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Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report

April - June 1978

Lloyd L. Bonzon, Kenneth T. Gillen, Lowell H. Jones, Edward A. Salazar

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

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QUALIFICATION TESTING EVALUATION PROGRAM: LIGHT WATER REACTOR SAFETY RESEARCH. QUARTERLY REPORT

APRIL - JUNE 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 R. Clough, and arranged and edited by L. Bonzon.

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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 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 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 of 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.

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" 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 to test the method postulated for combined environment accelerated aging.

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 factor; 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: On April 21, W. Rutherford requested information on qualification testing facilities capability and costing. This was followed, on May 2, with a telecon request by W. Rutherford, R. Feit, and G. Bennett to transmit the Class 1E equipment lists (developed by UEC under contract to Sandia) to assist Inspection and Enforcement (IE) staff in their response to the Commission directive to develop a "Plan for an Analysis of Alternatives for Conducting Independent Verification Testing of Environmentally Qualified Equipment." The requested information was transmitted by letter dated May 5, 1978.

A "Plan to Investigate the Adequacy of Quality Assurance Practices for NRC-Sponsored Confirmatory Research Programs" was discussed by NRC/DOE/Contractor staff in Silver Spring on May 11, led by G. Bennett. R. Luna and L. Bonzon attended this meeting as the Sandia representatives to provide the requested information on Sandia's formal QA practices with regard to these research programs.

On May 11, discussions centering on a proposed work statement to Sandia (to estimate the costing of a large-scale qualification test facility) and the Commission-requested connector tests were held with R. Feit and W. Rutherford in Silver Spring. W. Rutherford provided certain elaboration and clarification of the work scope and timetables. R. Feit discussed the connector-tests status and the necessary information required from the utilities and vendors before particular connectors could be selected and tested.

A review of the radiation qualification and simulator evaluation effort was made for P. Tam, H. Krug, A. Hintze, L. Soffer, F. Akstulewzcez, and R. Feit in Bethesda on May 12. The discussion included a brief review of the formal reports generated and work in progress to date with emphasis on the LOCA beta signature and simulator adequacy. The proposed study by NRR and Standards to estimate generic in-containment LOCA radiation dose and dose rates to Class 1E equipment was detailed by P. Tam. It is expected that a funded study will be requested by NRR/Standards to be programmed by RSR in the very near future.

On May 12, discussions were held with A. Hintze, at Nicholson Lane, on the status of and schedule for Regulatory Guide 1.89; it is currently being reviewed internally by NRR staff.

A review of staff responses to equipment (radiation) qualification was discussed with J. Slider in Bethesda on May 12. A discussion of the applicable Sandia work in progress was presented. (Mr. Slider requested several formal reports; these were transmitted to him on May 15.) Regulatory Guide 1.97 was also discussed, specifically the correctness of the 10^8 R/hr rate requirement for post-accident monitoring (PAM) equipment qualification. Several recent program reports,[1.5,1.10,1.11] are pertinent to scoping the rate requirement for PAM equipment.

On June 15, W. Rutherford visted Sandia to discuss the Commission-requested connector tests and the IE Alternatives Study. Significant effort was devoted to contacting the potential connector assesmbly suppliers to determine if their files were sufficient to allow the reconstruction of certain previously purchased assemblies. A review of utility-supplied connector purchase/test information was made to determine insufficiencies and to develop specific IE and Sandia action items to promote the testing efficacy. A detailed discussion of the IE Alternatives Study led to a list of action items for the IE and Sandia staffs; Sandia's principal effort will be to proceed to arrange for the necessary staff support, while the funding authorization proceeds through NRC and DOE.

On June 22, R. Scholl requested information on the availability of Class 1E electrical cable from prior test programs, and the scheduling of

LOCA-simulation equipment. NRR will ask Sandia to perform moisture migration tests in single conductor electrical cable.

Program Formalization: Two Buff-Book submittals were prepared and submitted on April 26 and June 29. These generally represented routine input, except that several potential programmatic difficulties were identified which could affect the milestone schedules for select subtasks; specifically, certain unexpected review/reporting requirements and significant additional tasks will require a revised priority schedule for all tasks.

A draft Schedule 189 for FY78 was prepared in early April. Internal review and sign-off was completed in early May and submitted to DOE/ALO on May 15. Official transmittal to RSR was completed by letter dated June 2 from DOE/ALO. This preliminary Schedule 189 outlines the major task efforts and cost estimates for the QTE program; the suggested funding level is somewhat reduced from the FY78 authorized funding.

The routine reporting of program activities is done through formal quarterly reports issued 4 to 6 months after the close of the reported quarter. The first quarterly [1.12] (October-December, 1977) completed sign-off on April 21 and was issued on May 15. A draft of the second quarterly[1.13] (January-March, 1978) was completed on June 28 and is undergoing internal Sandia-required review and sign-off.

Standards Committee and Meetings Participation: On March 22, S. Khalifa of TVA, as a member of IEEE Subcommittee 2, Working Group 2.11, requested that K. Gillen assist in developing a guide for typetests of Class 1E connectors, connections, and field splices. Mr. Khalifa is very interested in the proposed Sandia method for accelerated aging in combined stress environments,[1.12,1.14] since no other general approach currently exists. He intends to propose the Sandia method as an available technique to achieve simultaneous thermal and radiation aging and to estimate the qualified life. On April 3, Mr. Khalifa formalized his request (by letter) to provide assistance and information to the Working Group. After discussions with R. Feit, K. Gillen formally (letter dated May 22) agreed to work with the committee in the development of the guide.

