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QUALIFICATION TESTING EVALUATION PROGRAM
LIGHT-WATER REACTOR SAFETY RESEARCH
QUARTERLY REPORT

July - September 1979

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EXECUTIVE SUMMARY

The July-September 1979 quarter can be characterized as a period of formal reporting and continuing effort in the Qualification Testing Evaluation (QTE) Program.

The national and international interest in the program remains very high. Numerous requests for information and reports were processed. The invitation to visit Japan to participate in mutual discussion of equipment qualification issues was accepted and an itinerary scheduled.

Under Task 1, the principal effort was devoted to the preparation for continuation of verification tests of the qualification of Browns Ferry Unit 3 connector assemblies. Radiation and thermal-aging experiments were completed last quarter; checkout of the superheat test capability was continued in preparation for the accident-simulation test phase.

Under Task 3, effort was directed towards the effects of the sorption characteristics of a material with respect to quantitatively accelerating the degradation. When experiments are properly run and analyzed, Arrhenius behavior is often found.

The balance of the quarterly effort centered on continuation of ongoing projects. Within the methodologies task, the test-facility upgrade and preparation for the connector tests received primary emphasis. Within the radiation-source task, effort continued on the development of a "best-estimate" LOCA-radiation signature and the drafting of a "simulator-adequacy" assessment report. Within the accelerated-aging task, the primary emphasis was on fire-retardant aging tests, continued combined-environments testing of electrical cable materials, the evaluation of ambient-aged electrical cable samples, and preparation of test plans for seals/gaskets evaluations.

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CHAPTER 1. PROGRAMMATIC OVERVIEW

Programs were initiated in late 1974 to evaluate the significance of synergistic effects in post loss-of-coolant accident (LOCA) testing of Class 1E safety-related equipment. As a result of these activities, two more complementary tasks were identified and initiated in late 1975; these were (1) to evaluate and improve accelerated-aging methodologies and (2) to determine the nuclear-source term as specified in Regulatory Guide 1.89.[1.1] In late 1976, these three tasks were integrated into a broader program, Qualification Testing Evaluation (QTE); the goal was to evaluate the overall adequacy of the qualification testing of safety-related equipment and to resolve specific anomalies and uncertainties with qualification testing as outlined in IEEE-323-1974.[1.2]

The objectives of the QTE program are to obtain data needed for confirmation of the suitability of current standards and regulatory guides for Class 1E safety-related equipment and to obtain data that will provide an improved technical basis for modifications of these standards and guides where appropriate. Specific major objectives of the research are

- To provide assessments of post-LOCA qualification-testing methodologies, including a qualitative assessment of the synergistic effects resulting from the combined environmental testing of representative Class 1E equipment;
- To determine the radiation environment from the nuclear source term for a design basis LOCA and evaluate the adequacy of radiation simulators; and
- To provide methods that can be used to simulate the natural aging process of representative Class 1E materials by accelerated-aging methods.

This program addresses three distinct tasks of concern in the type-testing of Class 1E equipment that reflect the objectives stated above. Under Task 1, LOCA testing methodologies and anomalies will be studied to define testing details and to identify potential weaknesses in safety-system components and materials. For example, the possible existence of synergistic effects will be determined for a range of typical components. These synergisms would result from the simultaneous applications (on identical components) of the LOCA environments as compared to the sequential application of radiation, followed by the other LOCA environments.

The Task 2 effort involves an assessment of the magnitudes of the prescribed LOCA-radiation sources, an evaluation of existing radiation simulators, an evaluation of component response to the LOCA-radiation signatures, the development of guidelines and rationale for use of radiation simulators in typetesting of Class 1E components, and the definition of the LOCA-radiation signature based on "best estimates" of the accident progression and fission product-release sequences.

Typetesting requires a component that is "aged" to simulate normal degradation during its design life under exposure to the ambient environments existing in nuclear power plants. In Task 3, a proposed accelerated-aging method will be experimentally verified for single and combined stress (i.e., potentially synergistic) environments and, where available, "benchmark data" will be obtained from naturally-aged materials existing in nuclear power plants.

1.1 Task 1 -- Qualification-Testing Methodologies Assessment

The FY79 effort under this task is concentrated in seven broad areas and numerous related subtasks:

a. The High Intensity Adjustable Cobalt Array (HIACA) test facility upgrade is to be completed to (1) accommodate large and more diverse Class 1E test items and (2) allow selectable radiation dose rates and minimize radiation spatial gradients. The facility upgrade will include new radiation sources, source positioning equipment, facility shielding and cell modifications, test chambers, and diagnostic/test equipment.

b. At a specific request of the Commissioners,[1.3] additional tests of connectors in a simulated-LOCA environment may be defined and conducted using connectors qualified according to the IEEE-323 standards (see References 1.2 and 1.4).

c. Short- and long-range test plans, project descriptions, and quality assurance programs will be developed to accomplish the coordinated usage of the upgraded test facility.

d. A basis for the test plans is an evaluation of the apparent "LOCA-sensitivity" of safety-related equipment. An initial data base for this vulnerability evaluation has been obtained by subcontract to an architect-engineer for a generic pressurized-water reactor (PWR) and includes Class 1E equipment lists, manufacturers, normal- and accident-environments definition, and comprehensive-data packages for each equipment item. This data file may be complemented and updated (as required) by the acquisition of a generic boiling-water reactor (BWR) and "old" plant information.

e. Offsite data to complement the Sandia test data will be acquired, as availability allows, through subcontracts to manufacturers, testing laboratories, etc. As the testing schedule requires, offsite testing may be conducted to serve as benchmarks or supplements to the Sandia test effort.

f. Initial methodology testing will begin based on the test plan(s) development and approval; these tests could include synergistic effects tests on larger/other components (e.g., valve operators, sensors, penetrations, cable assemblies) or investigations into additional qualification concerns (e.g., superheated steam, pressure-rate phenomena, pressure-/temperature-magnitude effects, dose-rate effects, component configuration influences). Some effort may be devoted to developing requalification test methods using naturally-aged components; similarly, other specific confirmatory tests may be performed under this subtask. An initial attempt will be made to develop typetesting methodologies and standard laboratory techniques to assure comprehensive and repeatable test sequencing; this could include development of screening procedures (tests) to determine rank-order component vulnerability and appropriate failure criteria for components.

g. Initial effort may be directed toward the problem of statistical qualification such as evaluating over-test methods for "statistical" equivalence of multiple tests at less severe environments.

1.2 Task 2 -- Radiation-Qualification Source Evaluation

The FY79 effort under this task is concentrated in four broad areas and related subtasks:

a. The evaluation of radiation-simulator "adequacy" will be completed for exposed organic materials using dose rate, depth dose, and material damage parameters as indicators. This evaluation may be extended to include other safety-related equipment and material as well.

b. A "best-estimate" LOCA-radiation signature will be completed and will be based on the accident-time-release sequencing as specified in WASH-1400[1.5] and References 1.6, 1.7, and 1.8. The signature will eliminate several unrealistic but conservative assumptions specified in Regulatory Guide 1.89[1.9] and may be the basis of a revised guide. The "best-estimate" LOCA-radiation signature effort may continue by incorporating new data base. This signature may also be used where applicable to judge simulator adequacy.

c. Guidelines for the formulation of radiation-qualification testing specifications and dose and dose-rate estimates for a generic containment structure may be developed based upon USNRC-defined need and contingent upon available funding.

d. Depending on the results of these subtasks, effort may be directed toward (1) experimental verification of simulator adequacy, (2) tailoring/designing of simulators to achieve better duplication of the actual component-damage profiles, (3) devising benchmark calculations of LOCA-radiation environments and component damage to assist in the evaluation of the computational capabilities of Class 1E equipment qualifiers, and (4) developing guidelines and rationale for the use of simulators in type-testing.

1.3 Task 3 -- Accelerated-Aging Study

The FY79 effort under this task is concentrated in six broad areas and numerous related subtasks:

a. Single-environment aging tests will be continued on electrical cable materials; tensile properties are used as measures of damage in these tests. Single-environment acceleration functions of damage versus time will be obtained. Similarly, aging tests in combined radiation and temperature environments will be continued in order to determine the importance of "synergisms" and to test the method postulated for combined-environment accelerated aging.[1.10]

b. The study of the effects of aging with regard to the retention of flame-retardant additives in cable materials (begun in FY78) will be continued. Test specimens have been made of common polymer materials with known fire-retardant additives; these will be subjected to accelerated aging and undergo quantitative testing to determine change in fire-retardant content and flammability with age. This effort may be extended by including aging at temperatures near the ignition temperature(s) to assess the effectiveness of fire-retardant cable in the environs of a slowly developing fire. An extension of this work may include aging of fireproof coating materials.

c. Alternate indicators of damage will continue to be invesigated under this task; examples of such indicators are voltage withstand, mandrel-bend tests, dissipation factor, equivalent series/parallel resistance, on changes in concentrations of various chemical species. Some of these tests more closely parallel current industry-failure criteria that require "functionability" of electrical cable. The evaluation of alternate real-time damage indicators (to complement mechanical properties) will continue, aimed toward the development of a parameter indicative of age (e.g., dissipation factor) that can be evaluated on-line.

d. The acquisition and analyses of ambient-aged cable (when available) to serve as benchmarks to the accelerated-aging tests will continue. Prior experience indicates that the nuclear-plant ambient environments are poorly defined. Unless reliable environmental information can be obtained, ambient-aged cable samples are of limited value. Additional combined-environments tests will be run on Savannah River Plant reactor cables where important synergistic effects have been observed.

e. The task effort will be extended to include components and materials other than electrical cables. It is proposed that aging techniques be evaluated for elastomeric seals; epoxies and motor windings are also under consideration. Only preliminary evaluations, test plans, and some initial testing can be accomplished in FY79. Other environmental conditions (such as mechanical stress and gaseous pollutants) will also be included in these evaluations.

f. As an alternate to the accelerated-aging method, other methods of estimating age or equipment life can be evaluated. Such a method could employ "sacrificial samples;" resistance to aging degradation for short periods of time would be experimentally verified and requalification tests would be developed and used to extrapolate the remaining accept able "life" of the equipment.

1.4 Quarterly Programmatic and Common-Task Activities

The several programmatic activities necessary for continuity and development are highlighted in this section. Technical activities specific to each task are in Sections 2.1, 3.1, and 4.1, which follow.

Program Formalization -- Standard Buff Book submittals were prepared and submitted on July 5 and September 20. Until agreement is reached on the connector tests and the long-range program objectives among NRC staff, the submittals will continue to reflect that uncertainty.

The routine reporting of program activities is done through formal quarterly reports issued 4 to 6 months after the close of the reported quarter. A draft of the second quarter report (January-March 1979)[1.11] was completed in late June and proceeded through the review and sign-off process during this quarter.

Program Reviews with NRC Staff -- A complete program review was held for R. Feit and G. Bennett in Silver Spring, Maryland, on July 18-19. All major tasks, and program status, were reviewed. Under Task 1, the principal discussion included: joint U.S.-French tests to be conducted at the KALI Loop, Cadarache, facility; tentative schedule, agenda, and papers that are a part of the Japan trip; status of the connector assemblies tests and NRC concerns about the test profiles. Under Task 2, the status of the simulator adequacy evaluation/report and the "best-estimate" LOCA radiation signature effort were reviewed. Under Task 3, discussion centered on a test plan submitted by Naval Research Laboratories, the need for a test plan for accelerated aging evaluation of seals and gaskets, and the possibility of a "user's guide" for the aging methodology developed. In addition, some discussion of TMI activities and valve testing was held.

On July 18, discussion of TMI-related activities was held with C. Kelber, R. Foulds, and R. Feit in Silver Spring. At issue was the framework for coordination with Department of Energy staff and the need to develop test plans to evaluate TMI instrumentation and safety-related equipment.

A general program review was held for T. Murley, R. Feit, G. Bennett, J. Zwolinski, and W. Rutherford at Sandia on September 25-27. The principal review centered on the connector tests, along with a tour of the HIACA and test facility upgrade in TA-V. In addition, recent data from the fire-retardant aging study was discussed and the test plan for accelerated aging studies of seals/gaskets was reviewed.

Meetings and Conferences Participation -- On July 17-20, A. W. Snyder attended the 1979 Annual Conference on Nuclear and Space Radiation Effects and presented an invited paper[1.12] which summarized the QTE program and discussed the "effects" of the Three Mile Island (TMI) incident. The full text of that paper is reproduced as Appendix A of this report.

L. Bonzon attended the 9th Biennial Conference on Reactor Operating Experience, held August 5-8 in Arlington, Texas.

Foreign Interest -- On September 13, P. F. Melo of the Servico Publico Federal, Brazil, requested a number of QTE Program reports and also requested that he be added to the distribution for all future reports.

