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Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report

July - September 1978

Lloyd L. Bonzon, Kenneth T. Gillen, Douglas W. Dugan

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This report documents part of the Qualification Testing Evaluation (QTE) Program being conducted by Sandia Laboratories.

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U. S. NUCLEAR REGULATORY COMMISSION

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QUALIFICATION TESTING EVALUATION PROGRAM LIGHT WATER REACTOR SAFETY RESEARCH QUARTERLY REPORT

JULY - SEPTEMBER 1978

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AUTHOR CREDITS

Author credits for the abstracted topical sections are shown in the report. Section 1 and the general subsections of Sections 2 and 3 were written and edited by L. Bonzon. The general subsection of Section 4 was written by K. Gillen, E. Salazar, and/or R. Clough, and arranged and edited by L. Bonzon.

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LIGHT WATER REACTOR SAFETY RESEARCH
QUARTERLY REPORT

JULY - SEPTEMBER 1978

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 equipment. As a result of these activities, two 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 of which 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 as follows:

1. To provide assessments of post-LOCA qualification testing methodologies, including a qualitative assessment of the

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synergistic effects resulting from the combined environmental testing of representative Class 1E equipment;

2. To determine the radiation environment from the nuclear source term for a design basis LOCA and to evaluate the adequacy of radiation simulators; and

3. 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 typetesting of Class 1E equipment which 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 of the LOCA environments as compared with the sequential application of radiation, followed by the other LOCA environments, on identical components.

The Task 2 effort involves an assessment of the prescribed LOCA-radiation sources magnitudes, an evaluation of existing radiation simulators, an evaluation of component response to the LOCA-radiation signature, and the development of guidelines and rationale for use of radiation simulators in typetesting Class 1E components.

Typetesting requires a component which 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.

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1.1 Task 1 - Qualification Testing Methodologies Assessment

The FY78 effort under this task is concentrated in seven broad areas and numerous related subtasks.

1. LOCA typetests have been conducted, sequentially and simultaneously, on several Class 1E systems: electrical cables, connector assemblies, and splice assemblies. One additional test in this series is scheduled; tests interpretations, evaluations and documentation will be completed.

2. A major effort will be the completion and approval of a comprehensive test plan to be used as the long-range coordinating plan for the LOCA-testing methodologies study. This subtask is supported by the test facility upgrade and vulnerability evaluation subtasks.

3. A test facility upgrade proposal is being prepared and the test facility upgraded to (a) accommodate larger and more diverse Class 1E test items, and (b) 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.

4. A basis for the test plan is an evaluation of the apparent "LOCA-sensitivity" of safety-related equipment. A specific data base for this vulnerability evaluation is being obtained by subcontract to an architect-engineer, and will include Class 1E equipment lists, manufacturers, normal and accident environments definition, and comprehensive data packages for each equipment item.

5. Offsite data to complement the Sandia test data will be acquired, as availability allows, through subcontracts to manufacturers, testing laboratories, etc. Offsite expertise will be used to review the test plan and facility upgrade proposal as appropriate. As the testing schedule requires, offsite testing may be conducted to serve as benchmarks or

supplements to the Sandia test effort, especially during the period of test plan and facility development.

6. Initial planning and definition will be conducted under a subtask to provide an assessment of the design adequacy (DAA) of Class 1E systems subject to LOCA-caused nonrandom multiple failures, with the objective of evaluating qualification test methods as to their validity in predicting performance under LOCA-environment conditions. As currently envisioned, the subtask outlined includes (a) identification of "most important" and "most sensitive" safety-related equipment; (b) survey of manufacturers to identify representative designs/manufacturers used in current plant designs; (c) selection of generic designs by performing a preliminary DAA to identify potential failure modes; (d) conducting detailed DAA on these generic designs to identify failure modes; (e) conducting quality control audits during manufacture, installation, and use; (f) conducting tests and/or typetests to assess design adequacy.

7. At the specific request of the Commission, additional tests of connectors in a simulated LOCA environment will be defined and conducted using connectors qualified according to the IEEE-323 standards.[1.2,1.3]

1.2 Task 2 - Radiation Qualification Source Evaluation

The FY78 effort under this task is concentrated in five broad areas and related subtasks.

1. Additional source specification calculations based on the latest Regulatory Guide 1.89 guidelines,[1.4] and complementing previous calculations,[1.5] will be completed. These calculations will investigate, parametrically, various implicit, explicit, and unspecified assumptions in the Guide.

2. Based on these calculated radiation signatures (energy release rate, energy/number spectra, and particle type), a preliminary evaluation of the adequacy of currently used radiation simulators to duplicate the environments will be made. The evaluation will be based on "equivalence"

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of dose rate, depth-dose, and charged-particle distribution profiles in typical Class 1E equipment.

3. The assessment of radiation damage to Class 1E equipment from the dose-rate and depth-dose profiles will be completed. The objective of this subtask is to specify how closely the matching of dose rate, depth-dose, etc., must be accomplished to assure equivalence of damage by radiation simulators.

4. A "best-estimate" LOCA radiation signature will be defined and will be based on the accident-time-release sequencing as specified in WASH-1400[1.6] and other references.[1.7,1.8,1.9] The signature will eliminate several unrealistic, but conservative, assumptions specified in Regulatory Guide 1.89 and may be the basis of a revised Guide.

5. Depending on the results of these subtasks, effort may be directed toward (a) tailoring/designing of simulators to achieve better duplication of the actual component damage profiles, (b) 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 (c) developing guidelines and rationale for the use of simulators in typetesting.

1.3 Task 3 - Accelerated Aging Study

The FY78 effort under this task is concentrated in eight broad areas and numerous related subtasks.

1. Single environment aging tests are being conducted on electrical cable materials; elongation is used as the measure of damage in these tests. Single environment acceleration functions of damage versus time will be obtained.

2. Aging tests in combined radiation and temperature environments will be conducted in order to determine the importance of synergisms and

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to test the method postulated for combined environment accelerated aging.[1.10]

3. Tests to evaluate environmental rate effects will be conducted. Of particular concern are the rate effects associated with oxygen diffusion and radiation.

4. Alternate indicators of damage will continue to be investigated under this task. Examples of such indicators are voltage-withstand, mandrel-bend tests, and dissipation factors; these tests closely parallel current industry failure criteria which require "functionability" of electrical cable.

5. 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, ambiently aged cable samples are of limited value to the task.

6. As an alternate to accelerated aging methods, other methods of estimating age or equipment life will 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 utilized to extrapolate the remaining acceptable "life" of the equipment.

7. Preliminary extensions of all these efforts will be made, perhaps including extensions to other ambient environments (e.g., mechanical stress, other gaseous environs, humidity), other Class 1E equipment or materials, and to older style cables currently installed in nuclear power plants.

8. The effect of aging on the retention or degradation of fire-retardant additives in electrical cable will be investigated. Test specimens will be made of the common polymer materials and known fire-retardant additives; these will be subjected to accelerated aging and undergo quantitative testing to determine change in flammability with age.

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 Reviews with NRC Staff -- A conference call was held on July 10 with R. Feit and H. Schirling to discuss an NRC request to conduct moisture penetration tests in single-conductor electric cable. The discussion centered around an NRC draft test plan for the tests, the cable available at Sandia for testing, and other details of the test profile and test procedures. The result of this telecon was a directive from R. Feit to proceed with autoclave setup and to conduct scoping tests with the apparatus to test various cable/autoclave interface seals in an attempt to minimize cable "pinching" which would restrict moisture penetration (see also Section 2.2).

A general program and financial review was held July 25, in Silver Spring, for R. Feit and G. Bennett. The principal concern was the FY79 budget estimates and projected program tasks; other items discussed were distribution for all formal reports, QA practices for research programs, achievements of the accelerated aging task, the Commission-requested connector tests, and moisture penetration tests.

A FY79 program objective review was held for interested NRC staff in Bethesda on July 26. The review was based on the preliminary Schedule 189 submitted to RES and was presented to permit NRC staff to interact in the program direction.

On July 26, discussions were held with A. Szukiewicz concerning his preparation of a short-term action plan on safety-system equipment qualification and practices. Certain Sandia reports and subtasks (e.g., accelerated cable aging and simulator adequacy) do impact the plan; several newly published reports were forwarded to him for his review.

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A review of Sandia's preparations to conduct the moisture penetration tests in single-conductor electric cable was held with D. Davis, H. Schirling, M. Chiramal and other NRC staff in Bethesda on July 26.

A full program review was held at Albuquerque on September 6 for R. Feit, G. Bennett, and W. Rutherford. R. Feit provided a copy of the "Draft Interim Position -- Environmental Qualification (Task A-24)", dated August 22 for review and comment. In-depth discussions were concentrated on the simulator adequacy evaluation, the Savannah River Reactor cable evaluation, fire-retardant aging, the 6th WRSR meeting and presentations, and possible LOCA-tests of PVC-jacketed, PE-insulated, electric cable.

Separate tours of Sandia test facilities were given to two NRC groups on September 6. The low-intensity cobalt array (LICA) facility and the LOCA-test autoclave/pressure system were reviewed by C. Long and V. Benaroya. The LICA facility and the TA-V large cobalt facility were reviewed by G. Bennett and G. Lainis; subsequent to this tour, discussion was held on the results of the moisture penetration tests on electric cable.

K. Gillen made an in-depth technical presentation on the accelerated aging task to the (extended) NRC Qualification Review Group on September 13 in Bethesda at the specific request of R. Feit. Subsequent to the presentation, the ACRS Subcommittee presentation (of November 1) was discussed with G. Bennett, gas-blocked electric cable designs were discussed with W. Reeves of LLL, and the Task A-24 NRC short-term, action-plan draft was discussed with A. Szukiewicz.

A review of the electric cable moisture penetration tests results was held on September 14 in Bethesda for E. Butcher, D. Tondi, V. Thomas, G. Lainis, W. Reeves (LLL), C. Heit, M. Chiramal, R. Feit, and P. Shemanski. The group agreed that the test results indicated stranded cable leakage under the 70 psig driving potential. It was decided that a review of actual plant equipment installations was necessary to decide if such driving potentials could in fact be achieved during postulated high-energy line break accidents. Gas-blocked, splice-blocked, and solid conductor cables were discussed as possible installation upgrades.

The proposed study by NRR and Standards to estimate generic in-containment LOCA radiation dose rates and dose to Class 1E equipment was discussed with P. Tam on September 14. The proposal is being internally reviewed within NRR; M. Fleishman is now the Standards contact for this effort.

Program Formalization -- Based on the informal comments received on the preliminary Schedule 189 (officially transmitted on June 2), the final Schedule 189 for FY79 was prepared during August, with the Sandia signoff initiated on August 24. Formal submittal, through DOE/ALO, is expected in early October; the suggested funding level is approximately at the FY78 authorized level.

A Buff-Book submittal was prepared and submitted on August 24. This submittal represented routine input, except that several potential programmatic difficulties were identified which could affect the milestone schedules for selected subtasks.

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 quarterly[1.11] (January - March, 1978) was completed in late June and the report issued on September 26. A draft of the third quarterly[1.12] (April - June, 1978) was completed on August 23 and is undergoing internal Sandia-required review and signoff.

Standards Committee and Meetings Participation -- On August 16, K. Gillen (as a member) hosted a meeting of the IEEE Power Engineering Society Working Group 12-37 in Albuquerque. This Committee is charged with the study of, and preparation of recommendations on, methods of satisfying the aging requirements for qualification of safety-related electric cable for use in nuclear power generating stations.
K. Gillen, E. Salazar, and R. Clough presented the various aspects of the NRC-sponsored accelerated aging task.[1.13,1.14,1.15] The Committee also

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toured Sandia's aging facilities, including the recently completed LICA facility.

L. Bonzon attended the NRC/RES Fire Protection Research Review Group meeting, in Albuquerque on September 7. A number of industry contacts were made relative to the Qualification Testing Evaluation Program, particularly with regard to the connector tests and the possibility of obtaining naturally aged cable samples.

North Anna Unit 2 Plant Tour -- L. Bonzon and representatives of the NRC visited the 90+%-complete North Anna Unit 2 nuclear power station located near Mineral, Virginia, on September 19. Unit 1 (operating) and Unit 2 are twin 934 MW(e) Westinghouse PWRs. The purpose of the visit was to evaluate a new, typical, plant with regard to in-plant safety-related equipment installation and the potential for moisture penetration in electric cable because of pressure differentials during postulated high-energy line break accidents. In these plants, the owner/operator (Virginia Electric Power Company) has installed in-line butt splices in all safety-related cable, before their entry into sealed equipment, to act as a moisture block.

International Exchange -- Twelve abstracts (see Section 1.5) were submitted in mid-March to the International Meeting on Nuclear Power Reactor Safety, to be held October 16-19, 1978, in Brussels, Belgium; these covered all aspects of the NRC, IEEE, and Sandia (and subcontractor) programs. On May 24, notice was received that the Technical Program Committee had created a new session, "Environmental Equipment Qualification", and the (combined) papers were accepted.[1.12] Eight papers will specifically deal with the Sandia program and the full papers will be published in the proceedings of the meeting. Substantial effort was devoted to these papers during this quarter.

Foreign Interest -- By letter dated June 1, Dr. K. Yahagi (EE Dept., Waseda University, Tokyo, Japan) requested approval to visit Sandia Laboratories on November 10, 1978. Dr. Yahagi is Chairman of the 30-member Committee on the Ionizing Radiation Resistance of Electrical Insulating

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Materials. The Committee, sponsored by the Institute of Electrical Engineers of Japan, is chartered to prepare a comprehensive Japanese standard, paralleling IEEE-323 and IEEE-383, on the qualification of Class 1E equipment for use in Japanese nuclear power stations. His request was approved on June 28 and forwarded to him on July 5. Final confirmation of his planned visit was received from Dr. Yahagi on July 24.

Dr. Y. Nakase (Japan Atomic Energy Research Institute, JAERI) requested information (June 9) on the QTE program and on the International Meeting on Nuclear Power Reactor Safety; the preliminary program for that Brussels meeting was forwarded to him on July 17. He, in turn, provided preprint copies of a paper entitled "Radiation Resistance of Cable Insulating Materials for Nuclear Power Generating Stations", on August 16. At his (August 18) request, SAND78-0718,[1.16] was also forwarded to Dr. Nakase on August 23.

By letter dated August 31, Dr. S. Machi (JAERI) requested approval to visit Sandia Laboratories on October 27; he is to be accompanied by Dr. Munekata (JAERI), Dr. Oshima (Pilot Scale Research Station), and Mr. Tomita (Radia Industry). His request was approved by letter dated September 11, 1978.

Dr. Nakase also requested specific information on the application of IEEE standards to equipment qualification tests in an August 18 letter. Reply to his questions were made through Dr. F. Campbell (Naval Research Laboratories), who coordinated the response, on September 26.

To complement attendance at the International Meeting on Nuclear Power Reactor Safety, Brussels, in October, it was proposed to NRC, by letter dated July 31, that visits be made to several European facilities conducting significant, complementary, research. Included in the proposal were visits to (a) the Fire Research Station in Borehamwoods, England, (b) Framatome, in Paris, France, (c) Saclay near Paris, France, (d) Cadarache near Avignon, France, (e) Bugey Power Station near Lyon, France, (f) CERN in Geneva, Switzerland, (g) ASEA-KABEL in Stockholm, Sweden, (h) ASEA-ATOM in Vasteras, Sweden, and (i) Oy Stromberg Ab in Helsinki, Finland. Formal approval for the European trip was received by letter dated September 7.

Industry Liaison -- The general interest in the overall QTE program remains exceptionally high. A number of industry requests were received and processed during this quarter; these are briefly reviewed in Table 1.1.

TABLE 1.1

Industry Liaison

Date	Company	Requested: Prior Reports	Requested: Mailing List	Requested: Other
July 10	Sargent & Lundy			Concerning source signatures and associated assumptions in their calculations
July 13	Bendix - Sidney	SAND76-0746 SAND77-0511A SAND77-1075A SAND77-1654C SAND77-1713C		
July 17	Wyle			Reactor safety work at Sandia
July 31	SRP - duPont	Test VI Test VII Test VIII	X	Qualification of certain Bendix connectors
August 7	Wyle			Visit; discussed facility upgrade and connector tests
August 21	Westinghouse			Advice on transmitter thermistor radiation damage and levels
August 28	ORNL	SAND78-0161A		

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TABLE 1.1 (cont)

Date	Company	Requested Prior Reports	Requested Mailing List	Requested Other
August 31	ITT - Surprenant	SAND76-0715 SAND76-0740 SAND78-0091 SAND78-0341 SAND78-0718	X	
September 5	SAI - LaJolla			Information on future work
September 7	Duke Power			Obtain naturally aged cable; connector tests
September 7	WEP Company			Polyethylene cable aging damage
September 27	SAI - LaJolla			Information on future work

1.5 Publications/Presentations

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

-- L. J. Klamerus, "Tests I and II, Sequential Mode; Cables and Paints," Sandia Laboratories, Albuquerque, NM, January 1976.†

-- L. J. Klamerus, "Test III, Simultaneous Mode; Cables and Paints," Sandia Laboratories, Albuquerque, NM, April 1976.†

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

†These Quick-Look reports are available in the USNRC Public Document Room; test results have been incorporated into Sandia topical report SAND78-0067 (Reference 1.17).

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-- L. J. Klamerus, "Test IV, Simultaneous Mode; Cables and Paints," Sandia Laboratories, Albuquerque, NM, July 1976.†

-- L. L. Bonzon, "Test V, Simultaneous Mode; Cables and Paints," Sandia Laboratories, Albuquerque, NM, August 1976.†

-- L. L. Bonzon, "Test VI, Simultaneous Mode; Cables, Connectors, and Lubricants," Sandia Laboratories, Albuquerque, NM, January 1977.†

-- L. L. Bonzon, "Test VII, Sequential Mode; Cables, Connectors, and Lubricants," Sandia Laboratories, Albuquerque, NM, February 1977.†

-- K. T. Gillen, E. A. Salazar, and C. W. Frank, "Proposed Research on Class I Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments," SAND76-0715, NUREG-0100, Sandia Laboratories, Albuquerque, NM, April 1977.

-- L. L. Bonzon, "Design Basis Event (DBE) Testing," SAND77-0150C, <u>1977 Proceedings of the Institute of Environmental Sciences</u>, April 1977.

-- D. V. Paulson and S. P. Carfagno, "A Review of Class IE Cable Qualification Data," FIRL Final Report F-C4598-1, prepared for Sandia Laboratories, June 1977. (Incorporated into Sandia topical report, SAND78-0067, Reference 1.17.)

-- N. A. Lurie, "Definition of Loss-of-Coolant Accident Source for Radiation Component Qualification: Phase I Final Report," IRT 8167-001, prepared for Sandia Laboratories, June 1977. (This report intended for internal Sandia use only.)

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†These Quick-Look reports are available in the USNRC Public Document Room; test results have been incorporated into Sandia topical report SAND78-0067 (Reference 1.17).