K. Gillen (as a member) and E. Salazar hosted and attended the third meeting of IEEE/ICC Working Group 12-37, April 28, in Albuquerque. The group is charged with the study of environmental aging of electrical cable insulation and jacketing systems. The end objective is to determine methods that will allow cable to be "aged" to a condition representative of selected years of use under specified operating conditions. The meeting discussion included: a review of a proposed outline for "Electrical Insulation Selection Criteria for Nuclear Power Plant Service"; the proposed Delphi-survey to obtain the industry experience; and, a review of the Sandia program and results to date. After the meeting, committee members toured the Sandia aging facilities.

L. Bonzon attended the 24th Annual IES Meeting, in Ft. Worth on April 20, specifically the session on "Qualification of Components for Nuclear Power Plants." The session included papers on IEEE standards activities, radiation qualification, seismic qualification, and BWR-blowdown scale tests.

Diablo Canyon Plant Tour: Sandia staff visited the Diablo Canyon Nuclear Power Station on April 7. Located near San Luis Obispo, CA, the station consists of twin 1060 MW(E) Westinghouse PWRs. The purpose of the visit was to examine in-place Class 1E equipment with particular emphasis on interfaces, mounting, and location of the equipment. Pacific Gas and Electric served as its own architect-engineer and constructor for the project; since it is virtually complete, it is an ideal facility to inspect and is representative of the current state-of-the-art.

International Exchange: During the previous quarter, it was proposed to the Technical Program Chairman that a dedicated session on Qualification Testing Methodology be incorporated into the program of the International Meeting on Nuclear Power Reactor Safety, to be held October 16-19, 1978 in Brussels, Belgium. Concurrently, twelve abstracts were submitted in mid-March, covering all aspects of the NRC, IEEE, and Sandia (and subcontractors) programs. Ten abstracts were Sandia-sponsored; the remaining two were related, but independently submitted. (Sandia-sponsored abstracts, with author list and Sandia designation, are described in Section 1.5.)

On May 24, notice was received that the Technical Program Committee had created a new session, "Environmental Equipment Qualification," and that the papers were accepted but were to be combined to reduce the number. The committee's comments were incorporated, resulting in the session as outlined in Table 1.1, which was forwarded by letter dated May 30. Full papers will be included in the Proceedings of the meeting.

Foreign Interest: Dr. Y. Nakase, Japan Atomic Energy Research Institute, requested specific information (by letter, received April 4) on the test methods described in Reference 1.15, particularly the dose rate chosen in the testing. His inquiries were answered by letter dated April 10; certain additional formal reports were transmitted as enclosures. Dr. Nakase also requested information (June 9) on the October 1978 International Meeting on Nuclear Power Reactor Safety, which was forwarded to him.

On May 17, at their request, a brief program review was presented to A. Onodera (Hitachi), Y. Fujise (Mitsubishi), K. Ikeda (Kobe Steel), and Y. Maki (Central Research Institute); all are Japanese nationals.

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 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.

TABLE 1.1

Session: Environmental Equipment Qualification

Author	Submitted For	Presentation Time	Title
R. Feit	Publication/ Presentation	15 minutes	The Nuclear Regulatory Commission Research Program to Improve the Operational Safety of Nuclear Power Plants
H. Thornburg*	Publication/ Presentation	25 minutes	Quality Assurance and Its Effect on Safety
J. Fragola	Publication/ Presentation	20 minutes	Standards Activities Related to Class 1E Equipment Qualification
R. Luna L. Bonzon	Publication/ Presentation	15 minutes	Methodology Assessment: An Overview of the Qualification Testing Evaluation (QTE) Program
K. Gillen	Publication/ Presentation	20 minutes	A Proposed Method for Combined Environment Accelerated Aging
K. Gillen E. Salazar	Publication	--	Aging of Nuclear Power Plant Safety Cables
L. Bonzon	Publication/ Presentation	20 minutes	An Experimental Investigation of Synergisms in Class 1E Components
S. Kasturi G. Dowd L. Bonzon	Publication/ Presentation	20 minutes	Qualification of Class 1E Equipment: The Role of the Utility and Architect-Engineer
N. Lurie J. Naber L. Bonzon	Publication	--	The Hypothesized LOCA Radiation Signature and the Problem of Simulator Adequacy
L. Bonzon N. Lurie	Publication/ Presentation	20 minutes	The Best-Estimate LOCA Radiation Signature: What It Means to Equipment Qualification
R. Leadon**	Publication/ Presentation	25 minutes	Damage Mechanisms to Cables in Reactor Environments
L. Bonzon R. Luna S. Carfagno	Publication/ Presentation	20 minutes	Qualification Problems: The Rest of the Iceberg

*USNRC, independently submitted.

**IRT Corporation, independently submitted.

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 requests are briefly reviewed in Table 1.2.

Related Study for IE: In early May, IE requested Sandia assistance in support of a Commission-requested study of alternatives "...for conducting independent verification testing of environmentally qualified equipment which is required to operate in safety systems."[1.16] On May 5, Class 1E equipment lists were forwarded to W. Rutherford. On May 11, Sandia was asked to quote on a portion of the study to estimate the costs involved in constructing, equipping, and operating a test facility that could be used for conducting environmental tests in accordance with IEEE-323-1974 requirements on environmentally sensitive equipment. The Sandia quote was forwarded by letter, through R. Feit, on May 16.