By letter dated January 29, Dr. K. Yahagi (Electrical Engineering Department, Waseda University, Tokyo) had invited L. Bonzon and others to Japan in the fall of 1979 to discuss and review equipment qualification programs; Dr. Yahagi was acting in his capacity as Chairman of the Committee of the Institute of Electrical Engineers of Japan (IEEJ) on the Ionizing Radiation Resistance of Electrical Insulating Materials. Follow-up correspondence on this matter was contained in letters dated July 25, September 17, and September 21; in them, a trip agenda and program was confirmed which called for lectures to IEEJ Sections in Tokyo, Osaka, and Hitachi, and visits to major nuclear power research establishments. Also during this period, two (abstracts) papers[1.13,1.14] were prepared and sent to Japan as background for the invited lectures; these were sent by letters dated August 28 and September 17.

Additional correspondence with the Japanese occurred by letter dated August 1. In this, Dr. Y. Nakase of JAERI forwarded information on the activities of the IEEJ Committee.

Industry Liaison -- The general interest in the overall QTE program remains high. A number of industry requests were received and processed during this quarter; these are briefly reviewed in Table 1.1. In addition to the routine requests listed in the table, two deserve elaboration.

G. Slider (EPRI) and J. Wanless (NUS) visited Sandia Laboratories on August 30 to discuss the QTE Program. A number of reports were exchanged. Specifically, EPRI provided a copy of an internal report, "EPRI Equipment Qualification Program," which describes some applicable equipment qualification efforts to be implemented under EPRI funding.

Regulatory Guide 1.131[1.15] was also the subject of intense interest by the industry. Some discussion was held at the (above) August 30 meeting. Telecons were also made with NUS staff on September 17. The concern is for the statement that:

Table 1.1

Industry Liaison

Date	Company	Requests: Prior Reports	Mailing List	Other
6 July	Combustion Engineering			Radiation effects in electronic chips.
9 July	Nuclear Services Corp.			Information on work possibilities, based on an unsolicited proposal.
12,16 July	Bechtel-Gaithersburg	SAND78-0067		Testing of splice assemblies; moisture penetration through standard electrical cable.
24 July	Gulton			Update on the QTE program, status of connector assemblies tests.
30 July	Westinghouse-Arizona	SAND76-0740		Radiation qualification testing to 150MR(γ) and 900MR(β), applicability of LINAC irradiators, concern for γ and β spectra importance.
31 July	Gilbert Associates	SAND78-2254 SAND79-0761		
27 August	Ebasco	SAND76-0715 SAND76-0740 SAND78-0091 SAND79-0761	2 Names	
29 August	Westinghouse			Use of silicone materials in LOCA/radiation environments.
29 August	General Atomic			Arrange for presentation to IEEE WG 2.11 meeting in Hartford.
30 August	EPRI/NUS	SAND78-2254 SAND79-0761		Visit to Sandia to discuss QTE Program.
14 Sept.	GE, Wire & Cable			Recent developments in QTE Program which might impact GE qualification programs.
17 Sept.	NUS			Regulatory Guide 1.131, for comment, which requires evaluation of synergistic effects in accelerated aging of electric cable and splices.
20 Sept.	Sargent & Lundy		X	
20 Sept.	Puget Sound Power & Light		X	

"Aging data shall be developed to establish long-term performance of the insulation. Synergistic effects on aging due to simultaneous application of environmental conditions shall be considered in the accelerated aging program. Investigation shall be performed to determine if there are synergistic effects and, where identified, they shall be accounted for in the qualification program . . ."

The Guide further references SAND78-0799[1.16] as a basis for the statement.

TMI-2 Equipment Recovery and Evaluation -- A series of meetings were held this quarter to coordinate the effort on TMI-2 equipment evaluation. A Memorandum of Understanding to establish the organizational structure needed for TMI containment building re-entry operation was drafted on July 5. This was discussed with NRC staff in Silver Spring on July 18. A revised draft was prepared by Sandia staff and dated July 27. A presentation to DOE staff, including this effort, was made by Sandia staff on August 13-14; agreement was reached on ten tasks as documented in "TMI-Related Tasks Identified in 8/14/79 GPU/DOE Meeting as Being of Joint Benefit," one of these was concerned with "Instrumentation Survivability." This led to a task entitled "Instrumentation and Electrical Equipment Survivability Planning Group" on September 13, with R. Damerow, Sandia, as Chairman. The balance of the quarter was spent in defining the role and membership of the Planning Group.

Supplemental funding was received from NRC on September 21 to establish a fourth task in the QTE program, to "initiate a study of the vulnerability of the TMI-2 safety-related equipment." First efforts are to be directed to terminal blocks evaluations.

1.5 Publications and Presentations

The following is an inclusive list of formal publications and presentations for FY79 which detail aspects of the QTE Program. Those marked by an asterisk (*) became available during the reported quarter.

The first eight papers were presented and/or published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16 through 19 October 1978.

1. R. E. Luna and L. L. Bonzon, Methodology Assessment: An Overview of the Qualification Testing Evaluation (QTE) Program, SAND78-0342 (Albuquerque: Sandia Laboratories).

2. K. T. Gillen, A Method for Combined Environment Accelerated Aging, SAND78-0501 (Albuquerque: Sandia Laboratories).

3. K. T. Gillen and E. A. Salazar, Aging of Nuclear Power Plant Safety Cables, SAND78-0344 (Albuquerque: Sandia Laboratories).

4. L. L. Bonzon, An Experimental Investigation of Synergisms in Class 1E Components, SAND78-0346 (Albuquerque: Sandia Laboratories).

5. S. G. Kasturi, G. T. Dowd, and L. L. Bonzon, Qualification of Class 1E Equipment: The Role of the Utility and Architect-Engineer, SAND78-0347 (Albuquerque: Sandia Laboratories).

6. N. A. Lurie and L. L. Bonzon, The Hypothesized LOCA Radiation Signature and the Problem of Simulator Adequacy, SAND78-0348/IRT 8167-005 (Albuquerque: Sandia Laboratories).

7. N. A. Lurie and L. L. Bonzon, The Best-Estimate LOCA Radiation Signature: What It Means to Equipment Qualification, SAND78-0349/IRT 8167-006 (Albuquerque: Sandia Laboratories).

8. L. L. Bonzon, R. E. Luna, and S. P. Carfagno, Qualification Issues: The Rest of the Iceberg, SAND78-0350 (Albuquerque: Sandia Laboratories).

9. K. T. Gillen and E. A. Salazar, A Model for Combined Environment Accelerated Aging Applied to a Neoprene Cable Jacketing Material, SAND78-0559C (Albuquerque: Sandia Laboratories). Presented at and published in the Proceedings of the 1978 Conference on Electrical Insulation and Dielectric Phenomena, Pocono Manor, PA, 29 October through 2 November 1978.

10. L. L. Bonzon, Status of the Qualification Testing Evaluation (QTE) Program, SAND78-1884C (Albuquerque: Sandia Laboratories). Presented at USNRC Sixth Water Reactor Safety Research Information Meeting, 6 through 9 November 1978.

11. K. T. Gillen, Experimental Verification of a Combined Environment Accelerated Aging Method Applied to Electrical Cable Material, SAND78-1907C (Albuquerque: Sandia Laboratories). Presented at USNRC Sixth Water Reactor Safety Research Information Meeting, 6 through 9 November 1978.

12. L. L. Bonzon et al, Qualification Testing Evaluation Program, Quarterly Report, April-June 1978, SAND78-1452, NUREG/CR-0401 (Albuquerque: Sandia Laboratories, November 1978).

13. K. T. Gillen, Accelerated Aging - Principles and Techniques, SAND79-0034A (Albuquerque: Sandia Laboratories, January 1979). Invited Seminar, Los Alamos Scientific Laboratory.

14. N. A. Lurie, Calculations to Support Evaluation of Test Sources for Radiation Qualification of Class 1E Equipment, Final Report - Phase 5 and 6, IRT 8167-010(A), January 1979. (Report for Sandia use only; to be included in forthcoming SAND report.)

15. L. L. Bonzon, K. T. Gillen, and D. W. Dugan, Qualification Testing Evaluation Program, Quarterly Report, July-September 1978, SAND78-2254, NUREG/CR-0696 (Albuquerque: Sandia Laboratories, March 1979).

16. N. A. Lurie, Calculations to Support Evaluation of Test Sources for Radiation Qualification of Class 1E Equipment, Final Report - Phases 5 and 6, IRT 8167-010(B), April 1979. (Report for Sandia use only; to be included in forthcoming SAND report.)

17. R. L. Clough, K. T. Gillen, and E. A. Salazar, A Study of Strong Synergism in Polymer Degradation, SAND79-0924C (Albuquerque: Sandia Laboratories). JOWOG-28 Meeting, Albuquerque, 11 through 14 June 1979.

18. K. T. Gillen, Implications of Sorption Effects on Accelerated Aging Studies, SAND79-0925C (Albuquerque: Sandia Laboratories). JOWOG-28 Meeting, Albuquerque, 11 through 14 June 1979.

19. K. T. Gillen, A Method for Accelerated Aging Under Combined Environmental Stress Conditions, SAND79-0939C (Albuquerque: Sandia Laboratories). JOWOG-28 Meeting, Albuquerque, 11 through 14 June 1979.

20. R. L. Clough and E. A. Salazar, Comparison of Polymer Flammabilities and a Determination of the Loss of Fire-Retardant Additives with Aging, SAND79-0978C (Albuquerque: Sandia Laboratories). JOWOG-28 Meeting, Albuquerque, 11 through 14 June 1979.

21. L. L. Bonzon, K. T. Gillen, and E. A. Salazar, Qualification Testing Evaluation Program, Quarterly Report, October-December 1978, SAND79-0761, NUREG/CR-0813 (Albuquerque: Sandia Laboratories, June 1979).

*--22. N. A. Lurie, Best-Estimate LOCA Radiation Signature, Final Report, Phases 2 through 4, IRT 0056-005, July 1979. (Report for Sandia use only; to be included in forthcoming SAND report.)

*--23. A. W. Snyder, L. L. Bonzon, and B. L. Gregory, Potential Environments and Effects in Reactor Accidents, Invited Paper, 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, CA, July 17-20, 1979.

1.6 References

1.1 "Qualifications of Class 1E Equipment for Nuclear Power Plants," Regulatory Guide 1.89 (23 September 1974).

1.2 IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, IEEE Std 323-1974 (New York: The Institute of Electrical and Electronic Engineers, Inc., 1974).

1.3 "Memorandum and Order, In the Matter of Petition for Emergency and Remedial Action," USNRC Commissioners (13 April 1978).

1.4 IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations, IEEE Std 323-1971 (New York: The Institute of Electrical and Electronic Engineers, Inc., 1971).

1.5 "Appendix VII: Release of Radioactivity in a Reactor Accident," in Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, NUREG-75/014, U.S. Nuclear Regulatory Commission (October 1975).

1.6 D. Y. Hsia and R. O. Chester, A Study of the Fission Product Release From a Badly Damaged Water-Cooled Reactor, ORNL-TM-4702 (Oak Ridge: Oak Ridge National Laboratory, June 1974).

1.7 Core-Meltdown Experimental Review, SAND74-0382 (Rev), NUREG-0205 (Albuquerque: Sandia Laboratories, March 1977).

1.8 M. J. Kolar, J. R. McCarty, and N. C. Olson, "Post-Loss-of-Coolant Accident Doses to Pressurized Water Reactor Equipment," Nuclear Technology, 36 (November 1977) pp 74-78.1.9

1.9 "Qualification of Safety-Related Electric Equipment for Nuclear Power Plants," Regulatory Guide 1.89, Rev. 1, Draft (1 November 1976).

1.10 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, SAND76-0715, NUREG-0237 (Albuquerque: Sandia Laboratories, July 1977).

1.11 L. L. Bonzon, R. L. Clough, and E. A. Salazar, Qualification Testing Evaluation Program, Quarterly Report, January-March 1979, SAND79-1314, NUREG/CR-0970 (Albuquerque: Sandia Laboratories, November 1979).

1.12 A. W. Snyder, L. L. Bonzon, and B. L. Gregory, Potential Environments and Effects in Reactor Accidents, Invited Paper, 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, CA, July 17-20, 1979.

1.13 L. L. Bonzon, The Qualification Testing Evaluation (QTE) Program, SAND79-1553A (Albuquerque: Sandia Laboratories, October 1979).

1.14 L. L. Bonzon, Some thoughts on the Three Mile Island Accident and the Qualification of Safety-Related Equipment, SAND79-1554A (Albuquerque: Sandia Laboratories, October 1979).

1.15 Regulatory Guide 1.131, Proposed Revision 1 for Comment, "Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1979.

1.16 L. L. Bonzon, K. T. Gillen, and F. V. Thome, Qualification Testing Evaluation Program, Quarterly Report, January-March 1978, SAND78-0799, NUREG/CR-0276 (Albuquerque: Sandia Laboratories, August 1978).

CHAPTER 2. QUALIFICATION-TESTING METHODOLOGIES ASSESSMENT

The activities under Task 1 are numerous and diverse. The programmatic activities were discussed in Section 1.4; Section 2.1 highlights the various technical activities. The significant efforts were (1) the continuing work on the Browns Ferry Unit 3 nuclear station (BF3) connector tests, and (2) progress on the test facility upgrade.