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-- L. L. Bonzon, "Test VIII, Sequential Mode; Cables, Splices, and Connectors," Sandia Laboratories, Albuquerque, NM, July 1977.[†]

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

-- K. T. Gillen, "Accelerated Aging in Combined Stress Environments," SAND77-0511A, Proceedings of the International Conference on Environmental Degradation of Engineering Materials, Virginia Polytechnic Institute Press, October 10-12, 1977.

-- L. L. Bonzon, "Radiation Signature Following the Hypothesized LOCA," SAND76-0740, NUREG76-6521, Sandia Laboratories, Albuquerque, NM, Revised October 1977.

-- D. V. Paulson and S. P. Carfagno, "Apparatus and Procedures for Qualification of Class 1E Electrical Cable," FIRL Final Report F-C4598-2, prepared for Sandia Laboratories, October 1977. (This report intended for internal Sandia use only.)

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

-- K. T. Gillen and E. A. Salazar, "Accelerated Aging Studies of Electric Cable Material," SAND77-1654C, USNRC Fifth Water Reactor Safety Research Information Meeting, November 7-11, 1977.

[†]These Quick-Look reports are available in the USNRC Public Document Room; test results have been incorporated into Sandia topical report SAND78-0067 (Reference 1.17).

-- L. L. Bonzon, K. T. Gillen, and E. A. Salazar, "A Consolidated Program to Evaluate Class 1E Equipment Qualification Techniques," SAND77-1075A, Transactions of the American Nuclear Society, Vol 27, 1977.

-- N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source: Final Report--Phases 2, 3, and 4," IRT 8167-002, prepared for Sandia Laboratories, December 1977. (This report intended for internal Sandia use only.)

-- L. L. Bonzon, N. A. Lurie, and J. A. Naber, "Adequacy of Radiation Sources for Qualification of Class 1E Reactor Components," IRT 8167-003/SAND78-0161A, January 1978. Accepted for publication/presentation at the American Nuclear Society, 1978 Annual Meeting, San Diego, CA, June 1978.

-- K. T. Gillen, "Combined Environmental Accelerated Aging Model Applied to Electric Cable Material," SAND78-0178A, presented at the Materials Science and Materials Chemistry Special Seminar, Westinghouse R&D Center, Pittsburgh, PA, January 30, 1978.

-- L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source," SAND78-0090, Sandia Laboratories, Albuquerque, NM, February 1978.

-- N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source, Summary and Conclusions: Final Report--Phases 2, 3, and 4," IRT 8167-004, prepared for Sandia Laboratories, February 1978. (This report intended for internal Sandia use only.)

-- K. T. Gillen and E. A. Salazar, "Single and Combined Environmental Accelerated Aging of Electrical Cable Insulation and Jacketing Materials," SAND78-0559A, March 1978. Accepted for publication/presentation at the 1978 Conference on Electrical

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Insulation and Dielectric Phenomena, Pocono Manor, PA, October 29-November 2, 1978.

-- L. L. Bonzon, "Design Basis Event (DBE) Testing," SAND77-0150C, Technology Monograph, A Compilation of Qualification Considerations for Nuclear Power Generating Stations, Institute of Environmental Sciences, 1977.

-- The following ten (10) abstracts were submitted in March to the International Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, to be held October 1978:

- R. E. Luna and L. L. Bonzon, "Methodology Assessment: The Role of the Independent Laboratory," SAND78-0342A.

- L. L. Bonzon and R. E. Luna, "The Qualification Testing Evaluation (QTE) Program: An Overview," SAND78-0343A.

- K. T. Gillen and E. A. Salazar, "Aging of Nuclear Power Plant Safety Cables," SAND78-0344A.

- K. T. Gillen, "A Proposed Method for Combined Environment Accelerated Aging," SAND78-0501A.

- L. L. Bonzon, D. V. Paulson, and S. P. Carfagno, "An Evaluation of Synergistic Effects in Electrical Cables During LOCA Typetests, Using Historical Data," SAND78-0345A.

- L. L. Bonzon, "An Experimental Investigation of Synergisms in Class 1E Components," SAND78-0346A.

- L. L. Bonzon, S. G. Kasturi, and G. T. Dowd, "Qualification of Class 1E Equipment: The Role of the Utility and Architect-Engineer," SAND78-0347A.

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- L. L. Bonzon, N. A. Lurie, and J. A. Naber, "The Hypothesized LOCA Radiation Signature and the Problem of Simulator Adequacy," SAND78-0348A/IRT 8167-005.

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

- L. L. Bonzon, R. E. Luna, and S. P. Carfagno, "Qualification Problems: The Rest of the Iceberg," SAND78-0350A.

-- F. V. Thome, "Preliminary Data Report, Testing to Evaluate Synergistic Effects from LOCA Environments. Test IX. Simultaneous Mode; Cables, Splice Assemblies, and Electrical Insulation Samples," SAND78-0718, Sandia Laboratories, Albuquerque, NM, April 1978.

-- "Qualification Testing Evaluation Program, Light Water Reactor Safety Research, Quarterly Report, October - December 1977," SAND78-0341, Sandia Laboratories, Albuquerque, NM, April 1978.

-- L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions," SAND78-0091, Sandia Laboratories, Albuquerque, NM, May 1978.

-- R. L. Clough, "Computer Modeling of Polymer Radiation Chemistry," SAND78-0876A, May 1978. Presented at, and published in the Proceedings of, the Fifth International Symposium on the Chemistry of the Organic Solid State, Brandeis University, Waltham, MA, June 13-16, 1978.

-- K. T. Gillen and E. A. Salazar, "A Model for Combined Environment Accelerated Aging Applied to a Neoprene Cable Jacketing Material," SAND78-0559, June 1978. Accepted for publication/presentation at the 1978 Conference on Electrical

Insulation and Dielectric Phenomena, Pocono Manor, PA, October 29-November 2, 1978.

-- N. A. Lurie, J. A. Naber, and L. L. Bonzon, "Adequacy of Radiation Sources for Qualification of Class 1E Reactor Components," IRT 8167-003/SAND78-0161A, Transactions of the American Nuclear Society, Vol 28, 1978.

*-- 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. (This report intended for internal Sandia use only.)

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

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

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1.6 References

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1.6 "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants; Appendix VII: Release of Radioactivity in a Reactor Accident," WASH-1400, NUREG75-014, U.S. Nuclear Regulatory Commission, October 1975.

1.7 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 National Laboratory, June 1974.

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1.9 M. J. Kolar, J. R. McCarty, and N. C. Olson, "Post-Loss-of-Coolant Accident Doses to Pressurized Water Reactor Equipment," Nuclear Technology 36: 74-78, November 1977.

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, Sandia Laboratories, Albuquerque, NM, July 1977.

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1.14 E. A. Salazar, K. T. Gillen, and R. L. Clough, "Evidence of Synergistic Effects During Aging in Combined Radiation and Temperature Environments," SAND78-1634A, Sandia Laboratories, Albuquerque, NM, August 1978. Presented to IEEE WG 12-37, August 16, 1978, at Albuquerque.

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1.16 F. V. Thome, "Preliminary Data Report, Testing to Evaluate Synergistic Effects from LOCA Environments. Text IX. Simultaneous Mode; Cables, Splice Assemblies, and Electrical Insulation Samples," SAND78-0718, Sandia Laboratories, Albuquerque, NM, April 1978.

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

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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. Section 2.2 details a specific program aspect that engendered particular effort and achieved a significant milestone during the reported quarter.

2.1 Task 1 - Technical Activities Summary

Publications -- Final distribution of SAND78-0067, "An Experimental Investigation of Synergisms in Class 1 Components Subjected to LOCA Typetests,"[2.1] was made in late September. It incorporates the results from nine (sequential or simultaneous) synergistic typetests and includes the full text of the offsite synergism study performed by FIRL.[2.2]

The nine typetests were conducted on a variety of Class 1 equipment, but the major thrust of the evaluation was on Class 1E equipment. In this category, electrical cables from four manufacturers, cable connector assemblies constructed by three connector suppliers, and cable field-splice assemblies were tested and the results can be briefly summarized as follows:

- No functional synergisms were observed for any electrical cables; Some single-conductor cables failed functionally during specific typetests.

- Functional synergisms could not be evaluated for cable connector assemblies since most assemblies failed (functionally) in the typetests.

- No functional synergisms were observed for the cable splice assemblies.

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UEC Subcontract -- United Engineers and Constructors (UEC) is completing a four-concurrent-phase subcontract to assemble comprehensive data packages for all in-containment Class 1E equipment for a contemporary PWR nuclear power plant. Through June, 21 data packages had been completed,[2.3] with 4 or 5 data packages remaining to be completed.

During this quarter, a revision to the "Environmental Design Criteria" section was prepared and transmitted in mid-July. In late July, a meeting was held with UEC staff in Philadelphia to review the work to date and to discuss other particulars remaining to be completed. In August, UEC transmitted all computer sorts of Revision 5 of the Class 1E equipment lists and Revision 1 of the "Equipment Design Environmental Chart." Also in August, UEC was requested to address the final four or five generic data packages; they are currently evaluating and preparing these packages.

FIRL Subcontract -- The principal efforts under this contract during the quarter were the continuing review of the UEC subcontract submittals (see above) and the test facility upgrade proposal reviews required by NRC/RES staff. For the latter, FIRL completed the review of the first package (A General Plan to Facilitate Review by FIRL) in late June; in general, FIRL concurred with the conceptual philosophy of the design.[2.3] The second package (Autoclave and Ancillary Equipment) was submitted for FIRL review on July 5; the third and final package (High Intensity Adjustable Cobalt Array) was submitted on July 10.

In late July, a meeting was held with FIRL staff in Philadelphia to discuss: preliminary comments and questions on the two facility review packages; connector testing capability at FIRL; and historical data of radiation tests, at various levels, of electrical cable. As a result of this meeting, the FIRL staff formalized their comments on the facility review packages.[2.4,2.5] Their summary comment was that "... no serious difficulties were found with the design. Some minor suggestions..." were made.

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Connector Tests -- Based on the UCS petition of November 4, 1977, the USNRC Commissioners requested that "additional tests of connectors in a simulated LOCA environment be conducted by Sandia Laboratories using connectors qualified according to the IEEE-323 standard when a suitable test facility is available."[2.6] The significant effort last quarter involved the interpretation of, and response to, the Commissioners' memorandum by NRC (RES, NRR, IE) staff and internal review of proposed test plans and specific connectors/plants to be retested. By letter dated May 22, DOR staff recommended the following connectors:[2.7]

BWR	(1)	Peach Bottom-Pyle National connectors; or
	(2)	Browns Ferry-Bendix connectors
PWR	(1)	Palisades-Viking Spec connectors; or
	(2)	Oconee-Viking Spec connectors

The staff recommended considering other connectors, if necessary, "...due to availability of the connectors to be tested, extensive procurement time or other constraints." Based on these recommendations, connector procurement/test packages were obtained from these utilities and initial contacts made with suppliers.

During July, efforts were directed towards the Browns Ferry-Bendix connectors. The specific test plans, procedures, and test reports were reviewed by NRC and Sandia staff. Tentative arrangements were made with Browns Ferry (BF) staff to assemble the connectors onto BF-supplied electric cable. Following internal-NRC concurrence, authorization to purchase the necessary Bendix connectors was received, by telecon from R. Feit, on August 1; six of each connectors were ordered in early August with delivery expected in November: 10-214628-51S, 10-214028-51P, 10-214636-78S, 10-214036-78P.

On August 2-3, W. Rutherford and F. Jablonski of NRC visited Sandia to discuss the details of the Bendix-BF tests and to consider other connectors for additional test; particular consideration was given to the Palisades-Viking connectors.

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In early August, Sandia staff prepared a test plan for the Browns Ferry-Bendix connector tests modeled on Wyle test reports to the Tennessee Valley Authority. Working copies of the plan were submitted to R. Feit for internal-NRC review on August 14. Concurrently Sandia requested that NRC staff reconsider all aspects of the connector tests; the NRC-staff was advised that if NRC still wanted to conduct the connector tests, Sandia would implement the program with the understanding that Sandia recommended against it.

On September 6, further meetings were held between Sandia and NRC staff in Albuquerque on the Browns Ferry-Bendix tests; NRC had agreed to continue with this selection of plant and connectors. Similarly, W. Rutherford provided information on CONAX modules, qualified for use in Arkansas Nuclear One, Unit 2, as a potential test candidate.

Further discussions were held with NRC staff, R. Satterfield, A. Szukiewicz, W. Rutherford, and R. Feit, on September 14, in Bethesda. There was general agreement to proceed with the Browns Ferry-Bendix tests; the CONAX modules will also be considered further.

On September 18, a partial shipment (three of four items) of Bendix connectors was received; the remaining item is scheduled for delivery in November. On September 22, a shipment of Scotchcast #9 (XR 5240) Electrical Resin was received, to be used in the final connector/cable assembly potting.

Facility Upgrade -- The existing test facility will be upgraded to accommodate larger and more diverse Class 1E test items and to allow selectable radiation dose rates and stationary radiation spatial gradients. The upgrade includes (1) new radiation sources and source-positioning apparatus, (2) an electro-hydraulic control system to select irradiation rates and to interlock for safety, (3) facility shielding modifications, (4) irradiation cell modifications, (5) test chambers, and (6) diagnostic/test equipment. The major efforts this quarter are detailed below.

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The order for cobalt was placed with Neutron Products, Inc., for approximately 300 kilocuries in 32 source pencils; delivery is scheduled for October 16. The cobalt specific activity loading will be adjusted (over the pencil length) to provide a quasi-flat radiation profile.

Design was completed for the high-intensity adjustable cobalt array (HIACA) positioning apparatus and submitted for fabrication quotes in early September. The design consists of three stages of telescoping tubes. Nested, the height is about 5-1/2 feet; extended, the height is about 12 feet. The 128 permanent base tubes are on 4 radii positions that permit (manual) adjusting of the source radius. The 32 tubes/pencils will normally be utilized in 8-ganged groups of 4 to tailor the dose rate. Two tube designs are to be fabricated for testing in a preliminary test fixture. The final design will then be selected; the test fixture will also serve as a device to adjust tolerances, etc.

The autoclave test chamber is being fabricated and will consist of the following features:

- Overall height - 88 in.
- Lower chamber height - 65 in.
- Head height - 23 in.
- Overall diameter - 22 in.
- Inside diameter - 20-1/2 in.
- Volume - approximately 16 ft^3
- Head penetrations - 9
- Net weight - 550 lb
- Maximum working pressure - 150 psig at 386°F
- Safety factor - 11.6

A 206,000 Btu/hr (6 horsepower) boiler was purchased, as the steam system supply, as an ASME-code approved item. Two 20 ft^3 accumulators were also purchased to ASME-code. A chemical storage/mixing tank with a capacity of 750 gal is being fabricated and the associated spray pumps were ordered.

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The necessary design and safety-committee reviews were completed this quarter. This includes the Franklin Institute reviews (see above) as well as the internal Sandia Non-Reactor Safety Review Committee reviews specific to safety of the proposed design. For the latter, approval was received for the following designs: source pencils, dummy/spacer pencils, HIACA positioning apparatus, autoclave chamber, steam system, reactor fuel storage racks, power distribution upgrade, pencil storage rack, pencil handling tools, GIF pit cover, ventilation shaft radiation shields, and manipulator-port radiation shields.

A cobalt storage stand and handling tool were designed and fabricated this quarter. The stand provides interim storage for cobalt for shipping cask or HIACA transfers. The handling tool incorporates a quick-release mechanism for underwater transfer of the cobalt; the cobalt pencil design incorporates end pieces to adapt to the handling tool.

A number of facility improvements were initiated and/or completed this quarter. Two 440 vac, 3 phase, 100 amp receptacles were ordered to supply boiler power and to service experiments. To clear the water storage pool, reactor fuel storage racks were designed, fabricated, and installed to store the old Annular Core Pulse Reactor (ACPR) fuel elements; these attach to the pool sides, out of the way of the cobalt and HIACA. An order was placed to refurbish the cell manipulator and to add shielding to prevent radiation streaming through the ports; the order is scheduled for completion in January 1979. To complete the common-wall shielding requirements, a fixture was designed and fabricated that inserts into a ventilation shaft in the wall; depleted uranium plates will be fashioned for the fixture to provide a shield equivalent-thickness of 4 in.

2.2 Tests of Autoclave Sealing Systems to Provide a Capability for Single-Conductor Cable Moisture Penetration Tests*

On July 12, 1978, RES staff forwarded two documents to Sandia describing a desired testing capability to support licensing and review decisions within the USNRC.

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*This section prepared by L. Bonzon and D. Dugan, Division 4442.

- Electrical Conductor/Insulator Hydraulic Phenomenon Considerations for Scoping Tests and Followup Actions

- Electrical Conductor/Insulator Hydraulic Phenomenon, Test Program Outline.

In these reports, the NRC requested that Sandia establish a test capability to evaluate the migration of moisture through single-conductor electric cable.

Based on these reports and discussions with RES staff, Sandia conducted three tests during August to develop an autoclave sealing system to withstand 70-psig differential pressures without excessive pinching of exiting single-conductor electric cables. The three tests evaluated a two-part RTV epoxy (Test I), a non-pinching pseudo-compression fitting using elastomeric rubber stoppers (Test II), and a one-part epoxy (Test III). In an attempt to detect the extent of moisture migration in single-conductor cables, an intense dye was used, and in some cases posttest sectioning of the cable was done to inspect for dye/moisture migration.

2.2.1 Seal Systems -- Figures 2.1 and 2.2, from Test I, show the basic arrangement for all tests. Single-conductor No. 12 AWG wire was wrapped on mandrels (6.4-in. diameter) for placement in the autoclave chamber (Figure 2.1). Approximately 10 ft. of cable was inside the test chamber and 2 to 3 ft. of cable was external to the chamber.

Figure 2.2 shows the cable(s) end inside the chamber. Approximately 3 in. of insulation was stripped from the tinned copper conductor and a 6-in. length of shrink-tube was positioned over the end (extending about 3 in. onto the cable insulation) as a dye pouch. Crystal violet and metinal yellow dyes were used, with crystal violet proving to be the most intense.