On June 2, Sandia was asked to conduct the entire study as an expanded scope of work under an existing contract. Sandia responded affirmatively by letter dated June 5. Discussions were held with IE staff on June 15 and 29 to clarify and elaborate on the study. Subcontractors have expressed interest in assisting Sandia in this study. Funding for the study had not been received as of June 30.

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.†

†These Quick-Look reports are available in the USNRC Public Document Room; test results are being incorporated into a topical report for wide distribution.

TABLE 1.2

Industry Liaison

Date	Company	Requested: Prior Reports	Requested: Mailing List	Other
April 12	Bendix-Sidney			Information on connector tests
May 4	Bendix-Sidney			Information on connector tests
May 9	Battelle-Columbus			To provide testing service to Sandia
May 18	ITT-Phoenix	SAND78-0718		
May 19	SAI-LaJolla	SAND78-0718 SAND78-0341	X	
May 23	Bechtel-Ann Arbor	SAND78-0341		Chemical spray effects during LOCA simulation
May 24	B&W-Lynchburg	SAND77-1713C SAND77-1654C		
May 26	Hitachi-New York	SAND78-0718 SAND78-0341 SAND78-0091	X	
May 26	AECL-Ontario	SAND78-0718 SAND78-0341 SAND78-0091	X	
May 30	Sigmaform-Santa Clara	Test VIII		
May 31	CE-Windsor	SLA-73-5349A		Aging to MIL 217B experience
June 1	ITT-Phoenix			Typical ambient aging temperature and allowable acceleration
June 2	LASL-Los Alamos	SAND78-0091		
June 5	AEP-New York	SAND78-0718 SAND78-0341 SAND78-0091	X	Radiation damage to PAM equipment per RG 1.97
June 12	GE-Gaithersburg			General
June 12	GE-Bridgeport	SAND78-0718 SAND78-0341 SAND78-0091	X	
June 13	SAI-LaJolla			Information on future work
June 14	B&W-Lynchburg	SAND78-0341	X	Generic containment dose and rate values
June 16	B&W-Lynchburg	SAND77-1075A		
June 19	YAEC-Westboro			Extrapolation of test data to Connecticut Yankee
June 20	SAI-LaJolla			Information on future work
June 21	S&L-Chicago			LOCA-radiation source term
June 21	S&W-Boston			Information on future work
June 26	FIRL-Philadelphia			Dose conversion factors and available radiation test facilities

-- 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.

-- 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 1 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, 1977 Proceedings of the Institute of Environmental Sciences, April 1977.

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*These Quick-Look reports are available in the USNRC Public Document Room; test results are being incorporated into a topical report for wide distribution.

-- D. V. Paulson and S. P. Carfagno, "A Review of Class 1E Cable Qualification Data," FIRL Final Report F-C4598-1, prepared for Sandia Laboratories, June 1977. (Being incorporated into a Sandia topical report.)

-- N. A. Lurie, "Definition of Loss-of-Coolant Accident Source for Radiation Component Qualification: Phase 1 Final Report," IRT 8167-001, prepared for Sandia Laboratories, June 1977. (This report intended for internal Sandia use only.)

-- 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.)

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†These Quick-Look reports are available in the USNRC Public Document Room; test results are being incorporated into a topical report for wide distribution.

-- 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.

-- 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 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.

- 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.

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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 SAND78-0718, "Preliminary Data Report, Text IX. Simultaneous Mode; Cables, Splice Assemblies, and Electrical Insulation Samples,"[2.1] was made in early May. This report (see also Reference 2.2) describes the test and results and compares the test with Test VIII, its sequential complement. There were no functional failures in either test, with cables and cables with splices maintaining rated current and voltage loads through both tests. The preliminary results indicate that no significant functional or material synergism exists. All splicing materials endured the conditions of LOCA better than the insulation materials in the attached cable. Type A cable showed less degradation of its electrical properties than did types B or C. The insulation tensile-specimens did not exhibit any strong synergistic effects, but the data indicate the potential importance of accurate and adequate aging.

Connector Tests: A significant new task, identified during the previous quarter, requires additional testing of cable connector assemblies. 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."

The significant effort this quarter involved the interpretation of, and response to, the Commissioners' memorandum by NRC (RSR, 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:[2.3]

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. Since Viking no longer manufactures connectors, other (PWR) plants/connectors are also being considered by IE staff.

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 March, twenty data packages had been completed. During this quarter one additional data package (on motors) was transmitted, as well as extensive revisions to the Class 1E equipment list. Four data packages remain to be completed under the current contract scope. (See also Section 2.2).

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/RSR staff. For the latter, the general conceptual plan was forwarded to FIRL by letter dated May 30. FIRL responded in a letter dated June 27, which contained 15 points, comments, or considerations. In general they concurred with the conceptual philosophy of the design:[2.4]

> "The essential purpose of the upgrading is to achieve a more versatile testing facility suitable for experimental type testing rather than routine testing ... . The proposal appears to do this well.
>
> "Overall, it appears that the upgraded facility will be able to handle larger items at greater dose rate and with a larger selection of dose rates; it will also provide a more uniform dose rate than the present facility. These factors should lead to more useful results than are available at present."

Two other packages are being drafted to be forwarded to FIRL for review in early July: autoclave and ancillary equipment, and High Intensity Adjustable Cobalt Array (HIACA).