2.1 Task 1 -- Technical Activities Summary

UEC Subcontract -- United Engineers and Constructors (UEC) has effectively completed a four-concurrent-phase subcontract to assemble comprehensive data packages for all in-containment Class 1E equipment in a contemporary PWR nuclear power plant. The contract remains in force through FY79 to allow the necessary revisions and updates to be made in a timely manner. There was no specific effort on this contract relative to the QTE Program.

Franklin Institute Research Laboratories (FIRL) Subcontract -- This contract continues through FY79 for the FIRL staff to provide general assistance to Sandia in the evaluation of Class 1E equipment data packages, in the development and critique of test plans, and in other general matters pertaining to the QTE Program. There was no specific effort on this contract relative to the QTE Program.

Connector Tests -- Considerable effort was directed towards the connector retest program as directed by the NRC Commissioners.[2.1] Following NRC permission to initiate testing (in early April), three each of two types of the Browns Ferry Unit 3 nuclear station (BF3) connector assemblies were subjected to a repeated (partial sequence) LOCA/MSLB simulation test program.[2.2] The objective of this test program is to duplicate as closely as practicable the original qualification tests;[2.3] by so doing, it may be possible to verify the original qualification test program. During April, the sequence was completed through the "Postaging Functional/Mechanical" tests. The balance of that quarter involved (1) demonstration of a superheat capability, and (2) continuing NRC review and determination of the desired superheat, pressure and temperature time-history profiles.[2.2] No final agreement had been reached by July 1.

During July, a number of discussions and telecons with NRC staff were held to assist in establishing the desired profiles. This included a meeting with R. Feit and G. Bennett in Silver Spring on July 18-19. Partial agreement was reached by NRC internal memorandum dated August 1, NRR to RSR, and further requirements (of Sandia) were telecopied on August 2. Based on these, the test equipment was reassembled during August to try to achieve the new profile; additional clarifications were also made to the test profile during this period.

During September, a series of superheat test runs were made; most were marginal with respect to the 2-minute rise time requirement. On September 4, a fourth-draft revision to the Connector Assembly Test Plan was submitted for NRC review. On September 10, a copy of the test procedures was submitted for NRC review along with test data from the superheat runs. Also on September 10, J. Zwolinski visited Sandia to observe the testing and to examine the test apparatus. During the period September 12-21, additional test runs were made, NRC reviewed and clarified the test plans and procedures, and plans were made to conduct the LOCA/MSLB simulation during the week of September 24. By separate letters dated September 19, RSR and IE staff agreed to proceed with the tests (with additional clarification).

On September 25 and 26 three "dry-run" tests were performed for NRC staff, but were marginally slow in desired rise time. On September 27 and 28, discussions were held with Sandia staff and T. Murley, G. Bennett, R. Feit, J. Zwolinski, and W. Rutherford. It was agreed that the test apparatus was too marginal and that modifications would be made during October by installation of new test facility upgrade equipment.

Based on the tentative selection, by NRC staff, of the second in the series of connector assemblies tests, some preliminary effort was devoted to reviewing the qualification package for the Duke Power, D. G. O'Brien, McGuire penetration/connector assemblies.[2.4] Discussion of the tests was held with W. Rutherford in Bethesda on July 19.

Facility Upgrade -- The existing test capability at the Sandia Gamma Irradiation Facility (GIF) is being upgraded to accommodate larger Class 1 test items. Along with increased size, the principle capability improvement is the addition of the high-intensity adjustable cobalt array (HIACA). The HIACA will have the ability to select radiation dose rate while holding the spatial gradient stationary. The upgrade includes

a. new radiation sources and source-positioning apparatus

b. a new source elevator system for storing HIACA under water

c. an electrohydraulic control system to select irradiation rates and to interlock for safety

d. irradiation cell modifications

e. test chamber and steam supply system

f. diagnostic/test equipment.

The major efforts this quarter are detailed below.

The cobalt positioning fixture, also called the high-intensity adjustable cobalt array (HIACA), was completed and delivered this quarter. Because of problems associated with modifications to the elevator system, the fixture has yet to be installed and tested.

Elevator modifications were completed during the last quarter, but a change in the method of installing/assembling the HIACA fixture on the rolling cart, and the cart to the elevator, has resulted in further modifications. Originally the method was to assemble the HIACA outside the facility and on the rolling cart. This entire assembly was to be lowered into the GIF pool, outside the cell, and then rolled on its set of tracks onto the elevator. In the event that even small problems were encountered with the telescoping HIACA fixture, the entire procedure would have to be reversed in order to remove it from the pool. The elevator and cart were modified further to permit raising the elevator completely into the cell to allow installation and removal of HIACA from inside the cell. The primary difficulty and delay was caused by the hydraulic hoses which must remain attached to the cart while passing through the cell floor opening. Another nonrelated elevator modification was associated with the elevator dashpots. The bolts that keep the dashpot cylinders from falling off when the elevator is raised had a probability of shearing off with normal elevator operation with the dashpot under compression.

The winch used to raise and lower the elevator was mounted outside and on the GIF roof. This winch was designed to raise 3,000 lbs, to slip at 2,000 lbs through the use of an adjustable slip clutch, to operate manually with a hand crank, to have 2 speeds to accommodate two methods of elevator lifting, and to operate with limit switches adjustable within the desired operating range. The winch will be load tested during the next quarter. Another aspect of the elevator system that required further modifications was the winch cable support system. The cables cradle the elevator so that advantage of lifting the elevator close to the center of gravity is obtained. The cable support had to be raised about 14 inches off the floor in order to facilitate installing the elevator onto the rails without having to remove the cable system.

The HIACA Control System was fabricated and installed in a control panel that was mounted near the GIF cell window. All conduit and wires between the main GIF control and the HIACA control and terminal panels were run. Only part of the wiring was connected since the GIF interlocks must be altered before the entire system is ready for operation. The winch control power terminal/relay box was relocated and wired for the new winch.

GIF cell modification this quarter consisted of designing shield plugs. A combination radiation and insulating (steam) shield plug was designed to provide a 2-1/2-inch pipe steam access to the GIF cell. This plug will be located in the GIF roof. The order for fabrication has been placed. Brass shield plugs have been designed, and the order placed, for use in shielding the elevator lift cable pass-through holes. Before these plugs could be accurately dimensioned, stainless steel tubing was press fit into the holes that had been core drilled for this purpose. Once the lines were in place, they were sized to .002 inches to assure a tight fitting plug and reduce any radiation streaming problem.

Completion of the installation of the test chamber support and positioning apparatus this quarter finished the last of the in-cell modifications. Load testing of this equipment will be completed before the test chamber becomes operational inside the cell. The entire steam system has been tested outside the cell including the superheater capability. This testing was for a special NRC test and not related to the upgraded facility; as a result, a great deal of knowledge and experience about the capabilities of the new system was acquired.

It was previously reported that, based on limited experimental data, planar silicon photo-diodes offered the most promise as active radiation monitors for the HIACA facility. At that time, the main reservation about this type of monitor detector was whether the additional units ordered would exhibit uniform, device-to-device, sensitivity to radiation; i.e., output per unit absorbed dose rate.

Receipt of the additional units allowed for 1) device sensitivity determinations and 2) checkout of device compatibility with existing automated data acquisition equipment.

Device-to-device response to a given radiation dose rate was disappointing. The units could be assigned to two distinct sensitivity groups. However, calibration experiments showed that the two group sensitivity curves, dose rate versus output, track over all dose rates considered. Several options for dealing with the problem of nonuniform detector response are available. It is possible to 1) degrade the more sensitive devices to a lower output dose-rate state, 2) establish a calibration to be used with each group, or 3) explore the possibility of obtaining more uniform devices.

Our course will be to proceed with option 3 and, if necessary, incorporate (to some degree) option 2. Discussions with the device vendor indicate that the device uniformity desired can probably be met by minor changes in fabrication techniques. Partly on the basis of this, an order for additional diodes has been placed. Manufacture of the new diodes, based in part on the performance of our present device, will be from a single melt and with a closer tolerance applied to device thickness.

Receipt of the new diodes is expected about four weeks after placement of the order.

Documentation -- The paper[2.5] reproduced as Appendix A was given at the 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects; it documents a part of the overall Task 1 effort.

2.2 References

2.1 "Memorandum and Order, In the Matter of Petition for Emergency and Remedial Action," USNRC Commissioners (13 April 1978).

2.2 L. L. Bonzon, R. L. Clough, K. T. Gillen, and E. A. Salazar, Qualification Testing Evaluation Program, Quarterly Report, April-June 1979, SAND80-0276 (Albuquerque: Sandia Laboratories, March 1980).

2.3 "Qualification Test for Electrical Connectors Used at Browns Ferry Nuclear Power Plant Unit 3," Report #43854-2 (Huntsville, Alabama: Wyle Laboratories, March 28, 1978).

2.4 Duke Power Company, McGuire Nuclear Station, "Low Voltage and Instrumentation Electronic Penetration Qualification Test Report," Report No. ER252, Specification No. MCS-1361.00-00-0003 through Revision 10, dated March 8, 1978.

2.5 A. W. Snyder, L. L. Bonzon, and B. L. Gregory, Potential Environments and Effects in Reactor Accidents, Invited Paper, 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, CA, July 17-20, 1979.

CHAPTER 3. RADIATION-QUALIFICATION SOURCE EVALUATION

Various technical activities under Task 2 are generally discussed in Section 3.1; program activities were discussed in Section 1.4. Significant efforts were the continuation of the program to complete the "best-estimate" LOCA-radiation signature report by IRT Corporation and the drafting of a "simulator adequacy" assessment report.

3.1 Task 2 -- Technical Activities Summary

Simulator Adequacy Report -- With the completion of certain supporting reports,[3.1, 3.2] significant effort was devoted to documenting the available data to serve as basis for judging the adequacy of laboratory radiation sources in producing equivalent damage that would result from the expectable LOCA radiation sources. A draft-report[3.3] was completed on August 27 and forwarded to R. Feit on August 29. First-level internal Sandia review was completed this quarter.

An initial conclusion is summarized below, but we would caution against significant or universal extrapolation without consideration of the full and final report:[3.3]

> "It can be concluded that the standard gamma-radiation simulators can adequately duplicate the damage mechanism and damage in safety-related equipment which result from postulated nuclear plant ambient and accident radiation environments. The conclusion can be no stronger than that because the simulators must be intelligently used in an overall qualification program which implies combinations of environments, magnitudes, and other considerations. Specialized simulators, which more closely achieve the LOCA-radiation signature, are equally adequate, with similar provisos. However, there seems to be no reason to select one simulator over another. One recommendation is to overstress the equipment/material everywhere to greater total dose than expected from the combined LOCA-radiation signature. Simulator 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.
>
> "In summary, we have seen no evidence of unique damage mechanisms in exposed organic materials which demand unique radiation simulation techniques. But neither can radiation be arbitrarily applied to the test item, without consideration of the complete qualification program."

Publications -- Final distribution was made of IRT Report 0056-005[3.4] in July. This report and its Phase 1 complement completed earlier,[3.5] documents an effort to develop a "best-estimate" LOCA-radiation signature based on realistic time releases and release modes/magnitudes from the LOCA event. Preliminary results have been reported; see, for example, References 3.6 and 3.7.

Documentation -- The paper[3.8] reproduced as Appendix A was given at the 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects; it documents a part of the overall Task 2 effort.

3.2 References

3.1 N. A. Lurie, Calculations to Support Evaluation of Test Sources for Radiation Qualification of Class 1E Equipment, Final Report, Phases 5 and 6, IRT 8167-010(B), April 1979. (Report for Sandia use only; to be included in forthcoming SAND report.)

3.2 J. F. Colwell, B. C. Passenheim, and N. A. Lurie, Evaluation of Radiation Damage Mechanisms in a Reactor Power Cable in a Loss-of-Coolant Accident, IRT 0056-002A, August 1978. (Report for Sandia use only; to be included in forthcoming SAND report.)

3.3 L. L. Bonzon and W. H. Buckalew, Evaluation of Simulator Adequacy for the Radiation Qualification of Safety-Related Equipment, SAND79-1787, NUREG/CR-1184 (Albuquerque: Sandia Laboratories, to be published.)

3.4 N. A. Lurie, Best-Estimate LOCA Radiation Signature, Final Report, Phases 2 through 4, IRT 0056-005, July 1979. (Report for Sandia use only; to be included in forthcoming SAND report.)

3.5 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, and intended for internal Sandia use only; this report will be published in a full Sandia topical report.)

3.6 N. A. Lurie, J. A. Naber, and L. L. Bonzon, Adequacy of Radiation Sources for Qualification of Class 1E Reactor Components, SAND78-0161A/IRT 8167-003, Trans Am Nucl Soc, Vol 28 (1978).

