the radiation followed by thermal sequence. Significant sequential ordering effects usually occur in materials which also have large synergisms of radiation and temperature and large radiation dose-rate effects because all three effects can be manifestations of the same chemical degradation

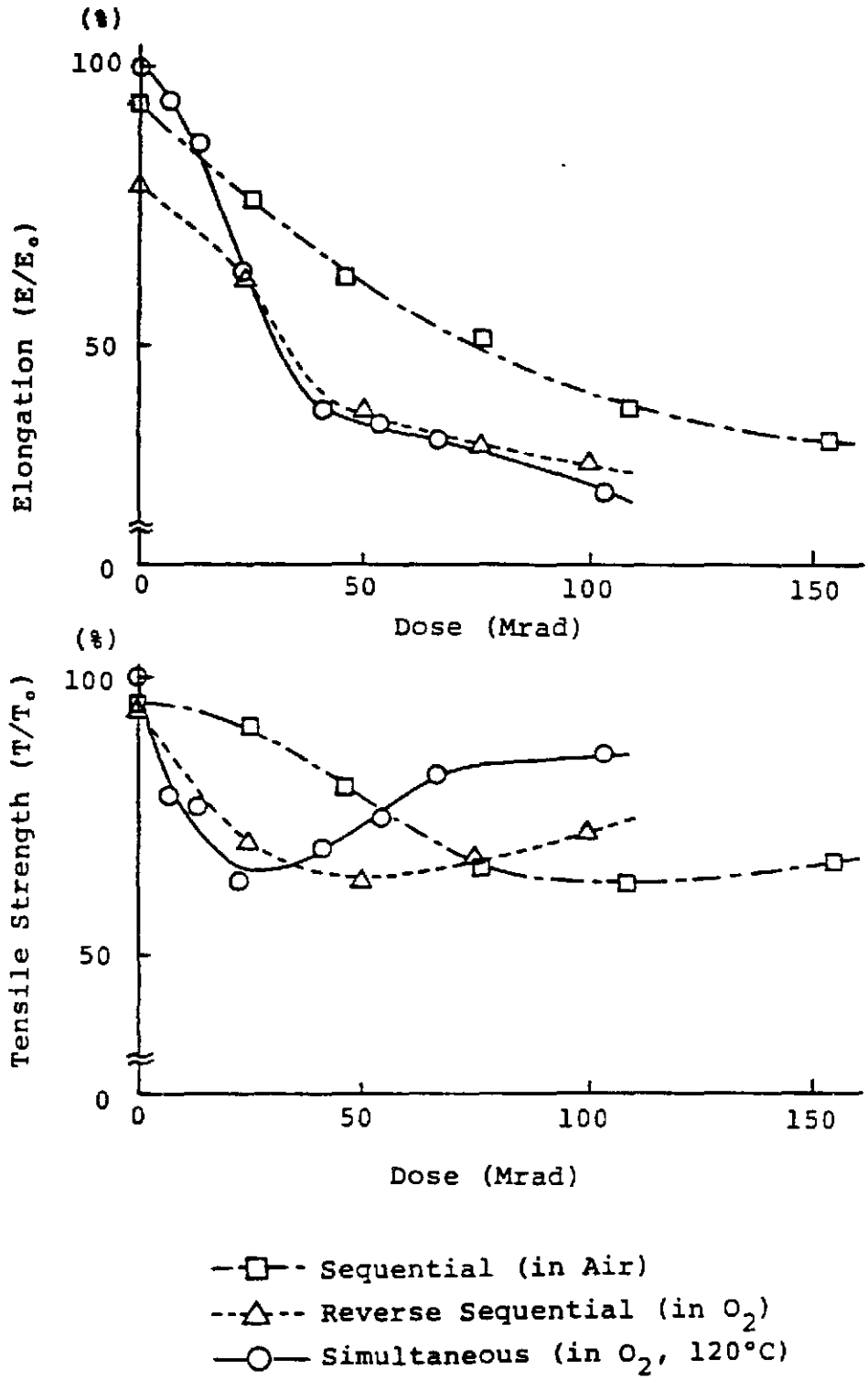

Figure 2.4.8 Sequential and Combination effect of thermal and radiation Stress on Degradation behavior of EPR Insulator from PH cables. (From Reference 2.4.11)

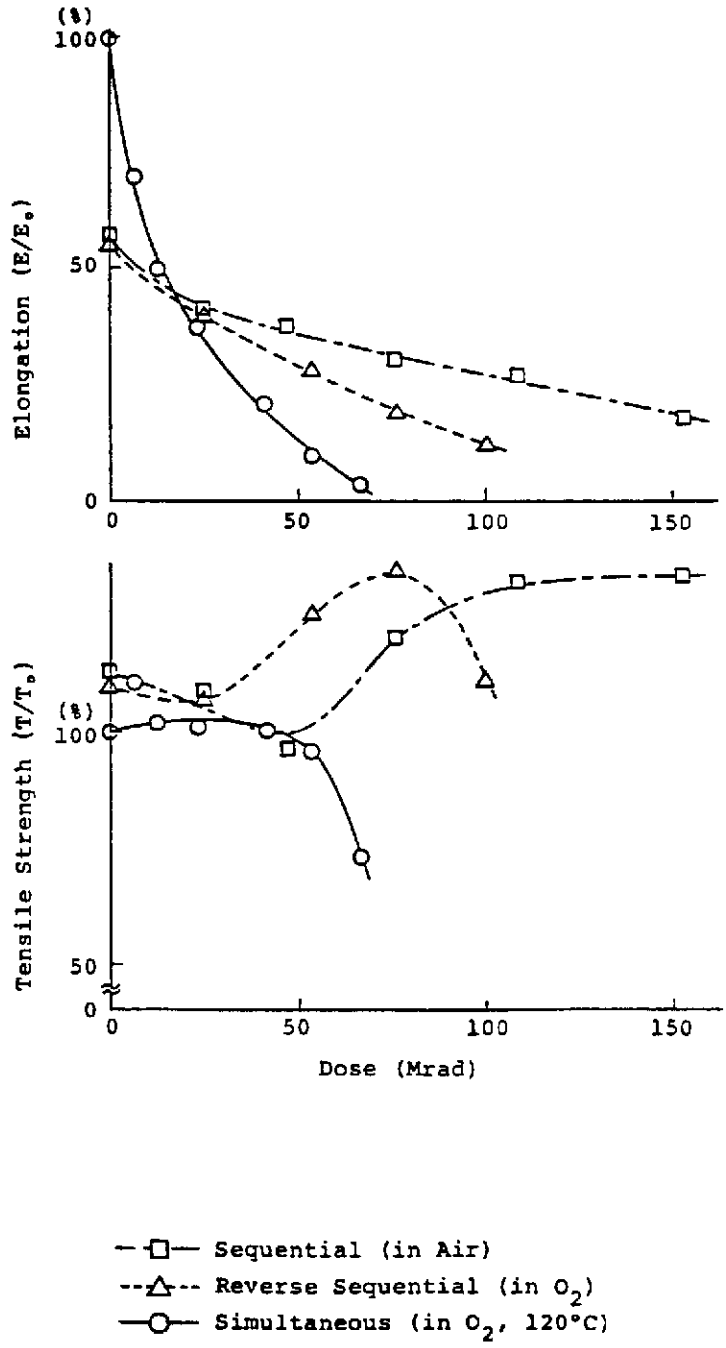

Figure 2.4.9 Sequential and Combination effect of thermal and radiation Stress on Degradation behavior of Hypalon Jacket from PH cables. (From Reference 2.4.11)

mechanisms. If sufficient knowledge exists of the mechanisms underlying dose-rate and synergistic effects, it is possible to select a sequential exposure which will simulate the material damage expected during real-time aging. This objective will come as a byproduct of the effort to simulate dose-rate effects (as illustrated in Figure 2.4.5). Thus the development of profiling techniques and the attempts to better understand diffusion and chemical mechanisms described in Issue 2.3 should significantly impact this issue as well.

By varying the sequence (or simultaneity) of exposures on identical test samples, the effect of the sequence on the test results can be observed and evaluated.

In summary then, the findings within this subissue are:

1. For several materials, tensile properties at completion of aging were only slightly affected by both the irradiation temperature and the order of the sequential aging environmental exposures. The importance of sequential effects is strongly dependent upon the polymer type and specific formulation.

2. In general, the choice of irradiation temperature was secondary to the choice of aging sequence in its effect on properties of certain materials.

3. When dose-rate or synergistic effects exist, sequential exposures do not always lead to material damage which is equivalent to simultaneous exposures.

4. The order used for sequential exposures can have a large influence on the observed material damage. The available evidence suggests that when differences occur, radiation followed by thermal aging is usually more damaging than thermal followed by radiation. In addition the damage caused in the radiation followed by thermal sequence more closely simulates the damage observed in simultaneous exposures.

5. In order to rigorously understand sequential aging, the same complications underlying dose-rate and synergistic effects should be considered, for instance the physical diffusion effects and any relevant chemical mechanism. As shown for the PVC material above, quantitative knowledge of these effects allows quantitative sequential aging predictions to be made.

6. With sufficient knowledge of the mechanisms (physical and chemical) responsible for polymer degradation in radiation environments, sequential aging conditions can be selected which will simulate the damage expected during a material's lifetime exposure to real-time combined environment conditions.

Electronics

Comparatively little information concerning the relative effects (and synergisms) of simultaneous versus sequential application of accelerated aging environments is available. Several worldwide research programs may produce useful data in this area.

In summary then, the findings within this subissue are:

> In most testing to date, the interaction of operational temperature and long-term irradiation have not been investigated. Because the amount of damage sustained by a semiconductor is influenced by its circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

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2.5 Mechanical Stress Effects

Issue Description

There are some Class 1 components which may experience mechanical stress loading in addition to other possible aging environments such as temperature, radiation and humidity. The effect of mechanical stress loading in combination with these other aging stresses is therefore an issue that should be addressed in any aging simulation. In addressing this issue, various candidate methods for simulating mechanical stress effects can be utilized. Seals, gaskets, and electric cables are examples of components that are subject to mechanical loading stress.

The IEEE standard governing type testing of electrical cables (383-1974), while permitting cold bending of cables during tests, does not provide for systematic variation of mechanical stresses.

This is the thrust of this effort to address the issue:

> "Is mechanical stress an important factor (e.g., in seals/gaskets) in the methods used for accelerated aging?"

Seals and Gaskets

At first glance this issue appears to be quite complicated, since it introduces an additional environmental stress (mechanical loading) to an already very complicated combination of other stresses (e.g., radiation, thermal, oxygen, humidity). Thus, in addition to the effects of mechanical stress alone, interactive or synergistic effects may occur. Fortunately, the situation is a good deal less complex than one might expect. For the typical elastomers used in seal and gasket applications, mechanical stress alone (e.g., at low temperatures) leads to one reversible phenomenon, physical relaxation. This mechanism, which is fairly well understood, leads to inconsequential long-term property changes. The addition of the other stresses lead to the chemical changes which are necessary for the material to fail. Many of these chemical changes are identical to or similar to the chemical changes responsible for radiation dose-rate effects or synergisms of radiation and temperature, so the knowledge gained in these areas is of great utility for understanding mechanical stress effects. There can be, of course, some additional complications. For instance, synergistic effects involving mechanical stress in combination with one of the other stresses can occur, as will be demonstrated below. Another additional complication comes from the different effects of similar chemical reactions on mechanically non-stressed versus stressed materials. Both scission and crosslinking processes will effect the mechanical properties of non-stressed materials, whereas crosslinking processes will have little effect on the sealing force for mechanically-stressed materials.

There are numerous polymeric seal and gasket materials used in nuclear power plant safety equipment and there is ample documentation indicating

a history of problems during real-time aging (see IE Bulletin 78-14 and IE Bulletin 79-01A, for example).

Numerous studies have evaluated candidate materials for seal and gasket applications in nuclear power plant applications, mostly through the use of compression set comparisons. Such studies have led to lists of materials which are recommended for radiation environments; these lists typically include ethylene propylene rubber, silicone, nitrile and viton materials. However, for the same generic type of material, some formulations are recommended and others are not. The most widely quoted study, Reference 2.5.1, recommends the following Parker compounds:

1. E740-75, an ethylene propylene rubber.

2. N674-70 and N-741-75, nitrile (Buna-N) rubber materials.

This study also states that

> "variations in compounding within the generic class can cause wide differences in properties. Early tests of nitriles, for example, discouraged their use in reactor environments for many years. However, later tests of other nitrile formulations showed their compression set properties were among the best when subjected to gamma radiation."

In general the most conservative simulation technique will lead to the highest level of confidence in the material chosen for an application. Currently the most widely-used method for screening candidate materials for seal and gasket applications is the compression set technique. As will be shown below, this method suffers from two potentially serious flaws.

In planning the experimental program, it was decided to concentrate on the most highly recommended formulations of each generic material available from Parker. These included

1. E740-75, the same EPR material mentioned above,

2. N741-75, one of the nitrile (Buna-N) materials mentioned above and

3. V747-75, a Viton material.

Additional materials of possible interest will also be screened in the program, e.g., silicone and neoprene materials.

Although some compression set measurements are planned, it was decided to devote more of the efforts towards stress-relaxation measurements. These measurements required development of a much more complicated experimental apparatus, but there were a number of important reasons for proposing this approach. Stress-relaxation techniques probe mechanical loading under tension, as opposed to the compression loading probed by compression set techniques. For typical seal and gasket applications,

the mechanical stress state is generally a combination of compression and tension (o-rings, for instance, are often stretched slightly in their o-ring groove). A distinct possibility exists that deterioration under mechanical tension may be much more rapid than under compression. For instance a small crack at the polymer surface, which would not propagate under compression, may propagate readily under tension, leading to failure. In this case the dominant degradation mechanism for a seal material exposed to a small component of tensile loading may be totally overlooked using compression set comparisons.

A second significant concern with compression set tests comes from the experience gained in radiation aging studies. This is the distinct possibility for important diffusion-limited oxidation effects during the high dose rate compression-set exposures. The large oxygen diffusion distances dictated by the compression set fixturing geometry almost guarantees diffusion effects. Since the long times associated with real-time exposures should allow the oxygen concentration in the seal material to remain at saturated equilibrium conditions, the compression set simulations may greatly underestimate the oxidative degradation. For this reason, the compression set measurements will be done using conditions which minimize diffusion-limited oxidation effects.

The configuration of the Sandia radiation-aging facility, with the Cobalt-60 permanently positioned under water, made the design of the in situ stress-relaxation apparatus a challenge. The sample clamping rig plus an apparatus for straining the sample, a transducer for monitoring the decay in stress and temperature control and gas flow capabilities all had to be included in a 6 inch diameter water tight aging can.

A prototype instrument, designed to meet the above objectives, has been built and successfully tested in a series of preliminary experiments. Based on these results, some minor modifications have been made to the apparatus and a second identical apparatus constructed. These instruments are now capable of carrying out in situ stress relaxation experiments in combined radiation/thermal environments under either flowing air or flowing nitrogen conditions. Long-term (many weeks) stress relaxation experiments in radiation environments, which have been virtually impossible until now, can be easily accomplished with this apparatus. Thus studies at low dose rates can be run to help develop methodologies for qualifying and screening seal and gasket materials. The air versus nitrogen capabilities will aid in understanding oxidation effects. Since the apparatus is temperature-controlled, it can easily be used to isothermally turn on and off the chemical degradation mechanisms (by inserting or removing the can from the Cobalt-60 environment). This will allow the combined effects of physical and chemical relaxation to be separated, leading to quantitative methods for understanding the chemical processes responsible for degradation, thus allowing for assessment of the adequacy of accelerated aging techniques.

Some of the early experiments using this versatile facility have already underscored the wealth of information available and have justified the

choice of stress relaxation measurements to follow mechanical stress effects.

One set of experiments compared radiation stress relaxation in air and nitrogen environments. The most dramatic difference occurred for the nitrile (Buna-N) rubber material. In nitrogen environments (Figure 2.5.1), rapid crosslinking occurred, which led to sample shrinkage and an eventual increase in stress. In air environments (Figure 2.5.2), the material developed numerous surface cracks which rapidly propagated until catastrophic sample failure occurred after a few Mrad total radiation dose. This same material performed extremely well in compression set experiments in which it was room-temperature aged for 16 days at 65 krad/h, as shown in Table 2.5.1 from Reference 2.5.2, NUREG/CR-3629 (SAND83-2651). In fact its compression set results under these conditions (R_{27} in the Table) was superior to all of the other materials tested. Thus the concerns mentioned earlier about possible differences between tensile and compressive loading is a reality for one of the formulations recommended for nuclear power plant seal and gasket applications.

Other early experiments have confirmed the existence of important oxygen diffusion-limited degradation during typical stress relaxation measurements in radiation. Figure 2.5.3 from Reference 2.5.3, NUREG/CR-4358 (SAND85-1557), shows a density profile of the EPR rubber material after radiation stress relaxation at 850 krad/h and 48°C. Enhanced density (oxidation) at the sample surfaces is evident. If such diffusion effects are present in stress-relaxation experiments, they should be much more important in compression set experiments since oxygen accessibility will be greatly reduced under similar accelerated radiation conditions.

Chemical switching experiments, in which the chemistry is turned on and off isothermally by inserting the can in and removing it from radiation, are also proving to be extremely helpful. Some typical results are shown for a viton material in Figure 2.5.4. Analysis of such experiments shows that from a practical point of view, the chemistry is turned on and off instantaneously. This allows quantitative separation of the chemical effects and should therefore allow us to understand and extrapolate the mechanisms of relevance to the functional deterioration of these materials.

Electric Cables

Electric cables are potentially subject to two failure mechanisms which can be caused by aging mechanical stress. First, if the cable is stretched by an applied force (or by its own weight) over an edge with small curvature, the metal wires will gradually creep through the soft polymeric insulation resulting in metallic contact between cable wires or a wire and the cable support, shorting the cable (creep shortout). Second, the polymeric materials embrittled by aging may crack under mechanical stress; in the presence of subsequent accident conditions

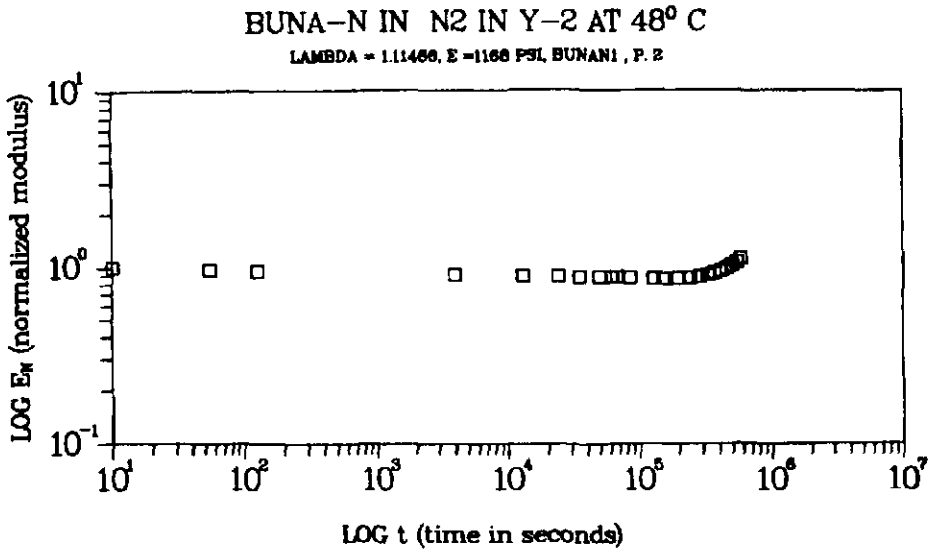

Figure 2.5.1 Stress relaxation data for nitrile rubber material aged in nitrogen at 806 krad/h and 48°C (unpublished data).

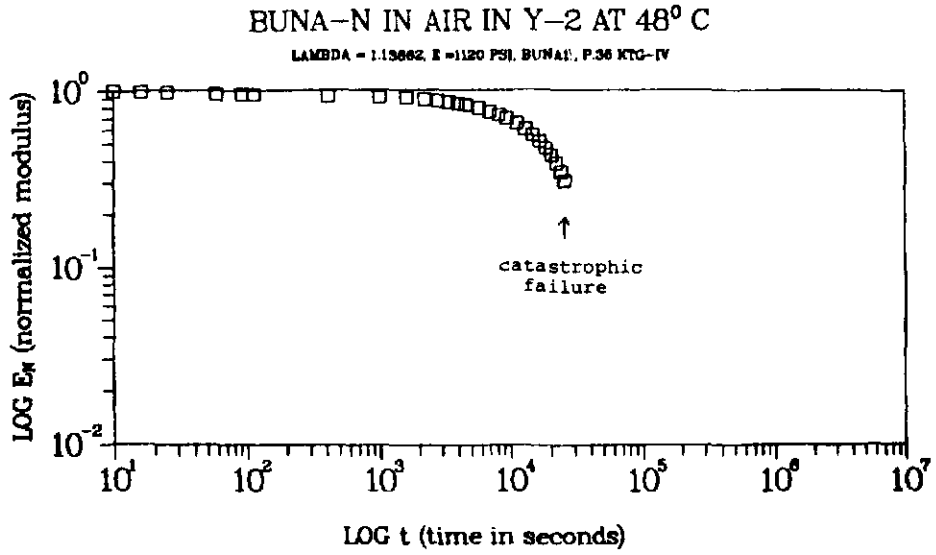

Figure 2.5.2 Stress relaxation data for nitrile rubber material aged in air at 818 krad/h and 48°C; failure occurred after about 5.8 Mrad total dose (unpublished data).

Table 2.5.1
U.S. Compression Set Test Results
(From Reference 2.5.2)

	EPR A	EPR B	BUNA N	SILICONE	VITON
Unaged (#38)	.09 ± .01	.11 ± .02	.07 ± .01	.07 ± .03	.17 ± .01
Partially Aged:					
R_{27} (#12)	.54 ± .04	.78 ± .03	.48 ± /01	.82 ± .02	.80 ± .05
120 (337)	.31 ± .07	.73 ± .05	.39 ± .02	.17 ± .01	.27 ± .03
Aged:					
R_{120} (#32)	.74 ± .03	1.0	.68 ± .01	1.0	.97 ± .02
$R_{27} \rightarrow 120$					
#13	.67 ± .02	.96 ± .03	.63 ± .01	.83 ± .02	.88 ± .01
#14	.67 ± .03	.98 ± .03	.59 ± .02	.86 ± .01	.89 ± .01
$120 \rightarrow R_{27}$					
#20	.68 ± .04	.91 ± .02	.68 ± .02	.72 ± .01	.80 ± .01
#50	.79 ± .06	.95 ± .01	.71 ± .02	.81 ± .01	.83 ± .02

*Errors Reflect ± σ for Four Samples. Absolute Error is ± .10.

$R_{70} \rightarrow 120°C$.: A 16-day irradiation at ~65 krd/h and 70°C followed by a 16-day thermal exposure at 120°C.

$R_{27} \rightarrow 120°C$: A 16-day irradiation at ~65 krd/h and ambient temperatures (~27°C) followed by a 16-day thermal exposure at 120°C.

$120°C \rightarrow R_{70}$: A 16-day thermal exposure at 120°C followed by 16-day irradiation at ~65 krd/h and 70°C.

$120°C \rightarrow R_{27}$: A 16-day thermal exposure at 120°C followed by a 16-day irradiation at ~65 krd/h and ambient temperatures (27°C).

R_{120}: A 16-day simultaneous exposure to 120°C thermal and ~65 krd/h irradiation environments.

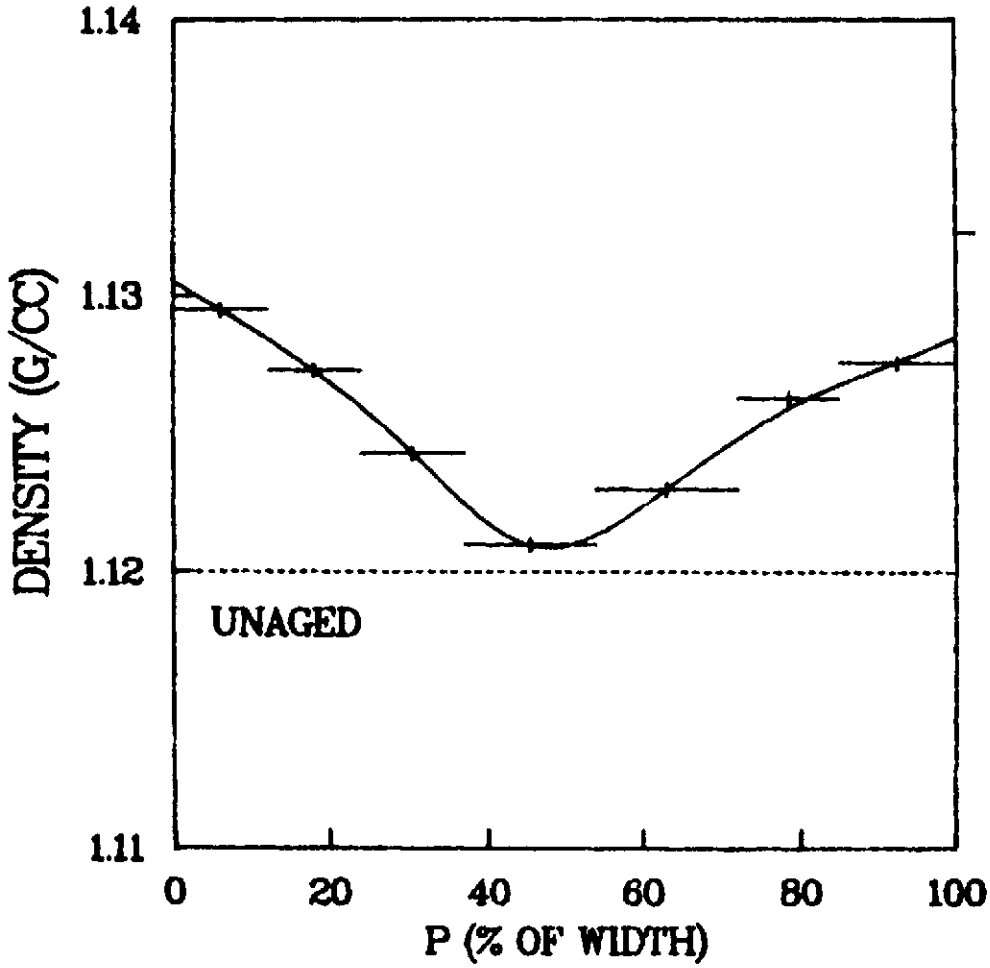

Figure 2.5.3 Density profile of an EPR O-ring material after exposure to mechanical tensile stress during radiation aging. (From Reference 2.5.3)

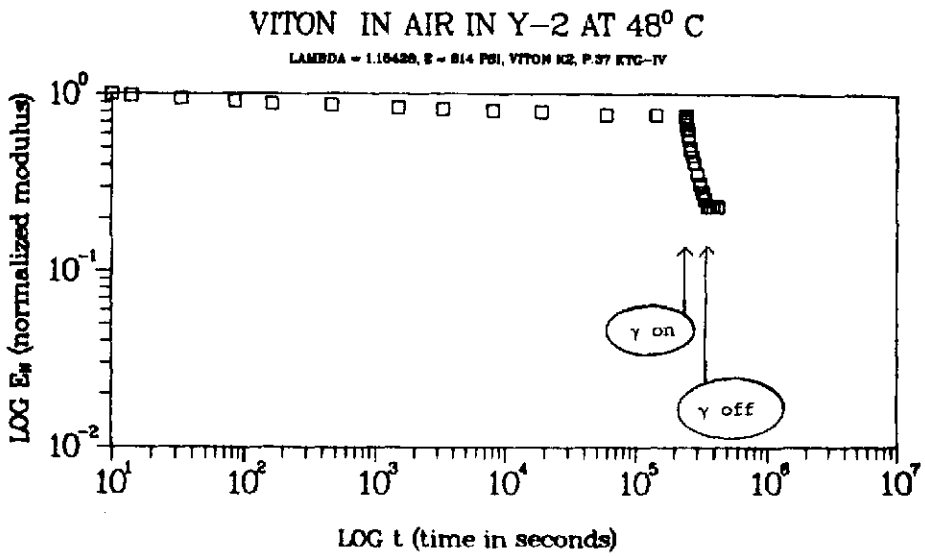

Figure 2.5.4 Chemical switching stress-relaxation results for a viton material aged at 732 krad/hr and 48°C. (Unpublished data.)

(steam and spray), strong leakage currents or shortout may occur. The two mechanisms are discussed in detail in Reference 2.5.4, NUREG/CR-3623 (SAND83-2622), and in Reference 2.5.5, NUREG/CR-4548 (SAND86-0494).

One test simplification employed by the program was the use of a single combination of materials (EPR for the insulating material and Hypalon for the jacket) and a single cable geometry (600 V, 20-A rated low-power cable); this cable type is typically used in nuclear power plants.

Systematic creep experiments were performed on cable samples that were bent in a U-shape over curved support rods, with curvature radii varying from 0.2 mm to 3.2 mm. In addition, each cable end was attached to a "stretching weight". The support rod diameters were much smaller than the usual mandrel diameters employed during cold bend tests (> 8 cable

diameters) so that the creep mechanism could be readily studied. Creep shortout was observed only at very high temperature (> 175°C) in combination with high stress (> 500 psi). It was found that some severe creep deformation occurred at temperatures of approximately 125°C.

An example of test results is shown in Figure 2.5.5. Cables were wrapped around a curvature of radius of 2.4 mm and stretched by 1-lb weights. The percentage of the reduction of thickness between cable conductor and support is plotted versus time, with exposure temperature as a parameter. The theoretical curve illustrates a uniform reduction in insulation thickness to 1% of its original value after 40 years. The

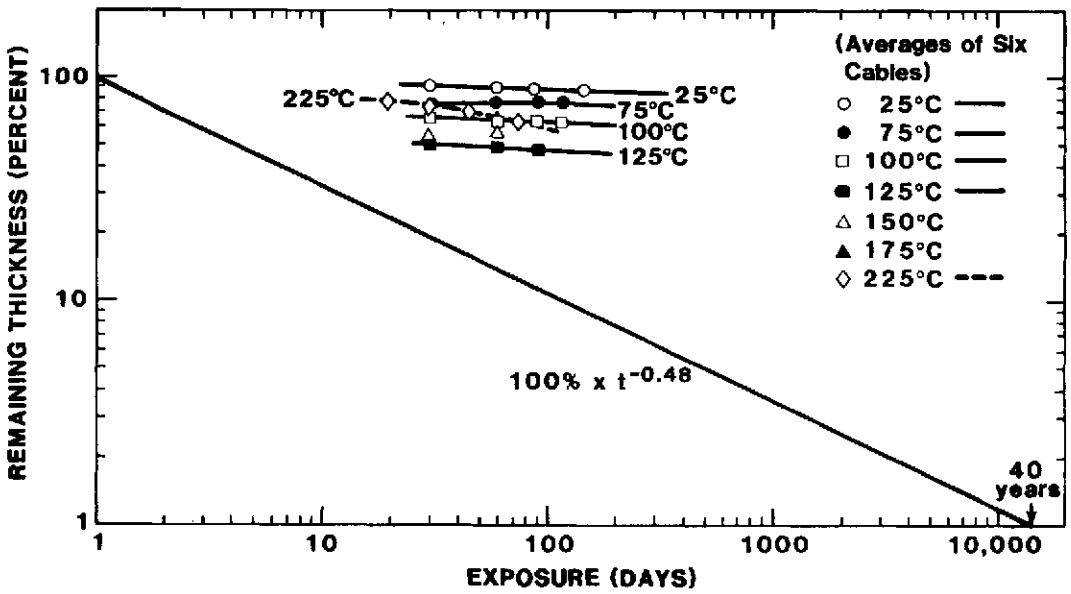

Figure 2.5.5 Average reduction for 6 cables. (From Reference 2.5.4)

measured data tend to move away from the theoretical curve; hence, no shortout is expected. (For 2-lb loads, however, this was no longer true.) It appears that the measured data reaches a minimum around 125°C; it is believed that the reason for this (at the higher temperatures) is because the rate of material hardening is increasing more than the softening rate.

For cracking assessment, three specific situations were considered:

a. Long cables (10m or more), lying undisturbed in conduits, may develop cracks under thermal cycling. For example, "through" cracks appeared in cables at 200°C after five days.

b. Cables that are sharply bent may develop cracks under mechanical stresses caused by thermal cycling or by weighting. Severe cracking tends to occur at temperatures above 175°C after several months.

c. Cables embrittled by aging may be stressed and cracked during maintenance activities. A simple model links the onset of cracks to the bending radius and the breaking strain at elongation (e/e_o) for the cable materials.

An example for case c above is presented in Figure 2.5.6. Cables were bent over a 4.8 mm diameter mandrel and thermally aged. Based on the simple model, cracks should not develop until degraded e/e_o values of

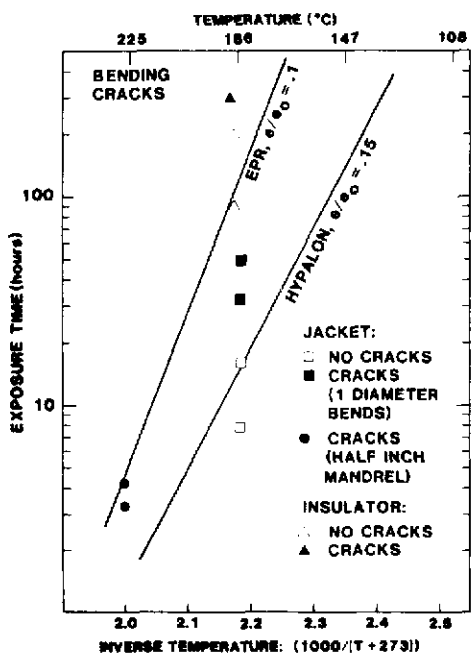

Figure 2.5.6 Crack occurrence versus temperature. (From Reference 2.5.4)

0.1 for EPR and 0.15 for Hypalon are reached. In Figure 2.5.6, theoretical estimates of the minimum time-temperature combinations to achieve these values are presented as straight lines. Combinations of time and temperature for which cracks were experimentally demonstrated are also illustrated. Results are consistent with the theoretical model limit for the onset of cracking.

The study made the following conclusions:

1. Above 175°C, cracks will occur even in straight cables, undisturbed in conduits except for thermal expansion cycles.

2. If cables hang over edges with small radius of curvature, and the overhang exceeds 15 m, creep shortout to the supporting edge may occur.

3. Cold handling of aged cables will promote cracks in the insulating layers.

Complex Equipment

Seals are generally used in many types of "complex" equipment to protect contaminant-sensitive components from the effects of the external environment. Reference 2.5.6 (RS4445/81/03) summarizes an equipment qualification test performed on a D. G. O'Brien Type K instrument penetration assembly.

This assembly consists of three major components, flange-mounted penetration modules, inner connectors -- mounted in containment and outer connectors -- mounted outside containment. The inner and outer connectors are enclosed in steel junction boxes. This penetration assembly has 16 flange-mounted modules and 16 each inner and outer connectors. Eight modules each have 14 pins (conductors) and 8 modules have 12 pins. Figure 2.5.7 shows the individual components which make up a typical module assembly. The two Sylgard® 170 components are intended to provide a water-tight seal -- thus protecting the conductor pins and avoiding electrical shorts -- during accident (LOCA/HELB) conditions.

As part of the test sequence, the entire penetration assembly was preconditioned at 150°C for seven days. During the postthermal aging inspection it was discovered that all module assembly coupling rings were loose; each coupling ring had been torqued to 25 foot-pounds prior to thermal aging. The report concluded that, due to the combined effects of the material's high coefficient of thermal expansion and the elevated temperature employed during aging, the Sylgard material extruded through the cable feed-through openings in the connector plug sleeve (i.e., mechanical stress relief). Consequently, the moisture seal was lost thus exposing the conductors to moisture during the steam test.

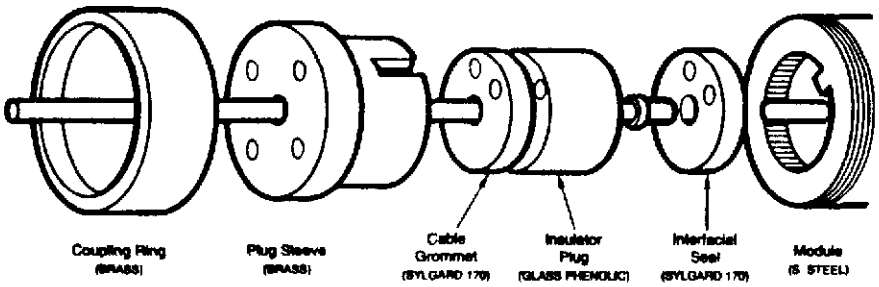

Figure 2.5.7 Connector-module assembly. (From Reference 2.5.6)

Reference 2.5.7, NUREG/CR-4147 (SAND85-0209), describes a series of tests performed to further study the mechanical stress relief phenomenon exhibited by the Sylgard 170 components. It notes that Sylgard 170's response to accelerated single stress thermal aging is not particularly age dependent. However, the elastomer response to an applied compressive force was strongly dependent on the environmental temperature and the degree of material confinement. Sylgard disks, similar to those used in the penetration assembly modules, were subjected to compressive stress (initial loading was 2000 lb_f) and aged at temperatures of 150°C, 43°C, and room temperature. The decrease in compressive stress was monitored as a function of time, and material extrusion was observed in all cases. Figure 2.5.8 shows the change in compressive load versus time for the three aging conditions.

One finding of this study was that the compressive force on the Sylgard disk would decay to about 50 lb_f in approximately 150 days at an aging temperature of 150°C, 610 days at 43°C, and 825 days at room temperature (approximately 27°C).

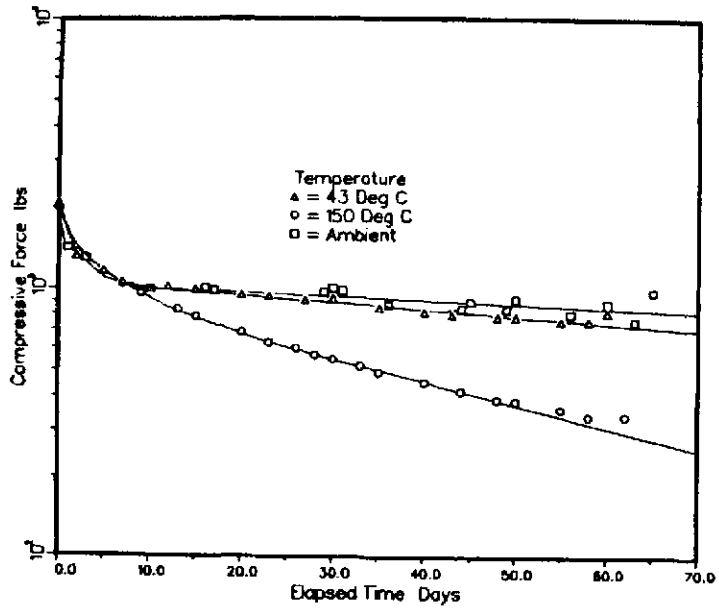

Figure 2.5.8 Comparison of Sylgard 170 mechanical stress relief during aging at 150°C, 43°C, and room temperature. (From Reference 2.5.7)

International and Other U.S. Programs

We know of two on-going programs which are studying the effects of mechanical stress on seal and gasket materials in radiation environments. The first is being conducted at the United Kingdom Atomic Energy Authority, Harwell by S. G. Burnay and J. W. Hitchon (Reference 2.5.8). This study is attempting to compare various techniques in their ability to predict seal lifetimes. Measurements are being made on various seals of leakage rates, sealing force and compression set over a limited range of temperature and irradiation conditions. Compression set measurement results are being empirically shifted using superposition concepts in order to make lifetime predictions.

A second program which has been going on for several years is being conducted by M. Ito of the Japanese Atomic Energy Research Institute in Takasaki. Using stress-relaxation results, a substantial amount of data has been generated for a number of materials in combined radiation and thermal aging environments (References 2.5.9 - 2.5.11). The data has been analyzed using mostly empirical approaches.

Finally, Reference 2.5.12, NUREG/CR-3424, suggests that mechanical stress effects may have been important towards the degradation of solenoid valves during an equipment qualification research program. It notes that the plausible principal causes for functional failure during testing included compressive set of seals and seats caused by gamma irradiation, and excessive damage of the organic seats and seals caused by cycling at the thermal aging temperature.

Issue Findings

By performing aging tests, with mechanical stress as a parameter, the significance of mechanical stress as an additive or synergistic degradation mechanism can be assessed.

Seals and Gaskets

Research efforts suggest that seals and gaskets are sensitive to environmental stress (e.g., temperature, radiation, humidity). The effect of mechanical stress loading in combination with these other aging stresses should be addressed.

In summary, the findings within this subissue are:

1. The most widely-used method for screening materials for seal and gasket applications is the compression set technique.

2. Diffusion-limited oxidation effects may cause compression set results to be non-conservative.

3. Small cracks at a polymer surface, which would not propagate under compression, may propagate under tension. Such effects could be significant for materials which have a tensile component in their mechanically-stressed state.

4. Stress-relaxation measurements may be more conservative than compression set measurements.

Electric Cables

Electrical cable failures (short circuits, cross-currents) can be caused by mechanical damage. Stresses may affect both material properties and geometric configurations of cables.

In summary, the findings within this subissue are:

1. Cracks will probably not occur in undisturbed cable insulation at temperatures below 150°C.

2. Creep shortout may occur in cables hung over edges with very small radius of curvature, much sharper than permitted in codes and standards.

3. Handling of aged cables may promote cracks in the insulating material.

Complex Equipment

Seals are often used in complex equipment to provide vapor/moisture barriers. Failure of the seal can lead to equipment failure.

In summary, the findings within this subissue are:

1. The combined effects of Sylgard 170's high coefficient of thermal expansion and an elevated temperature, caused material extrusion (through mechanical stress relief) and an observed loss of the moisture seal.

2. Sylgard was observed to undergo stress relief until the applied compressive force had decayed to background values.

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2.6 Oxygen Effects

Issue Description

A large percentage of the literature studies devoted to aging of polymers have concentrated on aging in an inert atmosphere, either nitrogen, argon or in a vacuum. A much more limited effort has been devoted to aging studies conducted in the presence of oxygen or air (~20% oxygen). There are a number of reasons for this situation, the main one being that inert atmosphere aging is already fairly complex for most materials and the addition of oxygen greatly increases the observed complexities. In addition, most aging studies concentrate on pure polymers, thus avoiding the added complications caused by the many fillers and additives present in commercial materials. Unfortunately, aging problems are of most practical concern for commercial materials exposed to air (oxygen-containing) environments. Moreover, of the limited number of studies conducted in air, many did not address (1) the possibility that the oxygen inside a small closed vessel could be used up in the course of the experiment and/or (2) the probability that at extremely high degradation rates characteristic of highly accelerated tests, oxygen diffusion effects can be so predominant that oxidation occurs only at the surfaces, resulting in "aged" samples for which the bulk of the material sees only anaerobic conditions.

The effects of oxygen on aging can be conveniently divided into two broad classifications-physical effects and chemical effects. Physical effects are caused by oxygen diffusion-limited degradation, a mechanism which has been observed for many years in various air-aging environments including heat and radiation. It occurs when the oxidation processes in a material use up dissolved oxygen faster than it can be replenished from the atmosphere surrounding the material by diffusion processes. This leads to a more heavily oxidized material near the sample surfaces and reduced or depleted oxidation in the sample interior, in other words a heterogeneously oxidized sample. Accelerated aging conditions will often result in heterogeneously oxidized samples, whereas the long times appropriate to real-time aging will typically allow sufficient time for diffusion to occur and therefore lead to homogeneous oxidation. Chemical effects refer to the multitude of new chemical reactions involving oxygen and oxidation products which will occur when oxygen is present in the material. Thus for simulating the aging of polymeric materials exposed to nuclear power plant environments, one must be concerned for both heat and radiation environments.

The requirement to consider oxygen effects during precondition aging for equipment qualification is only indirectly addressed in 10 CFR 50.49 and Regulatory 1.89. The applicable statement in 10 CFR 50.49, Section e.5, is "Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment."

Section 5.a of Regulatory Guide 1.89 states, "If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program."

This is the thrust of this effort to address the issue:

"What is the effect of oxygen in ambient and accelerated aging?"

Program Effort

The issue of oxygen effects in radiation environments is closely related to Issue 2.3 which is concerned with the importance and simulation of dose-rate effects. Because, as will become clear below, the presence of oxygen is a necessary requirement for the observation of dose-rate effects. In heat aging environments, the issue of oxygen effects is related to the question of the validity of the Arrhenius heat aging methodology, Issue 2.2. In both environments, heterogeneous oxidation in accelerated aging simulations caused by the physical diffusion mechanism will greatly complicate any attempts to simulate real-time aging. This potential concern was recognized in IEEE 323-1974, a document which recommended basic procedures for qualifying Class 1E equipment. When discussing radiation aging, it explicitly stated that "consideration shall be given to oxidation gas-diffusion effects."

The first effort, Reference 2.6.1, NUREG-0237 (SAND76-0715), discussed Class 1E equipment used in nuclear power plants, typical environments encountered by this equipment, and aging techniques used to qualify the equipment. Extended reviews of radiation chemistry and aging techniques were also included. The radiation aging section highlighted the various complications which can occur, especially for commercial (filled) polymeric materials in oxygen-containing environments. Based on these complications and the inadequacies of the existing aging techniques for Class 1E equipment, a proposal for an experimental program was presented. Among the main purposes of the proposed work was to obtain relevant data in two areas of particular concern--the effect of radiation dose rate on polymer degradation and the importance of synergism for combined thermal and radiation environments. Since oxidation effects were expected to dominate material degradation in air environments, both of these areas of concern involved oxygen effects.

In order to carry out the goals of this experimental program, an extremely versatile dedicated radiation-aging facility was constructed (see Issue 2.3 for a detailed description). This facility allowed us to carry out long term aging experiments in selectable combinations of radiation dose rate, temperature and surrounding atmosphere. To allow us to carry out thermal aging studies in either flowing air or flowing nitrogen environments, a heat aging facility was constructed as described in Reference 2.6.2, NUREG/CR-2877 (SAND81-2613).

After sufficient data at various dose rates had been collected for numerous polymeric materials, it became evident that radiation dose-rate effects are a very common occurrence. The dose-rate effects can range from very large to quite subtle, depending upon such factors as polymer type, aging conditions, sample geometry and degradation parameter monitored. Some examples from References 2.6.3, NUREG/CR-2156

(SAND80-2149) and 2.6.4, NUREG/CR-2157 (SAND80-1796) are shown in Figures 2.6.1-2.6.3. The tensile elongation results for the PVC material (Figure 2.6.1) are an example of very large dose-rate effects, whereas the EPR material in Figure 2.6.2 shows moderate dose-rate effects. Figure 2.6.3 gives reduced tensile strength and reduced elongation for a cross-linked polyolefin material. The tensile strength results indicate the existence of important dose-rate effects; the elongation results are more subtle and quite complex. As shown for these examples and in general, dose-rate effects usually indicate more mechanical degradation occurs for a given total dose as the dose rate is lowered. Therefore understanding the mechanisms responsible for these effects was important to developing aging methodologies. Comparisons of aging results in air versus nitrogen environments immediately implicated oxidation processes as critically important to these dose-rate effects, as is evident in Figures 2.6.2 and 2.6.3. In fact, no evidence exists for important dose-rate effects in a nitrogen environment. For these materials in the presence of oxygen, oxidative scission occurs and these scission processes become more important as the dose-rate is lowered. Reference 2.6.4 described results from a number of techniques which allowed us to reach this conclusion.

Data from one of these techniques, infrared spectroscopy, is shown in Figure 2.6.4 for EPR samples from Figure 2.6.2. The carbonyl peak which appears for the aged samples at ~1700 cm^{-1} is unambiguous evidence of oxidation and the larger peak at the lower dose rate is consistent with increased oxidation.

Experiments on the PVC material of Figure 2.6.1 and on a polyethylene material that also had large dose-rate effects showed that significant synergistic effects in combined radiation plus elevated temperature environments occurred (Figures 2.6.5 and 2.6.6). By utilizing the capability of our radiation-aging facility to control the surrounding atmosphere we were able to show that these synergistic effects completely disappeared in the absence of oxygen (Figures 2.6.7 and 2.6.8). Thus the presence of oxygen can lead to important synergistic effects.

This polyethylene material was identical to material sections of which were found to be mechanically degraded after removal from the Savannah River K nuclear plant (Reference 2.6.2). The degraded sections had seen an estimated 12 years of operating conditions equal to approximately 25 rad/h and 43°C, which gives a moderate total dose of approximately 2 Mrad. Infrared spectra of the degraded sections of material aged in the reactor showed strong carbonyl peaks which were absent in the undegraded material (Figure 2.6.9). Thus strong oxidation also occurs under ambient aging conditions.

The presence of oxygen can also have a tremendous influence on sequential aging exposures as shown in Table 2.6.1 from Reference 2.6.3 and on LOCA simulations with or without radiation preaging as described in Reference 2.6.5, NUREG/CR-2763 (SAND82-1071).

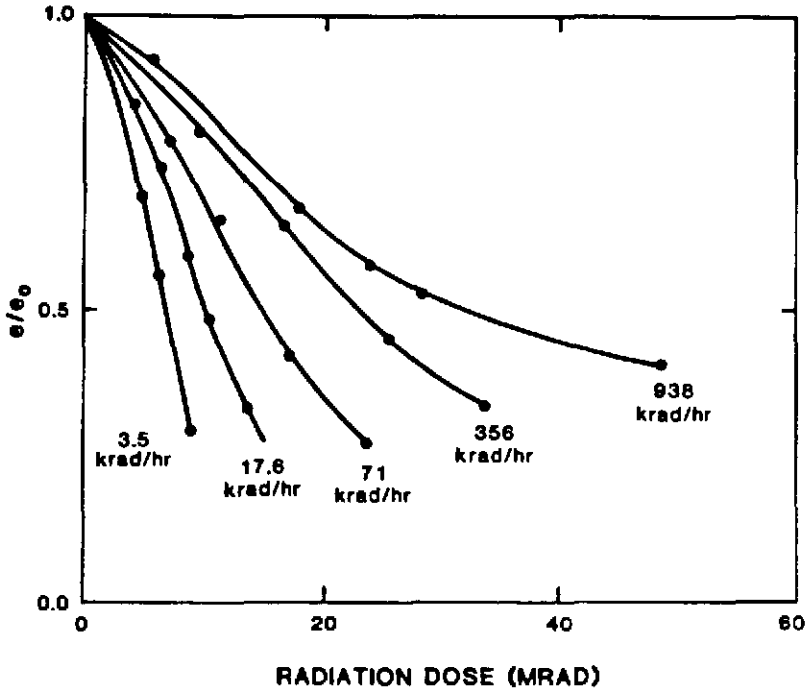

Figure 2.6.1 Aging of PVC at 60°C at a series of radiation dose rates. Sample tensile elongation divided by initial (unaged) elongation is plotted versus total absorbed radiation dosage. (From Reference 2.6.3)

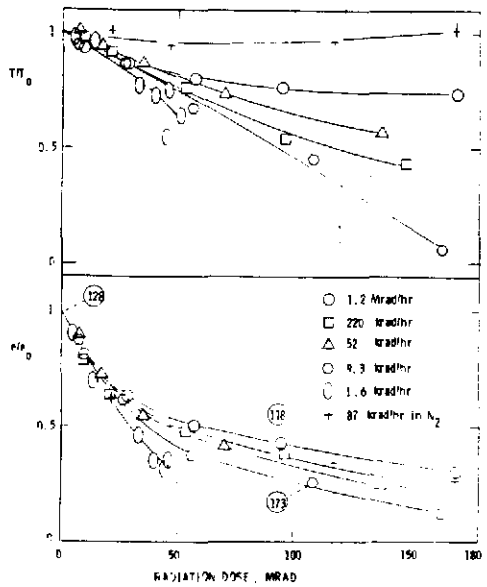

Figure 2.6.2 Aging of ethylene propylene rubber insulation. Explanation of figure is identical to that in Figure 2.6.3. (From Reference 2.6.4)

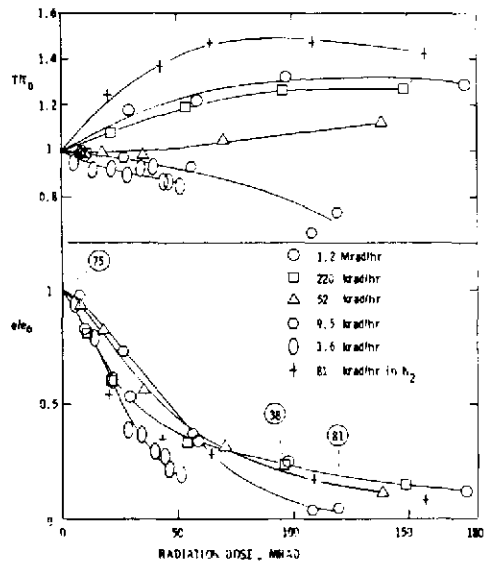

Figure 2.6.3 Aging of cross-linked polyolefin insulation. The tensile strength after aging divided by the tensile strength before aging (T/T_0) and the tensile elongation after aging divided by the tensile elongation before aging (e/e_0) plotted against the total integrated radiation dose at the various indicated dose rates. The circled numbers refer to the weight swelling ratios corresponding to the indicated experimental conditions. (From Reference 2.6.4)

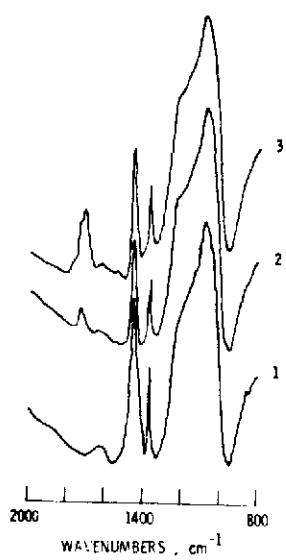

Figure 2.6.4 Infrared spectra of ethylene propylene rubber insulation. Curve 1 - unaged material. Curve 2 - aged in air at 1.2 Mrad/hr. Curve 3 - aged in air at 9.3 krad/hr. (From Reference 2.6.4)

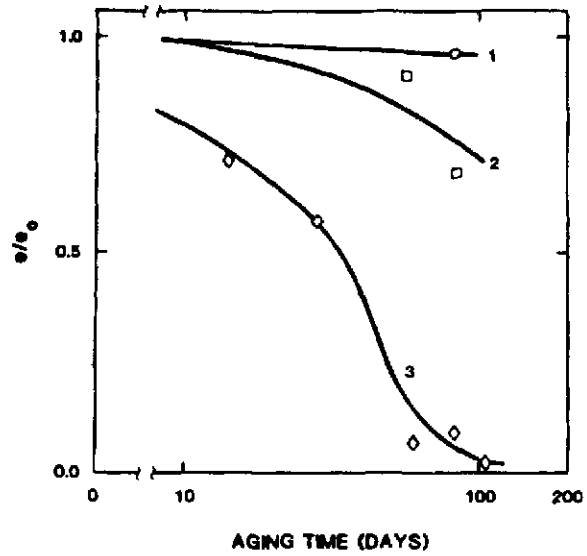

Figure 2.6.5 Tensile elongation results for PE insulation as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 5 krad/hr at 25°C; 3) radiation at elevated temperature, 5 krad/hr at 80°C. (From Reference 2.6.2)

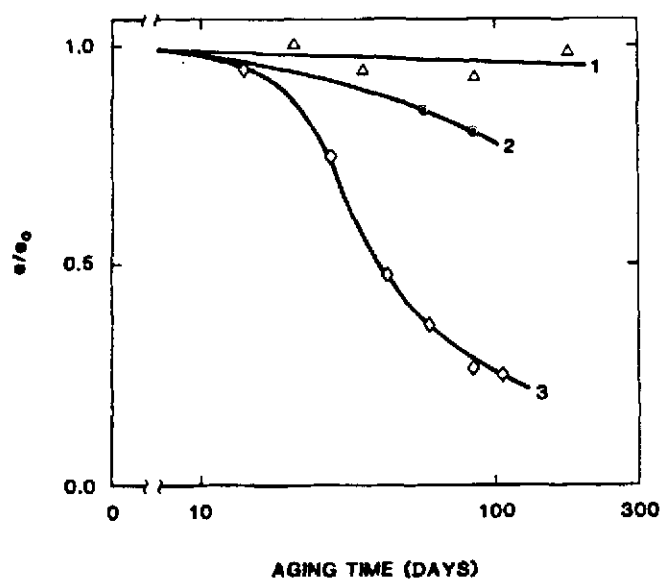

Figure 2.6.6 Tensile elongation results for PVC jacketing material as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 4.4 krad/hr at 25°C; 3) radiation at elevated temperature, 4.4 krad/hr at 80°C. (From Reference 2.6.2)

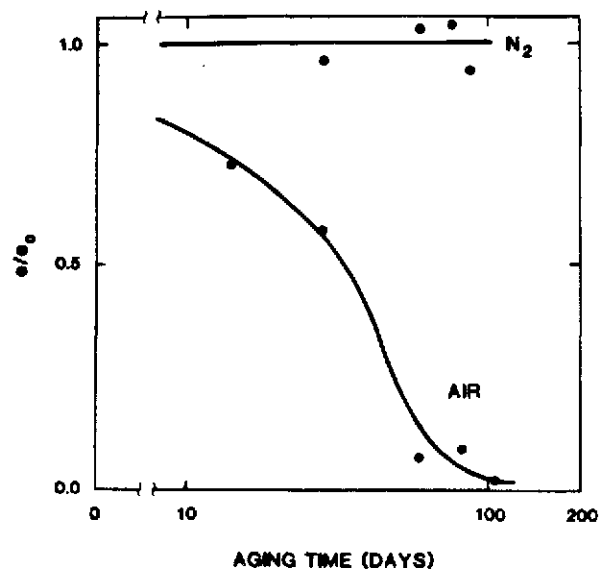

Figure 2.6.7 Tensile elongation results for PE insulation as a function of aging time at 5 krad/hr, 80°C in atmospheres of air and of N_2. (From Reference 2.6.2)

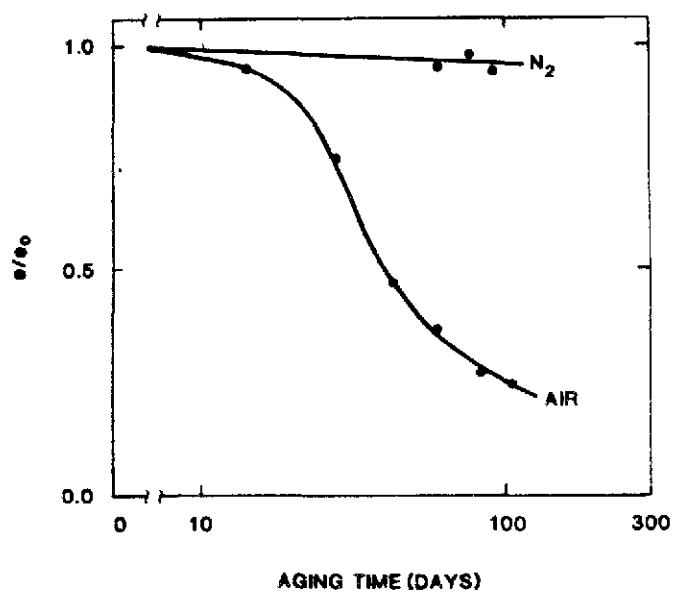

Figure 2.6.8 Tensile elongation results for PVC jacketing as a function of aging time at 4.4 krad/hr, 80°C in atmosphere of air and of N_2. (From Reference 2.6.2)

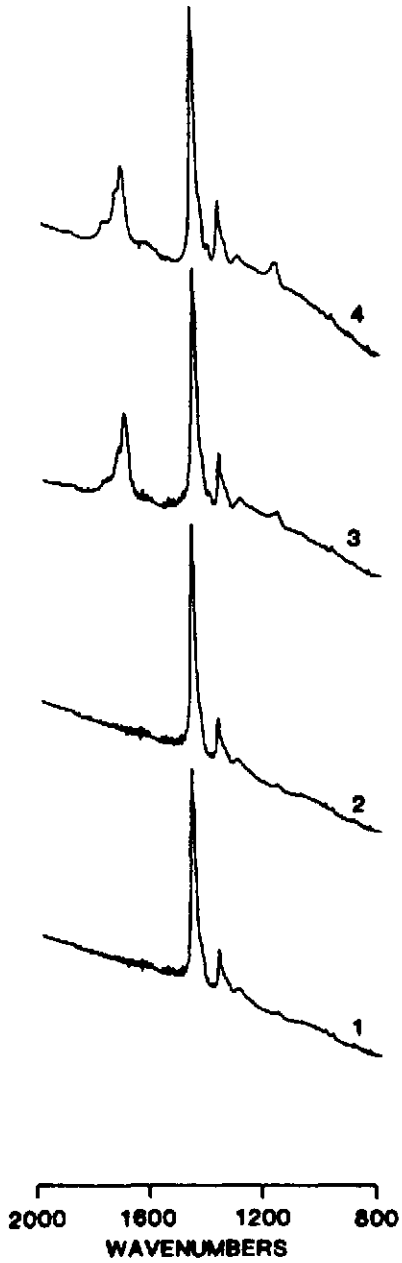

Figure 2.6.9 Fourier Transform Infrared Spectra of PE Insulation. Curve 1: Unaged laboratory sample. Curve 2: Sample removed from containment position H which exhibited no apparent deterioration. Curve 3: Sample removed from containment position F which exhibited substantial embrittlement. Curve 4: Laboratory aged sample (42 krad/hr, at 90°C to 17.9 Mrad total dose). Note the presence of the carbonyl bands near 1700 cm^{-1} for the aged samples. (From Reference 2.6.2)

Table 2.6.1 Sequential Aging Experiments: Tensile Elongation Data (From Reference 2.6.3)

EXPERIMENT*	e/e_o** PVC	e/e_o** PE
Unaged Material	1.0 ± 0.05	1.0 ± 0.1
Y; no subsequent T	0.80 ± 0.04	0.68 ± 0.09
T (in air); Y (in air)	0.68 ± 0.04	0.72 ± 0.07
Y (in air); T (in air)	0.32 ± 0.02	0.17 ± 0.04
Y (in N_2); T (in air)	1.02 ± 0.05	1.01 ± 0.1
Y (in air); T (in N_2)	0.83 ± 0.04	0.81 ± 0.08

* Y = gamma radiation - 4.5 krad/hr (for PE), 4 krad/hr (for PVC) - both at 25°C for 83 days
T = thermal exposure of 80°C for 83 days
** Sample tensile elongation divided by initial (unaged) elongation.
e/e_o = 310% for PVC: e_o = 540% for PE

Reference 2.6.6 (SAND83-0798J) is a review article that summarizes examples, from Sandia's program and from other researchers throughout the world, of the evidence that the presence of oxygen during radiation aging can lead to complicated dose-rate and synergistic effects.

Reference 2.6.7, NUREG/CR-3643 (SAND83-2493), gives an example of the effect of oxygen on polymer degradation which occurs with a Viton material. Here the material degrades in essentially opposite ways at high and low dose rate, becoming hard and brittle at high dose rate, but soft and stretchy at low dose rate. This trend is reflected in the dramatic differences seen in ultimate elongation and tensile strength changes seen at low and high dose rates (Figure 2.6.10). Note that the data shown in the figure for aging under inert atmosphere is shifted even beyond that of the high dose rate data. The reason behind the dramatic effects of dose rate is large differences in oxygen permeation depth at the dose rates employed. Table 2.6.2 gives oxidation depth as a function of dose rate as determined by cross-sectional polishing (see Issue 2.8). Further illustration of this strong oxygen effect is shown in Figure 2.6.11, which illustrates that under dose rates where the oxygen permeates throughout the material, the Viton undergoes net chain scission as it degrades. In contrast, under high dose rate irradiation, where oxygen can permeate only a short distance into the sample, the Viton undergoes net chain cross-linking as it degrades.

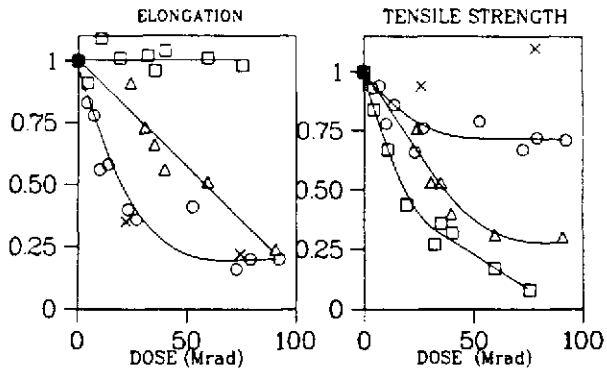

Figure 2.6.10 Change in ultimate tensile properties for a Viton material irradiated at 70°C using three different dose rates: ( ) 5.5×10^5 rad/h (in air), (Δ) 9.2×10^4 rad/h (in air), (□) 1.3×10^4 rad/h (in air), (X) 5.5×10^5 rad/h (in inert atmosphere). Left: reduced elongation (e/e_o), Right: reduced tensile strength (T/T_o). From Reference 2.6.7

Table 2.6.2

Oxidation Depth as Determined by Cross-Sectional Polishing of Viton Material Irradiated at 70°C, Using Different Dose Rates[a] (From Reference 2.6.7)

Dose rate (rad/h)	Oxidation depth[b]	Oxidation as % of total thickness
5.5×10^5	0.33 mm	35
1.8×10^5	0.42 mm	44
9.2×10^4	0.78 mm	82
1.3×10^4	complete	100

[a] Ring size was relatively dose-independent.