Synergism Topical Report: A topical report, "An Experimental Investigation of Synergisms in Class 1 Components Subjected to LOCA Typetests,"[2.5] was drafted and received preliminary review in late 1977. It incorporated the results of the synergistic typetests (through Test VIII) and included the full report of the offsite synergism study performed by FIRL.[2.6]

Following the completion of Test IX,[2.1] the test results and the data analyses were incorporated into the draft in April and internal Sandia reviews were initiated in May. Final reviews are expected to be completed in August with final printing and distribution scheduled for September.

QA Program for NRC-Sponsored Research: In a Memorandum and Order[2.7] issued by the NRC Commissioners on April 13, 1978, the NRC staff was directed to:

> "Develop a plan to investigate the adequacy of quality assurance practices for NRC-sponsored confirmatory research programs and provide recommendations to the Commission."

On May 11, R. Luna and L. Bonzon met with other NRC, DOE, and DOE-laboratory staff to discuss a May 9 draft plan developed by RSR personnel. Based on the meeting discussion, a second draft was forwarded to Sandia on May 15 with a request for comment by May 25.

Facility Upgrade: The existing test facility will be upgraded to accomodate 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 were concentrated in three areas: a redesign of the original concept of radiation source positioning; installation of additional facility shielding and cell modifications; and continued design definition of the test chamber and support equipment.

A new design of the HIACA was initiated to replace the excessively long (12 ft) and rigid test fixture design that was initially proposed. The new design allows precise visual assurance of cobalt source position; it consists of a hydraulically telescoping series of nested tubes that raise the 50-in. (double-length) cobalt pencils to the same 12-ft height, but is only 5-1/2 ft high when completely nested. Fabrication costs are reduced from the previous design and all radiation source transfers can be accomplished underwater in the open pool.

Additional shielding was attached to the common irradiation cell walls; this consisted of two inches of depleted uranium plate on each side of the wall.[2.8] In addition, two through-ports were core drilled into the cell ceiling to provide ingress/egress capability for experiment/loading/diagnostic equipment.

As described in a previous subsection (FIRL Subcontract), documentation of design objectives and designs was formalized to allow FIRL review. Three packages are to be prepared: general plan; autoclave and ancillary equipment; and HIACA. The first of these was forwarded to FIRL on May 30 and was formally reviewed by letter dated June 27;[2.4] in general, FIRL concurred with the conceptual philosophy of the design. The two other packages are being drafted to be forwarded in early June.

The bulk of the quarterly effort centered on the continuing design definition of the major equipment items and facility modifications. In some cases, long-lead-time items were identified and orders placed (e.g., autoclave dome heads, pumps, hydraulic hose). Major effort remains for the autoclave handling/support apparatus, HIACA, source-handling tools, elevator modification, and additional facility shielding.

2.2 Class 1E Equipment Identification*[2.9]

2.2.1 Study Scope

In order to provide a basis for equipment selected for use in LOCA-simulation methodology typetests, a comprehensive study was required. This included assembling and/or determining the following pertinent information:

(a) complete listing of typical Class 1 equipment

(b) realistic ambient and accident environments

(c) physical construction of equipment and materials list

(d) performance specification, service-life history, and maintenance schedule

(e) general equipment vulnerability and "weak-links" where known by prior use experience

(f) electrical/mechanical/environmental interfaces.

Three other objectives can be realized using the data obtained. First, the materials list and construction details should allow

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*This section prepared by L. Bonzon, Division 5432.

comparisons to be made with performance data available in the literature or with prior typetest results; this may preclude the necessity for typetesting all components or manufacturer's types with comparable materials. Second, the amount of equipment potentially requiring typetesting can be estimated based on the available data. Third, realistic ambient and accident environment specifications should allow optimum typetest environmental envelopes and procedures to be developed. This may lead to typetests based on realistic, not (conservative) envelope, considerations.

In late July 1977, UEC was authorized to conduct this study in a four-concurrent-phase subcontract for a contemporary PWR nuclear power plant for which UEC was committed to provide comprehensive equipment qualification certifications during the licensing process. The scope of the effort was limited to in-containment Class 1E equipment and that Class 1E equipment located just adjacent to containment in the penetration area. Through the course of the study UEC completed four distinct subtasks: description of systems; Class 1E equipment lists; environmental design criteria; and equipment data packages.

The description of systems delineates the major safety systems and their importance and function in mitigating accident consequences.

The Class 1E equipment list enumerates all equipment potentially exposed to harsh accident environments and provides information such as service description, plant location, accidents scenarios during which it functions, time at which it is required to start after an accident and the duration of function, environmental conditions, and number of cycles through which it will be operated throughout plant life.

The environmental design criteria describe the conditions of aging and accident environments in terms of (a) best estimate of the normal operating environment, (b) maximum service environmental conditions, and (c) qualification parameters, including margin. (A service environment chart and equipment location plans were also developed.)

An equipment description data package was compiled for each category of equipment. Non-proprietary information was obtained from 46 vendors, out of which 22 data packages were prepared. The data packages contain a description of the equipment, operating principle, installation and interface details, operating and environmental conditions, performance specification, materials of construction, maintenance procedures, periodic testing and calibration procedures, vulnerability details, and manufacturer's bulletins and test reports.