3.7 L. L. Bonzon, K. T. Gillen, and E. A. Salazar, Qualification Testing Evaluation Program, Quarterly Report, October-December 1978, SAND79-0761, NUREG/CR-0813 (Albuquerque: Sandia Laboratories, June 1979.)

3.8 A. W. Snyder, L. L. Bonzon, and B. L. Gregory, Potential Environments and Effects in Reactor Accidents, Invited Paper, 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, CA, July 17-20, 1979.

CHAPTER 4. ACCELERATED-AGING STUDY

The activities under Task 3 were numerous and diverse: programmatic activities were discussed in Section 1.4; Section 4.1 highlights the various technical activities; and Section 4.2 details a specific program aspect that engendered particular effort and achieved a significant milestone during the reported quarter.

4.1 Task 3 -- Technical Activities Summary

NRL Subcontract -- Since April 1977, the Naval Research Laboratories (NRL) has been subcontracted to provide radiation services related to the accelerated-aging study. A number of modern cable insulation and jacketing materials have been aged in various combinations of radiation, temperature, and humidity. This subcontract has been extended through FY79 to continue to provide a test facility which will complement the experiments being done at Sandia's low intensity cobalt array (LICA) facility.

On June 5, NRL submitted a proposal on "Determining the Susceptibility of Certain Polymeric Materials to the Effects of Oxygen at Low Dose-Rates." After review by Sandia and NRC staff, it was concluded in August that the proposal did not fit the charter of the accelerated aging program.

Cable Material Aging Experiments -- To support the accelerated-aging method analyses, the general single- and combined-environments aging experiments, on a variety of modern cable insulation and jacketing materials, continued this quarter. These experiments were performed at the Sandia LICA facility and, under subcontract, at the Naval Research Laboratories.

The thermal-aging facility is currently being used to age various materials at temperatures between 70° and 200°C. Radiation aging at room and elevated temperatures is also being carried out at the LICA facility.

Fire-Retardant Aging Program -- The investigation of the effects of radiation and thermal aging on fire-retardant additives in electric cable insulation was continued. Formulations of Hypalon and ethylene-propylene rubber (EPR) containing the antimony oxide-halocarbon retardant combination are being examined; a preliminary summary was given in Reference 4.1.

More extensive analyses of oven-aged, radiation-aged, and simultaneous thermal-radiation-aged samples have now been completed. These results indicate that in addition to substantial losses of the halogenated hydrocarbon additive found before, significant decreases in antimony can also take place in aging; the thermal environment seems to play a major role. The changes in antimony content have been documented both by chemical elemental analysis and by the neutron activation technique. A large number of samples, aged in various environments, have been submitted to Factory Mutual Corporation, where combustion tests will be carried out to measure and compare flammability characteristics. Further aging experiments are also underway.

On September 17, R. Clough met with A. Tewarson of Factory Mutual for discussions of the test methods to be used in this study.

Extension of Accelerated-Aging Investigation to Gaskets/Seals -- As a preliminary effort to expand the accelerated-aging task, discussions were held among Sandia staff on the extension of the work to include gaskets/seals. Based on (a) the United Engineers & Constructors study[4.2] which documents the extensive use of gaskets/seals in safety-related equipment, (b) documented failure history of seal systems,[4.3] and (c) the extensive applicable work by Sandia staff,[4.4-4.7] a test plan was developed and submitted to NRC staff by letter dated August 21.[4.8]

Extension to Other Aging Monitoring Techniques -- Discussions were held this quarter with R. Fei and Sandia staff (R. Clough and W. Johnson) on Mr. Johnson's proposal "A Laser-Based Measurement Technique for Applications to Aging Studies on Hydrocarbon Insulators Found Near Nuclear Reactors." The thrust of this program is the development of measurement techniques sufficiently sensitive to detect the changes from aging under actual in-use conditions. Such techniques would allow a direct correlation of accelerated and real-time experiments, and a comparison of various in-use environments. We will continue to monitor this on-going Sandia activity for applicability to the QTE program objectives; if appropriate, a request for NRC funding may be submitted in the circa-FY81 budget proposal.

Life Expectancy and Age Simulation Report -- Significant effort was devoted to documenting Sandia experience on life expectancy and age simulation. A complete-draft report[4.9] was completed in August with first-level internal Sandia review completed in September 1979. The abstract from the report summarizes its scope and content:

"This document outlines some of the types of experiments which can be used to improve reliability, simulate age, and predict life expectancy of complex equipment. Brief discussion is given of failure mode tests and compatibility tests, which often give useful qualitative aging information. A detailed discussion is presented on accelerated aging methods, emphasizing an approach based on kinetic rate expressions. This kinetic approach offers a convenient framework for describing the importance of competing reaction pathways, transitions in a material, diffusion effects, and sorption effects. It is concluded that, when properly conceived and carried out, accelerated aging studies of materials and simple components offer the best opportunity for making quantitative age simulations and lifetime predictions of equipment."

LICA Facility Upgrade -- Based on increasing need for higher dose-rate capabilities at LICA, a request to add cobalt to the existing facility was made to the Sandia Health Physics organization on July 10.

Approval to load up to 22-kilocuries Cobalt-60 equivalent was received on July 20.

A request-for-quote (RFQ) was prepared in July and sent to Atomic Energy of Canada Limited (AECL) in late August. Their quote was received on September 10 and accepted on September 18. Delivery of 12-kilocuries Cobalt-60 was scheduled for late 1979.

With the delivery of the cobalt, the maximum dose rate is expected to be (in excess of) 1 MR/hr, so that accelerated aging studies can span the range, 2 kR/hr to 1 MR/hr.

NRR Consulting Effort -- Based on the accelerated aging program effort, Sandia staff was asked to assist E. Butcher (NRR) in developing interim aging guidance for use in evaluating (prior-documented) equipment qualification programs. On September 18, K. Gillen visited with NRC staff in Bethesda, Maryland, to assist in this effort. Documents of tabulated data, prepared by NRR subcontractors, were discussed and evaluated; in some cases the data were found to be compared to inconsistent bases or differed from consensus data.

This work was further discussed with R. Feit and G. Bennett during a program review at Sandia on September 25.

Documentation -- The paper[4.10] reproduced as Appendix A was given at the 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects; it documents a part of the overall Task 3 effort.

4.2 Implications of Sorption Effects on Accelerated Aging Studies[4.11*]

Whenever a material degrades due to reactions with a surrounding gaseous medium, the sorption characteristics of the material can have a profound and largely unrecognized influence on any attempts at quantitatively accelerating the degradation. For oxidative degradation, the analysis presented in this section indicates that the activation energy obtained from an Arrhenius treatment is a linear combination of the true activation energy and the heat of sorption of oxygen. If material degradation is due to a reaction with a condensable such as water vapor, then a non-Arrhenius temperature dependence is expected for experiments run at constant gaseous concentration. If, however, the experiments are run and analyzed at constant relative pressure (relative humidity for water vapor), then Arrhenius behavior is often recovered.

Suppose a reaction occurs between a polymer and its surrounding gaseous environment (e.g., reaction with oxygen or water vapor) which adversely affects the polymer, leading to its degradation. In experiments designed to accelerate the degradation using elevated temperatures, the

*This section prepared by K. T. Gillen (Division 5813).

deterioration of the polymer is normally followed by monitoring the time-dependence of a macroscopic damage parameter, D. D is related to the degrading reactions because it is some function of the concentrations of the species in the polymer which are changing due to these reactions. Assume for the sake of argument that D is determined solely by the concentration of a species labelled A and that a single reaction dominates the degradation over the temperature range of interest. As examples, the species A could refer to a chemical group being destroyed or changed or to the chemical crosslink density. With C_A representing the concentration of the A species, D is assumed to be a smooth monotonically decreasing function of C_A,

$$D = f(C_A) \tag{1}$$

If after the ambient or use lifetime, a certain value of C_A (and hence a certain value of D) exists, then successful accelerated aging can be achieved by duplicating the final value of C_A (and hence D) in a shorter period of time by using a higher temperature.

To describe the changes in C_A with time under various conditions, it is convenient to use a standard kinetic treatment. In particular, if C_A changes due to a reaction with the surrounding gaseous medium, a typical rate expression is

$$\frac{dC_A}{dt} = k\ C_A^a\ C_B^b \tag{2}$$

where C_B is the concentration of the gaseous species sorbed in the polymer, k is the reaction rate constant which depends on temperature, and a and b are exponents which describe the order of the reaction with respect to C_A and C_B. We will further restrict our attention to the case where, at a given constant temperature, C_B is always at sorption quilibrium and therefore remains constant with time. This corresponds to the case where B is replenished by diffusion into the polymer from the atmosphere much faster than it is used up during the reaction. In practice, this condition is achieved by using sufficiently thin samples relative to the length of the aging exposures. When no time dependence in the C_B term exists, Eq. (2) can be rearranged and integrated from t = 0 (C_A°) to time t

$$\int_{C_A^o}^{C_A} \frac{dC_A}{C_A^a} = k\ C_B^b\ t \tag{3}$$

At a given temperature, let t_D be the time required for the damage parameter D to reach a particular value (e.g., a specified percentage of its original value). From Eq. (1) that value of D will correspond to a particular value of C_A, which implies a constant value for the integral on the left hand side of Eq. (3), independent of the temperature chosen. Therefore, for experiments run at different temperatures, the values of t_D will satisfy the relationship

$$k\, C_B^b\, t_D = \text{constant} \tag{4}$$

Thus the temperature dependence of the time to a specified value of the degradation parameter will result from the temperature dependence of the $k\, C_B^b$ term. The kinetic rate constant k is often found to have an Arrhenius dependence on temperature, i.e.,

$$k \propto \exp(-E_a/RT) \tag{5}$$

where R is the gas constant and E_a is the activation energy for the reaction. In addition, the term C_B^b can be temperature dependent due to the fact that equilibrium sorption of a gas in a polymer generally depends on temperature.

In order to discuss some of the subtle and interesting complications caused by this temperature dependent equilibrium sorption, first note that equilibrium sorption behavior is usually described in terms of sorption isotherms.[4.12] These give the relationship at constant temperature between the equilibrium concentration of penetrant in a material, C_B, and the partial vapor pressure, P_B, of gas surrounding the material and are written

$$C_B = \sigma(P_B)P_B \tag{6}$$

where σ is the solubility coefficient, which in general depends on P_B. Because the differences in sorption behavior usually found for permanent gases and condensable gases lead to quite different implications for accelerated aging studies, these two cases will be discussed separately.

Permanent gases -- For permanent gases such as oxygen (O_2) dissolved in polymers at pressures below a couple of atmospheres, solubilities are generally low and σ is independent of pressure.[4.12]

$$C_B = \sigma P_B \tag{7}$$

This linear relationship is referred to as Henry's Law behavior. The temperature dependence of the solubility coefficient is given by an Arrhenius-like expression.

$$\sigma = \sigma_o \exp(-\Delta H_s/RT) \tag{8}$$

where ΔH_s is the heat of solution. The absolute value of ΔH_s is typically small (less than a few kcal/mole) so that the equilibrium solubility either increases or decreases slightly with increasing temperature. When thermal oxidative degradation occurs ($B \equiv O_2$), accelerated aging usually involves air oven aging. In this case, P_{O_2} (but not C_{O_2}) will be independent of temperature and equal to approximately 0.2 atmospheres. From Eqs. (4), (5), (7) and (8), the temperature dependence of t_D will be given by

$$t_D \propto \exp\left(\frac{E_a + b\Delta H_s}{RT}\right) \tag{9}$$

In other words, the time to a specified level of degradation will have an Arrhenius dependence on temperature, but with an effective activation energy given by

$$E_{eff} = E_a + b\,\Delta H_s \tag{10}$$

Therefore, due to the Arrhenius dependence of sorption for permanent gases, conventional Arrhenius accelerated aging extrapolations are still viable even though temperature dependent sorption effects are present. It should be emphasized, however, that the measured activation energy does not represent the true activation energy for the thermoxidative process, as shown in Eq. (10).

Condensable gases -- Sorption isotherms for condensable gases are typically more complex than for permanent gases.[4.12] For condensable gases, the isothermal solubility coefficient, σ, is often dependent on the partial pressure, P, so that C is not linear in P. In addition, the solubility at constant P decreases rapidly as the temperature is increased.[4.12] An example of typical water sorption behavior is shown in Figure 4.1; such behavior greatly complicates accelerated aging extrapolations. If, for example, aging was carried out at the four temperatures shown using an atmosphere of constant water vapor concentration, P_{H_2O}, then the equilibrium water concentrations sorbed in the polymer would be given by C_1, C_2, C_3, and C_4, as shown in the Figure. The temperature dependence of the sorbed water concentration would, in general, be non-Arrhenius leading to non-Arrhenius results for the aging experiments. An example of this situation is shown in Figure 4.2 using literature data on the reversion of a polyurethane potting compound in humidity environments.[4.13] The study followed changes in durometer hardness (the macroscopic damage parameter, D) of the material at various combinations of temperature and humidity. The reversion time (t_D in our notation) was arbitrarily defined as the time required for the hardness to drop to 20. In Figure 4.2 the log of t_D is plotted versus inverse temperature for a constant water vapor concentration, P_{const}, of 4×10^{-3} moles/liter. The non-Arrhenius character of the data makes extrapolation difficult.