[b] Sample thickness = 1.91 mm.

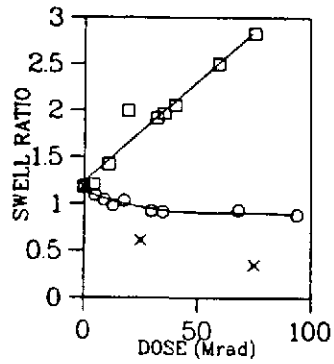

Figure 2.6.11 Solvent swelling data for the Viton material of Figure 2.6.10 (symbols have same meaning). Solvent was THF. Data calculated as: (Swelled weight minus dry weight) divided by dry weight. Increased swelling indicates chain scission; decreased swelling indicates cross-linking. (From Reference 2.6.7)

Reference 2.6.7 also calculated the total dose required to cause reaction of all the dissolved oxygen in various polymers. Table 2.6.3 indicates that total doses less than 0.1 Mrad would deplete all the initially dissolved oxygen in the interior of the polymers. Hence, unless the oxygen was replenished via diffusion at a rate faster than it was consumed by radiation, for the remainder of the aging irradiation the interior of the polymer would be starved of oxygen and oxidative degradation would be reduced. Reference 2.6.7 also summarized data on the depth of the oxidized regions for a collection of commercial polymer materials irradiated at several different dose rates and temperatures.

Reference 2.6.8, NUREG/CR-4008 (SAND84-1948), demonstrated that experimental profiling techniques (addressed in Issues 2.3 and 2.8) as well as theoretical calculations may be employed to estimate experimental conditions (i.e., polymer dimensions, oxygen pressure, dose rate, etc.) for which "homogeneous" oxidation aging will occur.

International and Other U.S. Programs

As mentioned earlier, numerous studies have been and are being conducted on oxidation effects on polymers. Far fewer studies are devoted to oxidation effects in radiation environments, especially for commercially

Table 2.6.3

Literature Values for Solubility and $G(-O_2)$ Yields, and Calculated Equivalent Dose Required to Cause Reaction of Dissolved Oxygen in Equilibrium With Air (.2 Atm O_2 pressure). (From Reference 2.6.7)

Polymer	S* (ref)	$G(-O_2)$ (ref)	Equivalent Dose (Mrad)
PE	2.2×10^{-6}		
		10 (23)	.04
	(18)	22 (24)	.02
PVC	7.5×10^{-6}	12 (25)	.15
		30 (25)	.05
	(22)	57 (26)	.03
EPR	5.4×10^{-6}	20 (27)	.05
		40 (27)	.03
	(22)	60 (27)	.02
CLPO	3.1×10^{-6} (22)	10 (28)	.06

* Expressed in mol/g•atm

useful materials. Reference 2.6.6 is a useful, recent review article which contains extensive references to relevant work.

Of particular interest is the effort being conducted at the Japanese Atomic Energy Research Institute (JAERI) in Takasaki, since its goals are close to those of the Sandia effort and its work is complementary. A substantial effort at JAERI is being directed at using a high pressure oxygen apparatus to study the effects of high oxygen pressures on radiation aging of polymers (Reference 2.6.9). The rationale for these studies is the anticipation that oxygen diffusion effects can be eliminated under higher dose-rate conditions for a given material and material geometry, thereby cutting the time necessary to simulate radiation aging. Figure 2.6.12 illustrates JAERI results for an EPR material irradiated under different oxygen pressure conditions. Elongation properties are similar for 1 Mrad/hr exposures with ambient air

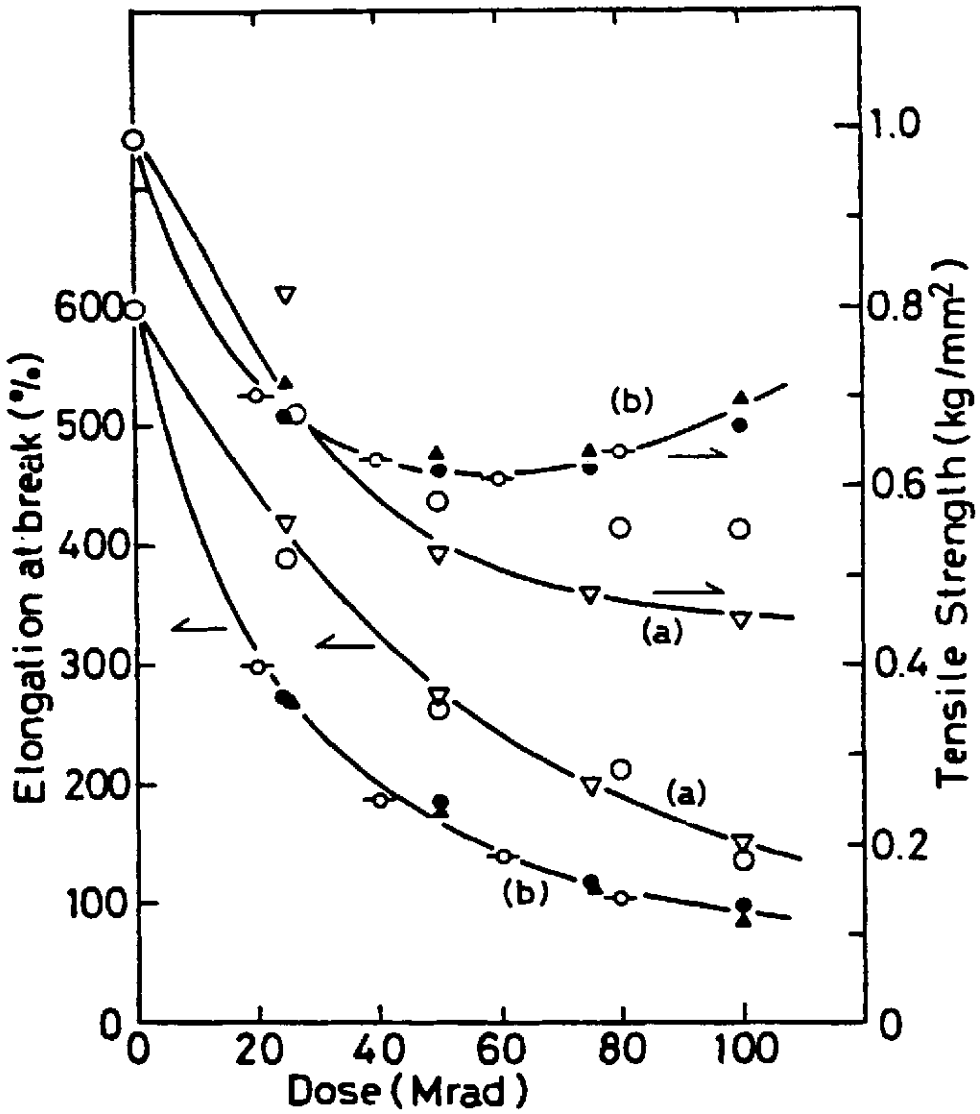

Figure 2.6.12 Elongation (E_b) and tensile strength (TS) at break against dose for formulated EPR irradiated in various conditions at room temperature. ▽ in vacuum with 1 Mrad/h; ○ in air with 1 Mrad/h; —○— in air with 5 krad/h; ● in O_2 5 atm with 0.5 Mrad/h; ▲ in O_2 5 atm with 0.1 Mrad/h. (From Reference 2.6.9)

or vacuum conditions. Degradation is more substantial for a 5 krad/hr exposure with ambient air pressures. By increasing the oxygen pressure to 5 atmospheres, 0.1 and 0.5 Mrad/hr exposures also produced the more severe degradation. Reference 2.6.9 concluded that degradation for this EPR material depended only on the thickness of the oxidation layer during irradiation. A second effort at JAERI is directed towards understanding the relationship between the thickness of the oxidation layer, the oxygen pressure during irradiation, and the dose rate (Reference 2.6.10). A third effort at JAERI is attempting to modify the additives used in polymers in order to eliminate any chemical dose-rate effects (Reference 2.6.10).

Issue Findings

The majority of this issue has been demonstrated since a preponderance of evidence indicates that the presence of oxygen can have large (often dominant) and complicated effects in both ambient and accelerated aging situations. The diffusion effects caused by oxygen were recognized as potentially significant to radiation aging simulations in IEEE 323-1974. The issue of critical importance is how to account for the effects of oxygen. This constitutes a major part of the question of how one simulates radiation dose-rate effects, discussed in detail in Issue 2.3.

By performing controlled oxygen and nitrogen accelerated-aging tests on identical test samples of exposed organic materials and equipment, the role of oxygen in the aging process can be evaluated.

In summary then, the findings within this issue are:

1. For the majority of common polymers the presence of oxygen dominates the degradation mechanism and degradation behaviors.

2. Oxidation processes will normally dominate ambient aging but, dependent upon the experimental conditions, these effects can range from insignificant to dominant during accelerated simulations.

3. Oxygen is required for dose-rate effects to occur.

4. Oxygen has a large influence on sequential aging experiments.

5. Oxygen can also lead to strong synergistic effects of radiation and elevated temperature.

6. The role of oxygen in degradation of a given material under given conditions can be evaluated by a number of chemical and spectroscopic techniques (see also Issues 2.3 and 2.8)

7. Because of the complex radiation results caused by the presence of oxygen, aging methodologies should be developed that consider these complexities. This problem is addressed in Issue 2.3, which

concerns the development of methods to simulate dose-rate effects. The discussion in Issue 2.3 describes in detail the mechanisms underlying dose-rate, synergistic and sequential ordering effects caused by the presence of oxygen and describes experimental techniques useful for monitoring and modelling such effects. Some of the experimental profiling techniques are extremely useful for estimating whether/when "homogeneous" oxidation aging will occur.

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Program Effort

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2.6.9 T. Seguchi, K. Arakawa, N. Hayakawa, and S. Machi, Radiat. Phys. Chem. 48, 671 (1981).

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2.7 Humidity Effects

Issue Description

There is literature available concerning the effects of humidity and water immersion on polymers in thermal aging environments. Numerous short- to intermediate-term laboratory studies plus a wealth of long-term data on field-aged samples have yielded lists of materials to avoid if high humidity is a potential concern. These results are invaluable aids to manufacturers when choosing materials to include in components used in elevated temperature, high humidity nuclear power plant environments. If, however, the manufacturer does not use this information or chooses to use a new material for which real-time, long-term data is nonexistent, concerns over methods to accelerate humidity aging emerge. For materials exposed to radiation as well as thermal and humidity stresses, a new concern occurs even for materials known to be resistant to humidity effects in thermal environments. This concern revolves around the possibility of synergistic effects in combined radiation-humidity environments.

The requirement to consider humidity during (accelerated) aging of equipment subjected to qualification testing is not specifically addressed in either 10CFR 50.49 or Regulatory Guide 1.89.

This is the thrust of this effort to address the issue:

"Does humidity affect aging, and how can it be accelerated?"

Program Effort

One of the first goals of the aging program was to screen for the possible presence of humidity effects for a wide variety of commercial cable insulation and jacketing materials. Since the initial screening tests showed no evidence of strong humidity effects, little further experimental effort was devoted to humidity effects. Reference 2.7.1, NUREG/CR-0813 (SAND79-0761), summarized extensive comparisons of radiation dose-rate experiments run under 0 and 70 percent relative humidity conditions. Figures 2.7.1-2.7.4 from this document show typical results for a chloroprene (neoprene) material, two chlorosulfonated polyethylene (hypalon) materials, a silicone and an ethylene propylene rubber. Even at the lowest dose rate, no evidence of humidity effects could be discerned. These results eliminated any major concerns about strong synergistic effects of radiation and humidity, implying that a material known to have good humidity resistance in thermal environments is probably humidity resistant in radiation environments as well.

The far more difficult task of developing a methodology for accelerating humidity aging effects was addressed in Reference 2.7.2, NUREG/CR-1466 (SAND79-1561). This paper outlined some of the types of experiments necessary to simulate age and predict life expectancy of complex equipment, concentrating on an approach based on kinetic rate

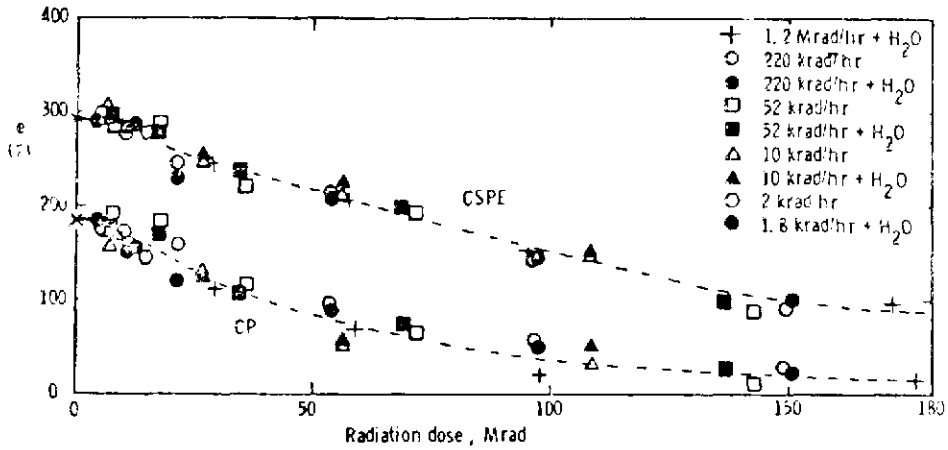

Figure 2.7.1 Ultimate tensile elongation (e) vs total radiation dose for chloroprene (CP) and chlorosulfonated polyethylene (CSPE) jacket materials at various dose rates under dry air and 70% relative humidity (H_2O) conditions as indicated in the figure. (From Reference 2.7.1)

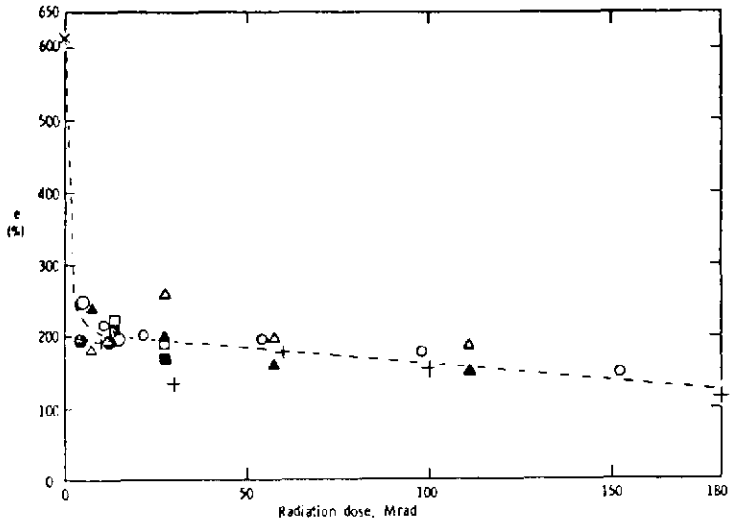

Figure 2.7.2 Ultimate tensile elongation (e) vs total radiation dose for chlorosulfonated polyethylene insulation material. See Figure 2.7.1 for meaning of symbols. (From Reference 2.7.1)

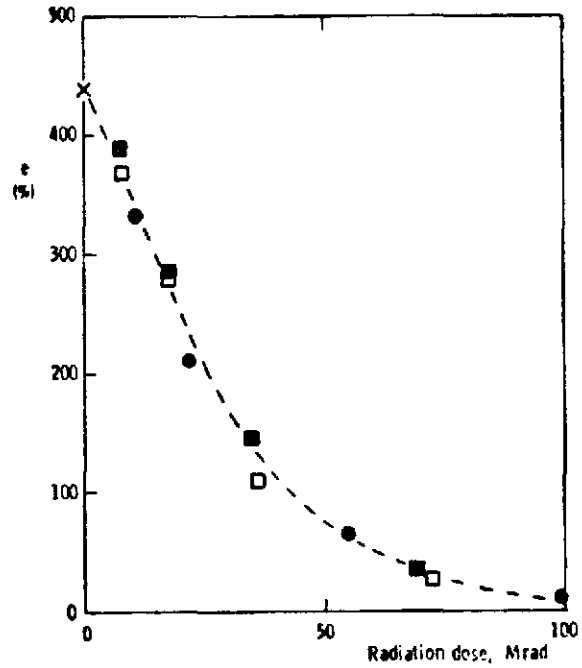

Figure 2.7.3 Elongation vs total radiation dose for silicone insulation. See Figure 2.7.1 for meaning of symbols. (From Reference 2.7.1)

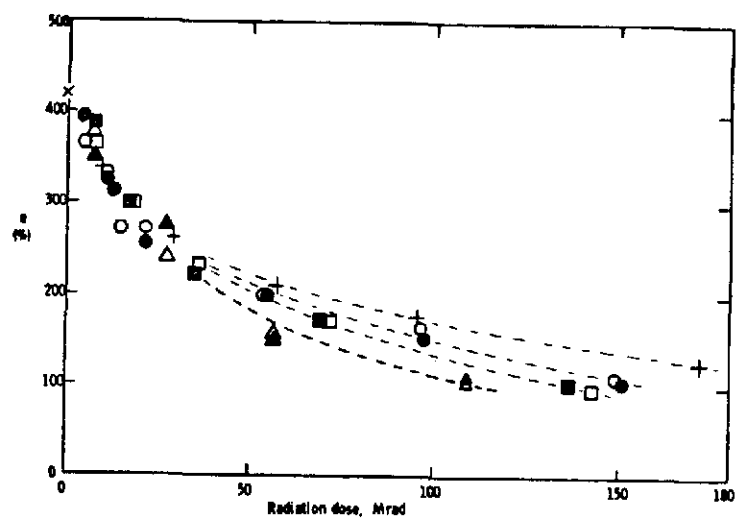

Figure 2.7.4 Ultimate tensile elongation (e) vs total radiation dose for ethylene propylene rubber insulation. See Figure 2.7.1 for meaning of symbols. (From Reference 2.7.1)

expressions. An innovative method for the acceleration of humidity aging was derived. Previous approaches for extrapolating accelerated humidity results in thermal environments to long-term ambient conditions involved totally empirical methods. The new approach was based on the observation that at a given constant relative pressure (relative humidity), the solubility coefficient of condensable gases have a small (and approximately Arrhenius) dependence on temperature (see Figures 2.7.5 and 2.7.6) even though the sorption isotherms are complex and very temperature dependent (see Figure 2.7.7). If E_c is defined as the activation energy for the solubility coefficient at constant relative humidity, the data of Figure 2.7.5 indicates that E_c is equal to zero for all values of relative humidity and the data of Figure 2.7.6 gives an E_c of 5 kcal/mole, also independent of humidity. These results are typical for water sorption in various materials. Using this observation together with a standard kinetic analysis allowed us to predict that degradation data analyzed under constant relative humidity conditions should be Arrhenius. Literature data on the reversion of a polyurethane potting compound was analyzed using this methodology; the results shown in Figure 2.7.8 offer excellent evidence for the validity of this approach.

Reference 2.7.3, NUREG/CR-3156 (SAND82-2559), identifies humidity as one (of several) environmental stress (aging mechanism) which can affect the performance of electronic components (e.g., semiconductors, capacitors, resistors) used in safety-related, Class 1E equipment. The authors recommend that an aging program, to investigate aging effects (including humidity) on electronic components, be undertaken.

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Reference 2.7.4 (SAND83-2165) examined moisture permeation into nuclear reactor pressure transmitters. Analytical solutions are presented for the humidity buildup in pressure transmitters of the ITT Barton and Foxboro types. Results indicate that 50% of the exterior humidity will be realized in six months and that the 90% level will be attained within a couple of years. Thus, even "closed" systems can be expected to achieve an equilibrium humidity condition.

Humidity and related water immersion studies on polymers and other materials in thermal environments are being conducted in hundreds of universities, industrial companies and research institutes throughout the world. Military specifications such as Mil-Std. 810 have been developed to identify equipment failures caused by combined environments that include humidity, temperature, and vibration. In radiation environments, however, we are not currently aware of any studies concentrating on humidity effects.

An example of a humidity study in thermal environments that is relevant to nuclear applications is described by Reference 2.7.5. This report discusses the hydrolytic degradation of polyimide (Kapton®) wire insulation which reduces the polymer chain length and renders the

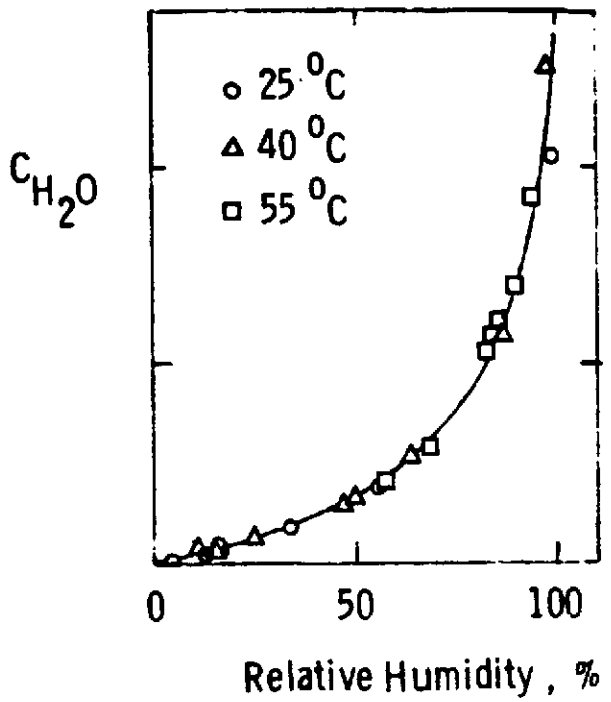

Figure 2.7.5 Sorption isotherms for water vapor in a polyurethane membrane, plotted vs relative humidity. The ordinate is the H_2O concentration in the polyurethane membrane. (From Reference 2.7.2)

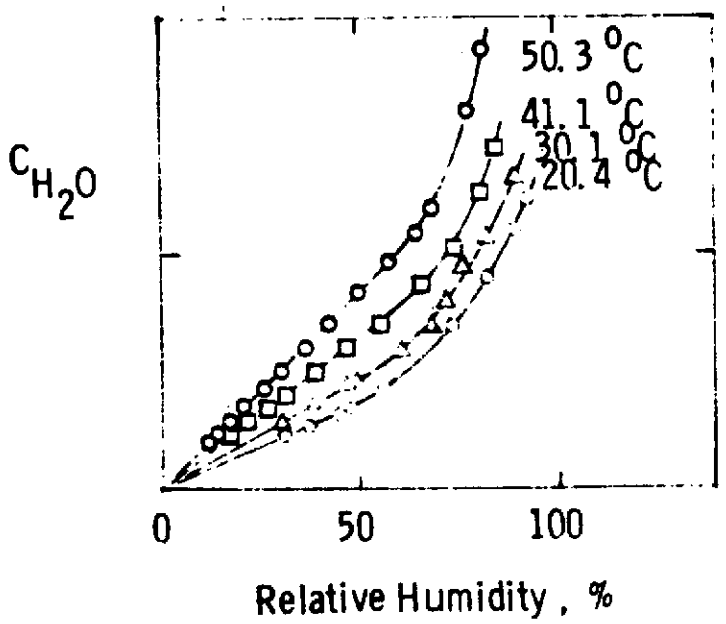

Figure 2.7.6 Sorption isotherms for water vapor in a silicone material, plotted vs relative humidity. The ordinate is the H_2O concentration in the silicone material. (From Reference 2.7.2)

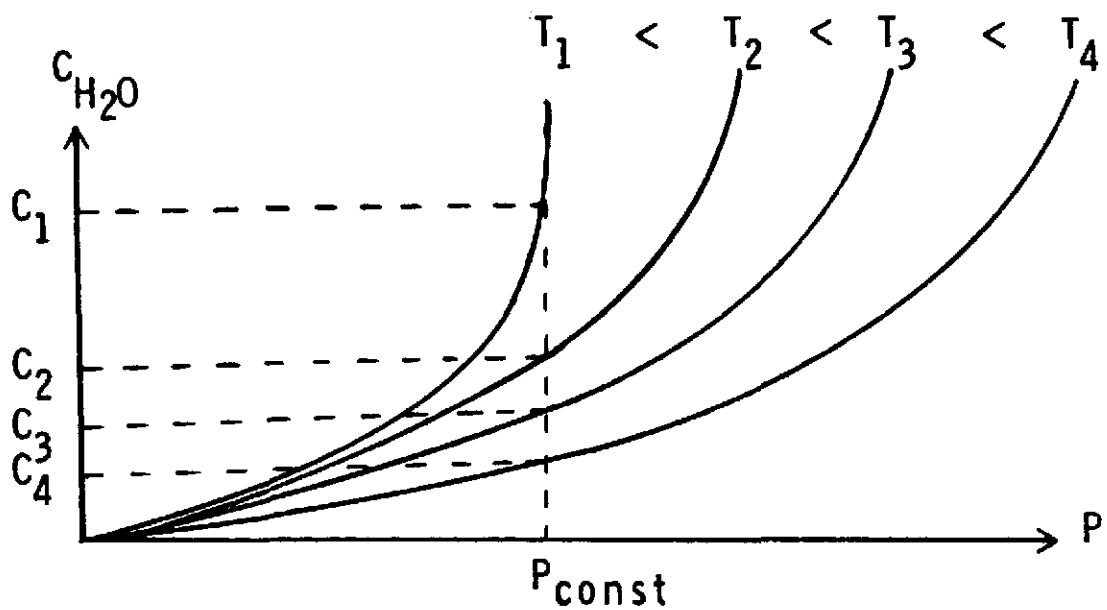

Figure 2.7.7 Typical isotherms for water vapor and other condensable fluids. (From Reference 2.7.2)

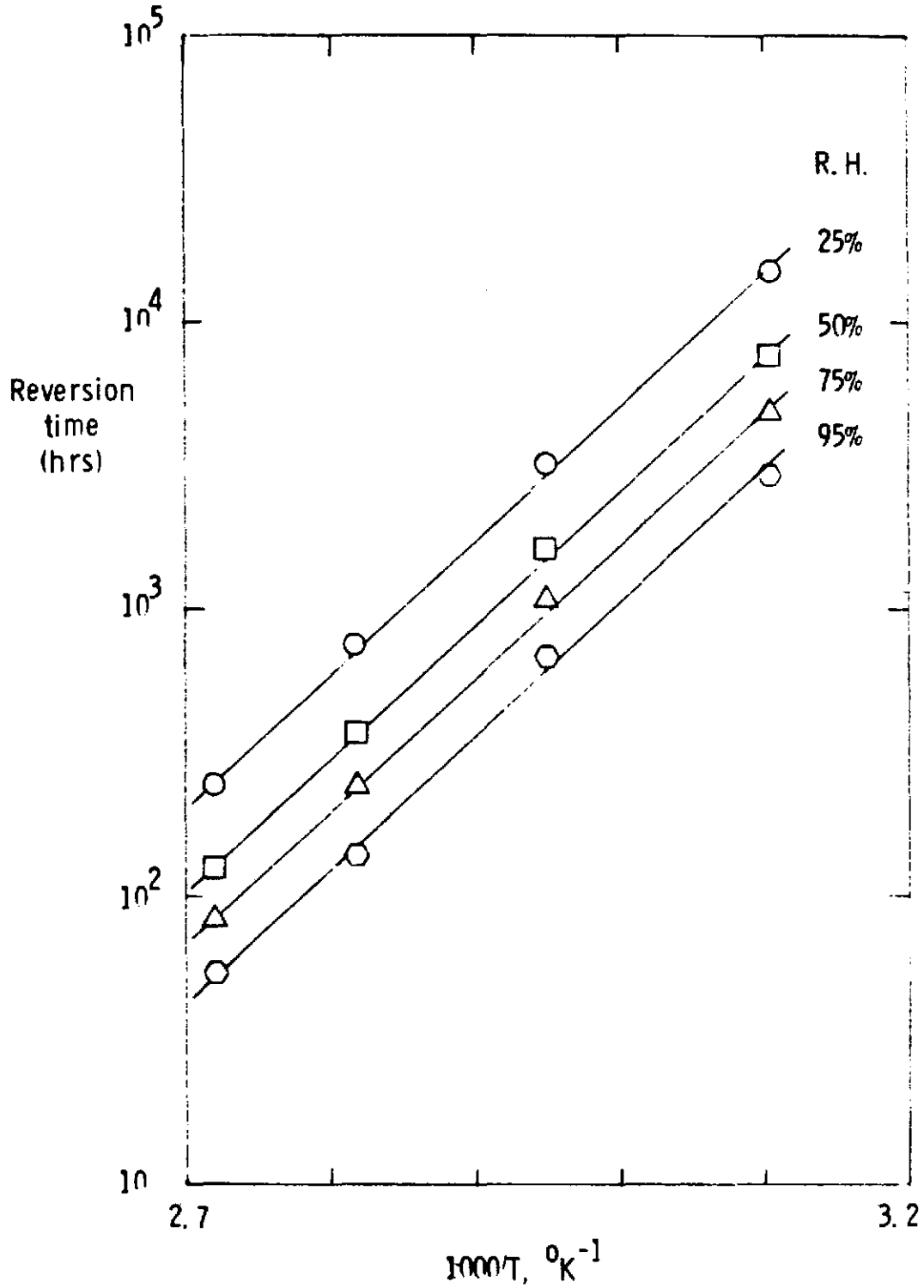

Figure 2.7.8 Arrhenius plot of data analyzed at various constant relative humidities. (From Reference 2.7.2)

initially strong material to a weak and brittle wire coating. The report noted that typical industry life-evaluation procedures are based on the recognition that degradation is the result of a chemical reaction such as oxidation that occurs at a rate that is temperature dependent (many times Arrhenius). This reaction then weakens the material's resistance to mechanical stress, moisture penetration, and electrical stress which ultimately produce system failure. For polyimide, the report noted that hydrolytic degradation due to moisture and high pH fluid exposures were the primary cause of wiring failures rather than oxidation degradation. Hence a study was initiated to determine whether humidity environmental conditions could be incorporated into standard life-evaluation test procedures. Initial experiments to determine whether Kapton's hydrolysis degradation can be described by an Arrhenius model are presented. An activation energy for hydrolysis degradation of 0.6 eV was determined.

Issue Findings

Humidity is one of several environmental stresses that can affect the performance of safety-related equipment. In this regard, Reference 2.7.1 concluded that a material having good humidity resistance in thermal environments is likely to be humidity resistant in radiation environments as well, and Reference 2.7.2 described a general (Arrhenius) method for extrapolating accelerated humidity aging effects (degradation) in thermal environments. Another humidity study (Reference 2.7.5) found that Kapton wire insulation is subject to hydrolytic degradation in humid-thermal environments and that such degradation can also be described by an Arrhenius model. However, Reference 2.7.4 indicated that humidity buildup can occur within "closed" systems (e.g., pressure transmitters) over time and eventually achieve equilibrium conditions.

In summary then, the findings within this issue are:

1. Humidity effects for polymeric materials are probably not of major importance in radiation environments as long as materials known to have humidity resistance in thermal environments are selected.

2. A general methodology for extrapolating accelerated humidity aging data in thermal environments to longer-term ambient conditions has been derived and verified.

3. Humidity screening tests would be useful prior to performing accelerated aging tests.

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Program Effort

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2.7.4 M. R. Keenan, Moisture Permeation into Nuclear Reactor Pressure Transmitters, SAND83-2165, Sandia National Laboratories, Albuquerque, New Mexico, March 1984.

2.7.5 F. J. Campbell, Temperature Dependence of Hydrolysis of Polyimide Wire Insulation, NRL Memorandum Report 5158, Naval Research Laboratory, Washington, D.C., August 4, 1983.

2.8 Analytical and Experimental Techniques for Correlation Between Natural and Artificial Aging

Issue Description

To correlate real-time and accelerated-aging results, it is necessary to develop complementary experimental and analytical techniques in an effort to select and prove the proposed accelerated-aging methods. Final "proof" of any proposed accelerated-aging method is its demonstration against naturally-aged, benchmark materials and equipment; this requires experimental techniques capable of correlating accelerated and natural aging conditions. Although there are countless experimental techniques which can be used to monitor the aging of materials, few have the requisite sensitivity to monitor the early stages of aging (often the only stages experimentally available for such naturally-aged samples) that ultimately lead to functional performance degradation. Thus the development of techniques sensitive to the earlier stages of material degradation is of critical importance.

Of the many techniques now used to monitor aging, some (e.g., mechanical property measurements) can be applied to quite a few materials, while others are only applicable to certain materials or under specialized environmental conditions. It is important to recognize the uses and limitations of existing techniques. This knowledge can then be utilized to develop new approaches/techniques capable of addressing such limitations.

This is the thrust of this effort to address the issue:

> "Are there analytical and experimental techniques which can be used to correlate real-time aging and accelerated aging (methods)?"

Polymeric Materials

There are numerous experimental techniques which can be used to follow aging of polymeric materials (see for example Reference 2.8.1). The selection of appropriate techniques depends upon the particular material being investigated and on the environments to which the material is exposed. During the chronology of the aging program, many of the available techniques have been screened for possible applicability and a number have given very useful information on aging and/or on the correlation of ambient and accelerated aging. Useful techniques included mechanical property measurements, infrared spectroscopy [Reference 2.8.2, NUREG/CR-2877 (SAND81-2613)], swelling and solubility measurements [References 2.8.3, NUREG/CR-2157 (SAND80-1796), 2.8.4, NUREG/CR-3643 (SAND83-2493) and 2.8.5, NUREG/CR-3151 (SAND82-1414)], emission spectroscopy (Reference 2.8.6, SAND82-1413), microprobe analysis (Reference 2.8.6, SAND82-1413J), oxygen uptake [Reference 2.8.7, NUREG/CR-4008 (SAND84-1984)] and chemiluminescence (Reference 2.8.7). Some representative data taken using infrared techniques is shown in Figure 2.8.1. These results offer excellent evidence that the oxidative

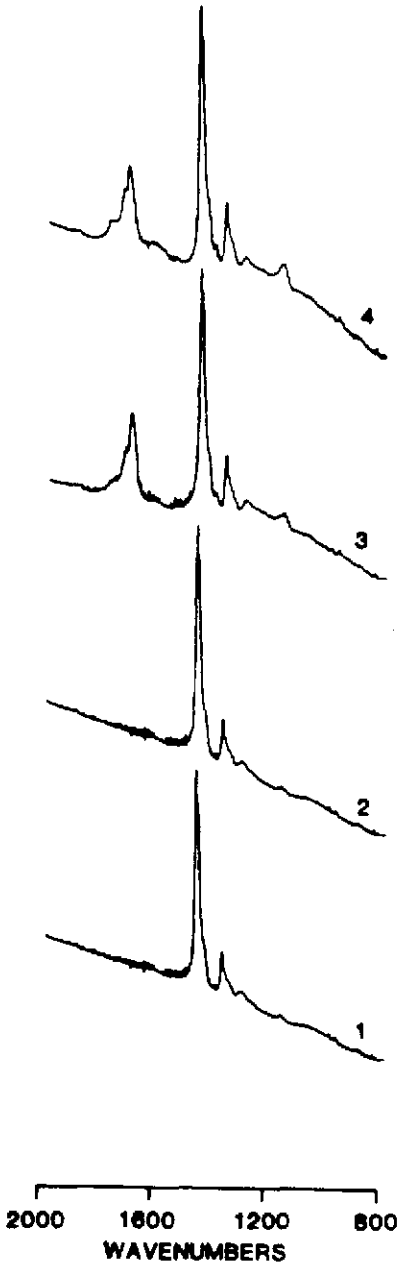

Figure 2.8.1 Fourier Transform Infrared Spectra of PE insulation. Curve 1: Unaged laboratory sample. Curve 2: Sample removed from containment position H of the Savannah River reactor which exhibited no apparent deterioration. Curve 3: Sample removed from containment position F which exhibited substantial embrittlement. Curve 4: Laboratory aged sample (42 krad/hr, at 90°C to 17.9 Mrad total dose). Note the presence of the carbonyl bands near 1700 cm -1 for the aged samples. (From Reference 2.8.2)

products generated in laboratory accelerated aging experiments are similar to those found in samples removed from a reactor environment.

One limitation is the scarcity of techniques sensitive to the earliest stages of degradation; such sensitivity is of paramount importance for monitoring real-time aging of long-lifetime materials. A second need is for techniques capable of monitoring heterogeneous oxidation across small polymer cross-sections, so that diffusion-limited oxidation complications (see Issues 2.2, 2.3 and 2.6) could be adequately addressed. A third need is for techniques applicable for commercial materials. Finally, some specialized chemical identification techniques were needed to help sort out the chemical steps underlying chemical dose-rate and synergistic effects (see Issues 2.3 and 2.4). This discussion will therefore concentrate on three main areas, the development of (1) profiling techniques, (2) chemical techniques and (3) techniques sensitive to the earlier stages of degradation.

Profiling Techniques

References 2.8.4, 2.8.6 and 2.8.7 discuss the causes and results of heterogeneous oxidative degradation due to oxygen diffusion which is an extremely important parameter in accelerated aging tests. Basically, at high dose rates, characteristic of accelerated tests, oxidation may proceed so rapidly that oxygen is used up faster than the rate at which it can diffuse into the material from the surrounding atmosphere. When this occurs, a heterogeneous degradation is the result; oxidation takes place only near the edges. This implies a severe restraint on the use of short-term, high-stress, accelerated tests and can result in many types of "dose-rate effects" when comparing accelerated aging data with ambient aging data.

Two new Sandia-developed techniques described in Reference 2.8.4 allow the determination of heterogeneous oxidation. The first is a rapid qualitative technique for identifying the depth of oxidation in materials which have strong oxidative heterogeneities. This technique involves mounting samples in epoxy, cutting them in cross-section, and polishing the cut surface using standard metallographic techniques. When examined under an optical microscope regions of markedly different oxidation exhibit a different degree of reflectivity. Figure 2.8.2 illustrates this effect.

A second technique, which is more quantitative and gives detailed information on the shapes of heterogeneities is relative hardness profiling. With this technique measurements of relative material hardness are made all across the cross-sectioned surface of a degraded sample. The measurements make use of a tiny, weighted probe; the depth of penetration into various positions of the cross-sectioned surface is measured. Figure 2.8.3 shows a relative hardness profile of a Viton material irradiated at an intermediate dose rate. Strongly heterogeneous degradation is seen. Near the surface where extensive oxidation has taken place, the material becomes softer as it degrades. At the

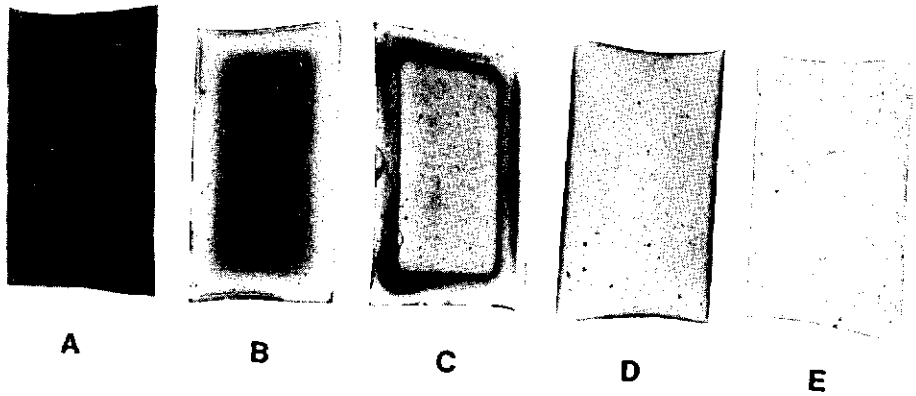

Figure 2.8.2 Cross-sectioned, polished samples of EPR which had been gamma-irradiated at 70°C. A: Unirradiated material. B: 6.7 x 10^5 rad/h (in air) to 165 Mrad. C: 6.7 x 10^5 rad/h (in air) to 297 Mrad. D: 1.1 x 10^5 rad/h (in air) to 175 Mrad. E: 1.1 x 10^6 rad/h (in vacuum) to 253 Mrad. All irradiations carried out at 70°C. Actual sample thickness = 3.15 mm. (From Reference 2.8.4)

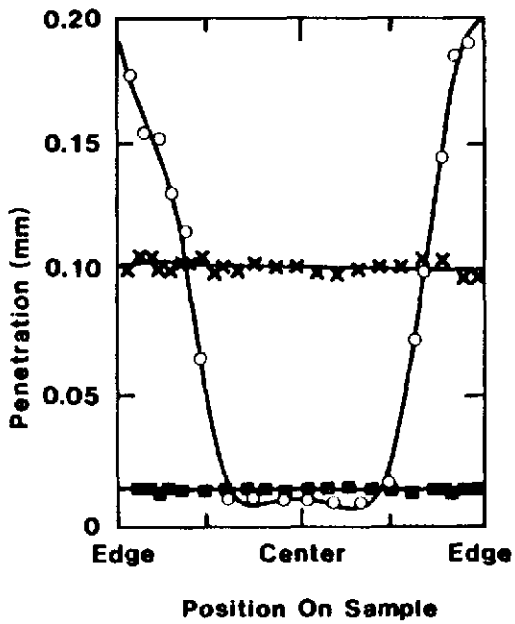

Figure 2.8.3 Hardness profiles for irradiated Viton samples. X = unirradiated material, ○ = 1.8 x 10^5 rad/h to 186 Mrad, □ = 9.0 x 10^5 rad/h to 191 Mrad (under vacuum). Probe load was 3 g. The samples had a thickness of 1.9 mm. (From Reference 2.8.4)

interior, where no oxygen penetrates, the material becomes harder as it degrades. The implications of this kind of result, for accelerated aging tests, is significant.

Figure 2.8.4 illustrates the very different degradation behavior of this material at high and low dose rates. At the highest dose rates, oxidative penetration is minimal, and the material becomes overall harder and brittle as it is irradiated. At the lowest dose rates where oxidation penetrates essentially throughout, the material becomes overall softer as it is irradiated.

Very recently, modifications to the hardness profiling technique have been made that result in an apparatus which allows a quantitative profile of a material's tensile modulus to be determined (Reference 2.8.4C). Some further details on this modulus profiling method are described in Section 2.3.

The extent of oxidative penetration in polymers subjected to irradiation in the presence of air depends on the oxygen penetration constant for the material, the rate at which the material oxidizes, and the material thickness. Table 2.8.1 gives data which we obtained for the depth of significant oxidation for a number of commercial materials subjected to irradiation in air as a function of dose rate and temperature.

References 2.8.6 and 2.8.8, NUREG/CR-4358 (SAND85-1557), describe the use of a density profiling technique which allows a detailed derivation of the shape of oxidation gradients in degraded polymers. The technique reflects the difference in the amount of chemically bound oxygen in different regions of the oxidized polymer. This technique operates by placing a series of tiny pieces of sample that have been cut from across the cross-section into a density gradient column. This reference illustrates that the technique is important not only for identifying heterogeneous oxidation due to diffusion effects, but can also identify heterogeneities due to other effects. Figure 2.8.5 shows a large oxidative heterogeneity in an EPR cable insulation material. Here, tremendously-enhanced oxidation has occurred only near the inside edge of the insulation, which had been adjacent to a copper conductor. The enhanced degradation was shown to be due to catalysis by copper ions which had migrated into the insulation material.

Chemical Techniques

Reference 2.8.9, NUREG/CR-2156 (SAND80-2149), describes techniques developed for identifying an important type of dose-rate effect which arises from a chemical mechanism. This mechanism involves the time-dependent thermal decomposition of peroxides which form upon radiation exposure in the presence of air. This peroxide breakdown constitutes a chain-branching step in the chemical kinetics. At high dose-rate, accelerated, test conditions, peroxide breakdown may be insignificant because of the short time periods involved. However, under

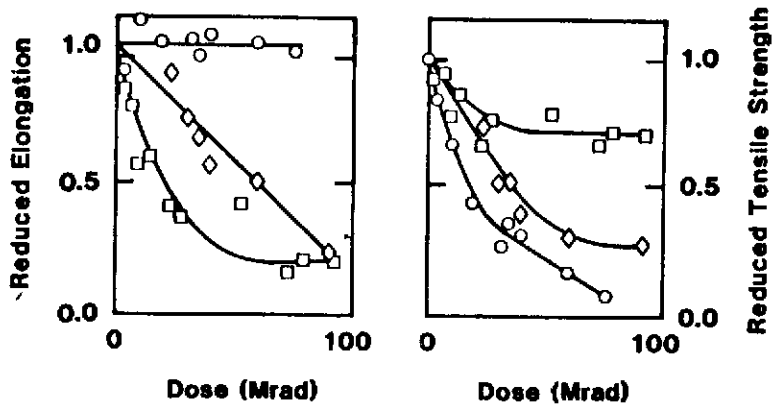

Figure 2.8.4 Change in ultimate tensile properties for a Viton material irradiated at 70°C using three different dose rates. □ = 5.5 x 10^5 rad/h (in air), ◊ = 9.2 x 10^4 rad/h (in air), ○ = 1.3 x 10^4 rad/h (in air). Left: reduced elongation (e/e_o), Right: reduced tensile strength (T/T_o). (From Reference 2.8.4)

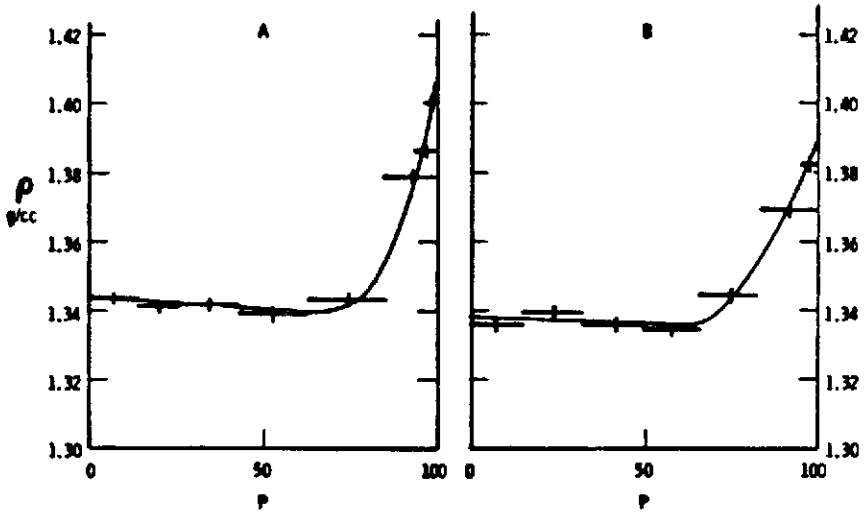

Figure 2.8.5 A: Density profile for EPR insulation aged to 135 Mrad at 52-krad/hr in air. B: Density profile for EPR insulation aged to 53 Mrad at 1.6 krad/hr in air. The percentage of the distance from the outside to the inside of the sample, P, is plotted as the abscissa. (From Reference 2.8.6)

Table 2.8.1
(From Reference 2.8.4)

Oxidative Degradation Depth (as Indicated by Ring Size) of Some Common Polymer Materials as a Function of Dose Rate and Temperature

Material type	Irradiation temperature (°C)	Literature oxygen permeation constant,[a] temperature, and (reference)	Material thickness (mm)	Oxidative penetration depth, dose rate, and dose			
EPR(I)	70	19.0 25°C (22)	3.15	0.54 mm 6.7×10^6 rad/h 165 Mrad	no rings 1.1×10^4 rad/h 121 Mrad		
EPR(II)	70	19.0 25°C (22)	1.91	0.56 mm 6.7×10^6 rad/h 226 Mrad	no rings 1.1×10^4 rad/h 143 Mrad		
	ambient				0.88 mm 1.1×10^4 rad/h 114 Mrad		
Viton	70	2.2 26°C (40)	1.91	0.33 mm 5.5×10^6 rad/h 178 Mrad	0.42 mm 1.8×10^6 rad/h 186 Mrad	0.78 mm 9.2×10^4 rad/h 130 Mrad	no rings 1.3×10^4 rad/h 76 Mrad
	43					.16 mm 9.6×10^4 rad/h 144 Mrad	
Crosslinked PE	43	—	.75	0.15 mm 8.9×10^6 rad/h 119 Mrad	0.35 mm 6.4×10^4 rad/h 116 Mrad	no rings 1.7×10^4 rad/h 114 Mrad	
Medium-Density PE	60	5.5 30°C (41[b])	.68	0.17 mm 3.1×10^6 rad/h 52 Mrad	no rings 2.4×10^4 rad/h 57 Mrad		
	43			0.09 mm 9.5×10^6 rad/h 135 Mrad	0.25 mm 6.0×10^4 rad/h 51 Mrad	no rings 2.0×10^4 rad/h 26 Mrad	
Polyurethane	70	—	2.01	0.30 mm 7.2×10^6 rad/h 156 Mrad			
Tefzel	70	0.6 25°C (42)	.28	0.08 mm 5.7×10^6 rad/h 27 Mrad	no rings 9.3×10^4 rad/h 16 Mrad		
	43				0.12 mm 1.0×10^6 rad/h 46 Mrad		
Chlorosulfonated PE	(ambient)[c]	3.3 30°C (40)	1.21	0.13 mm 9.6×10^6 rad/h 162 Mrad	0.27 mm 4.9×10^4 rad/h 134 Mrad		
Chloroprene	(ambient)[c]	4.0 25°C (40)	1.86	0.17 mm 9.1×10^6 rad/h 159 Mrad	0.22 mm 4.9×10^4 rad/h 135 Mrad	0.34 mm 4.7×10^4 rad/h 129 Mrad	0.72 mm 8.4×10^3 rad/h 149 Mrad
CLPO	(ambient)[c]	—	.75	0.06 mm 1.2×10^6 rad/h 175 Mrad	0.17 mm 2.2×10^6 rad/h 148 Mrad	0.22 mm 5.2×10^4 rad/h 141 Mrad	
PVC	43°C	1.0 25°C (40)[d]	1.43	0.12 mm 9.4×10^6 rad/h 223 Mrad			

[a] Units of: [(std cc) (cm thickness)/(s) (sq cm area) (cm Hg pressure drop)] $\times 10^{10}$.
[b] density = 0.922.
[c] Ambient temperature of the source: 10^6 rad/h was ca. 42°C, 10^5 rad/h was ca. 28°C, 10^4 rad/h was ca. 25°C.
[d] 20% plasticizer (DOP).

low dose-rate, long-term conditions, the breakdown of peroxides can become so important as to dominate the degradation mechanism. This can lead to much higher damage per equivalent dose under the long-term conditions. There are several tests for the importance of this peroxide breakdown in various polymeric materials. First, when this mechanism is important, samples irradiated in air at room temperature and then placed in elevated temperature are found to be sensitized. That is, the subsequent thermal exposure results in extensive degradation--much larger than similar thermal exposure of unirradiated materials. The role of peroxides in this sensitization process can be further confirmed by carrying out the irradiation in the presence of nitrogen. Without oxygen, no peroxides form. The result is that nitrogen-irradiated samples show no sensitization. A further test consists of air irradiation of a group of samples, followed by treatment of part of the samples with phosphine gas. The phosphine destroys the peroxides. The phosphine-treated samples exhibit no sensitization with respect to subsequent thermal exposure. Table 2.8.2 illustrates this phosphine blocking technique.

Table 2.8.2
(From Reference 2.8.9)

Sequential Experiments With and Without PH_3: Tensile Elongation Data

EXPERIMENT*	e/e_o**	
	PVC	PE
Unaged Material	1.0 ± 0.05	1.0 ± 0.1
γ	0.74 ± 0.04	0.71 ± 0.08
γ; T	0.38 ± 0.03	0.03 ± 0.03
γ; PH_3; T	0.74 ± 0.04	0.75 ± 0.08

* γ = gamma radiation - 4.5 krad/hr (for PE), 4 krad/hr (for PVC) - both at 25°C for 96 days in air
T = thermal exposure of 80°C for 25 days in air
PH_3 = 1.4×10^6 Pa (200 psi) of PH_3 at 25°C for 1 day
** Sample tensile elongation divided by initial elongation. e_o = 310% for PVC; e_o = 540% for PE

Reference 2.8.10, NUREG/CR-2962 (SAND82-2164), describes work on a characterization technique called thermally stimulated current, which was evaluated for use in studying polymer degradation. The goal was to explore this technique as another useful tool, but it was found that the broad current-release peaks which result from complex materials limited its usefulness.

Sensitive Techniques

Reference 2.8.11, NUREG/CR-3236 (SAND83-0635), describes developmental work on a new technique, multiphoton ionization, for identifying gaseous molecules in extremely low concentration. If this technique could be applied successfully to radiation aging of polymers, it could make it possible to monitor degradation rates (by sampling gaseous effluents) even under very low level ambient conditions, such as those which occur in a nuclear plant. The conclusion of the study was that while the technique is sufficiently sensitive, there are problems of selectivity. That is, it is difficult to see a small number of degradation product molecules among the background signal from high concentrations of oxygen.

Work on refining this technique, which includes further laser source development, is proceeding on non-NRC funds. It may be that this technique could be eventually used for in situ measurements to observe real-time degradation.

The density technique, mentioned earlier under Profiling Techniques, also represents a very sensitive analytical technique, which is often able to monitor the early stages of degradation. Figure 2.8.6 from Reference 2.8.8 shows density profiling data for an EPR cable insulation after thermal aging at 100°C. Significant density increases caused by the copper-catalysis mechanism are observable from density profiling before measurable changes in mechanical property data (Figure 2.8.7) occurs.

Since density measurements require very small sample slices, the technique possibly represents a sensitive and pseudo non-destructive means of in-plant monitoring of material condition.

To test this concept we compared density results from our accelerated-aging studies of Savannah River polyethylene insulation (References 2.8.2, 2.8.8 and 2.8.9) with results for material aged under natural aging conditions at Savannah River. The unaged red-colored insulation had a density of 0.925 g/cc. After only 2.8 Mrad of radiation at 80°C and 3.8 Krad/hr., the density increased to about 0.953 g/cc. This extreme sensitivity of density to small radiation doses in accelerated tests was confirmed using ambient-aged samples removed from Savannah River, as shown in Table 2.8.3 for red and orange insulation removed from three locations in Savannah River. At Location 1, the materials were exposed to zero radiation and 43°C for 12 years. For the red insulation little density change was found after 0.5 Mrad but a large increase occurred after 2.6 Mrad, corresponding to severe mechanical embrittlement. The orange-colored insulation, which was mechanically the

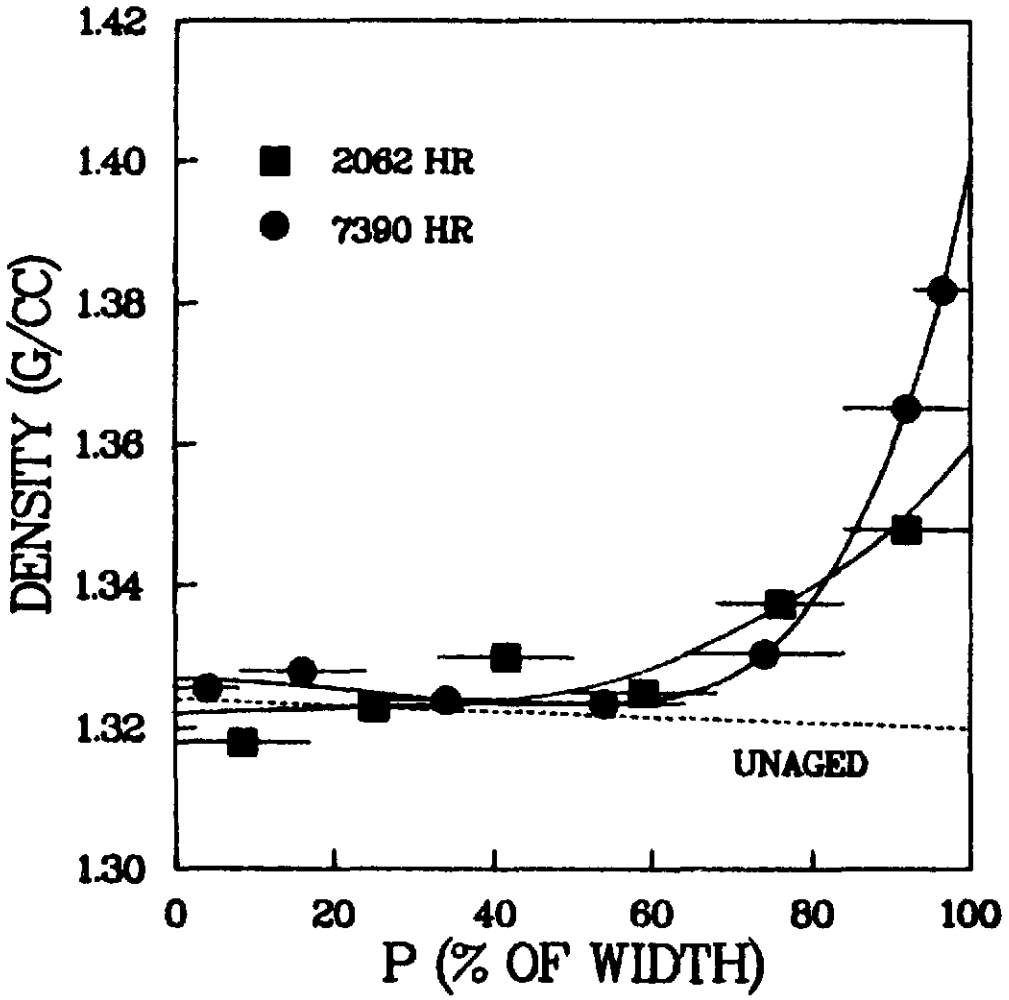

Figure 2.8.6 Density profiles for the EPR cable insulation material which had been heat aged in air at 100°C. (From Reference 2.8.8)

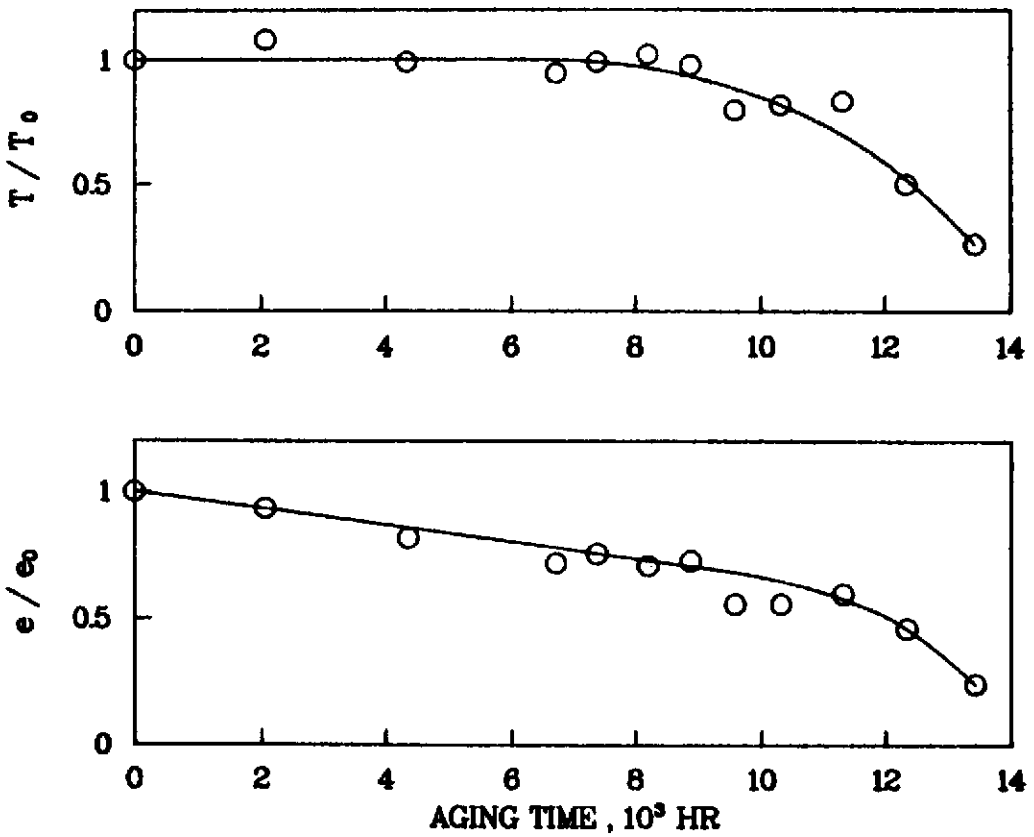

Figure 2.8.7 Mechanical property heat aging results at 100°C for the EPR cable insulation material. The tensile strength after aging divided by the tensile strength before aging (T/T_0) and the tensile elongation after aging divided by the tensile elongation before aging (e/e_0) are plotted against the aging time. (From Reference 2.8.8)

Table 2.8.3

Location	Dose (Mrad)	Density Red (g/cc)	Density Orange (g/cc)
1	0	0.929	0.926
2	0.5	0.929	0.947
3	2.6	0.969	-

most sensitive color of insulation, showed a dramatic increase in density after only 0.5 Mrad estimated dose. Mechanically, in this location, the orange insulation passed a mandrel bend test in which it was bent on a mandrel whose diameter was equal to the insulation diameter.

These results from Savannah River confirm the conclusions from the accelerated tests. Density can be extremely sensitive to low total doses of radiation and may be a good technique to both monitor early storage degradation and assess "nondestructively" the condition of materials aging in reactor environments.

Miscellaneous Analytical Techniques

Reference 2.8.12, NUREG/CR-3808 (SAND84-7135) describes an analytical technique based on a weak-link analysis approach and age-related degradation. While specifically related to aging-seismic correlations, it has a broader generic applicability through its Failure Modes and Effects Analysis (FMEA):

> "The purpose of this work is to develop a methodology for screening electrical equipment into various categories; the equipment will fall into groups that range between those in which no correlation between aging and seismic capacity exists and those in which a significant correlation exists.
>
> "The basis of the methodology for evaluating potential aging seismic correlation is the premise that equipment will fail during a seismic test when a "weak link" in the equipment mechanically fails by yielding or breaking. The probability of failure is calculated by using a normal probability function to define the relationship of the stress applied to the weak link and its frequency (or probability) of failing. If the weak links are fabricated from materials (e.g., elastomers, plastics) which degrade in strength with time and which may bear high loads during a seismic test, then a significant increase in the probability of failure with age may exist...

"The probabilistic approach to evaluating the effects of aging and seismic stresses on the operability of safety-related equipment is considered more realistic than the presently used deterministic approach. Another advantage in using the probabilistic method is that the results can be used in determining the effects of aging on the probability of the system being able to perform its safety functions; an item which should be considered when completing a Probabilistic Risk Analysis."

Equipment Studies

Electrical Cables

Mechanical stress is another environment which can affect the functionability of electric cable. A correlation between material-age and voltage-breakdown has been investigated. Electric cables were subjected to two failure mechanisms which can be caused by mechanical stress. First, if a cable is stretched by an applied force (or by its own weight) over an edge with small curvature, the metal wires will gradually creep through the soft polymeric insulation resulting in metallic contact between cable wires or a wire and the cable support, shorting the cable (creep shortout). Second, the polymeric materials embrittled by aging may crack under mechanical stress; in the presence of steam and spray, strong leakage currents or shortout may occur. The two mechanisms are discussed in detail in References 2.8.13, NUREG/CR-3623 (SAND83-2622), and 2.8.14, NUREG/CR-4548 (SAND86-0494). As an example of the testing itself, various cable sections were hung over support rods, weighted, and thermally aged, all simultaneously. Figure 2.8.8 shows the reduction of cable thickness, at the support point, for 1-lb. weighted cables. From this type of data, a mechanical stress, aging, voltage-breakdown correlation can be developed.

Batteries

The current method to accelerate the age of battery cells is by exposing the complete cell(s) to an elevated temperature for an extended time period. The aging period in days is determined by multiplying the desired age in years by published equivalent aging factors which are given in IEEE 535-1979. The rationale for this method is that the entire cell will be aged to the predominate aging failure mode which is positive grid growth and then plate failure. Existing data-base to support this methodology appears to be very sparse.