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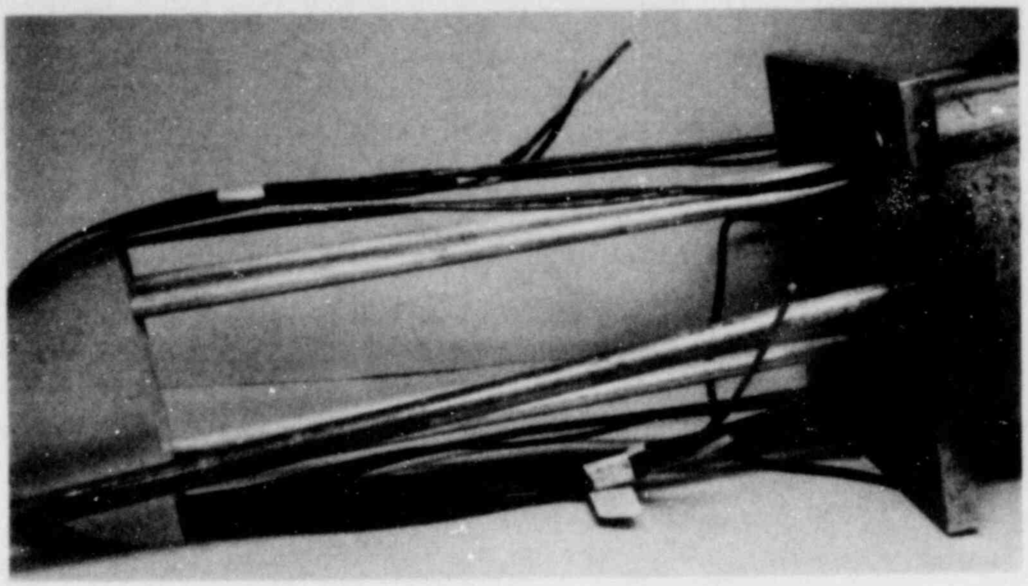

Figure 2.1. Typical Cable Wrap Around Mandrels for All Tests

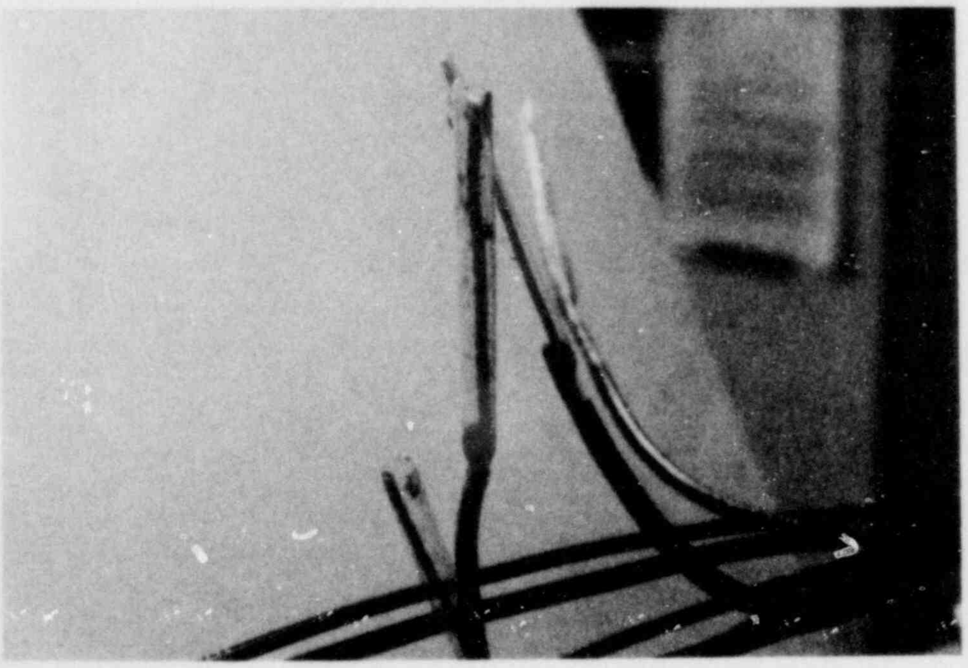

Figure 2.2. Typical Dye Pouch on High-Pressure Cable End, Using Shrink Tubing

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Figure 2.3 shows the seal system used in Test I. The exiting cables were potted in aluminum tubes and end caps used, on either end, to reduce leakage paths and help hold the epoxy in place during the pressure tests. The potted tubes were held in place by 1/2-in. Swagelock fittings. The aluminum tubes were cleaned and primed with GE-4155, potted with (a two-part) GE-RTV-630 epoxy, and cured at 70°C for 4 hr. In Test I, this sealing system held pressure for the test duration (1 hr and 35 min), including a rapid rise (less than 10 s) to the 70-psig test pressure. This sealing system is effective, then, for all cables not containing sulfur compounds (which prevent the setting of two-part epoxies).

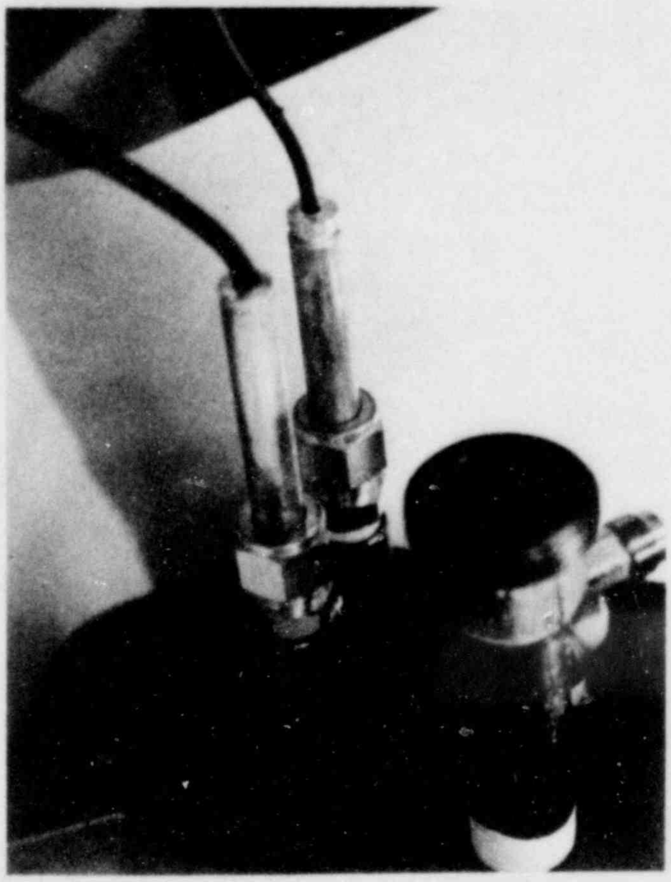

Figure 2.3. Test I Seal System, Two-Part Epoxy in Aluminum Tubing Held With Swagelock Fittings

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Figures 2.4 through 2.6 show the seal system in Test II. Swagelock fittings were used to hold a rubber stopper in place which served to provide the pressure seal. The rubber stopper was positioned, small-area side to the pressure, so the pressure would not wedge the rubber stopper and thus pinch the cable. The rubber-stopper/cable interface tightness was adjustable by the Swagelock cap and was adjusted to "minimum" tightness for these tests. This sealing system held pressure for the test duration (3 hr and 30 min) including a rapid rise (less than 10 s) to the 70-psig test pressure. This sealing system is effective, then, for all cables by adjusting the rubber stopper hole size and Swagelock fittings.

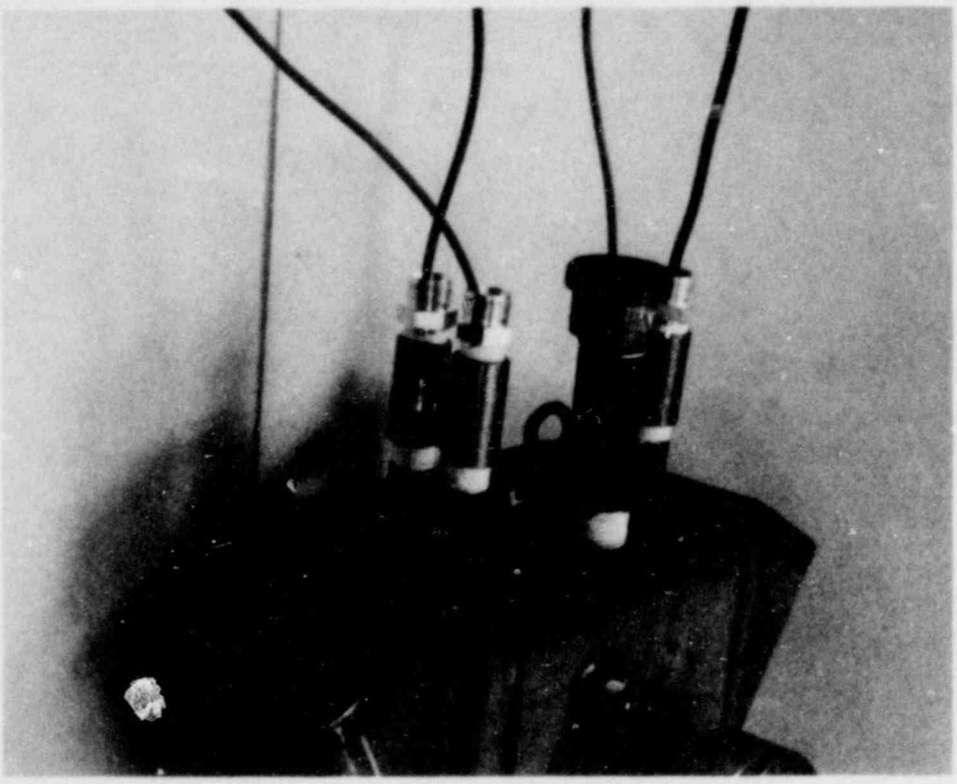

Figure 2.4. Test II Seal System: Existing Autoclave Chamber

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Figure 2.5. Test II Seal System: Rubber Stopper With Hole and Swagelock Caps

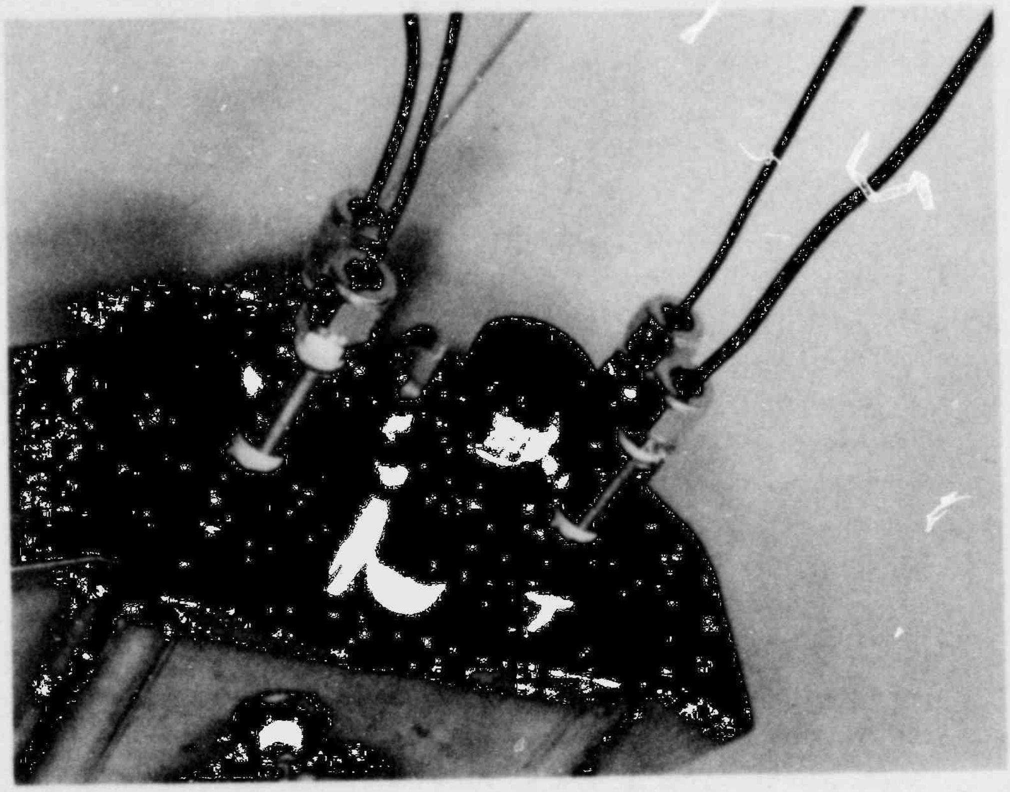

Figure 2.6. Test II Seal System Assembled

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Test III repeated the scheme used in Test I, except that a one-part RTV epoxy was used as the sealant. At 8 min into the test, steam leakage and sealant bubbling forced the test to be terminated. The one-part epoxy is not acceptable as a seal system at these pressures and temperatures.

In summarizing the seal systems tests, the rubber stopper/Swagelock system is best. It is easily adaptable to any cable size. The test results (amount of moisture leakage, discussed below, were not significantly different from the results using the two-part epoxy) and posttest examination of the cable indicated no excessive cable pinching.

The use of dye to evaluate moisture migration was only marginally successful. In some tests, a 1-ft sample was sectioned from the cable ends exposed to the high pressure. In only one cable was there evidence of the dye. Hence, it is not useful to rely on the dye as an "exact" indicator of the extent of moisture migration.

2.2.2 Moisture Leakage Observed -- Tests I and III can be discussed briefly in terms of moisture leakage. Test I was not designed to be quantitative. Two cross-linked polyethylene cables and two ethylene propylene rubber cables with stranded conductors were tested; all four cables showed leakage within 2 to 3 min of the pressure application. Test III, which was terminated after 8 min because of seal leaks, included an (old-style) polyethylene and two cross-linked polyethylene stranded cables and a neoprene solid cable; the solid cable did not leak, but the stranded cables began to leak within a few minutes of the pressure application.

Test II provided some quantitative information. Figures 2.7 through 2.12 show the sequence of events during the test. The cables were a solid-conductor neoprene and three stranded-conductor types: an (old-style) polyethylene and two cross-linked polyethylenes. All stranded-conductor cables began to leak within 1 min of the pressure application and continued to leak throughout the 3.5-hr, 70-psig test. The solid-conductor cable never leaked. The table below summarizes the amount of leakage. At about 30 min into Test II, the leakage from the polyethylene cable was largely steam, indicating some cable insulation failure. This was

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confirmed by visual observations, as indicated in Figure 2.9 and subsequent figures; the side of the graduated beaker shows condensation.

TABLE 2.1

Moisture Leakage (cc), Test II

Time	Neoprene Solid	Old-Style Polyethylene	XLP-1	XLPE-2
(Leak Start)	(--)	(34 sec)	(20 sec)	(55 sec)
5 min	--	1	2	1
10	--	2	2.5	2
15	--	2.5	3	2
30	--	2.5-3.0*	4	3
1 hr	--	3	6.5	6
1.5	--	3.5	9	9
2	--	4	11	11
2.5	--	4	12.5	12.5
3	--	4	15	15
3.5	--	4	17	18

*At about 30 min, the leakage from the polyethylene cable was largely steam.

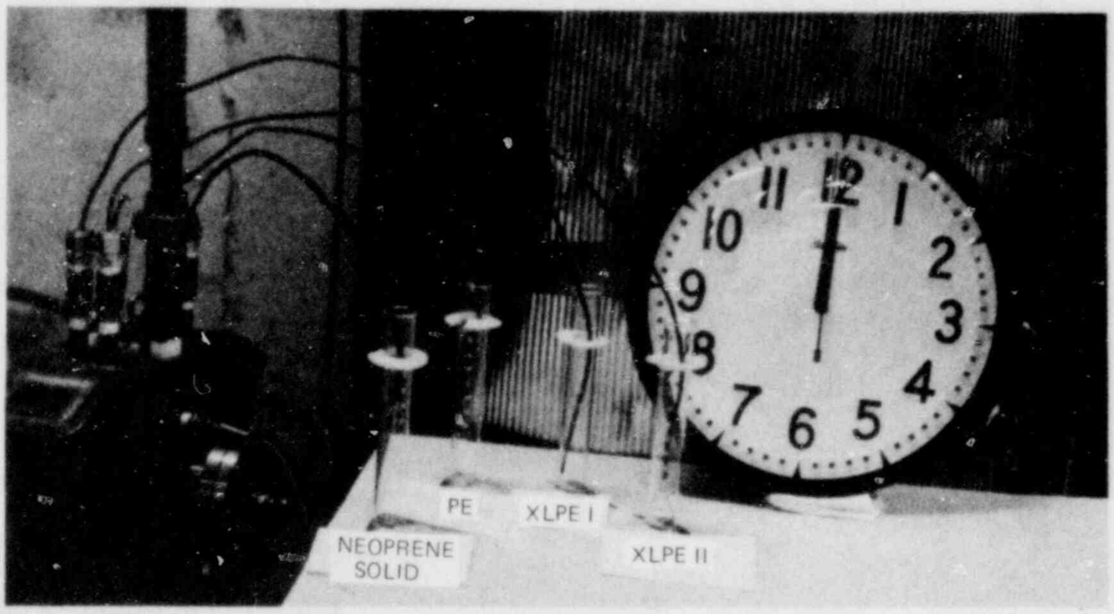

Figure 2.7. Test II Prior to Start of Test

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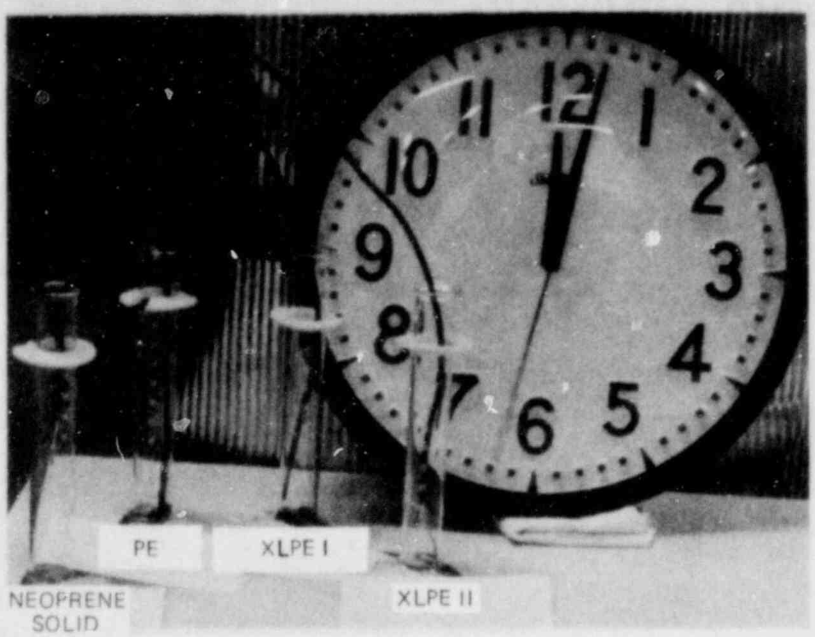