2.2.2 Systems Descriptions and Environmental Design Criteria

In this generic PWR design, some 20 systems within containment have Class 1E equipment as integral part(s) of the system. These systems, their function, and their Class 1E function, are briefly described in Table 2.1. While some of the systems provide an active Class 1E function (spec., CAH, CBS, CS, EDE, FW, MM, RH, SI), the majority containing Class 1E equipment provide containment-isolation or reactor-trip functions only. In terms of the complexity of the system, relative to numbers of Class 1E equipment, the reactor coolant (RC) system is the most complex; it contains 12 flow transmitters, 8 flow relays, 3 level transmitters, 6 pressure transmitters, 25 temperature elements, 12 pneumatic or motor-operator valve actuators, 12 position switches, and associated cabling, connectors, enclosures, etc. Out of all these items, only nine are located just outside containment (in the equipment tunnel up to the first isolation valve outside containment).

The electrical/mechanical equipment located within the containment and the area outside but adjacent to the containment penetration must be designed to withstand the normal environmental conditions of pressure, temperature, humidity, and radiation, and then operate during a design-basis accident, without failure. The design also considers environmental conditions prevalent during certain tests, e.g., periodic containment leak-rate testing. Specific maximum/normal/minimum environments were detailed for all plant areas; Table 2.2 summarizes the in-containment information.

TABLE 2.1

In-Containment Systems Description

System	Function	Class 1E Function
CAH - Containment Air Handling	Air cooling/circulation for hydrogen mixing.	Recirculation portion is an engineered safety feature (ESF).
CAP - Containment Air Purge	Purge air prior to personnel entry at cold shutdown.	CAP isolation components are active safety.
CBS - Containment Building Spray	Sprays atmosphere with sodium hydroxide, borated water solution to remove iodine and reduce containment pressure after an accident.	CBS system is an ESF.
CC - Primary Component Cooling Water	Cooling system for all safety-related or (potentially) contaminated systems or components.	CC components providing containment isolation are active safety.
COP - Containment On-line Purge	Limit containment radioactivity concentrations during normal operation.	COP isolation components are active safety.
CS - Chemical and Volume Control	Maintain inventory and chemistry of the reactor coolant. Charging pumps provide high pressure injection for ECCS.	High pressure injection is a safety function.
EDE - Electrical Distribution-Emergency	Provides electrical power for all safety-related equipment.	Provides electrical power for all safety-related equipment.
FW - Feedwater	Provide water for the first level heat sink to the primary systems via the steam generators. Emergency feedwater is an ESF providing decay heat removal.	Emergency feedwater is an ESF providing decay heat removal. FW components provide input to Reactor Protection System.
MM - Miscellaneous Equipment	Includes connectors, terminal blocks, NEMA enclosures.	Includes connectors, terminal blocks, NEMA enclosures.
MS - Main Steam	Transports energy from the RC system to the turbine generator.	MS instrumentation provides inputs to the Reactor Protection System. MS isolation components are active safety.
NG - Nitrogen Gas	Nitrogen gas is supplied to accumulators, pressurizer relief tank, and the main steam lines for blanketing and pressurization.	NG isolation components are active safety.
NI - Nuclear Instrumentation	Provides input to the reactor control system during normal operation.	NI provides input to the Reactor Protection System.
RC - Reactor Coolant	Cooling of reactor core and heat transfer to the steam system.	RC components provide containment isolation and inputs to the Reactor Protection System.
RH - Residual Heat Removal	Provides decay heat removal to bring plant to cold shutdown.	RH is an ESF; provides low pressure injection for ECCS.
RM - Radiation Monitoring	Provides radiation monitoring function throughout the plant.	RM provides inputs to the Reactor Protection System.
RMW - Reactor Make-up Water	Provides reactor quality water for chemical and volume control in primary systems.	RMW isolation components are active safety.
SB - Steam Generator Blowdown	Provides bleed-off of water on secondary side for control of water chemistry.	SB isolation components are active safety.
SI - Safety Injection	Provides medium pressure core injection for ECCS.	SI is an ESF.
VG - Equipment Vent	Provides a collecting system for gases from radioactive or hydrogenated components and for evacuation of piping systems during filling.	VG isolation components are active safety.
WLD - Equipment and Floor Drains	Collects all liquids from systems for processing.	WLD isolation components are active safety.

TABLE 2.2

Containment Environmental Parameters for the Generic PWR

	Normal	Accident	Post-Accident
Pressure (PSIG)			
	60 (test)		
Maximum	1.5	52	6
Normal	0.5	N/A	N/A
Temperature (F)			
	100 (test)		
Maximum	120	278*	160
Minimum	50	50	
Humidity (%)			
Maximum	90	Steam/Air	Steam/Air
Minimum	5	Mixture	Mixture
Containment Spray	None	Boric Acid 1.2% Wt PH 7.5 - 10.5 NAOH Used to control PH	None

*Temperature transient of 296°F is reached for first minute.

To analyze the normal radiation environment expected, the containment is divided into three radiation areas: inside the primary shield, inside the secondary shield, and outside the secondary shield to the containment wall. Inside the primary and secondary shield the radiation dose rate during normal plant operation is estimated at 50 rad/hr; integrated over 40 years, this gives 2×10^7 rads. For reactor coolant piping instrumentation in this area, the radiation dose rate during normal plant operation is estimated at 820 rad/hr; integrated over 40 years this gives 3×10^8 rads. Outside the secondary shield to the containment wall the radiation dose rate during normal plant operation is estimated at

150 mrad/hr; integrated over 40 years this gives 5.3×10^4 rads. However, for equipment in close proximity to radioactive pipes or sumps the radiation could be much higher and is conservatively estimated to be 50 rad/hr. Complete equipment location drawings have been provided to allow estimates of ambient radiation environments for each specific item; these radiation values are also reflected in the Class 1E equipment lists (Section 2.2.3).