Natural- and accelerated-aged battery tests are currently underway as described in Reference 2.8.15, NUREG/CR-3923 (SAND84-1737). [Other pertinent references are Reference 2.8.16, NUREG/CR-4095 (SAND84-2628), Reference 2.8.17, NUREG/CR-4096 (SAND84-2629), Reference 2.8.18, NUREG/CR-4097 (SAND84-2630), Reference 2.8.19, NUREG/CR-4098 (SAND84-2631), and Reference 2.8.20, NUREG/CR-4099 (SAND84-2632).] From this type of effort will come a specific data-based aging method applicable to nuclear station batteries. In addition, an objective is to

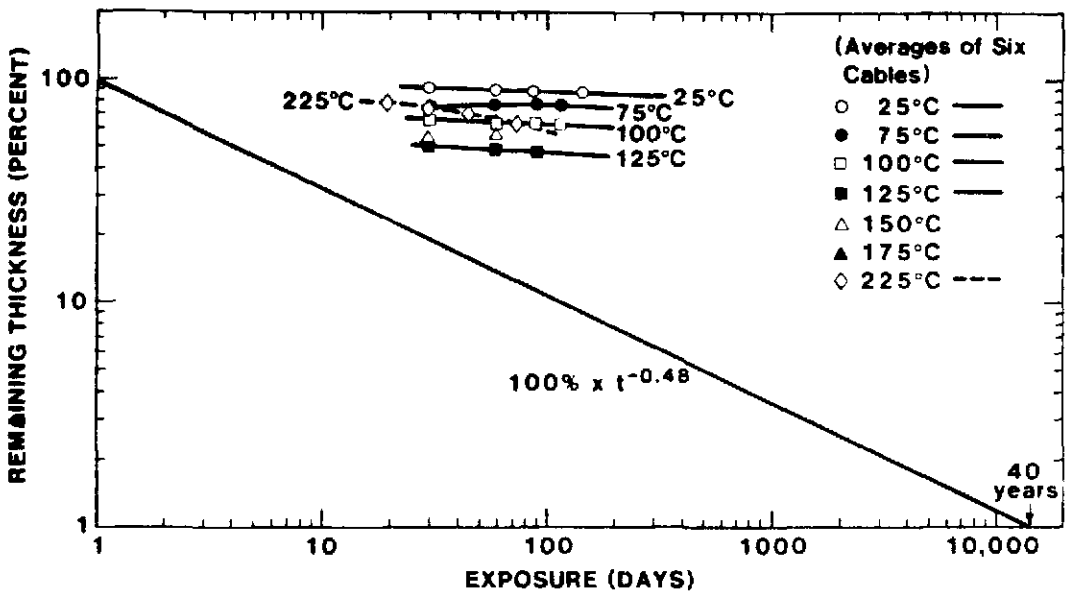

Figure 2.8.8 Average reduction of cable thickness for 6 cables hung over support rods, weighted, and thermally aged, all simultaneously. (From Reference 2.8.13)

identify and develop appropriate (and in situ) degradation indicators, not as a replacement for, but to complement, the current capacity discharge testing. (Currently, a small effort is underway at Westinghouse-Pittsburgh, but is unpublished to date.)

International and Other U.S. Programs

The only work we are aware of that is directly addressing the limitations of current techniques relative to aging is being done at the University of Connecticut (Reference 2.8.21). This group is screening various sensitive analytical techniques to look for chemical degradation products resulting from accelerated and natural aging of electric cables.

Several EPRI and other NRC programs results could eventually provide data which relate to this activity.

Issue Findings

Polymeric Materials

We are currently applying the polymeric materials techniques described above to characterize the degradation process of major polymer types. This, in turn, is allowing us to construct models for extrapolation from accelerated aging test results.

The techniques are also now being used by workers outside Sandia who are addressing the same issues. For instance, we have been informed that the Japan Atomic Energy Research Institute is making use of our techniques for studying heterogeneous oxidation in their program on materials aging and reactor safety. We have also been told that workers at the University of Connecticut, who are being funded by EPRI to study radiation aging, are also employing our profiling techniques.

In summary, the findings within this subissue are:

1. Several new techniques have been developed to identify and study heterogeneous oxidation which often occurs in accelerated aging tests:

   - metallographic polishing
   - density-gradient profiling
   - relative hardness profiling
   - tensile modulus profiling.

2. Techniques have been developed to monitor/understand the chemical mechanisms important to chemical dose-rate effects:

   - sequential studies in air/nitrogen
   - phosphine treatment studies
   - kinetic modelling studies.

3. Thermally-stimulated current measurements are currently too difficult to employ in degradation evaluations of complicated materials.

4. Multiphoton ionization would have to be developed to a higher selectivity before it would be useful in studying low-level "ambient" aging.

Analytical Techniques

In summary, the findings within this subissue are:

1. A probabilistic approach to evaluating the effects of aging and seismic stresses on the operability of safety-related equipment is considered more realistic than the presently used deterministic approach.

2. The probabilistic method appears to be useful in determining the effects of aging on the probability of the system being able to perform its safety function.

Equipment Studies

In summary, the findings within this subissue are:

1. A correlation between mechanical stress, aging, and voltage-breakdown is being developed.

2. A study of artificial battery cell aging techniques is underway. The results are being compared to naturally-aged battery cells in an attempt to identify and evaluate all degradation indicators.

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Polymeric Materials

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2.9 Comparison of Artificially- and Naturally-Aged Equipment

Issue Description

In a test to determine if a specific equipment item is qualified for Class 1 service, part of the test methodology includes age preconditioning. Such preconditioning generally includes radiation and thermal aging, cycle aging, vibration aging, and similar tests. Since most qualification tests are not performed on naturally-aged equipment and components, the end-of-installed life condition is usually achieved through artificial (accelerated) means.

The assumption, of course, is that accelerated preconditioning will result in the equipment being in the same state as if it had aged naturally. Proof that artificial-aging methods are valid can only come through comparisons with naturally-aged equipment. Further, the sensitivity of the equipment to accident tests may be strongly influenced by proper preconditiong (see Issue 1.9).

An equivalence between natural and accelerated aging is implied by Section e.5 of 10 CFR 50.49, which states

> "Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment."

Conversely, Regulatory Guide 1.89, Section 5.a requires

> "If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program. Synergistic effects known at this time are dose-rate effects and effects resulting from the different sequence of applying radiation and (elevated) temperature."

This is the thrust of this effort to address the issue:

> "What aging and degradation experiences are available from actual plant use, and can they be correlated with accelerated-aging methods?"

Batteries

To adequately determine the qualification of nuclear station batteries, representative cells which compose the battery must be exposed to simulated seismic events. The seismic events must be representative of the seismic conditions to which the inservice batteries could be subjected. For a complete qualification, the cells should be exposed to the simulated events at ages which represent their expected end-of-qualified-life condition. While naturally-aged cells may be used for qualification testing, such cells are not usually available and time

constraints may not permit waiting until such cells are available. To circumvent the problem of nonavailability of naturally-aged cells, cells aged by accelerated methods may be used.

However, testing accelerated-aged battery cells is of no value if the natural aging and degradation experiences are not duplicated in accelerated-aged cells. Hence, it is desirable to compare the relative degradation and the seismic performance of naturally-aged batteries to accelerated-aged batteries.

The initial effort was a survey to determine the availability of naturally-aged battery cells. The concept was that through seismic-fragility testing and pre- and postseismic inspections and chemical analyses of cell internal components, aging and degradation mechanisms of naturally-aged cells could be determined. This effort would then be followed by purchasing new cells of the same type as the naturally-aged cells. Some of these new cells then would be exposed to simulated seismic events to establish baseline data, while others would be aged to the same age as the naturally-aged cells. These accelerated-aged cells would then experience the seismic-fragility testing and pre- and postseismic inspections and chemical analyses to determine their aging and degradation mechanisms. The two sets of aging and degradation data would be compared to determine their correlation, if any.

Reference 2.9.1, NUREG/CR-3923 (SAND84-1737), provides the results of the first set of cells to be tested, 12-year old Gould NCX-2250 cells. References 2.9.2, NUREG/CR-4095 (SAND84-2628), 2.9.3, NUREG/CR-4096 (SAND84-2629), and 2.9.4, NUREG/CR-4097 (SAND84-2630), report on other cells - 10-year old Exide FHC-19, 10-year old C&D LCU-13, and 27-year old Exide EMP-13 cells, respectively.

To establish the aging degradation mechanisms that are susceptible to a seismic event, the electrical properties including discharge capacity of the cells were monitored through a graduated series of g-level step increases until either the shake table limits were reached or electrical or mechanical "failure" of the cells occurred. Battery performance was monitored both during the seismic event and during postseismic tests. IEEE Std. 535-1979 indicates that there should be no abrupt changes of more than five percent of the current or voltage of the battery discharge during the seismic test. During postseismic tests the battery discharge capacity should be 80% or more of the cell's rated discharge capacity.

The results of the four series of tests have been documented and will be used later for comparison to seismic testing results for similar cells aged by accelerated methods. The summarized results of the four tests follow:

- The 12-year old NCX-2250 cells exhibited a 66 percent electrical failure rate during the seismic-fragility testing phase (with g-levels in excess of 1-g ZPA). Subsequent examination showed a

common failure mode of cracking at the abnormally brittle, positive lead bus bar-terminal post interface. No significant plate growth was observed on any of the positive plates. Posttest capacity tests demonstrated that the three cells that did not fail during seismic simulation did have capacities greater than the accepted minimum value of 80 percent.

- The 10-year old FHC-19 cells did not "fail" during exposure to two different series of g-level step increases (to above 2-g ZPA) but did have a 75% "failure" rate during postseismic discharge capacity measurements. Although no positive plate growth was noted, the positive and negative plates were cracked and brittle. The positive bus bar was also brittle and corroded. The corrosion had reduced the bus material available for conduction to just over 50 percent of the original area.

- The 10-year old LCU-13 cells also did not fail during exposure to two different series of g-level step increases (to above 2-g ZPA) and also exhibited a relatively low (22 percent) "failure" rate when their posttest capacities were determined. The two cells that did "fail" the posttest capacity tests were the two that had been exposed to the second series of higher g-level step increases. All plates were in good condition with the positive plates showing minimal loss of active material and no significant plate growth. No corrosion or cracks were noted on the positive bus bars.

- The EMP-13 cells, even though 27-years old, did not "fail" during the seismic events (to above 1-g ZPA), but the postseismic tests showed capacities of only 33 percent and lower. Disassembly of cells showed the negative plates to be very weak and brittle and severely erupted on both sides. Positive plates, while solid and now warped, did have the lead buttons clogged with active lead oxides and showed evidence of corrosion products being dislodged from the positive grids. The positive bus material was brittle and showed evidence of chemical attack but no evidence of cracks were observed in either the post or bus material.

These preliminary results suggest that plate growth may not always be the dominant aging mechanism affecting battery performance during seismic events. IEEE Std. 535-1979 was apparently developed with the assumption that plate growth is the dominant aging mechanism and that elevated temperature exposures would accelerate this phenomena. Further evaluation of cell degradation mechanisms and how cell degradation is impacted by seismic events is provided by Reference 2.9.5, NUREG/CR-4099 (SAND84-2632). It also provides a baseline against which future accelerated-aged results can be compared.

To establish an unaged (neither accelerated- nor naturally-aged) baseline, new NCX-2250, FHC-19, and LCU-13 cells will be exposed to seismic testing consisting of two different series of g-level step increases. Only the first phase of testing at the lower g-level steps

have been accomplished. None of the cells failed during the seismic testing and all cells had posttest capacities of more than 80 percent. The results of the testing at both series of g-level step increases will be published as NUREG/CR-4098 (SAND84-2631), and is shown as Reference 2.9.6.

To provide accelerated-aged cells, new cells of the same type as the naturally-aged batteries are being thermally aged to 10 and 12 years, respectively. They will then be exposed to seismic g-level step increases as were the naturally-aged cells and the results will be compared. Additional future efforts include accelerating the age of both the 10- and 12-year naturally-aged cells as well as the accelerated-aged cells to an age of 18 years; additional seismic tests will then be performed and the results compared.

Organic Materials

Organic materials are used extensively in Class 1 equipment: Cable insulation and jacketing, seals, gaskets, and O-rings, for example. Thus the aging behavior of organic materials is of particular importance to equipment qualification testing.

Reference 2.9.7, NUREG/CR-2877 (SAND81-2613), describes an investigation of the deterioration of PE and PVC cable materials which occurred in the containment building of the Savannah River nuclear reactor. Radiation dosimetry and temperature mapping data indicated that the maximum dose experienced by the cables was only 2.5 Mrad at an average operating temperature of 43°C. The amount of material degradation observed seemed surprising, considering the relatively moderate environment.

An experimental program was initiated, on the commercial PE and PVC materials used at the plant, to investigate their degradation behavior under combined gamma irradiation and elevated temperature conditions and various irradiation and temperature sequences. The important aging effects observed included:

1. A strong synergism between radiation and temperature. (See Figures 2.9.1 and 2.9.2.)

2. Large dose-rate dependent effects which occurred over a wide range of dose rates. (See Figures 2.9.3 and 2.9.4)

3. A dependence of the degradation observed on the ordering of the sequential radiation, elevated-temperature exposure.

It was established that the material deterioration observed at the plant resulted from radiation-induced oxidation and that the degradation rate could be correlated with local levels of radiation intensity in the containment area. The report concluded that test methods for aging and qualification testing of organic materials should be designed to test for

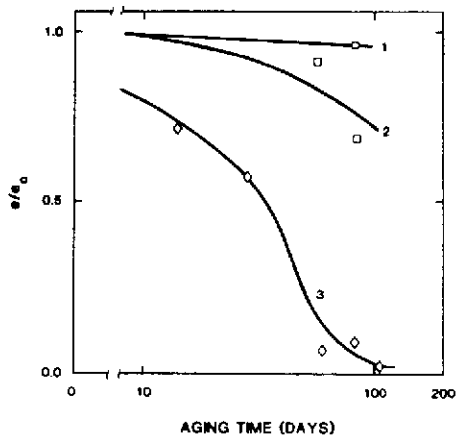

Figure 2.9.1 Tensile elongation results for PE insulation as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 5 krad/hr at 25°C; 3) radiation at elevated temperature, 5 krad/hr at 80°C. (From Reference 2.9.7)

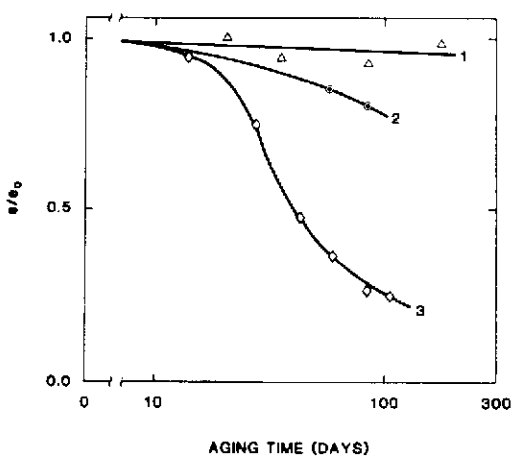

Figure 2.9.2 Tensile elongation results for PVC jacketing material as a function of aging time in three different laboratory environments: 1) elevated temperature, 80°C; 2) radiation at room temperature, 4.4 krad/hr at 25°C; 3) radiation at elevated temperature, 4.4 krad/hr at 80°C. (From Reference 2.9.7)

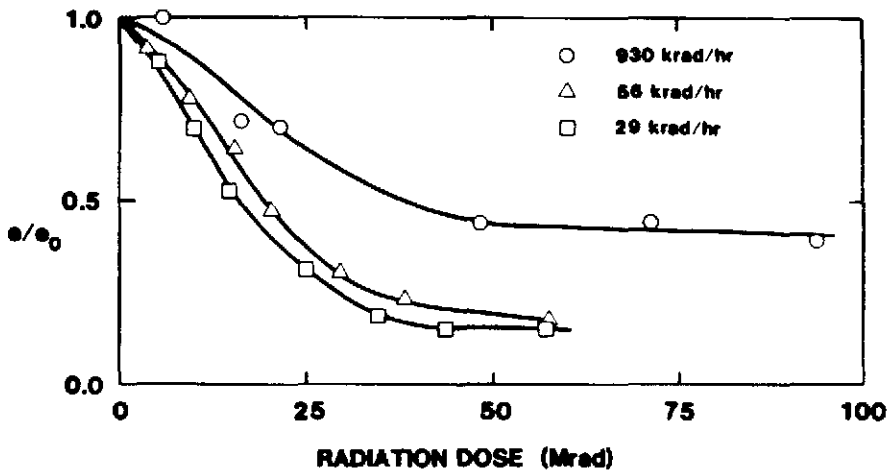

Figure 2.9.3 Tensile elongation results for PVC jacketing as a function of dose at 43°C using three different dose rates. (From Reference 2.9.7)

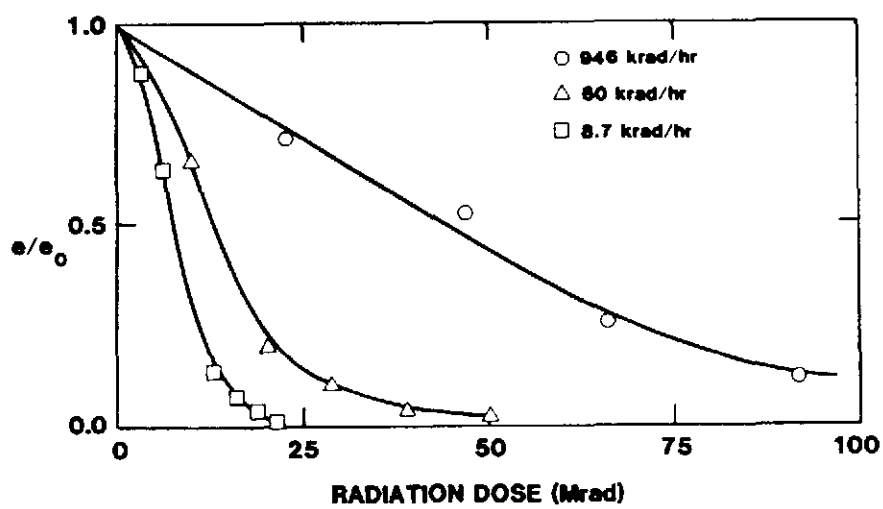

Figure 2.9.4 Tensile elongation results from PE insulation as a function of dose at 43°C using three different dose rates. (From Reference 2.9.7)

dose-rate effects and synergisms. The report also noted that in the past, studies intending to survey the radiation stability of organic materials were typically performed over short time scales at very high dose rates. Reference 2.9.7 noted that such survey results may be inappropriate when selecting materials for low-dose-rate, long-term applications.

Reference 2.9.8, NUREG/CR-4008 (SAND84-1948), provided a general extrapolation model to explain the PVC and PE behavior that was described by Reference 2.9.7. The model allows for laboratory accelerated-aged-data to be extrapolated to temperatures and dose rates observed at the Savannah River reactor. Hence, comparison between artificially- and naturally-aged cable properties was possible. Figure 2.9.5 illustrates model predictions for PVC tensile strength and compares results to naturally-aged material. Excellent correlation was obtained. Similar results were noted for PE.

International and Other U.S. Program

Batteries

The French are known to be interested in the aging of battery cells as indicated in References 2.9.9 (EDF HM64/313) and 2.9.10 (EDF HM64/312). The references indicate, however, that the artificial methods are acceptable for testing and naturally-aged cell tests are not required. Reference 2.9.9 describes and summarizes three different aging methods but all appear to be based on cycling (charge and discharge) and overcharging (either an overcharging by current or voltage). Reference 2.9.10 is qualification specifications that have resulted from investigative tests over the last seven years. The related reports unfortunately contain manufacturers' proprietary information and are not available to us (Reference 2.9.11).

We are not aware of any other international activities comparing accelerated- and naturally-aged battery cells.

Reference 2.9.12 (Report No. 44681-2) reports on tests of artificially-aged and naturally-aged battery cells. The intent of the effort was to age 82 different cells to 5, 10, 15, and 20 years but because of anomalies occurring during the 5- and 10-year aging sequence, only 32 cells were aged and then only to 5 and 10 years. As a result, no cells were aged to comparable ages of available (8-, 10-, and 16-years) naturally-aged cells, thus voiding the possibility of any comparison of like cells.

Reference 2.9.13 (SAND82-7049), is a Sandia funded effort that studied battery accelerated-aging techniques. It provides an overview of battery aging methodologies but does not consider a comparison of naturally- and accelerated-aged cells. The report does acknowledge the fact that there is a need to develop accelerated-aging techniques so that cycle life data can be acquired in less time than real time testing would require. The

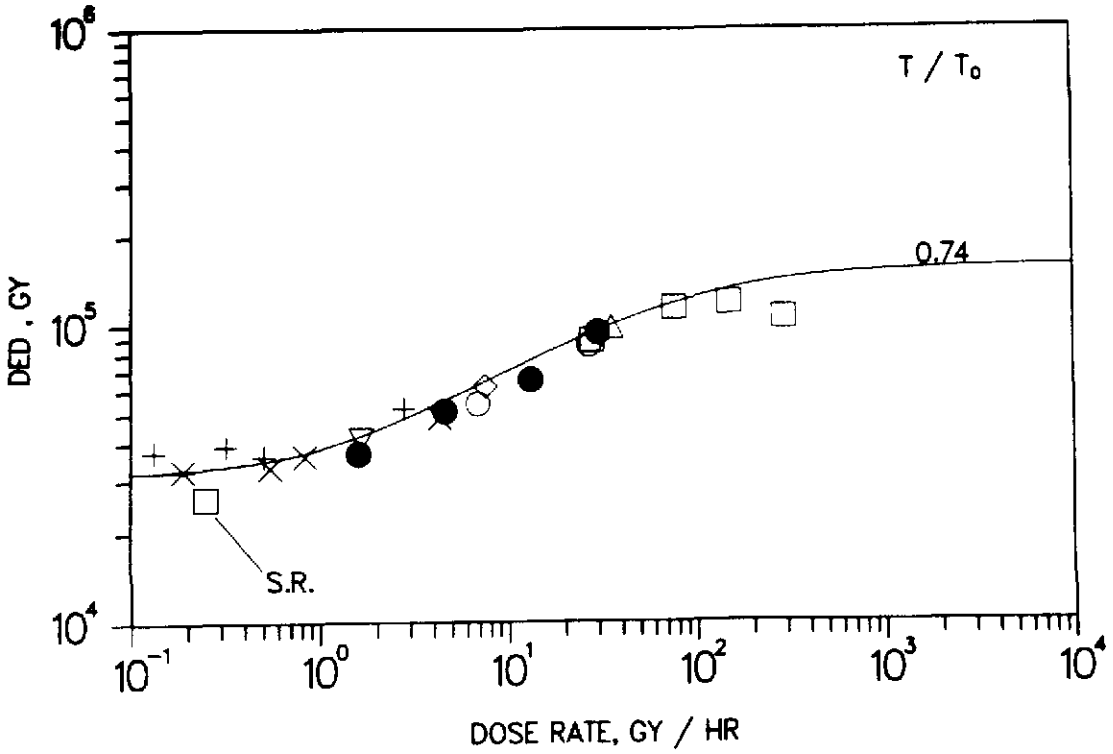

Figure 2.9.5 Required dose to achieve normalized tensile strength (T/T_o) of 0.74 for various dose-rate exposures. S.R. denotes naturally-aged data after a 12-year exposure at Savannah River. Other experimental data points represent laboratory data shifted (by a model) to a reference exposure temperature of 43°C (appropriate for the Savannah River naturally-aged cable). Solid curve is a theoretical fit based on the theoretical model which allows high temperature, high dose rate conditions to be shifted to equivalent lower temperature, lower dose rate conditions. (From Reference 2.9.8)

report advocates a team approach to design accelerated-aging tests (methods), concludes that it is not known whether or not a good accelerated (aging) test can be developed, and recommends that an accelerated (aging) test program be defined and that batteries be purchased to carry out the program.

Organic Materials

Reference 2.9.14, CERN 83-08, compares long-term irradiation results obtained at the CERN Super Proton Synchrotron (SPS) accelerator to short term irradiation results employed to predict cable insulation and jacket behavior. Short-term irradiations were performed at 300 krad/hr (50 Gy/sec) while long-term irradiations were performed between .03 krad/hr (.005 Gy/sec) to 1.2 krad/h (.2 Gy/h). The report concludes that for some types of low-density polyethylene (PE) the reduction of elongation at break at the same dose may be up to a factor of 10 higher when irradiated at the CERN accelerator at low dose rates compared with the irradiation at high dose rates. The same applies for the tensile strength of PVC. For other types of PE and PVC, and also for EPR, the difference in measured values at the same dose was considerably smaller or even negligible.

Reference 2.9.15 notes that KWU-Erlangen personnel and facilities are participating in an in situ cable study. Cables from several manufacturers have been placed in two operating nuclear reactors around the recirculating loop so as to give a dose rate of 80-150 rd/h. In one plant, this experiment has been underway for 5 years. The same cables have been placed in controlled conditions at the KWU-Erlangen Co-60 source at approximately the same dose rates. One goal of the experiment is to determine whether the more harsh conditions of the power plant (higher temperatures, humidity, etc.) produce different cable degradation then the controlled radiation conditions at KWU-Erlangen. Some results are expected in the next year.

Reference 2.9.16 reports on an ongoing cable monitoring effort at the Oconee Nuclear Station. Various types of galvanized-steel sheathed cable were installed in cable life-evaluation circuits in the reactor building. At five year intervals over the life of the plant, cable samples are to be removed from the life-evaluation circuits and tested to determine the effects of the reactor building environment on the integrity of the cable. Reference 2.9.16 reports on cable conditions after five years and ten years of operation. Average radiation exposures for these two time periods were estimated to be 1.8×10^4 Rem and 3.5×10^4 Rem, respectively. Test specimens included PE, XLPE, EPR, and HTK insulation and PVC and neoprene jackets. The report notes that visual examination and physical and electrical data obtained from various tests indicate that all cable samples were flexible and in good condition overall. No comparison between naturally-aged and artificially-aged specimens was reported.

A program with EPRI sponsorship at the University of Connecticut has started placing polymer materials in a number of nuclear power plants to generate long-term, natural-aging data for comparison with accelerated-aged results. (Reference 2.9.17)

Other Equipment

Reference 2.9.18, NUREG/CR-3424, is a report documenting the results of an equipment qualification research tests performed on nine ASCO solenoid valves. The purpose of the program was to provide independent verification of previous testing performed on similar solenoid valves. In addition, the substitution of two naturally-aged valves for specimens that had malfunctioned after thermal aging, allowed a comparative evaluation of the behavior of artificially aged valves to those that were naturally aged.

The solenoid valves were (accelerated) aged to 50 Mrad gamma radiation followed by thermal aging at 268°F to simulate a 4-year life at 140°F. Two thousand operational cycles were performed during thermal aging. Two of the aged valves were removed from testing and replaced by valves that had been naturally aged (3 years at 140°F while energized and pressurized with N_2). All valves were cycled 2000 times at room temperature, then exposed to vibration aging, seismic testing, 150 Mrad gamma irradiation, and a 30-day LOCA/MSLB simulation.

Two valves leaked excessively after thermal aging. All valves performed satisfactorily during seismic testing (a tenth, unaged, valve was included in this test and the results did not reveal any difference in behavior between unaged, naturally aged, and artificially aged valves). Of the seven valves subjected to the LOCA/MSLB exposure: One performed satisfactorily, two performed their safety function but leaked, and four failed during transients. Although the naturally-aged valves were subjected to less severe aging then the artificially aged valves, they failed earlier in the LOCA/MSLB test.

The valve that functioned properly throughout the test series without leaking had a metal valve seat, all of the other valves had seats made of EPDM.

The reference suggested that some elements of the accelerated-aging test program were nonconservative and others were overly conservative. One nonconservative element was the use of nitrogen as the process fluid during thermal aging (instead of compressed air as in nuclear power plant applications). This had the effect of suspending the oxygen-dependent degradation. Therefore, the real time simulated by the accelerated thermal aging was considered to be less than the four years used to define the thermal aging conditions.

Reference 2.9.18 noted that the exhaust path materials of a naturally-aged valve showed a much higher level of degradation than did the artificially-aged valve parts "wetted" by nitrogen during aging.

It was also concluded that cycling the artificially aged valves at the elevated, accelerated-aging temperature (131°C) was an overconservative element of the test. This procedure probably overstressed the softened organic seats and seals so that the EPDM degraded in a way that is not representative of its performance in normal service. Reference 2.9.18 noted that the guidance of IEEE Std. 382-1980 with regard to cycling during thermal aging should be used with caution

Very recently, we have been informed (Reference 2.9.19) that an EPRI-sponsored seismic test of artificially-aged and naturally-aged equipment has been completed. Most (of the tested) equipment showed no differences in test responses. However, differences were observed for some limit switches, pressure switches, and rotary switches.

Issue Findings

The ultimate proof of an accelerated-aging method involves its demonstration against real time aging experiences. By carefully selecting naturally-aged materials and equipment taken from nuclear power plants it should be possible to correlate the observed accelerated-aging techniques with the real degradation or failure observed. This is best accomplished by taking like materials or equipment and attempting to accelerate the age. The results of these demonstrations directly suggest the validity of the model, or may suggest adjustments in the simulation model. Test results have demonstrated the importance of this type of effort. Seismic testing of aged batteries suggested that plate growth may not always be the dominant aging mechanism affecting battery seismic fragility failure. Examination of naturally-aged cable suggested that the equal radiation dose-equal damage hypothesis does not apply to all cable materials. Finally, examination of naturally-aged and artificially-aged solenoid valves indicated that the use of nitrogen gas as the process fluid during thermal aging was nonconservative.

Batteries

In summary then, the findings within this subissue are:

1. Corrosion to varying degrees of both the positive bus bars and the positive bus bar/terminal post interface region.

2. Complete separation (breakage) of terminal post from the bus bar.

3. A high rate of self-discharge of cells exposed to a second series of simulated seismic events.

4. "Failure" of many cells exposed to a second series of simulated seismic event in that many of the cells had a posttest capacity of less than the accepted minimum value of 80 percent.

5. A gross mechanical/electrical failure of one type of cells due to separation of terminal post from bus bar.

6. No common indicator immediately following the simulated seismic event results which would predict the 3-hour posttest capacity of the cells.

7. Substantial degradation of some plates (both negative and positive plates) of some type cells.

Organic Materials

In summary then, the findings within this subissue are:

1. Savannah River cable material deterioration resulted from radiation-induced oxidation.

2. The degradation of the naturally-aged (Savannah River) cable materials could be correlated with local levels of radiation intensity.

3. The degradation of the naturally-aged (Savannah River) cable materials could be correlated to predictions based on accelerated-aged data and on general extrapolation model for an important chemical dose-rate effect.

4. Accelerated-aging methods for organic materials should take into consideration the significance of dose-rate and sequence effects to the safety-related performance of the materials.

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3.1 Complementary Test Approaches and Considerations

Issue Description

Regulatory Guide 1.89 defines qualification as "...as a verification of design limited to demonstrating that the electric equipment is capable of performing its safety function..." According to Section 5.1, IEEE 323-1974, "Type testing...is the preferred method." Furthermore, IEEE 323-1974, Section 6.3.1, states that, "The type test shall be designed to demonstrate that the equipment performance meets or exceeds the requirements of the equipment specifications for the plant."

Current equipment qualification is based on type testing, which provides assurance that the equipment will not fail due to common-mode mechanisms. And it must incorporate margin to account for uncertainties in production of equipment and in (reasonable) errors when defining success criteria.

Some testing experiences suggest that multiple sample tests and fragility tests may serve to complement type-testing to achieve a broader safety perspective. Multiple sample tests may be appropriate in cases where the failure modes are affected by non-manufacturing considerations; the specifics of plant applications is one example. Fragility testing can help identify failure modes and failure thresholds, giving data-based assurance, for example, that accelerated aging programs have addressed the appropriate system (or equipment) "weak-links."

Other overstress test methods may be useful (to attempt to compensate for dose rate effects, for example), but only when the limits of applicability are thoroughly understood.

This is the thrust of this effort to address the issue:

> "Are there test approaches (such as multiple sample, fragility and overstress testing) which can complement qualification type-testing to achieve a broader safety perspective?"

Limitations of Overstress Approaches

Reference 3.1.1 (SAND83-2085C) discusses the limitations to using "dose overstress" (a higher total dose) to simulate dose-rate effects during radiation testing. The significant points of concern were:

- Dose-rate effects are now widely known. One potential response is to use a larger total dose to account for this.

- The concept of such an overstress approach can be inappropriate. First of all, dose-rate effects vary greatly in magnitude, and based on high dose-rate testing, poor materials with large dose-rate effects may be selected over better materials with small effects. Also, in certain cases, material properties have been

found to level out (as with PVC) or reverse trend (as with Buna-N) at high doses, so that such overstress may be ineffective, misleading, or counterproductive. For Viton®, the material properties were found to change in opposite directions at high and low dose rates, which could not be accounted for by dose overstress.

- The concept of overstress carries the implicit assumption of similarity in polymer degradation behavior. In fact, this is not the case: different polymer types exhibit dramatic differences in degradation phenomena. Furthermore, a given material often exhibits major differences in degradation phenomena under different conditions such as dose rate or temperature. The fundamental problem is that differences both in materials and aging conditions can lead to very fundamental differences in the degradation mechanisms underlying the macroscopic changes. (Examples of such differences can be surface oxidation versus oxidation throughout, or cross-linking as the predominant molecular-level change versus chain scission as the predominant change.)

- The underlying problem with this concept is that differences in radiation conditioning can lead to fundamental differences in degradation mechanisms. These differences in mechanism cannot always be reconciled by the use of extra dose at higher dose rates.

Multiple Sample Testing

The objective of equipment qualification is to assure that equipment will perform, when needed, in a given environment primarily through type-testing. Multiple sample testing may provide additional useful information, however.

The following points, from References 3.1.2 through 3.1.6, show that a different perspective may be obtained using multiple sample testing.

- Terminal Blocks: There exists a continuum of IR values for terminal blocks during a test. Furthermore, the IR value at a particular moment is determined by the interaction of many factors. If the test only collects attribute data (example: 1 amp fuse blown), it may not provide a representative picture of terminal block performance nor allow a defendable argument to be made about the appropriateness of a terminal block for a particular application (Reference 3.1.2, NUREG/CR-3418, SAND83-1617).

- Transmitters: One ITT Barton Model 763 transmitter was tested in each of five different environments to identify individual and synergistic effects of environmental stresses. The primary failure was a thermal instability problem resulting from a degraded performance of potentiometers in the transmitter's circuit. This common-mode failure was identified by a nuclear

utility and Sandia; however, this common-mode failure was not identified in Barton and Westinghouse tests. A secondary failure mode was found which involved corrosion due to a lubricant applied during the manufacturing process. (This corrosion caused the resistive element of the calibration potentiometers to open.) However, two lubricants were apparently used interchangeably in manufacturing. Only one lubricant caused corrosion, so this potentially common-mode failure would only be found in a proportional number of transmitters (Reference 3.1.3, NUREG/CR-3863, SAND84-1264).

- D. G. O'Brien Penetrations: Sandia tests simulated aging and accident conditions. Two degradation mechanisms were found which involved extrusion of the penetration module grommet material due to thermal expansion during the high temperature portions of the aging and accident environments: (1) stripping of insulation from the conductors in the penetration module and (2) intrusion of moisture at the mating connector faces. The failures may have been introduced by the high temperatures used to accelerate aging of assembled units since these high temperatures would not have been seen in a natural aging environment. However, a Wyle test, using a modified (unassembled) aging procedure, also showed moisture intrusion at the mating connector faces. Therefore moisture intrusion is a problem which cannot be dismissed and appears to be statistically significant but perhaps not common-mode (Reference 3.1.4, RS4445/81/03; Reference 3.1.5; and Reference 3.1.6).

- Cables: From examining several Franklin cable qualification reports (not referenced here), there were 41 groups of cables where multiple samples had been tested. In 26 groups, all cables succeeded; in eight groups, all cables failed; and in seven groups, a mixture of successes and failures were found. Testing multiple samples can help identify random successes or random failures.

Fragility Testing

Fragility tests may be used to identify failure modes and thresholds. The following reports describe a potentially common failure mode revealed after seismic-fragility test of naturally-aged batteries (to high g-levels) and previously unreported. Plate grid growth, the basis for accelerated-aging methods (IEEE 535-1979), was found for only one battery type, but lead bus bar corrosion was observed in several battery types.

Reference 3.1.7, NUREG/CR-3923 (SAND84-1737), covers the first test series using 12-year old lead-calcium Gould NCX-2250 cells from the James A. Fitzpatrick Nuclear Power Station. The seismic-fragility tests were done to determine the actual failure modes and threshold. The electrical properties including discharge capacity of cells were monitored through and after a graduated series of g-level step increases until either the

shake table limits were reached or until electrical "failure" of the cells occurred. (This test procedure is applicable to all tests discussed here.)

The results of the test are as follows:

- Six of the nine 12-year old cells (selected at random from approximately 60 cells retired from service) failed under electrical load during seismic simulation.

- Electrical failure of the test cells was caused by severe cracking of abnormally brittle positive bus material and/or disconnection of positive posts from the bus material.

- Decohesion, leading to fracture, occurred mostly along the boundaries of extremely large lead grains and was assisted by chemical corrosion. Coarse grain structure can be attributed to abnormalities in the bus casting process.

- Internal components and connections in two cells without bus defects were extremely durable. These cells survived violent and repeated seismic testing and were capable of meeting the acceptance criterion of 80 percent of rated capacity after the test.

- Plates and separators were generally in very good condition, with no significant plate growth. However, the presently accepted accelerated-aging methods for qualifying batteries, per IEEE 535-1979, are based on plate or grid growth.

- Failures in some cell jars also occurred but only after repeated testing at high acceleration levels; these failures lead to electrolyte leakage.

Reference 3.1.8, NUREG/CR-4095 (SAND84-2628), covers the testing of 10-year old lead-calcium Exide FHC-19 cells from the Calvert Cliffs Nuclear Power Station. The results of the test are as follows:

- The ten-year old Exide FHC-19 cells showed signs of age-related degradation. Some of these signs (plate warping and sedimentation) could be observed from the outside of the cells, while other indications (corrosion damage and brittleness) were discovered when the cells were disassembled after the seismic tests.

- All of the cells that were tested delivered uninterrupted power during repeated strong shaking. There was, however, significant loss of electrical capacity. Four of the six cells failed to deliver 80 percent of their rated capacity (the normal acceptance level for emergency power supply batteries) after being shaken to

a ZPA level of about 1.5 g. Two cells were shaken to a higher level (ZPA approximately 2 g) and both showed a considerable loss of capacity.

- Internal components of these cells had various types of age-related damage, including corrosion, plate swelling, cracking and blistering. The amount of sediment present was uncharacteristically large for lead-calcium alloy cells. (The fact that these cells showed a slight grid growth, may indicate a relationship to the presently accepted accelerated-aging methods for qualifying batteries.)

- There were no signs of damage to the jars during any of the tests. This includes tests where a cell was free to move in the test frame (and hence would be subject to impact loads) and tests with multicell racks.

Reference 3.1.9, NUREG/CR-4096 (SAND84-2629), covers the testing of 10-year old lead-calcium C&D LCU-13 cells from the North Anna Nuclear Power Station. The results of the test are as follows:

- The ten-year old C&D LCU-13 cells that were the subject of this study were in good condition. They were all able to survive repeated strong shaking without sudden failure. There was some loss of electrical capacity noted due to the seismic tests.

- Internal components of all of the cells that were disassembled appeared to be in good condition with little plate swelling, cracking or pitting. There was only a modest amount of sediment.

- Subsequent observation of the exposed cross-sections of the bus bars of two cells showed no signs of corrosion in the conductive area - the surfaces all had the characteristic metallic luster of lead. None of the positive plates showed signs of grid growth. IEEE 535-1979 states plate growth is the determining aging factor for subsequent battery qualification.

- There were no signs of damage to the jars during any of the tests. This includes test where a cell was free to move in the test frame (and hence would be subject to impact loads) and tests with multicell racks.

Reference 3.1.10, NUREG/CR-4097 (SAND84-2630), covers the testing of 27-year old lead-antimony Exide EMP-13 cells from the recently decommissioned Shippingport Atomic Power Station. The results of the test are as follows:

- The twenty-seven-year old Exide EMP-12 cells were in poor condition. They were all able to survive repeated strong shaking without sudden failures, but all had suffered from major losses in electrical capacity.

- Internal components of the cells that were disassembled were in poor condition with extensive corrosion and embrittlement. The seismic testing caused corrosion products and active plate material to break away from the plates and lead to a loss in capacity. However, this type of battery cell (modified Plante construction) would not see the typical plate grid growth commonly found in cells composed of flat pasted plates.

- There were no signs of damage to the jars during any of the tests. This includes tests with multicell racks.

International and Other U.S. Programs

We are not currently aware of any other significant related activities.

Issue Findings

Limitations of Overstress Approaches

There is the possibility that overstressing of one variable may account for the effect of a second variable, but the limitations of such an approach must be understood. In one instance, dose versus dose-rate, the findings within this subissue are:

1. Dose-rate effects can vary greatly in magnitude and direction for different materials.

2. Differences in materials and aging conditions can lead to differences in the degradation mechanism.

3. "Dose-Overstressing" (increased total dose) may not adequately account for dose-rate effects, thus consideration of dose-rate effects, aging conditions, and material differences is important before applying overstress in testing.

Multiple Sample Testing

Multiple sample testing may serve to complement type-testing to achieve a broader safety perpective. In summary then, the findings within this subissue are:

1. Multiple sample testing may reveal application-dependent failures.

2. Multiple sample testing may identify failures which are statistically significant, but not "common-mode."

3. Multiple sample testing may identify an equipment performance problem due to a manufacturing process that occurs in a random number of components.

Fragility Testing

Fragility testing of equipment has not been routinely used, as a supporting basis of data, in equipment qualification programs. The potential benefit of fragility testing is in identifying failure thresholds and margins in Class 1E equipment. However, fragility testing may subject test specimens to racheting effects that are produced by the differences in the "transient" application of the test environment versus the "real" environment.

Based on the battery seismic-fragility tests presented here, the findings within this subissue are:

1. Plate-growth was not universally observed in the four types of naturally-aged battery cells tested, although this is the aging mechanism used in IEEE-535-1979.

2. For the high g-levels used in the battery fragility tests, the principal (catastrophic) failure mode was fracture of the bad post/bus-bar interface.

3. Lead post/bus-bar corrosion was observed in three of four battery types evaluated.

References

Limitations of Overstress Approaches

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3.2 TMI-2 Experiences

Issue Description

The TMI-2 accident provides a unique opportunity to evaluate equipment qualification issues. In general, three categories of information can be identified: equipment response and survivability; fission products distribution and source term; and the hydrogen burn environment. Numerous research activities are underway in each of these areas sponsored by many national and international organizations. The combined knowledge and its application in future equipment qualification programs should provide increased assurance of safety-related equipment adequacy.

Regulatory Guide 1.89 (June 1984) recognizes that future TMI-2 findings may impact the selection of the source term (see Paragraph B and Appendix D, for example).

This is the thrust of this effort to address the issue:

> "What can we learn from the TMI-2 accident and research regarding simulation of accident conditions?"

Terminal Block Study

Reference 3.2.1, NUREG/CR-1682 (SAND80-1957), discusses a study of terminal blocks (TB) used in TMI-2 and tested to TMI-2-like accident environments. (Some indirect evidence exists to suggest that such blocks were replaced with in-line splices just prior to the accident.) Under high humidity and in the presence of contaminants, the TBs exhibited leakage currents, and even at relatively low TMI-2 temperatures, occasional electrical breakdown (Figure 3.2.1). The estimated probability of breakdown versus temperature is shown in Figure 3.2.2; for the TMI-2 accident temperature (about 86°C), the breakdown probability for a protected block is on the order of 1 percent or less. The results of the study with respect to accident simulations are:

1. The TB materials of construction are not critical to electrical breakdown phenomena, compared to surface conditions.

2. The design of the blocks, their mounting arrangement, their function (e.g., voltage and currents) and their protection arrangement (e.g., boxes, weepholes) are important parameters in electrical breakdown.

3. Under high humidity and/or contaminant conditions, even relatively low temperature conditions can lead to electrical breakdown.

TMI-2 Data Evaluation

Available TMI-2 data continues to be evaluated, under a contract with IRT Corporation. Reference 3.2.2 (IRT 6597-001) includes reviews of over 170

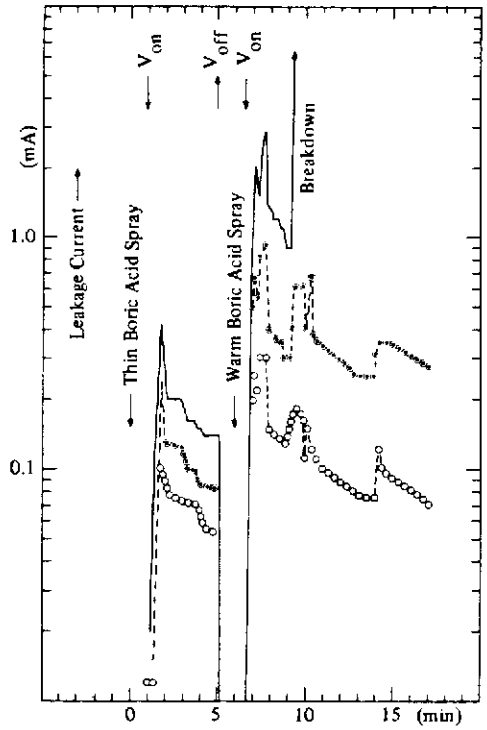

Figure 3.2.1 Leakage currents to ground versus time. Three neighboring contacts on the same block are shown. The three currents fluctuate proportionally with contaminants and environment. A complete breakdown on one connector occurs at 9 minutes. (From Reference 3.2.1)

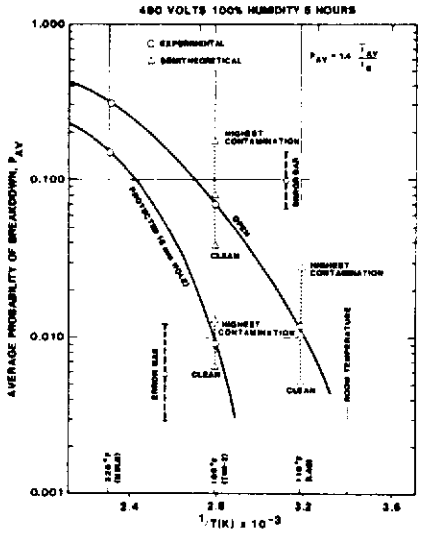

Figure 3.2.2 Breakdown probability after 5 hours exposure to steam and contaminants. Upper curve-open blocks, lower curve-blocks in enclosure with 6 mm weephole. Averages for extreme contaminant levels are indicated. "Highest contamination" means steam saturated with conductive spray. (From Reference 3.2.1)

TMI-2 related reports (through April 1983); it was hoped that the accident would serve the purpose of increasing the equipment qualification information data base.

Forty-four reports related specifically to equipment evaluations (e.g., radiation monitors, transmitters, amplifiers); most were not Class 1E devices. Some of the findings include:

- Cracked GM tube (HP-R-213 area radiation monitor), possibly due to mechanical shock from the hydrogen burn that occurred about 10 hours after the accident began.

- Failed output transistor (HP-R-211), caused by a momentary short of the 600 V power line to the signal line due, perhaps, to water intrusion into an improperly mated connector.

- Multivalued signal output indications by area radiation monitors due to impedance mismatches and radiation-degraded transistors.

- Instrument failure due to radiation-degraded MOS field effect transistors.

- Problems due to wet cables, corrosion, and moisture intrusion.

- Common mode failures (e.g., many instruments that were wired into the same terminal block or were located around the same reactor coolant pump failed or did not yield a proper output).

Fifty-three reports related specifically to fission products. At that time, the best estimates of releases were as shown in Table 3.2.1.

Table 3.2.1
Summary of Activity Release From Fuel on March 28, 1979
(From Reference 3.2.2)

Nuclide	Percent Release
Kr-85	71
Xe-131M	70
Xe-133	68
I-131	59
Cs-134	76
Cs-136	57
Cs-137	60
Sr-89	<0.01
Sr-90	<0.07

Reference 3.2.2 recommends that:

1. A comprehensive mass balance of the fission products be made.

2. A full mapping of the TMI-2 containment building be made to help analysts verify existing transport codes and fission product release behavior.

3. The raw release data should be analyzed and correlated to predict the time-dependent fission product releases. (This would be an important contribution to equipment qualification as it would allow estimation of radiation dose levels to equipment...).

Nineteen reports related specifically to the hydrogen burn phenomenon. Reference 3.2.2 concluded that:

1. Such an event could have serious implications for equipment not properly designed to withstand the potential mechanical shock and thermal transients.

2. Determining the characteristics and behavior of the TMI hydrogen burn is perhaps the most important issue; physical inspection of more TMI-2 components is required to assess the effects of the burn.

3. Results of the Hydrogen Burn Survival Program at Sandia indicate that "fully exposed" components may experience temperatures above current LOCA qualification guidelines; however, the local environment may provide sufficient thermal protection to assure survival. Thermal analyses appear necessary to determine the response of a given component to a hydrogen burn environment, but major redesign will probably not be necessary.

The reports collection effort continues, Reference. 3.2.3. As of October 1, 1984, the TMI-2 Related Reports Collection contains 261 entries.

International and Other U.S. Programs

The reports, summarized below, provide information in three areas: equipment evaluation, radiation levels, and the hydrogen burn environment.

In Reference 3.2.4 (SAND84-1520), terminal blocks and splices were exposed to beta radiation at levels expected at TMI-2. Several techniques including FTIR spectroscopy, RAMAN spectroscopy, visual observation, and density profiling were used in an effort to detect beta radiation surface damage. These techniques could not detect any surface damage at doses less than or equal to 10 Mrad, i.e., doses comparable to those from the TMI-2 accident environment.

Reference 3.2.5 (GEND-INF-069) reports on an examination of TMI-2 cable for physical abuse, effects of radiation, and effects of the hydrogen burn.

Physical abuse: The portion of the polar crane cable lying on the D-ring showed clear signs of damage from maintenance. This section of cable had cuts, abrasions, and even protective sleeves covering the cable jacket which had apparently been used to reseal a severely damaged section of cable.

Effects of radiation: The radioactive contamination present on the portion of polar crane cable lying horizontally on the D-ring apparently did not cause any dramatic changes in the material or electrical properties of the cable when compared to a contiguous section of the same cable with a vertical orientation even though the contamination was approximately 10 times greater. Therefore, the polar crane pendant cable permitted an evaluation of "high" versus "low" radioactive contamination of a cable sample as well as a comparison of thermal effects (char, discoloration, normal).

Effects of the hydrogen burn. The several findings included:

- Char pattern indicated the direction of the burn front.

- Heat flux tests on the cable jacket demonstrated that the char was caused by the large volume of gases resulting from the burn and not caused by the burn front itself.

- Char pattern indicated that heat sinks and shielding played an important role in determining if the cable survived.

- Little or no changes in electrical and material properties of the polar crane cable could be observed in representative cable samples taken from each region of polar crane cable. The charred region of the cable actually ignited and burned producing a char on the jacket surface and yet the mechanical and electrical properties of this cable region were not greatly different from those of the undamaged region.

Reference 3.2.6 (GEND-INF-072) reports on TMI-2 cable sections, connected to the HP-R-214 Dome Monitor, that were removed after the accident. The testing showed:

1. No detectable difference between cable in conduit or out of conduit.

2. No damage to the cable compared to a virgin sample.

Reference 3.2.7, (SAND84-0559), reports on analyses of the dome radiation monitor at TMI-2; this was the only instrument inside containment capable of measuring the high radiation levels which might be present during a loss-of-coolant accident (LOCA). The detector failure modes included moisture intrusion into the electronics package, DC feedback in the preamplifier, MOS transistor degradation, and electrolytic capacitor failure.

The examination results strongly suggested the following design changes to improve high level radiation monitoring.

1. Fabricate the detector to be more nearly hermetically sealed (to avoid moisture intrusion).

2. Do not use the detector inside a thick, lead-shielded vessel since it is impossible to predict levels outside such a shield.

3. Do not use MOS transistors or MOS integrated circuits in any application were radiation exposure is a possibility.

4. Use military grade, or better, components in the electronics package. Mil Standard 883 Class B components should be sufficient for this application.

Using transistor current gain degradation and elastomeric material degradation properties, the total gamma radiation dose received by the Dome Monitor (HP-R-214) electronics inside the stainless steel vessel and the dose to the multiconductor cable outside the vessel was estimated. Table 3.2.2 lists the doses received by the radiation detectors which have been analyzed at SNL.

Table 3.2.2
Total Estimated Gamma Radiation Doses Received by TMI-2 Radiation Detectors (From Reference 3.2.7)

Containment Elevation (Feet)	Instrument	Dose (rads)
305	HP-R-211	2.5 x 10E5
305	HP-R-212	4.5 x 10E5
347	HP-R-213	9.9 x 10E5
372	HP-R-214 Cable	7.9 x 10E6
372	HP-R-214 Detector	2.2 x 10E5

The original Dome Monitor stripchart recording was found to be erroneous because the output was plotted on five decade log paper rather than on eight decade paper and the recorder was improperly scaled. Corrections have been applied to the Dome Monitor output. During the first 800 hours of the accident, the available data supports two hypotheses regarding accuracy. Hypothesis 1 proposes that the Dome Monitor was relatively accurate even though some moisture had probably entered the detector

housing. Hypothesis 2 says that the monitor was inaccurate for substantially all of the accident because of moisture intrusion into the detector as early as 7:00 a.m. on the day of the accident. Gamma radiation rates as a function of time both inside the vessel shield and in the outside containment atmosphere are estimated in Figure 3.2.3. This figure is based on assuming Hypothesis 2 to be correct. This time history is based on total gamma dose measurements only.

A particularly active program has been the GEND effort, and several reports are summarized here.

Reference 3.2.8 (GEND-INF-056) summarizes results of diagnostic tests conducted on selected cable channels within the TMI-2 Reactor Building. A total of 233 TMI-2 Reactor Building cables were tested in situ; anomalous electrical behavior was observed in 103 (44%) of these cables. Of these, 57 cables (24%) contained circuits classified as inoperable. Table 3.2.3 presents a summary of inoperable cables, by penetration, and also includes some general environmental conditions.

In Reference 3.2.9 (GEND-INF-010), a short length of Raychem Flamtrol cable, which was connected to the HP-RT-211 radiation detector from the TMI-2 containment building, was examined for electrical and mechanical degradation. The ultimate tensile strength, the percent elongation at break, and the insulation resistance were measured. All three techniques yielded similar results for the TMI cable and for a comparison "virgin" specimen; no damage could be detected.

Reference 3.2.10 (GEND-INF-041) reports on two pressure switches (NM-PS-1454 and NM-PS-4174) used to actuate alarms in the TMI-2 Nitrogen Manifold System that were selected for investigation. They were subjected to in situ electrical tests, removal, and detailed examination. One unit exhibited no anomalies. The other unit exhibited an anomaly during in situ testing but appeared to have been mishandled. Both units showed evidence that contaminated air entered the switch enclosure, and the second unit also showed evidence of a significant incursion of moisture and corrosion, possibly from hot water spraying during the Reactor Building gross decontamination experiment. (Therefore, it is recommended that instruments such as these pressure switches be carefully sealed when installed to prevent the intrusion of moisture.)

The most common finding at TMI-2 appears to be moisture intrusion into sealed housings. Reference 3.2.11 (GEND-INF-029) is a typical reference. Two Foxboro transmitters were reported as operational, while two Bailey transmitters "...failed as a result of water damage to the signal conditioning electronics located inside the transmitter housing."

Reference 3.2.12 (GEND-INF-058) summarizes the results of a test program designed to resolve whether or not pressure transmitters located on the core flood tanks at TMI-2 performed within their calibration limits during the periods of high radiation. Two Foxboro transmitters and one Bailey transmitter, each with associated cabling, were subjected to

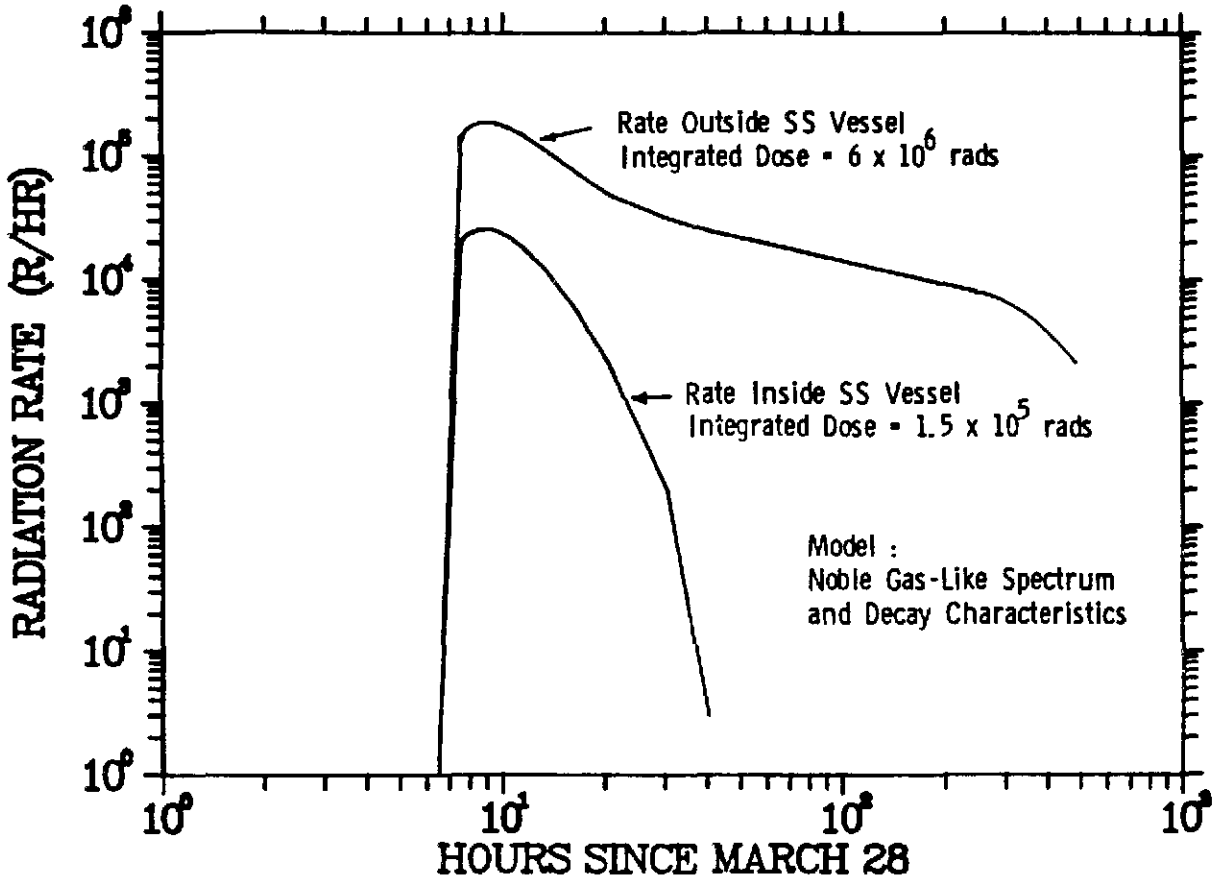

Figure 3.2.3 TMI-2 gamma radiation time history. This is our best estimate of radiation history both inside and outside the stainless steel vessel (Hypothesis 2). Noble gas spectrum and decay characteristics were assumed in order to calculate the level outside the vessel. (From Reference 3.2.7)

Table 3.2.3
Penetration Environment and Damage Summary
(From Reference 3.2.8)

Penetration Section	Penetration Elevation (ft)	Irradiation (R/h)	Cable References to Water			Inoperable (%)
			Number of Cables	Cables Marginal[a]	Cables Below[b] Water	
R400, R402, R407	292	20 to 50	117	117	0	4.3
R405	292	50 to 1000	5	4	1	80.0
R406	292	20 to 50	6	4	2	16.7
R504	323	20	10	4	0	30.0
R505	319	20	10	3	0	10.0
R506	323	20	19	4	0	31.6
R534	300	20	14	6	0	35.7
R607	292	20 to 50	52	38	14	61.5

a. Cables which were partially above and below water level.

b. Peak water level 292-ft elevation.

irradiation. The irradiation test was designed to simulate, as closely as possible, the accident conditions experienced by the actual transmitter at TMI. The three transmitters functioned without failure when subjected to a total gamma dose of 1×10^7 rads. The Bailey Meter Co. transmitter exhibited a definite sensitivity to total integrated dose with a 4 percent zero shift at 5.5×10^4 rads and increasing to 16 percent at 5.4×10^6 rads. The visual inspection revealed that insulation on the transmitter's internal wiring had become brittle. The Foxboro units had negligible shift. In comparing the response time data from before and after the irradiation for the three transmitters, a degradation of about 15 percent was noted for each transmitters.

In Reference 3.2.13 (GEND-INF-023 Vol. VI), conclusions on the hydrogen burn were made, including:

1. Hydrogen concentration in the reactor building prior to burn was confirmed to be about 8 percent, as calculated by analyses of TMI-2 pressure and temperature records.

2. The most probable ignition site for the hydrogen burn was in the basement volume; radial location is not defined.

3. Thermal degradation of most susceptible materials on all levels is consistent with direct flame contact from hydrogen fire.

4. Polar-crane pendant cable thermal damage indicates intense exposure to a hydrogen-fire plume.

5. Burn patterns in the reactor building indicate that the dome region above the 406-ft level was uniformly exposed to direct hydrogen flame, the region between the 406-ft level and the top of the D-shield was partially exposed to hydrogen flame (most likely in the south and east quadrants), and the damage on the 305-ft level was geometrically similar to that above the 347-ft level but of less degree.

These are only representative samples of the GEND effort, and further information is available from TI & EP, P.O. Box 88, Middletown, PA 17057.

Issue Findings

Terminal Block Study

In general, industry practice has been to remove terminal blocks from safety-related circuits based on the results of this study and other industry experiences. The most important research finding is that terminal blocks will electrically break down under severe and sometimes even mild (TMI-2-like) conditions with some probability.

TMI-2 Data Evaluation

From an equipment perspective: (1) it appears that installation and/or maintenance practices are very critical to equipment survivability particularly due to moisture intrusion; and (2) some effect of mechanical shock and (3) thermal transients from a hydrogen burn should be considered.

International and Other U.S. Programs

From a safety and qualification standpoint, industry research has considered the survivability of components in accidents. These include findings on component performance, recommended practices, radiation levels, hydrogen burn conditions, and the TMI-2 accident.

In summary then:

1. Maintenance activities may compromise the functionability of components. [Reference 3.2.5]

2. Components exposed to a hydrogen burn may be charred by the large volume of gasses resulting from the burn rather than the burn front itself. [Reference 3.2.5]

3. Heat sinks and shielding may determine component survival during a hydrogen burn. [Reference 3.2.5]

4. For the TMI-2 dome monitor cable, no difference was found between cable in-conduit or out-of-conduit. [Reference 3.2.6]

5. The dome radiation monitor, the only instrument inside containment capable of measuring high radiation levels, failed during the TMI-2 accident. [Reference 3.2.7]

6. Radiation doses received by TMI-2 equipment were estimated to be between 2.2×10^5 and 7.9×10^6 rads. [Reference 3.2.7]

7. From in situ tests, 44 percent of the cables tested had anomalous electrical behavior (of these, 24 percent were inoperable). [Reference 3.2.8]

8. Sensitive devices should be carefully sealed when installed to prevent the intrusion of moisture. [Reference 3.2.10]

9. Some transmitters are sensitive to irradiation, causing output shifts. [Reference 3.2.12]

10. Hydrogen burns conditions at TMI-2: (1) hydrogen concentration was 8 percent, (2) ignition site was the basement volume, and (3) no defined path was found for the hydrogen propagation. [Reference 3.2.13]

References

Terminal Block Study

3.2.1 O. M. Stuetzer, Electrical Insulators in a Reactor Accident Environment, NUREG/CR-1682, SAND80-1957, Sandia Laboratories, Albuquerque, New Mexico, January 1981.

A. O. M. Stuetzer, Preliminary Examination of TMI-2 Equipment: Terminal Blocks, SAND80-2447A. Presented at the USNRC Eighth WRSR Information Meeting, Gaithersburg, Maryland, October 27-31, 1980.

TMI-2 Data Evaluation

3.2.2 D. D. Thayer, TMI-2 Literature Search and Review With Application to Equipment Qualification Issues, IRT 6597-001, June 1983. Prepared by IRT Corporation under Contract 37-4505 to Sandia Laboratories, Albuquerque.

3.2.3 Letter Bonzon (SNLA) to Farmer (NRC), dated January 25, 1984, Status Report on IRT Efforts on TMI-2 Evaluations, with four attachments.

Attachment 1: IRT 6597-001, June 1983.
Attachment 2: Suggested Statements of Work
Attachment 3: Attempt to Model the Fission Product Releases from Three Mile Island Unit 2, IRT Report, January 12, 1984
Attachment 4: TMI Reports List

International and Other U.S. Programs

3.2.4 P. R. Bennett, Progress Report on the TMI-2 Equipment Examination Program: Beta Radiation Damage Assessment, SAND84-1520, Sandia National Laboratories, Albuquerque, New Mexico, March 1985.

3.2.5 R. Trujillo, et al, Analysis of the Polar Crane Pendant Cable from Three Mile Island-Unit 2, GEND-INF-069, Sandia National Laboratories, Albuquerque, New Mexico, January 1986.

3.2.6 E. H. Richards and V. J. Dandini, Analysis and Testing of the HP-R-214 Dome Monitor Cable From Three Mile Island Unit 2, GEND-INF-072, Sandia National Laboratories, Albuquerque, New Mexico, March 1986.

3.2.7 M. B. Murphy, et al, Analysis of the TMI-2 Dome Radiation Monitor, SAND84-0559, Sandia National Laboratories, Albuquerque, New Mexico, January 1985.

3.2.8 H. J. Helbert, et al, TMI-2 Cable/Connections Program FY-84 Status Report, GEND-INF-056, EG&G Idaho, Inc., Idaho Falls, Idaho, September 1984.

3.2.9 V. J. Dandini and L. D. Bustard, HP-RT-211 Cable Analysis, GEND-INF-010, EG&G Idaho, Inc., Idaho Falls, Idaho, September 1981.

3.2.10 J. A. Gannon, Evaluation of TMI-2 Pressure Switches NM-PS-145A and NM-PS-4174, GEND-INF-041, EG&G Idaho, Inc., Idaho Falls, Idaho, November 1983.

3.2.11 M. E. Yancey and R. C. Strahm, TMI-2 Pressure Transmitter Examination and Evaluation, GEND-INF-029, Vol. II, EG&G Idaho, Inc., Idaho Falls, Idaho, April 1984.