Figure 2.8. Test II at 1-1/2 Minutes into 70-psig Test

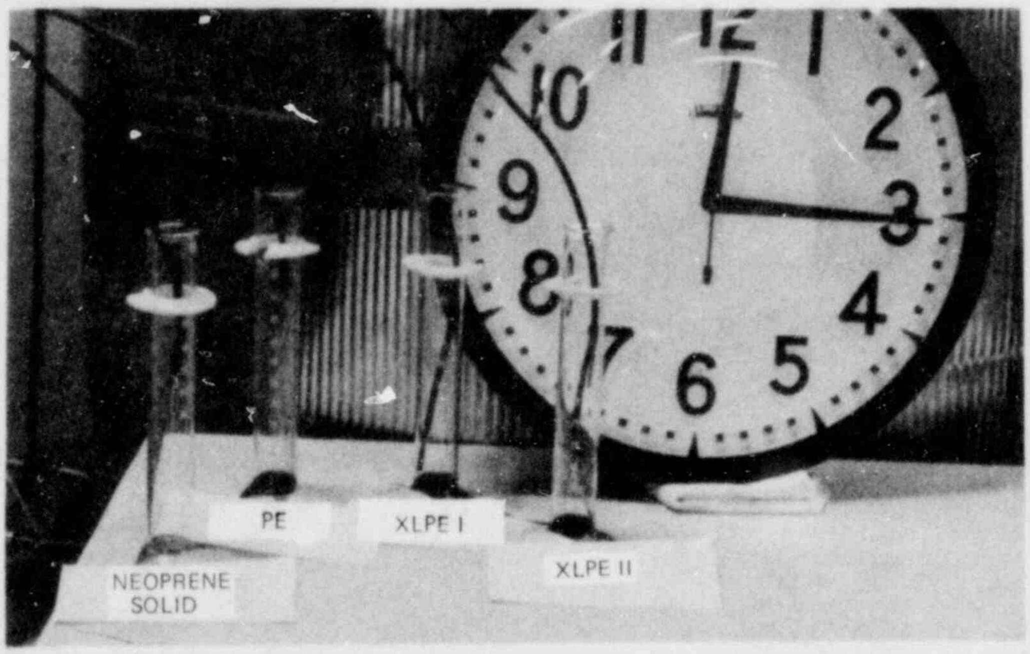

Figure 2.9. Test II at 15 Minutes

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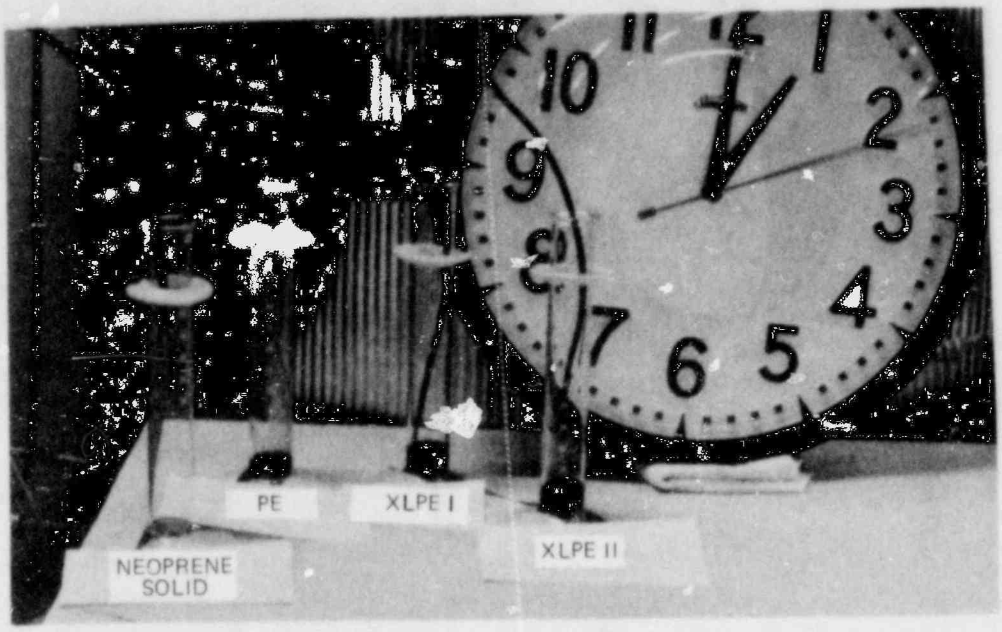

Figure 2.10. Test II at 1 Hour

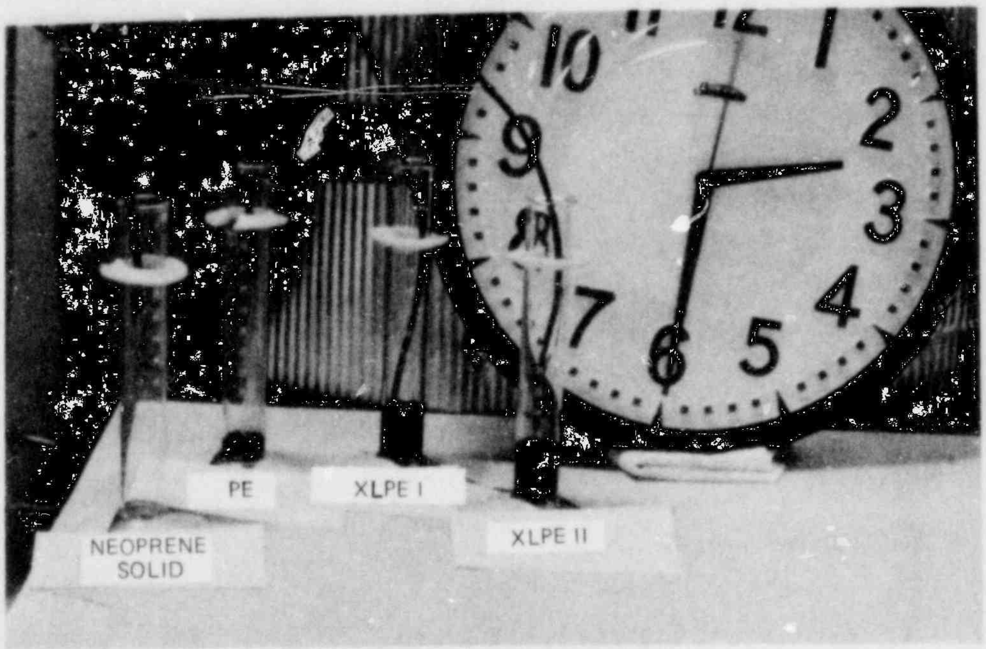

Figure 2.11. Test II at 2-1/2 Hours

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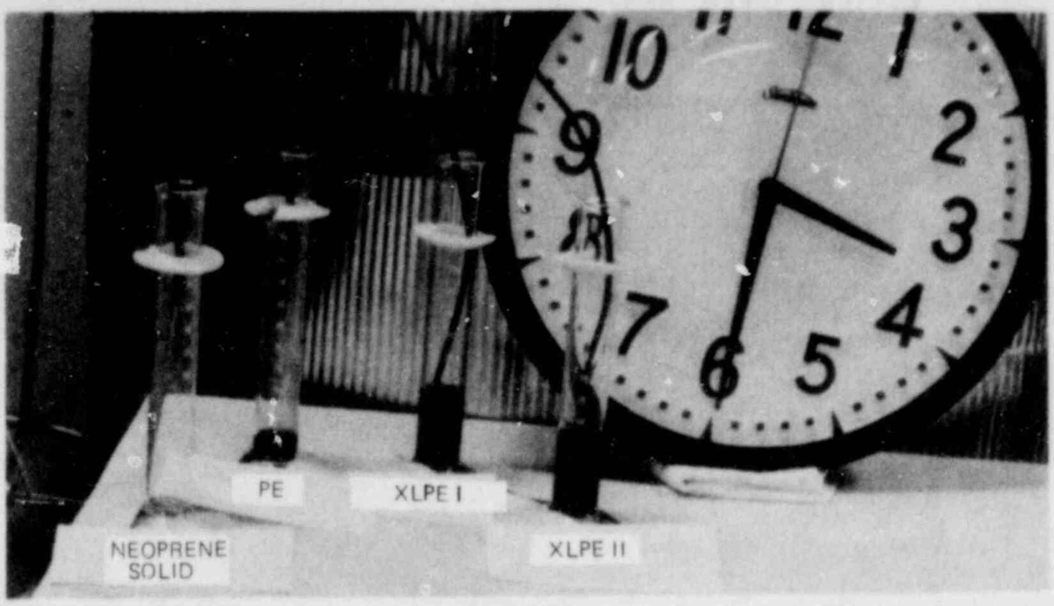

Figure 2.12. Test II at 3-1/2 Hours

One other test deserves mention. Using the rubber-stopper/Swagelock seal system of Test II, about 13 feet of various cables were attached to an air pressure system. The free cable end was submerged in several inches of water. Air pressure was then slowly increased until air leakage was observed by bubbling of the water. All stranded-conductor cable leaked air with 10-25 psig; the highest pressure was necessary for one type of stranded-conductor cable to which the insulation had been "molded". The neoprene solid-conductor cable was pressurized to 70 psig for almost 16 hr with no observed leakage.

It can be concluded from these leakage results that:

- Stranded-conductor cable will leak under very minimal pressure.
- The primary leak path is through the strands of the conductor, not between the conductor/insulation interface, as evidenced by the solid-conductor cable response.

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2.3 References

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

2.2 D. V. Paulson and S. P. Carfagno, "A Review of Class 1E Qualification Data," FIRL Final Report F-C4598-1, prepared for Sandia Laboratories, June 1977. (Incorporated into SAND78-0067.)

2.3 L. L. Bonzon, K. T. Gillen, L. H. Jones, and E. A. Salazar, "Qualification Testing Evaluation Program, Quarterly Report, April-June, 1978," SAND78-1452, NUREG/CR-401, Sandia Laboratories, Albuquerque, NM, November 1978.

2.4 Letter dated July 28, 1978, R. J. Gibson (FIRL) to L. L. Bonzon (SLA), "FIRL Comments and Review of High Intensity Adjustable Cobalt Array (HIACA)."

2.5 Letter dated July 28, 1978, R. J. Gibson (FIRL) to L. L. Bonzon (SLA), "FIRL Comments and Review of Autoclaves and Ancillary Equipment."

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

2.7 Letter dated May 22, 1978, D. G. McDonald (DOR/NRC) to R. Feit (RSR/NRC), "Connectors Recommended to be Subjected to Verification Testing in Accordance with the Commission Directive."

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CHAPTER 3.

Radiation Qualification Source Evaluation

The various technical activities under Task 2 are generally discussed in Section 3.1; the program activities were discussed in Section 1.4. Section 3.2 details a specific program aspect that engendered particular effort and achieved a significant milestone during the reported quarter.

3.1 Task 2 - Technical Activities Summary

IRT Subcontract -- Based on previous subcontract experience and the developed calculational techniques, a second contract was awarded to IRT Corporation in late March. The subcontract is in two parts: Part I, in four phases, is a preparation of a "best-estimate" LOCA radiation signature, dose and rate calculations for a generic containment structure, and calculations of selected Class 1E equipment response to the signature. Part II requests IRT to provide additional assistance to supplement, expand, update, and reevaluate prior submittals as deemed necessary. The "best-estimate" signature will be generally based on the accident-time-release sequencing as specified in WASH-1400[3.1] and other recent literature. (See also Reference 3.2.)

The principal effort this quarter was devoted to completion of a previous draft report[3.3] on simulator adequacy calculations (see also Reference 3.4), completion of the Phase 1 "best-estimate" report[3.5] and work on the balance of the phases, and completion of work and a report[3.6] on radiation damage mechanisms in electric cable.

The "best-estimate" Phase 1 LOCA-signature report was transmitted in draft form on July 6 and in final form on August 2. The objective of the overall study is to formulate a "best-estimate" of fission product releases resulting from a LOCA, based on the best available data; the source may then be compared with the Regulatory Guide[3.7] sources. In the Phase 1 report, the accident scenario and time sequencing and the fission

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product release data are discussed and selections made for use in the balance of the study. Figure 3.1 summarizes the time sequencing: gap release is assumed linear in time from 30 to 150 s following LOCA initiation; meltdown releases are assumed to occur at a constant rate starting at 1 min with completion in 30 min; vaporization releases are assumed to begin 60 min after meltdown, to be complete 3.5 hr later, with exponential release and a 30-min characteristic release half-time; the oxidation release could potentially occur any time after appreciable amounts of core melting and its placement will be a parameter of the calculation.

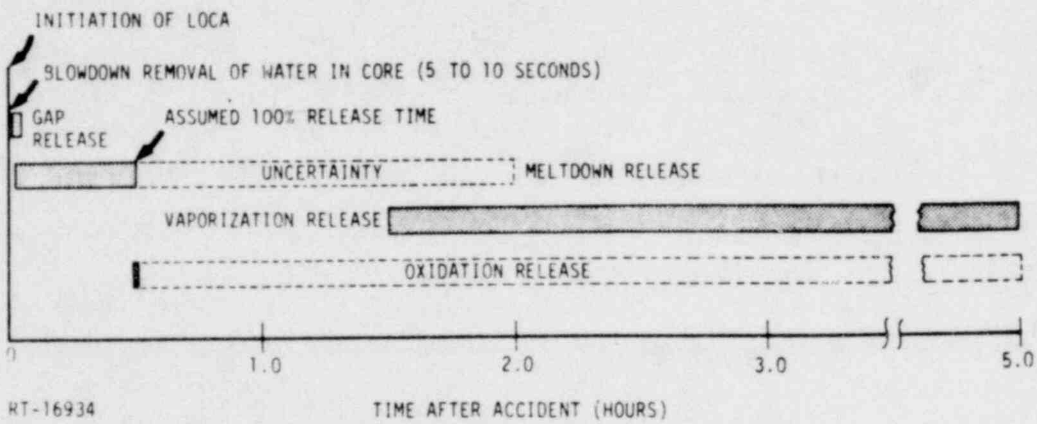

Figure 3.1 Time Sequence of Fission Product Release for an Unterminated LOCA

A subcontract performance review was held in Albuquerque with IRT staff on July 19. That discussion included the best-estimate signature study, the cable damage mechanisms study, and the Sandia comments to IRT 8167-010.[3.3] Target completion dates were decided.

By letter dated June 26, IRT was requested to perform a LOCA-radiation damage assessment of electrical cables and an assessment of radiation simulator adequacy, to be based on the previous work under IRT subcontracts. This additional work was to be limited in scope and to be completed by mid-August. The draft report was received on August 28; the

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final report[3.6] was received on September 8. The report is intended as support information for the evaluation of radiation simulator adequacy being conducted by Sandia personnel. The abstract and conclusions section of Reference 3.6 is reproduced below; see also Section 3.2.

"This analysis of the response of a reactor power cable to a loss-of-coolant accident (LOCA) discusses several different possible failure mechanisms.

"In the first few fractions of a minute the electrical leakage currents flow and trapped electrons may discharge, inducing noise pulses into the cable. Quantitative analysis shows these effects are not serious for a power cable.

"If the cable has poor heat contact with the external environment, within a few minutes radiation energy (dose) will raise the temperature from an already elevated temperature to approach the maximum service temperature for the insulators.

"After a few days the accumulated dose is enough to deteriorate the cable insulation. This chemical and mechanical deterioration thus represents the ultimate failure mode of the cable.

"It is noted that gamma ray simulators may understress the electrical effects and overstress the temperature effects. It is also noted that several parameters used in this study were extrapolated from the values for roughly similar materials in roughly similar environments. A final, definitive study might have to include a program to measure these parameters in the materials of interest."

<u>Evaluation of Radiation Simulator Adequacy, A Status Report</u> -- Sandia personnel are pursuing the question of simulator adequacy, and a number of supporting studies were completed in this fiscal year.

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The preliminary basis for the adequacy task is documented in several reports issued this year:

- L. L. Bonzon, "Radiation Signature Following the Hypothesized LOCA," SAND76-0740, NUREG76-6521, Sandia Laboratories, Albuquerque, NM, Revised October 1977.

- L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source," SAND78-0090, Sandia Laboratories, Albuquerque, NM, February 1978.

- L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions," SAND78-0091, Sandia Laboratories, Albuquerque, NM, May 1978.

- N. A. Lurie, "Evaluation of Test Sources for Radiation Components Qualification," IRT 8167-010 Draft, prepared for Sandia Laboratories, March 1978. (This report intended for internal use only.)

Even more specifically, during this quarter two independent studies using these reports as a basis were made to specifically address any, and all, damage mechanisms from LOCA and simulator radiation sources and to evaluate those mechanisms dependent upon, or peculiar to, the type of source:

- 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, prepared for Sandia Laboratories, August 1978. (This report intended for internal use only.)

- "Co-60 Simulation of LOCA Exposure," Sandia Laboratories internal memorandum dated August 21, 1978, L. A. Harrah to L. L. Bonzon.

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Generally these latter two reports did not uncover any major concerns about simulator adequacy, but a thorough review and analysis will be required before a Sandia consensus is available.

3.2 Evaluation of Radiation Damage Mechanisms in Safety-Related Electric Cable[3.6]

To support the evaluation of radiation simulator adequacy in producing equivalent damage to Class 1 safety-related equipment (as compared to the LOCA-radiation signature), IRT Corporation prepared the Reference 3.6 study on radiation damage mechanisms in a typical power cable. It represents but one part in a larger task on simulator adequacy to be completed by Sandia personnel. The report is presented in its entirety as an Appendix to this quarterly report because of its potential interest to the general readership.

For a general perspective on the report and its conclusions and for the casual reader, the section entitled "Summary and Conclusions," is reproduced here:

"The most serious problems that will occur with the cable are chemical and mechanical deterioration which are expected to occur after a few hours to a day in LOCA environment. Impact strength and elongation changes by 50 percent within ten hours. In order to properly evaluate these effects the expected vibrational and bending stresses should be included in the specification of the LOCA environment.

"The temperature rise in the insulation layers can approach the recommended service limit of Hypalon because of the energy deposited in the Hypalon and in the copper by the radiation field.

"It is not expected that leakage resistance will be less than a megohm. Whether this impedance is a problem or not depends on what sort of circuits the cable connects. Voltage noise spikes will probably be of short duration. The amplitude of these spikes depends on the cable load impedance.

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"The gradient of dose rate into the cable, which is not simulated by standard gamma sources, has two consequences. The first is to allow the radiation to deposit charge into the insulator because the back and forward flow of secondary emission is not equal. The main role of this trapped charge is to create voltage noise spikes. These spikes may not be reproduced by a simulator. The second consequence of the $\dot{\gamma}$ gradient is that more energy is deposited in the interior by the nonattenuated simulator than is deposited by the LOCA radiation field. The simulator may exaggerate the interior temperature compared with the actual situation.

"Several material parameters such as electrical conductivity and the radiation G values had to be interpolated from similar materials or from adjacent temperatures. If more accurate estimates than are presented here are ever needed, then a measurement program would be required."

3.3 References

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

3.2 L. L. Bonzon, K. T. Gillen, and F. V. Thome, "Qualification Testing Evaluation, Quarterly Report, January-March, 1978," SAND78-0799, NUREG/CR-0276, Sandia Laboratories, Albuquerque, NM, August 1978.

3.3 N. A. Lurie, "Calculations to Support Radiation Simulator Adequacy Assessments for Class 1 Equipment," Draft IRT 8167-010, April 1978. (Draft report for Sandia review and comment.)

3.4 L. L. Bonzon, K. T. Gillen, L. H. Jones, and E. A. Salazar, "Qualification Testing Evaluation Program, Quarterly Report, April-June, 1978," SAND78-1452, NUREG/CR-401, Sandia Laboratories, Albuquerque, NM, November 1978.

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. (This report intended for internal Sandia use only.)

3.6 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, prepared for Sandia Laboratories, August 1978. (This report intended for internal Sandia use only.)

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

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CHAPTER 4.