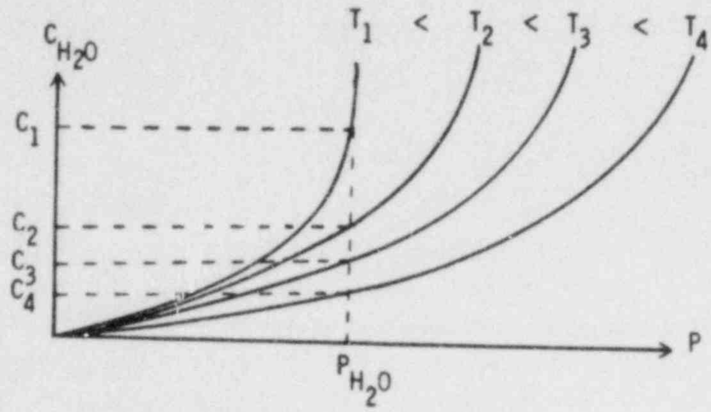

Figure 4.1. Typical Isotherms for Water Vapor and Other Condensable Fluids

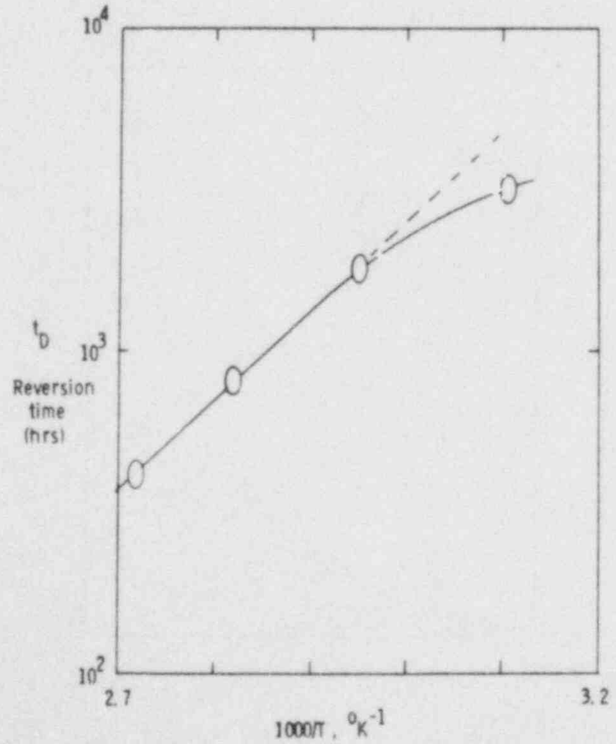

Figure 4.2. Arrhenius Plot of the Reversion Time for a Polyurethane Potting Compound Plotted Versus Inverse Temperature for a Constant Water Vapor Concentration of 4×10^{-3} Moles/Liter. The data comes from Reference 4.13.

Analogous to the situation discussed earlier for permanent gases, it would be possible to make predictions for materials reacting with condensable gases if the accelerated experiments could be carried out under conditions in which the temperature dependence of the concentration of the sorbed species was Arrhenius. Much data exist which indicate that at a given constant relative humidity (RH), the solubility coefficient and hence C_{H_2O} has a small dependence on temperature and is approximately Arrhenius, i.e.,

$$\sigma(RH) = \sigma_o(RH) \exp(-E_c/RT) \tag{11}$$

E_c, the activation energy for the solubility coefficient at constant RH, is often found to depend only on the system studied, not on the particular value of the relative humidity. As evidence for this phenomenon, some representative literature sorption data are shown in Figures 4.3[4.14] and 4.4,[4.15] where equilibrium water sorption is plotted versus RH at a number of temperatures. The data of Figure 4.3 indicate that E_c is equal to zero for all values of RH; analysis of the data of Figure 4.4 gives an E_c of +5 kcal/mole, also independent of RH. For water sorption in various materials, E_c is usually found in the range of 0 to +5 kcal/mole.

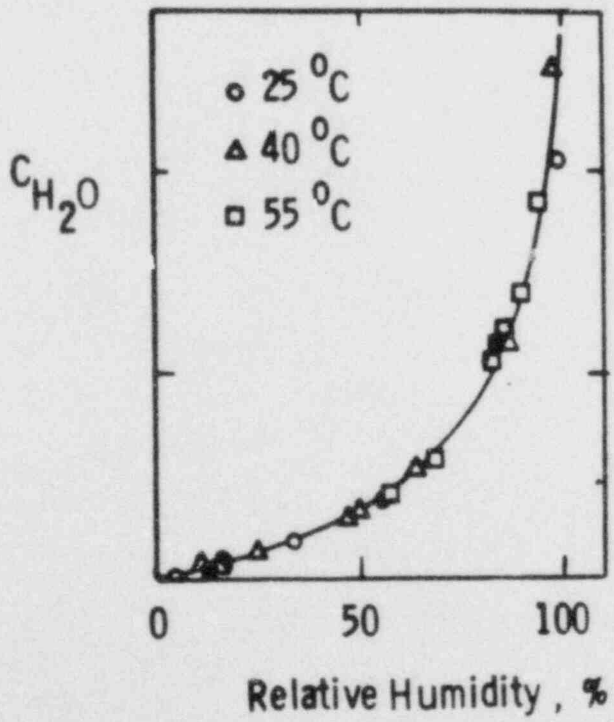

Figure 4.3. Sorption Isotherms for Water Vapor in a Polyurethane Membrane, Plotted Versus Relative Humidity. Data from Reference 4.14.

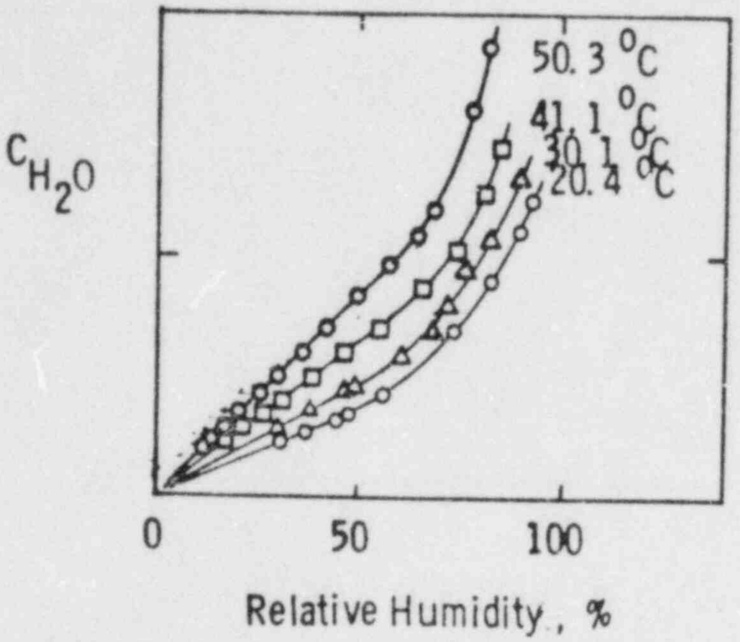

Figure 4.4. Sorption Isotherms for Water Vapor in a Silicone Material, Plotted Versus Relative Humidity. Data from Reference 4.15.

The observation that C_{H_2O} approximates Arrhenius behavior at constant RH makes the analysis of condensable gas data at constant RH completely analogous to permanent gas data analysis at constant P. The same reasoning that led to Eq. (10) indicates that degradation data analyzed under constant RH conditions should be Arrhenius with the experimentally measured effective activation energy given by

$$E_{eff} = E_a + b\ E_c \tag{12}$$

An instructive example of the above comes from a reanalysis at constant RH of the data from Reference 4.13. This data gave non-Arrhenius behavior when analyzed at constant water vapor concentration, as shown in Figure 4.2. The reanalysis is shown in Figure 4.5, where the log of the reversion times is plotted versus inverse temperature at four different relative humidities (25, 50, 75, and 95 percent). This analysis gives the expected Arrhenius behavior with E_{eff} equal to 21 kcal/mole. Extrapolation of the data to other conditions is now possible. It is clear from Eq. (12) that in order to obtain the true E_a for hydrolysis, detailed kinetic experiments as well as extensive sorption studies must be carried out. To make aging predictions, however, these studies are often unnecessary.

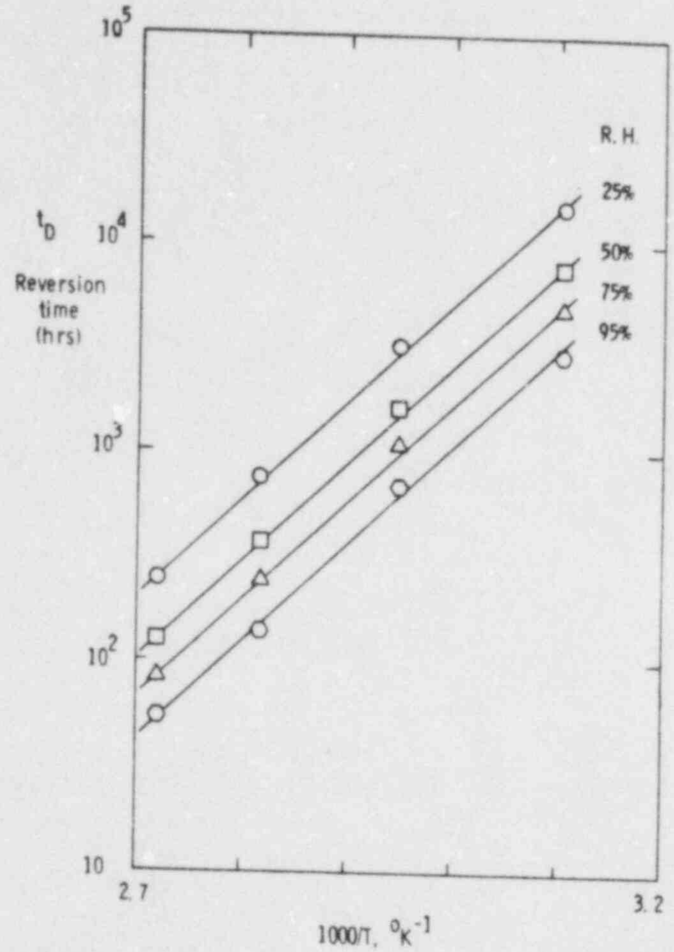

Figure 4.5. Arrhenius Plot of the Data From Reference 4.13 Analyzed at Various Constant Relative Humidities.

Although the preceding discussion dealt primarily with humidity effects, similar sorption behavior is found for other condensable fluids. Therefore, when a condensable vapor is important to degradation, accelerated aging studies should be done at constant relative pressure P/P_o of the condensable. Arrhenius behavior at constant relative pressure will likely result.

In conclusion, it is important to be aware of the profound influence of temperature dependent sorption when attempting either to extrapolate accelerated aging data or to interpret any resulting Arrhenius activation energy.

4.3 References

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4.2 United Engineers & Constructors submittals under J.O. 6602.002, "Sandia Laboratories Class 1E Equipment Data Package."

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4.5 J. G. Curro and E. A. Salazar, "Physical and Chemical Relaxation of Elastomers," J Appl Polymer Sci, Vol 19, 1975.

4.6 J. G. Curro, "A Technique for Determination of the Chain Length Distribution From Stress Relaxation Measurements in Certain Elastomers," J Appl Polymer Sci, Vol 14, 1976.

4.7 E. A. Salazar, J. C. Curro, and K. T. Gillen, "Physical and Chemical Stress Relaxation of a Fluoroelastomer," J Appl Polymer Sci, Vol 14, 1976.

4.8 Letter dated August 21, 1979, L. Bonzon (SNL) to R. Feit (NRC), "Extension of the Accelerated Aging Task Effort."

4.9 K. T. Gillen and K. E. Mead, Predicting Life Expectancy and Simulating Age of Complex Equipment Using Accelerated Aging Techniques, SAND79-1561 (Albuquerque: Sandia Laboratories, January 1980).

4.10 A. W. Snyder, L. L. Bonzon, and B. L. Gregory, Potential Environments and Effects in Reactor Accidents, Invited Paper, 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, CA, July 17-20, 1979.