3.2.12 M. E. Yancey, Irradiation Test Report--Foxboro E11GM, Bailey BY3X31A, and Flame Retardant Ethylene Propylene Instrumentation Cable, GEND-INF-058, EG&G Idaho, Inc., Idaho Falls, Idaho, August 1984.

3.2.13 N. J. Alvares, Assessment of Extent and Degree of Thermal Damage to Polymeric Materials in the Three Mile Island Unit 2 Reactor Building, GEND-INF-023, Vol. VI, Lawrence Livermore National Laboratory, Livermore, California, February 1984.

3.3 Evaluation of Qualification Procedures for Specific Equipment Types

Issue Description

The USNRC sponsors a number of programs at Sandia National Laboratories (SNL) specifically addressing safety-related equipment survivability. The major thrust of these programs has been the physical testing of equipment. Test results illustrate the importance of a dedicated equipment design effort giving particular attention to the safety implications of the equipment operation. Several equipment survivability tests here have revealed equipment design and test-related deficiencies. Additionally, accident-simulation testing of hardware provides actual, physical "proof" of equipment survivability and functionability; whereas testing of materials serves to indicate "trends" of behavior, and may give insight to (reasonable) test approaches that could be applied for more complex components.

These test programs should be continued in order to bring about a final, conclusive resolution of the various issues being addressed that have not yet been closed. This situation exists due to several factors, among which are: (1) the evolutionary nature of equipment development, (2) the in-plant experiences of equipment degradation and/or failures, (3) the "accident" experiences of the industry, (4) the continuing research activities especially in the areas of operational and accident environment definition, and (5) the evolutionary nature of applicable standards, guides, and rulemakings.

This is the thrust of this effort to address the issue:

> "Are the procedures given in the IEEE standards for qualifying specific types of electric equipment adequate?"

Electric Cables

Reference 3.3.1, NUREG/CR-0275 (SAND78-0067), summarizes the results of nine LOCA typetests performed on a number of Class 1 components. Cables were one component included in each of the nine tests.

The primary purpose of this study was to determine, experimentally, the existence and magnitude of synergistic effects that result from the simultaneous application of simulated loss-of-coolant accident (LOCA) environments on Class 1 equipment. This evaluation involved performing two types of tests (a sequential and a simultaneous application of radiation in sequence or combination with the other LOCA environments) on typical and identical Class 1 equipment.

Test conditions employed are summarized in Figures 3.3.1 and 3.3.2. Saturated steam conditions existed throughout the accident simulations.

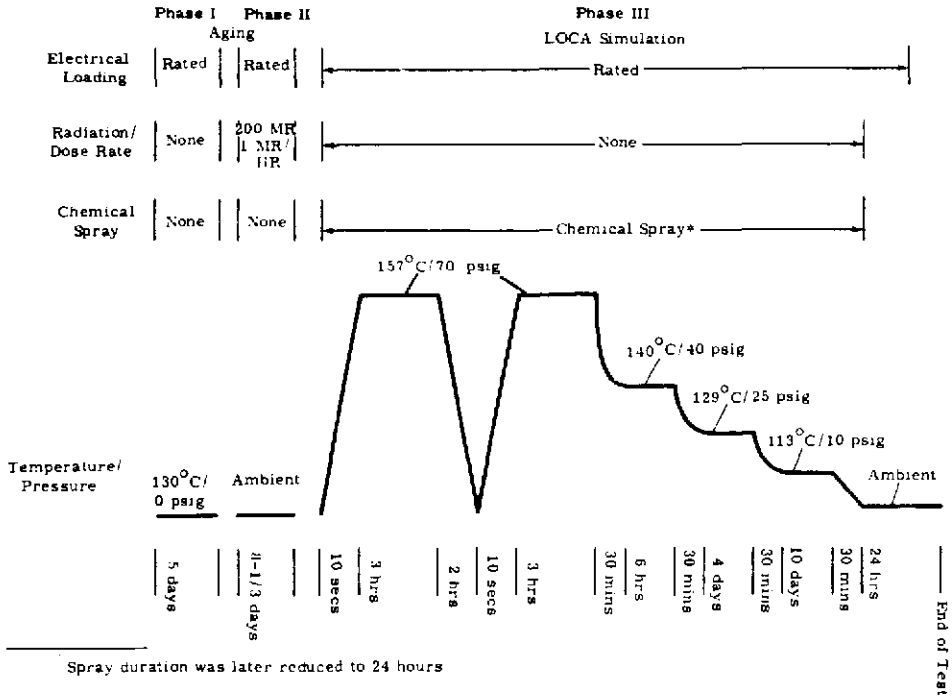

Figure 3.3.1 Sequential Test Profile (from Reference 3.3.1)

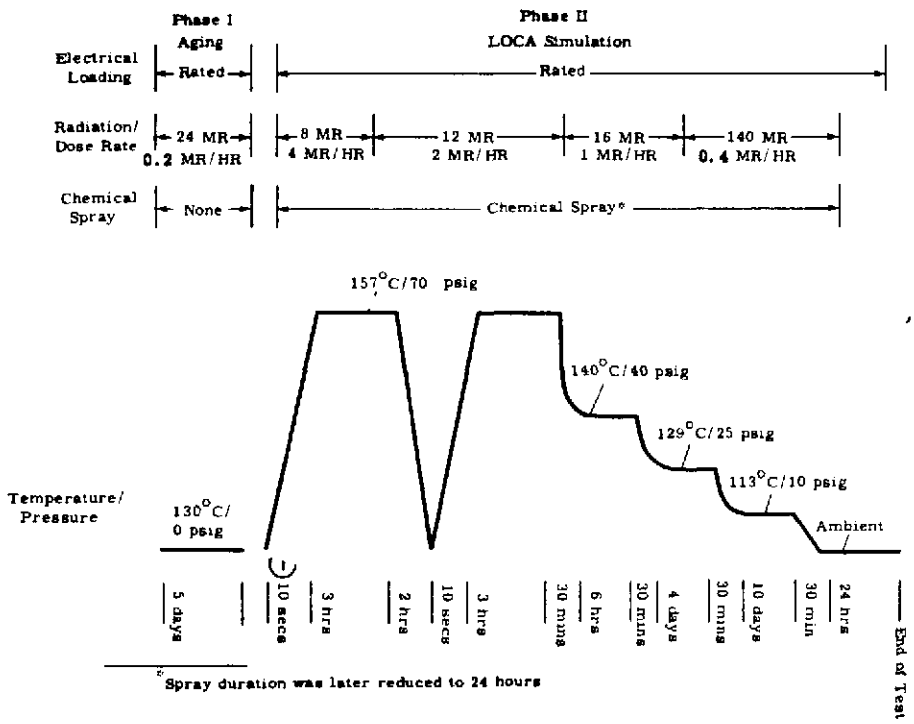

Figure 3.3.2 Simultaneous Test Profile (from Reference 3.3.1)

All electrical cable and cable assemblies used No. 12 AWG wire in single- or three-conductor cable. Rated single-phase current/voltage loads were applied continuously during the test. Current loads were maintained at approximately 20 amperes (60 Hz); 600 volts were applied separately between the conductor(s) and electrical ground.

The significant results of the cable type tests included:

1. Failures of some single conductor cables were observed in the sequential tests. Some single-conductor cables apparently failed during one sequential test as was verified in posttest electrical measurements; however, similar cable, used in splices, did not fail during another sequential test. These failures are troublesome, but numerical statistics are too minimal to make definitive conclusions.

2. Failure criteria must be adequately established since failure can adversely affect related tests and equipment. Although "functionability" is the correct criterion, cable shrinkage, expansion, or cracking (i.e., failure) and/or cable material can affect the "functionability" of cable assemblies, such as connectors or splices.

3. Synergistic effects do not appear significant in LOCA type tests of electrical cable. This conclusion is based on these test data and the off-site experience of FIRL; this experience should be representative of general industry experience.

Reference 3.3.2, NUREG/CR-3538 (SAND83-1258), discusses the effects of aging and LOCA simulations (sequential and simultaneous) on the electrical and mechanical properties of seven commercial EPR materials. Both cables and separate tensile specimens were tested.

In addition to performing LOCA research tests on cables as received from the factory, single conductor cables were also tested; these conductors were obtained by carefully removing the multiconductor outer jacket and sheaths and then separating the individual conductors from each other.

There were four major goals for the EPR experimental program:

1. To experimentally determine whether qualification testing of EPR single conductors is more severe than EPR multiconductor testing.

2. To investigate if cable electrical performances and insulation mechanical properties are sensitive to whether simultaneous or sequential stress exposures are employed during simulations of aging and accident environments.

3. To monitor insulation moisture absorption and tensile properties to gain insight concerning mechanical property changes that may cause cable electrical degradation.

4. To determine if the results obtained for one EPR formulation could be applied to other EPR formulations, and/or to other organic materials in general.

The intended sequential and simultaneous accident test profiles are similar to the IEEE 323-1974, Appendix A profile, but also different in several respects, most notably:

1. After four days of steam exposure we interrupted the steam exposure to remove baskets containing tensile specimens.

2. A 104°C saturated steam exposure was used after four days until the end of the test.

3. Chemical spray was not applied during the exposure.

4. The transient ramps were not started at 60°C.

The conclusions reached as a result of this test were:

1. Future EPR cable qualification tests should not employ single conductor test specimens to establish qualification for multiconductors. Both jacket-insulation interaction effects and helicity of multiconductor geometries need to be considered in a qualification program.

2. EPR cable qualification tests should correlate test conditions to use conditions. An example is the test bend radius used for a qualification test and the minimum bend radius used during cable installation.

3. EPR qualification tests or analysis should not rely on referenced behavior of other different EPR products. We observed a large variation in EPR behavior; generic EPR response does not occur.

4. Some EPR qualification tests need not employ simultaneous thermal, radiation, and steam test conditions. However, one EPR multiconductor cable, under simultaneous testing techniques, produced more cable damage than under sequential procedures, as indicated in Figures 3.3.3 and 3.3.4.

Reference 3.3.3, NUREG/CR-3588 (SAND83-2406), (a companion report to Reference, 3.3.2) discusses the effects of aging and LOCA simulations (sequential and simultaneous) on the electrical and mechanical properties of three commercial XLPO materials. Both cables and tensile specimens were tested.

In addition to performing LOCA research tests on cables received from the factory, single conductor cables were also tested. These conductors were obtained by carefully removing the multiconductor outer jacket and sheaths and then separating the individual conductors from each other.

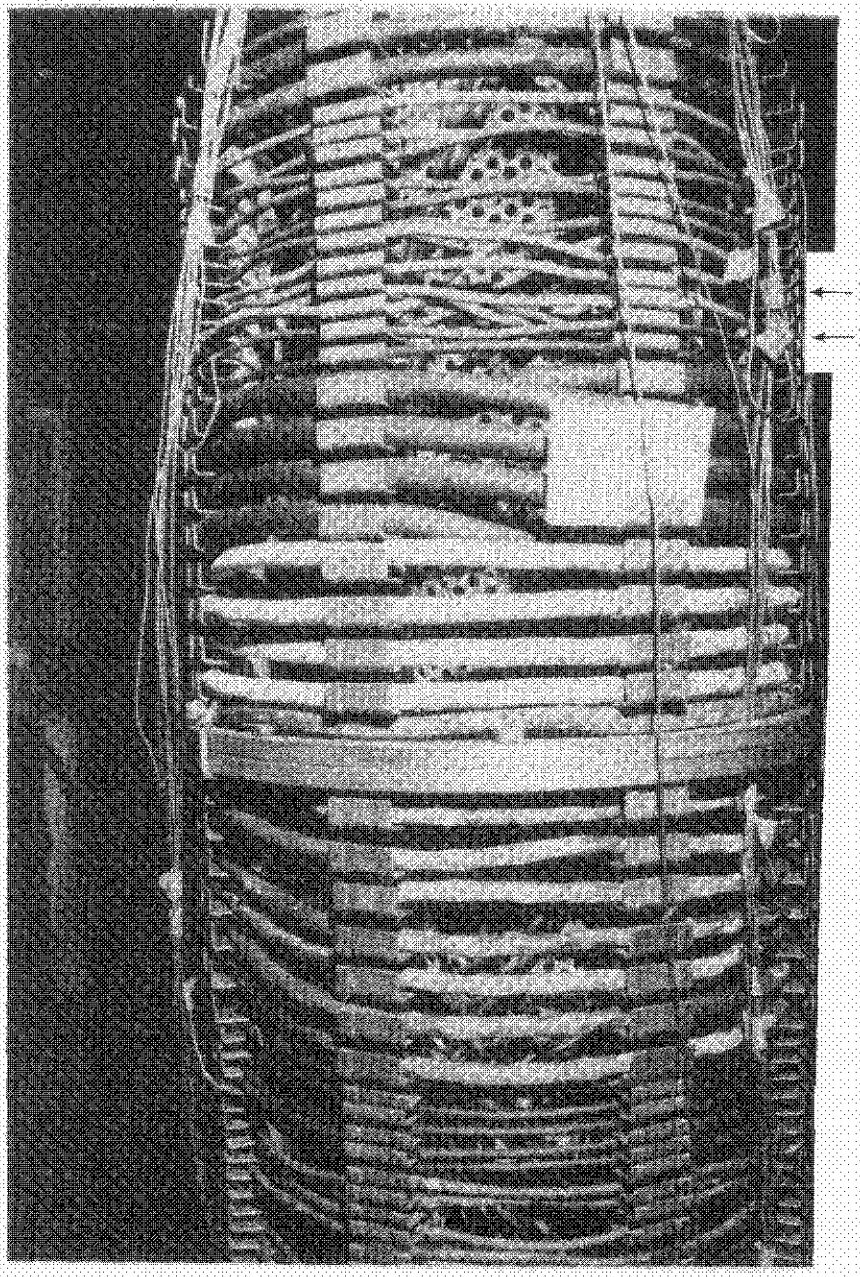

Figure 3.3.3 Cables at Completion of the Sequential Test. (From Reference 3.3.2)

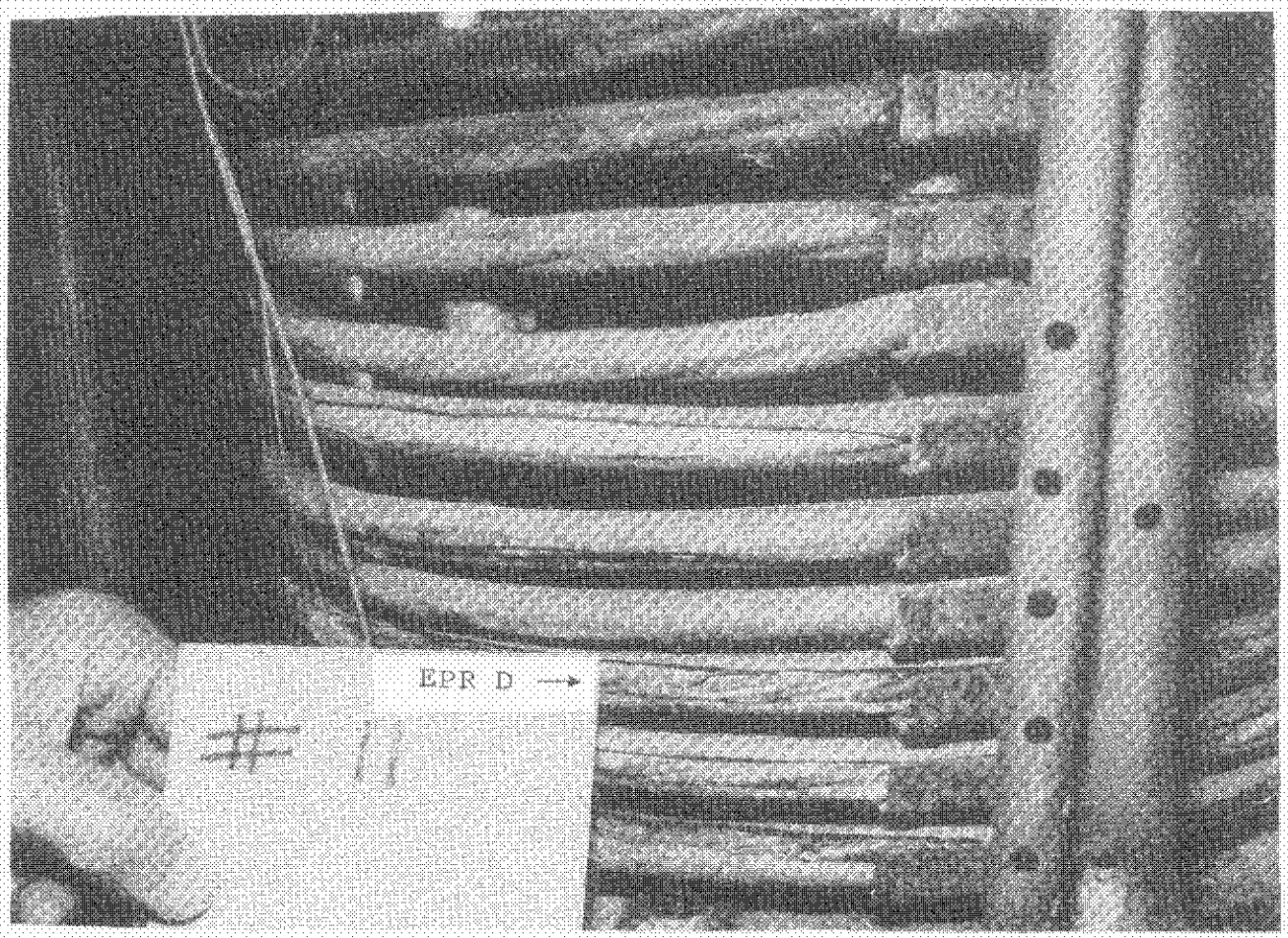

Figure 3.3.4 Cables on Day 18 of Simultaneous Test #2. (From Reference 3.3.2)

There were three major goals for the XLPO experimental program.

1. To experimentally determine whether qualification testing of XLPO single conductors is more severe than XLPO multiconductor testing.

2. To investigate if cable electrical performances and insulation mechanical properties are sensitive to whether simultaneous or sequential stress exposures are employed during simulations of aging and accident environments.

3. To monitor insulation moisture absorption and tensile properties to gain insight concerning mechanical property changes that may cause cable electrical degradation.

The intended sequential and simultaneous accident test profiles are similar to the IEEE 323-1974, Appendix A profile, but deviated in the same respects as noted in Reference 3.3.2.

The experimental results indicated:

1. The electrical properties for XLPO cable products did not depend on whether single conductor or multiconductor testing was performed.

2. Electrical performance of one XLPO cable type, during the simultaneous tests was similar to that achieved during the sequential test. For the other XLPO cable types, electrical performance during the simultaneous tests was similar to that achieved during the manufacturer's sequential tests.

3. Dimensional swelling for the XLPO insulations was not sufficient to cause mechanical degradation that affected electrical properties.

4. The XLPO multiconductor cable constructions included chlorosulfonated polyethylene (CSPE) and Neoprene outer jackets. Substantial visual degradation of these jacket materials was observed during the simultaneous testing exposures (see Figures 3.3.5 and 3.3.6).

Reference 3.3.4 describes the effect an intense superheat environment (340°F superheat) had on cable segments. Forty-two 12-inch long cable segments (21 types, from 9 manufacturers) were hung in a test chamber. Figure 3.3.7 shows the posttest condition of the cables along with a virgin sample (center position) for comparison. Most were badly damaged -- exposed conductors and carbonized jackets and insulation.

Since this was a screening test only, no data on damage thresholds for time or temperature or pressure is available. However, these tests suggest that equipment survival may be of very short duration. This has

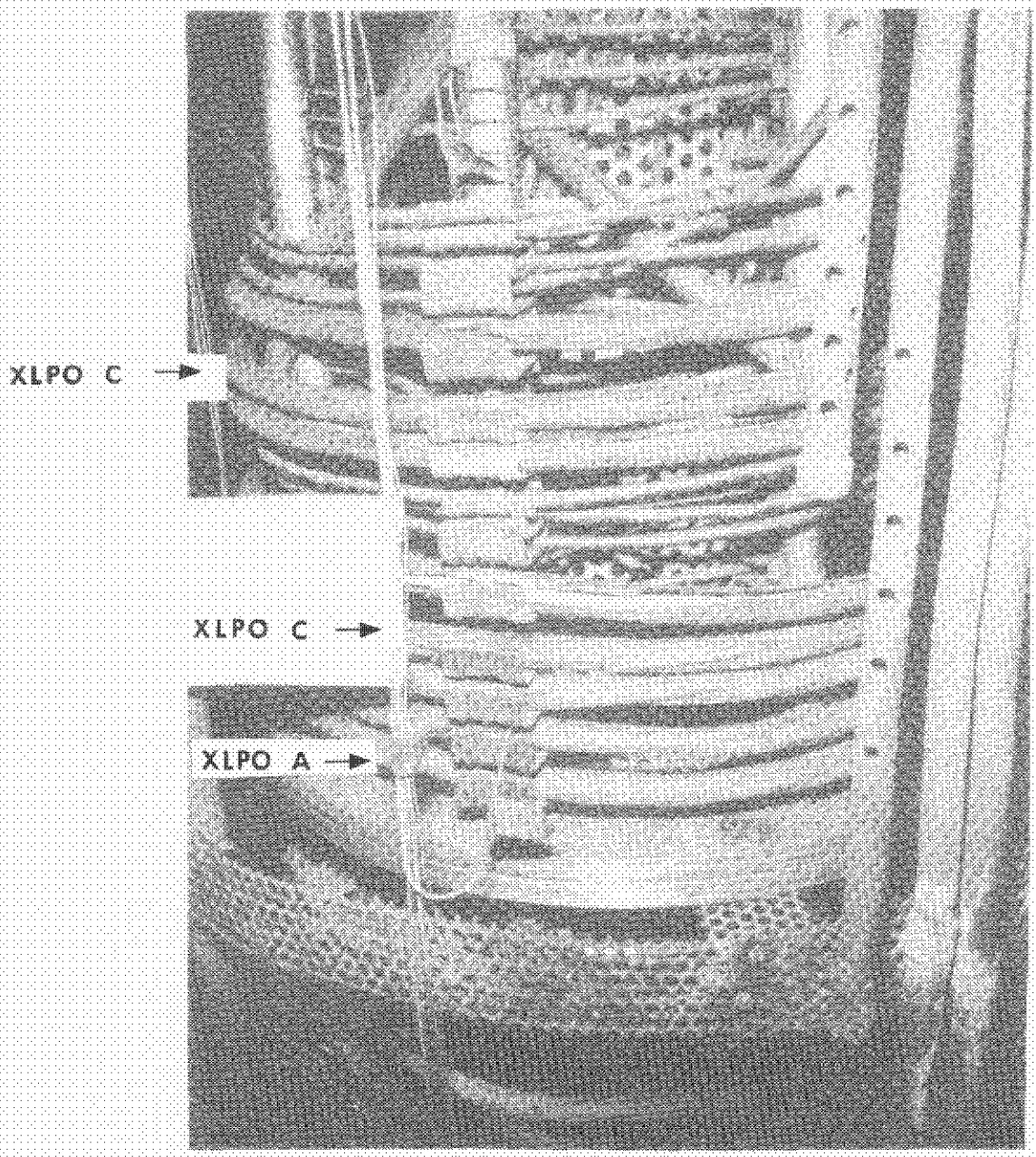

Figure 3.3.5 CSPE jacket condition (XLPO A and XLPO C cables) at completion of simultaneous test #2. (From Reference 3.3.3)

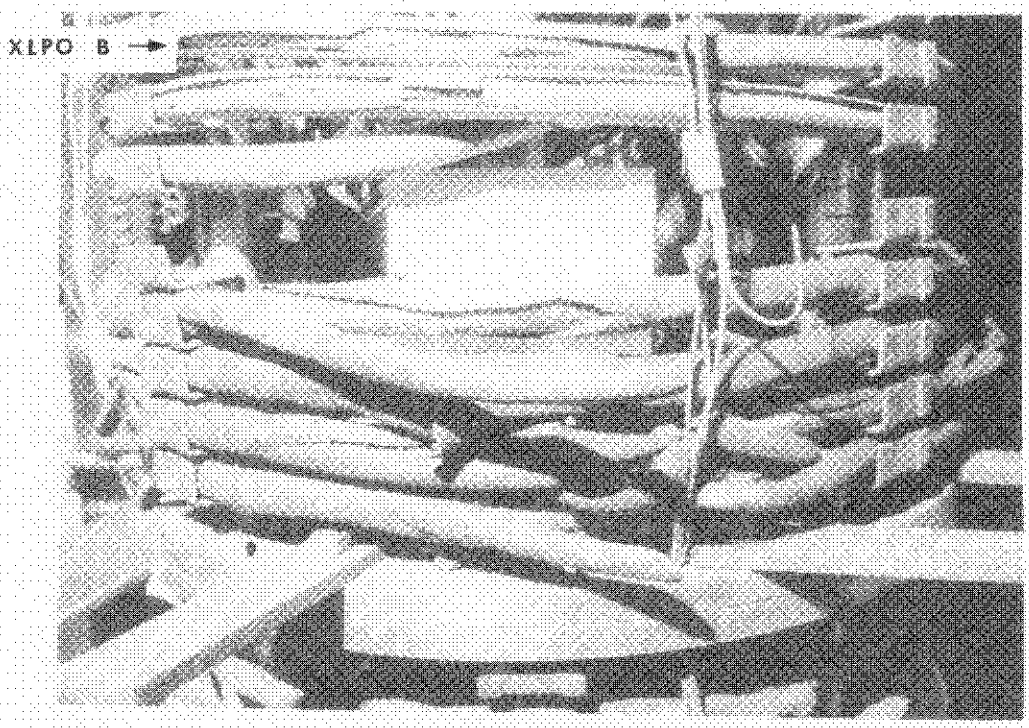

Figure 3.3.6 Neoprene jacket condition (XLPO B cable) at completion of simultaneous test #1. (From Reference 3.3.3)

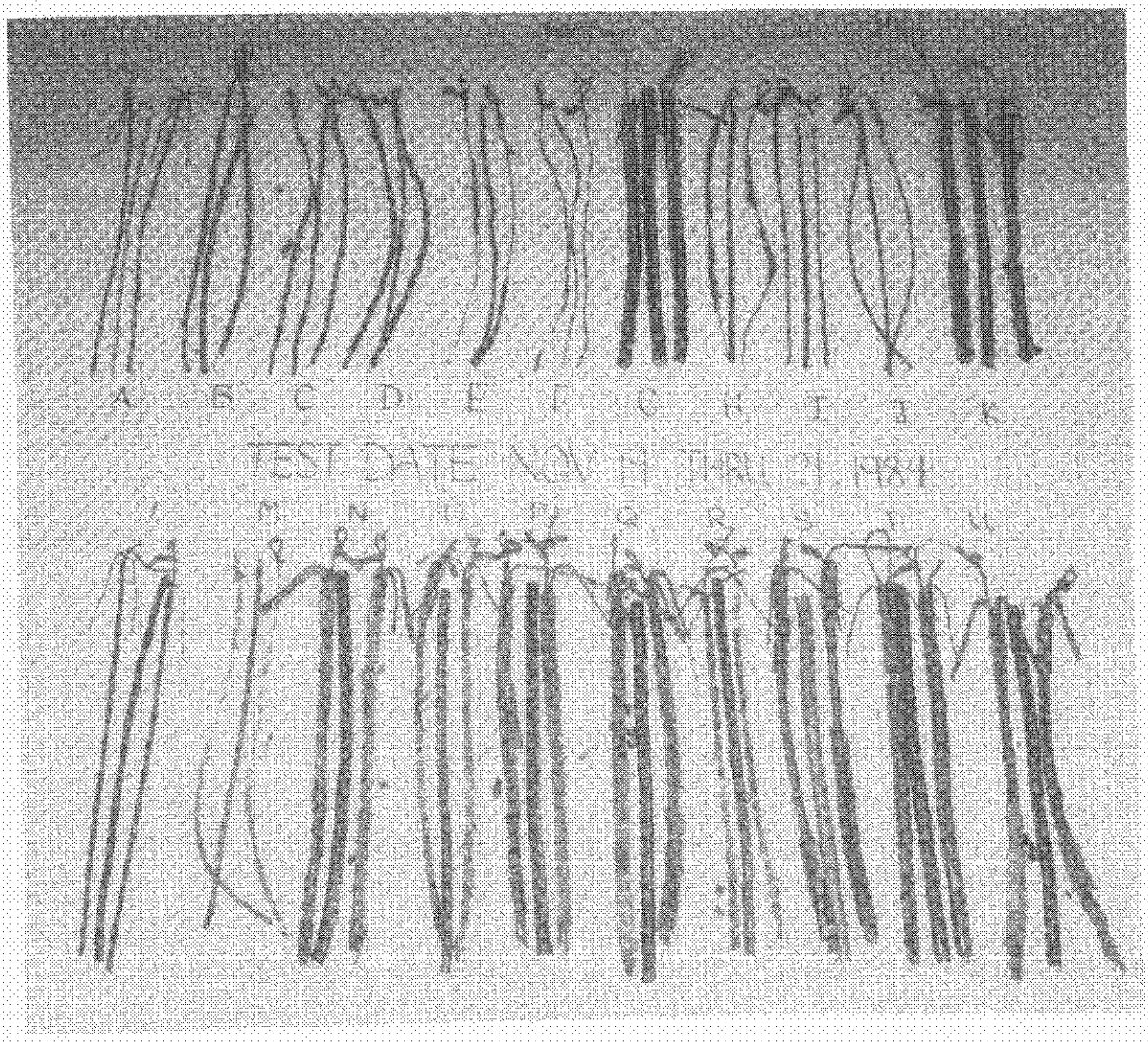

Figure 3.3.7 Posttest comparison of cables, exposed to 700°F/140 psig (340°F superheat) superheat steam, and virgin (center positions) samples. From Reference 3.3.4)

implications to equipment survivability specifically, but also to the concern for accident management through reliable in-containment instrumentation and data signals.

Reference 3.3.5, NUREG/CR-4536 (SAND86-0450), discusses the results of a superheat steam test conducted on three (different) EPR cable products. Unaged and simultaneously aged (282°F for 168 hours and 40 Mrads, 40-year equivalent) cable specimens were tested in a simultaneous radiation and steam environment. The cables were energized at 480V and 0.6A throughout the accident exposure, and IR measurements were made periodically during the exposure.

The report compares the results obtained to those reported for the previous cable test described in reference 3.3.2 in which similar EPR cable products were tested under saturated steam conditions. The cables were exposed to the same (IEEE 323-1974) temperature profiles in both tests (340°F, peak) but the chamber pressure was maintained at 77 psia for the two peaks and first (320°F) plateau during the superheat test. Cable specimens included single conductors and multiconducjtors in each test; however, only one cable set was the same for both tests. Despite this, reference 3.3.5 lists the following comparisons of results for the two cable tests:

(1) single conductor cables exhibited high IR values and low leakage currents in both tests; and

(2) both tests showed the following results for the multiconductor cables:

- the jacket had a longitudinal split;
- the exposed gap in the jacket was approximately 0.6 cm wide;
- bare conductors were visible;
- the circumference of the jacket increased during the test, although the increase was not large enough to contain the bundle of conductors;
- the cables had large leakage currents after one minute at the following voltages:

saturated steam		superheated steam	
600 V	180-750 mA	600 V	2 mA
		1200 V	5 mA
		1800 V	750 mA

- the IR values followed the same pattern throughout the tests except that in the superheated steam test the IR readings were below the instrument range at 19-21 days, whereas in the saturated steam test the readings were below the instrument range at approximately 8, 12, and 18 days into the accident.

Based on the results of both tests, reference 3.3.5 goes on to make the following general conclusions:

(1) Results from single conductor cable tests may not be conservative when trying to qualify multiconductor cables.

(2) Superheat appears to have little effect on cables other than slightly delaying the time to failure.

(3) The differences between the electrical degradation of single and multiconductor cables does not appear to be generic to all cables.

International and Other U.S. Programs

Electric Cables

Reference 3.3.6/1, NUREG/CR-2932/1 of 2 (SAND 81-2027/1 of 2), reports on a methodology test of nine seven-strand No. 6 AWG and nine seven-strand No. 12 AWG electric cables manufactured by The Rockbestos Company with chemically crosslinked polyolefin insulation containing factory-made center conductor splices and insulation repairs.

A companion report, Reference 3.3.6/2, NUREG/CR-2932/2 of 2 (SAND81-2027/2 of 2), discusses a methodology test of nine seven-strand No. 12 AWG electric cables manufactured by General Electric Company with chemically crosslinked polyolefin insulation containing factory-made center conductor splices and insulation repairs.

Two series of tests are described in the reports: (1) Tests of cable specimens exposed to aging and LOCA/MSLB environments to evaluate the methods for measuring cable performance and degradation given by IEEE 383-1974, and (2) tests of cable insulation samples to determine the most severe aging environment and exposure sequence.

Test measurements performed on cable specimens encompassed x-ray, visual inspection, electrical continuity, and insulation resistance measurements prior to and during the test sequence. Hipot and mandrel bend tests were conducted after the exposures. The cables were exposed in the following sequence:

1. Low dose rate radiation aging (62 krad/hr* for 50.3 Mrad)

2. Thermal aging [~846 hrs at 302°F (150°C)]

*Air equivalent dose

3. LOCA radiation (0.77 Mrad/hr* for 146 Mrad)

4. Steam and chemical spray exposure [33-day LOCA/MSLB profile (cables energized-480 VAC, 11 amperes)].

The Rockbestos cables maintained their current-carrying capability during and following the LOCA steam and chemical spray exposure. Insulation resistance measurements taken to evaluate insulation material degradation dropped from greater than 10^{12} ohms for virgin materials to approximately 5.5×10^5 ohms for No. 6 AWG cable and 1.5×10^6 ohms for No. 12 AWG cable (see Figure 3.3.8) during the initial accident profile pressure and temperature peaks. After one week into the accident profile, the insulation resistance increased to approximately 4.0×10^7 ohms and 1.0×10^8 ohms, respectively, and remained fairly constant through the remainder of the exposure.

The G.E. cables maintained their current-carrying capability during and following the LOCA steam and chemical spray exposure. Insulation resistance measurements taken to evaluate insulation material degradation dropped from greater than 10^{12} ohms for virgin materials to approximately 2.0×10^6 ohms (see Figure 3.3.9) during the initial accident profile pressure and temperature peaks. After one week into the accident profile, the insulation resistance increased to approximately 5.0×10^8 ohms and and remained fairly constant through the remainder of the exposure.

Insofar as other cable qualification tests are concerned, there are many (more than 1500 separate cable test summaries are listed in the January 1985 Equipment Qualification Data Bank Quarterly Report, prepared for EPRI by NUS Corp.). However, from the reports we have seen, the SNL experience, in general, parallels industry's experience in cable qualification testing.

Internationally, we are aware that several nations are performing qualification tests on cables, most notably the French, Japanese, and Germans. Again, the number of such test reports is very large, but also appears (in general) to parallel the SNL experience.

Issue Findings

Electric Cables

In summary, the findings within this subissue are:

1. Synergistic effects did not appear significant in one series of LOCA type tests of electric cable, but did appear significant for one multiconductor cable in another test.

*Air equivalent dose

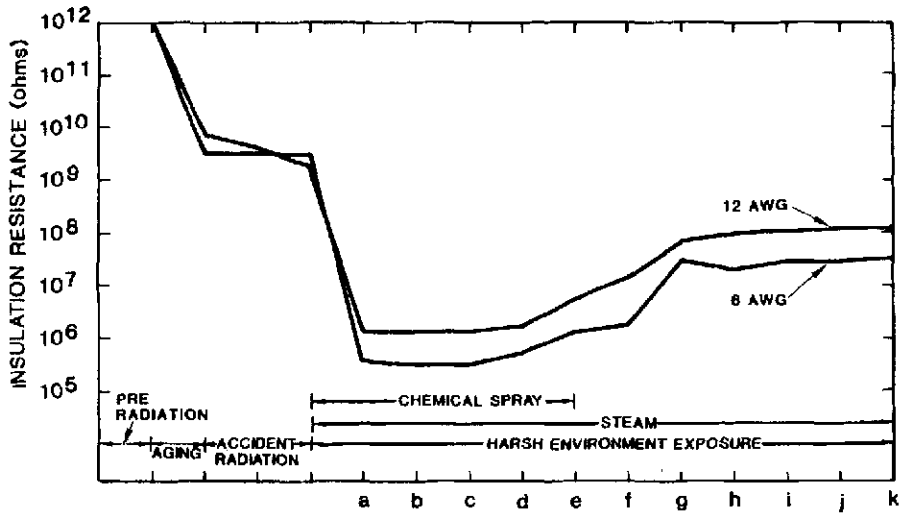

Figure 3.3.8 Insulation resistance profile. (From Reference 3.3.6/1)

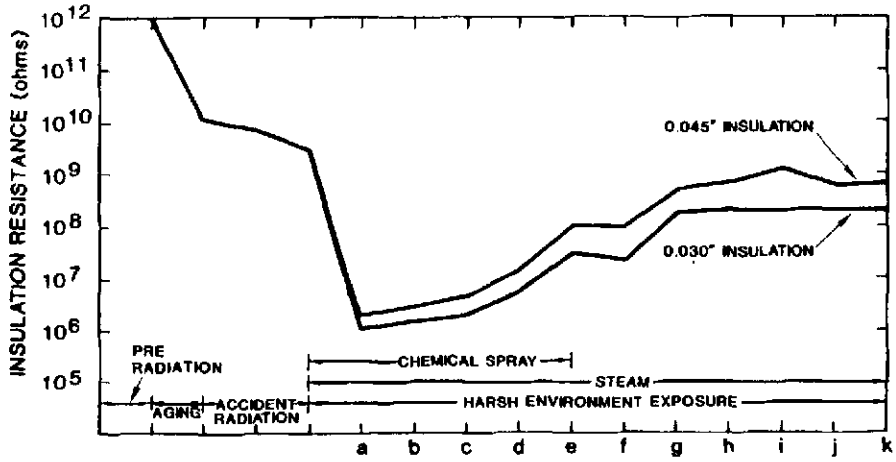

Figure 3.3.9 Insulation resistance profile. (From Reference 3.3.6/2)

2. Failure criteria must be adequately established, and should address the questions of cable "functionability" and "survivability."

3. Qualification tests should correlate test conditions to use conditions.

4. Electric cable qualification tests for one jacket-insulation formulation should not rely on referenced behavior of other different products.

5. Superheat appears to have little effect on some cables other than slightly delaying the time to failure.

References

Electric Cables

3.3.1 L. L. Bonzon, An Experimental Investigation of Synergisms in Class 1 Components Subjected to LOCA Type Tests, NUREG/CR-0275, SAND78-0067, Sandia Laboratories, Albuquerque, New Mexico, August 1978.

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3.3.4 Letter L. Bonzon (SNLA) to W. Farmer (NRC), dated December 3, 1984, Severe Accident Effects on Electrical Cable.

3.3.5 P. R. Bennett, S. D. St. Clair and T. W. Gilmore, Superheated-Steam Test of Ethylene Propylene Rubber Cables Using a Simultaneous Aging and Accident Environment, NUREG/CR-4536, SAND86-0450, Sandia National Laboratories, Albuquerque, New Mexico. (To be published.)

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Electric Cables, Other Programs

3.3.6/1 E. E. Minor and D. T. Furgal, Equipment Qualification Research Test of Electric Cable with Factory Splices and Insulation Rework, Test No. 2, Report No. 1, NUREG/CR-2932/1 of 2, SAND81-2027/1 of 2, Sandia Laboratories, Albuquerque, New Mexico, September 1982. (EQRT Program results).

3.3.6/2 E. E. Minor and D. T. Furgal, Equipment Qualification Research Test of Electric Cable with Factory Splices and Insulation Rework, Test No. 2, Report No. 2, NUREG/CR-2932/2 of 2, SAND81-2027/2 of 2, Sandia Laboratories, Albuquerque, New Mexico, September 1982. (EQRT Program results).

Cable Connector Assemblies

Reference 3.3.7, NUREG/CR-0275 (SAND78-0067), summarizes the results of nine LOCA type tests performed on a number of Class 1 components. Connectors were included as one component in three of the nine tests.

The primary purpose of this study was to determine, experimentally, the existence and magnitude of synergistic effects that results from the simultaneous application of simulated loss-of-coolant accident (LOCA) environments on Class 1 equipment. This evaluation involved performing two types of tests (a sequential and a simultaneous application of radiation in sequence or combination with the other LOCA environments) on typical and identical Class 1 equipment.

The test conditions employed were the same as for the electric cable tests (Figures 3.3.1 and 3.3.2).

To simulate normal usage conditions, the connectors were cycled during the aging and accident exposures. A cycle consisted of unmating, separating, remating and tightening the connector; 60 cycles was chosen as the expected lifetime value for connectors based on informal discussions with industry and certain proprietary test reports. Thirty of these 60 cycles were completed prior to thermal and/or radiation aging. The remaining 30 cycles were performed at natural breaks in the test profiles.

All cable connector assemblies used No. 12 AWG wire in three-conductor cable. Rated single-phase current/voltage loads were applied continuously during the test. Current loads were maintained at approximately 20 amperes (60 Hz); 600 volts were applied separately between the conductor(s) and electrical ground.

For at least a portion of each test, the 600-Vdc load was removed from each connector assembly because of an electrical failure of the assembly or another test component loaded on the same circuits. Electrically, at least one connector of each type failed and was unable to carry the rated voltage and current loads.

The significant conclusions reached as a result of the connector tests included:

1. Failures may be a function of the assembly, not just the connector.

2. No strong indications of synergistic effects were observed within the very limited test data.

Reference 3.3.8, NUREG/CR-1191 (SAND79-2311), describes a qualification verification test of six electrical connectors. Three each of two different types of electrical connectors were tested. Both types are manufactured by Bendix and used inside containment at Browns Ferry, Unit 3 (BF3).

The 250-Vdc connectors are equipped with 12 pins. The wiring configuration was such that all pins were used during testing. The 480-Vac connectors are equipped with 14 pins: 12 were wired into a single-phase circuit. In each case, the 12 pins were wired to form a series/parallel circuit that gave a voltage difference between pins within the connector as well as passing current to simulate a load.

The connector assemblies were exposed to Cobalt-60 radiation doses totalling 68.5 Mrads (air) at an averaged dose rate of 0.76 Mrad/h. All of the electrical connectors were then subjected to a two-part simulated aging test. The specified aging times/temperatures/tolerances consisted of 60 h in air at 217°F (intended to simulate 60 mo in air at 130°F) and 180 h in nitrogen at 217°F (intended to simulate 18 mo in 130°F nitrogen with a maximum of 4% oxygen). The humidity was not controlled (i.e., was ambient).

The electrical connector test assemblies were next subjected to the accident test profile shown in Figure 3.3.10. The 250-Vdc connector assemblies were energized throughout the entire test. The 480-Vac connector assemblies were energized during the first 5 minutes, between minutes 15 and 18, and between minutes 57 and 60.

All aspects of the test were considered satisfactorily completed. All connector assemblies passed the specifications of the test series; in particular, the assemblies carried their specified current/voltage loads during the accident-test phase. Posttest inspections indicated little or no degradation of the connectors.

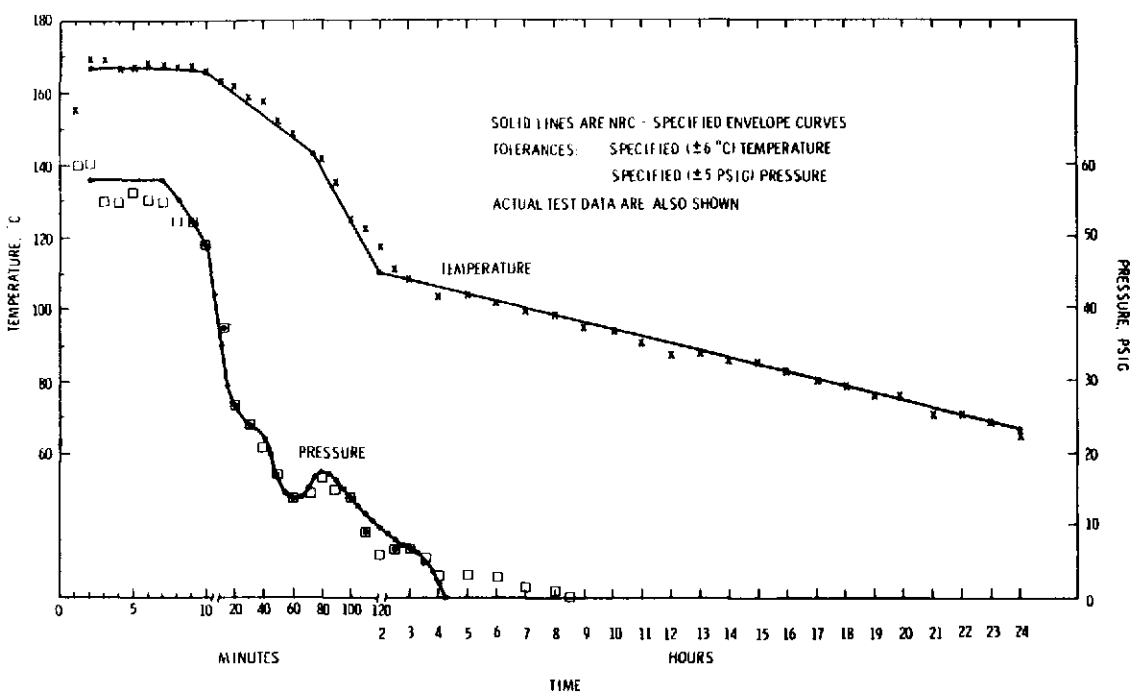

Figure 3.3.10 Specified BF3-connector accident test profile and data from November 1 and 2, 1979 test. (From Reference 3.3.8)

International and Other U.S. Programs

Cable Connector Assemblies

The SC-2 Qualification Subcommittee Working Group 2.11 of IEEE has been working on a draft standard (572-198x) which is intended to augment the recommendations given in IEEE Std 383-1974, by providing guidance for the qualification of connection assemblies.

Qualification of connectors is a continuing effort in the industry; however, we are not aware of any other significant national or international research efforts in this area.

Issue Findings

Cable Connector Assemblies

In summary then, the findings within this subissue are:

1. Failures may be a function of the assembly, not just the connector.

2. Connectors did not exhibit strong synergistic effects.

3. Mild superheat conditions did not appear to affect connector performance.

References

Cable Connector Assemblies

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Terminal Blocks

Reference 3.3.9, NUREG/CR-1682 (SAND80-1957), describes a comprehensive experimental program in which more than 600 terminal were exposed for an average of five hours to heat, steam, varying power cycles, contamination, different protective arrangements, etc. The conditions were selected to imitate the Three Mile Island accident (i.e., temperatures below 90°C, blocks made of phenolics and the contaminants most often boric acid and the standard containment spray).

The purpose of the work was a broad investigation of insulator deterioration under accident conditions to provide data for circuit analysis and to propose improvements in present and future installations.

All experiments were done under as high humidity as obtainable inside the chamber but outside the protective circuit boxes. The three kinds of phenolic blocks present in Three Mile Island Unit 2 were exposed. Voltage, temperature, and contamination were varied; the leakage currents to ground were measured over time spans from 4 to 6 hours. Three series of experiments were performed.

1. In preliminary measurements the temperature rose to about 43°C (110°F) in about 20 minutes. Leakage currents and fluctuations increased with increasing humidity and temperature. Breakdowns occurred during this phase. When the temperature stabilized, the leakage currents as well as the noise dropped, often by one decade; this constitutes a healing phenomenon. A.C. voltages of 480V, 240V, and 120V were applied. Turning the voltage on after the humidity had risen increased current and noise noticeably well above those situations where the voltage had been applied before the experiment was started, as shown in Figure 3.3.11. Over two hundred terminals were measured; seven total breakdowns were observed; six of these were coincident.

2. Forced breakdown was studied by applying contaminants in solutions with an eye dropper. If the insulator was noticeably hotter than the contaminant, the contaminant droplet was simply blown off, clearly helped by the applied electric field, and no breakdown could be obtained. Otherwise, strong contaminants (such as the standard containment spray solution) produced breakdowns within seconds.

   We were surprised to observe that broken-down terminals tend to regenerate when exposure to humidity and voltage ceases. Of 50 terminals broken down to a few ohms, 33 returned to 10 MΩ or more (measured with 4.5 volt ohmmeter) after drying for a few days with no voltages applied.

3. In a final experimental series, the Three Mile Island accident was closely imitated: the temperature rise was generated by injected steam and approached 86°C (186°F) in about 30 minutes.

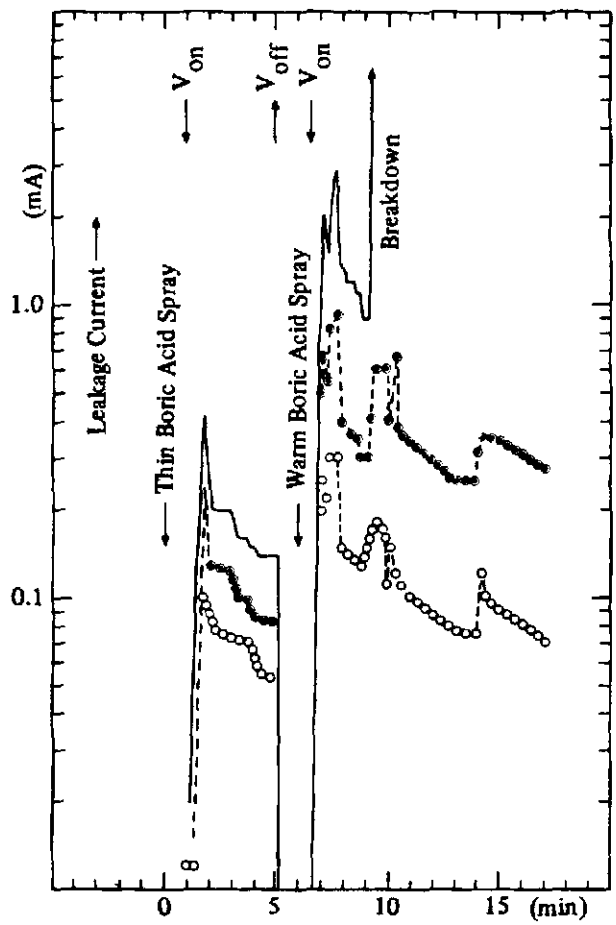


Figure 3.3.11 Leakage current vs time showing breakdown. (Block #2 temperature about 95°F, relative humidity 70-95%, V = 480 volts). (From Reference 3.3.9)

Contaminants were injected with the steam for the first 30 minutes of the first steam cycle. To incorporate the experiences of the preliminary tests, the terminals were exposed to three steam cycles and five voltage cycles during five-hour experiments (see Figure 3.3.12). Impurities were added at concentrations believed to have existed at Three Mile Island or taken from a general contamination listing. Readings for up to 12 terminals during each experiment were recorded every minute (some fast fluctuations were missed thereby). Three hundred and nineteen terminals were exposed to full cycle programs, and 108 additional ones to partial or altered programs. In part of the experiments, a protective enclosure with 6 mm weepholes was provided for the blocks. Twenty-three breakdowns were obtained, almost all at the beginning of the first or second voltage application. Average leakage resistances, mostly taken at 480V, varied from 5 MΩ for a well protected clean terminal to 200 kΩ for a dirty, unprotected terminal. Instantaneous values were occasionally lower by a factor of 10 and, thus, possibly inadmissible for certain circuits.

The study made recommendations for reducing or eliminating the danger from terminal block deterioration; they include:

- For new reactors or for thorough retrofit situations, elimination of terminal boxes as in safety-related circuits (as already accomplished at TMI) should be considered for all those circuits which are of conceivable use during and after an accident.

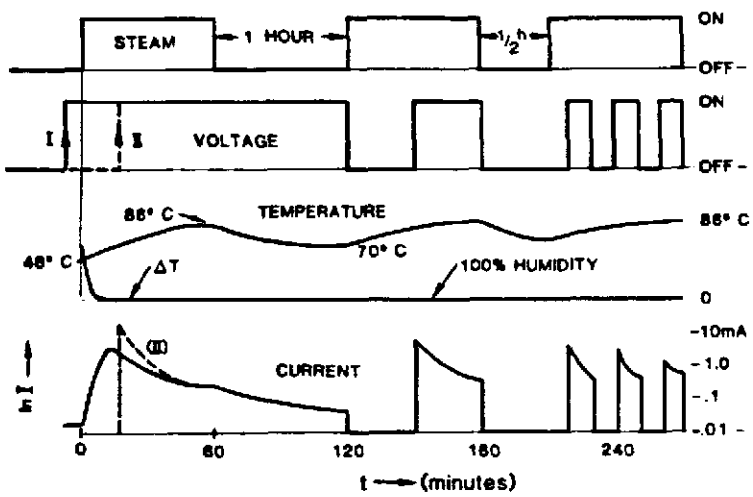


Figure 3.3.12 Steam, voltage, and temperature cycles with representative leakage current. (From Reference 3.3.9)

- Special attention should be given to penetration boxes and instrument boxes on the containment wall; they are connected to a heat sink, and the blocks inside will suffer higher water adsorption. High voltage circuits (480V) and those circuits where the power is cycled are most endangered.

- For a complete assessment of terminal block malfunctioning, a detailed circuit analysis has to be performed.

- For new reactors or for thorough retrofit situations, elimination of terminal blocks in safety-related circuits (as already accomplished at TMI) should be considered for all those cirucits which are of conceivable use during and after an accident.

International and Other U.S. Programs

Terminal Blocks

A LOCA-simulation test of terminal blocks was performed under a prior NRC-funded program (Independent Verification Testing Program), and is summarized in Reference 3.3.10, NUREG/CR-1952 (SAND81-0151).

Twenty known-defective sliding (or "U") link terminal blocks were subjected to an 80-psig, 10-second saturated steam thermal shock test followed by a 2-hour steady-state environment typical of LOCA simulations. The terminal blocks were originally manufactured by the States Company and were removed, for this test, from the Midland Nuclear Plant. The defects consisted of various-sized cracks between the threaded screwhole and side of the U-shaped link, which is designed for quick electrical disconnect.

The objective of this test program was to evaluate the potential for propagation of the cracks as a result of the LOCA, or LOCA simulation. In particular, the concern was for crack propagation to electrical failure during the thermal-shock phase of the LOCA, or LOCA simulation. As a basis for the test, terminal block qualification tests performed for the Alabama Power Company's Joseph M. Farley Nuclear Plant were used to establish typical temperature/pressure profile histories.

Based on the NDEs and visual inspection, the clip defects were categorized according to the extent of the defect. Five categories of defects were recognized; the descriptors are relative comparisons:

Category	Number of Clips
Failures	4
Very large radial cracks	8
Large radial cracks	1
Small radial cracks	3
Excess material, minor visual, other	18

The assembled terminal blocks were wired and installed in a NEMA-4 enclosure modified with four 1-inch holes cut into its bottom. The holes allowed cabling/thermocouple access and simulated the "leaky" enclosures used in previous tests.

It had been generally agreed that thermal shock is the primary (potential) mechanism for crack propagation as a result of a LOCA event. Therefore, other damaging conditions of a LOCA simulation, such as radiation, chemical spray, or temperature/pressure soak, were not a part of this test series.

Figure 3.3.13 shows the desired and actual profiles for the test. Rise time (to 80 psig) was about 6 seconds as measured by pressure transducer. Rise time (to 156°C) was about 10 seconds as measured by the environment-monitoring thermocouples. Saturated steam conditions prevailed during the test. Figure 3.3.14 shows the posttest view of the terminal blocks. All U-links remained intact and in place.

The circuit was energized with a 250 VDC, 1 amp, power supply just prior to, and for the duration of, the thermal shock test. Strip-chart recordings of the voltage and current were made to verify the loading and/or any electrical failures during the test. The circuit remained energized during the 2-hour test.

Insulation resistance (IR) measurements were made prior to and following the test. The measurements were made between the interconnecting wire termination and electrical ground. IRs ranged between ~ 10^{10} ohms at 500 VDC (pretest), to ~ 10^{8} ohms (upon completion of LOCA), and recovered to ~ 10^{10} ohms within 24 hours.

The report concludes that LOCA simulation thermal shock will not propagate the defects and/or result in an open circuit condition.

Reference 3.3.11, NUREG/CR-3418 (SAND83-1617), describes a two-phase series of tests on terminal blocks. Twenty-four terminal blocks (TBs) (five models from four manufacturers) were tested in simulated Design Basis Event (DBE) Loss-of-Coolant Accident (LOCA) environmental conditions. The environmental exposure profiles closely followed the recommended qualification profiles for temperature of IEEE 323-1974 Appendix A.

The primary objective of this test was to determine the failure and degradation modes of TBs when exposed to simulated LOCA conditions. The terminal blocks tested were those commonly used in nuclear power plants. They were tested in a configuration representative of plant installations and powered at voltages typical of RTD, pressure transmitter, and control circuits. Secondary objectives were to (1) investigate and compare performance differences in different TB designs, and (2) characterize insulation resistance in terms of leakage currents as functions of environment temperature, pressure, circuit voltage, and the presence of chemical spray.

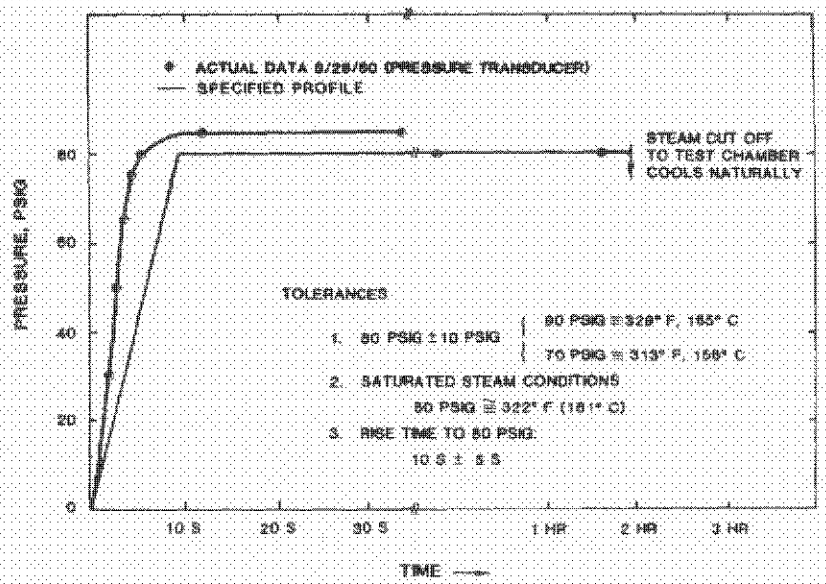


Figure 3.3.13 Specified and actual test profile. (From reference 3.3.10)

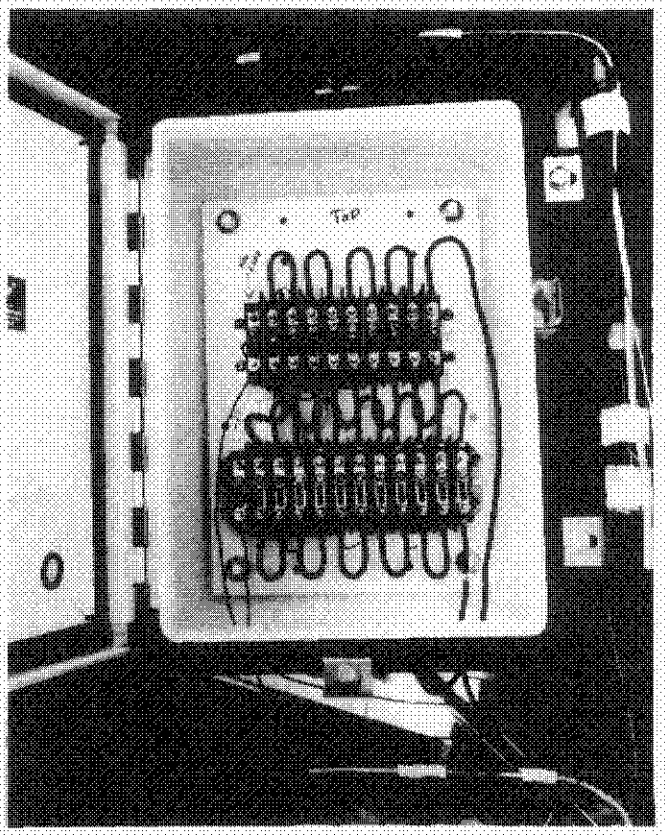


Figure 3.3.14 Posttest view of terminal blocks. (From Reference 3.3.10)

The terminal block tests were conducted in two phases. Each phase used new terminal blocks in the "as-received" condition. No thermal or seismic aging was conducted and no radiation was applied either as an aging environment or during the LOCA simulation. Phase I consisted of an 11-day exposure to a steam only environment. Phase II consisted of approximately one day of exposure to a simultaneous steam and chemical spray environment followed by a 5-day exposure to a steam only environment. Saturated steam conditions were maintained throughout both test phases.

The terminal blocks were powered at voltages of 4 Vdc, 45 Vdc, and 125 Vdc. Resulting currents associated with these voltage levels were 1.8 mA, 20 mA, and 1 A, respectively. Terminal-to-terminal and terminal-to-ground leakage currents were monitored on a discrete time basis throughout the test. Based on these measurements, insulation resistances were calculated. During exposure to the LOCA steam environment insulation resistance was observed to decreased from initial values of 10^8 to 10^{10} ohms to 10^2 to 10^5 ohms. These decreases in IR are interpreted as being caused by condition in surface moisture films rather than bulk conduction through the insulation material. There was a noticeable dependence of IR on temperature. Figures 3.3.15 and 3.3.16 show the temperature dependence of insulation resistance for Terminal Block 1.

During the chemical spray periods of the Phase II tests, no effect of the chemical spray was observed. This finding was somewhat surprising since it was expected the chemical spray would enter the conduit, penetrate down through the conduit-cable interstitial space, and drip onto the terminal blocks. It was hypothesized that the introduction of Na^+ and OH^- ions to the surface film would enhance the conductivity of the film. The lack of any observed change in leakage currents initially indicated that the NEMA-4 enclosures with unsealed conduit entrances provided adequate protection against the intrusion of chemical spray.

To check this result, a submergence experiment was conducted to observe the performance of blocks positively known to be spray contaminated. In this test three blocks were submerged in a chemical spray and steam condensate solution and three blocks were left unsubmerged. IRs in a steam environment after the submergence were compared. They indicated that there was only slight difference between submerged and unsubmerged blocks, with the unsubmerged blocks being slightly better.

Some of the conclusions reached, as a result of these tests were:

- Surface moisture films are the most probable explanation for the observed degradation of terminal block performance. Because films are a transient phenomena, the performance of the terminal blocks change with changing environmental conditions. Test conditions which do not adequately simulate the dominant accident conditions may bias the results of the test. For example superheated test conditions may not accurately reflect the terminal block's performance in a saturated environment.

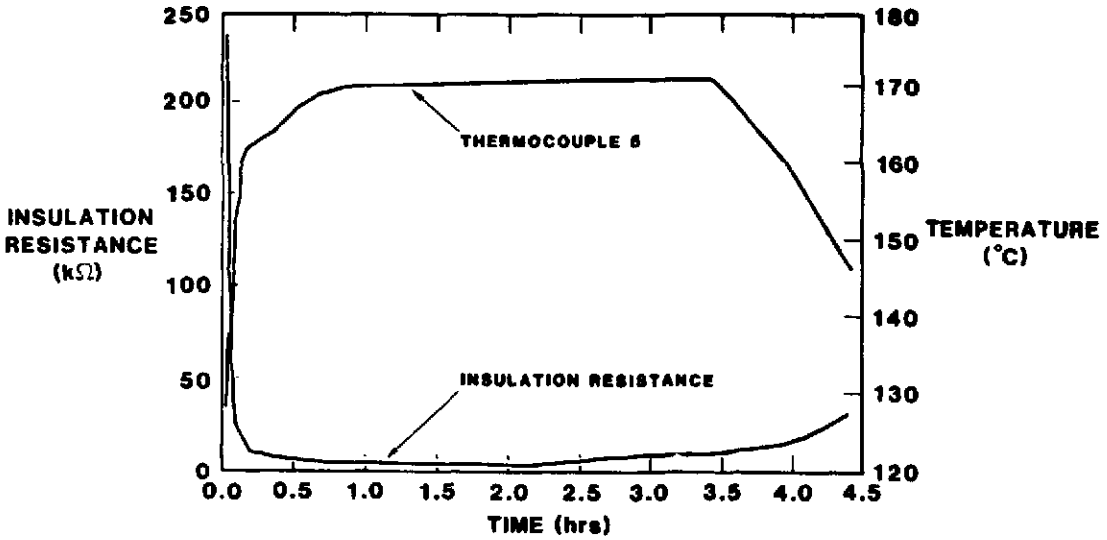


Figure 3.3.15 Insulation resistance for terminal block 1 during first steam ramp and 172°C plateau. (From Reference 3.3.11)

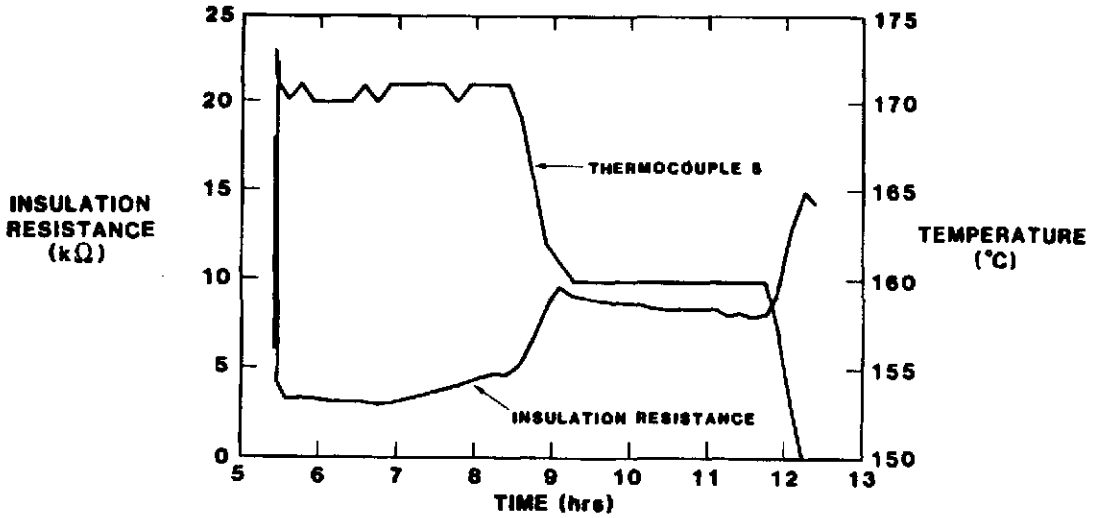


Figure 3.3.16 Insulation resistance for terminal block 1 from near the beginning of the first 172°C plateau through the 161°C plateau. (From Reference 3.3.11)

- Exposure to the LOCA environment caused some permanent degradation to the terminal block surface IRs, decreasing posttest and cooldown period IRs by one to two orders of magnitude.