Accelerated Aging Study

The activities under Task 3 were numerous and diverse. The programmatic activities were discussed in Section 1.4; Section 4.1 highlights the various technical activities. 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 in Summary

Fire-Retardant Aging Program -- The study to investigate the effects of aging on fire retardants in electric cable is progressing. An initial round of combustion tests was carried out at the Smithers Laboratory facility on samples of EPR and Hypalon formulations that were in an unaged condition. Two formulations were tested for each type of cabling material, one containing the full range of additives and fillers excepting fire retardants, the other with the same formulation but including an Sb_2O_3-halocarbon flame retardant. Oven aging of samples was started; experimental preparation for radiation aging and combined radiation/temperature aging tests has been started, and these will be underway in the near future. Combustion data on the unaged samples will be used for comparison with data on samples having varying degrees of accelerated aging. Chemical-analytical data on flame retardant retention will also be obtained on the aged samples as they become available, using techniques which will include mass spectroscopy.

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, temperaure, and humidity. In late September, this subcontract was extended through FY79 to continue to provide a test facility to do complementary experiments to those at the Sandia (LICA) irradiation facility.

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Embrittled Polyethylene Cable Evaluation -- Accelerated aging experiments on cables obtained from the Savannah River Plant reactor building had been essentially completed during the previous quarter. The experiments were designed to determine if deterioration observed in the polyethylene insulation of cables was primarily due to thermal environments or if extremely large synergistic effects of combined radiation and temperature environments might have been a major factor in their deterioration. Polyvinyl chloride-jacketed, polyethylene-insulated cable, which had not been exposed to the reactor environment, was exposed to single, sequential, and combined thermal and/or radiation environments. During this quarter, the results were evaluated and their analyses indicate that strong synergistic effects exist both for the polyethylene insulation and for the polyvinyl chloride jacket. A full report on this activity is to be issued during FY79.

Computer Calculations on Polyethylene Radiation Degradation -- Minor work has continued on a computer program to model the structural changes that occur in polyethylene exposed to high-energy radiation under a variety of radiation and temperature conditions. The program is based on kinetic parameters for free radical mediated chemical reactions which have been taken from the literature. Computed results on yields of hydrogen evolution and crosslinking, including the effects of temperature, have been obtained. Some dose-rate effects data have also been generated. Further calculations on these and other related phenomena are planned. Parameters for inclusion of molecular oxygen in the reactions and for allowance of a wider range of initial structural features in the unirradiated polymer are being examined for inclusion in the computations.

4.2 Description of a Method for Combined Environments Accelerated Aging*

Since one requirement for the qualification of safety-related components for the nuclear power industry is that consideration be given

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*This section prepared by K. Gillen, Division 5813.

to aging of the component prior to any tests in a simulated accident environment, there is considerable interest in accelerated aging techniques. One of the more difficult tasks of accelerated aging occurs when attempting to simulate the ambient deterioration of a component which degrades due to a combination of two or more environmental stresses. This situation holds for many components in nuclear power plants; in particular, components inside containment may exist in significant radiation, thermal, and humidity environments. Synergism is potentially important in combined environment situations so that the deteriorating effects of the various environments may not be additive. These problems can greatly complicate accelerated aging.

One accomplishment of the accelerated aging program at Sandia was the development of a method potentially applicable to combined environment aging when synergism is important.[4.1-4.3] The purpose of the following sections is to describe the method, indicate how it can be used to carry out accelerated aging in combined environment situations, and, using literature data and data generated in the program, show how the assumptions of the method can be tested.

4.2.1 Single Environment Accelerated Aging -- A brief review of some of the principles of single environment accelerated aging is necessary before describing the combined environment method. For both single and combined environments, the constant overstress technique of accelerated aging is used. In this technique, the environmental variable or variables of interest are raised above ambient values to a constant level and the aging is followed with time. Figure 4.1 summarizes the method by which single environment accelerated aging is normally carried out. At each of the constant overstress states, S_i, S_j, S_k, ... , the normalized degradation of a material or component, D_s, is followed with time. D_s could represent, for example, the fraction of good components remaining or the fraction of some material variable remaining. In many instances, constant acceleration factors will relate the decay in one overstress state to the decay in a second. When this occurs, the decays at the various overstress conditions

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will be superimposable by horizontal shifts on the log time axis (see Figure 4.1). Where S is a thermal stress, this corresponds to so-called time-temperature super-position used in numerous polymeric systems. The functional relationship between time and stress will define the accelerating function, $A_s(S)$. If the functional form of $A_s(S)$ can be extrapolated to the use condition, Su, degradation predictions can be made at the use environment as shown in Figure 4.1.

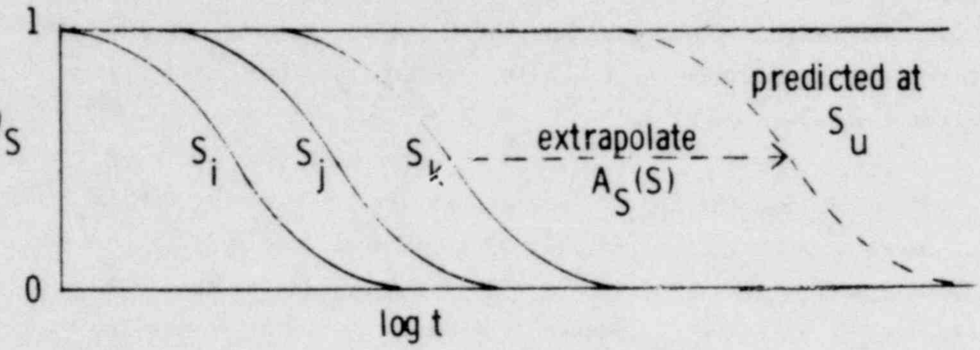

Figure 4.1 Conventional Accelerated Aging. Plot of normalized degradation D_s versus log time at three constant stress levels and extrapolation to use conditions.

4.2.2 <u>Combined Environments Accelerated Aging Method</u> -- Suppose the simulation of the ambient aging of a component in a combined stress environment is required. For a nuclear power plant component, the ambient (i.e., use) environment might include, for example, a temperature, T_u, and a radiation dose rate, R_u. To accelerate the combined environment degradation by a factor X, the proposed method suggests the use of a temperature, T_X, which accelerates the thermal degradation by the factor X, simultaneously with a radiation dose rate, R_X, which accelerates the radiation degradation by the same factor X. The appropriate values of T_X and R_X are obtained from knowledge of the single environment acceleration functions, $A_T(T)$ and $A_R(R)$, which are obtained from single environment aging studies. This relatively simple concept is similar to the idea used by Paloneimi[4.4] for accelerating all thermal reactions equally through control of the gaseous environment.

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It is next assumed that any synergistic reactions are accelerated or scaled by the same factor X; this is the key assumption underlying the approach. The set (T_X, R_X) is called a matched set to the use conditions (T_u, R_u). By carrying out combined environment aging under a number of matched set conditions, the validity of the scaling assumption can be ascertained. For example, Figure 4.2 shows the results of two hypothetical matched set experiments carried out by using acceleration factors of X = 20 and X = 80. The combined environment degradation, D_c, is plotted versus log time with the solid curves representing the accelerated degradations. By multiplying the times in the X = 20 and X = 80 accelerated degradations by factors of 20 and 80, respectively, predictions under the use conditions will result. If these and similar extrapolated predictions from other matched set experiments agree, then good evidence exists for the validity of the scaling assumption. In other words, the method predicts consistency of matched set conditions. This implies that when the method is appropriate, the time dependent degradation of a material carried out under a given set of combined stress variables can be used to predict the time dependent degradation of that material under any set of stress conditions which is a matched set to the experimental conditions.

Experimental data will be used in the next section to show how the consistency of matched set conditions can be conveniently checked in order to verify the scaling assumption.

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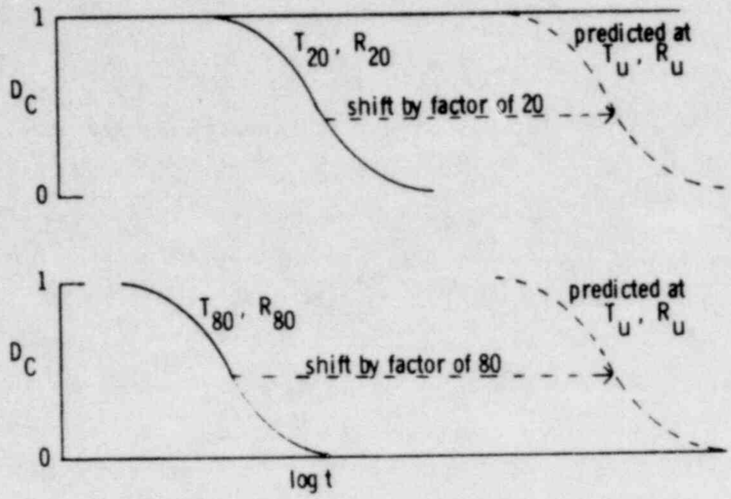

Figure 4.2 Combined Environments Accelerated Aging Approach. Normalized degradation D_c vs log time using 20 times (above) and 80 times (below) matched set conditions to use conditions.

4.2.3 Analyses of Cable Aging Data -- An extensive experimental aging program is currently underway on cable jacketing and insulation materials stripped from low-voltage electrical cables which are used for safety applications in nuclear power plants.[4.1,4.3,4.5] Aging is being carried out in single and combined radiation and thermal environments, with the degradation monitored by following the ultimate tensile elongation versus time. Results (Figure 4.3) for thermal aging of a chloroprene jacketing material at temperatures ranging from 363 to 413°K indicate that constant acceleration factors relate the decays under various constant temperature conditions and that the thermal acceleration function, $A_T(T)$, has an Arrhenius form with an activation energy of 21 kcal/mole (Figure 4.4). In addition, as shown in Figure 4.5, room temperature radiation aging results for the chloroprene material indicate that the degradation depends only on the integrated radiation dose. The radiation acceleration function, $A_R(R)$, is, therefore, proportional to the radiation dose rate. Combined environment (temperature simulataneous with radiation) experiments are being carried out in Sandia's radiation facility,[4.6] as well as at the Naval Research Laboratories' facility.[4.7] Some typical results for the chloroprene material in a combined environment comprising 361°K and 95

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krad/hr are shown in Figure 4.6. The filled circles represent the experimental fractions of ultimate tensile elongation remaining versus time for the combined 361°K plus 95 krad/hr environment. The solid curves, marked D_T and D_R, represent the fractions of elongation remaining versus time for the single thermal environment of 361°K and the single radiat..n en..ironment (low temperature) of 95 krad/hr, respectively. The dash.. line approximates the expected decay in the combined environment in the absence of synergistic effects. Comparison of this result with the experimental combined environmental data indicates that synergistic effects are found for the chloroprene material.

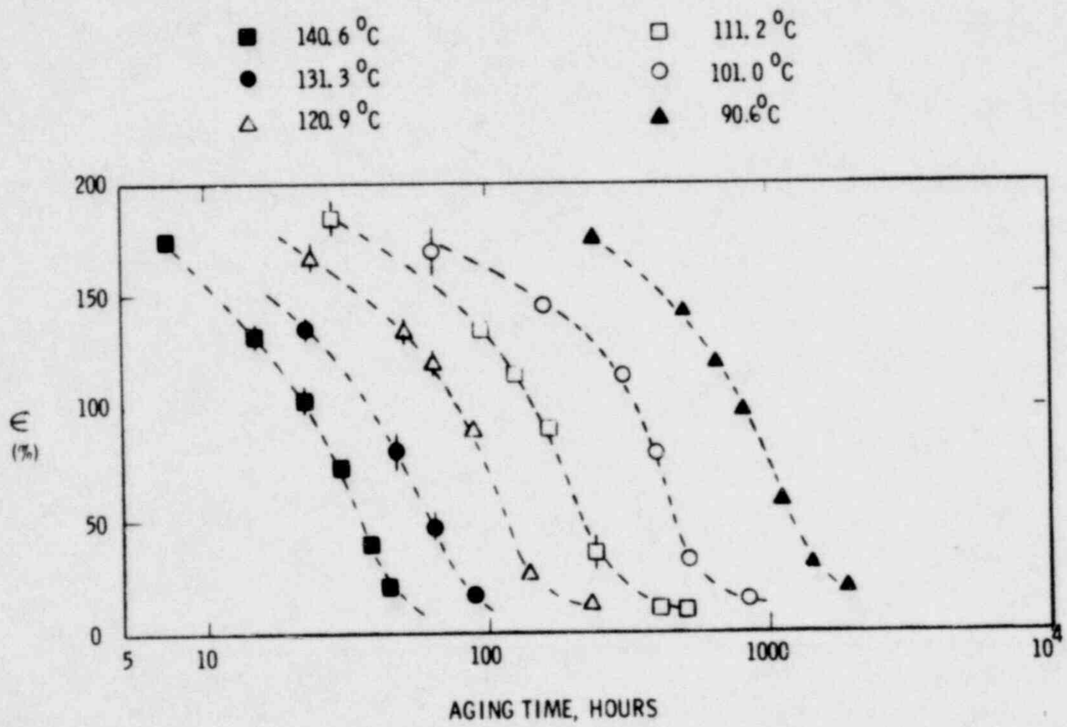

Figure 4.3 Ultimate Tensile Elongation (e) vs Aging Time for Chloroprene Material at the Six Indicated Temperatures

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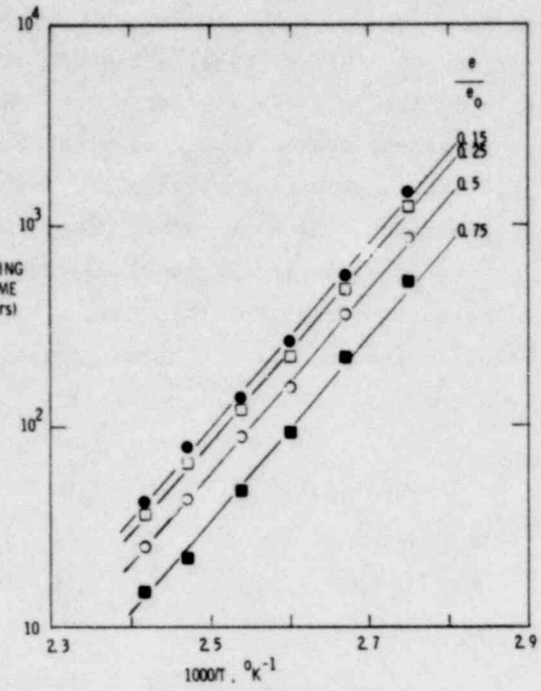

Figure 4.4 Arrhenius Plot of Chloroprene Data at the Four Values of e/e_o Indicated

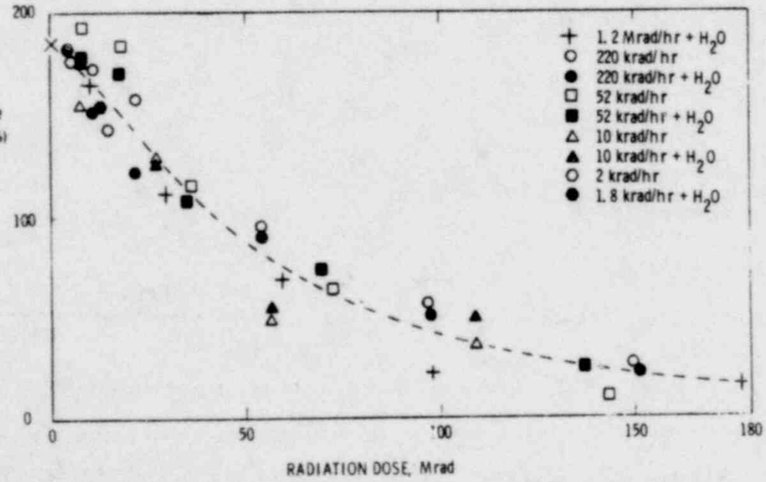

Figure 4.5 Ultimate Tensile Elongation (e) vs Total Radiation Dose for Chloroprene Material at Various Dose Rates Under Room Temperature Dry Air and 70% Relative Humidity (H_2O) Conditions, As Indicated in the Figure.

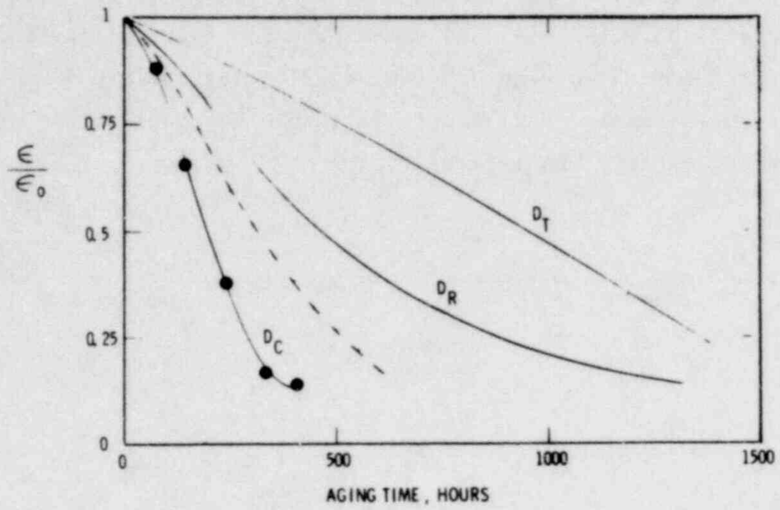

Figure 4.6 Aging of Chloroprene: D_T, 361°K; D_R, 95 krad/hr; D_C, 361°K combined with 95 krad/hr; dashed curve, prediction without synergism.

For the chloroprene data, Figure 4.7 shows how the single and combined environment studies can be analyzed according to the formalism of the proposed method. Contours of the time required (0.1 yr and 1 yr) for the elongation to decrease an arbitrary amount (to 50% of initial in this figure) are plotted versus the log of the radiation dose rate on one axis and inverse temperature on the other axis. The horizontal portions of the contours are in regions (low temperature) where the radiation environment dominates the degradation. From the room temperature radiation results (Figure 4.5), it takes 0.1 yr for the elongation to decrease to 50% of initial at 55 krad/hr. This determines the horizontal portion of the 0.1 yr curve. Since $A_R(R)$ is proportional to the radiation dose rate, R, the horizontal portion of the 1-yr curve will occur at 5.5 krad/hr. At low radiation dose rates and sufficiently high temperatures, the thermal environments dominate the degradation; thus, the single environment (thermal) results lead to the vertical portions of the contours. For

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example, for Figure 4.4, it takes 0.1 yr at 363°K for the elongation to decrease to 50% of initial. From the result that $A_T(T)$ is Arrhenius with an activation energy of 21 kcal/mole, one can locate the vertical portion of the 1-yr curve at 336°K. To connect the two single-environment-dominated extremes, experiments must be carried out in the region where both radiation and temperature are important.