The qualification parameters were established using the margins as follows:

Pressure	= Maximum pressure + 10%
Temperature	= Maximum temperature + 15%
Humidity	= Maximum humidity
Gamma Dose	= Maximum + 10%
Beta Dose	= Maximum + 10%

Figure 2.1 shows the accidents and envelope curves for containment pressure and temperature. The integrated accident dose is listed in Table 2.3.

2.2.3 Class 1E Equipment Lists

The Class 1E equipment list delineates all Class 1E equipment located inside and outside containment in the penetration area. For each piece of equipment this list provides information such as service description, plant location, accidents during which it functions, time at which it is required to start after an accident and the duration for which it functions, environmental conditions, number of cycles through which it will be operated throughout plant life, its safety function, and the supplier/model of the equipment item.

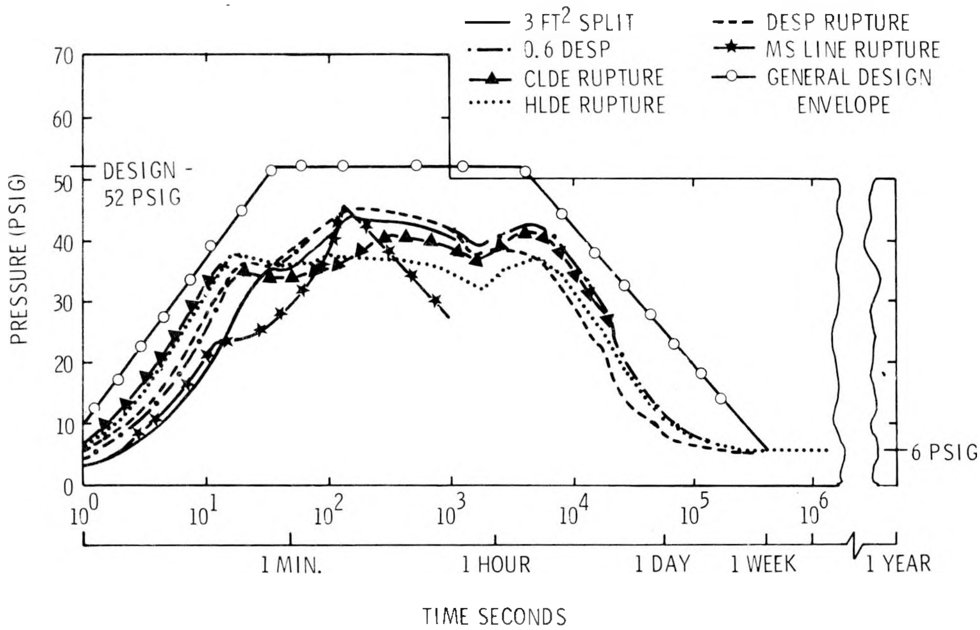

Figure 2.1a. Containment accident design envelope, pressure-time

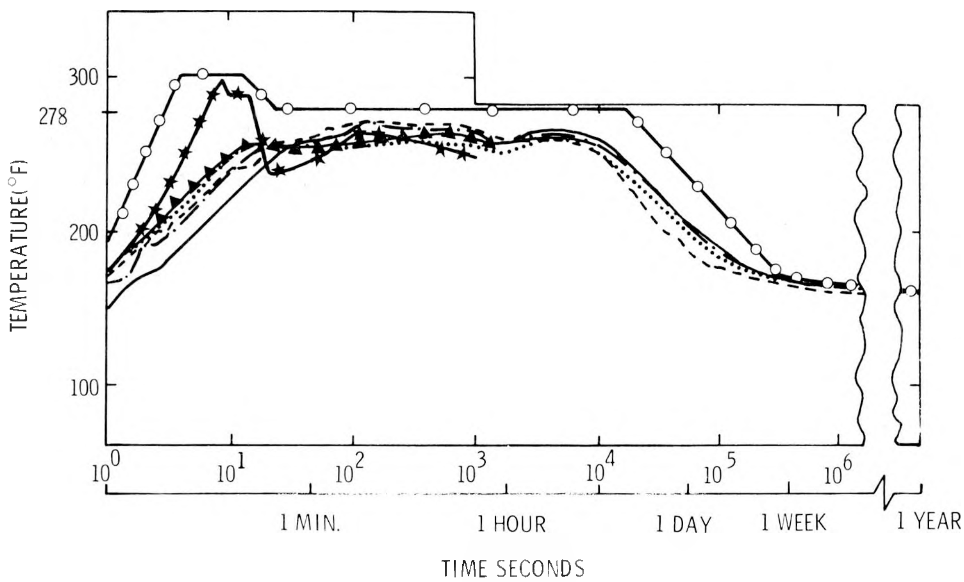

Figure 2.1b. Containment accident design envelope, temperature-time

TABLE 2.3

Accident Integrated Air Dose

Time After LOCA	Beta (rad)	Gamma (rad)	Total (rad)
1 hour	5.2(+6)*	1.1(+6)	6.3(+6)
12 hours	1.8(+7)	3.2(+6)	2.1(+7)
1 day	2.5(+7)	4.1(+6)	2.9(+7)
10 days	8.0(+7)	1.1(+7)	9.1(+7)
1 month	1.0(+8)	1.3(+7)	1.1(+8)
6 months	1.2(+8)	1.3(+7)	1.3(+8)
1 year	1.4(+8)	1.3(+7)	1.5(+8)