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APPENDIX A

POTENTIAL ENVIRONMENTS AND EFFECTS IN REACTOR ACCIDENTS

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ABSTRACT

The radiological safety of contemporary commercial nuclear power reactors depends principally upon multiple-barrier containment of fission products. The reliability of the containment system depends, in part, upon the tolerance of active and passive hardware to potentially detrimental effects from the composite environmental stress from cumulative normal operation and from the environments produced by accidents. Recent studies estimate that contemporary plants are constructed from an assortment of about 140 different safety and safety-related components ranging in electrical complexity from organic dielectric insulation to small-scale integrated solid-state circuits of signal processors and transmitters. Qualifying these safety components to perform reliably during and subsequent to hypothesized accidents requires exposure, or simulated exposure, to severe compounded environments of temperature, humidity, radiation, and chemical sprays. Additionally, accelerated aging to condition components to the equivalent of a forty-year in-service life before exposure may be required. Within the past few years, much greater attention has been given to improving estimates of the radiation field environments which would accompany accidents, to understanding the role of aging and accelerated aging in simulated operational environments, to determining the relative significance of synergistic effects in compounded environments, and to evaluating the validity of currently specified methods for qualifying safety equipment. The status and a critique of these efforts is given.

Note: This invited paper was presented at the 1979 IEEE Annual Conference on Nuclear and Space Radiation Effects, Santa Cruz, California, July 17-20, 1979. This paper summarizes part of the Qualification Testing Evaluation (QTE) Program (A1051-0) being conducted by Sandia National Laboratories for the United States Nuclear Regulatory Commission.

INTRODUCTION

The radiological safety of contemporary commercial nuclear power reactors depends principally upon reliable containment of the radioisotopes produced by fission of the uranium fuel. The containment function is performed by passive plant hardware and structures and by active electrical and mechanical components. The containment system performs functions both during normal operation and during and after accidents which breach one or more of the containment barriers. The reliability of the containment system depends, in part, upon the tolerance of active and passive hardware to effects from environmental stress from cumulative normal operation and from the environments produced by accidents. The objective of this paper is to identify the safety and safety-related equipment of commercial nuclear power plants and to characterize the environments and the effects in potential reactor accidents.

RADIOLOGICAL CONTAINMENT AND CONTROL/ENGINEERED SAFETY FEATURES

There are three barriers in a contemporary U.S. commercial pressurized water reactor* (PWR) to contain the fission product sources of radioactivity. The first barrier is the tubular zirconium-alloy clad enclosing typically 40,000, 0.5-inch diameter, 12-foot long "pins" of uranium oxide fuel pellets. The second barrier is the reactor vessel, piping, valves, etc., which constitute the primary coolant pressure boundary. The third barrier is the containment building. The fuel clad, the first barrier, does not have active penetrations. The primary coolant pressure boundary and the containment building barriers have numerous valved penetrations. Many valved pipe penetrations of the primary pressure boundary are followed in-line by valved penetrations through the containment building barrier. The valved penetrations of both the primary pressure boundary and the containment building are required to permit numerous systems to perform normal and/or emergency cooling functions. Additional penetrations in the containment building are required for instrumentation and control, resulting, for example, in passive cable penetrations through the containment barriers. Figures A.1 through A.3

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*Generally, U.S. pressurized water reactors have more equipment potentially exposed to severe environments from accidents than do boiling water reactors due to equipment placement and containment system configuration.

list typical systems which penetrate the containment barriers. In Figures A.1 through A.3 the systems are shown in two categories, viz, those systems which are explicitly for safety and those systems which are required for normal operation but include, for example, valved penetrations which are safety related. Those systems which are explicitly for safety are commonly referred to as the engineered safety features to distinguish them from other less tangible features of safety achieved inherently through design, redundancy, quality assurance, training, and operational procedures. The overall concept for safety involving multiple barriers and systems to reduce stress to the barriers and to mitigate the consequences of accidents is commonly referred to as defense-in-depth, with the engineered safety systems providing explicit backup defense. The control of the systems noted in Figures A.1-3 requires electrical equipment. The electrical equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment or reactor heat removal, or otherwise are essential in preventing significant releases of radioactive material to the environment are defined as Class 1E.[A.1] While Class 1E was originally meant to designate electrical equipment and electrical systems, more recently the explicit identification and separation of safety and safety-related electrical equipment from other nonelectrical equipment also having safety functions has diminished and the safety equipment classification will likely be redefined.

PRIMARY REACTOR COOLING:

- HIGH PRESSURE EMERGENCY WATER INJECTION AND WATER LEVEL AND CHEMISTRY
- MEDIUM PRESSURE EMERGENCY WATER INJECTION
- LOW PRESSURE EMERGENCY WATER INJECTION AND RESIDUAL HEAT REMOVAL
- REACTOR-QUALITY WATER SUPPLY
- REACTOR COOLING SYSTEM
- SYSTEM COMPONENT-COOLING SUPPLY
- NITROGEN-GAS PRESSURIZATION SUPPLY
- EQUIPMENT VENTING AND GAS COLLECTION

Figure A.1. In-Containment, Safety (and Related) Systems; Model PWR

CONTAINMENT BUILDING ENVIRONMENTAL CONDITIONING:

- AIR RECIRCULATION/MIXING AND COOLING
- SPRAY, PRESSURE REDUCTION AND IODINE REMOVAL
- GAS PURGE (PRIOR TO PERSONNEL ENTRY)
- AIR DECONTAMINATION (DURING POWER OPERATION)
- EQUIPMENT AND FLOOR DRAINS

Figure .2. In-Containment, Safety (and Related) Systems; Model PWR

STEAM/TURBINE:

- FEED WATER, AUXILIARY AND MAIN
- MAIN STEAM SUPPLY
- WATER QUALITY AND LEVEL CONTROL

CONTROL AND MONITORING:

- RADIATION MONITORING
- NUCLEAR INSTRUMENTATION
- EMERGENCY ELECTRICAL DISTRIBUTION

Figure A.3. In-Containment, Safety (and Related) Systems; Model PWR

Figures A.4 and A.5 identify generic components which are elements within the safety, and safety-related, systems. For each generic component there are numerous suppliers, and each supplier may have several models. A recent study[A.2] estimated that nuclear power plants now in advanced states of construction and scheduled for initial operation before 1983 are constructed from an assortment of about 140 different safety, and safety-related, items. The electrical and electronic complexity of the safety-related equipment and components in Figure A.4 extends from dielectrics to small-scale integrated solid-state circuits of signal processors and transmitters.

ELECTRICAL:

- TRANSMITTERS/AMPLIFIERS
- ACTUATORS
- THERMOCOUPLES AND RESISTIVE TEMPERATURE DETECTORS
- RADIATION DETECTORS
- LIMIT SWITCHES
- PRESSURE SWITCHES
- LEVEL SWITCHES
- ROTOMETERS
- MOTORS
- CONNECTORS AND PENETRATIONS
- WIRE AND CABLE

Figure A.4. Generic Safety (and Related) Equipment/Components

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NON-ELECTRICAL:

- ACTUATORS, PNEUMATIC
- VALVES
- ENCLOSURES

Figure A.5. Generic Safety (and Related) Equipment/Components

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ENVIRONMENTAL QUALIFICATION OF SAFETY EQUIPMENT

The components and subsystems of engineered safety, and safety-related, systems are currently qualified in environments approximating (or simulating) the environments estimated to be produced by a hypothesized loss-of-coolant accident (LOCA), followed by failure of the emergency core cooling systems (ECCS). This hypothesized event would result in breaching of the zirconium-alloy clad barrier and the escape of fission products to, but confined within, the free space of the containment building. (A breach of the primary coolant pressure boundary is a necessary condition for the loss-of-coolant accident.) The combined environment in the containment building would consist (at different times after the accident) of highly localized superheated steam for a very brief period, a more extended period of saturated steam, a very lengthy period of high humidity, plus chemical spray, radiation, and pressure and temperature; the latter produced predominantly by the initial release of steam. The post-accident containment environment includes chemical spray intentionally released to precipitate radioactive iodine. Very severe equipment environments would result from the hypothesized LOCA, failure of the ECCS, and failure of most of the other engineered safety systems, while retaining containment building integrity. This condition would have the greatest impact on the post-accident availability of monitoring equipment to support post-accident recovery. Numerous additional accidents can be hypothesized which would produce less severe environments but would require different post-accident recovery operations and different dependencies on engineering safety systems. For example, in the Three Mile Island (TMI) accident, the post-accident containment building environment was dominated by radiation, but the primary pressure boundary and the engineered safety systems remained largely intact. The containment building is contaminated and the components of the intact engineered safety systems will be subjected to further degradation until the recovery operation is completed. Additionally, the TMI accident revealed the potential threat to engineered safety systems from combustion of hydrogen and from the possible detonation of hydrogen confined in the containment building. Even prior to the TMI accident, the uncertainties in the definition of the extreme accident environments were acknowledged and the validity of the tests to qualify safety equipment were being examined. The experience of TMI accident has at least further enlarged the scope of the required environmental definitions; has identified the need to be more specific about the time-dependent roles of safety-related equipment following a wide range of potential accident conditions; and has identified the need to re-examine the validity of simulation of time-intensity profiles and test procedures.

The environmental qualification of safety equipment requires assurance of the performance of the equipment in the most severe, but plausible, compound environments of accidents which might occur anytime during the projected forty-year life of a plant. The principal specific technical details which must be resolved for each engineered safety, or safety-related, system are:

- the parametric definition (e.g., intensity versus time) of the specific environments for the range of plausible accident conditions

- the definition of one or more test conditions to encompass the range of plausible accident environments

- the definition of the sequence of exposures to the several test environments, or to combined environments if synergistic effects are determined to be significant

- the accelerated aging, if important, of the materials and components prior to exposure of the accident environment

- the scale of integration of the tested materials and components into operable subsystems, or a full system, to assure reliable performance of the operating system in the real accident environment.

THE THERMODYNAMIC ENVIRONMENT

Considerable effort, both analytical and experimental, has been devoted to understanding the thermal-hydraulic behavior of PWRs under a large number of assumed primary pressure boundary and main steamline breaks in containment. These efforts have also provided the basis for deriving very good pressure-time and temperature-time data on the containment environment. The pressure-time data have in particular, been used to set containment design-pressure specifications. Design pressure is typically 60 psig, but a specific design will depend, for example, upon the primary cooling system volume, the containment building free volume, and other plant-specific features such as available pressure suppression systems. Typically, containment pressures from accidents are predicted to be in the range 30-50 psig and persist for several hours to a day following a LOCA. Typically, containment temperatures are predicted to be in the range 300-380°F, depending upon the initial assumptions, and likewise persist for several hours to a day. Typical test requirements for temperatures are shown in Figure A.6.[A.3]

THE RADIATION ENVIRONMENT

The radiation environment has been defined with considerably less precision than the pressure/temperature environment. In contrast to the readily determinable steam source for the pressure and temperature environment, the determination of the radiation source term for the radiation-field environment is subject to numerous uncertainties. Even assuming total core meltdown, the precision in determining the times and the sources of release, the quantities of radioactive materials released to containment, and the physical state and location of the materials in containment is currently very limited. The first source term shown in Figure A.7.[A.4] is currently assumed for the qualification of equipment. This source term assumes complete failure of the ECCS and failure of most other safety-related equipment, except that the containment barrier is not breached. The severity of this accident likely precludes the need to use any equipment after the accident except the monitoring and the containment heat removal equipment. The second, lesser source term (not officially adopted to date) assumes partial failure of the ECCS, but with the safety-related electrical systems and the containment building environmental conditioning equipment being initially intact, but subjected thereafter to the environment. The first source is assumed, for specification of the radiation field environment, to occur instantaneously with the LOCA. This conservative feature and other features of Source 1 will be commented upon later.