- IRs were observed to be inversely related to the temperature of the environment.

- The chemical spray and submergence test results indicate that little change in the film conductivity may be expected as a result of chemical spray.

Reference 3.3.12, NUREG/CR-3691 (SAND84-0422), is a compendium of terminal blocks usage in the nuclear industry, including factors affecting their performance, and their possible impact on plant operations. This document includes a discussion on nine separate LOCA simulation tests performed by the industry for terminal block qualification.

The report makes several observations concerning the use, qualification, methodology, and performance factors of terminal blocks. Among these are:

- The primary application of terminal blocks in the nuclear power industry is instrumentation and control circuits.

- Most industry qualification tests do not continuously monitor for low level leakage currents during LOCA simulation tests of terminal blocks. Without quantitative knowledge of these leakage currents, adequate analyses of their effects on instrumentation and control circuits cannot be performed.

- The use of voltage levels above actual use conditions in qualification tests of terminal blocks may be nonconservative with respect to the measurement of low level leakage currents which are the primary degradation mode of terminal blocks.

- Terminal block leakage currents in a steam environment may degrade performance of instrumentation and control circuits to an extent sufficient to cause erroneous indications and/or actions.

Qualification of terminal blocks is a continuing effort in the industry; however, we are not aware of any significant national or international research efforts in this area.

Issue Findings

Terminal Blocks

In summary, the findings within this subissue are:

1. Elimination of in-containment terminal blocks in safety-related circuits should be considered.

2. Terminal blocks should not be mounted in such a manner that condensate can collect and form a breakdown/leakage path.

3. Sliding-link terminal blocks did not appear to be affected by thermal shock.

4. Surface moisture films are the most probable explanation for degradation of terminal block performance.

5. Terminal block surface IRs were observed to be inversely related to the environment temperature, and suffered some permanent degradation, by one to two orders of magnitude, in the posttest and cooldown period.

6. Chemical spray appeared to have little effects on the surface film conductivity.

7. Most industry qualification tests do not continuously monitor for low level leakage currents during LOCA simulation tests of terminal blocks. Without quantitative knowledge of these leakage currents, adequate analyses of their effects on instrumentation and control circuits cannot be performed.

References

Terminal Blocks

3.3.9 O. M. Stuetzer, Electrical Insulators in a Reactor Accident Environment, NUREG/CR-1682, SAND80-1957, Sandia Laboratories, Albuquerque, New Mexico, January 1981.

A. O M. Stuetzer, Preliminary Examination of TMI-2 Equipment: Terminal Blocks, SAND80-2447A. Presented at the USNRC Eighth WRSR Information Meeting, Gaithersburg, Maryland, October 27-31, 1980.

Terminal Blocks, Other Programs

3.3.10 L. L. Bonzon, et al, LOCA-Simulation Thermal-Shock Test of Sliding Link Terminal Blocks, NUREG/CR-1952, SAND81-0151, May 1981. (Independent Verification Testing Program results.)

3.3.11 C. M. Craft, Screening Tests of Terminal Block Performance in a Simulated LOCA Environment, NUREG/CR-3418, SAND83-1617, Sandia National Laboratories, August 1984. (CAP Program results)

A. C. M. Craft, Performance and Effects of Terminal Blocks in a Loss of Coolant Accident Environment, NUREG/CR-1965C. Presented at the 11th USNRC Water Reactor Safety Research Information Meeting, October 24-28, 1983, Gaithersburg, Maryland. Published in NUREG/CP-0048, vol. 5, pp. 261-282, January 1984. (CAP Program results)

3.3.12 C. M. Craft, An Assessment of Terminal Blocks in the Nuclear Power Industry, NUREG/CR-3691, SAND84-0422, Sandia National Laboratories, September 1984. (CAP Program results)

Cable Splices

Reference 3.3.13, NUREG/CR-0275 (SAND78-0067), summarizes the results of nine LOCA type tests performed on a number of Class 1 components. Cable splices were included as one component in three of the nine tests.

The primary purpose of this study was to determine, experimentally, the existence and magniture of synergistic effects that result from the simultaneous application of simulated loss-of-coolant accident (LOCA) environments on Class 1 equipment. This evaluation involved performing two types of tests (a sequential and a simultaneous application of radiation in sequence or combination with the other LOCA environments) on typical and identical Class 1 equipment.

The test conditions employed were the same as for the electric cable tests (Figures 3.3.1 and 3.3.2)

Rated single-phase current/voltage loads were applied continuously during the test. Current loads were maintained at approximately 20 amperes (60 Hz); 600 volts were applied separately between the conductor(s) and electrical ground.

The test data indicated that the splice assemblies functioned satisfactorily during both the sequential and simultaneous tests. Although there were some deterioration detected, no indication of significant synergistic effects were observed.

International and Other U.S. Programs

Cable Splices

Reference 3.3.14, NUREG/CR-2932/1 of 2 (SAND81-2027/1 of 2), reports on an equipment qualification research test of electric cables with factory splices.

Nine seven-strand No. 6 AWG and nine seven-strand No. 12 AWG electric cables manufactured by The Rockbestos Company with chemically crosslinked polyolefin insulation containing factory-made center conductor splices and insulation repairs were subjected to a methodology test of IEEE Standard 383-1974.

The cables were exposed to the following sequence of tests:

1. Low dose-rate radiation aging (62 krad/hr for 50.3 Mrad)

2. Thermal aging [approximately 846 hrs at 302°F (150°C)]

3. LOCA radiation (0.77 Mrad/hr for 146 Mrad)

4. Steam and chemical spray exposure [33 day LOCA/MSLB profile (cables energized-480 VAC, 11 amperes)].

Reference 3.3.15, NUREG/CR-2932/2 of 2 (SAND81-2027/2 of 2), also reports on an equipment qualification research test of electric cables with factory splices.

Nine single conductor, seven-strand, No. 12 AWG electric cables manufactured by General Electric Company with chemically crosslinked polyolefin insulation containing factory-made center conductor splices and insulation repairs were subjected to a methodology test of IEEE Standard 383-1974. The cables were exposed to the same test sequence as the Rockbestos cables discussed above (Reference 3.3.14).

Analysis of the test data indicated that the cables could function and support control operations during and after a LOCA/MSLB of the severity simulated by the tests. Further, the presence of center-conductor splices and insulation repairs did not appear to degrade cable performance in either test.

Qualification of cable splices is a continuing effort in the industry; however, we are not aware of any significant national or international research efforts in this area.

Issue Findings

Cable Splices

In summary then, the finding within this subissue is:

Splice assemblies did not exhibit significant synergistic effects.

References

Cable Splices

3.3.13 L. L. Bonzon, An Experimental Investigation of Synergisms in Class 1 Components Subjected to LOCA Typetests, NUREG/CR-0275, SAND78-0067, Sandia Laboratories, Albuquerque, New Mexico, August 1978. (Note: This is the same document as Reference 3.3.1.)

A. L. L. Bonzon, Test VIII, Sequential Mode; Cables, Splices, and Connectors, Sandia Laboratories, Albuquerque, New Mexico, July 1977.

B. F. V. Thome, Preliminary Data Report, Testing to Evaluate Synergistic Effects from LOCA Environments. Test IX. Simultaneous Mode; Cables, Splice Assemblies, and Electrical Insulation Samples, SAND78-0718, Sandia Laboratories, Albuquerque, New Mexico, April 1978.

C. L. L. Bonzon, An Experimental Investigation of Synergisms in Class 1E Components, SAND78-0346C (Albuquerque: Sandia Laboratories). Presented and published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, October 16-19, 1978.

Cable Splices, Other Programs

3.3.14 E. E. Minor and D. T. Furgal, Equipment Qualification Research Test of Electric Cable with Factory Splices and Insulation Rework, Test No. 2, Report No. 1, NUREG/CR-2932/1 of 2, SAND81-2027/1 of 2, Sandia Laboratories, Albuquerque, New Mexico, September 1982.

3.3.15 E. E. Minor and D. T. Furgal, Equipment Qualification Research Test of Electric Cable with Factory Splices and Insulation Rework, Test No. 2, Report No. 2, NUREG/CR-2932/2 of 2, SAND81-2027/2 of 2, Sandia Laboratories, Albuquerque, New Mexico, September 1982.

Electrical Penetration Assemblies

Reference 3.3.16 (RS 4445/81/03) reports on an equipment qualification research test of a Type K instrumentation penetration assembly manufactured by the D. G. O'Brien Company. The objective of the test was to subject a complete, fully-assembled penetration assembly to thermal and radiation aging and LOCA/MSLB steam exposure sequences. Penetration assembly performance was evaluated by insulation resistance measurements taken between conductors and ground.

The assembly tested was a D. G. O'Brien Type K penetration manufactured to Duke Power specifications. This assembly consists of three major components, 16 flange-mounted penetration modules, 16 inner connectors -- mounted in containment and 16 outer connectors -- mounted outside containment. The inner connectors were enclosed in a steel junction box.

The penetration assembly was thermally aged at 150°C for 168 hours in a large environmental chamber. Insulation resistance values were in the range of 10^8 ohms. Postthermal aging inspection revealed that all of the module connector coupling rings were loose. Prior to thermal aging they had been torqued to 25 foot-pounds. The coupling rings were all retorqued to 25 foot-pounds.

The penetration assembly was then radiation aged at 0.72 Mrad/hr to a total integrated dose of 183 Mrad(air). Insulation resistance values were on the order of 10^9 ohms. All module coupling rings were found to be loose during postradiation aging inspection, they were again retorqued to 25 foot-pounds.

The inner section of the penetration assembly was exposed to a LOCA/MSLB profile for approximately 140 hours. The temperatures at the first two plateaus were superheated, enhanced 50°C and 28°C, respectively, above the 15 psig saturated steam conditions. The remainder of the exposure was made at saturated steam conditions, except that after a thermal equilibrium of 121°C (saturated) was reached, the penetration assembly was given a thermal shock by spraying it with 4°C water for two minutes. Following thermal shock, the assembly was sprayed continuously with a chemical solution for the remainder of the exposure sequence.

During LOCA steam exposure, the first insulation resistance measurements were taken four hours into the sequence. The measurements showed that 12 circuits had either erratic or below minimum insulation resistance readings. At the end of the LOCA steam exposure, electrical shorts or near short circuit conditions were still indicated on three circuits and another three circuits still had insulation resistance values of less than 1.0×10^6 ohms. Insulation measurements taken after LOCA exposure cool-down showed that 86 of the 104 circuits had insulation resistance values below 5.0×10^6 ohms.

Visual inspection of the penetration assembly after the exposure sequences revealed the following:

1. Most of the connector coupling rings were loose -- finger tight.

2. The Sylgard 170 cable grommet material had extruded through the cable feed-through openings in the connector plug sleeves. (See Figure 3.3.17.)

3. Where grommet extrusion occurred, the cable insulation was necked-down, and in some instances, the insulation was pushed back, exposing the conductor. (See Figure 3.3.18.)

4. In some instances, the extrusion process exposed the cable conductors in locations near the connector plug sleeve creating the possibility for a short circuit.

5. Moisture intrusion at the connector-module interface was observed in many instances.

International and Other U.S. Programs

Electrical Penetration Assemblies

We are not aware of any significant international effort in this area.

There have been a number (approximately 360 reports are listed in the January 1985 Equipment Qualification Data Bank Quarterly Report, prepared for EPRI by NUS Corp.) of qualification tests performed on electrical penetrations. Three reports are significant in that they also involved D. G. O'Brien Type K penetration assemblies. The first -- Reference 3.3.17 (Report No. ER 252) -- provides the data acquired during qualification tests performed by the D. G. O'Brien Company. However, in these tests, the grommet and interfacial seal were not exposed to the thermal and radiation aging environments. Reference 3.3.18 (Report No. 45869-1: Vol. I) and Reference 3.3.19 (Report No. 45869-1: Vol. II) describe an environmental qualification test made on an electrical penetration assembly having Type B through Type M connector modules. The Type K plug module is reported to have "...performed satisfactorily during the Accident Test..." However, the plug kits were disassembled during thermal and radiation aging.

Reference 3.3.20 describes a new program to study the potential for leakage through containment electrical penetration assemblies under severe accident conditions beyond the design basis events.

Issue Findings

Electrical Penetration Assemblies

Regulatory Guide 1.63 addresses the qualification requirements for electric penetration assemblies. Basically, it endorses the procedures set forth in IEEE Std. 317-1976, subject to seven additional guidelines.

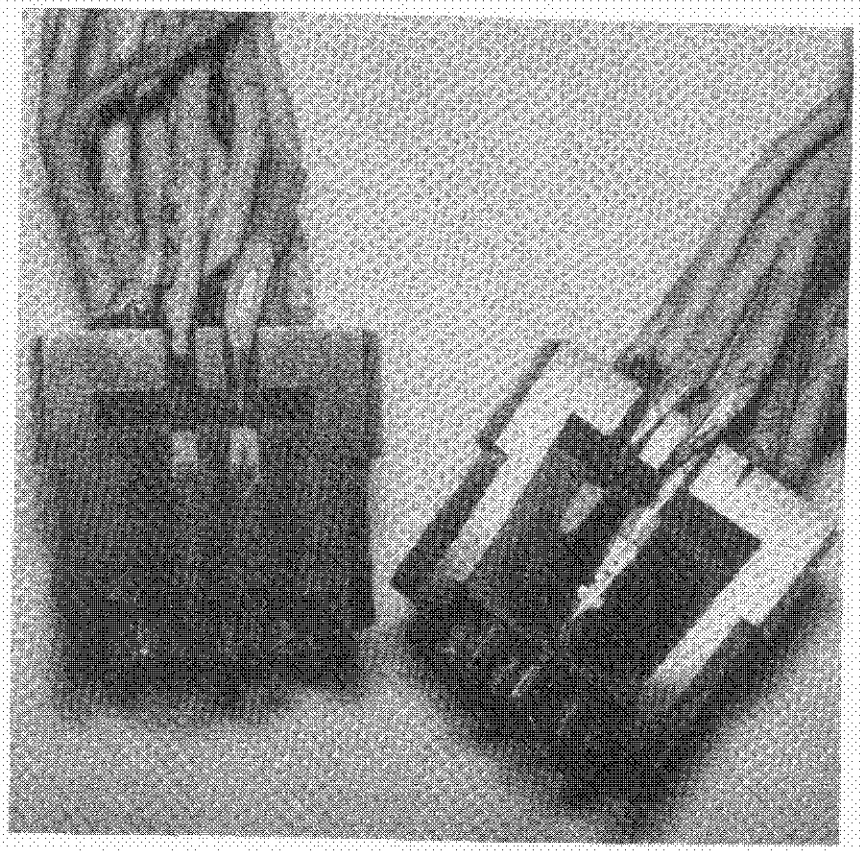


Figure 3.3.17 Sectioned inner connectors showing elastomer extrusion around the conductors. (From Reference 3.3.16)

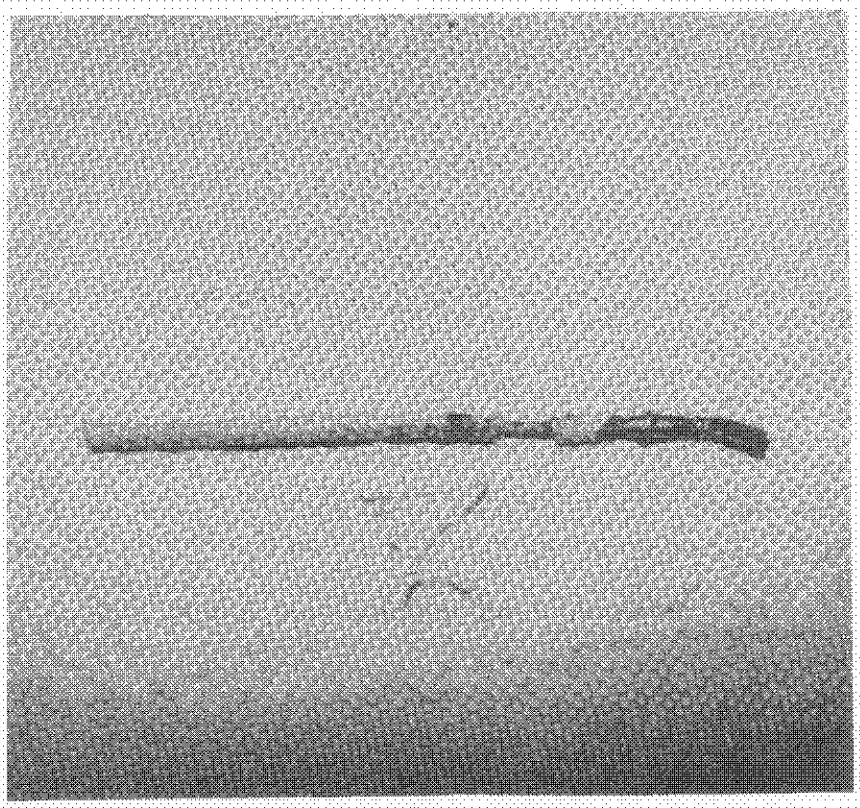


Figure 3.3.18 Single conductor, removed from outer connector assembly, showing the effect of elastomer extrusion on the insulation. (From Reference 3.3.16)

In summary, the findings within this subissue are:

1. Sylgard 170 cable grommet material extruded through the cable feed-through openings in the connector plug sleeves, stripping conductor insulation and exposing the conductor in locations near the plug sleeve.

2. Moisture intrusion at the connector-module interface was observed after the LOCA steam exposure.

References

Electrical Penetration Assemblies

3.3.16 W. H. Buckalew and F. V. Thome, Equipment Qualification Research Tests of the Duke Power/D. G. O'Brien Type K Instrumentation Penetration Assembly, RS4445/81/03, Sandia Laboratories, Albuquerque, New Mexico, March 1982. Limited Distribution; obtain through W. Farmer, USNRC.

Electrical Penetration Assemblies, Other Programs

3.3.17 Low Voltage and Instrumentation Electric Penetration Qualification Test Report, Report No. ER 252, D. G. O'Brien, Inc. (No Date.)

3.3.18 Environmental Qualification Test Report on Electrical Penetration Assembly, Type B-M for Duke Power Company: Volume I, Report No. 45869-1, Wyle Laboratories, Huntsville, Alabama, July 21, 1982.

3.3.19 Analysis of Cables and Connectors Following Accident Test Concluded on March 31, 1982 and Extended Accident Test Conducted From May 10 Through 18, 1982 on Electrical Penetration Assembly, Type B-M For Duke Power Company: Volume II, Report No. 45869-1, Wyle Laboratories, Huntsville, Alabama, September 10, 1982.

3.3.20 Letter C.Subramanian (SNLA) to W. Farmer (NRC), dated February 24, 1984, Electrical Penetration Assemblies (EPAs) Study Program (FIN No. A-1364) Test Plan.

Pressure Switches

No specific testing of pressure switches has been conducted to date, the effort has been only to monitor the results of other national and international programs.

International and Other U.S. Programs

Pressure Switches

Reference 3.3.21, NUREG/CR-3630 (SAND83-2652), discusses the results of seismic and of loss-of-coolant accident (LOCA) simulation tests made on pressure switches procured from two different equipment manufacturers.

Two separate tests were conducted. The unaged equipment baseline evaluation test program was performed to serve as a screening tool to identify unanticipated problem areas and to assess the general functional capabilities of the equipment when subjected to design-basis event environments only. Seismic and LOCA environments were incorporated sequentially on the same switches. The seismic simulation spectrum was a composite of generic required response spectra (RRS). The LOCA simulation generically followed the typical IEEE-323 accident profile.

The follow-up test to the design-basis event test involved a LOCA simulation only. It was performed to help identify primary failure modes as well as failure thresholds. The LOCA environment was simulated by using saturated steam in a step-function profile for the temperature and pressure.

The results showed the primary mode of failure to be the rupture of the elastomeric gaskets and seals. In some instances, an almost simultaneous failure of the diaphragms (also elastomeric) was observed. The failure threshold was shown to be less than 10 psig to 30 psig pressure. In both tests (baseline evaluation and follow-up) failures resulted from deformed or torn gaskets or seals that allowed steam and chemical spray into the switch housing. The failures were manifested by shorting of the electrical system. (See Figure 3.3.19.) Ruptured diaphragms resulted from environmental test chamber pressure seeking relief through the weak, unsupported inlet side of the pressure switch diaphragm and out the pressure-sensing line.

Qualification of pressure switches is a continuing effort in the industry; however, we are not aware of any other significant national or international research efforts in this area.

Issue Findings

Pressure Switches

Current work to date only suggests that additional testing should be conducted on pressure switches.

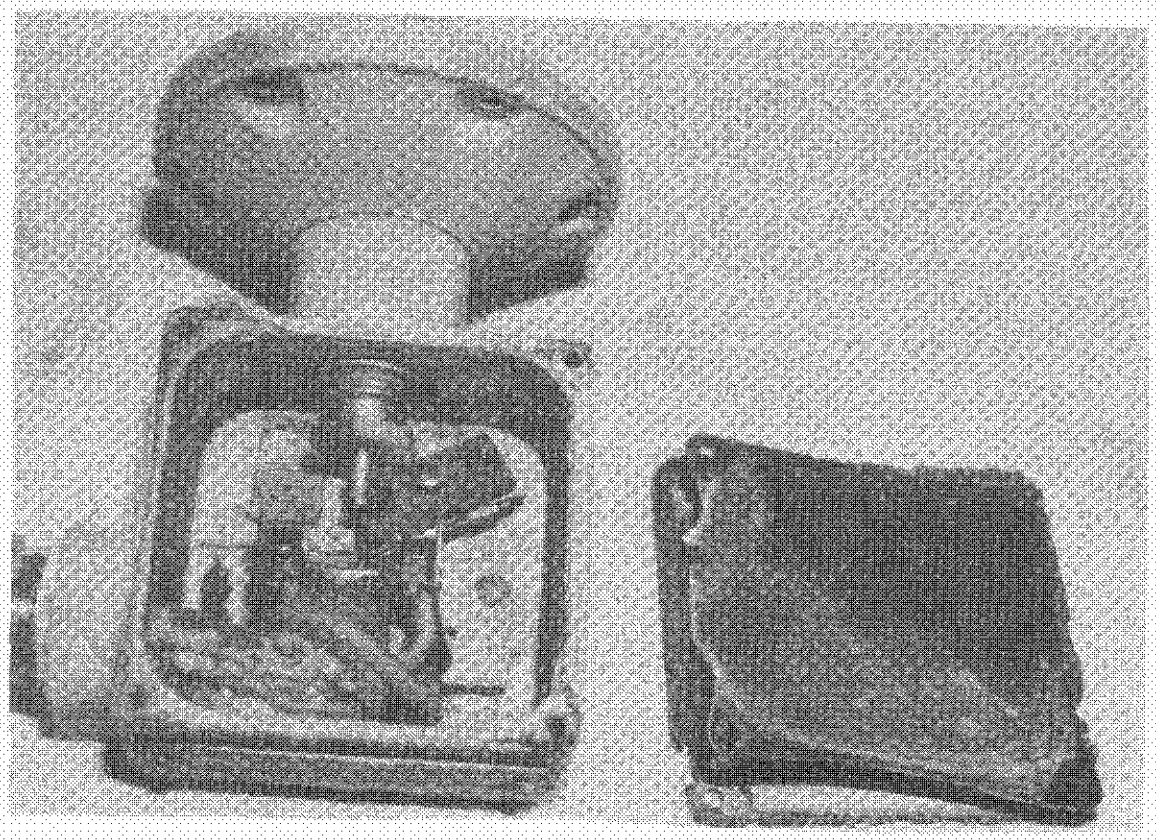


Figure 3.3.19 PostLOCA view of one failed pressure switch showing internal damage. (From Reference 3.3.21)

References

Pressure Switches, Program Effort

none

Pressure Switches, Other Programs

3.3.21 E. A. Salazar, Equipment Qualification Methodology Research: Tests of Pressure Switches, NUREG/CR-3630, SAND83-2652, Sandia National Laboratories, Albuquerque, New Mexico, March, 1984. (EQRT Program results.)

A. E. A. Salazar, Quick-Look Report, Pressure Switch Test, December 14, 1982. (EQRT Program results.)

B. E. A. Salazar, Quick-Look Report, Follow-Up Test on Baseline Evaluation Test Using Pressure Switches, January 27, 1983. (EQRT Program results.)

RTDs

No specific testing of RTDs has been conducted to date, the effort has been only to monitor the results of other national and international programs.

International and Other U.S. Programs

RTDs

Reference 3.3.22, NUREG/CR-3597 (SAND84-0938), discusses the results of loss-of-coolant accident (LOCA) environment simulation test made on resistance temperature detectors (RTDs) procured from three different manufacturers.

The purpose of the test was to "screen" RTDs from three different suppliers to assess the functional capabilities of unaged equipment and to determine the primary failure modes of RTDs. Ten unaged RTDs were exposed to the LOCA environment, which consisted of saturated steam and chemical spray. The test lasted about 24 hours and was of short duration to isolate any obvious problem areas. The LOCA environment caused functional failures and some physical damage in four of the RTDs tested.

The results showed that the primary mode of failure for Supplier A RTDs to be the leakage allowed by the head seal gasket material. In one case the failure occurred at 15 psig pressure; in two cases the failures probably did not occur until after the peak (115 psig) pressure condition had passed. The failures were manifested by shorting of the electrical connectors in the RTD head. Figure 3.3.20 shows water inside the head area of one RTD found during posttest inspection.

Considering the failure mode of these unaged components, it appears that the failures are related to the seal material rather than seal design (a contributing cause also appears to be poor quality control in the head casting process). Figure 3.3.21 shows evidence of water leakage past the cap seal.

Supplier B and Supplier C RTDs functioned properly throughout all phases of this test.

Qualification of RTDs is a continuing effort in the industry; however, we are not aware of any other significant national or international research efforts in this area.

Issue Findings

RTDs

Current work to date only suggests that additional testing should be conducted on RTDs.

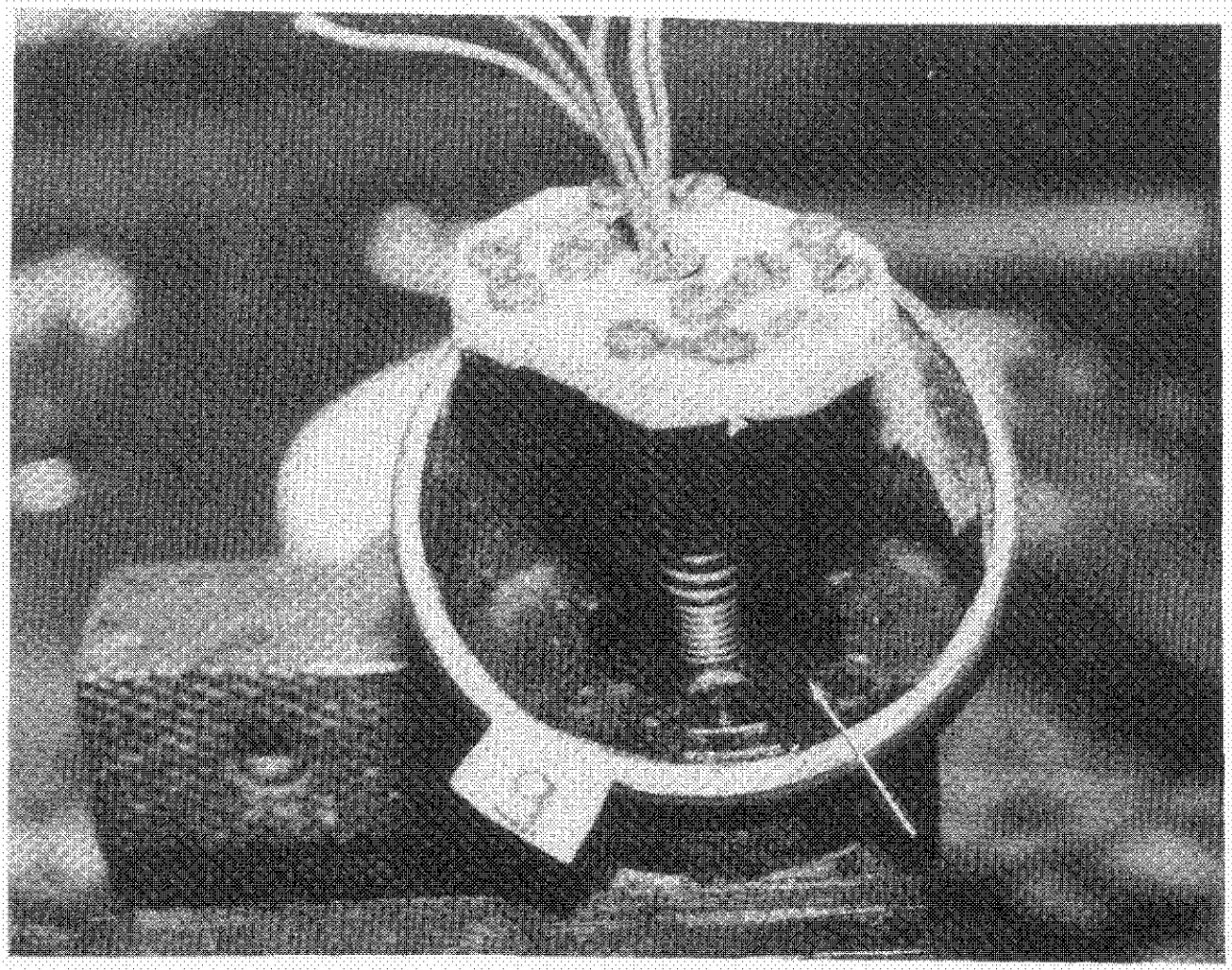


Figure 3.3.20 RTD No. 1, showing water inside the thermowell (arrow).
(From Reference 3.3.22)

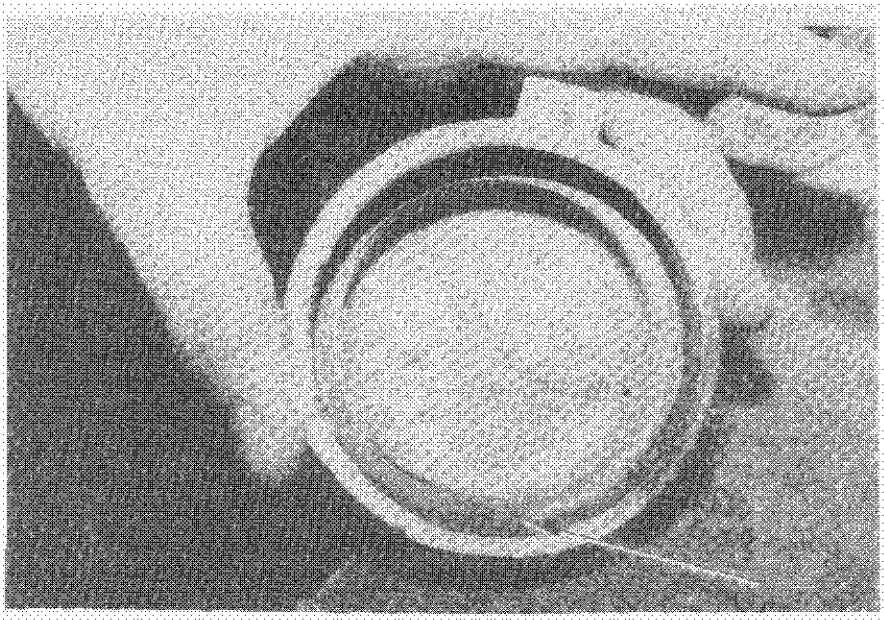


Figure 3.3.21 RTD No. 1, showing discoloration of cap seal. (From Reference 3.3.22)

References

RTDs, Program Effort

none

RTDs, Other Programs

3.3.22 F. J. Wyant and E. E. Minor, Equipment Qualification Methodology Research: Tests of RTDs, NUREG/CR-3597, SAND84-0938, Sandia National Laboratories, Albuquerque, New Mexico, September 1984. (EQRT Program results.)

A. E. E. Minor, Quick-Look Report, RTD Screening Test, April 25, 1983. (EQRT Program results.)

Pressure Transmitters

No specific testing of pressure transmitters has been conducted to date, the effort has been only to monitor the results of other national and international programs.

International and Other U.S. Programs

Pressure Transmitters

Reference 3.3.23, NUREG/CR-3863 (SAND84-1264), discusses an experimental investigation into the performance of Class 1E electronic pressure transmitters exposed to environments within and beyond the design basis. Emphasis was placed on determining the instruments' failure and degradation modes in separate and simultaneous environmental exposures. Five unaged ITT Barton Model 763 pressure transmitters were tested; each transmitter was exposed to a unique environment. The environments were (T1) simulated loss-of-coolant accident (LOCA) steam/chemical spray conditions along, (T2) temperature alone, (T3) radiation alone, (T4) simultaneous radiation and LOCA temperature (no steam) conditions, and T(5) simultaneous radiation and simulated LOCA steam/chemical spray conditions.

Specific objectives were (1) to determine and isolate the effects of individual and simultaneous environmental stresses, (2) to address severe accident questions by testing beyond the normal design basis environmental envelope, (3) to investigate the current qualification test methodology as it pertains to electronic components, and (4) to identify and analyze any weak-link circuit components for the mechanisms contributing to their degradation.

The response of the transmitters showed that temperature is the primary environmental stress affecting performance. Initial large errors that decreased with time-at-temperature were observed. The source of these errors seem to be a common mode design weakness in the transmitter's calibration potentiometers resulting from a dependency of material dielectric properties on temperature. The modification recommended by the manufacturer to reduce this temperature effect proved to work reasonably well in the long term. A potential second common failure mode which activates slowly with time-at-temperature was also identified. The operation of this failure mechanism is catalyzed by the presence of lubricants used in the production of some potentiometers. The design of this transmitter proved to be exceptionally hard to radiation effects and there appeared to be no significant synergistic effects between radiation and temperature.

Reference 3.3.24, NUREG/CR-4324 (SAND85-1481), discusses hydrogen burn survival testing of pressure transmitters. These tests subjected artificially aged and unaged Barton Model 763 pressure transmitters to heat flux pulses simulating increasingly severe hydrogen burns. The initial base heat flux pulse used in the tests was representative of a

hydrogen deflagration in a large dry containment building resulting from a 75 percent metal-water reaction. Starting with the base pulse, the test specimens were exposed to increasingly severe heat flux pulses until a pulse whose heat flux levels were three times those of the base pulse was reached. Specimen performance was checked during and after exposure to each of the simulated hydrogen burns.

Transmitter output measurements performed between exposures to the heat flux pulses indicated slight changes in transmitter calibration from test to test for both specimens. These changes were generally larger in the unaged transmitter.

Calibration checks conducted in a pressure lab at the conclusion of the entire test series showed that the performance of both transmitter specimens (aged and unaged) fell outside of manufacturer's technical specifications. For laboratory conditions, the accuracy should have been ±0.8 percent of full scale. The maximum deviation was -2.85 percent of full scale for the unaged transmitter at 1000 psig (the upper end of its operating span) and -2.15 percent for the aged transmitter (also at 1000 psig). Figures 3.3.22 and 3.3.23 show the calibration check results.

These tests demonstrate that transmitters, in both unaged and aged condition, can withstand exposure to very severe simulated hydrogen burn heat plus pulses and remain functional.

Qualification of pressure transmitters is a continuing effort in the industry; however, we are not aware of any other significant national or international research efforts in this area.

Issue Findings

Pressure Transmitters

Current work to date only suggests that additional testing should be conducted on pressure transmitters.

References

Pressure Transmitters, Program Effort

none

Pressure Transmitters, Other Programs

3.3.23 D. T. Furgal, C. M. Craft, and E. A. Salazar, Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests, NUREG/CR-3863, SAND84-1264, Sandia National Laboratories, March 1985. (CAP Program results.)

A. D. T. Furgal, C. M. Craft, Performance Assessment of Class 1E Pressure Transmitters Subjected to Environmental Stresses, SAND84-1655C. Presented at the 12th USNRC Water Reactor Safety Research Information Meeting, October 22-26, 1984, Gaithersburg, Maryland. (CAP Program results.)

3.3.24 V. J. Dandini, Testing of Nuclear Qualified Cables and Pressure Transmitters in Simulated Hydrogen Deflagrations to Determine Survival Margins and Sensitivities, NUREG/CR-4324, SAND85-1481, Sandia National Laboratories, Albuquerque, New Mexico, December 1985.

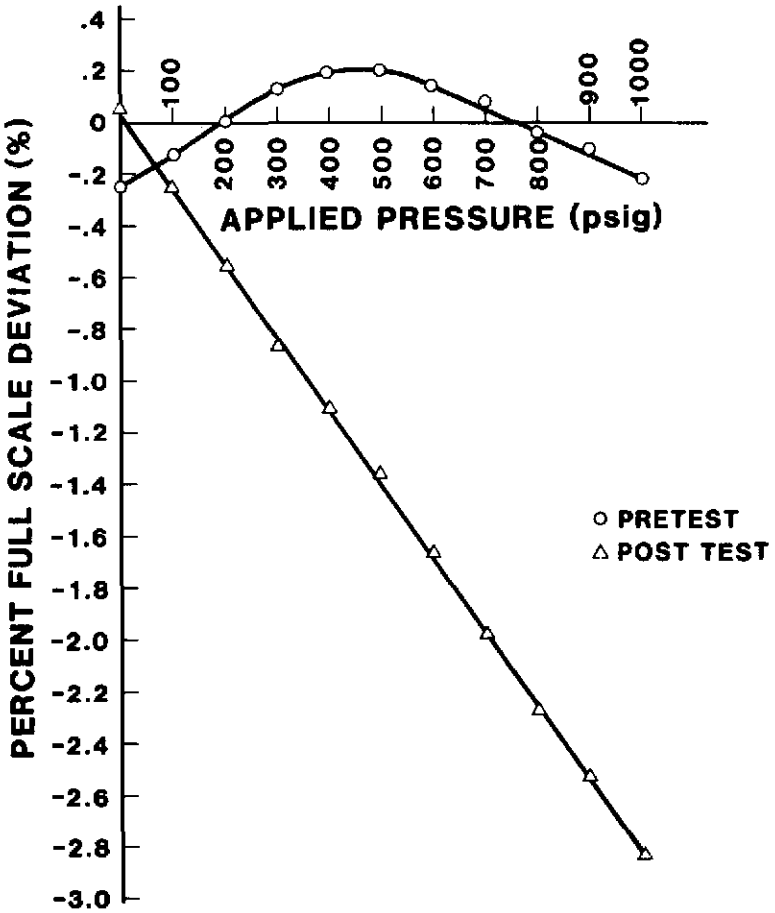


Figure 3.3.22 Unaged Transmitter Percent of Full-Scale Deviation From Pretest Calibration [Lab Check Results]. (From Reference 3.3.24)

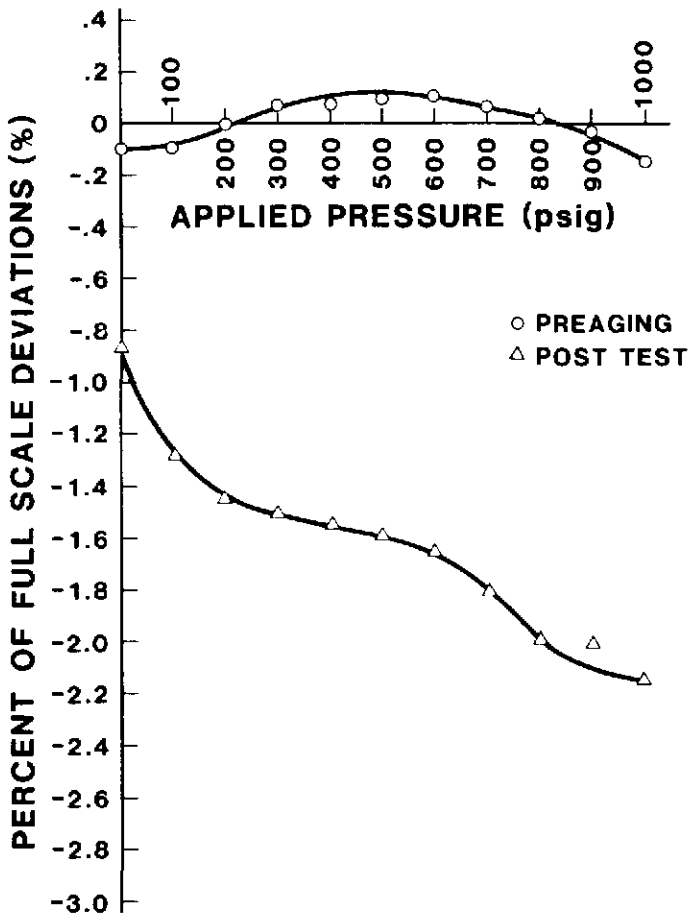


Figure 3.3.23 Aged Transmitter Percent of Full-Scale Deviation From Preaging Calibration [Lab Check Results]. (From Reference 3.3.24)

Valve Operators

No specific testing of valve operators has been conducted to date, the effort has been only to monitor the results of other national and international programs.

International and Other U.S. Programs

Valve Operators

Reference 3.3.25, NUREG/CR-3424, is a report documenting the results of an equipment qualification research tests performed on nine ASCO solenoid valves. The purpose of the program was to provide independent verification of previous testing performed on similar solenoid valves. In addition, the substitution of two naturally aged valves for specimens that had malfunctioned after thermal aging, allowed a comparative evaluation of the behavior of artificially aged valves to those that were naturally aged.

The solenoid valves were (accelerated) aged to 50 Mrad gamma radiation followed by thermal aging at 268°F to simulate a 4-year life at 140°F. Two thousand operational cycles were performed during thermal aging. Two of the aged valves were removed from testing and replaced by valves that had been naturally aged (3 years at 140°F while energized and pressurized with N_2). All valves were cycled 2000 times at room temperature, then exposed to vibration aging, seismic testing, 150 Mrad gamma irradiation, and a 30-day LOCA/MSLB simulation.

Two valves leaked excessively after thermal aging. All valves performed satisfactorily during seismic testing (a tenth, unaged, valve was included in this test and the results did not reveal any difference in behavior between unaged, naturally aged, and artificially aged valves). Of the seven valves subjected to the LOCA/MSLB exposure: One performed satisfactorily, two performed their safety function but leaked, and four failed during transients.

The valve that functioned properly throughout the test series without leaking had a metal valve seat, all of the other valves had seats made of EPDM.

Qualification of valve operators is a continuing effort in the industry; however, we are not aware of any other significant national or international research efforts in this area.

Issue Findings

Valve Operators

Current work to date only suggests that additional testing should be conducted on valve operators.

References

Valve Operators, Program Effort

none

Valve Operators, Other Programs

3.3.25 D. V. Paulson, et al, Equipment Qualification Research: Test Program and Failure Analysis of Class 1E Solenoid Valves, NUREG/CR-3424, Franklin Research Center, Philadelphia, Pennsylvania, November 1983.

Materials

Reference 3.3.26, NUREG/CR-0275 (SAND78-0067), reports on simultaneous and sequential tests which were performed on two compounds of typical radiation crosslinked, highly flame-retardant polyolefins which were used in insulating systems for nuclear power applications. One objective of the tests was to evaluate for synergisms in the cable insulation materials. A second objective was to evaluate the effect of "adequate" sample age. Reference 3.3.26 concludes that no significant material synergism for elongation existed for these two materials. Figures 3.3.24 and 3.3.25 illustrate elongation data for the two compounds during the sequential qualification exposure. The results for compound A (Figure 3.3.24) are not logically self-consistent; but the effect of aging is apparent, since in general, the aged samples are more severely degraded. The compound B data (Figure 3.3.25) has several interesting features. First, increasingly severe degradation correlates logically with the corresponding increasingly severe environment or combination of environments. For example, "Aging + LOCA" is more severe than "Aging" or "LOCA" separately. The other combinations are similarly logical. Next, the preaged samples are always more severely degraded. This illustrates the potential importance of accurate and adequate aging. Similar results were obtained during simultaneous testing (Figures 3.3.26 and 3.3.27).

Reference 3.3.27, NUREG/CR-2763 (SAND82-1071) presents the results of a U.S.-French test program. During this test program, nine materials commonly used as electric cable insulations and jackets in nuclear power plant applications were monitored in terms of mechanical properties, weight increases, solubility measurements, and infrared spectroscopy. Three different HELB simulation tests were employed. The only planned difference among the three tests was the composition of the overpressure gas (gas pressure was constant for the three tests) that was present in addition to the steam environment. The three gas compositions were nitrogen (0% oxygen), and 10% oxygen, and air (21% oxygen).

Reference 3.3.27 indicated that the oxygen concentration during a LOCA simulation can have a significant influence on the chemical and physical changes occurring in certain polymeric materials. It noted that the chloroprene material (See Figure 3.3.28) studied appeared to suffer significantly more damage in LOCA simulations containing oxygen, a conclusion which was valid even for samples which were not aged. For aged EPR insulation samples (See Figure 3.3.29), substantially decreased tensile properties resulted when oxygen was included in the HELB simulations. Solubility, swelling, and Fourier Transform Infrared Spectroscopy results on these EPR materials indicated that increased oxidative scission was responsible for the enhanced degradation. For a PVC material (See Table 3.3.1),the results indicated that the damage became more severe as the oxygen content was lowered (slightly greater mechanical damage, substantially greater swelling). Reference 3.3.27 concludes that it is possible that a HELB simulation without oxygen may underestimate the changes which occur in some materials while overestimating the changes which occur in other materials.

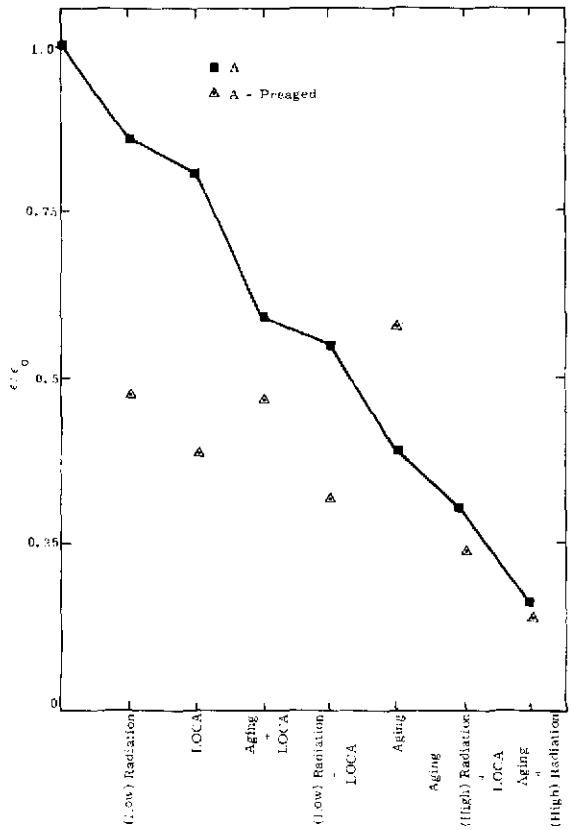


Figure 3.3.24 Normalized elongation (VIII) for the Type A insulation samples versus increasingly severe environments. (From Reference 3.3.26)

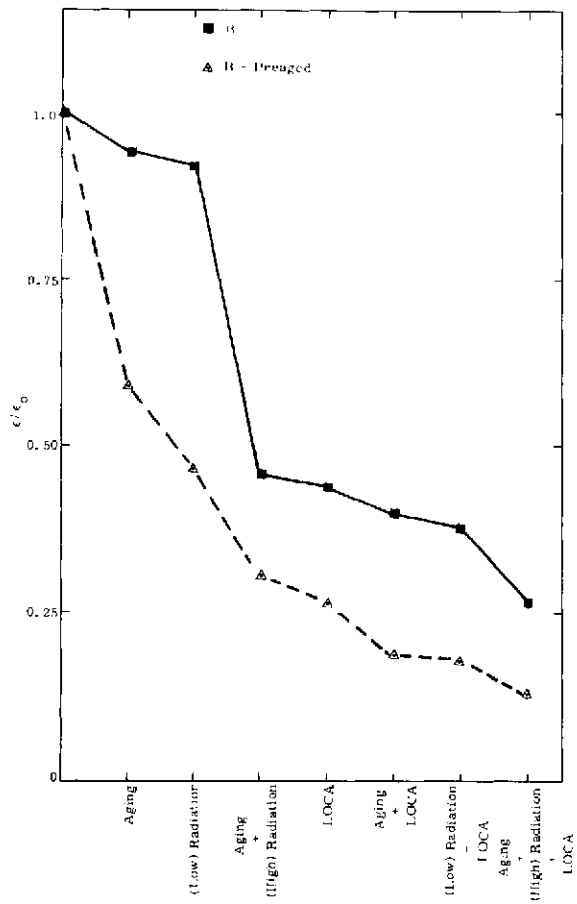


Figure 3.3.25 Normalized elongation (VIII) for the Type B insulation samples versus increasingly severe environments. (From Reference 3.3.26)

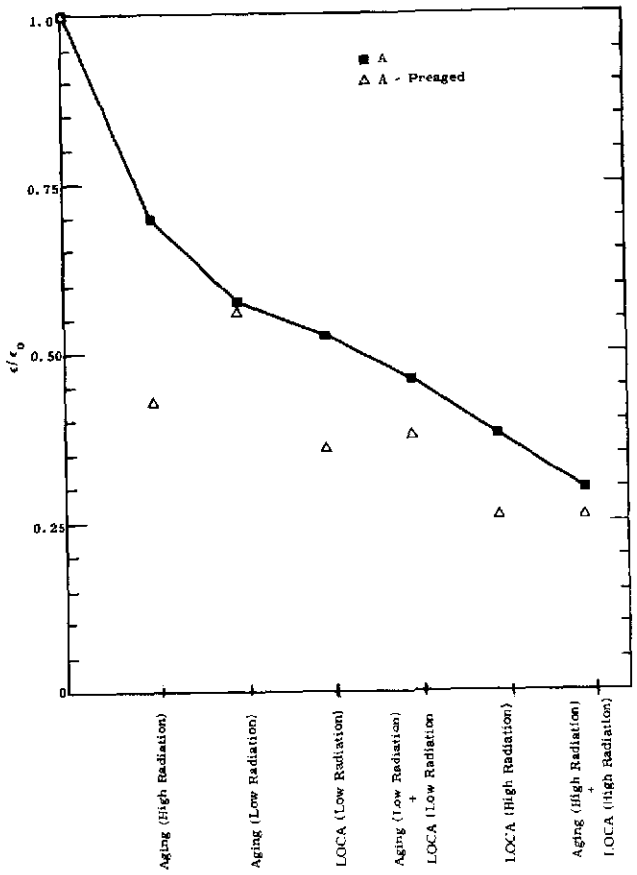


Figure 3.3.26 Normalized elongation (IX) for the Type A insulation samples versus increasingly severe environments. (From Reference 3.3.26)

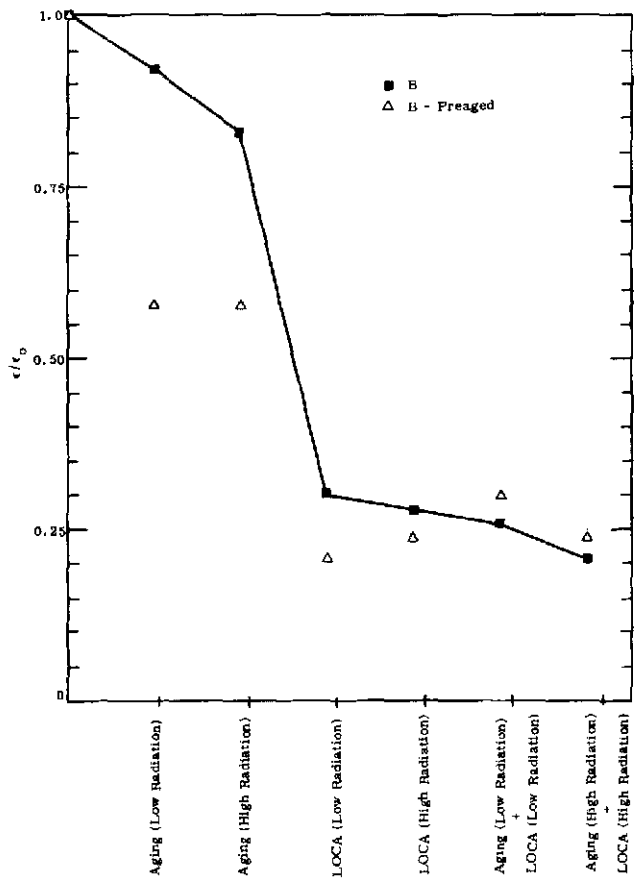


Figure 3.3.27 Normalized elongation (IX) for the Type B insulation samples versus increasingly severe environments. (From Reference 3.3.26)

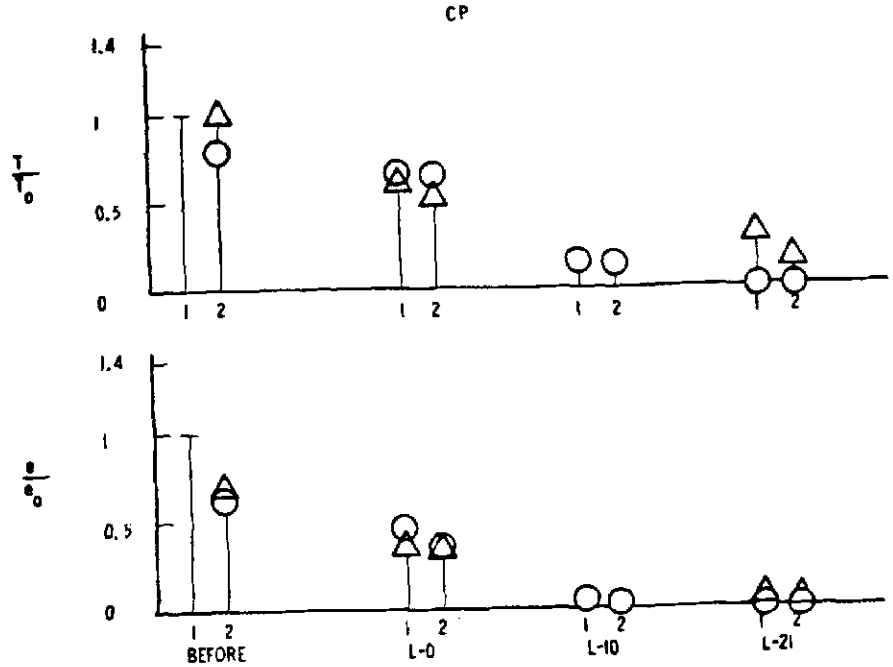


Figure 3.3.28 Tensile property results for the chloroprene (CP) material. The tensile strength, T, divided by its unaged value, T_o, and the ultimate tensile elongation, e, divided by the unaged value, e_o, are plotted after the aging exposure and after the 3 HELB simulations. The numbers 1 and 2 refer to the two aging conditions: unaged, and 20 Mrad at 830 krad/h, respectively. BEFORE refers to the results before the HELB simulation, L-0, L-10, and L-21 refer to results after the HELB simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 3.3.27)

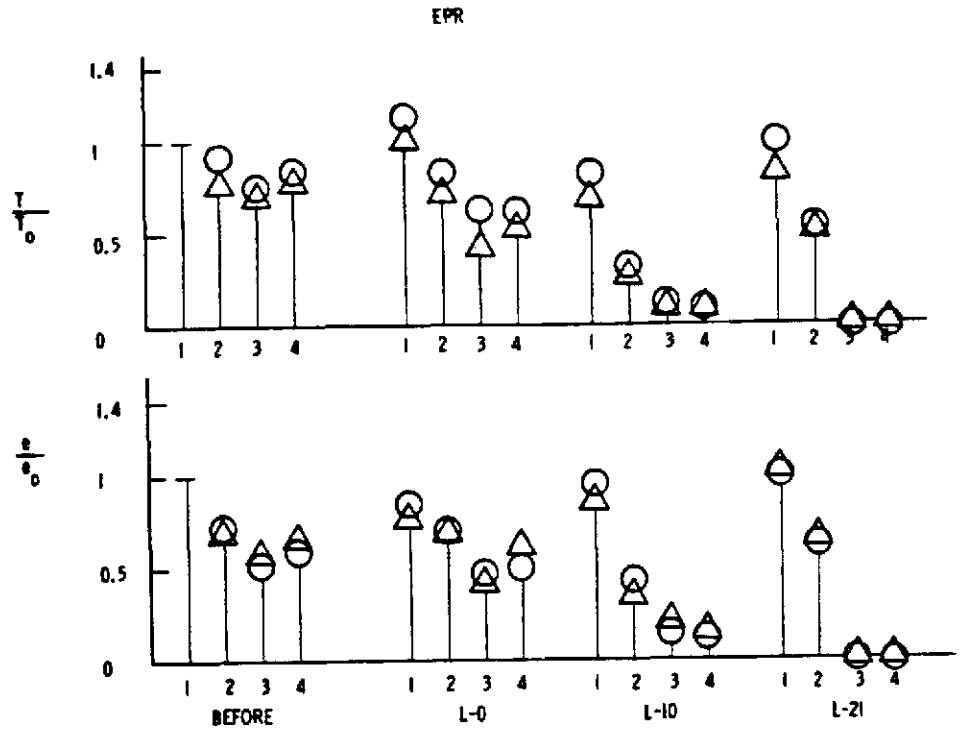


Figure 3.3.29 Tensile property results for the EPR material. The tensile strength, T, divided by its unaged value, T_o, and the ultimate tensile elongation, e, divided by the unaged value, e_o, are plotted after the aging exposure and after the 3 HELB simulations. The numbers 1, 2, 3, and 4 refer to the four aging conditions: unaged, 22 Mrad at 910 krad/h; 47 Mrad at 930 krad/h; and 26 Mrad at 27 krad/h, respectively. BEFORE refers to the results before the HELB simulation, L-0, L-10, and L-21 refer to results after the HELB simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 3.3.27)

Reference 3.3.27 also noted that aging conditions are an important parameter that helps define material response during subsequent HELB simulations. Figures 3.3.29 and 3.3.30 illustrate test results for EPR and CLPO respectively. For both materials ultimate tensile elongation properties depended on the aging technique (total dose and dose rate) employed prior to the HELB simulation. Table 3.3.1 illustrates that the moisture absorbed by PVC during the HELB simulation depended strongly on the aging technique.

Materials properties for several EPRs were examined by the test efforts described by Reference 3.3.28, NUREG/CR-3538 (SAND83-1258). A purpose of the material tests was to monitor insulation moisture absorption and tensile properties to gain insight concerning mechanical property changes that might cause cable electrical degradation. Functional performance of electric cables with the same insulation materials was also monitored.

Reference 3.3.28 did note substantial electrical degradation for one of the EPR multiconductor cables during simultaneous accident simulations. Examination of materials' properties indicated that this product experienced substantial moisture absorption and dimensional increases

Table 3.3.1
(From Reference 3.3.27)

MATERIAL	HELB TEST % O_2	AGING LOT	PERCENT WEIGHT CHANGE AFTER 6 HR	1500 HR	5000 HR
PVC	0	1	+93	+14	-4
		2	+220	+50	-1
		3	+347	+46	-4
		4	+330	+62	-7
	10	1	+53	-2	
		2	+150	-4	
		3	+360	-3	
		4	+230	-4	
	21	1	0	-1	-1
		2	+18	0	-2
		3	+39	+4	-2
		4	+35	+4	-3

Weight Change Due to HELB Simulation versus Time After LOCA Simulation. Aging Lot #1 refers to unaged PVC material. Lot #2 was aged to 20 Mrad at 830 krad/h. Lot #3 was aged to 42 Mrad at 830 krad/h. Lot #4 was aged to 23 Mrad at 24 krad/h.

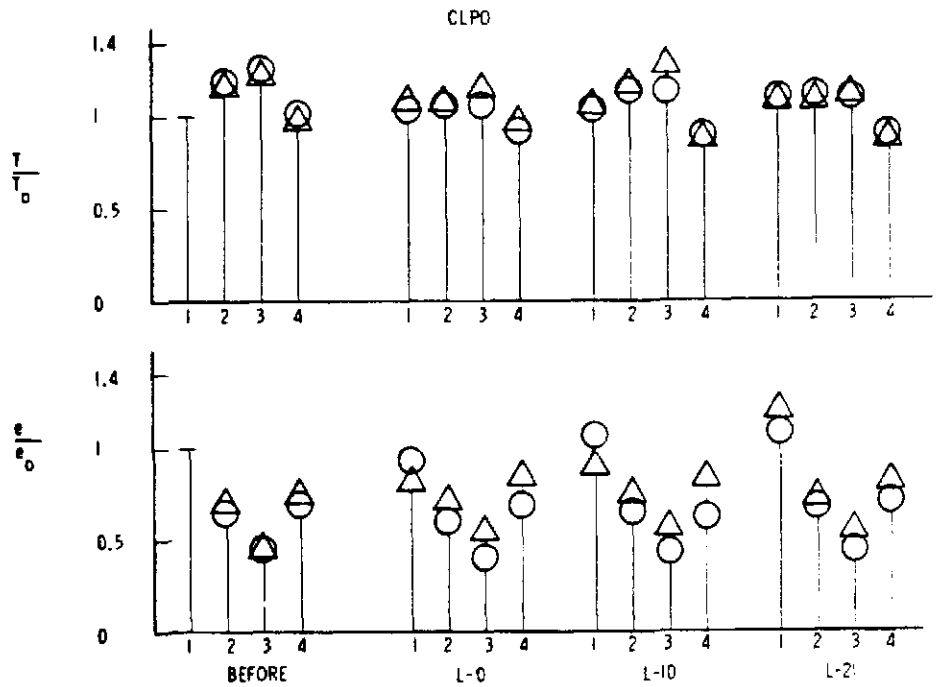


Figure 3.3.30 Tensile property results for the CLPO material. The tensile strength, T, divided by its unaged value, T_0, and the ultimate tensile elongation, e, divided by the unaged value, e_0, are plotted after the aging exposure and after the 3 HELB simulations. The numbers 1, 2, 3, and 4 refer to the four aging conditions: unaged, 22 Mrad at 910 krad/h; 47 Mrad at 930 krad/h; and 26 Mrad at 27 krad/h, respectively. BEFORE refers to the results before the HELB simulation, L-0, L-10, and L-21 refer to results after the HELB simulations containing nitrogen, 10 percent oxygen and air, respectively. The circles give the results measured at Sandia, the triangles denote the results obtained at CEA-ORIS-STBR. (From Reference 3.3.27)

during accident simulations, particularly during the simultaneous accident simulations (Table 3.3.2). No other multiconductor product experienced as dramatic moisture absorption as did EPR D.

The simultaneous tests employed by Reference 3.3.28 for EPR D (and other EPR materials) included simultaneous aging techniques prior to the simultaneous accident simulation. To differentiate between between aging and accident influences on test conclusions, insulation material properties were examined by efforts described in Reference 3.3.28 for several combinations of aging and accident conditions. Table 3.3.3 illustrates that moisture absorption for several of the EPR materials (including EPR D) depended strongly on the accelerated preconditioning applied prior to the simultaneous accident simulation. Those specimens aged to a "40 year equivalent" most severely absorbed moisture during subsequent simultaneous accident simulations. Figure 3.3.31 demonstrates for one of the EPR materials listed in Table 3.3.3 (EPR-1483) that dimensional increases were most severe for simultaneous aging techniques that simulate a "40 year equivalent life". Moreover, Figure 3.3.31 suggests that sequential accident simulation techniques may be suitable provided appropriate age preconditioning techniques are employed.

To summarize, concurrent examination of materials properties in addition to functional testing of cables provided necessary information (1) to allow for the development of an hypothesis regarding the cable degradation mode, (2) to hypothsize that age preconditioning was very important to the observed degradation, and (3) to demonstrate that appropriate sequential qualification techniques might have also produced this degradation mechanism.