*Read as 5.2×10^6

These lists are computer generated, and sorts are provided as an aid to the user by (1) system, (2) location, and (3) device type. A typical example of such a list is shown in Table 2.4 (nitrogen gas system). The Class 1E equipment is composed of flow relays (FY), pneumatic valve actuators (V), and position switches (ZS); the function of this equipment is isolation (ISOL) and post-accident monitoring (PAM). Other information useful in interpreting the table is:

1. Accident code 60; LOCA without containment spray, including:
   - small primary system pipe rupture
   - steam generator tube rupture
   - rupture of a control rod drive mechanism housing;

2. For the pressure/temperature/humidity/dose columns:

   - line 1 is the best estimate of the normal (i.e., typical) operating, or service, environments
   - line 2 is the maximum realistic service environmental conditions including the accident
   - line 3 is the qualification parameters which include margin over the realistic estimates in line 2.

A very interesting feature of the list is the "realism" in the table entries; the impact of realism on qualification (capability) is great. Besides the maximum service environmental conditions, which are certainly important in qualification, the start and duration of operation are also crucial aspects. As an example, the flow relay (NG-FY-4609) must start to function 70 seconds after initiation of accident code 60 (see Table 2.4) and complete its function 1 second later. However, the architect-engineer requires qualification of this equipment for 1300 seconds; this qualification duration accounts for margin, certain unspecified accident occurrence and sensing scenarios, and general reluctance to assume the realistic, but short, duration. A purpose of this study is, in fact, to contrast the realistic and qualification parameters.

2.2.4 Generic Equipment Data Packages

A descriptive data package was compiled for each generic category of equipment. To date, a total of 46 vendors have been contacted for non-proprietary information from which 22 data packages have been prepared. The data packages contain a description of the equipment, operating principle, installation and interface details, operating and environmental conditions, performance specification, materials of construction, maintenance procedures, periodic testing and calibration procedures, vulnerability details, and manufacturer's bulletins and test reports.

JOB NO. 6602-002

CLASS 1E EQUIPMENT LIST

SORT NO. 01   SHEET 25

TABLE 2. 4

NG System Class 1E Equipment

DWG. 6602-M-500000

REV. 05

DATE 06/30/78

SYSTEM: NG - NITROGEN GAS

REV NO.	EQUIPMENT ID	SERVICE LEGEND	MFGR MODEL	LOC TRN	ACC CODE SFTY FUCT	START DUR: EST QUAL UNITS	NO OF CYCL EST QUAL	BEST EST-NORM OPER ENVIR / MAX SERV ENVIR COND / QUALIFICATION PARAM PRESS PSIG	TEMP F	HUM %	INTEGRATED DOSE B-RADS	G-RADS	DESN LIFE QUAL LIFE	IS AGING REQD	METH OF AGING	REMARKS
05	NG -FY - 13	NG-V-13 SOV	W ASCO	TUN-	60	70	500	0.0	104	95	N/A	5.3E04				
			FT8316	NEL		71		.006	148	95	N/A	1.1E06				
					ISOL	1300	1000	.006	163	100	N/A	1.2E06				
				B		SEC										
05	NG -FY - 36	NG-V-36 SOV	W ASCO	TUN-	60	70	500	0.0	104	95	N/A	5.3E04				
			LB8316	NEL		71		.006	148	95	N/A	1.1E06				
					ISOL	1300	1000	.006	163	100	N/A	1.2E06				
				A		SEC										
05	NG -FY -4609	N2 SUPPLY ISOL	W	CNTN	60	70	500	0.5	120	90	N/A	5.3E04				
		IRC NG-V37 SOV		-26		71		60	278	100	5.2E06	1.1E06				
					ISOL	1300	1000	66	293	100	5.7E06	1.2E06				
				B		SEC										
04	NG -V - 13	N2 SUPPLY ISO	W C-V	TUN-	60	70	500	0.0	104	95	N/A	2E07				PNEUMATIC VALVE
		ORC NG-V13	100	NEL		75		.006	148	95	N/A	2.1E07				ACTUATOR NOT 1E
					ISOL	1300	1000	.006	163	100	N/A	2.3E07				
				B		SEC										
05	NG -V - 36	N2 SUPPLY ISO	W	TUN-	60	70	500	0.0	104	95	N/A	2E07				PNEUMATIC VALVE
		ORC NG-V36		NEL		75		.006	148	95	N/A	2.1E07				ACTUATOR NOT 1E
					ISOL	1300	1000	.006	163	100	N/A	2.3E07				
				A		SEC										
05	NG -V - 37	N2 SUPPLY ISO	W	CNTN	60	70	500	0.5	120	90	N/A	2E07				PNEUMATIC VALVE
		IRC NG-V37		-26		73		60	278	100	5.2E06	2.1E07				ACTUATOR NOT 1E
					ISOL	1300	1000	66	293	100	5.7E06	2.3E07				
				B		SEC										
04	NG -ZS - 13	NG-V-13 POS	W NAMC	TUN-	60	CONT	CONT	0.0	104	95	N/A	2E07				
			EA170	NEL		1		.006	148	95	N/A	3.3E07				
					PAM	1		.006	163	100	N/A	3.6E07				
				B		YEAR										
04	NG -ZS - 36	NG-V-36 POS	W NAMC	TUN-	60	CONT	CONT	0.0	104	95	N/A	2E07				
			EA170	NEL		1		.006	148	95	N/A	3.3E07				
					PAM	1		.006	163	100	N/A	3.6E07				
				A		YEAR										
05	NG -ZS - 37	NG-V-37 POS	W	CNTN	60	CONT	CONT	0.5	120	90	N/A	2E07				
				-26		1		60	278	100	1.4E08	3.3E07				
					PAM	1		66	293	100	1.5E08	3.6E07				
				B		YEAR										