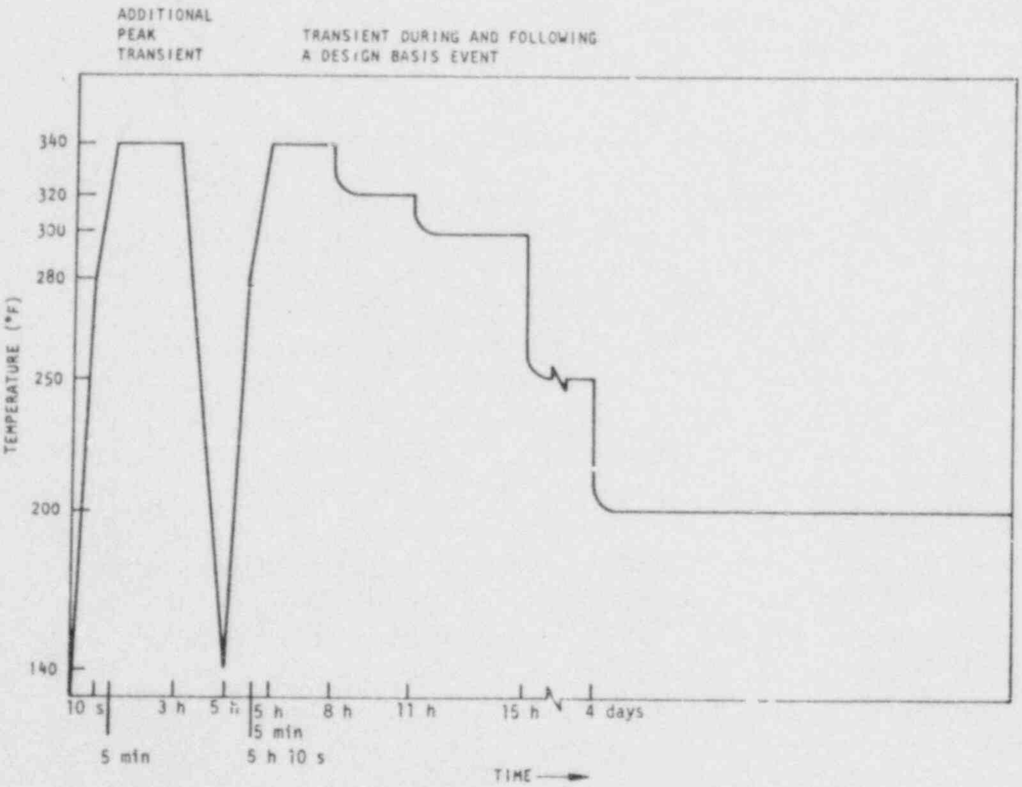


Figure A.6. Test Chamber Temperature Profile for Environment Simulation (Combined PWR/BWR) (from Reference A.3)

SOURCE 1	AIRBORNE	100% NOBLE GASES, 25% IODINES
(CONTAINMENT HEAT REMOVAL SYSTEMS, ETC.)	PLATE-OUT	25% IODINES, 1% SOLIDS
	WATERBORNE	50% HALOGENS, 1% SOLIDS
SOURCE 2	AIRBORNE	10% NOBLE GASES (EXCEPT ^{85}Kr), 30% KRYPTON-85, 5% IODINES
(SAFETY-RELATED ELECTRICAL SYSTEMS)	PLATE-OUT	5% IODINES
	WATERBORNE	10% HALOGENS

Figure A.7. Source Types and Distribution Categories (Reference A.4)

Within the past few years, a marked increase in attention has been given to improving the estimates of the radiation source terms, to understanding the role of aging and accelerated aging of materials, to determining the relative significance of synergistic effects in compound environments, and to evaluating the validity of currently prescribed tests (or simulations) in qualifying Class 1E equipment. The recent works of Lurie, Naber, and Bonzon[A.5,A.6] have dealt with the radiation source definition. Gillen and Salazar[A.7, A.8] have experimentally studied and modelled aging and accelerated aging in organic dielectrics. Bonzon[A.9] has studied synergistic effects and the validity of current LOCA-qualification testing methods. Recent analytical work on the retention of fission products within components of the breached primary coolant system has been done by Gieseke, et al.[A.10] Sandia National Laboratories has just begun to investigate experimentally the transport and deposition of fission products in the primary cooling system under various postulated light water reactor accidents.

A few examples of the work of Lurie et al will indicate the current status of radiation source term estimates. Figure A.8[A.6] shows the approximate time sequence of fission product release to containment for an unterminated LOCA and failure of ECCS. The initial releases are the gases normally confined in the gap between the uranium-oxide fuel and the zirconium-alloy clad. At elevated temperatures of the oxide fuel, in the absence of primary coolant, the structural strength of the zircalloy clad is reduced, the zirconium oxidizes in the presence of steam, and in continued absence of any significant cooling, the zircalloy clad melts. Gap releases can plausibly begin to occur as early as a few seconds following the LOCA in the absence of emergency core cooling. The indicated meltdown releases in Figure A.8 are very conservative. The assumed "100% release time" is the expected time for initial fuel slumping and certainly not complete melting. Increasing temperature of the melt evolves increasing quantities of the lower vapor pressure chemical species (shown as "vaporization release"). On contact of molten fuel with coolant condensate, the phenomena of steam explosions could conceivably disperse finely divided fuel, further oxidize fission products, and release additional products into the containment building atmosphere. Steam explosions are the consequence of contact between a hot liquid and a cold liquid. The rapid specific volume increase from transformation is referred to as an "explosion." Steam explosions require an enhanced transfer of heat from the melt by rapid fragmentaton and an increase of the heat transfer surface area of the hot liquid. Enhanced oxidation also results from the enhanced surface area.

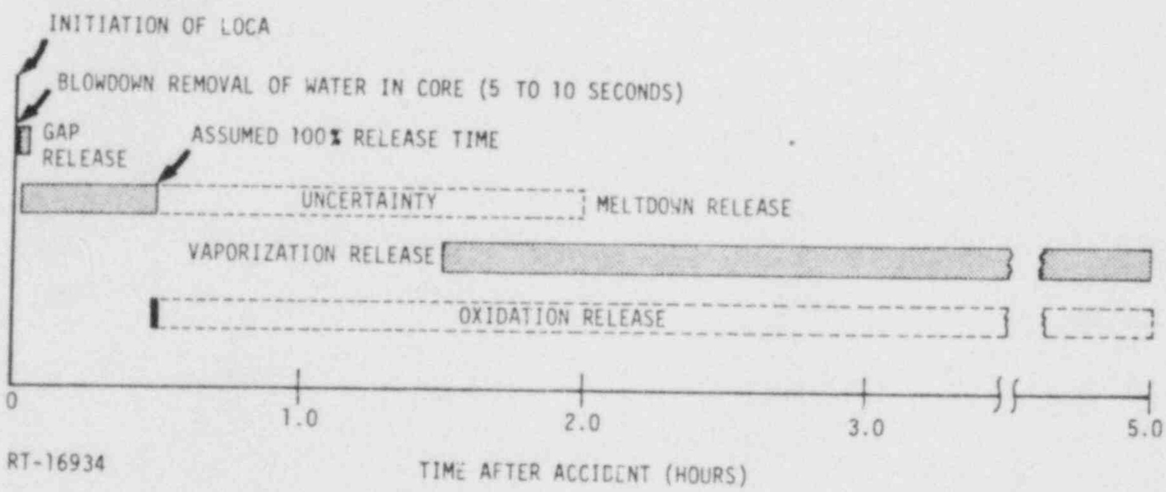


Figure A.8. Time Sequence of Fission Product Release for an Unterminated LOCA Without Emergency Core Cooling (from Reference A.6)

Figure A.9[A.6] shows the fractions of the total fission product inventories existing within the fuel clad at the time of a postulated accident that are released by the events identified in Figure A.8. These fractions were derived principally to support the analyses of accident consequences to public health and safety in the Reactor Safety Study.[A.11] Recent experimental results[A.12, A.13] indicate the gap fractions noted in Figure A.9 are conservatively large. In general, the derivation of these fractions are a composite of numerous, limited experiments; applications of related chemical data; and experienced judgements, tempered with a conservative margin. Thus, the fractions are more likely higher than would be observed in directly applicable experiments. The significance of core meltdown to large release fractions and to the worst initial environmental conditions for safety equipment is obvious. Also, the relative magnitude of the source term in the TMI accident is apparent from the fact that very little, if any, of the core melted and the release was from the fuel-clad gap. The seriousness of TMI to the safety equipment was not the initial radiation level but rather the long-term exposure of equipment that is vital to decontamination and recovery of the physical facility. The highest reported radiation exposure dose level in the TMI containment was about 30 krads/hr from krypton, xenon, iodine, and a small fraction of cesium. The oxidation release is very difficult to estimate because steam explosions are events very dependent, for example, upon geometries of the contact between the liquids, and geometries of the "container." The phenomenon is simply not fully understood.

Fission Product Species	Gap Release	Meltdown Release	Vaporization Release	Oxidation Release
Noble Gases (Xe, Kr)	0.030	0.873	0.097	(X)(Y) 0.90
Halogens (I, Br)	0.017	0.885	0.098	(X)(Y) 0.90
Alkali Metals (Cs, Rb)	0.050	0.760	0.190	---
Tellurium Group (Te, Se, Sb)	0.0001	0.150	0.849	(X)(Y) 0.60
Noble Metals (Ru, Rh, Pd, Mo, Tc)	---	0.030	0.049	(X)(Y) 0.90
Alkaline Earths (Sr, Ba)	0.000001	0.100	0.009	---
Rare Earths (Y, La, Ce, Nd, Pr, Eu, Pm, Sm, Np, Pu)	---	0.003	0.010	---
Refractories (Zr, Nb)	---	0.003	---	---

X = Fraction of core involved in steam explosion

Y = Fraction of inventory remaining for release by oxidation

Figure A.9. Best Estimate LOCA Fission Product Release Fractions (from Reference A.6)

Figure A.10[A.6] shows the gamma-ray energy release rates per watt of reactor thermal power (at the time of the LOCA) versus the time after a LOCA. (The magnitudes and the curve shapes for the beta-ray energy release rates are quite similar.) Comparison is shown between the gamma-ray energy release rates for the more recent total "best-estimate" (Figure A.9) and for the current qualification testing criteria, Source 1 (Figure A.7). The "Source 1" assumptions of instantaneous release are very conservative for periods earlier than approximately 10^3 seconds following a LOCA. A consequence of this assumption is that some Class 1E equipment which is only required to function at the moment of a LOCA is now being qualified in radiation environments. For periods greater than approximately one hour, the energy release rates for the "Source 1" criteria are less than the "best-estimate," due largely to the omission of large fractions of the alkali metals, the tellurium group, and the noble metals. As indicated previously, recent preliminary experimental data on the gap fraction are as much as an order of magnitude less than even the "best-estimate" values. More definitive experimental data are needed, at least on meltdown release fractions.

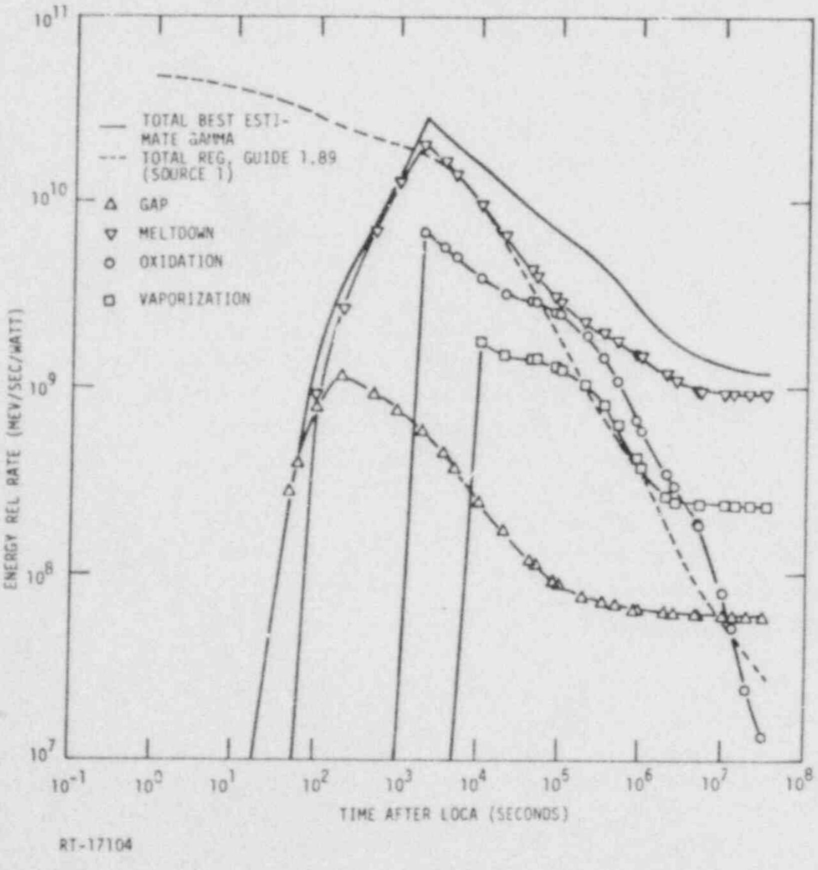


Figure A.10. Gamma-Ray Energy Release Rate for "Best-Estimate" Source Showing the Contributions of the Constituents, and Compared to the Total Regulatory Guide 1.89 Source Result (from Reference A.6)

Bonzon[A.14] has computed time-dependent dose rates for the airborne component of "Source 1," assuming a free-volume containment building of typical dimensions. The computed initial axial centerline maximum dose rate in the containment building is about 3.5 megarads/hour and the integrated 30-day dose is about 28 megarads. Including a uniformly deposited "plate-out" and a "waterborne" source under a set of reasonable assumptions of spatial distribution results in a maximum initial dose rate of about 4.5 megarads/hour and an integrated 30-day dose of about 45 megarads. Currently, standards[A.1] for exposure levels for the qualification of equipment specify an integrated dose over the first hour of 4 megarads, 55 megarads through the first thirty days, and 150 megarads through the first year. Common industry practice exposes Class 1E equipment to perhaps 200 megarads.