Table 3.3.2
Percentage Weight Increases for EPR and XLPO Tensile Specimens.
(From References 3.3.28 and 3.3.29)

Cable Material	Sequential Test	Simultaneous Test #1	Simultaneous Test #2
EPR A	+50	?	
EPR B	+4	-1	
EPR C	+9	+23	
EPR D	+121	+173	+172
EPR E	0	+7	
EPR F			+94
EPR G			
XLPO A			+15
XLPO B			+58
XLPO C			+25

EPR Material	Accelerated Age	Moisture Absorption (% Weight Increase)
EPR D	Unaged	16
	40-yr equiv*	172
EPR F	Unaged	20
	40-yr equiv*	94
Japanese EOR-5	Unaged	49
	40-yr equiv*	77
EPR-1483	Unaged	17
	5-yr equiv**	22
	40-yr equiv***	67

* A 7d 139°C thermal exposure with simultaneous irradiation for 6 d to 40 Mrd (air-equiv.).

** A 94 d simultaneous exposure to 120°C and 4.9 Mrd (air-equiv.).

*** A 30 d simultaneous exposure to 120°C and 39 Mrd (air-equiv.).

Table 3.3.3 The Effect of "Accelerated Age" on EPR Moisture Absorption During Simultaneous Steam and Irradiation LOCA Simulation. (From Reference 3.3.28)

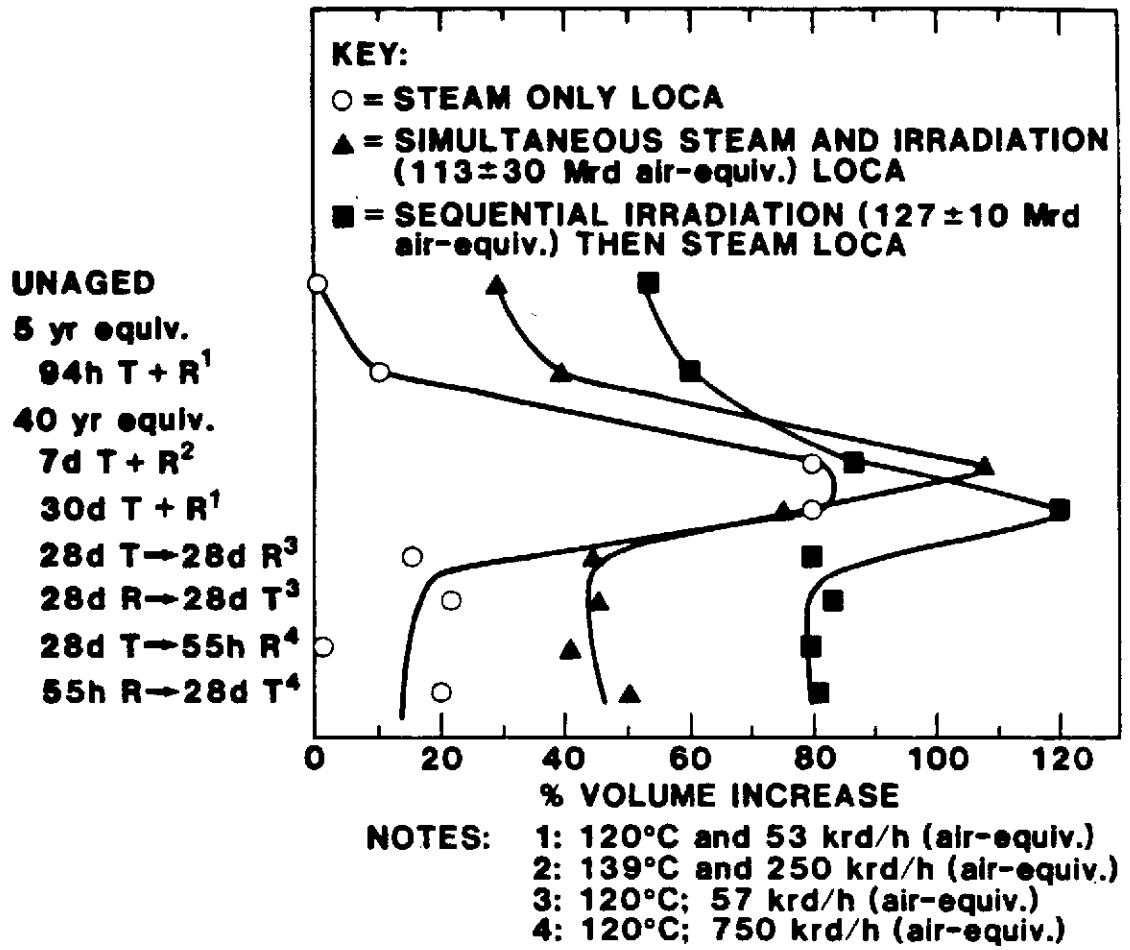


Figure 3.3.31 Effect of aging and accident techniques on % volume change of EPR-1483. (From Reference 3.3.29D)

Reference 3.3.29 examined the implications of the EPR conclusions for XLPO cables. Reference 3.3.29 did not observe for XLPO cables any functional performance differences caused by simultaneous versus sequential testing or resulting from the use of multiconductor test specimens. Based on materials' properties measurement, it was concluded that this was not surprising. Substantial moisture absorption (Table 3.3.2) and dimensional swelling of XLPO cable insulation during simultaneous accident simulations were not observed. Moreover, ultimate elongation for all three XLPO products at completion of the simultaneous tests were higher than for the EPR D product (Table 3.3.4).

Material property response of several polymer materials to variations in aging and accident simulation techniques was reported by Reference 3.3.30, NUREG/CR-4091 (SAND84-2291). The overall research goal was to evaluate whether some combinations of alternative aging and accident simulation techniques were better suited for qualification activities than other combinations of simulation techniques. The research effort included the following experimental variables: (1) sequential versus simultaneous accelerated aging exposures; (2) the order of the sequential aging exposures; (3) ambient versus 70°C irradiation temperatures during the sequential aging exposures; (4) the presence and absence of oxygen during the accident simulations; (5) sequential versus simultaneous accident HELB simulations; and (6) ambient versus 70°C irradiation temperatures during the sequential accident irradiations.

Reference 3.3.30 noted that the presence or absence of air (oxygen) during accident simulations can influence the degree of degradation in some materials. The US EPR and TEFZEL materials (Figures 3.3.32 and 3.3.33 and Table 3.3.5) were examples where degradation was enhanced when air was present during the accident simulations. The US XLPO materials (Figures 3.3.34 and 3.3.35) are examples where degradation was reduced by the presence of air. Reference 3.3.30 concluded that since most reactor containments are not inerted, a conservative accident simulation for qualifying materials would include air in the LOCA test chamber.

Product	% Weight Gain	% Outer Diameter Increase	e/eo*	e* (%)
XLPO A	15	6	.16 ± .05	60 ± 20
XLPO B	58	22	.30 ± .12	96 ± 38
XLPO C	25	12	.17 ± .06	56 ± 20
EPR D (Ref. 1)	173	53	.13 ± .02	31 ± 5

*Measurements performed after moisture desorption had stabilized.

Table 3.3.4 Insulation Specimen Properties at Completion of Simultaneous Testing. (From Reference 3.3.29)

Table 3.3.5
(From Reference 3.3.30)

a. TEFZEL 1: Bend Radii By Which All Samples Cracked. Table entries are expressed as multiples of the TEFZEL 1 sample radius.

ACCIDENT SIMULATIONS

	R70→ST(AIR)	R70→ST(N_2)	R28→ST(AIR)	R28→ST(N_2)	R + ST(AIR)	R + ST(N_2)
UNAGED	69.00	75.00	11.00	6.00	69.00	
R120	75.00	75.00	75.00		75.00	75.00
R70→120	75.00	75.00	69.00	50.00	75.00	44.00
120→R70	75.00	75.00	44.00	31.00	75.00	69.00
R27→120	75.00	75.00	44.00	75.00	75.00	22.00
120→R27	75.00	75.00	75.00	75.00	75.00	56.00

b. TEFZEL 2: Bend Radii By Which All Samples Cracked. Table entries are expressed as multiples of the TEFZEL 2 sample radius.

ACCIDENT SIMULATIONS

	R70→ST(AIR)	R70→ST(N_2)	R28→ST(AIR)	R28→ST(N_2)	R + ST(AIR)	R + ST(N_2)
UNAGED	6.00	6.00			75.00	
R120	75.00	75.00	75.00	75.00	75.00	75.00
R70→120	75.00	75.00	44.00	75.00	75.00	56.00
120→R70	22.00	75.00	75.00	56.00	75.00	75.00
R27→120	11.00	11.00	11.00	11.00	75.00	22.00
120→R27	22.00	11.00	11.00	22.00	75.00	44.00

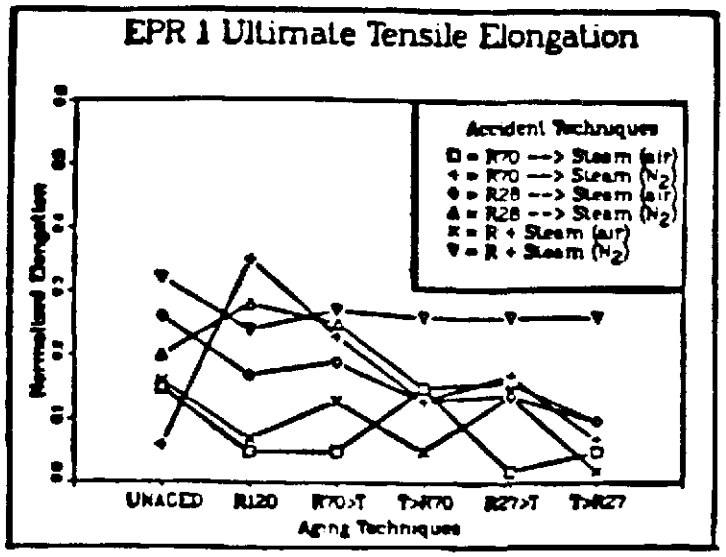


Figure 3.3.32 Ultimate tensile strength of EPR 1 at completion of accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R28 refers to an ambient temperature irradiation. T indicates an elevated temperature exposure. Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 3.3.30)

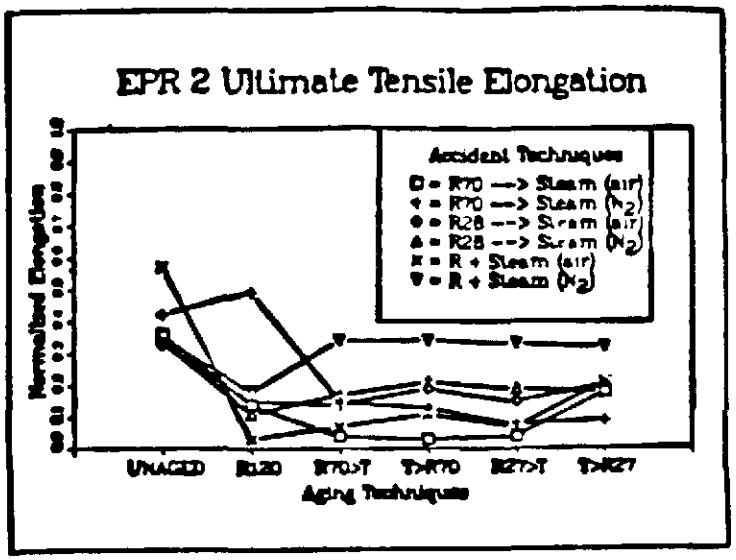


Figure 3.3.33 Ultimate tensile strength of EPR 2 at completion of accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R28 refers to an ambient temperature irradiation. T indicates an elevated temperature exposure. Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 3.3.30)

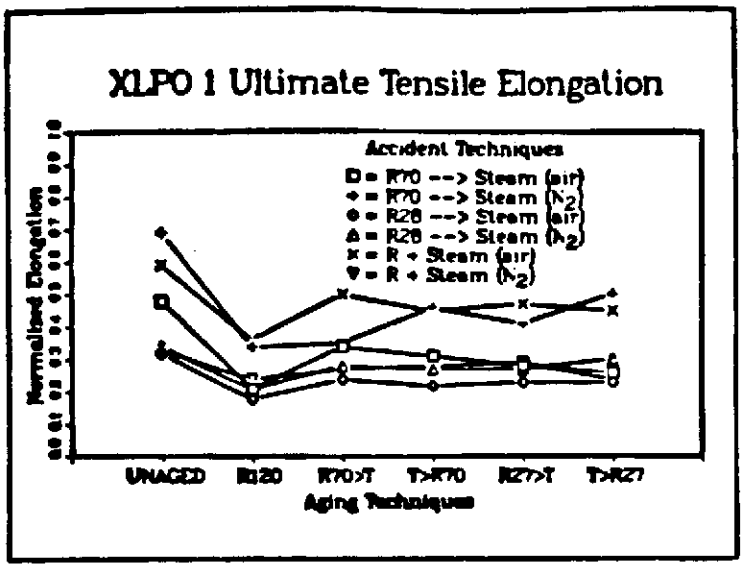


Figure 3.3.34 Ultimate tensile strength of XLPO 1 at completion of accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R28 refers to an ambient temperature irradiation. T indicates an elevated temperature exposure. Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 3.3.30)

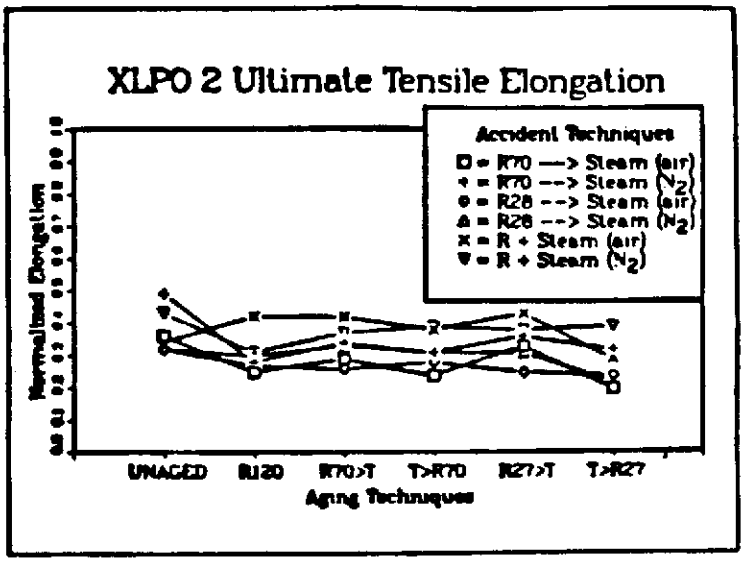


Figure 3.3.35 Ultimate tensile strength of XLPO 2 at completion of accident simulation. Various aging techniques employed prior to the accident simulation are displayed along the x-axis. R28 refers to an ambient temperature irradiation. T indicates an elevated temperature exposure. Steam(air) and steam(N_2) refer to accident steam simulations with air and nitrogen, respectively. (From Reference 3.3.30)

Reference 3.3.30 also noted that materials properties during accident simulations are sensitive to the age preconditioning technique. Some examples are illustrated in Figures 3.3.36 through 3.3.39. It concludes that a radiation followed by a thermal exposure is a more conservative choice for an aging sequence than would a thermal followed by radiation aging sequence.

Finally, Reference 3.3.30 noted that perceptions regarding the importance of accident synergistic effects depended on the previously mentioned variables of (1) oxygen presence during the accident simulation and (2) the age preconditioning technique. Reference 3.3.30 concluded that for most of the materials it studied that sequential qualification techniques could be used. Table 3.3.6 identifies the most appropriate sequential qualification procedures for each material included in the Reference 3.3.30 test program. These sequential techniques produced degradation similar to that achieved during simultaneous irradiation and LOCA (with air) accident simulation test exposures except for a fire-retardant French EPDM material (Figure 3.3.38). Reference 3.3.30 considered the simultaneous accident simulation to be the best representation of postulated design basis event accident conditions among those included in the test program.

Reference 3.3.31 examined the behavior of Sylgard® 170 Silicone elastomer during accelerated thermal aging, during radiation exposures, and during applied compressive forces. Experiments were also performed with simultaneous application of compressive stress and elevated temperature. During these latter experiments, the degree of elastomer confinement was also a variable.

Sylgard 170 response to elevated temperature exposures included an initial loss of elongation followed by very little additional degradation. Figure 3.3.40 illustrates a composite plot of Sandia and Dow Corning elongation data. Material properties were not particularly age dependent. Radiation exposures, however, produced significant, monotone changes in both the elongation and hardness with increasing radiation dose. Results are illustrated by Figure 3.3.41.

Sylgard 170's response to an applied compressive force was strongly dependent on the environmental temperature and the degree of material confinement. Figure 3.3.42 illustrates the compressive force versus elapsed aging time for the first 70 days of aging. The 150°C aging exposure was more degrading than was the ambient or 43°C exposure. After 135 days, the 150°C test specimen had decayed to a background stress of 50 lbs or less. After 187 days, the 43°C test specimen had a remaining compressive stress of approximately 500 lbs. The 43°C aging exposure was then terminated. These two specimens and an unaged specimen were then exposed to a LOCA temperature profile (no steam nor chemical spray) and the compressive force was monitored. The unaged and 43°C exposed disc behaved very similarly. Results are shown in Figure 3.3.43. Compressive forces during the early stages of the accident temperature simulation reached 2500 lbs. For the specimen aged at 150°C (Figure 3.3.44),

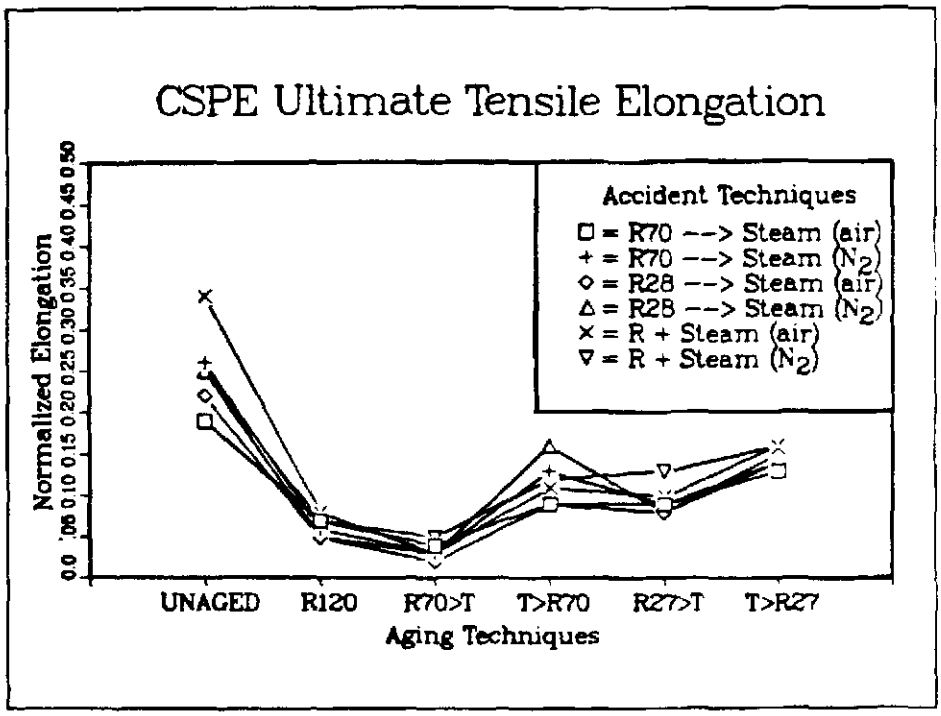


Figure 3.3.36 Ultimate tensile elongation of CSPE at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure. (From Reference 3.3.30)

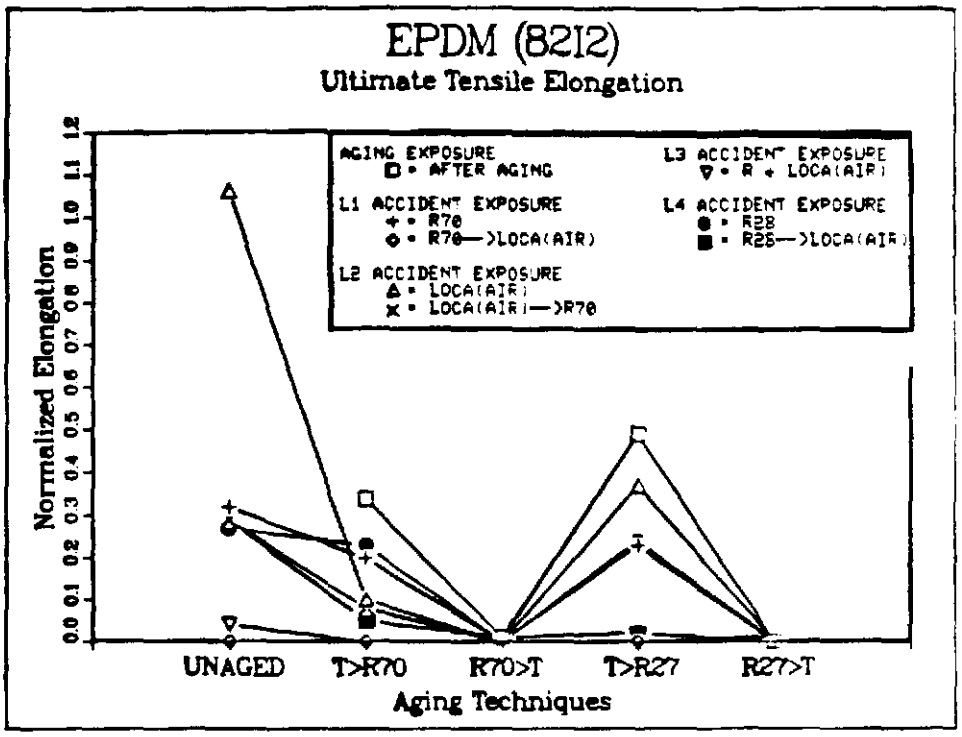


Figure 3.3.37 Ultimate tensile elongation of EPDM (82I2) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure. (From Reference 3.3.30)

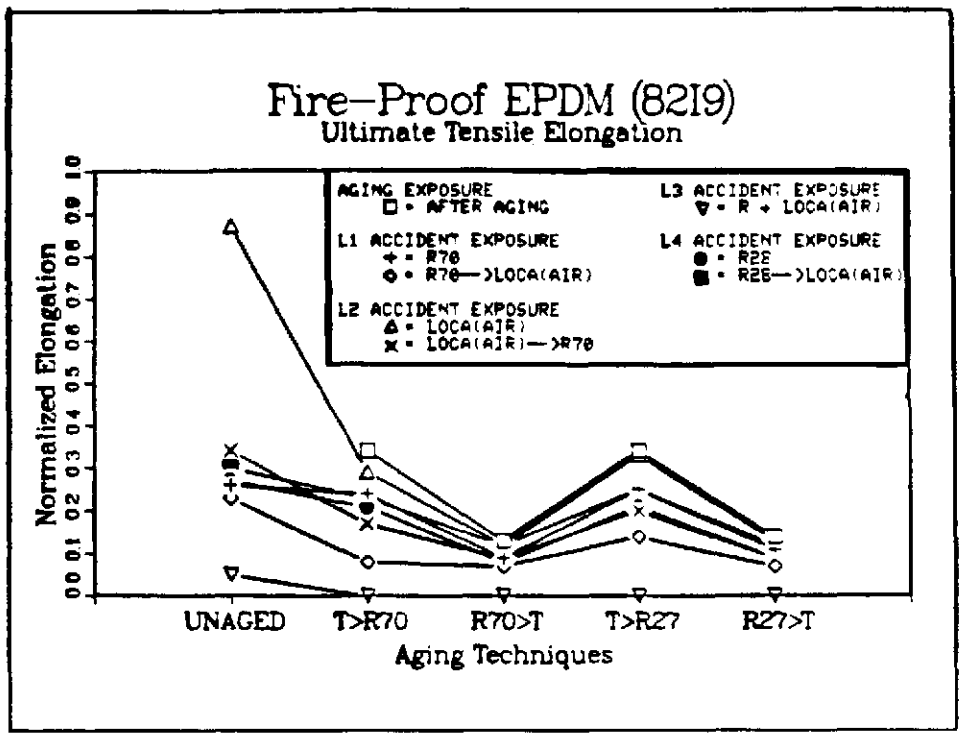


Figure 3.3.38 Ultimate tensile elongation of Fire-Proof EPDM (8219) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure. (From Reference 3.3.30)

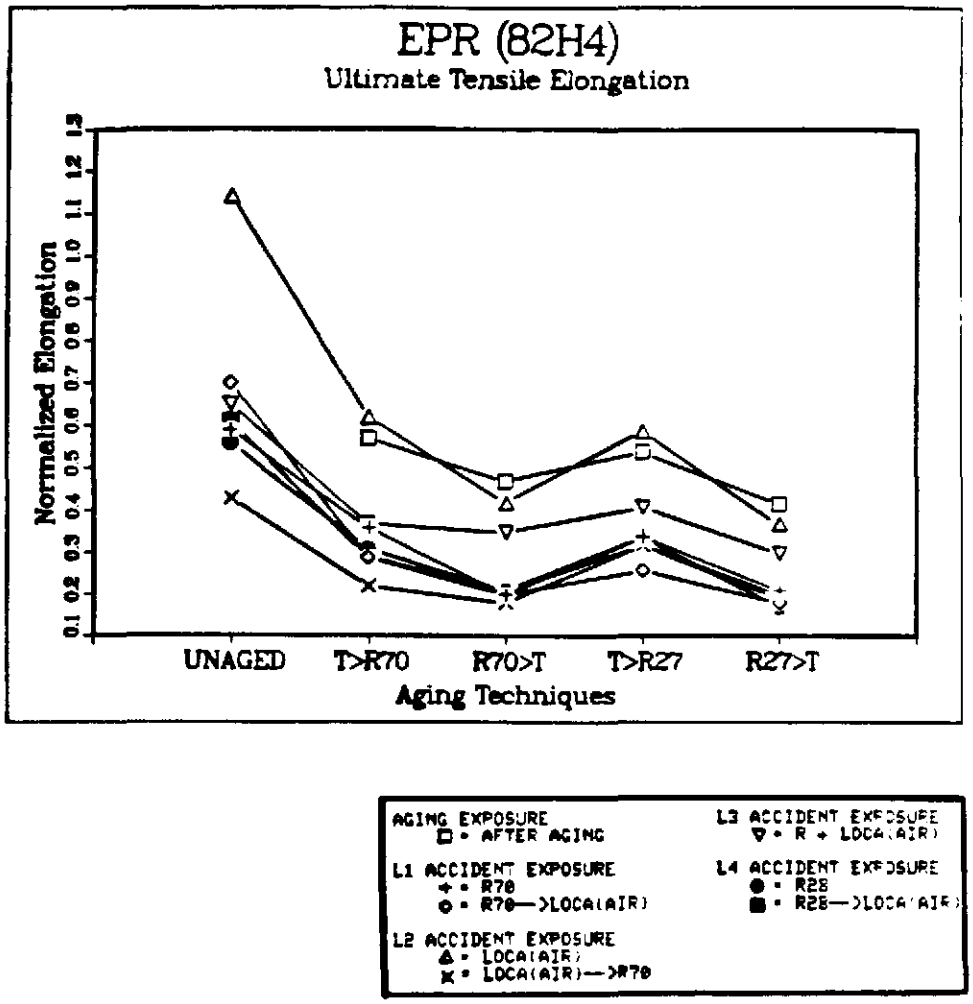


Figure 3.3.39 Ultimate tensile elongation of EPR (82H4) at completion of various phases of the accident exposure. Various aging techniques employed prior to the accident simulation are displayed along the X-axis. R28 refers to an ambient temperature irradiation. R70 refers to a 70°C irradiation. T indicates an elevated temperature exposure. (From Reference 3.3.30)

Table 3.3.6
Appropriate Sequential Qualification Procedures to Simulate a Simultaneous Accident Radiation, Steam, and Air Environment.
(From Reference 3.3.30)

Material	Appropriate Sequential Qualification Procedures
Cross-linked Polyolefins	
XLPO 1	Any sequential simulation
XLPO 2	Any sequential simulation
PEC (82I1)	Any aging simulation followed by the R70→LOCA(air) accident simulation.
Ethylene Propylene Rubbers	
EPDM (82I2)	A R→T aging sequence followed by any accident simulation
EPDM(82I9)	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR (82H4)	Any sequential simulation
EPR 1	A R→T aging sequence followed by a R70→LOCA(air) accident simulation
EPR 2	A R70→T, R27→T, or T→R70 aging sequence followed by R70→LOCA(air) accident simulation
TEFZEL	
TEFZEL 1	Elevated temperature irradiations for sequential aging and accident exposures
TEFZEL 2	R70→T aging sequence followed by an accident simulation
Chlorosulfonated Polyethylene	
HYPALON (82G10)	Any sequential simulation
CSPE	R70→T aging sequence followed by any accident simulation.
O-ring Materials	Any sequential simulation
Connector Materials	
PPS (82H6)	Any sequential simulation
Polydiallylphtalate (82H5)	Any aging simulation followed by the R28→LOCA(air) or R70→LOCA(air) accident simulations
VAMAC	Any sequential simulation
CPE	R70→T aging technique followed by any accident simulation

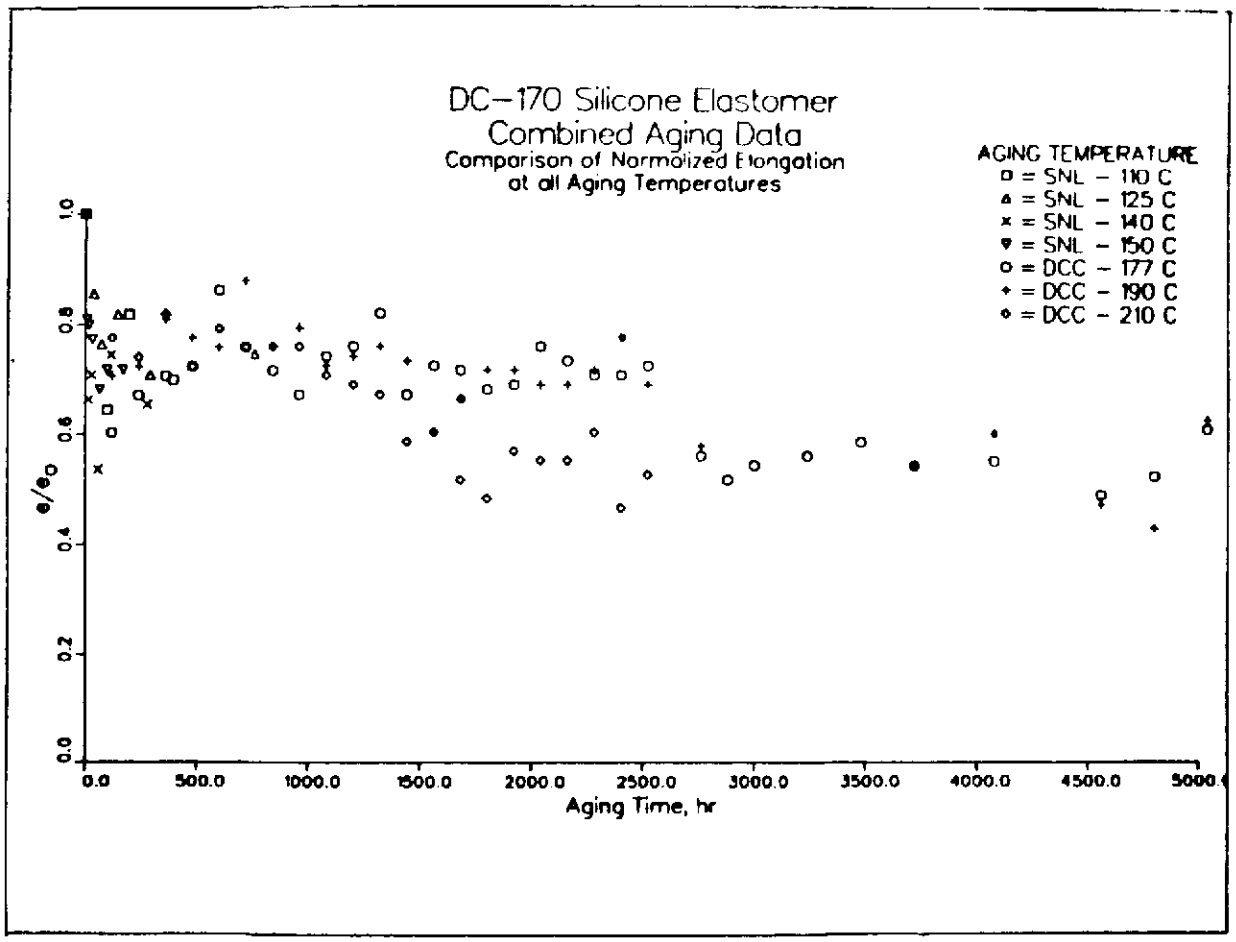


Figure 3.3.40 Comparison of Sandia and Dow Corning Elongation Data. (From Reference 3.3.31).

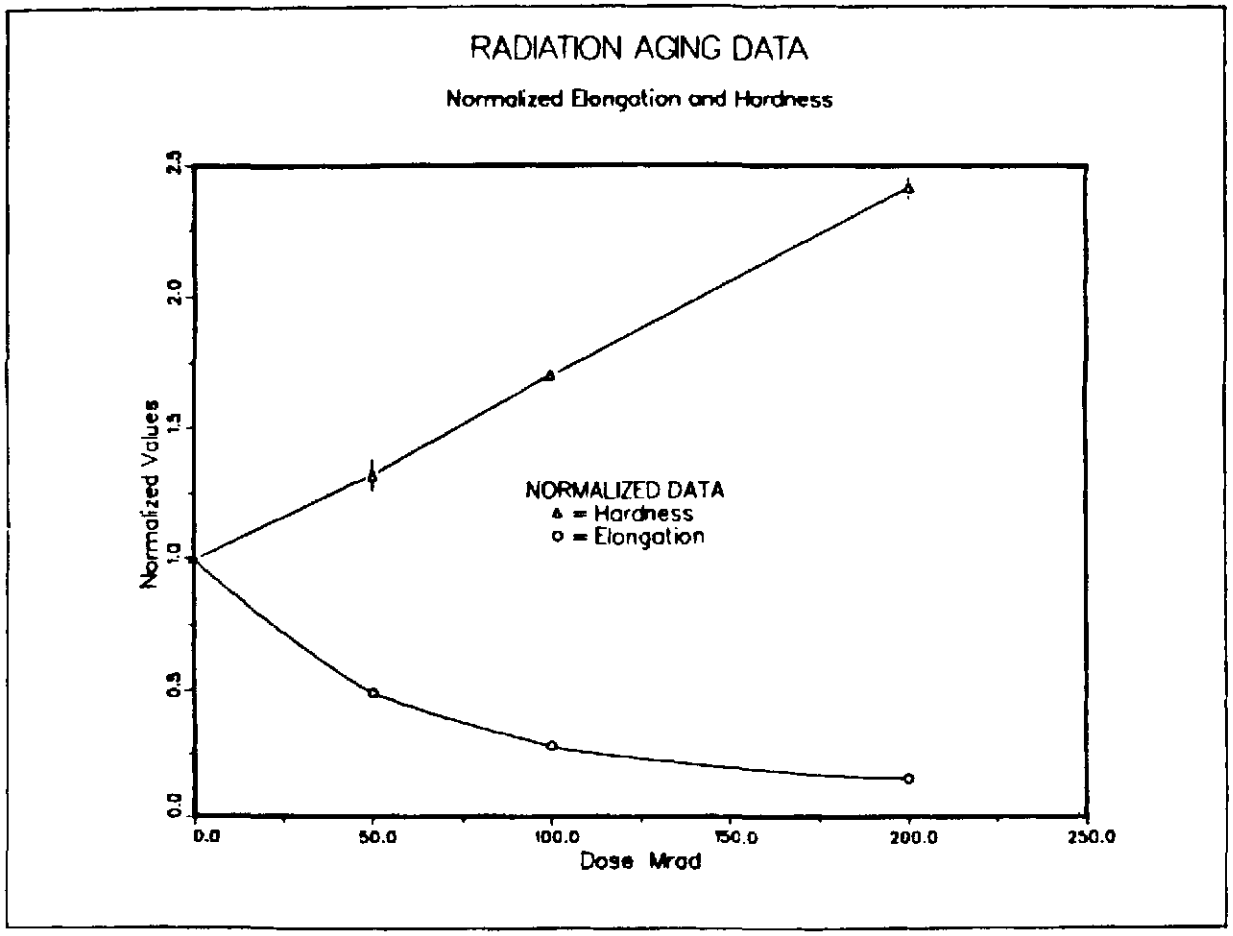


Figure 3.3.41 Radiation Aging Data - Normalized Elongation and Hardness. (From Reference 3.3.31)

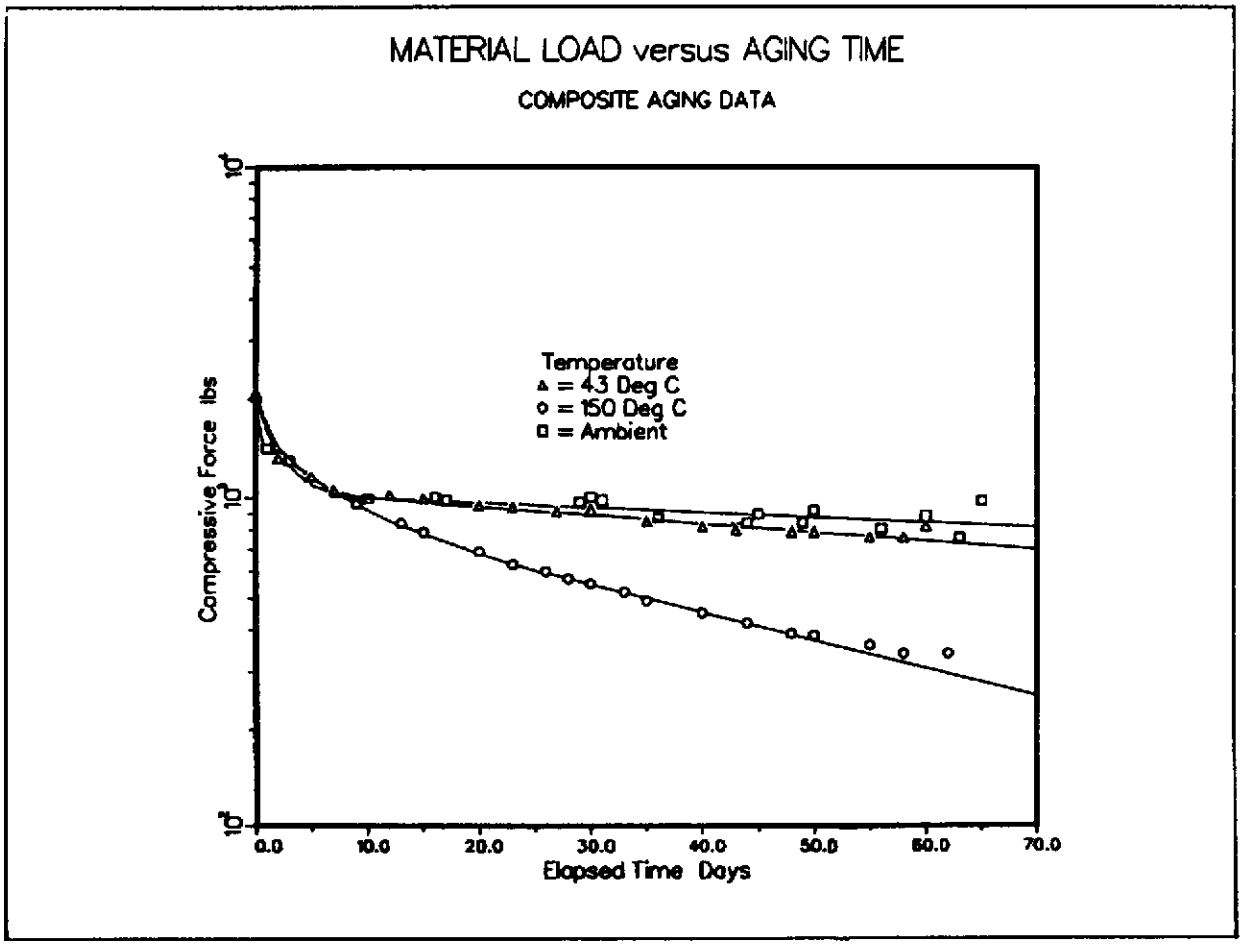


Figure 3.3.42 Compressive force versus elapsed aging time for Sylgard 170 elastomer confined in a 12-hole fixture. (From Reference 3.3.31)

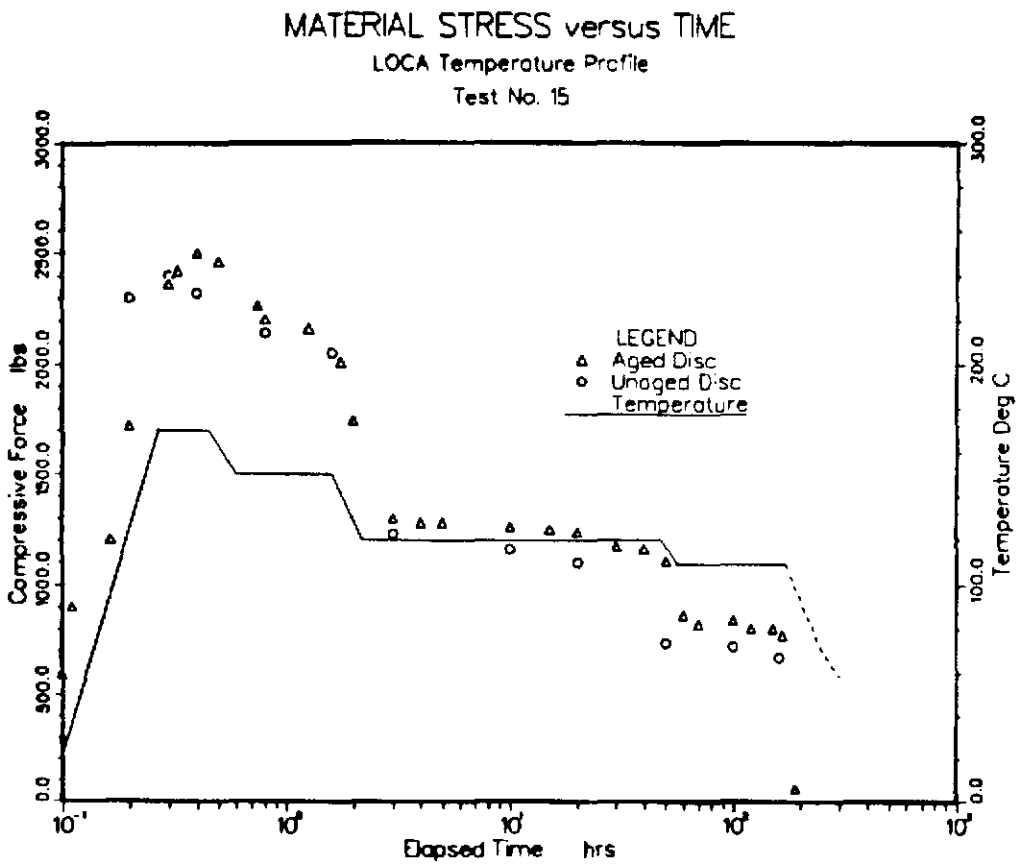


Figure 3.3.43 Compressive force versus elapsed LOCA temperature exposure time for Sylgard 170 elastomer confined in 12-hole fixtures. Aged disc was aged for 187 days at 43°C. Compressive force for aged disc at prior to the LOCA exposure was approximately 500 lbs. The unaged disc was torqued to 2000 lb. compressive force prior to the LOCA exposure. (From Reference 3.3.31)

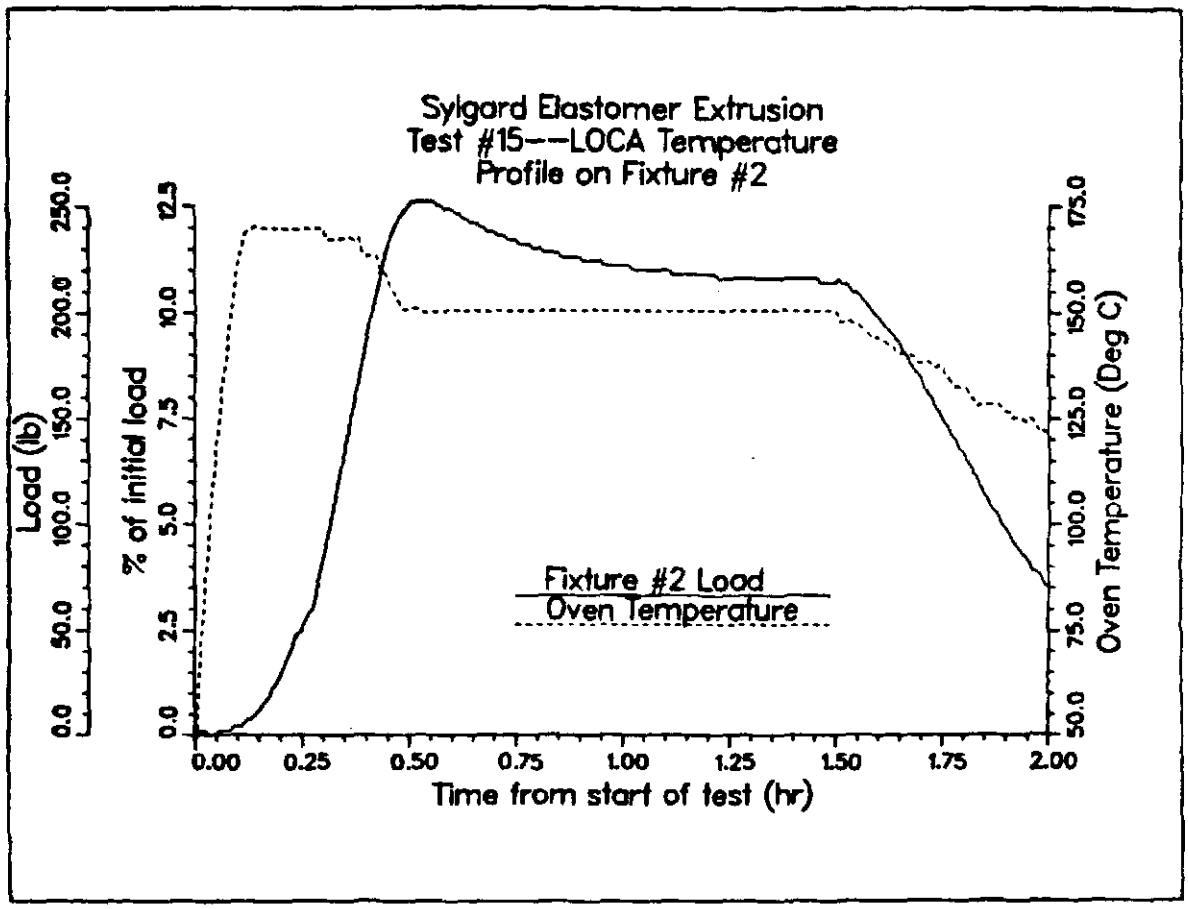


Figure 3.3.44 Compressive force versus elapsed LOCA temperature exposure time for Sylgard 170 elastomer confined in 12-hole fixtures. Aged disc was aged for 135 days at 150°C. After the aging exposure and prior to the LOCA temperature exposure, the compressive load on the disc had decayed to less than 50 lbs. (From Reference 3.3.31)

compressive forces did not exceed 250 lbs during the LOCA temperature simulation. Thus the method and/or extent of aging (i.e., the amount of residual compressive stress at completion of aging) strongly affects the material's response during a subsequent accident simulation.

Reference 3.3.31 noted that stress relief (as illustrated by Figures 3.3.42 through 3.3.44 may proceed by three methods: relaxation, material flow, and material extrusion. In addition it was observed that (1) total relief of any applied compressive stress will occur when any relief path is present, and (2) the compressive relief rate increases, from the ambient value, with increasing environment temperature. Finally it was demonstrated that insulated electrical conductors in intimate contact with the highly confined elastomer at an elevated temperature, in a simulated electrical penetration-grommet seal, will undergo catastrophic damage (stripping) to the insulation material. Based on its material studies, Reference 3.3.31 concludes that stress relaxation and elastomer extrusion should be considered in developing seal designs which require long-term maintenance of compressive forces. Also, the potential for damaging insulation on wires passing through the elastomeric seal should be considered if the material may possibly experience an elevated (LOCA) temperature transient.

International and Other U.S. Programs

Materials

Reference 3.3.32 presents data to compare tensile properties of dumbell specimens to cable specimens during a simultaneous qualification test exposure. The authors note that during the simultaneous test, the physical configuration and environmental conditions were different for the dumbbell specimens as compared to the cable specimens. The dumbbell samples were free of electrical stress while the cable insulation (protected by a jacket) was not exposed to free air, humidity and chemical spray. Reference 3.3.32 reports that the tensile strengths of the dumbbell samples were considerably lower than those from the cable, but the percent elongations were essentially the same.

Tensile properties for two EPR sheet materials were monitored for simultaneous, sequential, and reverse sequential LOCA tests. Results are presented in Reference 3.3.33, NUREG/CR-0023, Vol.2. In the sequential method, the EPR sheets were exposed to a PWR LOCA steam and chemical spray profile after a 150 Mrad irradiation. For the reverse sequential method, the steam and chemical spray profile was before the irradiation. For the simultaneous exposure, irradiation, steam, and chemical spray exposures were combined. With regards to ultimate tensile elongation properties for the two EPR materials, there was little difference between the three accident simulation methods. Differences between the three simulation techniques were observed for the ultimate tensile elongation. The sequential method was most destructive, while the reverse sequential method had a minor effect on the ultimate tensile strength. The degree

of deterioration caused by the simultaneous technique was intermediate between the two sequential methods.

References 3.3.34 and 3.3.35, NUREG/CR-2932, Vol. 1 and 2 (SAND81-2027, Vol. 1 and 2) employed preliminary materials tests to choose the most appropriate aging conditions for an equipment qualification research test on cables. Tensile data for two chemically cross-linked polyolefin insulations were obtained. Both materials were exposed to three different aging exposures

1. A high dose rate (865 krad/h) irradiation to a total dose of 42.5 Mrad followed by thermal aging for 240 hours at 150°C.

2. Thermal aging for 240 hours at 150°C followed by a high dose rate (865 krad/h) irradiation to a total dose of 42.5 Mrad.

3. A low dose rate (48 krad/h) irradiation to a total dose of 42.5 Mrad followed by thermal aging for 240 hours at 150°C.

At completion of the aging exposures, tensile properties for some of the specimens were measured. The remaining specimens were irradiated at 865 krad/h, in increments of 50 Mrad, until a total aging and accident dose of 200 Mrad was achieved. At each 50 Mrad increment, tensile properties were monitored. Results for one of the materials are presented in Figures 3.3.45 and 3.3.46. The effects of alternative aging techniques is clearly evident at completion of the aging exposures. However, by completion of the accident irradiation there was very little life remaining and all three aged sample groups were approaching the same level of mechanical degradation. The second material exhibited similar behavior but with much larger scatter in the data. Based on these material test results, References 3.3.34 and 3.3.35 employed a low dose rate followed by thermal aging sequence as part of qualification research exposure for electrical cables.

Reference 3.3.36, NUREG/CP-0041, Vol. 5, employed materials testing to examine three LOCA research questions:

1. What is the effect on material properties of the LOCA testing period?

2. What is the effect on material properties of the heating and cooling rate during the LOCA transients?

3. What is the effect of oxygen presence during simultaneous and sequential LOCA simulations?

Figure 3.3.47 illustrates the effect of LOCA testing period on various materials and demonstrates that the importance of LOCA testing period depends on the tested material. Reference 3.3.36 concludes that the mechanical properties of HYPALON are influenced by the LOCA exposure

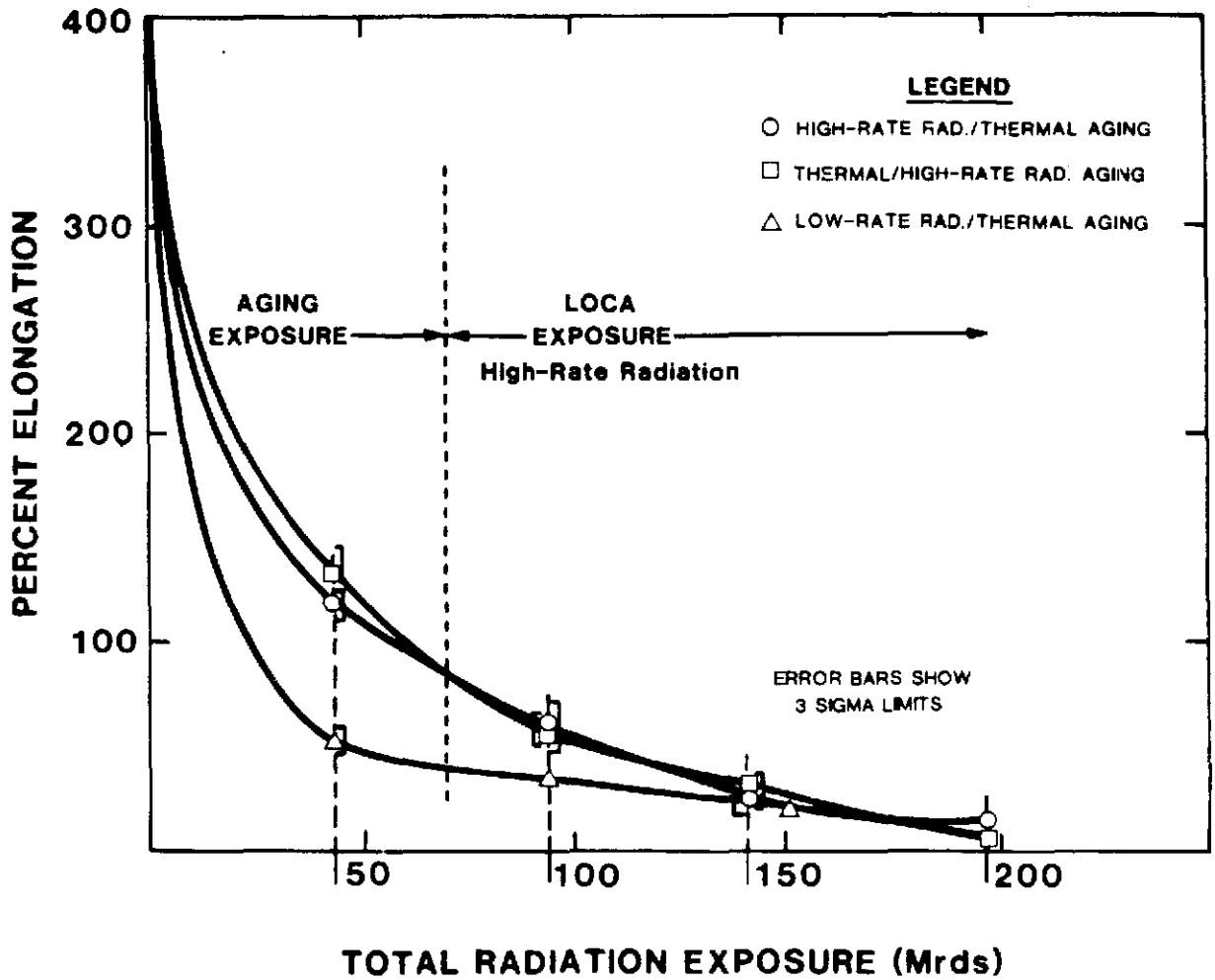


Figure 3.3.45 Ultimate tensile elongation for a chemically crosslinked polyolefin insulation after various phases of aging and accident simulations. (From Reference 3.3.35)

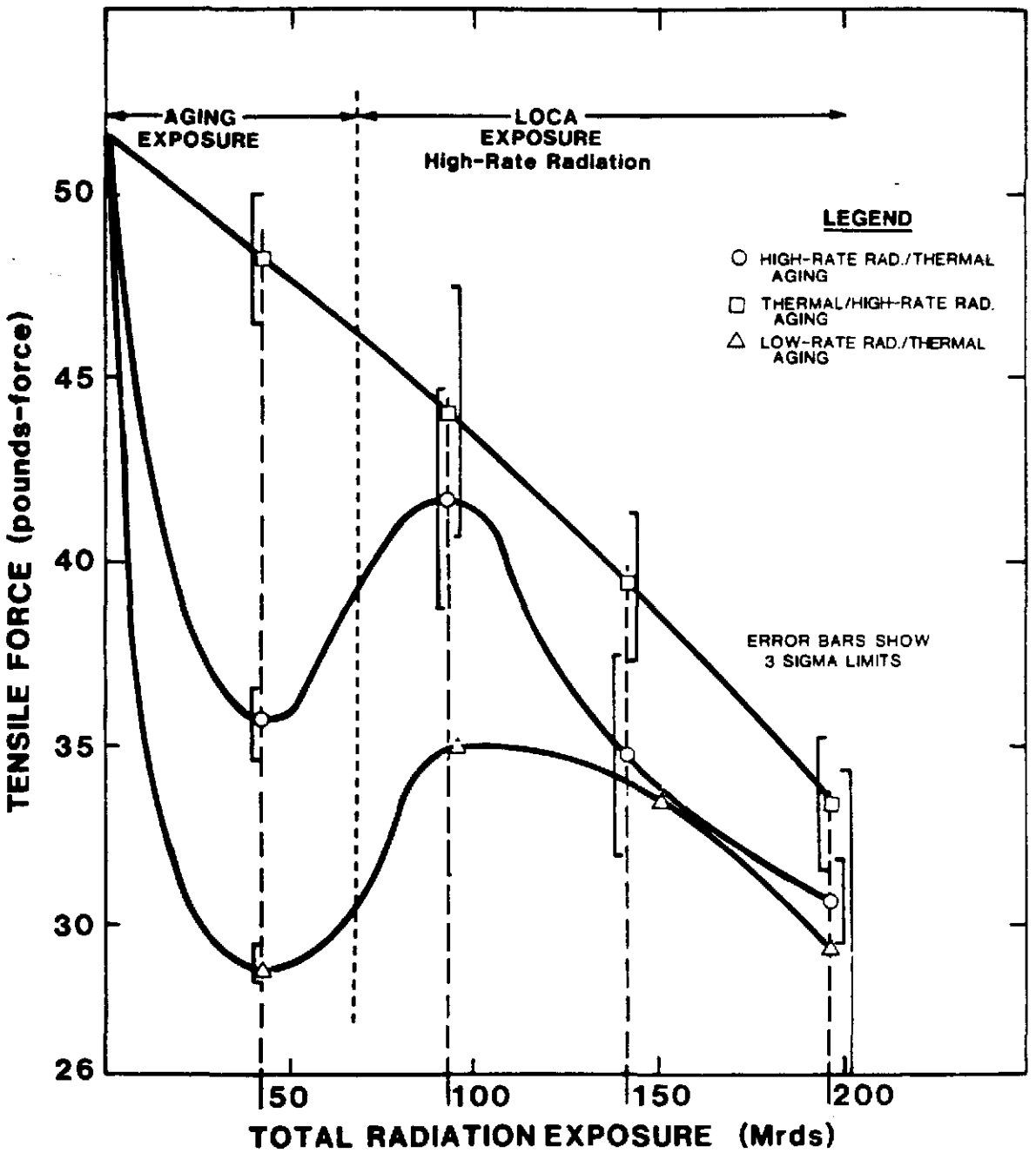


Figure 3.3.46 Ultimate tensile force for a chemically crosslinked polyolefin insulation after various phases of aging and accident simulations. (From Reference 3.3.35)

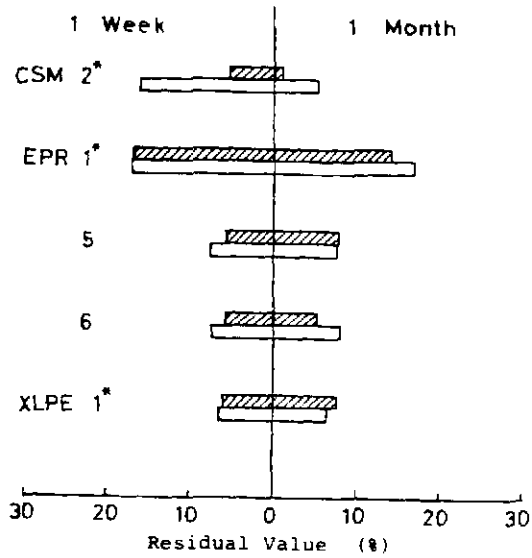


Figure 3.3.47 Residual value of toughness obtained for various kinds of insulating materials under 1 week and 1 month tests. (From Reference 3.3.36)

time. However, those of EPR and crosslinked polyethylene are mainly influenced by radiation. This suggests that EPR and XLPE are stable against exposure to chemical spray and heat (120°C) in a saturated steam (no air) LOCA exposure.

Figure 3.3.48 shows the log mean of volume resistivity and residual toughness of six kinds of fire-retardant EPR at various stages of the LOCA tests. Volume resistivity and residual toughness decrease with time. The values obtained under one week and one month are plotted in the figure. For EPR, exposure time does not significantly influence either the mechanical nor the electrical properties.

Reference 3.3.36 also examined the effect of the heating and cooling rate during the LOCA transients. It concludes that the heating and cooling rate during a LOCA transient has little or no influence on the deterioration of the insulating materials EPR, XLPE, and Silicone that were studied.

Tensile property and volume resistivity data after LOCA testing (with and without air) is also presented in Reference 3.3.36 for several polymer materials. Both PWR and BWR LOCA conditions were simulated. The total accident radiation exposure for PWR conditions was 150 Mrad while the

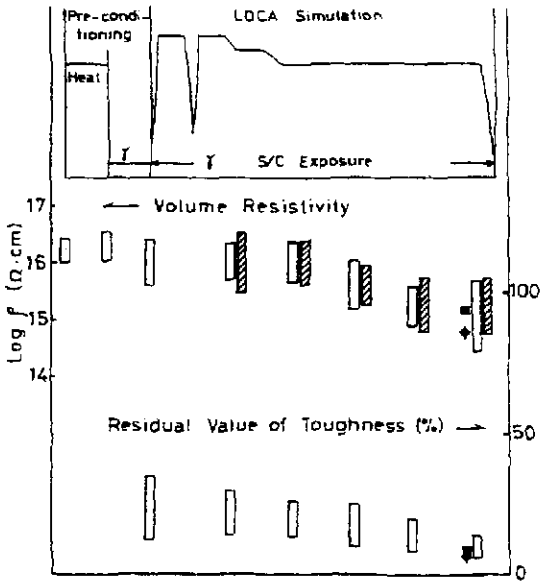
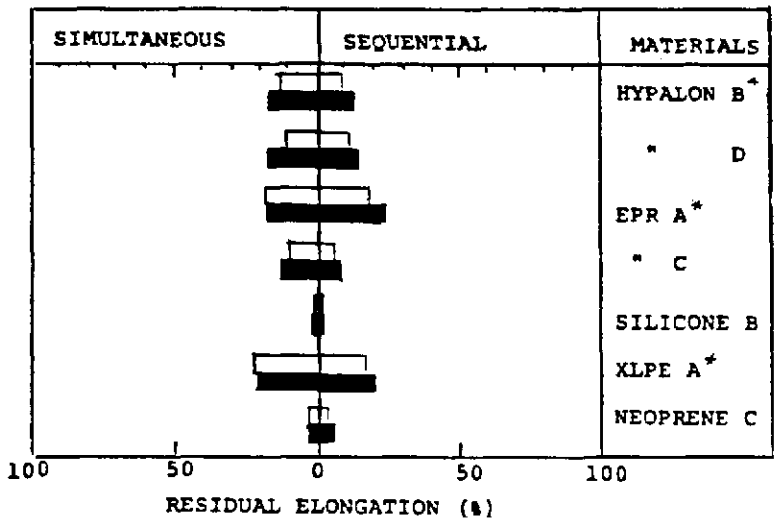
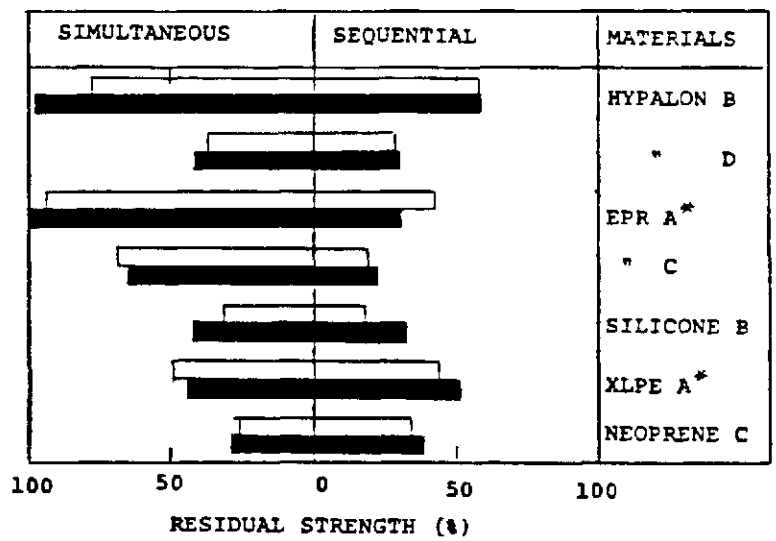


Figure 3.3.48 Log mean values of volume resistivity and residual value of toughness at various stages of LOCA simulation tests. (From Reference 3.3.36)

total accident radiation exposure for the BWR conditions was 26 Mrad. Simultaneous and sequential accident simulation techniques were employed. Both preconditioned and nonpreconditioned specimens were tested. Preconditioning for all specimens was a sequential exposure of thermal aging followed by irradiation (50 Mrad).