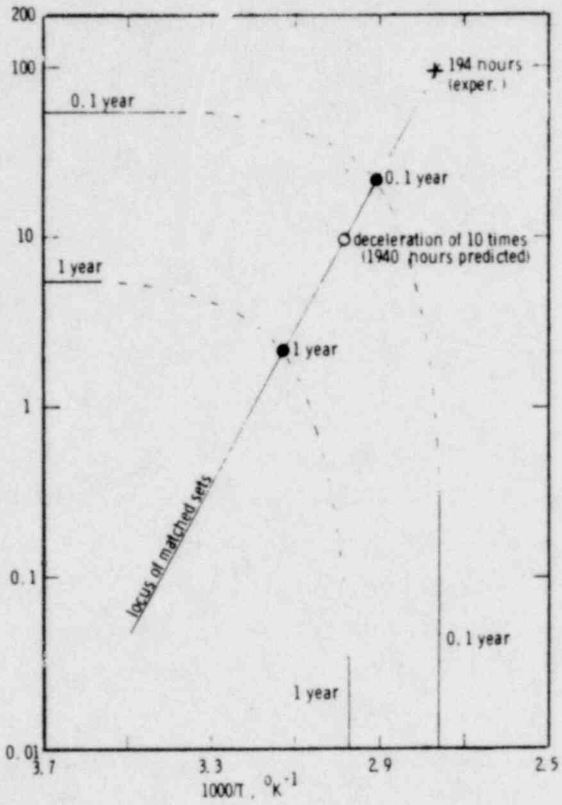

Figure 4.7 Analysis of Chloroprene Aging Data Using Proposed Method. Contours of time required for $e/e_o = 0.5$.

From Figure 4.6 approximately 194 hr are required in the combined environment of 95 krad/hr and 361°K for the elongation to decay to 50% of original. This point is marked on Figure 4.7 with a cross, and the line through it represents the locus of all matched sets (T, R) to the experimental condition of 95 krad/hr and 361°K. It should be noted that in

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this instance the locus of matched sets is a straight line because $A_T(T)$ is Arrhenius (linear on abscissa scale) and $A_R(R)$ is proportional to dose rate (linear on ordinate scale); in the general case where dose rate effects and/or non-Arrhenius behavior exists, the locus of matched sets will not be a straight line. For the chloroprene material, the matched set with a deceleration factor of 10 with respect to the experimental conditions (361°K, 95 krad/hr) is given by (334.7°K, 9.5 krad/hr); this result is obtained from $A_T(T)$ and $A_R(R)$ and its location is denoted by the open circle in Figure 4.7. The proposed method predicts that the time corresponding to this point would be 10 times 194 hr, or 1940 hr. In the same way, predictions (solid circles) corresponding to 0.1 yr, 1 yr, or any other specified time, can be determined. By performing other combined environment experiments under conditions in which both environments are important, the remainder of the time contours (dashed curves) can be generated and the method assumptions checked. For instance, Figure 4.8 summarizes the results of 11 different combined environment experiments run on the chloroprene material, where again the time contours for decay to 50% of original elongation are plotted. The locations of the crosses and their corresponding number denote the combined environment conditions and the experimental times required for the elongation to decrease to 50% of original. The circles are the 0.1-yr points predicted by using the approach outlined in Figure 4.7.

The large number of experiments scattered throughout the region where both environments are important allow the complete contour to be constructed. The temperature and radiation conditions for nine of these experiments were chosen such that three groups of approximately matched set experiments were carried out, each group comprising three matched set conditions which are connected by dashed lines in Figure 4.8. The superposition within experimental uncertainty (10%) of the 0.1-yr predictions (filled circles) derived from the matched set conditions verifies the scaling assumption underlying the proposed method. As indicated earlier, this verification implies that experimental results under a set of specified radiation and temperature conditions can be successfully used to predict results under conditions which are a matched set to the original conditions. It should be emphasized, however, that in general, it is not necessary or practical to carry out exact matched set experiments.

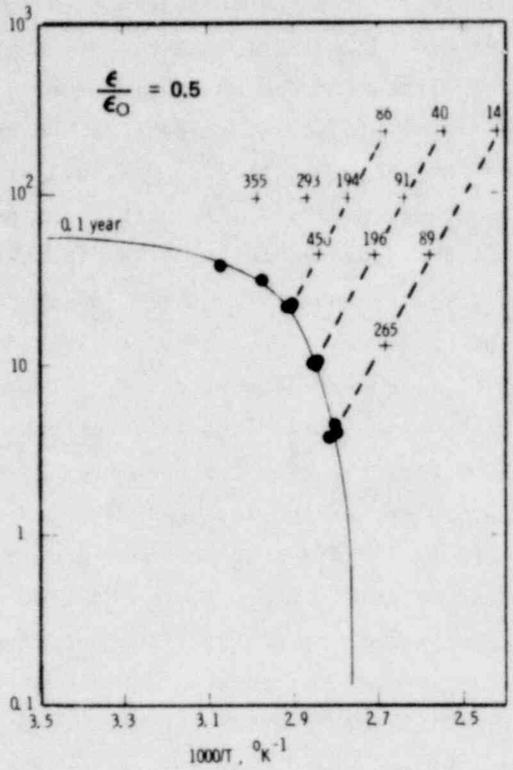

Figure 4.8 Analysis of Chloroprene Data Using Proposed Method. The crosses represent the experimental radiation and temperature coordinates. The numbers above them represent the time (hrs) required for e/e_o to reach 0.5. The filled circles are the prediction for the 0.1 year contour using the approach outlined in Figure 4.7. The crosses connected by dashed lines are approximately matched sets to each other.

It is only necessary to perform a number of combined environment experiments at enough properly chosen conditions to generate the contours and at enough different accelerations to verify that the predicted contours remain unchanged as the aging conditions become less severe. It should also be noted that Figure 4.8 actually represents a slice through a three-dimensional plot at a constant value of e/e_o (0.5). Similar figures can be constructed for other values of e/e_o; in the case of the chloroprene material, analysis of data at other values of e/e_o also confirms the

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scaling assumption. Once contours similar to those in Figures 4.7 and 4.8 have been constructed and confirmed for a given material, they can be used to predict damage to the material under various radiation and temperature conditions. In addition, the matched set approach can be used to accelerate any ambient aging conditions by a chosen factor.

A second test of the proposed method utilizes literature data[4.8] on the thermoradiation sterilization of a biological material. For this system, decay in the various environments was always first order, so a contour of the predicted first order rate constant ($k = 0.1\ hr^{-1}$) is plotted versus radiation and temperature in Figure 4.9. The experimental combined environment points are denoted by crosses, and the points predicted for $k = 0.1\ hr^{-1}$ (again using the approach outlined in Figure 4.7) are denoted by solid circles. Synergism of radiation and temperature was important for this biological system; the self-consistency of the predicted [illegible]s offers convincing evidence for the validity of the proposed method. As pointed out earlier, exact matched set experiments are neither practical nor necessary to confirm the scaling assumptions when the data is analyzed in the above manner.

4.2.4 Summary and Conclusions -- A method for carrying out accelerated aging in combined environment situations is discussed. Data on single and combined environmental aging of chloroprene are presented. The results for this material indicate that synergistic effects exist for combined radiation and temperature environments. The chloroprene data and literature data for a biological material are analyzed using the proposed method; the results offer convincing evidence in support of this approach.

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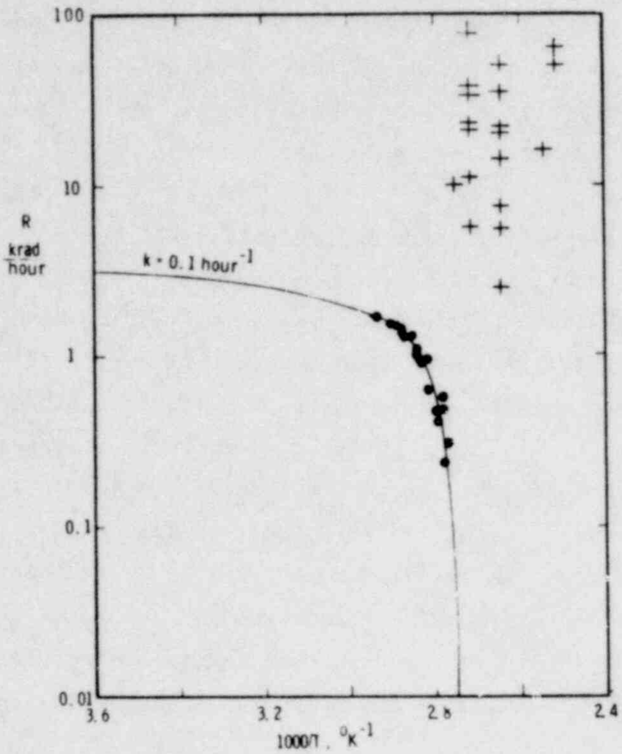

Figure 4.9 Analysis of Literature Data Using the Proposed Combined Environment Accelerated Aging Method. The crosses represent the experimental radiation and temperature coordinates. The filled circles are the predictions for $k = 0.1\ \text{hour}^{-1}$ using the approach outlined in Figure 4.7

4.3 References

4.1 K. T. Gillen, E. A. Salazar, and C. W. Frank, "Proposed Research on Class 1 Components to Test a General Approach to Accelerated Aging Under Combined Stress Environments," SAND76-0715, April 1977.

4.2 K. T. Gillen, "Accelerated Aging in Combined Stress Environments," Proceedings of Conference on Environmental Degradation of Engineering Materials, VPI Press, 1977.

4.3 K. T. Gillen, "A Method for Combined Environment Accelerated Aging," Transactions of the ENS/ANS International Topical Meeting on Nuclear Power Reactor Safety, Oct. 16-19, 1978, Brussels, in press.

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4.4 P. Paloniemi and P. Lindstrom, 1976 IEEE International Symposium on Electrical Insulation, Montreal, p. 28.

4.5 K. T. Gillen and E. A. Salazar, "Aging of Nuclear Power Plant Safety Cables," Transactions of the ENS/ANS International Topical Meeting on Nuclear Power Reactor Safety, Oct. 16-19, 1978, Brussels, in press.

4.6 L. L. Bonzon, K. T. Gillen, L. H. Jones, and E. A. Salazar, "Qualification Testing Evaluation Program, Light Water Reactor Safety Research, Quarterly Report April-June 1978," SAND78-1452, NUREG/CR-401, Sandia Laboratories, Albuquerque, NM, November 1978.

4.7 F. J. Campbell, IEEE Trans. on Nuclear Science NS-11, 123 (1964).

4.8 M. C. Reynolds and J. P. Brannen, Radiation Preservation of Food (Proc. Symp. Bombay, 1978), IAEA, Vienna (1972) and references therein.

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APPENDIX

IRT 0056-002A

EVALUATION OF RADIATION DAMAGE MECHANISMS IN A REACTOR POWER CABLE IN A LOSS-OF-COOLANT ACCIDENT

Prepared by

J. F. Colwell, B. C. Passenheim, and N. A. Lurie

This report documents part of the Qualification Testing Evaluation (QTE) Program (A 1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

Prepared for

SANDIA LABORATORIES
Albuquerque, New Mexico

Under

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ABSTRACT AND CONCLUSIONS

This analysis of the response of a reactor power cable to a loss-of-coolant accident (LOCA) discusses several different possible failure mechanisms.

In the first few fractions of a minute the electrical leakage currents flow and trapped electrons may discharge, inducing noise pulses into the cable. Quantitative analysis shows these effects are not serious for a power cable.

If the cable has poor heat contact with the external environment, within a few minutes radiation energy (dose) will raise the temperature from an already elevated temperature to approach the maximum service temperature for the insulators.

After a few days the accumulated dose is enough to deteriorate the cable insulation. This chemical and mechanical deterioration thus represents the ultimate failure mode of the cable.

It is noted that gamma ray simulators may understress the electrical effects and overstress the temperature effects. It is also noted that several parameters used in this study were extrapolated from the values for roughly similar materials in roughly similar environments. A final, definitive study might have to include a program to measure these parameters in the materials of interest.

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1. INTRODUCTION

The present work explores the expected failure modes of a power cable subjected to a loss-of-coolant accident (LOCA) environment in a nuclear power plant. The work is a more detailed numerical analysis of many of the effects outlined by Leadon in his preliminary analysis of the problem (Ref 1). An important source of input data for our present work is the energy deposition study carried out by Lurie, et al. (Refs 2,3).

The model for the cable which we use is identical to the one for which Lurie did the irradiation calculation. It is a copper conductor with an ethylene-propylene rubber (EPR) insulator covered by a chloro-sulfonated polyethylene (trade-name Hypalon) jacket. It is shown in Figure 1. The material properties for the various regions of the cable are listed in Table 1.

Table 1. Material Properties of the Cable

	Region Within the Cable		
Parameter, Symbol, Units	Copper	EPR	Hypalon
Outer Radius, r, cm	0.729	0.947	1.1304
Electrical Cond., σ, $(\text{ohm-cm})^{-1}$	5.7×10^{5}	6.4×10^{-10}	1.1×10^{-8}
Dielectric Strength, volt/cm	N/A	1.8×10^{-7}	1.0×10^{-5}
Dielectric Const., k, pure number	N/A	2.3	6.0
Dielectric Relaxation Time, sec	1.6×10^{-19}	2.2×10^{-4}	4.8×10^{-5}
Density, ρ, gm/cm^3	8.94	0.92	1.25
Thermal Cond., κ, watt/cm$^{\circ}$C	4.19	3.47×10^{-3}	1.9×10^{-3}
Tensile Strength, 1000 psi		<1	4.0
Service Temperature, max. $^{\circ}$C for continued use		177	163

In this task it is first necessary to define the environment to which the cable is expected to be subjected and then to calculate how this environment will influence the

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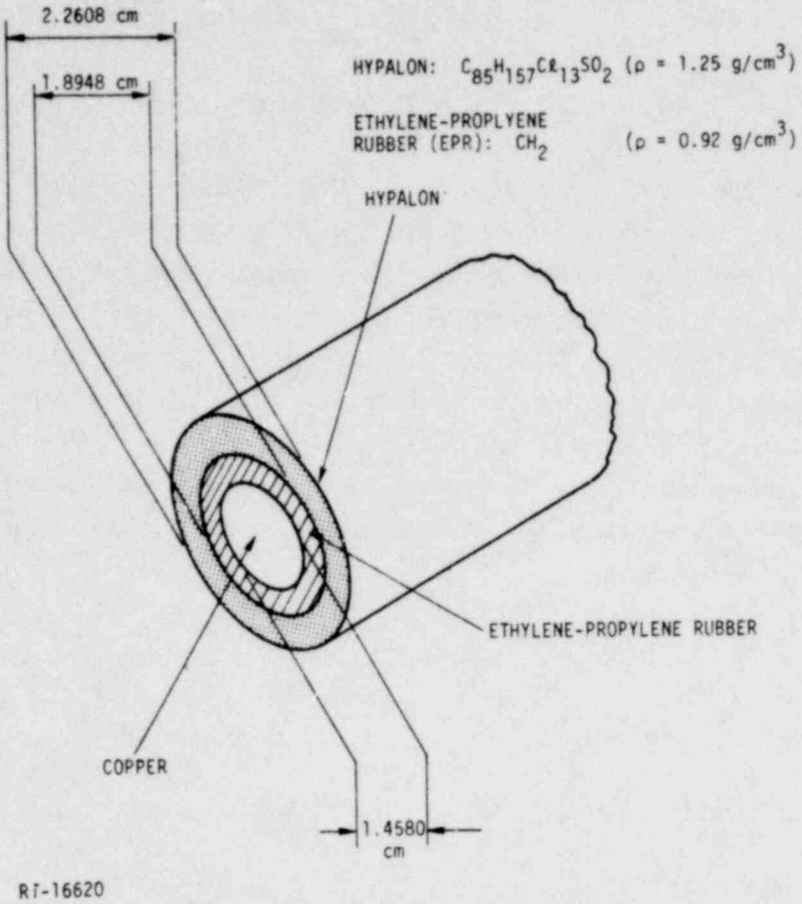

Figure 1. Model of the reactor power cable

operation of the cable. The definition of the assumed LOCA environment will be found in Section 2 of this report. The bulk of Section 2 is devoted to adapting the dose calculations of References 2 and 3 to the specified usage needed here. Section 3 is devoted to a calculation of the changes in electrical conditions. These changes include leakage current through the insulation because of its radiation and temperature enhanced conductivity, induced voltages from trapped charge, and the possibility of breakdown due to fields generated by this charge.

Section 4 briefly discusses temperature conditions. It should be remarked that both the electrical and temperature conditions depend on whether or not the cable is in good enough contact with some external sink to exchange charge and thermal energy respectively. There will undoubtedly be some regions in poor contact with any ground plane where the fields and temperature can rise because there is no electrical or heat current flowing through the jacket. Sections 3 and 4 consider cases of both good and poor ground contact.

The mechanical and chemical deterioration which will occur are treated in Section 5. It turns out that disintegration or temperature and radiation enhanced deformation are the most likely failure modes of the cable.

This report follows the temporal development of effects in the cable after the LOCA environment is established. First the high dose rate will change the electrical conditions in seconds to fractions of a minute. The cable survives this challenge. Within a few minutes the radiation heat input will raise the insulator temperature near or above its normal service limit. The radiation and mechanical deteriorations induced by the radiation become serious only after many hours to a few days when the total accumulated dose has had a chance to build up to high levels.

2. CABLE ENVIRONMENT

For simplicity in discussing the attenuation of energy deposition from the cable edge into the cable interior, we have specialized our analysis to a single case: the airborne source case shown in Figures 8 and 10 of Reference 3. All sources discussed in Reference 3 attenuate in roughly the same manner as the airborne sources. An average over all sources will give substantially the same relative results as found for this one case. We also limit our analysis to a reactor power of 4,000 megawatts. All doses and dose rates are proportional to the reactor power so the final results presented here can be scaled up or down to fit a different particular case.

It is convenient to normalize the depth-dose profile to unity at the cable surface. The normalized doses per source particle are shown in Figure 2 for gamma radiation, and in Figure 3 for beta radiation. To get absolute dose deposition profiles, these energy loss curves must be multiplied by the total number of source particles at the cable surface.

Of course, not all of the fission product radiation impinges on the cable, so that an estimate of the local dose rate is also necessary. This has already been carried out by Bonzon (Ref 4); his results are shown in Figures 4 and 5. Note that these doses are due to combined environments from airborne, plate-out and waterborne sources. Values read from Figures 4 and 5 and converted to rads(Hypalon) are given in Table 2. These surface dose rates are then multiplied by the normalized dose-depth profiles shown in Figures 2 and 3 to obtain the total dose rates within the cable. The results of this procedure is exhibited in Figure 6. The leveling off a the EPR-Hypalon boundary is caused by backscatter into the Hypalon. The very rapid attenuation into the copper is caused by the rapid attenuation of betas, leaving only gammas which penetrate this far. The total gamma deposition is down from the total beta deposition by a factor of ten, so that the removal of betas drops the dose rate by a large factor.