AGING AND ACCELERATED AGING

Environmental qualification of safety, and safety-related, components requires that the components be put in a condition equivalent to the end-of-life condition before being subjected to a test in the simulated environment of a hypothetical accident. Aging to the equivalent of forty years of in-service use necessitates accelerated aging techniques. Severe aging environments exist within the containment building where the combined environmental stresses of temperature, humidity, and radiation prevail. In containment, a nominal average temperature of about 100°F is assumed for aging; humidity is very variable but 90% relative humidity is assumed for aging; radiation exposure during the forty-year life is very dependent upon location. In the general free space above the reactor vessel shield, the forty-year dose is typically 20 megarads; in the free space in the vicinity of reactor coolant piping the typical dose is 160 megarads; on cooling piping, such as for instrumentation, the typical dose is 300 megarads.[A.2]

Accelerated aging in thermal environments has been extensively studied, particularly in organic materials. Much work remains to be done on the effects of humidity on aging. Radiation dose-rate effects, important to accelerated aging in (room temperature) radiation environments, has been extensively studied to examine the validity of the reciprocity of the product of dose rate and exposure duration for times much shorter than forty years. Much less work has been done to address the particulars of radiation aging for the forty years of reactor in-service use. In combined environments the synergistic effects on aging need to be understood.

Gillen and Salazar[A.7] have conducted an extensive study of aging in the combined environments of radiation and humidity in polymeric materials contained in electrical cabling of nuclear power plants typical of those plants currently under construction. Quoting from the abstract of their paper,[A.7] "Mechanical damage was monitored after room temperature aging in a Co-60 gamma radiation source at various humidities and radiation dose rates ranging from 1.2 megarad/hr to 2 kilorads/hr. For chloroprene, chlorosulfonated polyethylene, and silicone materials, the mechanical degradation was found to depend only on the total integrated radiation dose, implying that radiation dose rate effects are small. On the other hand, strong evidence for radiation dose rate effects were found for an ethylene propylene rubber material and a cross-linked polyolefin material.

Humidity effects were determined to be insignificant for all the materials studied." The data on ethylene propylene are shown in Figure A.11.[A.17] Shown are the percent changes in ultimate tensile elongation versus radiation dose for several dose rates both with and without a concurrent humidity environment of 70% relative humidity. Even though there is a measurable effect, the effect may be largely insignificant in practice to the "failure" of safety-related equipment.

The synergistic effects of radiation and temperature are much more pronounced in the aging of polymeric materials. An example, shown in Figure A.12,[A.7] is the degradation of chloroprene, showing the experimentally determined fractions of ultimate tensile elongation remaining versus time for radiation environments, at high and low temperature and for the high temperature environment alone. Gillen[A.8] has developed an accelerated aging method for use in combined radiation and temperature environments. The method is an extension of the overstress method of accelerated aging in a single environment. For radiation exposure where reciprocity of the product of dose rate and time is valid for the observed property degradation, the acceleration of aging (or the time production ratio) is simply the ratio of the dose rates. For temperature exposure the acceleration of aging (or the time reduction ratio) is Arrhenius.

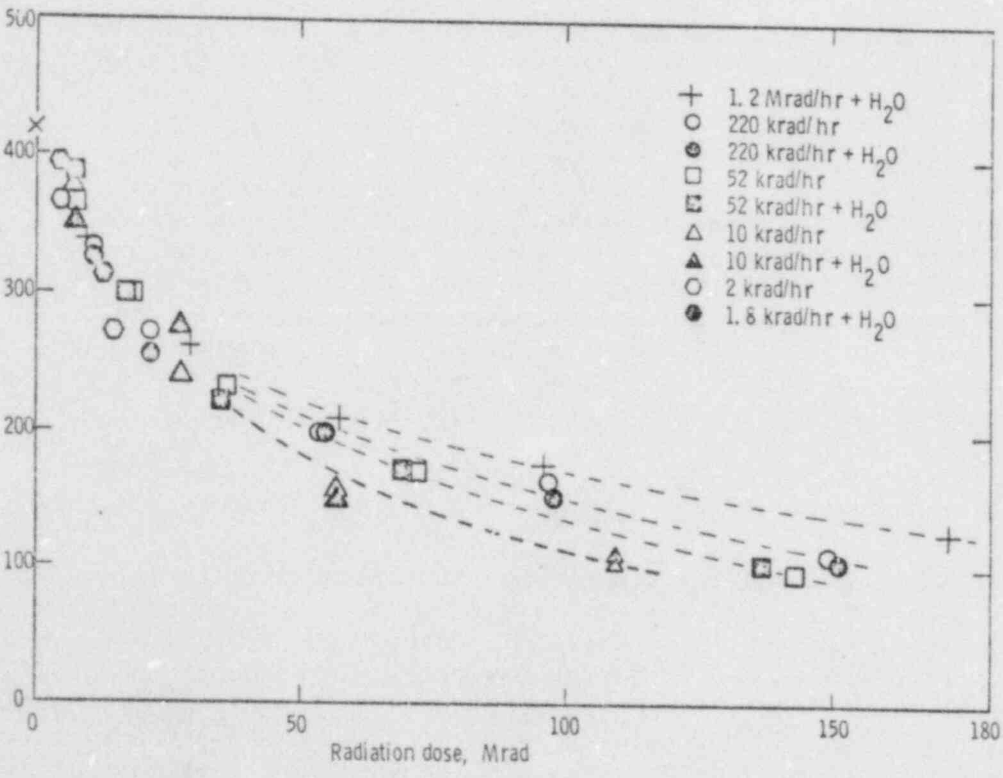
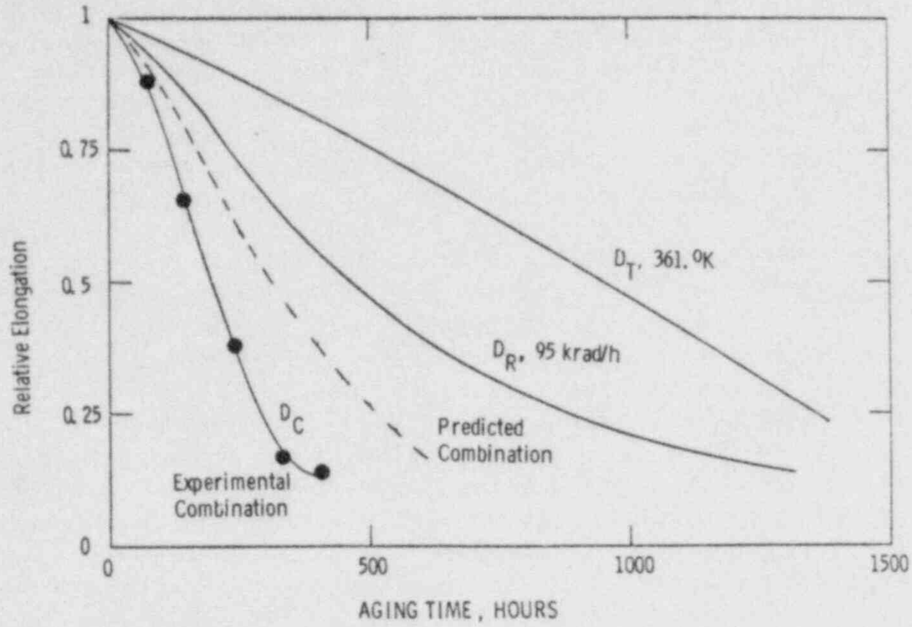


Figure A.11. Dose Rate Effects in EPR (from Reference A.7)



Aging of chloroprene: D_T, 361 °K; D_R, 95 krad/hour; D_C, 361 °K combined with 95 krad/hour; dashed curve, prediction without synergism.

Figure A.12. Combined Environment Effects in Chloroprene (rom Reference A.8)

To accelerate the degradation in a combined environment by a desired factor, the method proposed by Gillen[A.8] suggests the use of overstress temperature which accelerates the thermal degradation by the same factor in combination with an over-stress dose rate which accelerates the radiation degradation by the same factor. The values of overstress temperature and dose rate are obtained from the single environment accelerations. The key assumption is that the "synergistic" effects are accelerated by the same factor. Figure A.13[A.8] shows the method graphically.

SEQUENTIAL VERSUS SIMULTANEOUS ENVIRONMENTAL TESTS

Under current standards for the qualification of Class 1E equipment, the various LOCA-type environmental tests may be conducted sequentially in single environments, but in a specified order. Bonzon[A.9] has been conducting an experimental LOCA-simulation investigation to determine if significant synergistic effects are being overlooked by the sequential application of different environmental stresses. Nine generic LOCA-type tests have been conducted on single and multiconductor electrical cables, cable connector assemblies, and cable splice assemblies to evaluate synergistic effects. In addition, Franklin Institute Research Laboratories has conducted for Sandia National Laboratories a review of the results of sixty-six tests conducted for the nuclear power industry. No evidence has yet been found for significant functional synergistic effects in these components, during LOCA-simulation tests. (But strong synergisms have been observed in aging of cable materials.)

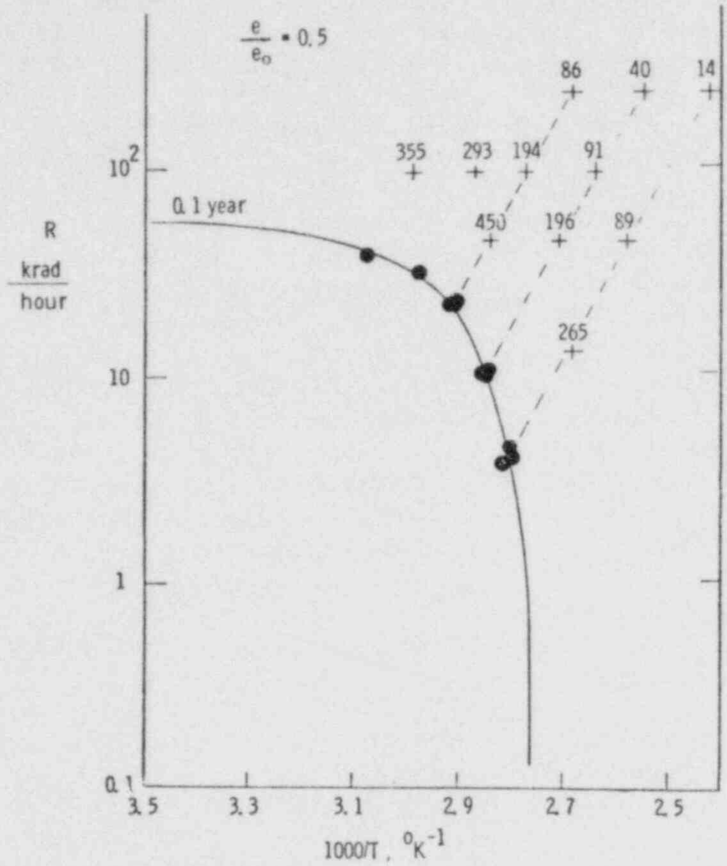


Figure A.13. Demonstration of the Method for Chloroprene (from Reference A.8)

SUMMARY/CRITIQUE

The reliability of the engineered safety systems of commercial nuclear power plants is vital to public safety throughout the nominal expected forty-year life of the plants. Also, in the wake of the Three Mile Island accident, the systems are also requisite to the least costly recovery of a plant for continued operation following a radiological accident. Even though the engineered safety systems of nuclear power plants are intrinsically far less complex electrically and electronically than are the space and defense systems developed to survive radiation environments, the requirements of forty-year reliability in radiological accident environments, preceded by aging in adverse operational radiation environments, is stringent. An additional complexity is the compound environment in both normal operation and in accident conditions.

Currently assumed radiation source terms to derive radiation field environments for the qualification of safety equipment appear by all evidence to be very conservative. Assumed release fractions of isotopes from their origins, such as from the fuel-clad gap and from core meltdown, appear conservatively large; retention within the breached primary pressure boundary is ignored; and the compartmentalization and topological features which offer significant shielding within the containment building are not generally accounted for in the radiation field calculations.

Although no significant synergistic effects have been observed in specific materials and elemental components such as cables and connectors in LOCA-simulations of compound accident environments, the question of synergistic effects in more complex components, subsystems, and systems remains open; the very conservative over-testing in radiation fields through the very conservative source term definition might be offsetting synergistic effects yet to be identified by further studies.

A method to derive accelerated aging criteria in compound radiation and temperature environments for a single electrical materials exhibiting synergistic effects has been empirically derived.[A.8] An exhaustive examination of the full complement of electrical materials potentially susceptible to synergistic effects has not been done, but strong synergistic effects have been observed in some of these materials.

The experience from the Three Mile Island accident raises questions pertaining to the qualification of safety, and safety-related, equipment to post-accident environments:

- should such equipment be assured to perform reliably subsequent to transient overpressures arising from combustion of gases generated by the accident?
- what should be the requirements on such equipment for extended times subsequent to "low-order" accidents?

In the concept of defense-in-depth there is an implicit assumption of reliability at each line of defense. The engineered safety systems are the final lines of defense against an unlikely, but possible, accident causing public consequences. Qualification of safety, and safety-related, equipment in tests founded upon defendable technical bases should be a minimum condition for assured reliability of safety equipment. Continued critical examination of the technical bases is warranted, coupled with more realistic determinations of the environments to be expected over the full range of plausible accidents and coupled with better definitions of the time-dependent requirements of the engineered safety systems.

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