Figures 3.3.49 and 3.3.50 illustrate the comparison of mechanical properties for the test materials after simultaneous and sequential PWR accident simulations without and with air, respectively. Residual elongation of the insulating materials, except for silicone rubber, decreased significantly under air containing conditions in both simultaneous and sequential PWR simulation methods. Note that when air is not included during the PWR accident simulations that sequential techniques seem appropriate for most of the materials including EPRA and XLPE A. However, when air is included during the PWR accident simulations, simultaneous techniques become more severe. Figures 3.3.51 and 3.3.52 illustrate the comparison of mechanical properties for the test materials after simultaneous and sequential BWR accident simulations without and with air, respectively. Some oxygen effects were also noted.





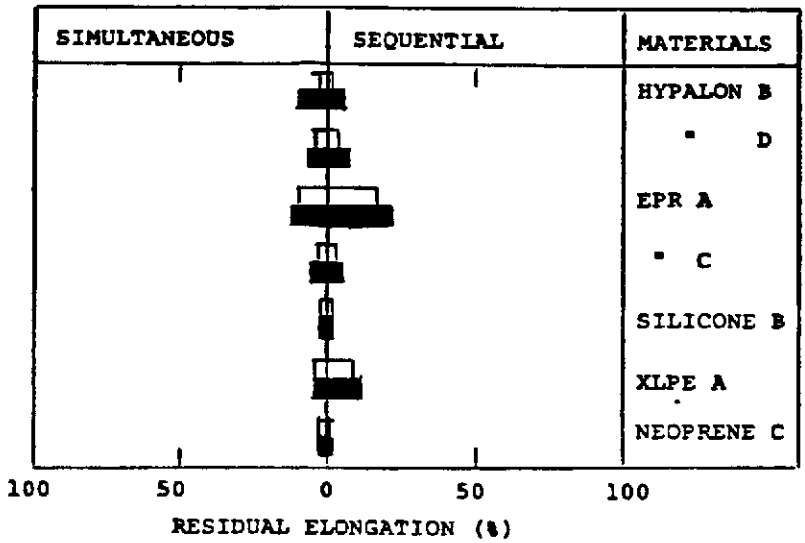
Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (PWR conditions).

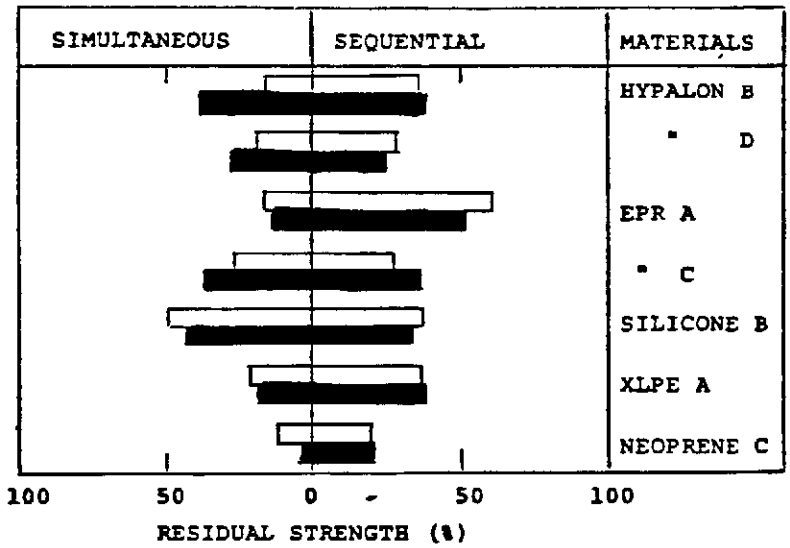
☐ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 3.3.49 (From Reference 3.3.36)



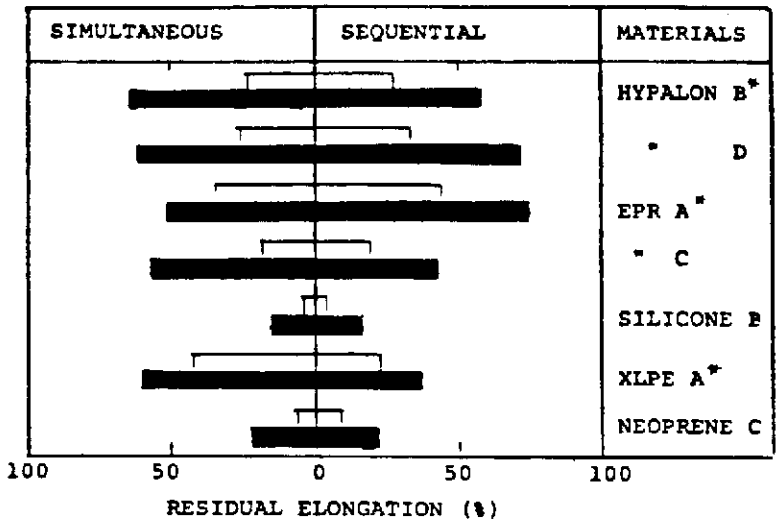


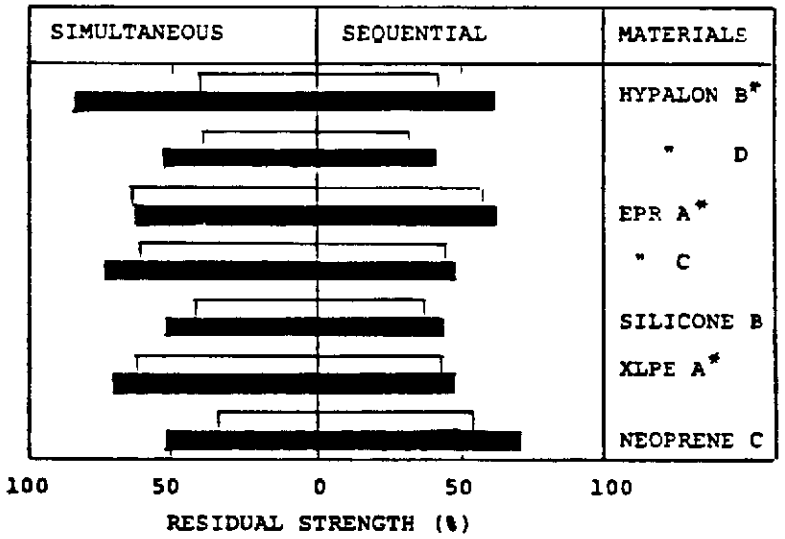
- COMPARISON OF MECHANICAL PROPERTIES OF THE MATERIALS AFTER SIMULTANEOUS AND SEQUENTIAL LOCA TESTING (PWR CONDITIONS CONTAINING AIR)

PRE-CONDITIONED

NOT PRE-CONDITIONED

Figure 3.3.50 (From Reference 3.3.36)





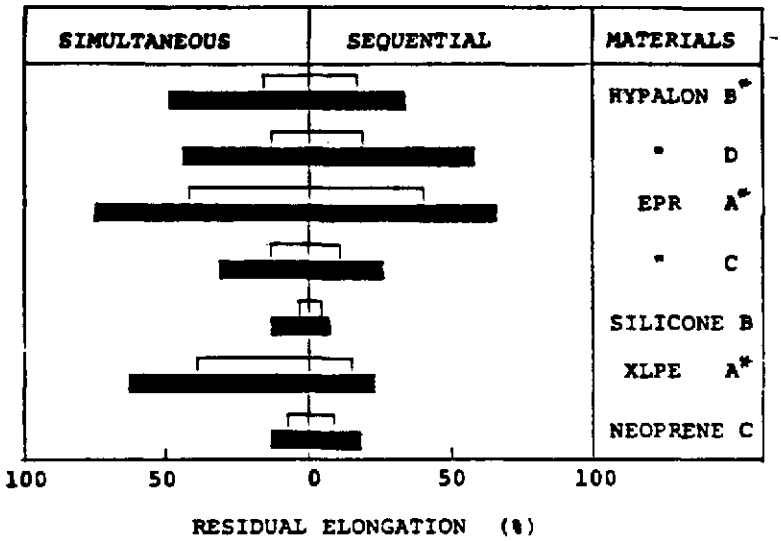
Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions).

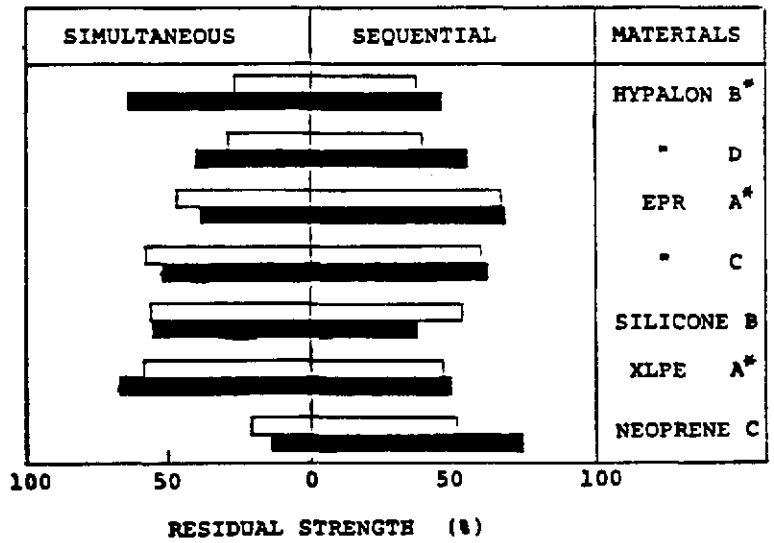
☐ ; Pre-conditioned

■ ; Not pre-conditioned

Without air

Figure 3.3.51 (From Reference 3.3.36)





Comparison of mechanical properties of the materials after simultaneous and sequential LOCA testing (BWR conditions containing air).

☐ ; Pre-conditioned

■ ; Not pre-conditioned

Figure 3.3.52 (From Reference 3.3.36)

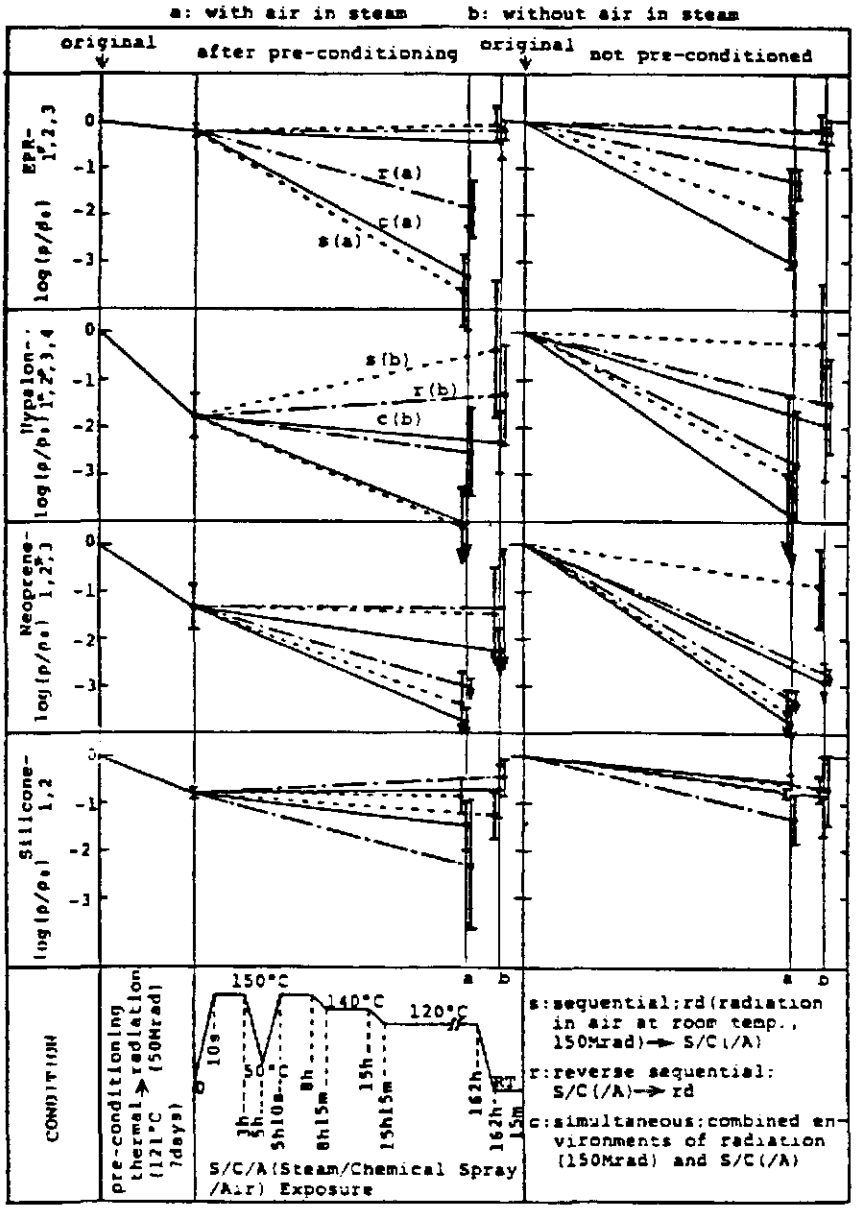
Figure 3.3.53 presents Reference 3.3.36 volume resistivity data for several insulating materials at completion of LOCA simulation tests. Simulations were performed both with and without air in the steam. For EPR, Hypalon, and Neoprene, the volume resistivity was degraded by two to three orders of magnitude by the air containing simulations but only degraded by zero to two orders of magnitude by the non-air simulations. For the air containing simulations, the simultaneous method was most degrading for the resistivity of EPR, Hypalon, and Neoprene unaged specimens while the reverse sequential method was least degrading. For aged specimens, the simultaneous and sequential simultaneous produced similar degradation of the resistivity. Silicone volume resistivity was not strongly affected by the presence or absence of air during the LOCA simulation. Reference 3.3.36 noted a relationship between swelling of materials during the LOCA tests and decreases in volume resistivity. It states: "Materials which show a larger drop in volume resistivity swelled easily under LOCA conditions, especially with air. This suggests that radiation-induced oxidation during LOCA tests will result in the formation of hydrophilic part in the materials which results in swelling."

Reference 3.3.37 provides additional data comparing simultaneous and sequential LOCA simulations. PWR LOCA simulations were performed for both air and non-air conditions. Figures 3.3.54 and 3.3.55 illustrate tensile strength test results at completion of the accident simulations. The authors note that when the LOCA environments were simulated by saturated steam conditions, most materials were slightly more degraded in tensile strength by sequential simulation techniques. However, when air was included in the steam environment, then the simultaneous simulation techniques were slightly more severe.

Reference 3.3.37 also provides degradation behavior data for a Hypalon and an EPR material. Results are illustrated in Figures 3.3.56 and 3.3.57. Reference 3.3.37 notes for both materials that the ultimate tensile elongation was dominated by the total radiation dose and affected only slightly by the test sequence. However, the ultimate tensile strength for both materials was affected considerably by the test sequence. Simultaneous simulations with air present were the most degrading. Simultaneous simulations without air were least degrading.

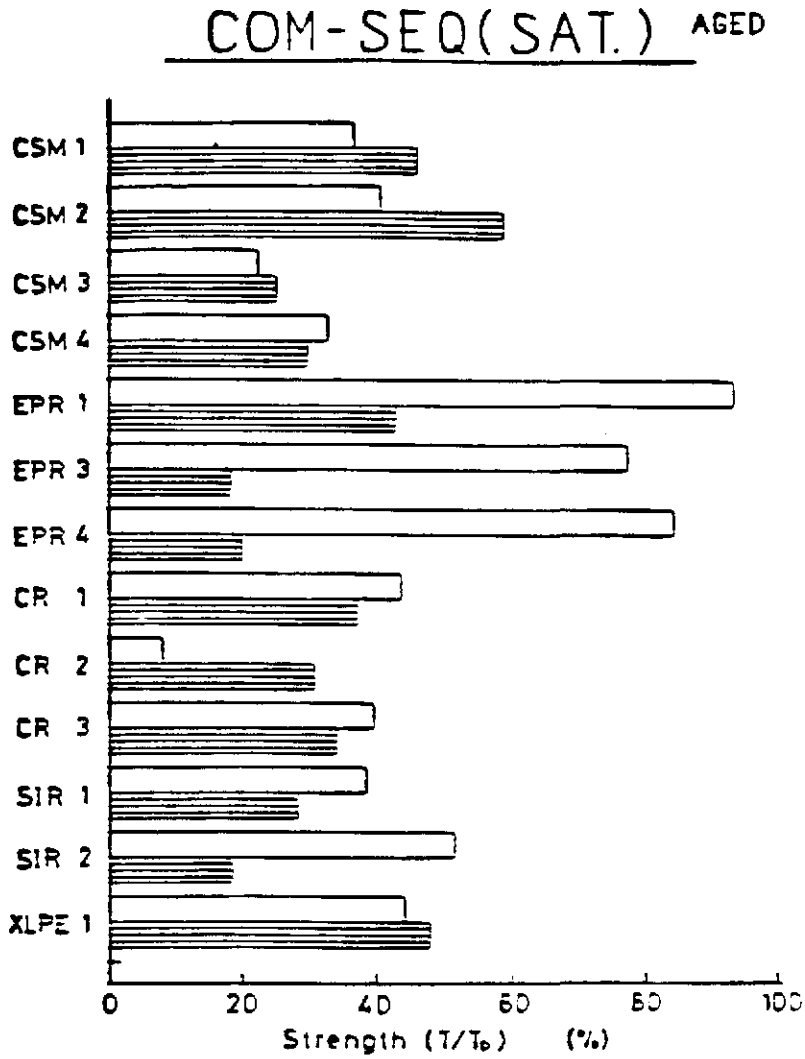
Reference 3.3.38 compares the behavior of cable insulation and jacket materials to the electrical performance of cables. Accident simulations are performed with and without air and for two different time periods (one week and three months). The purpose of varying the accident length was to evaluate the ability to accelerate accident simulations. Air presence was considered an important variable to be considered in this evaluation. For insulation and jacket sheet materials tensile and volume resistivity properties were monitored. For the cable specimens insulation resistance was determined.

For sheet specimens of EPR insulation and Hypalon jacket materials, air presence was important to both the tensile and volume resistivity



Log mean values of volume resistivity for some insulating materials after simultaneous, sequential and reverse sequential LOCA tests.

Figure 3.3.53 (From Reference 3.3.36)

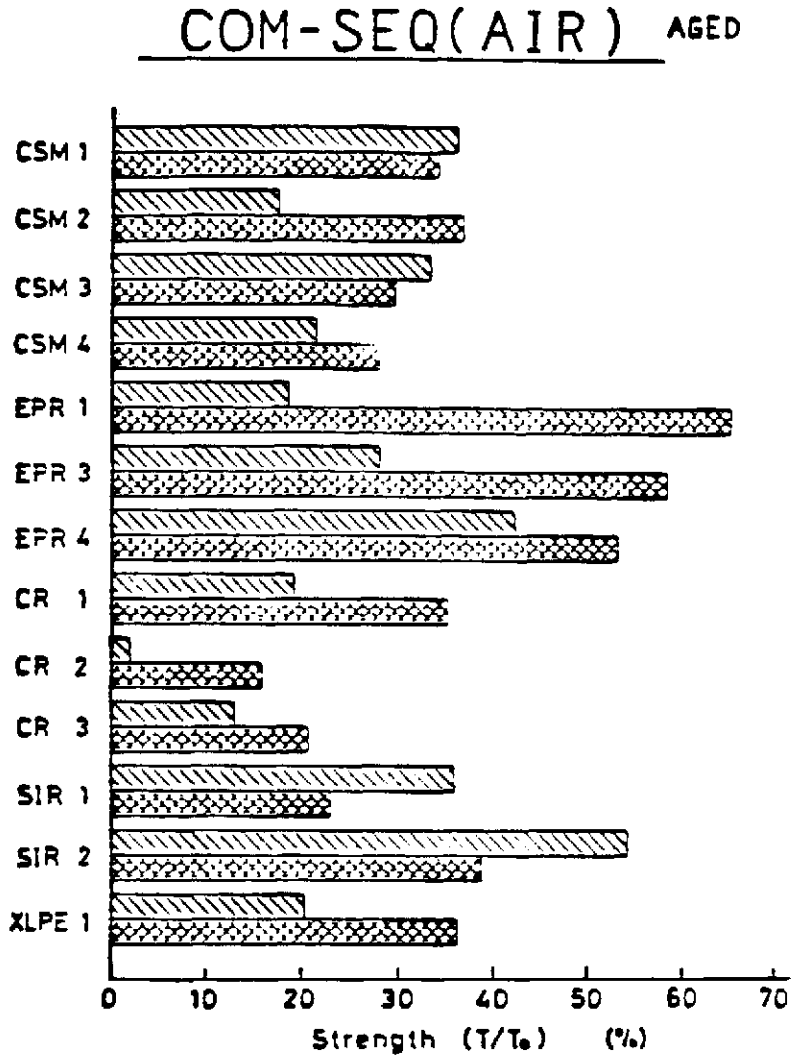


Comparison of Strength of Pre-conditioned (Aged) Materials Exposed to Simultaneous and Sequential LOCA testing for PWR conditions simulated by saturated steam.

Simultaneous testing

Sequential testing

Figure 3.3.54 (From Reference 3.3.37)

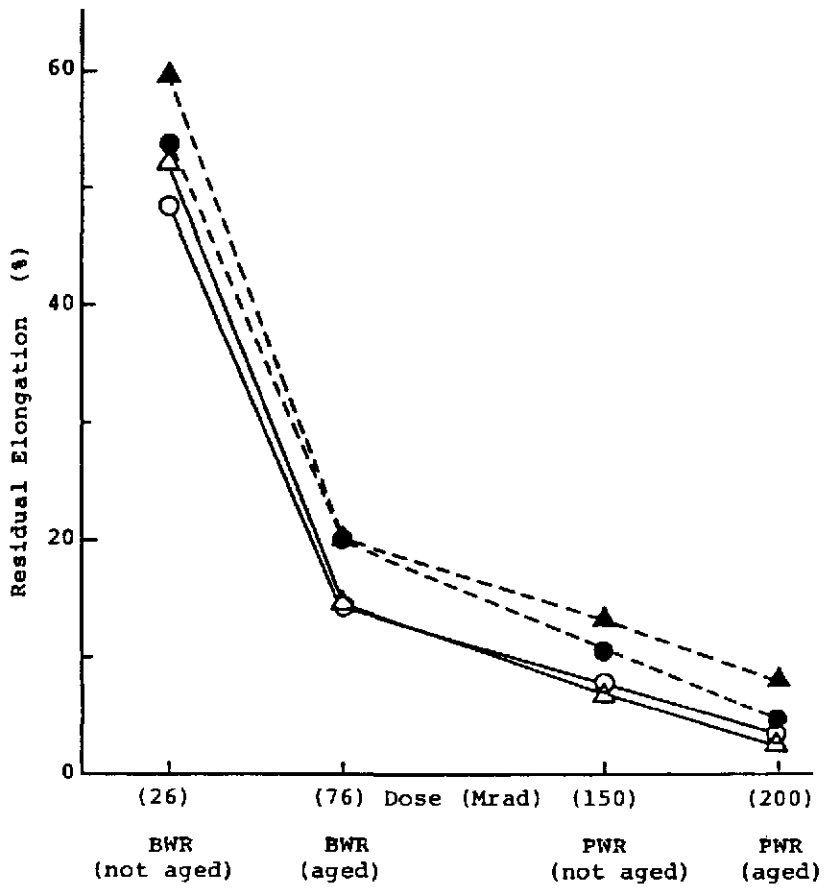


Comparison of Strength of Pre-conditioned (Aged) Materials Exposed to Simultaneous and Sequential LOCA testing for PWR conditions simulated by air containing steam.

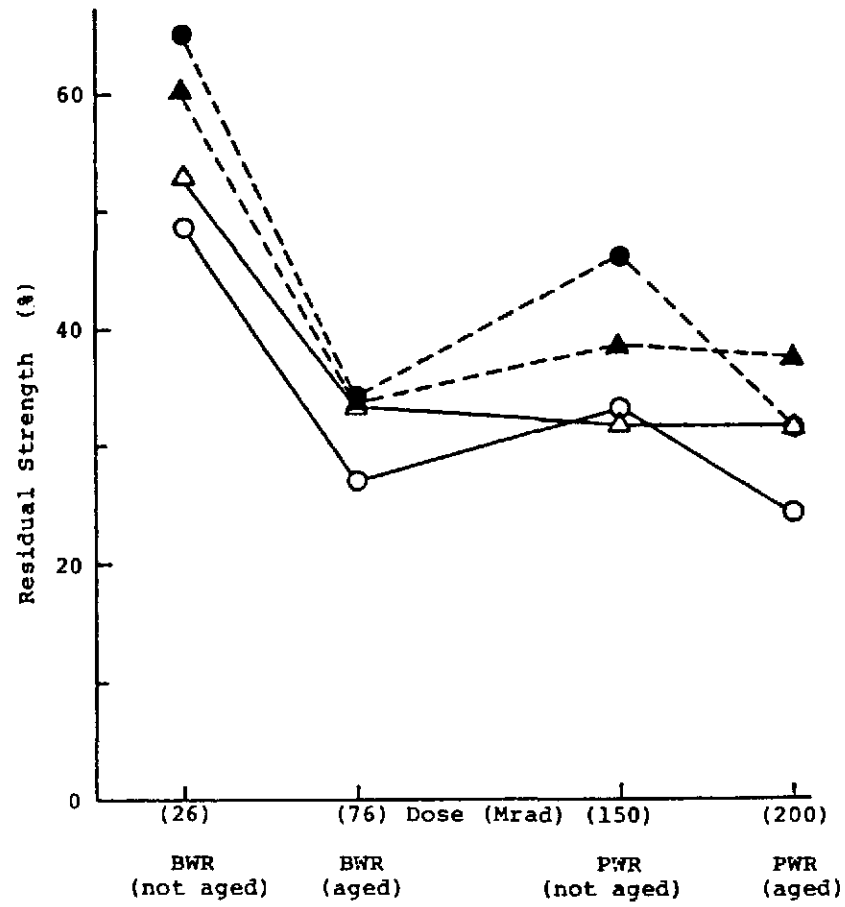
Simultaneous testing

Sequential testing

Figure 3.3.55 (From Reference 3.3.37)



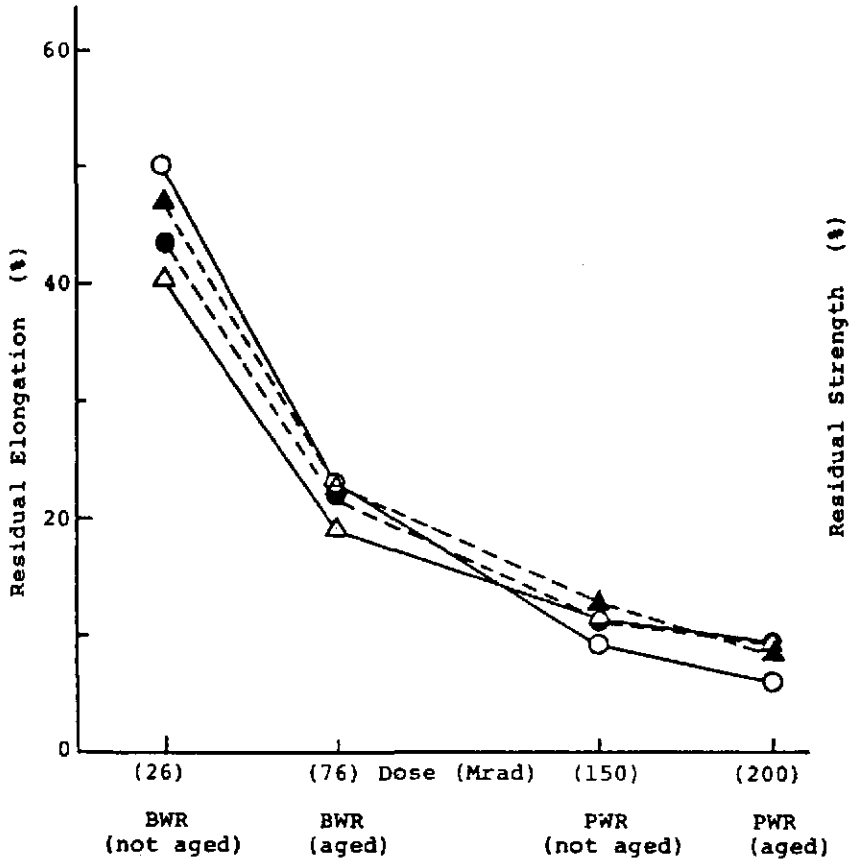
Changes in Elongation of Hypalon under various simulated LOCA conditions.



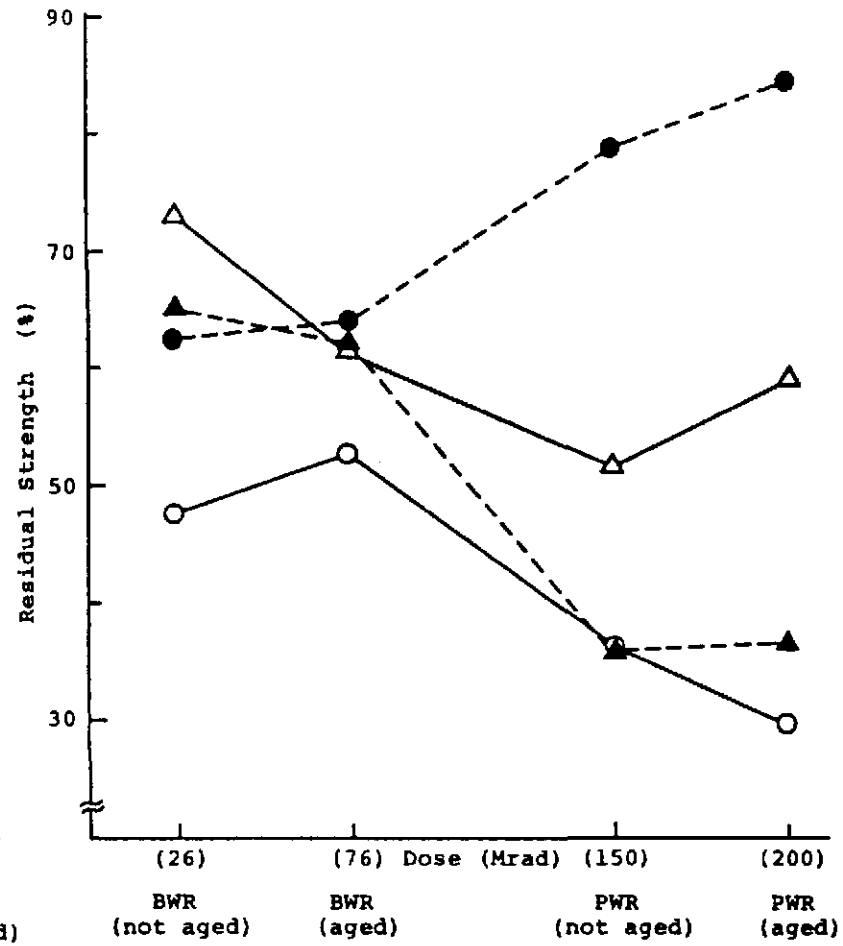
Changes in Tensile Strength of Hypalon under various simulated LOCA conditions.

Symbol	Method	Air
—○—	Simultaneous	Yes
- -●- -	"	No
—△—	Sequential	Yes
- -▲- -	"	No

Figure 3.3.56 (From Reference 3.3.37)



Changes in Elongation of EPR under various simulated LOCA conditions.



Changes in Tensile Strength of EPR under various simulated LOCA conditions.

Symbol	Method	Air
—○—	Simultaneous	Yes
--●--	"	No
—△—	Sequential	Yes
--▲--	"	No

Figure 3.3.57 (From Reference 3.3.37)

parameters. Table 3.3.7 summarizes relevant data. In steam with air environments, the volume resistivity was substantially more degraded by the three month LOCA exposure than by the one week LOCA exposure. This effect was not noted for the non-air exposures.

For cable specimens, insulation resistance did not exhibit similar large effects. Figures 3.3.58 and 3.3.59 demonstrate that insulation resistance differences between air and non-air LOCA simulations are small. Moreover, Reference 3.3.38 illustrates that the cable insulation material experienced less degradation of the elongation than did the sheet material (Figure 3.3.60). Reference 3.3.38 concludes that the cable construction enhances the performance of the insulation during air containing LOCA simulations.

Reference 3.3.39 discusses simultaneous and sequential tests performed in France on nine polymer materials. Mechanical and electrical measurements for most of the materials indicated that sequential accident exposures were more severe than simultaneous accident exposures. An exception was Hypalon. For this material, simultaneous testing was most severe. Test materials included two polyamide-imide varnishes used in motors and coils; one epoxy resin, glass fiber insulation, three elastomeric materials, a silicon thermoset, VAMAC and others.

Issue Findings

Materials

Testing of materials rather than Class 1E components is an economical method for evaluating numerous experimental variables for several types of materials. Insights gained through this experimental approach are amply demonstrated by the materials' tests summarized in this section. Testing has been performed to evaluate:

1. the importance of oxygen effects during accident simulations.

2. the importance of accident synergistic effects.

3. the sensitivity of accident results to age preconditioning techniques.

4. the importance of accident test periods.

5. the importance of irradiation temperatures during sequential accident simulations.

6. the effect of heating and cooling rates during LOCA transients.

The insights gained via these test programs are useful towards resolving which test simplifications are reasonable for qualification of Class 1E equipment.

Table 3.3.7 (From Reference 3.3.38)

Rate of decrease in ultimate elongation under LOCA condition. (1/100Mrad)

	Non Air		Air 0.05MPa	
Sample	One Week	Three Months	One Week	Three Months
EPR-A	0.40	0.40	0.83	1.57
EPR-B	0.46	0.46	0.74	1.46
CSPE-A	0.56	1.00	0.77	1.60
CSPE-B	0.80	1.11	1.09	2.14

Decay curves are taken as Maxwellian.

Rate of decrease in volume resistivity under LOCA condition. (1/100 Mrad)

	Non Air		Air 0.05MPa	
Sample	One Week	Three Months	One Week	Three Months
EPR-A	<1.0	<1.0	2.2	7.1
EPR-B	<1.0	<1.0	2.2	11.2
EPR-C	<1.0	<1.0	1.3	6.6

Decay curves are taken as Maxwellian.

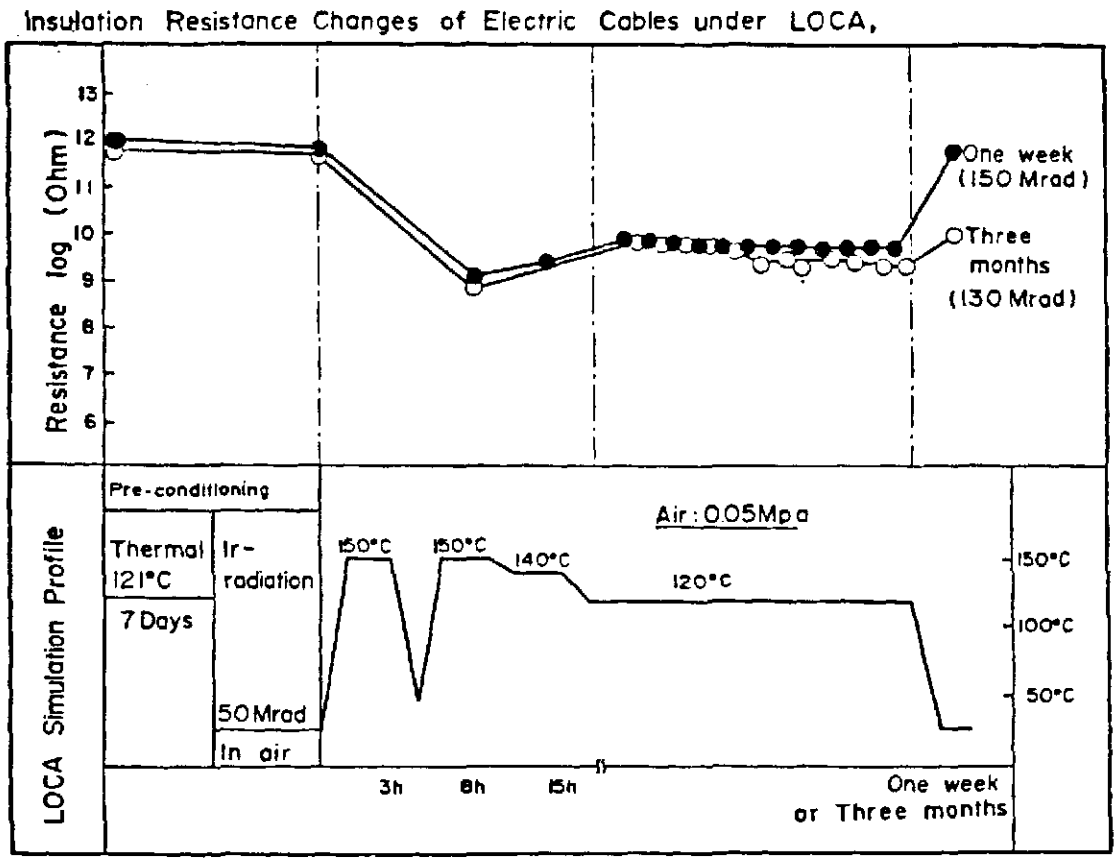


Figure 3.3.58 Electrical cable performance during an air LOCA exposure. Figure 1.6.17 illustrates cable performance during a non-LOCA exposure. (From Reference 3.3.38)

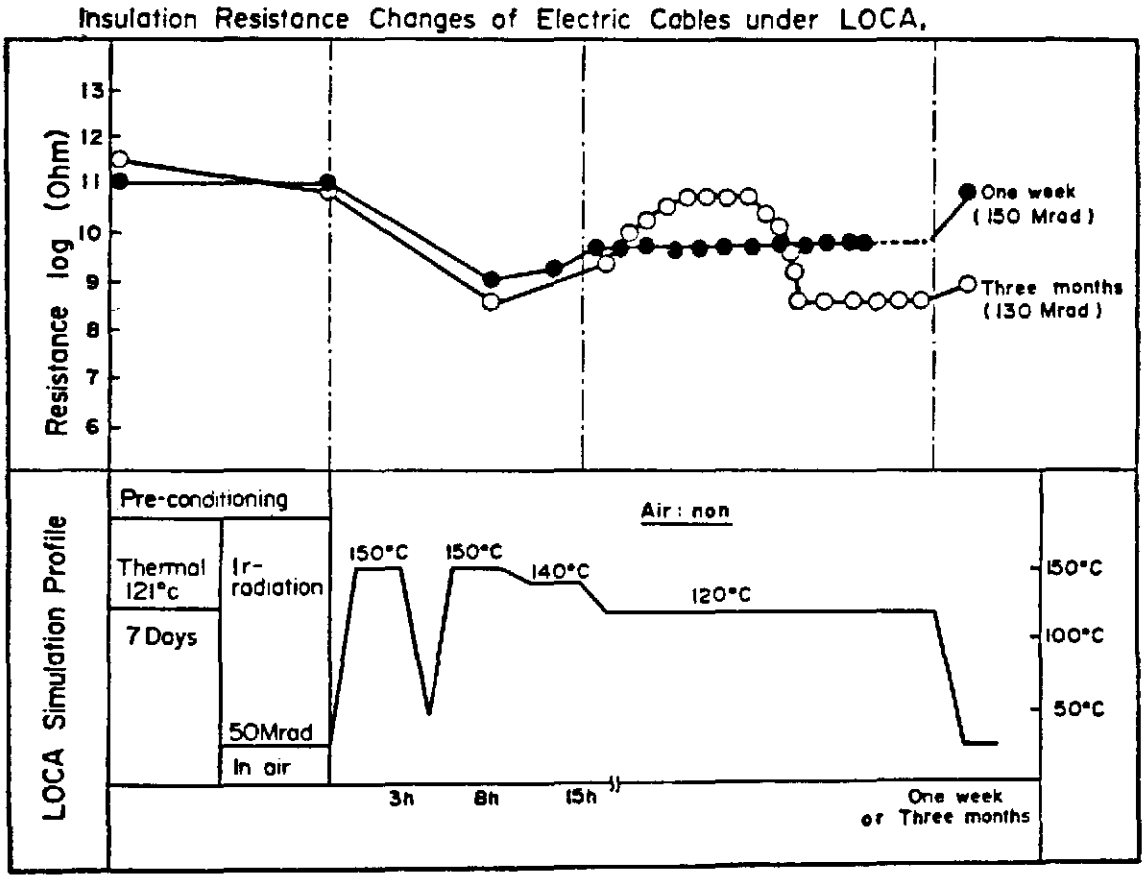


Figure 3.3.59 Electrical cable performance during a non-air LOCA exposure. Figure 1.6.16 illustrates cable performance during an air LOCA exposure (From Reference 3.3.38)

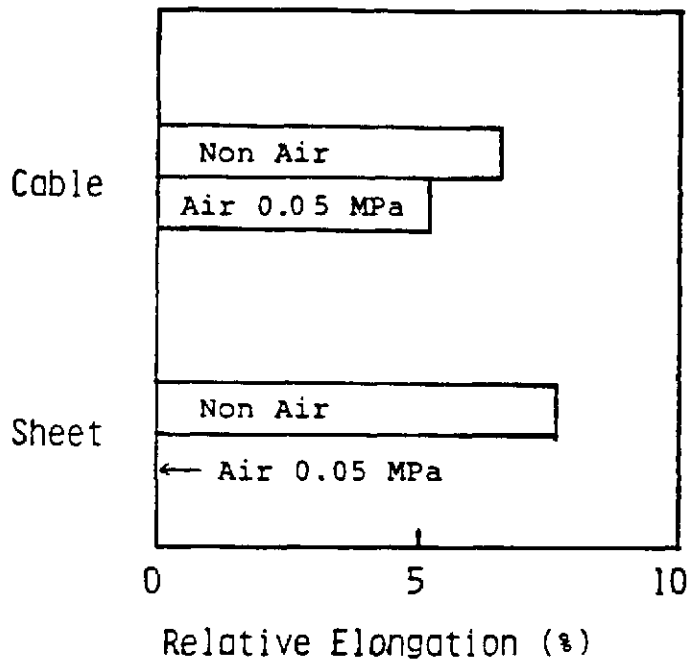


Figure 3.3.60 Comparison between degradation of sheet shape samples and degradation of cables under three months LOCA test. (From Reference 3.3.38)

Of importance, is the question whether material test performance is representative of equipment performance. References 3.3.28, 3.3.29, 3.3.32, 3.3.34, 3.3.35, and 3.3.38 most directly address this issue. In Reference 3.3.28 materials behavior during LOCA simulations was employed to explain EPR cable functional degradation that had occurred. Reference 3.3.29 used materials data to explain why similar degradation was not noted for XLPO cables. Reference 3.3.32 compared tensile properties of dumbbell specimens with tensile properties of insulation removed from a cable. Differences for tensile strength were noted but not for tensile elongation. References 3.3.34 and 3.3.35 employed material tests in order to choose the most appropriate aging sequence for cable qualification research tests. Finally, Reference 3.3.38 compared electrical and mechanical properties for cable insulation and sheets of the same material. The protective nature of the cable configuration reduced the degradation of the cable insulation compared to the sheet material during air containing LOCA simulations. These results suggest that materials studies are very valuable, but may not predict exactly equipment performance.

In summary, the findings within this subissue are:

1. The oxygen concentration during a LOCA simulation can have a significant influence on the chemical and physical changes occurring in certain polymeric materials.

2. Moisture absorption of some materials may depend strongly on the accelerated preconditioning applied prior to a simultaneous accident simulation.

3. For those reactor containments which are not inerted, a conservative accident simulation for qualifying materials would include air in the LOCA/MSLB test chamber.

4. Stress relaxation and elastomer extrusion should be considered in developing seal designs which require maintaining long-term compressive force.

References

Materials

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Materials, Other Programs

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Related Experiences

Sometimes, during the process of testing equipment, auxiliary components -- used in support of the test -- behave unexpectedly. At times, this unexpected behavior has implications beyond the scope of the test. For example, Reference 3.3.40 discusses the unexpected sensitivity of Raychem sleeving to tearing. This anomaly was observed during screening tests made to assess the high temperature capabilities of auxiliary materials to be used in a research test on pressure switches. The screening tests involved exposing mock-up electrical connection assemblies (flexible hose and applicable hardware covered with heat-shrinkable tubing) to 140°C for 17 hours in a air-circulating oven. Unexpectedly, splitting of the heat-shrinkable tubing was observed in three of the four assemblies tested. The heat-shrinkable material used in the screening test had been procured five years previously, and had been stored in plastic bags at ambient conditions.

A follow-up test was made on "new" material using the same environmental conditions that had been used initially. In addition, the hardware (nuts, flanges) and tubing was carefully inspected to detect sharp edges or roughness in the metal and nicks or tears on the ends of the heat-shrinkable tubing. A tear did develop in one tube, however, during heat shrinkage, propagating from a minute tear on the edge and running the length of the tube. The assemblies were completed and subjected to the 140°C temperature. No tube failures were observed.

A third test, using two assemblies with "new" and two with "old" tubing, was conducted at 140°C for 17 hours; visual inspection revealed no damage. The same assemblies were later exposed to a LOCA (steam) test. No tearing, splitting, cracking, or blistering was observed in either the "new" or "old" material during the posttest inspections.

The results of the tests show

1. This material exhibits a very low hot tear strength and any nick, tear, or puncture can lead to failure.

2. The "old" material exhibits the same behavior as the "new" material under similar fabrication procedures.

3. There was no indication that storage for five years at ambient conditions had detrimental effects on the "old" material.

Reference 3.3.41 relates several instances in which Anaconda metal host (Type NWC-6/1/82) exhibited abnormal behavior. The unaged flexible metal hose was used as support equipment only and was never intended to be an active part of the test. However, during posttest inspections following four environmental (steam) tests, the flexible hose insulation was observed to have experienced abnormal shrinkage, bubbling, excessive ballooning, and some minor tearing. The effect of these exposures on unaged hose was totally unanticipated.

In another related issues area, Reference 3.3.42 summarizes the effort to evaluate the naturally-aged (10 years) condition of eight ITT Barton, Model 332 differential pressure transmitters. The transmitters were removed from the Beznau Power Station in Switzerland. A calibration check of the as-received instruments showed reasonable agreement to the calibration data taken in 1979 (supplied by Beznau staff).

Calibration checks have also been performed at elevated temperature (122°F), the results of these tests showed that two transmitters exhibited considerable temperature effects, moderate effects were observed in two others, and the remaining instruments exhibited little or no temperature effect. However, some amount of mechanical hysteresis behavior was observed in all of the transmitters during the elevated-temperature checks.

Piecepart evaluations have been done and comparisons have been made with similar, new pieceparts and will be reported.

International and Other U.S. Programs

Related Experiences

We are not aware of any significant findings in this area.

Issue Findings

Past experience has shown that observations of unexpected equipment or material behavior can lead to previously unanticipated concerns.

Related Experiences

In summary, the findings within this subissue are:

1. Determinations should be made concerning the influence of the fabrication processes, surface preparations, and application procedures on the ability of heat-shrinkable tubing to provide a positive environmental seal.

2. The question of the behavior of flexible metal hose in a LOCA environment after aging should be addressed.

3. Room temperature calibration checks on naturally-aged (10 years) differential pressure transmitters agreed well with calibration data taken in 1979, but showed some temperature effects when the checks were performed at an elevated temperature.

4. All of the naturally-aged transmitters exhibited some hysteresis during the elevated-temperature calibration checks.

References

Related Experiences

3.3.40 Heat-Shrink Tubing Anomalies Investigation, letter L. Bonzon (SNL) to W. Farmer (NRC), April 8, 1983.

3.3.41 E. A. Salazar and E. E. Minor, Effects of Simulated LOCA Environments on Flexible Metal Hose, Memo to W. Farmer (NRC), May 26, 1983. (EQRT Program results.)

3.3.42 Letter L. Bonzon (SNL) to W. Farmer (NRC), Status Report on Ambient-Aged, Beznau Transmitter Evaluations, October 5, 1983.

3.4 Regulatory Guide 1.97 Requirements

Issue Description

Regulatory Guide 1.97 (May 1983) sets forth requirements for postaccident monitoring of containment and system conditions. It requires certain instrumentation, qualified for the accident conditions, to (1) provide information required to permit the operator to perform preplanned manual actions to accomplish safe plant shutdown; (2) determine whether the reactor trip, engineered safety features and manually initiated safety systems and other systems important to safety are performing their intended functions; and (3) provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release and/or to determine if a breach has occurred.

The equipment necessary to accomplish these actions must survive some of the most severe conditions encountered in a nuclear plant accident. Recently, questions have been raised concerning the adequacy of the procedures for equipment qualification to meet the requirements of RG 1.97, and whether it is even possible to meet the requirements. Research is just now beginning to address these questions.

This is the thrust of this effort to address the issue:

> "Is postaccident monitoring (PAM) equipment adequately qualified to meet the requirements of Regulatory Guide 1.97?"

Program Effort

No specific research testing of methodology to meet Regulatory Guide 1.97 requirements has been conducted to date. However, testing of a high-range radiation monitor has been conducted, as proposed by Reference 3.4.1.

International and Other U.S. Programs

Some evaluation of the requirements of Regulatory Guide 1.97 have been conducted under the USNRC-sponsored, Nuclear Power Plant Instrumentation Evaluation (NPPIE) program; see References 3.4.2 and 3.4.3. Its objectives were:

1. To identify problems facing the nuclear industry in meeting 10 CFR 50, Appendix A, Criteria 13, 19, 64, with regard to Accident Management Instrumentation.

2. To find practical, cost effective solutions to the problems identified in the first objective.

3. To examine Regulatory Guide 1.97 in detail to determine the adequacy of the current version and then to recommend changes to that guide as appropriate.

The five areas of primary concern were identified as:

1) reactor coolant level instrumentation

2) core exit thermocouples

3) containment area radiation monitors (to 10^7 R/hr)

4) halogen and particulate sampling

5) coolant activity measurements

Issue Findings

Current work to date only reveals that limited evaluations have been conducted on Regulatory Guide 1.97 requirements.

References

Program Effort

3.4.1 Letter, E. Richards (SNL) to W. Farmer (NRC), Test Plan for High-Range Radiation Monitor, December 19, 1985.

International and Other U.S. Programs

3.4.2 J. A. Rose, Nuclear Power Plant Instrumentation Evaluation, Tenth WRSR Information Meeting, NUREG/CP-0041, Vol. 3, October 12-15, 1982.

3.4.3 A. C. Williams, Assessment of PWR Core Exit Thermocouples During Accident and Postaccident Situations, Eleventh WRSR Information Meeting, NUREG/CP-0048, Vol. 5, October 24-28, 1983.

3.5 Advanced Systems Qualification Issues

Issue Description

Advanced light-water reactor designs, and reactor system upgrades in the future, are considering the use of advanced concepts such as fiberoptics and multiplexers in place of hardwired systems. In the latter case, the impetus is the need for more information channels, but the available containment penetrations are limited.

Viewed in the broadest context, any new safety-related system or concept must be judged for safety adequacy. The basis for that judgment must be established; in this case, the applicable and peculiar qualification methods must be established and verified.

This is the thrust of this effort to address the issue:

"What are the applicable qualification methods for advanced systems?"

Program Effort

To date this effort has merely involved the random collection of pertinent literature, which has come to our attention.

International and Other U.S. Programs

Both the Japanese and the French are known to have active programs in fiberoptics, relative to nuclear plant applications.

Reference 3.5.1, reports on Japanese fiber development and their radiation resistance. Of particular interest is their reporting of a dose-rate effect, as shown in Figure 3.5.1. The implications cannot be judged, but there is an apparent dependency on the test method employed.

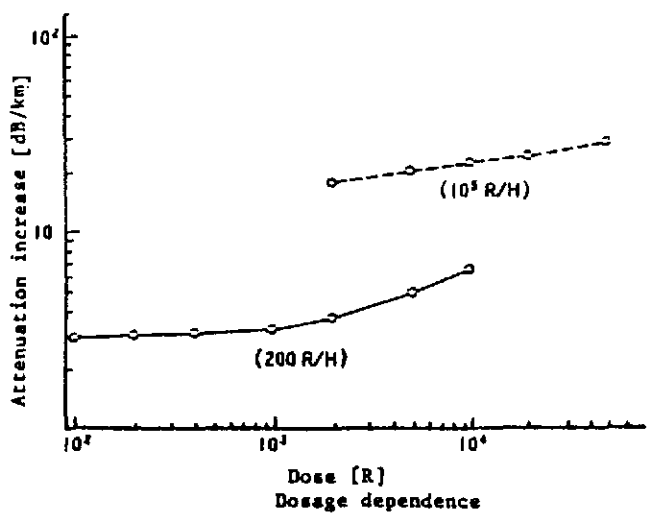

Figure 3.5.1 Dose and rate dependence of one optical fiber. (From Reference 3.5.1).

The French CEA, Reference 3.5.2, reported an active program with the FY-85 goals of:

- obtaining an irradiation acceleration method
- choosing a representative set of degradation parameters.

A variety of destructive and non-destructive methods will be used. Dose rates will be varied between .2 - 2 krad/hr. Many national and international references are known; several involve radiation tolerance of fiberoptics.

Conax Buffalo Corporation markets a fiberoptics feedthrough (Reference 3.5.3). Although it is recommended for extreme environments applications, no specific nuclear plant applications are mentioned. Environmental parameters listed include:

Hydrostatic pressure:	15,000 psi
Temperature range:	-300°F to 500°F (construction material dependent)
Radiation resistance:	225 Megarads (gamma)
Seismic capability:	15g Spectra/2.5g ZPA.

Issue Findings

Advanced light-water reactor designs, and reactor system upgrades in the future, are considering the use of advanced concepts such as fiberoptics and multiplexers in place of hardwired systems.

The preliminary review of the literature leads us to the finding:

Fiberoptics systems may be test method sensitive (i.e., dose-rate sensitive).

References

Program Effort

none

International and Other U. S. Programs

3.5.1 Y. Chigusa, et al, Radiation Resistant Characteristics of Image Fiber. Presented at the Conference on Electrical Insulation and Dielectric Phenomenon, October 22-26, 1984.

3.5.2 J. Henry, Private Communication. U.S./French Information Exchange. Sandia Laboratories, Albuquerque, October 18, 1984.

3.5.3 Fiber Optic Feedthroughs, Conax Buffalo Corporation brochure. Not dated, but data sheet insert is dated 11/21/83.

3.6 Realistic Accident Environments and Calculational Models

Issue Description

To be able to adequately assess equipment qualification under the accident environment, an accurate definition of that environment is necessary. This definition should include the source term, time dependent doses and rates, temperature/pressure profiles, and the steam/air/spray conditions, which will exist during an accident.

The continuing controversy on source term definition is recognized in RG 1.89 (June 1984) and 10 CFR 50.49 and it centers on the issue of fission product releases and their mechanistic distributions and time histories. The TMI-2 experiences are one impetus. Clearly the source term definition efforts (e.g., at various National Laboratories and the American Nuclear Society study) also impact this effort. But in addition to the mechanistic fission product releases, distributions, and time histories, there is also a need to develop calculational models accounting for equipment locations and shielding to determine applicable accident dose and dose-rate ranges.

A steam environment (superheated or saturated) is postulated to exist throughout the LOCA scenario, and therefore is included in the accident simulations for equipment qualification. Spray, too, is required to be considered as part of such tests. Additionally, research has begun to demonstrate that oxygen effects during accident simulations may be important. Thus it is desirable to accurately characterize the realistic time-dependent combined steam/air/spray conditions.

This is the thrust of this effort to address the issue:

> "What are the realistic accident environments, and how do they impact the choice of simulation methods?"

"Best-Estimate" Radiation Signature

The first effort to introduce more realism into the accident-radiation release scenario used the WASH-1400 time and magnitude scenarios. That is to say that the release assumptions are more mechanistic in nature, rather than the earlier hypothesized scenarios based on arbitrary (siting) source terms.

Reference 3.6.1, NUREG/CR-1237 (SAND79-2143), was the first report of its kind to attempt this mechanistic approach. Figure 3.6.1 shows the time sequence assumed for each of the four mechanistic release components employed, and Figures 3.6.2 and 3.6.3 the gamma and beta energy release rates from the various contributors. The effects of the uncertainties in the release fraction as well as variations in the time dependence of the releases were investigated;

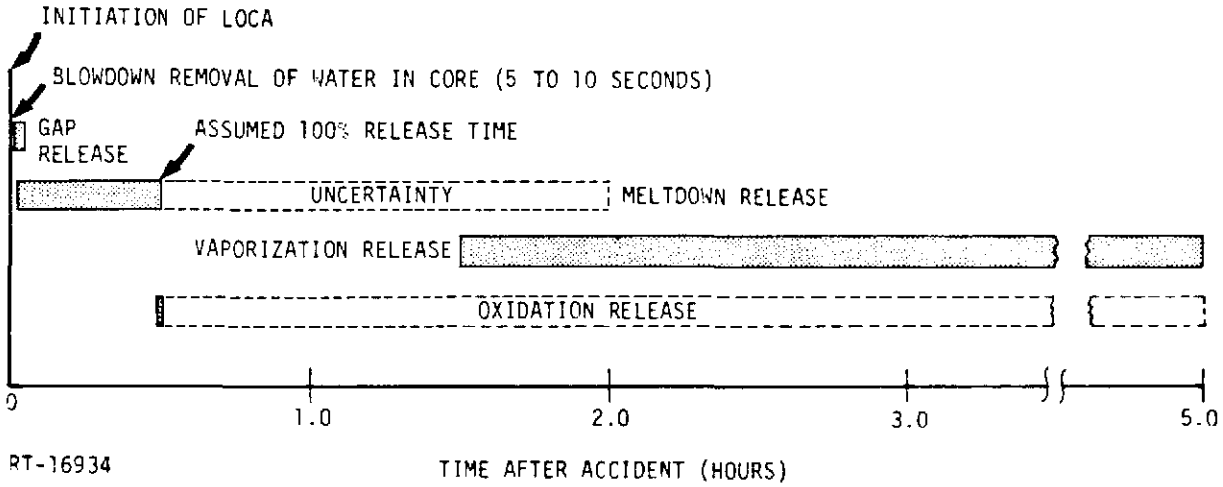

Figure 3.6.1 Time sequence of fission product release for an unterminated LOCA without emergency core cooling. (From Reference 3.6.1).

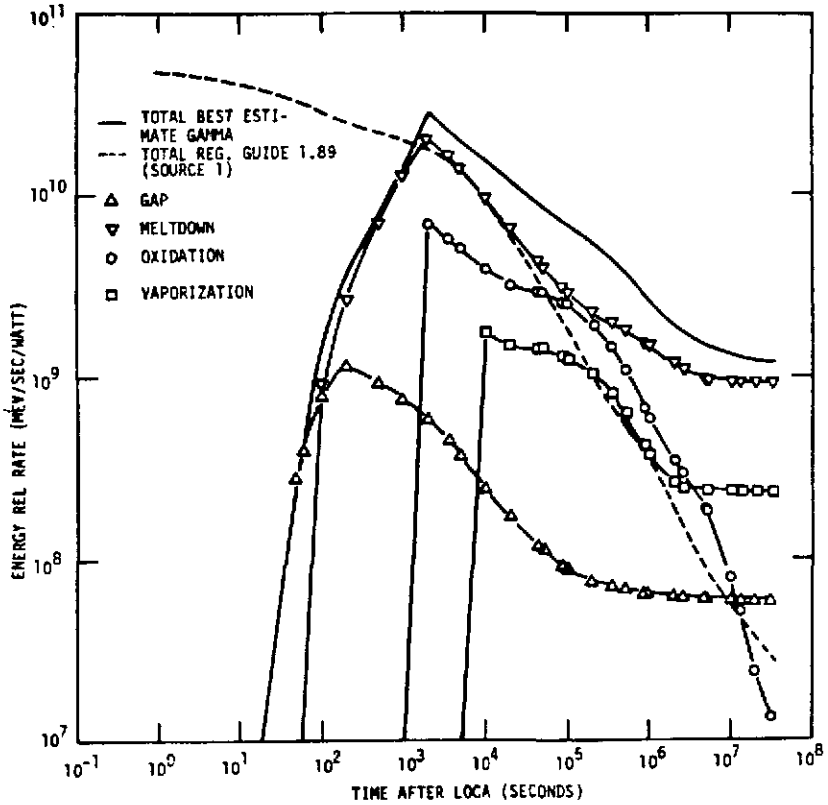

Figure 3.6.2 Gamma-ray energy release rate for "best-estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result. (From Reference 3.6.1).

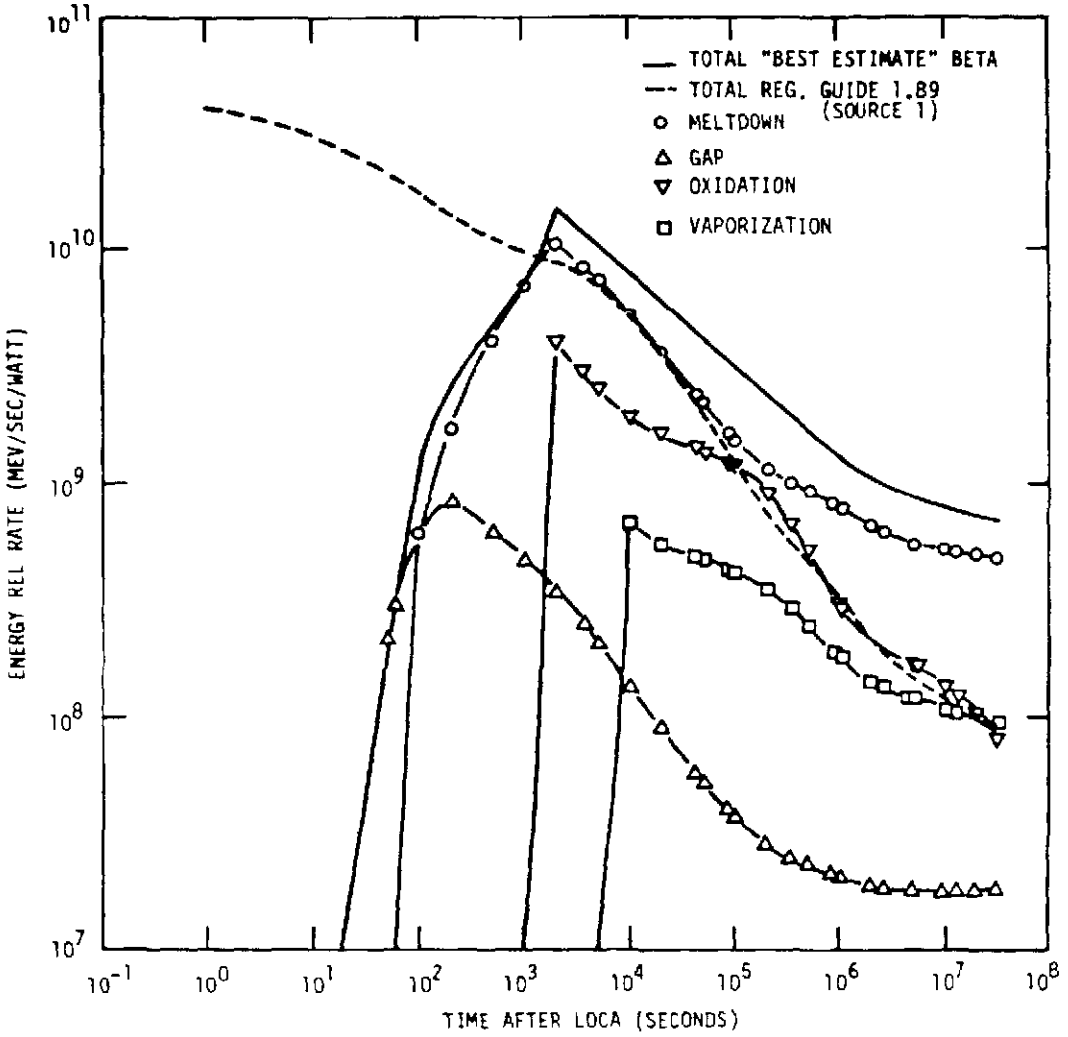

Figure 3.6.3 Beta energy release rate for "best-estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result. (From Reference 3.6.1).

a wide uncertainty band exists for the "best-estimate" releases. The study contained significant findings and recommendations:

- As far as the energy release rates are concerned, the dominant component is the meltdown term for all times greater than about 60 seconds after an unterminated LOCA without emergency core cooling. Except for the time interval 30-60 seconds, the gap release is the least significant. In an accident that is successfully terminated (i.e., does not result in significant core degradation), the gap release would be the only fission product source.

- Comparisons of the total "best-estimate" source with the total Regulatory Guide 1.89 source (Source-1) show that the latter source is very much larger and decreasing until about 1000 seconds (about 15 minutes) after the accident. At this time the "best-estimate" energy release is larger and remains larger for all later times.

- Spectrum calculations for the "best-estimate" sources were made analogous to the spectra generated for the Regulatory Guide sources. The behavior for the two cases is generally similar, including the behavior of the average particle energy.