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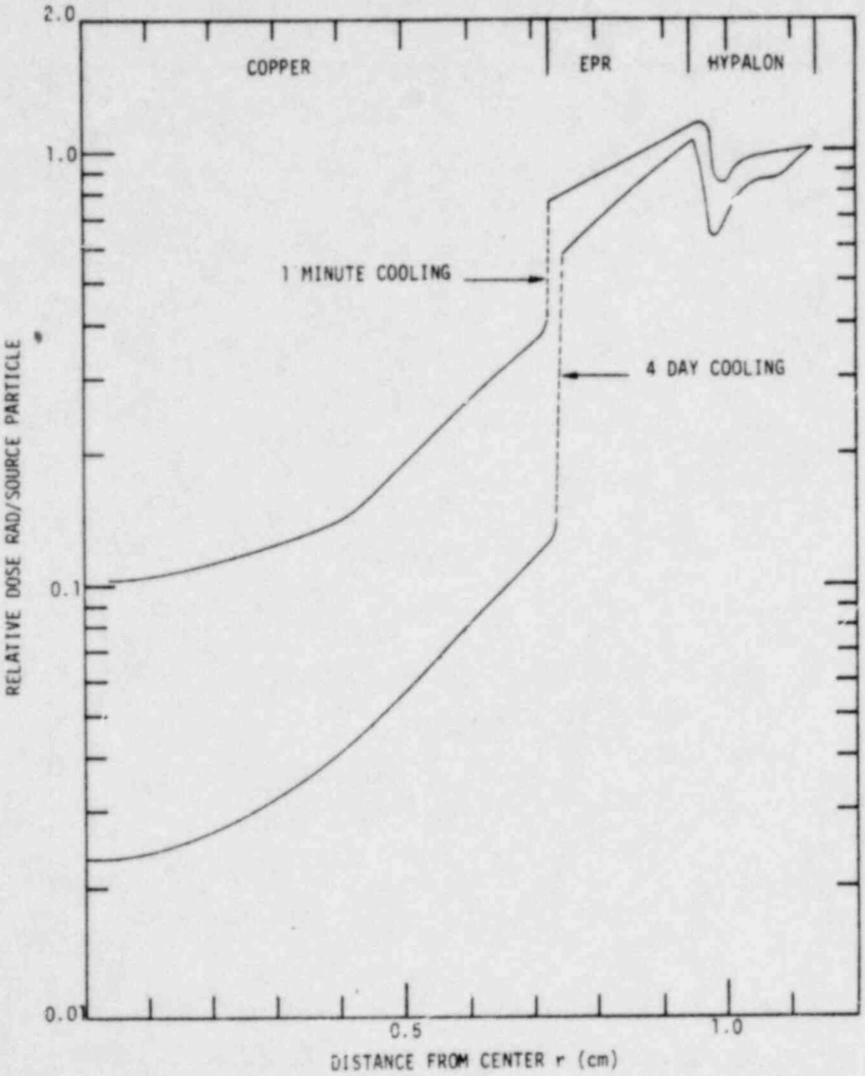

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Figure 2. Normalized dose deposition profiles in a cable by gamma radiation in the LOCA radiation environment

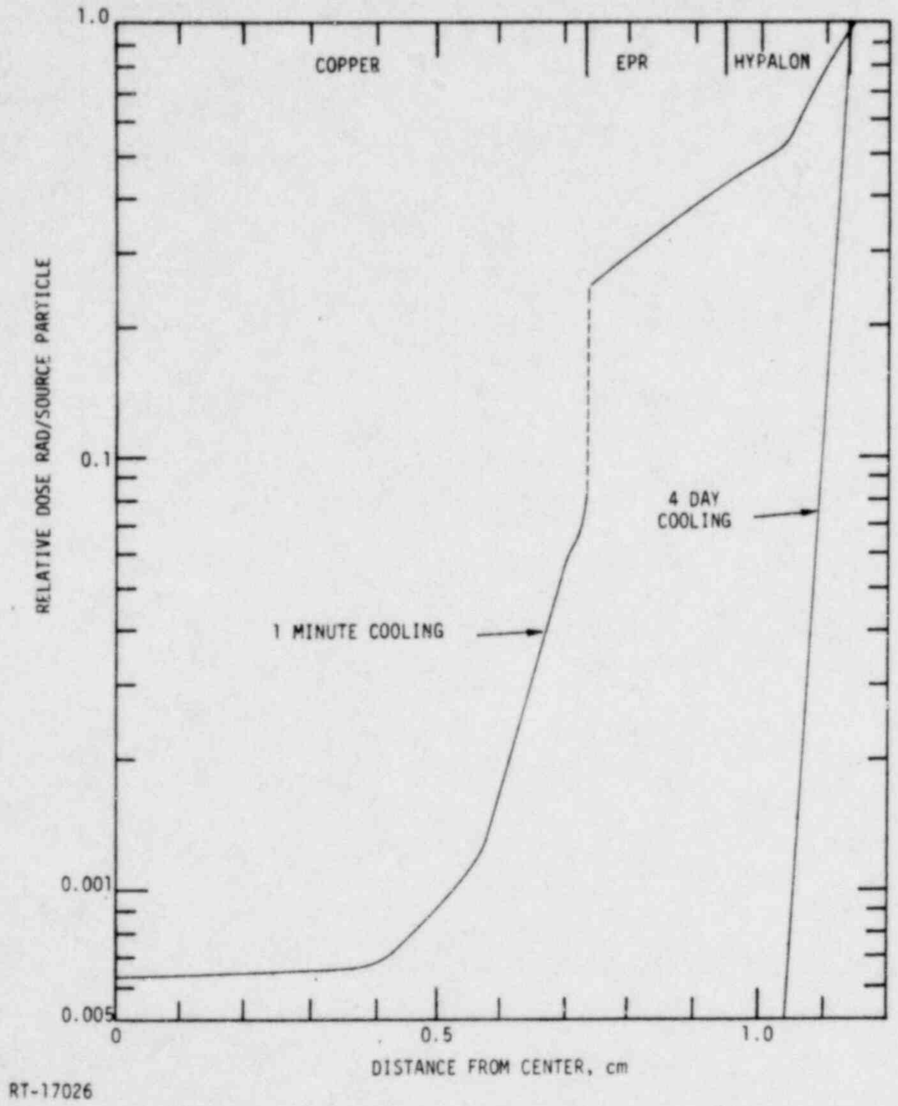

Figure 3. Normalized dose deposition profiles in a cable by beta radiation in a LOCA radiation environment

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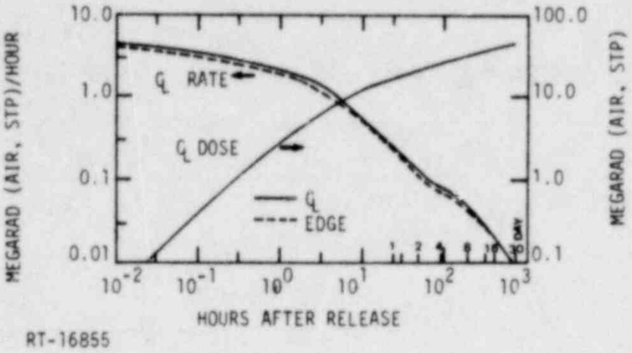

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Figure 4. Typical dose rate and dose to air from the combined gamma sources; airborne source uniformly distributed in the containment volume, plate-out source uniformly distributed on the containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume.

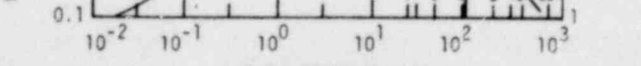

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Figure 5. Typical dose rate and dose to air from the combined beta sources; airborne source uniformly distributed in the containment volume, plate-out source uniformly distributed on the containment sidewall surface, 1/10 of the waterborne source assumed to be uniformly distributed in the containment volume.

Table 2. Radiation Dose Rate in Rad (Hypalon)/Sec at Surface of Cables

Time After Accident	Radiation Source		
	Gamma	Beta	Total
1 minute	1.9×10^2	1.4×10^4	1.5×10^4
4 days	33	310	343

Other important environments include the external temperature which is taken to be steam at 60 psig. This results in a temperature of 143°C, which is rather high but still below the maximum service temperature shown in Table 1. The current expected is about 20 amps and the voltage is presumed to be a nominal 220V rms (i.e., 310V peak). The above data are summarized in Table 3. As a convenience in later discussions, the average dose rate in the different materials, as taken from Figure 6, is appended to Table 3.

Table 3. Worst Case Environment of the Cable

Reactor Power	4,000 Mwatt
External Temperature	140°C Peak
Potential of Inner Conductor	310 Volts Peak
Current Carried by Conductor	20 amps

Average Radiation Dose Rate Within the Given Region in Rad(Material)/Sec

Time After Accident	Copper	EPR	Hypalon
1 minute	585.0	6.4×10^3	1.1×10^4
4 days	1.9	26.0	100.0

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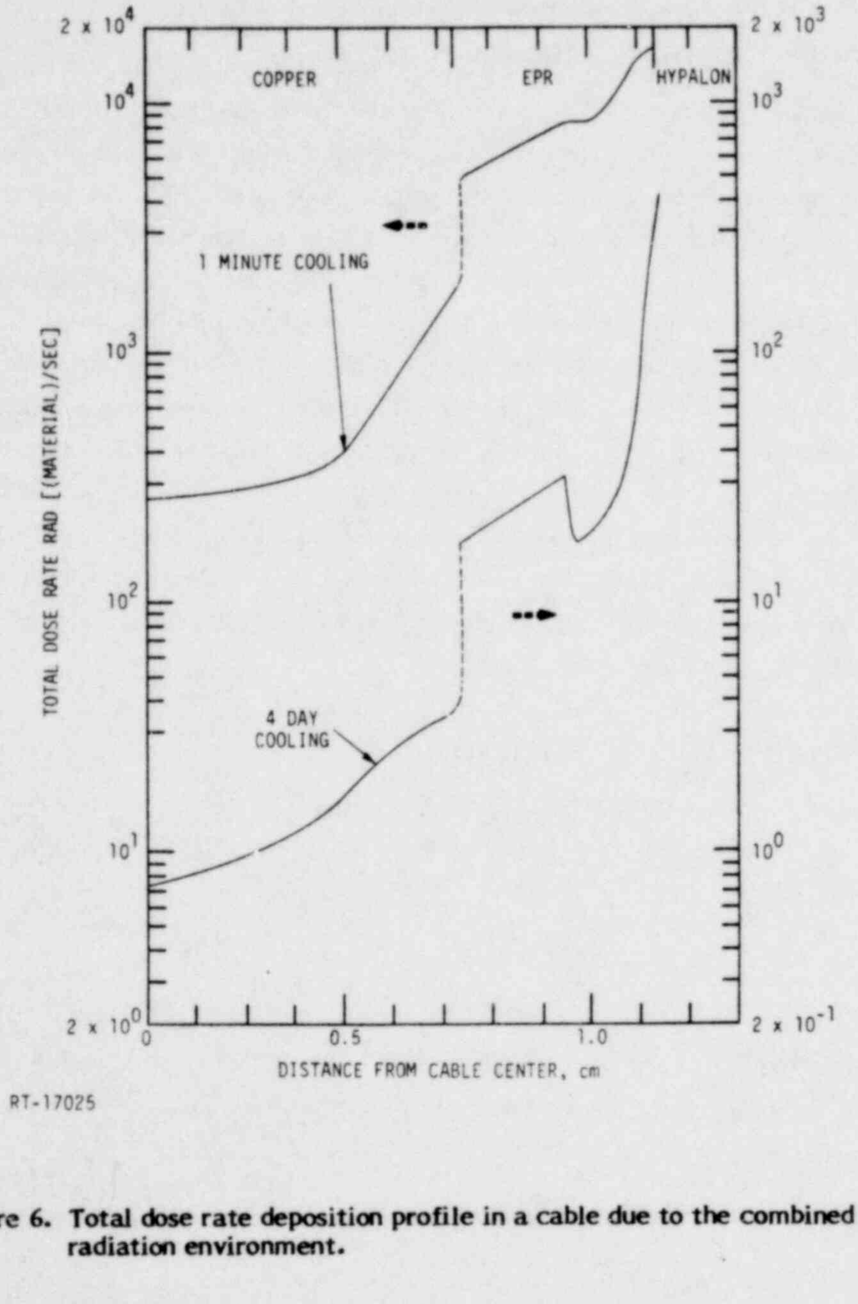

Figure 6. Total dose rate deposition profile in a cable due to the combined LOCA radiation environment.

3. ELECTRICAL CHANGES IN THE CABLE

3.1 SUMMARY OF THE ELECTRICAL PROBLEM

The dose rate, which is established soon after the accident will have several electrical consequences. The radiation induced conductivity, σ, will turn the pure insulator into a shunt resistor (albeit, still a very large one). This effect must be calculated from the value of conductivity of EPR at the radiation level and at the temperature listed in Table 3. Such data for EPR could not be found, but had to be estimated by comparison with data at the correct temperature for similar materials and for data at lower temperature for the same material. This procedure is acceptable since the electrical deviations remain small, although measurements on the material of interest in the environment of interest would be more satisfying. This work is described in subsection 3.2. The leakage current is the product of σ times the electric field, E. The field is derived in subsection 3.3, and σ and E are combined into an effective lumped shunt resistor in subsection 3.4. Noise pulses caused by release of trapped charge are discussed in subsection 3.5. Subsection 3.6 mentions differences between the LOCA radiation and gamma simulators.

3.2 DERIVATION OF THE CONDUCTIVITY

Several workers (Refs 5,7) have determined the electrical conductivity of a plastic insulator in a radiation environment. They all agree that the radiation induces a conductivity of the form

$$\sigma = A\dot{\gamma}^{\delta} \quad , \tag{1}$$

where $\dot{\gamma}$ is the dose rate, A is of order 10^{-15} to 10^{-20} (ohm-cm rad/sec)$^{-1}$ and δ is approximately 1.0. Assuming that δ equals 1 is accurate enough for our purpose.

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The conductivity at an elevated temperature must also be found. Theoretically, the conductivity should be the product of the charge per carrier, e, the mobility, μ, and the numbers of carriers,n.

$$\sigma = n e \mu \quad . \tag{2}$$

The carriers are generated thermally as well as by the radiation

$$\sigma = n e \mu (n_{th} + n\dot{\gamma}) \quad , \tag{3}$$

where $n_{th} \approx n_o \exp(-E/kT)$.

The data of Harrison and Proulx (Ref 6) are plotted in Figure 7 as a function of inverse temperature (1000/T). This plot indicates that A has a temperature independent background plus an exponential variation. This is what would be expected from Equation 3 if μ is not a strong function of temperature. However, an analysis by Adamee and Calderwood (Ref 8) indicates that μ itself is an exponential function of T. Thus, it is safest to assume that the temperature dependence of A is parallel to the temperature dependence of σ. As an aid to this end, data from an NBS Monograph (Ref 9) are also plotted in Figure 5. It appears that at 143°C (1000/T = 2.40), A will be of the order of 10^{-13} for EPR. Hypalon is a better conductor than EPR so that, under the same temperature conditions the parameter for Hypalon (A) will be larger. We take A to be 10^{-12} for Hypalon. The values of σ quoted in Table 1 are found using the above values of A along with the 1 minute values of $\dot{\gamma}$ in the appropriate regions presented in Table 3. In Section 4 of this report, it is concluded that the temperature of the EPR may rise by as much as 15°C which could increase σ by a factor of 4. Nevertheless, as we are about to show, the conductivity does not appear to be large enough to cause any problem.

3.3 DERIVATION OF THE ELECTRIC FIELD

The electric field is the superposition of the field set up by any potential applied to the conductor and a field set up by charges trapped in the insulator. We will show that the field generated by trapped charge is insignificant compared to the normally applied voltage.

The trapped charge is the charge deposited per unit time, as calculated in Ref 3, times the charging time. The charging time is taken to be the dielectric relaxation

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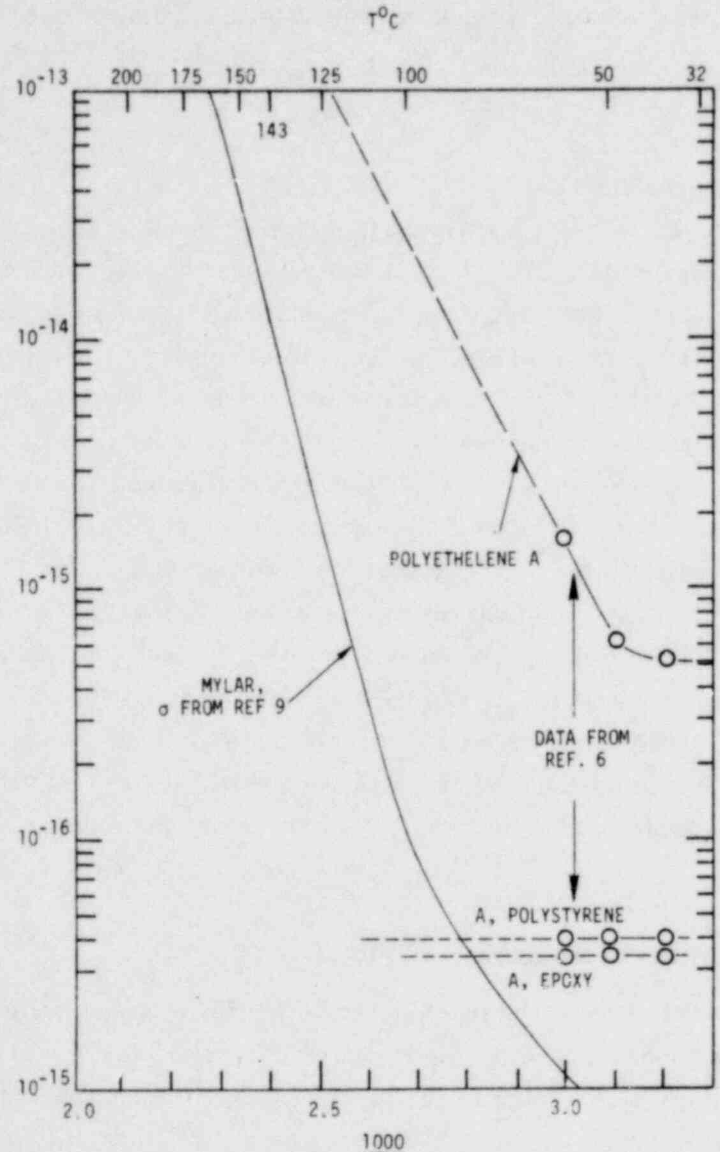

Figure 7. Electrical conductivity σ, and A as a function of temperature, T.

time presented in Table 1. Rather than treat each region separately, an average of 10^{-4} sec is used. For longer times, the field will be large enough to sweep most of the charges out of the cable. The total charge deposited within the copper is assumed to leak off immediately. This current represents a possible noise signal which has not been analyzed here because this is a power cable rather than a signal cable. Figure 8 shows the total charge per unit length $Q_T(r)$, in the dielectric from the copper out to radius, r. The charge starts at zero at r = 0.729 and goes negative because it is the accumulated electronic charge. The displacement, D, is found from the total charge by Gauss' Law applied to a unit length of the cable

$$2\pi r D_r = Q_T + Q_C \quad , \tag{4}$$

where Q_C is added to include image charges on the copper. The electric field is

$$E_r = D_r/\epsilon\epsilon_o = (Q_T + Q_C)/2\pi r\epsilon\epsilon_o \tag{5}$$

where ϵ is the dielectric constant of the medium and $\epsilon_o = 8.84 \times 10^{-14}$ F/cm is the permittivity of free space. The dielectric relaxation time is calculated from $\epsilon\epsilon_o/\sigma$. The constant Q_C in Equation 4 must be found from the voltage, V, across the cable. It is not a simple problem to decide what this voltage should be. In some places the outer jacket will make good contact with ground and will be at zero potential. Most places, it will charge up to an unknown voltage and then sweep the charge out. We shall examine both cases.

The relation between E and V is

$$E_r = -\,dV/dr \quad . \tag{6}$$

The solution to Equation 6 is written in terms of integrals over Q_T. To simplify notation we use r with subscripts C, E, and H to signify the outer radius of the three regions. Also subscripts E and H are added to ϵ to signify dielectric constants in these two regions. The final result is

$$V = V_C - \frac{1}{2\pi\epsilon\epsilon_o}\left[\int_{r_C}^{r} Q_T\,dr/r + Q_C \ln (r/r_C)\right] . \tag{7}$$

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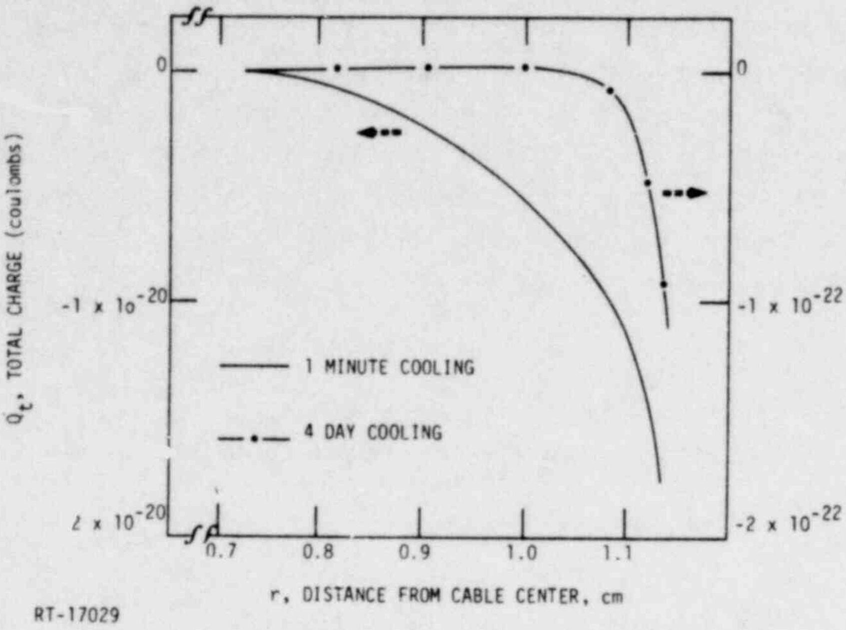

Figure 8. Total charge trapped in the cable insulation, Q_T, from the copper out to radius r.

The voltage at the Hypalon surface is

$$V_H = V_c - \left(\frac{1}{2\pi\epsilon\epsilon_0}\right)\left\{\frac{1}{\epsilon_E}\left[\int_{r_c}^{r_E} Q_T\, dr/r + Q_c \ell n\left(r_E/r_c\right)\right] + \frac{1}{\epsilon_H}\left[\int_{r_E}^{r_H} Q_T\, dr/r + Q_c \ell n\left(r_H/r_E\right)\right]\right\} . \quad (8)$$

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The solution for Q_C is

$$Q_C = \left[(2\pi\epsilon_o \epsilon_E)\ (V_H - V_C) + \int_{r_C}^{r_E} Q_T\, dr/r + 0.38 \int_{r_E}^{r_H} Q_T\, dr/r \right] / 1.1 \quad ,$$

where numerical values of the radii have been inserted into Equation 8.

This value of Q_C is inserted into Equation 5 to find the electric field,

$$E = \left(V_H = V_C + \frac{6.2 \times 10^{-21}}{\epsilon_E \epsilon_o} \right) / 1.1 = 266 + 2.6 \times 10^{-8} \text{ V/cm.}$$

The second term in E, caused by the trapped charge, is much less than the first term, caused by the applied voltage. The insulator is in no danger of breaking down if leakage current can drain off trapped charge as we assume here. The case where trapped charge builds up is discussed in subsection 3.5.

Also, this field will create a Joule heating of σE^2 which is about 5×10^{-5} watt/cm^3. This is much less than the power deposited by the radiation so there is no danger of "hot spots" driven by the positive feedback between increased σ with temperature and increased Joule heating with increased σ.

3.4 SHUNT RESISTANCE REPRESENTATION OF LEAKAGE CURRENT

The leakage current found above can be represented by a lumped shunt resistor, R_S, across the line:

$$R_S = \left(V_H - V_C \right) / \pi r L \sigma E = 1/\pi r L \sigma \approx 0.44/L\sigma \tag{9}$$

where L is the length of the cable. The numerical value of R_s is about 70 MΩ if L is 10 cm, and it is 7 MΩ if L is 100 cm. If σ is increased because of increased insulator temperature then R_S will decrease by the same factor. All of these shunt resistances are high enough to be of no concern for the power cable considered here. It is possible, however, that they could represent an unacceptable loss in a high impedance measuring circuit.

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3.5 THE NOISE PULSE GENERATED BY RELEASE OF TRAPPED CHARGE

We recognize that noise pulses from trapped charge discharges are probably not important to a power cable but would be for a signal cable. For completeness, we consider this topic. An estimate of this effect can be found by supposing that the field builds up to its breakdown limit, E_B, in some section of the cable which is not in good enough contact with a ground plane to allow ordinary leakage current to dispose of the trapped charge. The total charge built up per unit length, Q_B, under this circumstance can be found by inverting Equation 5 with $E_r = E_B \approx 10^6$ V/cm

$$Q_B = 2\pi r \epsilon\epsilon_o E_B \approx 1.3 \times 10^{-6} \text{ C/cm} \quad . \tag{10}$$

When the cable breaks down, the charge Q_B is transferred from the insulation to the copper where it neutralizes an equal image charge. In principle, the current flow during breakdown is perpendicular to the cable and it will not tend to induce an emf around the circuit to which the cable is connected. The exact magnitude of any induced emf depends on the geometry of the termination of the cable. If the charge transit time for the discharge is of order 10^{-6} sec, and the cable is a meter or more long, then it is unlikely that the induced emf would exceed a volt. The main noise will thus be a return current to establish a new image charge after the discharge. This current will flow over the period of about 10^{-6} sec so the 10^{-6} C/cm will be replaced by a current which is about one amp per cm of cable which failed. This current can be supplied from ground through the resistive load termination of the cable. If this load is very big then a large voltage has to be developed across the load. The charge can also be supplied from stray capacitance of the load termination. If this load capacitance is as big as the cable capacity of a few pf/cm, then it can supply the charge without changing its voltage by too much.

In summary, the amplitude of noise pulses built up by breakdown in the cable depends almost completely on the exact details of the cable termination. If the termination is either a small resistive load or a large capacitive load, then the noise spikes will not be large.

3.6 POSSIBLE ERRORS IN SIMULATION OF LOCA CABLE FAILURE

The trapped charge in the insulator is largely a consequence of the nonuiform dose deposited in the cable. The mechanism that applies here is that the secondary emission

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leaving a volume element is not balanced by secondary emission from adjacent points because the adjacent points are not receiving the same primary radiation. The gamma source simulators do not show such a strong attenuation as does the true LOCA source so they will not properly simulate the trapped charge and any resultant noise spikes. However, because radiation induced leakage currents are judged to be inconsequential we suggest that the LOCA environment is adequately simulated by Co^{60}, as long as average doses are matched.

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4. TEMPERATURE CHANGES IN THE CABLE

The total energy deposited inside any region of the cable is found by taking the dose rate within the region from Table 3 and converting the rad/sec to erg/gm-sec. This is then multiplied by density and converted to watts. The net result for copper is about 0.052 watt/cm^3. There is also a Joule heat generation from ½ I^2R losses in the copper. This is only about 10^{-3} watt/cm^3 when I is the nominal maximum of 20 amps; it is safely ignored compared with the radiation energy.

Two regions of the cable should be considered. The region considered first is in the middle where heat flow is radially through the insulator. The region near the ends is considered in the next paragraph.

Select a section of cable L cm long so the total power deposited into the cable is ($\pi r^2 L$) 0.052 watt. At equilibrium there will be a temperature gradient, ΔT, across the EPR which is big enough to conduct this heat to the external environment. In other words ΔT must satisfy

$$\frac{\Delta T}{\Delta r}(2\pi r L)\kappa_E = 0.052\,\pi r^2 L \quad . \tag{12}$$

The radius r is 0.729 cm and the thickness, Δr, is 0.18 cm which yields a value of ΔT of

$$\Delta T = \frac{0.052 \times 0.729 \times 0.18}{2 \times 3.47 \times 10^{-3}} = 1.0^{\circ}C \quad . \tag{13}$$

The radiation energy deposited in the Hypalon is 0.14 watt/cm^3. If this much heat flows radially out through the Hypalon itself, then a temperature change of about 7°C must exist within the Hypalon region. Such a rise above an ambient of 143 degrees is still safely below the service temperature of both EPR and Hypalon.

If the cable is in poor thermal contact at its outer radius, then the heat must flow into the copper and the heating problem becomes more serious. The heat must then flow out lengthwise to the ends of the cable or to regions where radial flow is again permitted. For the case of linear flow in the copper, Equation 12 is recast as

$$\frac{\Delta T}{\Delta L} \pi r^2 \kappa_c = 0.04 \pi r^2 L$$

$$\Delta T = 0.04(L \Delta L)/4.19 \quad . \tag{14}$$

As an illustration, take $L \simeq \Delta L = 10$ cm and ΔT is about a degree. If L were a meter and if this were the only heat current allowed, then T could rise by ten to one hundred degrees, which does approach the temperature limit of the insulators.

On the other hand, the temperature rise may not be quite as large as indicated above since the two transport mechanisms will operate in parallel and allow the temperature to reach some lower temperature than that calculated by either mechanism by itself. In any event, however, some sections of the cable may approach the maximum service temperature of the insulators.

The ^{60}Co and ^{137}Cs simulation radiation sources do not reproduce the true LOCA attenuation into the cable so that the temperature profiles established by these simulators will not accurately reproduce the true LOCA profiles. In particular, the simulation fields are not attenuated in the cable as the true LOCA spectrum. Thus, the simulator will deposit too much power in the copper and inner portions of the Hypalon compared with the power deposited into the outer surface. If the simulation is performed so that the total doses are equivalent, then the simulator will exaggerate the heating of the inner insulator region. This is the inverse of the case of electrical effects generated by trapped charges where the simulator understresses the effect.

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5. CHEMICAL AND MECHANICAL CHANGES IN THE CABLE

Several different chemical changes occur in a polymer or an elastomer (cf. Refs 1, 10-12) upon exposure to radiation: evolution of gases, crosslinking, and scission of bonds (also called unsaturation). The magnitudes of the effects are usually given by a G factor which gives the number of such events produced by 100 eV of absorbed radiation. The number of events per rad can be found by multiplying G/eV by 6.24×10^{11} eV/erg if desired. The values of G that are used here are given in Table 4. Some analysis, as discussed next, was done to obtain these numbers.

Table 4. G, the Number of Chemical Changes per Absorbed 100 eV at 140°C

Process	Polyethylene	Polypropylene
Crosslinking	3.3	1.0
Hydrogen Evolved	6.0	1.0 to 2.0
Unsaturation	3.0	0.8 to 1.0

The values for G for crosslinking and hydrogen evolution in polyethylene are given by Bolt and Carroll (Ref 10) as a function of temperature. We have used these values in Table 4. Reference 9 also contains G unsaturation for polyethylene, G crosslinking, and G unsaturation for polypropylene at 20°C. The increase in G with temperature will be similar to polyethylene so we have scaled these values up similar to the increase of polyethylene. Kircher and Bowman (Ref 11) indicate that an unsaturated hydrocarbon (polypropylene) evolves hydrogen at one-half to one-sixth the rate of a saturated hydrocarbon (polyethylene). This allows filling in the final entry in Table 4.

Bolt and Carroll's data indicate that the largest increase in G is already achieved by 140°C; a further increase to 200°C raises G by only five percent. There will not be a large synergistic effect between radiation and temperature above the steam temperature.

This analysis requires the total dose rather than the dose rate used heretofore. There are two factors to consider in integrating the dose rate over time to obtain dose.

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The first factor is that the dose rate is steadily decreasing with time. This has already been accounted for by Bonzon (Ref 4) who presents total dose at the cable surface as a function of time. This can be used directly for total dose in the Hypalon region. The second factor is the energy degradation of the LOCA spectrum so that there is greater attenuation into the cable with increasing time. This factor was accounted for in the EPR layer by simply interpolating the attenuation factor between the one-minute and four-day cases. For later times the original data from Reference 2 is used to extrapolate the attenuation factor. The total dose so determined is shown in Figure 9.

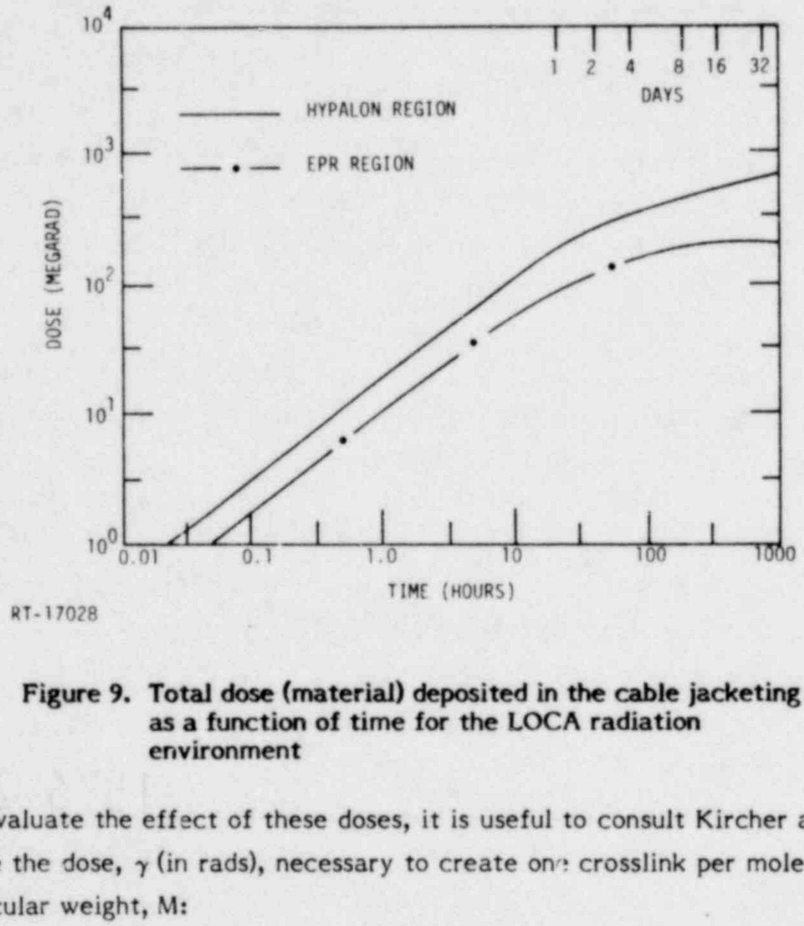

Figure 9. Total dose (material) deposited in the cable jacketing as a function of time for the LOCA radiation environment

To evaluate the effect of these doses, it is useful to consult Kircher and Bowman who derive the dose, γ (in rads), necessary to create one crosslink per mole in material with molecular weight, M:

$$\gamma = 4.8 \times 10^{11}/MG_{c\ell} \quad . \tag{15}$$

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$G_{c\ell}$ will be between 1.0 and 3.3. For this example a G of 2.0 was selected. The value of γ to produce a crosslink per monomer (small molecule) with molecular weight M midway between polyethylene 14 and polypropylene 28, is about 10^4 Mrad. The unsaturation G is about the same as the crosslinking G, while the H_2 evolved is much larger. Thus by 10^4 Mrad (on the average), every monomer in the organic material has suffered a scission, or a crosslink. By the time this dose has accumulated, the polymer will be nearly disintegrated. It may be arbitrarily assumed that the material will retain some semblence of its structure if only one out of twenty monomers is decoupled or crosslinked. In this case, the acceptable dose is about 500 Mrad. This dose is reached in the Hypalon after approximately ten days in a LOCA environment, and may not be attained in the EPR region at all.

The mechanical evaluation in References 10 and 11 is also of interest in this regard. It essentially tells the same story as above. Hypalon starts to deteriorate at 90 Mrad (about ten hours) and loses stability seriously at 300 Mrad (eight days). The tensile strength of polyethylene and polypropalene will maintain up to 80 percent of their original value through 100 Mrad (one to two days) and will drop to 50 percent or less after 500 Mrad. The elongation and impact strength will deteriorate by 90 percent or more after 50 Mrad (which is accumulated after ten hours in a LOCA radiation environment). These later properties will determine the cable's resistance to vibration, which radiation environment should be specified in a LOCA.

We note that there is some uncertainty in the literature concerning the dose level at which Hypalon becomes seriously deteriorated. The values reported by Bolt and Carroll (Ref 10) are about a factor of 100 less than those given by Kircher and Bowman (Ref 11). The latter values are more consistent with failure levels in similar materials, and therefore they have been adopted here.

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6. SUMMARY AND CONCLUSIONS

The most serious problems that will occur with the cable are chemical and mechanical deterioration which are expected to occur after a few hours to a day in LOCA environment. Impact strength and elongation changes by 50 percent within ten hours. In order to properly evaluate these effects the expected vibrational and bending stresses should be included in the specification of the LOCA environment.

The temperature rise in the insulation layers can approach the recommended service limit of Hypalon because of the energy deposited in the Hypalon and in the copper by the radiation field.

It is not expected that leakage resistance will be less than a megohm. Whether this impedance is a problem or not depends on what sort of circuits the cable connects. Voltage noise spikes will probably be of short duration. The amplitude of these spikes depends on the cable load impedance.

The gradient of dose rate into the cable, which is not simulated by standard gamma sources, has two consequences. The first is to allow the radiation to deposit charge into the insulator because the back and forward flow of secondary emission is not equal. The main role of this trapped charge is to create voltage noise spikes. These spikes may not be reproduced by a simulator. The second consequence of the $\dot{\gamma}$ gradient is that more energy is deposited in the interior by the nonattenuated simulator than is deposited by the LOCA radiation field. The simulator may exaggerate the interior temperature compared with the actual situation.

Several material parameters such as electrical conductivity and the radiation G values had to be interpolated from similar materials or from adjacent temperatures. If more accurate estimates than are presented here are ever needed, then a measurement program would be required.

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