Reference 3.6.2, NUREG/CR-2367 (SAND81-7159), was an update of the earlier study intended to incorporate new fission product release research. Of the four release stages (i.e., gap, meltdown, oxidation, vaporization), the only new information was for gap release; this is shown in the following table:

Table 3.6.1
Old "Best-Estimate" versus New "Best-Estimate"
Gap Release Fractions (From Reference 3.6.2)

Fission Product Species	Old Total Gap Release Fraction	New Total Gap Release Fraction
Noble Gases (Xe,Kr)	0.03	0.20
Halogens (I,Br)	0.017	0.30
Alkali Metals (Cs,Rb)	0.05	0.25
Alkaline Earths (Sr,Ba)	0.000001	0.0
Tellurium Group (Te,Se,Sb)	0.0001	0.0

The report demonstrates that further refinement of the "best-estimate" source term will come primarily from improvements in the meltdown release data.

Reference 3.6.3, NUREG/CR-2593 (SAND82-7013), and Reference 3.6.4, NUREG/CR-2594 (SAND82-7014), document the computer codes used in these "best-estimate" studies. RIBD/IRT calculates isotopic concentrations from various power histories and outputs time-dependent inventories, activities, decay powers, and energy releases for 800 fission products. GABAS calculates the time-dependent beta and/or gamma spectra from (decaying) fission products. Both codes are available from the Radiation Shielding Information Center (RSIC).

TMI-2 Accident Release Data

Available TMI-2 data continues to be evaluated, under a contract with IRT Corporation. Although that effort includes a broad review of equipment qualification issues, in particular it includes information on fission products.

Reference 3.6.5 (IRT 6597-001) reviews over 170 TMI-2 related reports (through April 1983) of which 53 are related to fission products. At that time, the best estimates of releases were as shown in the table:

Table 3.6.2
Summary of Activity Release from Fuel on March 28, 1979
(From Reference 3.6.5)

Nuclide	Percent Release
Kr-85	71
Xe-131M	70
Xe-133	68
I-131	59
Cs-134	76
Cs-136	57
Cs-137	60
Sr-89	<0.01
Sr-90	<0.07

Reference 3.6.5 recommends that:

1. A comprehensive mass balance of the fission products be made.

2. A full mapping of the TMI-2 containment building be made to help analysts verify existing transport codes and fission product release and behavior.

3. The raw release data should be analyzed and correlated to predict the time-dependent fission product releases. (This would be an important contribution to equipment qualification

as it would allow estimation of radiation dose levels to equipment...).

Some follow-on effort was made as documented in Reference 3.6.6, in particular, an "Attempt to Model the Fission Product Releases from TMI Unit 2." However, it was concluded that:

1. No information was available giving actual radioactive release data as a function of (early) time during the accident. In fact, much data is considered unreliable by on-site personnel.

2. "Experts" agree that one cannot realistically model the TMI accident with present state-of-the-art computer codes (primarily because the accident was "wet", and the codes model "dry" primary systems).

As of October 1, 1984, the TMI-2 Related Reports Collection contains 261 entries, with 88 being in the Fission Product category.

Dose/Rate Modeling

At the request of the Accident Evaluation Branch (AEB/NRR), the PATH code (Reference 3.6.7) was transferred to NRC to perform certain calculations of dose and rate received by equipment inside the containment building for various accident scenarios and source terms. PATH is a General Atomics proprietary code; it handles various spatial geometries, radiation source types, source geometries, source inputs and single shield materials.

We maintain the capability at Sandia, in parallel with the NRC (National Institute of Health computer) version.

Steam/Air/Spray Model

Regulatory Guide 1.89 (Appendix C) provides a number of references for methods which may be used to calculate the mass and energy release to determine the LOCA/MSLB environment. In addition, Appendix C of IEEE Std. 323-1974 presents a short discussion on a method to maintain saturated (100% RH) conditions in a combined steam/air test environment. However, with this method the test cannot include superheat conditions at the beginning of the LOCA simulation.

The question thus arises as to what are the real steam (superheat/saturated), oxygen partial pressure, and spray profiles to be expected in an accident? Clearly, additional research and modeling in this area would be beneficial.

International and Other U.S. Programs

A very large effort on realistic fission product releases and transport are underway around the world. Basic research is typically done at the National Laboratories, and sponsored by the NRC. The IDCOR and American Nuclear Society source term efforts represent the U.S. industry responses to this concern. The eventual "consensus" source term should be applicable to equipment qualification issues.

Modeling efforts are also a significant effort. A few of the major activities are listed here from Reference 3.6.6:

Primary System Fission Product Transport Codes -

TRAP-MELT- this code is being developed by Battelle for the NRC and is not available to the public. Its intended use is to model the transport of fission products through the primary system of a nuclear reactor. The code is capable of modeling accidents which are characterized as "dry" accidents.

RETAIN- this code is being developed by IMPELL Corporation for the Industry Degraded Core Rulemaking group (IDCOR). It is only available to members of IDCOR. It is apparently based on the same models used in TRAP-MELT.

Reactor Building Containment Codes -

MARCH- developed by Battelle for the NRC. A thermal-hydraulic code that describes the physical processes governing the progression of reactor meltdown accidents from the initiating event through attack of the concrete basemat by the molten core debris. It does not model the fission product transport. The output from MARCH is used as input to containment codes which cannot model thermal-hydraulic conditions.

CORRAL- developed by Battelle. This code calculates the fission product transport and deposition in the containment system of water-cooled reactors. It requires input from a thermal-hydraulic code, like MARCH, to perform its calculations.

CONTAIN 1.0- developed by Sandia for NRC. This is a coupled fission product transport and thermal-hydraulic containment code. It has the advantage over MARCH and CORRAL in that it does what they do in one code

instead of two codes. It is a state-of-the-art severe accident code and has many improvements over previous codes.

Integrated Severe Accident Computer Codes -

MELCOR- This is a code "package" being developed by Sandia for the NRC. It provides a system of severe accident computer codes to (1) model appropriately all phenomena essential to the description of severe LWR accidents, (2) provide "best estimate" consequence predictions for severe accidents, (3) permit meaningful estimates of the uncertainties associated with those predictions to be made, and (4) facilitates incorporation of new or alternative phenomenological models.

MAAP- This is IDCOR's counterpart to MELCOR and is not available to non-IDCOR members.

The TMI-2 lessons learned activities are generally funded by the Department of Energy under the GEND program. Numerous reports have been produced by this organization; References 3.6.5 and 3.6.6 summarize some of this effort.

EPRI has funded numerous studies on various aspects of TMI-2, including fission product releases and dispersal, but most are aimed at off-site dose estimate calculations.

Issue Findings

Extensive research continues throughout the world on defining fission product release fractions (and transport). By basing this effort on the consensus in-containment source term to be set by national research and debate, the applicable radiation signature for safety-related equipment can be finally established. With knowledge of the source term, calculational models accounting for equipment locations (and shielding) can be developed to determine applicable dose and dose rate ranges. The result will be a generic radiation signature to be used in equipment qualification tests based on mechanistic and realistic phenomena.

Likewise, a realistic model of the steam/air/spray conditions expected in containment during a LOCA/MSLB needs to be defined and modeled.

"Best-Estimate" Radiation Signature

In summary then, the findings within this subissue are:

1. As far as the energy release rates are concerned, the dominant component is the meltdown term for all times greater than about

60 seconds after an unterminated LOCA without emergency core cooling. Except for the time interval 30-60 seconds, the gap release is the least significant. In an accident that is successfully terminated (i.e., does not result in significant core degradation), the gap release would be the only fission product source.

2. Comparisons of the total "best-estimate" source from previous SNL studies with the total Regulatory Guide 1.89 source (Source-1) show that the latter source is very much larger and decreasing until about 1000 seconds (about 15 minutes) after the accident. At this time the "best-estimate" energy release is larger and remains larger for all later times.

3. Spectrum calculations for the "best-estimate" sources were made analogous to the spectra generated for the Regulatory Guide sources. The behavior for the two cases is generally similar, including the behavior of the average particle energy.

4. Further refinement of the "best-estimate" source term will come primarily from improvements in the meltdown release data.

Steam/Air/Spray Model

A realistic steam/air/spray model should be defined and evaluated.

TMI-2 Accident Release Data

In summary then, the findings within this subissue are:

1. No information was available giving actual radioactive release data as a function of early time during the accident.

2. "Experts" agreed that one cannot accurately model the TMI accident with present state-of-the-art computer codes (primarily because the accident is "wet", and the codes model "dry" primary systems).

References

"Best-Estimate" Radiation Signature

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3.6.2 D. D. Thayer, D. H. Houston, and N. A. Lurie, Updated Best-Estimate LOCA Radiation Signature, NUREG/CR-2367, SAND81-7159, IRT Corporation Contractor Report. Prepared for Sandia Laboratories under Contract #74-7074, August 1981.

A. N. A. Lurie, Best-Estimate LOCA Radiation Signature; Phase 1, Suggested Accident Scenario and Source Definition, IRT 0056-001, prepared for Sandia Laboratories, June 1978. (Incorporated into Sandia Topical report, SAND79-2143.)

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C. N. A. Lurie and L. L. Bonzon, Best-Estimate LOCA Radiation Signature for Equipment Qualification, SAND80-0888C, IRT 4617-044 (Albuquerque: Sandia Laboratories, April 1980). Presented at the ANS-ENS Meeting on Thermal Reactor Safety, Knoxville, Tennessee, April 8-11, 1980.

3.6.3 D. D. Thayer and N. A. Lurie, A User's Manual for Computer Code RIBD/IRT, NUREG/CR-2593, SAND82-7013, IRT Corporation for Sandia Laboratories, Albuquerque, New Mexico, January 1982.

3.6.4 D. D. Thayer and N. A. Lurie, A User's Manual for the GABAS Spectrum Computer Code, NUREG/CR-2594, SAND82-7014, IRT Corporation for Sandia Laboratories, Albuquerque, New Mexico, January 1982.

TMI-2 Accident Release Data

3.6.5 D. D. Thayer, TMI-2 Literature Search and Review with Application to Equipment Qualification Issues, IRT 6597-001, June 1983. Prepared by IRT Corporation under Contract 37-4505 to Sandia Laboratories, Albuquerque.

3.6.6 Letter Bonzon (SNLA) to Farmer (NRC), dated January 25, 1984, Status Report on IRT Efforts on TMI-2 Evaluations, with four attachments.

Attachment 1: IRT 6597-001, June 1983.
Attachment 2: Suggested Statements of Work
Attachment 3: Attempt to Model the Fission Product Releases from Three Mile Island Unit 2, IRT Report, January 12, 1984.
Attachment 4: TMI Reports List

Dose/Rate Modeling

3.6.7 Letter J. Bryson (SNLA) to W. Farmer (NRC) dated December 1, 1982, Summary of the PATH Computer Code Procurement and Implementation Project.

3.7 Criteria for Selecting Simulation Methods

Issue Description

Regulatory Guide 1.89 (June, 1984) indicates that the procedures described by IEEE Std. 323-1974 are acceptable (with a few exceptions) for qualification of electric equipment. IEEE 323-1974 implies that the method of type-testing should be selected to produce the most "severe" condition. In Section 6.3.2, Test Sequences, it states that "the type-tests shall be run on the equipment in a specified order. For most equipment and applications, the following constitutes the most severe sequence; however, the sequence used shall be justified as the most severe for the item being tested." The concepts of "...justified as the most severe..." implies that a literature examination or pretesting may be required; that is, data-based criteria for selecting simulation methods are necessary.

Other guidance in Regulatory Guide 1.89 (June 1984) also implies that a technical data base and/or test criteria may be important towards selecting the appropriate qualification method. Section 5 includes:

"a. If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program...

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

This is the thrust of this effort to address the issue:

"Can criteria be established for selecting the appropriate simulation method for equipment and materials?"

Aging Simulation Criteria

Reference 3.7.1 is a summary of materials aging methodologies and issues by SNL staff. It provides several recommendations regarding the generation of data to support qualification test strategy decisions.

1) If humidity is considered to be a possible stress, then screening tests can be conducted in an autoclave at elevated temperatures and 100 percent relative humidity.

2) If humidity effects are found, then development of technical basis for extrapolation of humidity effects to ambient aging conditions may be possible by carrying out aging tests at various temperatures keeping the relative humidities constant, as discussed in Reference 3.7.2, NUREG/CR-1466 (SAND79-1561).

3) As a general rule, the extrapolation factor should not exceed the data range factor. For example, if the data extends from 0.1 - 1 year, then an extrapolation beyond 10 years should not be made.

4) It is suggested that radiation experiments be performed to some constant dose, utilizing two different dose rates which differ by at least two orders of magnitude, the highest not to exceed 2 Mrad/hr.

5) For radiation plus temperature environments, synergistic interactions can be screened for by comparing a moderate radiation dose rate at room temperature with the same dose rate at a "mildly elevated temperature."

Reference 3.7.2 includes a discussion of several considerations in aging and life prediction: compatability tests; failure mode tests; accelerated aging studies; and life tests.

Reference 3.7.3 updates the work in Reference 3.7.1. It endorses the prior work and encourages the need for determining whether homogeneous or heterogeneous degradation is occurring. Several suggested screening techniques are (1) density profiling, (2) metallographic polishing, (3) gel permeation chromotography, and (4) swelling/solubility measurements. Reference 3.7.3 also suggests specific tests for:

- dose-rate effects; take data at 1 Mrad/hr, 150 krad/hr and 20 krad/hr to the same total dose

- synergistic/sequential effects; take data at 20 krad/hr at ambient and 70°C temperatures to the same total dose.

Reference 3.7.4 is a summary of the major international efforts on tests for predicting radiation degradation of organic materials. It is very useful as a survey article.

Reference 3.7.5, NUREG/CR-3643 (SAND83-2493), and Reference 3.7.6, NUREG/CR-4358 (SAND85-1557), discuss three useful techniques for determining heterogeneous/homogeneous degradation:

1) metallographic polishing
2) microhardness
3) density-gradient column

Figures 3.7.1-3.7.3 show typical results of these three techniques.

Reference 3.7.7, SAND83-1584C, approaches the screening criteria issue from the standpoint of using the results to select the best materials or equipment, even under rather complex aging situations (e.g., dose-rate effects). Examples are given in the reference.

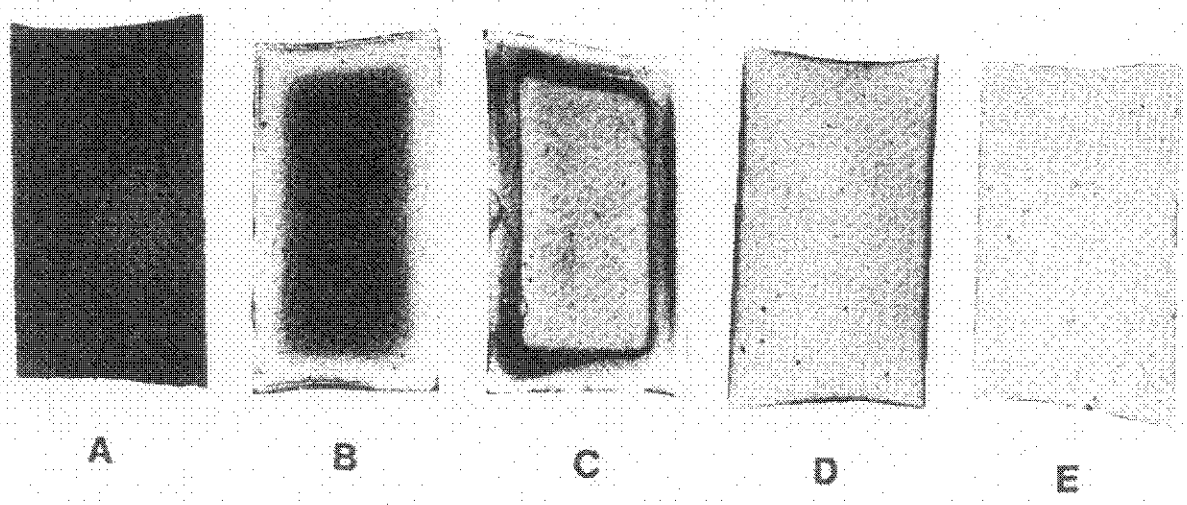

Figure 3.7.1 Cross-sectioned, polished samples of gamma irradiated EPR.
A: Unirradiated material
B: 6.7×10^5 rad/h (in air) to 165 Mrad
C: 6.7×10^5 rad/h (in air) to 297 Mrad
D: 1.1×10^5 rad/h (in air) to 175 Mrad
E: 1.1×10^6 rad/h (in vacuum) to 253 Mrad
All irradiations carried out at 70°C. Actual sample thickness = 3.15 mm. (From Reference 3.7.5)

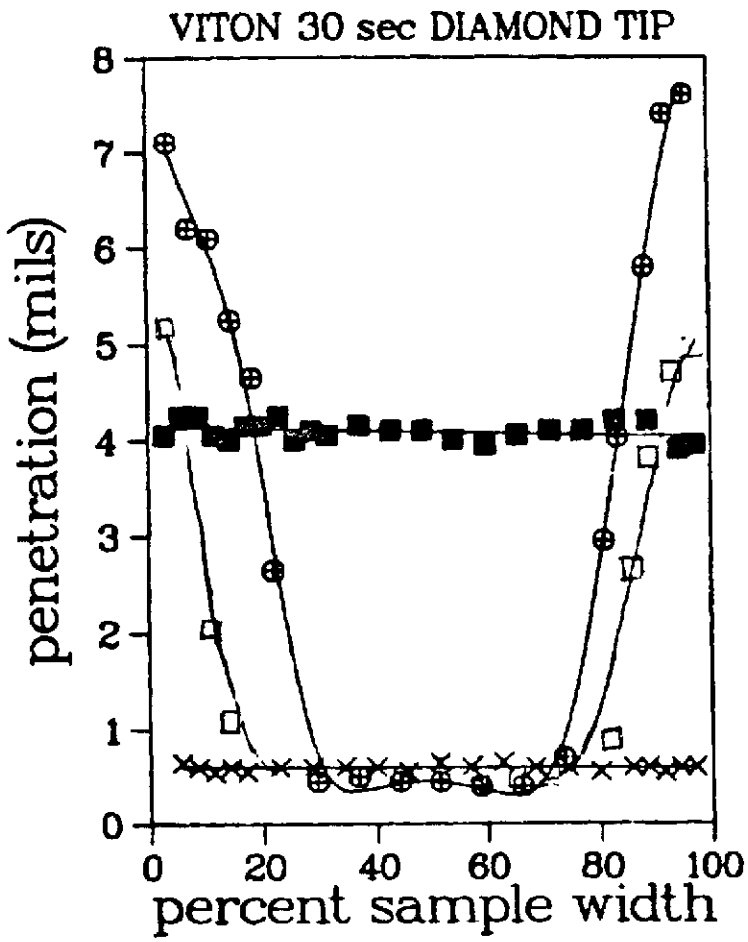

Figure 3.7.2 Hardness profiles for irradiated Viton® samples.
□ = Unirradiated material
□ = 5.5 x 10^5 rad/h (in air) to 165 Mrad
+ = 6.7 x 10^5 rad/h (in air) to 297 Mrad
X = 1.1 x 10^5 rad/h (in air) to 175 Mrad
Experimental load was 3 g. (From Reference 3.7.5)

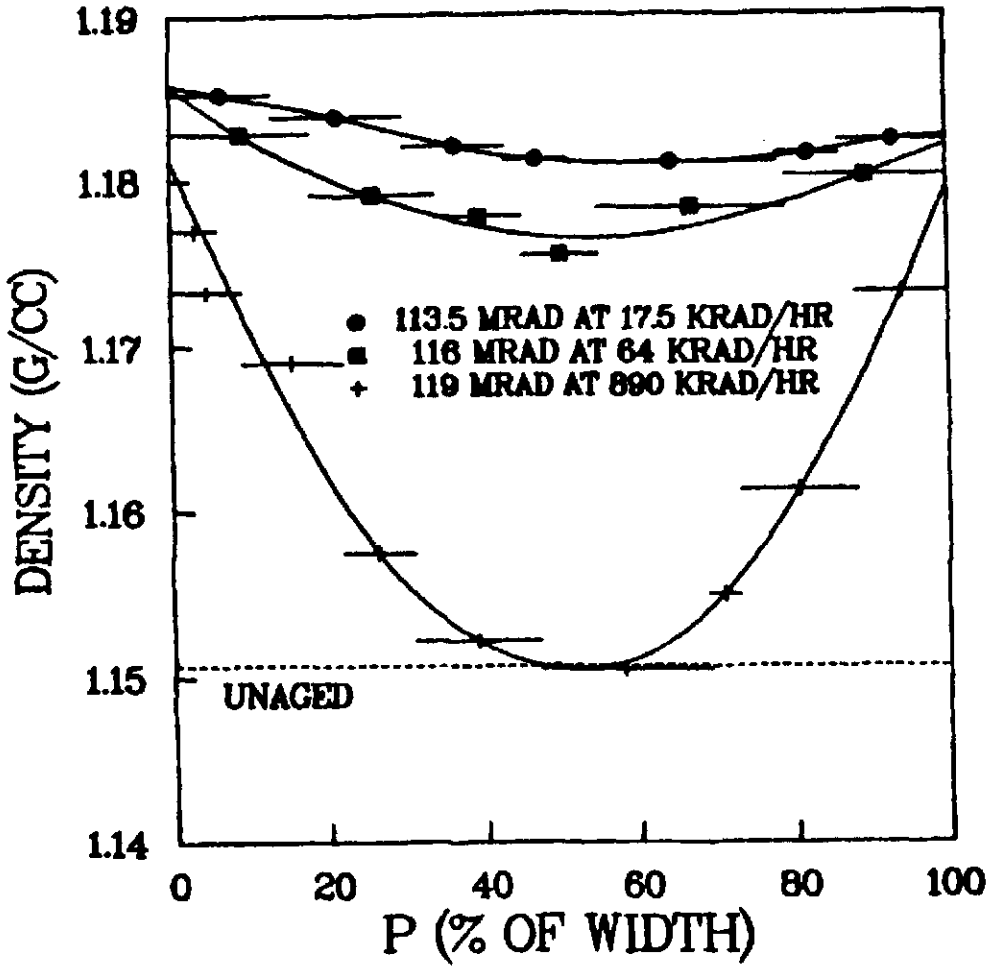

Figure 3.7.3 Density profiles for an unaged sample of the chemically crosslinked polyethylene material and for three samples radiation-aged in air to approximately the same total dose but at the three indicated dose rates. The percentage of the distance from the outside to the inside of the sample, P, is plotted as the abscissa. For this material, this distance averaged approximately 0.8 mm. (From Reference 3.7.6)

Reference 3.7.8, NUREG/CR-4147 (SAND85-0209), is an example of a product design influencing the choice of test method. The particular material, Sylgard 170, is rather insensitive to accelerated aging. However, in oneapplication it was confined even though it has a very large thermal coefficient of expansion. So aging at elevated temperature would result in unrealistic forces on the assembled component; in fact, such damage was observed and reported, in Reference 3.7.9, RS4445/81/03.

Reference 3.7.10, NUREG/CR-3808 (SAND84-7135), is yet another approach to developing screening criteria, based on a weak-link analysis approach and accounting for age-related degradation. While specifically related to aging-seismic correlations, it has a broader generic applicability through its Failure Modes and Effects Analysis (FMEA):

> "The purpose of this work is to develop a methodology for screening electrical equipment into various categories; the equipment will fall into groups that range between those in which no correlation between aging and seismic capacity exists and those in which a significant correlation exists.
>
> "The basis of the methodology for evaluating potential aging seismic correlation is the premise that equipment will fail during a seismic test when a "weak link" in the equipment mechanically fails by yielding or breaking. The probability of failure is calculated by using a normal probability function to define the relationship of the stress applied to the weak link and its frequency (or probability) of failing. If the weak links are fabricated from materials (e.g., elastomers, plastics) which degrade in strength with time and which may bear high loads during a seismic test, then a significant increase in the probability of failure with age may exist...
>
> "The probabilistic approach to evaluating the effects of aging and seismic stresses on the operability of safety-related equipment is considered more realistic than the presently used deterministic approach. Another advantage in using the probabilistic method is that the results can be used in determining the effects of aging on the probability of the system being able to perform its safety functions; an item which should be considered when completing a Probabilistic Risk Analysis."

Accident Simulation Criteria

Reference 3.7.11, NUREG/CR-2763 (SAND82-1071), shows an example where a simple air-oven test may screen for the importance of oxygen in LOCA testing. Table 3.7.1 illustrates this point. Results from a LOCA simulation with 10% oxygen content are compared to (145°C) air-oven results. Clearly, the same oxygen-dependency trends, even with aging methods (i.e., dose-rate and total dose) as a parameter, are evident.

Reference 3.7.12, NUREG/CR-3538 (SAND83-1258), reports on EPR cable performance during LOCA simulations. Table 3.7.2 illustrates the variability of test results with material and with test method employed

Table 3.7.1

Sequential Screening Comparisons

(From Reference 3.7.11)

MATERIAL	LOT	e/e_o AIR OVEN	e/e_o L-10
CLPE	1	0.65	0.68
	3	0.05	0.42
	4	0.56	0.53
TEFZEL®	1	0.92	1.02
	4	0	0.04
SILICONE	1	0.93	0.96
	2	0.24	0.38
	4	0.17	0.36
CLPO	1	0.79	0.98
	2	0.33	0.70
	3	0.24	0.49
	4	0.26	0.72
CSPE	1	0.40	0.74
	2	0.27	0.61
	3	0.19	0.48
	4	0.15	0.58
PVC	1	0.57	0.53
	2	0.18	0.09
	4	0.04	0.03
CP	1	0.01	0.05
	2	0.01	0.03
EPR	1	0.77	0.89
	2	0.49	0.38
	3	0.12	0.17
	4	0.14	0.12

Lot #1:
1: Unaged
2: 21 Mrad at 880 krad/hr
3: 45 Mrad at 880 krad/hr
4: 23 Mrad at 24 krad hr

Table 3.7.2

(From Reference 3.7.12)

Insulation Specimens: Percentage Weight Increases and Dimensional Increases (O.D./Length or Width/Thickness/Length

Cable Material	Sequential Test *	Simultaneous Test #1*	Simultaneous Test #2**
Percentage Weight Increases			
EPR A	+50	?	
EPR B	+4	-1	
EPR C	+9	+23	
EPR D	+121	+173	+172
EPR E	+0	+7	
EPR F			+94
EPR-5			+77
EPR-1483	+55	+45	
Percentage Dimensional Increases			
EPR A	+19/0	?	
EPR B	-18/0	-5/0	
EPR C	+5/+20/0	+7/+20/+5	
EPR D	+38/+5	+53/+35	+51/+42
EPR E	+2/0/0	0/0/0	
EPR F			+31/+19
EPR 5			+18/+30/+21
EPR-1483	+17/+46/+7	+20/+46/+16	

* Both the sequential and simultaneous #1 LOCA profiles were interrupted at day 9 by an unanticipated steam cooldown. The test was continued and measurements were made at the end of 21 days of steam exposure.

** Measurements made during unanticipated steam cooldown starting at day 16 of LOCA profile.

(simultaneous or sequential). In particular, note the response of EPR D. That cable actually failed electrical performance tests. This can be correlated with its excessive weight increases (i.e., moisture absorption) and dimensional increases. Hence, the results suggest that a moisture absorption test may be useful in selecting an appropriate accident methodology.

Fragility Tests

Selection criteria may also be established through fragility testing. Very limited efforts to data have been conducted, and these are mostly on seismic tests of batteries; see for example, Reference 3.7.13, NUREG/CR-3923 (SAND84-1737). The benefit of such testing is to focus on the actual failure modes and thresholds. The appropriate qualification test method can then be chosen.

International and Other U.S. Programs

While there is a significant level of activity in testing of safety-related materials and equipment, we are not aware of any other specific effort to select the appropriate simulation method by establishing data-based criteria.

Issue Findings

Under the overall QTE Program effort, some techniques for developing a data base and associated test criteria have come to light.

By evaluating certain screening criteria, which are suggested by analyses of test data and results, it should be possible to develop these to a sufficiently advanced level that they can be used to decide the appropriate accident or aging simulation test method for the specific equipment item. As test results continue to be generated in other parts of this program, other criteria may be pursued in a similar manner. Closure of this issue may be based on demonstration of whether the individual criterion prove to be satisfactory or unsatisfactory.

In summary then, the findings within this issue are:

1. Humidity screening may be accomplished in 100 percent relative humidity autoclaves.

2. Dose-rate effects may be determined by conducting experiments at rates covering two orders of magnitude.

3. Synergistic/sequential effects may be determined by conducting moderate dose-rate experiments in ambient and elevated (70°C) temperatures.

4. Heterogeneous degradation may be determined through simple techniques like metallographic polishing, microhardness, and density-gradient columns.

5. "Best" materials can be selected using these same techniques.

6. Manufacturer's product description literature may affect the choice of simulation methods.

7. Probabilistic, weak-link, age-degradable analyses may provide insights into failure mechanisms and hence test methods selection.

8. Simple, air-oven, tests may indicate a preferred LOCA-oxygen simulation test method.

9. Swelling, moisture-absorption, dimensional tests may indicate a preferred LOCA simultaneous/sequential test method.

10. Fragility testing may provide the focus on a preferred qualification test method.

References

Aging Simulation Criteria

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3.7.4 R. L. Clough, et al., Accelerated-Aging Tests for Predicting Radiation Degradation of Organic Materials, Nuclear Safety, Vol. 25, No. 2, March-April 1984, pp. 238-254.

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Fragility Tests

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3.8 Review of Standards and Guides

Issue Description

The thirty issues being addressed in this report were selected from the specifications in Regulatory Guides and industry standards (and from typical industry test practices). They are key issues in equipment qualification and/or require supportive development and data bases.

However, equipment qualification methods and requirements are affected by the continual development and updating of Guides and Standards and by methodology research activities.

Through a continuous, in-depth review of the standards and regulations governing equipment qualification issues, other generic issues not currently addressed in this program may be identified and necessary investigations initiated and pursued to conclusion.

This is the thrust of this effort to address the issue:

"Do the Standards and Guides inclusively and adequately encompass the equipment qualification issues?"

Program Effort

The evolution of standards and regulations pertaining to equipment qualification issues has been significant and is continuing. For example, IEEE-323 has progressed through 1971, 1974, and 1983 versions as well as an intermediate (intended as a substitute) standard, IEEE-627-1980. And Regulatory Guide 1.89 has also evolved in a parallel manner (and is again slated for revision in late 1985).

Similarly, research and experience have also affected qualification methods. The most recent and most visible development may be the TMI-2 source term data and subsequent national review.

To keep abreast of these developments, Sandia staff serve on various national and international standards organization. In addition, pertinent research reports, as well as Guides and Standards, are collected. Sandia staff also assist NRC/IE in technical reviews of vendor and utility qualification programs. These activities are a bases for identifying new issues requiring resolution.

Typical of those issues are the following, which have recently been identified as potentially important and deserving of evaluation:

- moisture intrusion into "sealed" systems
- realistic accident pressure/temperature conditions in containment
- realistic oxygen partial-pressure time history living accidents
- 1-hour time margin requirements

- equipment research, coupled with a systems survivability concept approach
- HELB phenomena, and effect on equipment outside containment
- the "conservatism" in sequential testing with all radiation (ambient plus accident) applied first in the sequence
- naturally-aged equipment evaluation and tests, and comparisons to artificially-aged equipment
- aging (and accident) degradation of equipment and materials in inert environments typical of some BWR plants
- margin; what is adequate margin?
- how do electrical monitoring techniques in simulation tests apply to inplant circuits?
- the significance of systems testing compared with individual components tests
- definitive specifications for acceptance criteria
- "scaling" effects; e.g., smaller, but same-type components and prototypes
- tests which evaluate the significance of "proper" aging simulation on the functionability of equipment during accident simulation
- steam and radiation (e.g., "beta cloud" propagation into closed, but not sealed, enclosures.

International and Other U.S. Programs

In 1979, the Office of Standards Development (SD/NRC) requested a study be done intended to serve as a broad framework to enable the USNRC/SD to choose from, to study further, and to implement select, programs that would advance the state-of-the-art of safety-related equipment qualification through further Standards and Guides development. Reference 3.8.1, NUREG/CR-0988 (SAND79-0944), had these objectives:

- Compile existing directives, in-progress directives, and community experience pertaining to Class 1 component qualification testing.

- Enumerate those qualification testing areas requiring standards development.

- Evaluate the necessary supportive program for each standard's development.

- Prioritize the necessary effort, construct schedules and cost estimates for standards development.

- Prepare a final report, summarizing the program results with suggestions for follow-on activities.

The study's general recommendations were:

-- Maintain the status quo on standards for an interim period (while the goals and programs are reevaluated).
-- Support data-base generation activities.

-- Reduce cross-referencing.
-- Increase reliance on stable, data-base literature.
-- Separate problem definition from problem solution.
-- Redefine working groups' charters.
-- Strive for uniformity of standards information content.
-- Return to numerical environment definition.
-- Adopt applicable existing procedures and specifications.

Besides Standards and Guides, industry experiences are a useful barometer for needed activities. Reference 3.8.2 (FRC Project C5257) is an excellent summary of such experience; it summarizes the findings of NRC reviews of licensees and applicants to meet the requirements set forth in the DOR Guidelines and NUREG-0588 (for safety-related electrical equipment located in harsh environments).

Internationally, several nations are developing their own equipment qualification standards and doing the necessary supportive research, most notably the French, Japanese, and Germans. While the IEEE standards were the original bases, more recently specific and very technical standards and procedures are being developed.

Issue Findings

Through an in-depth continuing review of the standards and regulations governing equipment qualification issues, other generic issues not currently addressed in this program may be identified.

References

Program Effort

none

International and Other U.S. Programs

3.8.1 L. L. Bonzon, Recommendations for Programs Development for Safety-Related Equipment Qualification Standards and Guides, NUREG/CR-0988, SAND79-0944, Sandia National Laboratories, December 1979. (USNRC/SD Program)

3.8.2 G. J. Toman, et al, Implementation Guidance for New and Corrective Equipment Environmental Qualification, FRC Project C5257, NRC Contract No. NRC-03-79-118, Franklin Research Center, April 22, 1983.

3.9 Battery Aging Methods

Issue Description

To adequately determine the qualification of nuclear station batteries, representative cells which compose the battery must be exposed to simulated seismic events. The seismic events must be representative of the seismic conditions to which the inservice batteries could be subjected. For a complete qualification, the cells should be exposed to the simulated events at ages which represent their expected end-of-qualified-life condition. While naturally-aged cells may be used for qualification testing, such cells are not usually available and time constraints may not permit waiting until such cells are available. To circumvent the problem of nonavailability of naturally-aged cells, cells aged by accelerated methods may be used.

The current method to accelerate the age of cells is by exposing the cell(s) to an elevated temperature for an extended time period. The aging period in days is determined by multiplying the desired age in years by published equivalent aging factors which are given in IEEE 535-1979. The rationale for this method is that the entire cell will be aged to the supposed predominate aging failure mode which is based on positive grid growth and then plate failure.

However, another battery failure mode, other than what was expected was found to be accelerated during a series of aging tests (i.e., corrosion of the post bar rather than plate growth).

This is the thrust of this effort to address the issue:

> "What are the appropriate accelerated-aging methodologies for nuclear station batteries?"

Program Efforts

Thermal aging efforts on battery cells in the late 1960s indicated an acceleration factor of 40 and was based upon a ten percent total growth of a positive plate. Thus the accepted acceleration factors of 18 and 36 from IEEE 535-1979 appear to be realistic values. Subsequent efforts, however, in the mid-1970s involving thermal aging and cycling at elevated temperatures indicate acceleration factors that are in the range of 2 to 12. Other work by personnel at the Battelle Columbus Laboratories indicate that valid acceleration methods are usually less than 10. These latest efforts indicate the accepted acceleration factors are high by factors of anywhere from 1.5 to 18 with the probability that the higher factors dominate.

Although the accepted aging temperatures and accelerated aging factors are stated to be based on the results of tests performed by battery manufacturers, that test data has not been available for verification to date. Discussions with members of the American National Standards

Committee C40, Storage Batteries, and a battery manufacturer have indicated a far from overwhelming endorsement of this "accepted" thermal aging method. In addition, some battery cells that had been thermally aged to 20 years looked as good as new ones when filmed during battery seismic qualification tests.

The initial effort to evaluate the presently suggested accelerated aging methods and to begin a determination of a rigorous data-based accelerated-aging methodology started with a series of seismic-fragility tests on naturally-aged battery cells. The tests established a baseline for failure modes and degradation thresholds experienced by these cells against which accelerated-aged cells will be compared and evaluated.

Reference 3.9.1, NUREG/CR-3923 (SAND84-1737), is the first report in the test series and reports on the seismic fragility of 12-year old lead-calcium Gould NCX-2250 cells. As in all cases in this test series, the test philosophy was to monitor the electrical properties including discharge capacity of the cells through a graduated series of g-level step increases until either the shake table limits were reached or electrical "failure" of the cells occurred. It must be noted that these are intended to be fragility tests, and in general the acceleration levels reached are well in excess of plant-specific qualification levels. The results of this first test provided the following significant findings:

- Of nine electrically active cells, six failed during the seismic testing phase when exposed to a series of g-loads in excess of 1-g zero period acceleration (ZPA).

- Of the three cells that did not fail during the seismic testing phase, all had posttest capacities greater than minimum accepted value of 80 percent.

- Posttest examination revealed a common failure mode of cracking at the abnormally brittle, positive lead bus bar/terminal post interface. One cell also experienced the failure at the negative bus bar/terminal post interface. Subsequent metallurgical analysis showed the brittle, cracked area to be characterized by an extremely coarse grain size material and extensive corrosion.

- None of the positive plates showed any significant growth, contrary to the currently accepted accelerated aging method which is specifically based upon plate growth.

Reference 3.9.2, NUREG/CR-4095 (SAND84-2628), is the second report in the test series and documents the seismic-fragility tests of 10-year old lead-calcium Exide FHC-19 battery cells. The time sequence of the g-level inputs varied slightly in these tests. The cells were first exposed to a simulated seismic event (g-level) with a ZPA of approximately 1.5 g. Of the cells that survived that level, two were selected and exposed to an even higher simulated seismic intensity with a ZPA of approximately 2.0 g. The results of this testing showed:

- Of the eight electrically active cells (six at the lower and two at the higher ZPAs), none failed during the seismic testing phase.

- Subsequent capacity tests showed only two of the six cells tested at the lower ZPA and neither of the two cells tested at the higher ZPA had capacities greater than the minimum accepted value of 80 percent.

- Disassembly of some of the cells showed active material lost from the positive plates and the remaining material hard and cracked. The positive bus bar material was corroded and brittle, with the corrosion having reduced the area available for conduction to just over 50 percent of original. (No breakages occurred during testing however.)

- The negative plates were in poor condition with the active material cracked and blistered.

- Specific gravity measurements made between the low and high ZPA testing indicated a high rate of self-discharge for one of the two cells tested at the higher ZPA.

Reference 3.9.3, NUREG/CR-4096 (SAND84-2629), is the third report in the test series and documents the seismic-fragility tests of 10-year old lead-calcium C&D LCU-13 cells. The results of this testing showed:

- Of nine electrically active cells (seven at the lower and two at the higher ZPAs), none failed during the seismic testing phase.

- Subsequent capacity tests showed all of the seven cells tested at the lower ZPA, but neither of the two cells tested at the higher ZPA, had capacities greater than the minimum accepted value of 80 percent.

- All plates were in good condition with the positive plates showing minimal loss of active material and no significant plate growth.

- Specific gravity measurements made between the low and high ZPA testing indicated a high rate of self-discharge for one of the two cells tested at the higher ZPA.

- There was no evidence of corrosion or cracks in the bus bars.

Reference 3.9.4, NUREG/CR-4097 (SAND84-2630), is the fourth report in the test series. The cells tested were 27-year old lead-antimony Exide EMP-13 cells. These cells were a modified Plante construction - a high purity lead strip rolled into a spiral button and inserted into holes cut in heavy antimony alloy grid. They were included in this test series because their age was far greater than anything else available and there was a need to evaluate advanced-age degradations. The results showed:

- Of nine cells electrically active cells, none failed during the seismic testing phase when exposed to a series of g loads in excess of 1-g ZPA.

- Subsequent capacity tests on five of the cells showed posttest capacities of 33 percent and lower.

- Disassembly of three of the cells showed them to be in very poor condition. Negative plates were weak and brittle and severely erupted on both sides. Positive plates, while solid and not warped, did have the lead buttons clogged with active lead oxides and showed evidence of corrosion products being dislodged from the positive grids.

- The positive bus material was brittle and showed evidence of chemical attack, but no evidence of cracks was observed in either post of the bus material.

In summary, we have seen (to varying degrees) chemical attack/corrosion of the positive bus bars, which in one case has been attributed to large grain size in three of the four cell types tested. In three of the four test series, the cells did not fail during the seismic testing. The posttest capacity tests produced mixed results and gave no indication of a common factor which would indicate which cells would meet the 80 percent posttest capacity and which would not. For example:

- Some Gould cells failed during the seismic testing phase, but all cells that did not fail had posttest capacities greater than 80 percent.

- None of the Exide cells failed during the seismic testing phase, but only one-third had posttest capacities greater than 80 percent.

- None of the C&D cells failed during the seismic testing phase and all had posttest capacities greater than 80 percent.

- None of the Exide and C&D cells failed during a second (higher g-level) series of seismic tests, but none of them had posttest capacities greater than 80 percent.

Plate growth was seen in only one of the four cases but even that growth was not substantial (compared to IEEE 535-1979 positive plate growth basis).

To more explicitly evaluate these results in terms of cell degradations and their impact on aging methodology, Reference 3.9.5, NUREG/CR-4099 (SAND84-2632), is presently being prepared. It will evaluate the overall effort and provide a baseline against which future results can be compared.

To establish a baseline, new cells of the same type of lead-calcium cells that were previously tested were exposed to simulated seismic events. They, like the naturally-aged FHC-19 and LCU-13, will be exposed to g-levels of two different ZPA values. Only the first phase of testing at the lower ZPA level has been accomplished to date, but none of the cells failed during the seismic testing and all cells had posttest capacities greater than 80 percent. The results of the testing at both ZPA levels will be published as Reference 3.9.6, NUREG/CR-4098 (SAND84-2631).

International and Other U.S. Programs

Both the French and Canadians are known to be interested in aging of battery cells and subsequent seismic testing for qualification. References 3.9.7 (HM64/313) describes the French endurance tests to accelerate the aging of battery cells and Reference 3.9.8 (HM64/321) describes the specification for qualification of battery cells. They are the result of seven years of testing by different manufacturers (Reference 3.9.9). Unfortunately, the results contain proprietary information and are not available to us. Interestingly, the French do not appear to use thermal heating of the cells as a method of aging. Their efforts seem to focus on cycling and overcharging.

While the Canadians have no direct interest in investigating aging methodologies, they have been conducting seismic qualification work over the last eight years. Reference 3.9.10 (77-221-H) reports on the earliest effort of which we are aware. The report states the tests "...are exploratory...to provide initial test data and general information on the performance...of batteries of various ages under simulated seismic excitation." Tested were lead-antimony cells that were new, and 9 and 16 years old, all in a three cell rack configuration. None of the cells "failed" during seismic testing, even though some of the plates of the 16-year old cell fractured and fell to the bottom of the cell jar. The 16-year old cells had a posttest capacity of only 31 percent because of the structural damage to the plates during the tests. The other two sets of cells both had posttest capacities greater than 80 percent. The report did not comment on any plate growth evaluations.

Reference 3.9.11 (C84-5-K) documents a later effort by the Canadians. This effort involved the testing of 13-year old lead-antimony cells that were selected from two cell banks at the Pickering A Nuclear Generating Station. Eight cells, all of which had pretest capacities greater than 80 percent, were exposed to the seismic test. None of the cells failed during the seismic testing, but two had posttest capacities less than 80 percent. Although none of these cells were disassembled, cell inspection and measurements showed the positive plates had increased in thickness by an average of over 40 percent.

Most of the early U.S. efforts of which we are aware concerned qualification of specific batteries for a specific plant application using thermally-aged cells. Reference 3.9.12 (43291-1) reports on two thermally-aged (to 20 years) C&D KC-19 cells exposed to a simulated

seismic program where the Test Response Spectra (TRS) exceeded the Required Response Spectra (RRS) for the San Onofre Nuclear Generating Station. No condition of the cells either pre- or posttest is given. The cells did not fail during the seismic testing and the statement is made "...that the specimens possessed sufficient integrity to withstand, without compromise of structure or electrical function, the prescribed simulated seismic environment..." Reference 3.9.13 (43450-1) reports the results of a seismic qualification test that included two C&D LC-21 cells that had been thermally aged to 20 years and two C&D CF cells that were naturally aged to 25 years. Half way through the series of tests one of the thermally-aged cell jars cracked. Both were subsequently replaced for the balance of the tests. The report concluded that except for the cracked cell jar, the cells demonstrated integrity to withstand the prescribed simulated seismic environment. A generic RRS was used that was the composition of eight plant specific RRS. No other pre- or posttest conditions or data were provided.

Reference 3.9.14 (44466-1) reports the results of seismic testing on a group of C&D cells (three KC-9, two KC-13, two KC-17 and three KC-21) that had been thermally aged to 20 years. All were exposed to a TRS that exceeded a generic RRS. None of the cells failed during the seismic testing and again the statement that the specimens possessed sufficient integrity to withstand the seismic environment was made. As before, no pre- or posttest conditions or data were provided.

Reference 3.9.15 (44681-1) reports the results of seismic testing on three each Gould naturally-aged, 16-year old FPR-23 and 10-year old FPS-25 cells. Again no cells failed during the seismic testing. In addition to the statement that the specimens possessed integrity to withstand the simulated seismic environment, pre- and posttest capacities were provided; all had capacities over 100 percent. The oldest cells had the highest pretest capacities, lost the least amount of capacity, and ended with the highest capacities.

Reference 3.9.16 (44681-2) reports on tests that included cells which had been aged by an accelerated (thermal) aging method. The intent was to age different cells to 5, 10, 15 and 20 years, but because of anomalies that occurred during the 5- and 10-year aging sequence, the aging was terminated by the customer (Gould) at the 10-year stage. Seismic testing was then done using sets of cells that had been aged to 5 and 10 years and also 8-, 10-, and 16-year old naturally-aged cells. Although one cell (a 10-year old naturally-aged FPS-25 cell) experienced bent positive terminal posts, there is no information to indicate that the cell was disassembled to look for corrosion on the positive bus bar. The posttest capacities varied. All 5-year accelerated-aged cells that survived the seismic testing had posttest capacities above 80 percent. All 10-year accelerated-aged cells failed the 80 percent criteria. Two cells (10 years old) of the 18 naturally-aged cells failed the 80 percent criteria; the other 16 all had capacities above 90 percent.

Reference 3.9.17 (SAND82-7049) is an overview of battery aging methodologies. It documents the present status of accelerated aging of batteries that are pertinent to photovoltaic applications. It provides a background on accelerated aging, gives the results of a literature and manufacturers' survey, presents a review of the general concepts of accelerated aging, examines the presently accepted method to accelerate age, offers a team concept to design an accelerated life test, and finally includes a brief description of a recently developed method for accelerated life testing. The report concludes that it is not known whether a good accelerated test (aging) for life prediction can be developed. It recommends the effort should be continued on lead-acid batteries so that development of a new accelerated aging procedure can capitalize on the progress made.

Reference 3.9.18 (A-472-82) is a qualification report of three battery types which used (approximately) 20-year old accelerated aged C&D cells. All cells were exposed to a seismic TRS that exceeded the specific RRS for Hope Creek. None of the cells failed during the seismic testing and all had posttest capacities greater than 80 percent.

Reference 3.9.19 (58836) reports on the results of seismic testing of 14-year old naturally-aged C&D LCU-27 cells. One cell, in a three- cell rack test configuration, cracked during the seismic testing and was replaced. Additional restraint (a band) was installed to prevent movement and the test was continued. No other mechanical or electrical failures occurred. Pre- and posttest data were not provided.

Reference 3.9.20 (SAND83-7122) is an independent review and analysis of the accelerated aging procedures and test data obtained by Exide over a 3-year program to develop advanced lead acid batteries for utility load leveling. Of interest to the aging methodologies is the deep-discharge cycling tests at elevated temperatures. For stationary batteries used in nuclear power plants, cycling new cells may be a reliable method to age the cells for subsequent qualification tests. Exide also looked at temperature acceleration factors but only as an additional stress during the cycling of the cells. Battelle questioned the value Exide used as a temperature acceleration factor and recommended a program to verify or determine the acceleration factor. A successful conclusion to such a program could be of help in arriving at a valid, data-based, accelerated aging method.

Issue Findings

Nuclear station batteries are a specific vital safety-related system which may be vulnerable to aging degradation.

As references 3.9.7 through 3.9.20 indicate, there are several worldwide research programs investigating various battery aging methodologies.

In summary then, the findings within this issue are:

1. Corrosion, to varying degrees, of the positive bus (bars) and the interface region of the positive bus bar/terminal post has been observed. (In some cases, this resulted in a breakage of the lead post during high-g seismic fragility tests.)

2. Minimal positive plate/grid growth has been found in some batteries, contrary to the currently accepted accelerated aging method which is specifically based upon plate growth.

3. A high rate of self-discharge of one-half of the cells exposed to a second simulated seismic event.

4. "Failure" of all cells exposed to a second series of simulated seismic-fragility events to retain a capacity of the accepted minimum value of 80 percent.

5. One type of cell tested showed gross mechanical (and electrical) failure during the seismic-fragility test, as a result of breakage of the bus bar/terminal post interface.

6. There was no common indicator immediately following the simulated seismic-fragility event results which would predict the 3-hour posttest capacity of the cells.

7. Some plates from some types of cells did show substantial degradation.

References

Program Effort

3.9.1 L. L. Bonzon, et al, Test Series 1: Seismic-Fragility Tests of Naturally-Aged Class 1E Gould NCX-2250 Battery Cells, NUREG/CR-3923, SAND84-1737, Sandia National Laboratories, Albuquerque, New Mexico, September 1984.

A. L. L. Bonzon, et al, Seismic Fragility Testing of Naturally-Aged, Safety-Related, Class 1E Battery Cells, SAND84-1652C, October 1984. Presented at the 12th WRSR Information Meeting, October 22-26, 1984.

3.9.2 L. L. Bonzon and D. B. Hente, Test Series 2: Seismic-Fragility Tests of Naturally-Aged Class 1E Exide FHC-19 Battery Cells, NUREG/CR-4095, SAND84-2628, Sandia National Laboratories, Albuquerque, New Mexico, March 1985.

3.9.3 L. L. Bonzon and D. B. Hente, Test Series 3: Seismic-Fragility Tests of Naturally-Aged Class 1E C&D LCU-13 Battery Cells, NUREG/CR-4096, SAND84-2629, Sandia National Laboratories, Albuquerque, New Mexico, March 1985.

3.9.4 L. L. Bonzon and D. B. Hente, Test Series 4: Seismic-Fragility Tests of Naturally-Aged Class 1E Exide EMP-13 Battery Cells, NUREG/CR-4097, SAND84-2630, Sandia National Laboratories, Albuquerque, New Mexico, March 1985.

3.9.5 L. L. Bonzon and D. B. Hente, Age-Related Degradation of Naturally-Aged Class 1E Battery Cells, NUREG/CR-4099, SAND84-2632, Sandia National Laboratories, Albuquerque, New Mexico, April 1986.

3.9.6 L. L. Bonzon and D. B. Hente, Seismic-Fragility Baseline Tests of New Class 1E Battery Cells, NUREG/CR-4098, SAND84-2631, Sandia National Laboratories, Albuquerque, New Mexico. (To be published)

International and Other U.S. Programs

3.9.7 J. Marie, Description of Some Electrical Endurance Tests for the Aging of Lead Storage Cells, HM 64/313, Electricite de France, Research Division, Paris, France, May 1984.

3.9.8 J. LeRoy, Technical and Qualification Testing Specification for Stationary Lead or Nickle-Cadmium Storage Cells for In-House Emergency Supply, HM 64/321, Electricite de France, Research Division, Paris, France, June 1984.

3.9.9 Letter Ph. Roussaire (Electricite de France) to L. Bonzon (SNL), September 24, 1984, Aging Tests of Batteries.

3.9.10 T. Lowen, Seismic Qualification Tests for Nuclear Generating Station Batteries, 77-221-H, Ontario Hydro Research Division, Toronto, Canada, May 17, 1977.

3.9.11 J. W. Janis, Seismic Testing of Aged Nuclear Generating Station Batteries - Cell Inspection and Capacity Tests, C84-5-K, Ontario Hydro Research Division, Toronto, Canada, February 21, 1984.

3.9.12 Wyle Laboratories, Seismic Simulation Test Program on Three KCY-23, Three KCY-25 and Two KC-19 Battery Cells Contained in a Type K Battery Rack, 43291-1, Huntsville, Alabama, June 24, 1976.

3.9.13 Wyle Laboratories, Seismic Simulation Test Program on a Battery Rack and Batteries, 43450-1, Huntsville, Alabama, December 7, 1976.

3.9.14 Wyle Laboratories, Seismic Simulation Test Program on a Two-Step Battery Rack and KC-9, KC-13, KC-17, and KC-21 Battery Cells, 44466-1, Huntsville, Alabama, March 12, 1979.

3.9.15 Wyle Laboratories, Seismic Simulation Test Program on One Battery Rack Containing Three (3) 60FPS-25 Battery Cells and One Battery Rack Containing Three (3) 60FPR-23 Battery Cells, 44681-1, Huntsville, Alabama, May 15, 1981.

3.9.16 Wyle Laboratories, Nuclear Environmental Qualification Program on Lead Acid Storage Batteries, 44681-2, Huntsville, Alabama, October 27, 1981.

3.9.17 J. E. Clifford and R. E. Thomas, Study of Battery Accelerated Testing Techniques Final Report, SAND82-7049, Contractor Report, Battelle Columbus Laboratories, Columbus, Ohio, April 1982.

3.9.18 B. P. Harraurod and T. M. Maynard, Jr., Nuclear Environmental Qualification Report of KC-9, KC-21, and LC-25 Battery Racks and Cells for Hope Creek Generating Station Units 1 and 2, A-472-82, Corporate Consulting and Development Company, Ltd., Raleigh, North Carolina, June 1982.

3.9.19 Wyle Laboratories, Seismic Multifrequency Random Test of Class 1E Equipment to be Used in the Diablo Canyon Site for Pacific Gas and Electric Company, 58836, Norco, California, March 1983.

3.9.20 J. E. Clifford and R. E. Thomas, Final Report on Analyses of Lead-Acid Battery Accelerated Testing Data, SAND83-7122, Contractor Report, Battalle Columbus Laboratories, Columbus, Ohio, August 1983.

3.10 Radiation Damage Thresholds

Issue Description

Item (8) of Regulatory Guide 1.89 (June, 1984), Section C.2.c. states that:

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

Section B, "Discussion" further elaborates on the position:

> "Item (8) of Regulatory Position C.2.c addresses qualification of equipment exposed to low-level radiation doses. Numerous studies that have compiled radiation effects data on all classes of organic compounds show that compounds with the least radiation resistance have damage thresholds greater than 10^4 rads and would remain functional with exposures somewhat above the threshold value. Thus, for organic materials, radiation qualification may be readily justified by existing test data or operating experience for radiation exposures below 10^4 rads. However, for electronic components, studies have shown failures in metal oxide semiconductor devices at somewhat lower doses. Therefore, radiation qualification for electronic components may have a lower exposure threshold."

A universal and accepted radiation "threshold" would benefit the industry and standardize the regulation process.

This is the thrust of this effort to address the issue:

> "Can radiation damage thresholds be established for organic materials, electronics, and equipment generally?"

Organic Materials

References 3.10.1 and 3.10.2 summarize the Sandia experiences on age-related degradation of polymeric materials and discuss the general methodology for accelerated aging. Both references "imply" that a total dose of about 10^5 rads could be considered a quasi-threshold value (for the materials evaluated by Sandia under this program). They also recognize and caution that material aging behavior is very material-formulation specific.

Electronics

Reference 3.10.3, NUREG/CR-3156 (SAND82-2559), attempted a survey of the state-of-the-art in aging of electronics. The study contained these findings:

1. Semiconductor and integrated circuits are the most sensitive and vulnerable parts in an electronic circuit.

2. Typical tolerances for present electronics are about 10^4 rad (Si) gamma and about $10^{13}n/cm^2$ for commercial hybrids and integrated circuits.

3. For specially fabricated or selected radiation-hardened devices, tolerances of about 10^6 rad(Si) gamma and about $10^{14}n/cm^2$ are typical.

4. In most testing to date, the interaction of operational temperature and long-term irradiation have not been investigated.

5. To understand the nuclear plant radiation environment effects on semiconductor devices, more work is needed on low dose-rate, long-term radiation effects.

6. Because the amount of damage sustained by a semiconductor is influenced by its circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

7. Both discrete devices and integrated circuits (op amps) are being used in some instrumentation in containment.

8. Combined neutron and gamma radiation environments should be examined in simulating radiation effects.

9. More work is needed to understand synergistic reactions and on predicting the effects of a combination of two or more stresses (e.g., radiation, temperature, humidity, chemical contamination, electrical stress, mechanical stress and other environmental factors).

Reference 3.10.4 reports on the operational testing of ambient-aged ITT Barton Model 332 differential pressure transmitters; the supplier estimated that 1 Megarad total dose was experienced over the in-plant life of the transmitters. While all instruments performed generally satisfactorily, the zero and range potentiometers were set close to one end of full travel, perhaps indicating a common mode change in the equipment. (Piecepart degradation evaluation have been completed and results will be reported.)

<u>International and Other U.S. Programs</u>

The Electric Power Research Institute (EPRI) has funded several studies on radiation effects.

Portions of the Reference 3.10.5 (RP 1707-3) abstract are presented here:

"Organic polymers are often identified as the weak elements in equipment. Data on radiation effects are summarized for 50 generic name plastics and 16 elastomers. Coatings, lubricants, and adhesives are treated as separate groups.

"Some semiconductor devices and electronic assemblies are extremely sensitive to radiation. Any damage threshold including these would be too low to be of practical value. With that exception, equipment exposed to less than 10^4 rads should not be significantly affected. Equipment containing no Teflon should not be significantly affected by 10^5 rads.

"Data concerning synergistic effects and radiation sensitization are discussed. The authors suggest correlations between the two effects."

Reference 3.10.6 (NP-4172M) offers these (selected) conclusions reached in compiling and reviewing available radiaton effects data on synthetic organic polymers:

"Both the lowest reported threshold dose and the 25% change dose level vary widely within each of the five categories of synthetic polymers investigated.

"This study confirms the results from the previous investigation (Reference 3.10.5) in that virtually all of the materials have thresholds above 10^5 rads. Five of the materials were found to have thresholds below 10^5 but above 10^4 rads, namely:

(1) Fluorocarbons
(2) Nylons
(3) Epoxy resins
(4) Polyethylene oxides
(5) Poly-alpha-methyl chloroacrylate.

"The radiation data in this report must be applied on a case-by-case basis. These data present radiation responses of pure materials in a simple environment, i.e., radiation. In real qualification applications, complicating material response factors, i.e., synergisms, sequential responses and sensitizations may become important."

EPRI is currently establishing a compendium of publicly available materials data which have been used in aging calculations for qualification of nuclear power plant equipment, Reference 3.10.7. Currently limited to thermal data, radiation records are now being assembled.

The Atomic Industrial Forum, Reference 3.10.8, has made the following recommendation with discussion:

"Recommendation

"That equipment located in areas where total integrated gamma dose levels over the life of the plant or equipment including the post accident doses do not exceed 10E5 RADS (conservatively estimated to be one order of magnitude less than the threshold of detectable effects) should be exempt from requirements to demonstrate its radiation aging capabilities by type testing and/or analysis. For solid state electronic devices, the threshold level should be 10E3 RADS gamma.

"In the context used herein, threshold damage radiation dose level is defined as that amount of integrated gamma dose which when absorbed by the material at a prescribed dose rate, causes degradation as measured by loss of 25% engineering properties of interest. Review of available data in various equipment qualification documents from various test facilities indicate that if the dose rate(s) at which this integrated dose is applied is within the range of 0.1 to 1 Megarads/Hr, no significant dose-rate dependent effects may be present. It is, however, noted that dose-rate effects are not well understood and defined for all materials and that further generic study is required to establish the proper range of dose rates."

The French are known to have a program devoted to radiation effects on sensitive electronics, but complete details are lacking. References 3.10.9 (CEA Report #175) and 3.10.10 (CEA Report #176) are listed merely to acknowledge their efforts.

Reference 3.10.11 discusses the determination of radiation thresholds by using thermogravimetric analysis (TGA) techniques. Samples of phenolic material were irradiated to various doses and then TGA curves were obtained for each sample. From these curves an activation energy was determined. The radiation threshold was evaluated by plotting the activation energy versus the radiation dose. Figure 3.10.1 demonstrates results for a phenolic material.

Reference 3.10.12 summarizes some radiation testing experiences for electronic components, especially capacitors, resistors, and transistors, and notes that many of these components have been qualified to extremely high radiation levels such as 2×10^8 rads. Table 3.10.1 summarizes radiation levels to which some electronics are qualified. Reference 3.10.12 describes a test program to evaluate the radiation level to which a CMOS integrated curcuit would operate. The program concluded that below 2.6×10^3 rads, the CMOS IC would operate properly for qualification purposes.

Issue Findings

References 3.10.1, 3.10.2, 3.10.3, 3.10.5, 3.10.6, 3.10.11, and 3.10.12 indicate the strong (industry wide) interest and effort in developing a radiation threshold data base for materials and some equipment types.

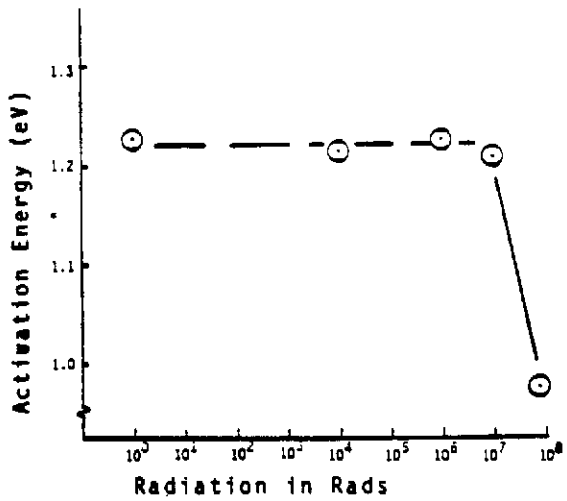

Figure 3.10.1: Radiation threshold for phenolic determined by thermogravimetric analysis. (From Reference 3.10.12)

Type of Electronic Component	Radiation Level to Which Qualification Has Been Determined (rad)
Resistors	
Carbon composition	2×10^8
Wire wound	2×10^8
Metal film	2×10^8
Capacitors	
Ceramic	2×10^8
Aluminum	2×10^8
Paper	2×10^7
Polyester	2×10^7
Tantalum	2×10^8
Diodes, silicone	2×10^8
Transistors, silicone	2×10^8
Integrated circuits	
CMOS	2.6×10^3
TTL	5×10^7
Op amps	5×10^7
Optical couplers	1×10^6
NMOS	5×10^2

Table 3.10.1 Radiation levels to which some electronics are qualified (From Reference 3.10.12)

The intent is to make this information available for use in developing adequate equipment qualification tests.

While much data potentially applicable to radiation threshold determination has been and is being assembled, a thorough independent review of the data sources should be conducted to determine any limits of applicability.

In addition, by performing low dose, and low dose rate, tests on generic and identical test samples of organic materials and electronics considered to be the most potentially sensitive to radiation degradation, it should be possible to assess whether universal "threshold" values can be established. The test results, in concert with engineering judgment, may then be used to determine an applicable accelerated radiation aging method, or whether exemptions to radiation aging can be generically allowed, and at what levels.

Organic Materials

There seems to be a general consensus that a radiation damage threshold can be established for most organic materials in the range of 10^4-10^5 rads. Regulatory Guide 1.89 suggests a value of 10^4 rads.

In summary then, the findings within this subissue are:

1. A radiation damage threshold may be established.

2. The effort to develop a radiation threshold data base for materials should continue.

3. An independent review of the data sources and limited testing should be conducted to determine the limits of applicability and/or the accuracy range.

Electronics

Numerous reports suggest that electronics are sensitive to radiation damage. A threshold value has not been determined. Several worldwide research programs may produce useful data in this area.

In summary then, the findings within this subissue are:

1. Both discrete devices and integrated circuits (op amps) are being used in some instrumentation in containment. To understand the nuclear plant radiation environment effects on semiconductor devices, more work is needed on low dose-rate, long-term radiation effects.

2. In most testing to date, the interaction of operational temperature and long-term irradiation has not been investigated. Because the amount of damage sustained by a semiconductor is influenced by its

circuit application (powered-forward or reversed bias, or not powered), it is important to consider how a device is used...when evaluating aging phenomena.

3. Combined neutron and gamma radiation environments should be examined in simulating radiation effects.

4. More work is needed to understand synergistic reactions and on predicting the effects of a combination of two or more stresses (e.g., radiation, temperature, humidity, chemical contamination, electrical stress, mechanical stress and other environmental factors).

5. The effort to develop a radiation threshold data base for electronic components should continue.

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Appendix

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Since its inception in 1975, the Qualification Testing Evaluation (QTE) Program has produced numerous results pertinent to equipment qualification issues. Many have been incorporated into Regulatory Guides, Rules, and industry practices and standards.

This report summarizes the numerous reports and findings to date. Thirty separate issues are discussed encompassing three generic areas: accident simulation methods; aging simulation methods; and, special topics related to equipment qualification. Each issue-specific section contains: (1) a brief description of the issue; (2) a summary of the applicable research effort; and (3) a summary of the findings to date.

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