In some cases, because of proprietary information restrictions, UEC staff developed a generic data package based on known industry capability. Data packages have been completed for:

Limit Switches	600 Volt Control Cable
Solenoid Valves	600 Volt Power Cable
Containment Electrical Penetrations	300 Volt Instrumentation Cable
Terminal Blocks	300 Volt Thermocouple Extension Cable
Pneumatic Actuators	Motor Operated Actuators
Resistance Temperature Detectors	Level Switches
Thermocouples	Differential Pressure Switches
Terminal Blocks	Transmitters
Terminal Lugs (Manufacturer 1)	Electrical Enclosures
Wiring Ducts (Deleted)	Motors
Terminal Lugs (Manufacturer 2)	Radiation Monitors

Of these items, four are generic: penetrations, DP switches, transmitters, and motors. In addition, four more packages remain to be completed: flow relay, neutron flux element, pressure relay, and connectors.

2.2.5 Summary and Discussion

This Class 1E equipment compilation and data package development represent the first consolidated effort to completely describe the equipment item and its normal and accident environments and functions. The study provides a substantive data base for various aspects of the QTE program; in particular, it will:

- provide a basis for equipment for the LOCA simulation methodology tests;
- allow analyses of equipment functionability when exposed to the specified environments through knowledge of construction, interface, and material data;
- allow establishment of test regimen from the function, construction, size, and interface data;
- provide contrast between "realistic" and qualification parameters;
- allow the detailing of optimum typetest envelopes and procedures based on realistic parameters.

2.3 References

2.1 F. V. Thome, "Preliminary Data Report, Test IX. Simultaneous Mode; Cables, Splice Assemblies, and Electrical Insulation Samples," SAND78-0718, Sandia Laboratories, Albuquerque, NM, April 1978.

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

2.3 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."

2.4 Letter dated June 27, 1978, R. J. Gibson (FIRL) to L. L. Bonzon (SLA), " FIRL Comments and Review of Test Facility Upgrade Proposal."

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

2.6 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.)

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

2.8 L. L. Bonzon and D. W. Dugan, "Preliminary Test Facility Upgrade Proposal," Sandia Laboratories Internal Memorandum, prepared for review by Burns & Roe, Inc., November 1977.

2.9 UEC, Inc., J.O. 6602-002. Submittals to Sandia Laboratories under Contract 07-3426.

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

Publications and Presentations: Final distribution of SAND78-0091, "Definition of Loss-of-Coolant Accident Radiation Source: Summary and Conclusions,"[3.1] was made in May. This report, prepared by IRT Corporation under contract to Sandia Laboratories, details the radiation energy release rates and spectra corresponding to the sources specified in Regulatory Guide 1.89,[3.2] for the radiation qualification of Class 1E equipment. The effects of several parameters not completely specified in the Guide, such as reactor fuel composition, operating duration and power level, and treatment of the progeny, were evaluated. The results are presented as time-dependent beta and gamma ray energy release rates and spectra, and are fundamental quantities that are generally applicable to any nuclear power station. The full report, SAND78-0090,[3.3] is principally useful as a source of additional data.

A paper was presented by N. Lurie at the 1978 Annual Meeting of the American Nuclear Society, San Diego, on June 21.[3.4] It describes calculations of the time-dependent LOCA radiation signature (as specified in Regulatory Guide 1.89), and calculations of depth-dose damage profiles in Class 1E components which can be used to evaluate simulator adequacy.

IRT Subcontract--1: A subcontract with IRT Corporation to perform parametric calculations of the LOCA radiation signatures and depth-dose calculations in selected Class 1E components from the LOCA and simulators signatures was effectively completed this quarter. Phases 1-4 results were documented as References 3.1 and 3.3. A draft report on Phases 5-6 was completed in April for Sandia review and comment.[3.5] These latter phases involve the calculation of depth-dose and charged-particle distribution profiles in modeled equipment exposed to radiation from the LOCA source and simulator sources. The calculations will be used to evaluate simulator adequacy as discussed in Reference 3.4. Comments on the Phases 5-6 draft were forwarded to IRT by letter dated May 26. These are currently being addressed by IRT staff, with a final report to be issued in July. Receipt of the draft concludes the subcontract, except for incorporation of the comments and corrections which will be accomplished under Subcontract--2.

IRT Subcontract--2: Based on previous subcontract experience and the developed calculational techniques, a contract was awarded to IRT Corporation in late March to develop a "best-estimate" LOCA radiation signature. The signature will be generally based on the accident-time-release sequencing as specified in WASH-1400[3.6] and other recent literature. (See also Reference 3.7.)

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 effectively extends the contract through September, and requests IRT to provide additional assistance to supplement, expand, update, and reevaluate prior submittals as deemed necessary.

A preliminary draft of the Phase 1 report was discussed with IRT staff on June 20 at San Diego. Based on these discussions, a comment copy will be forwarded to Sandia in early July.

By letter dated June 26, IRT was requested to perform a LOCA-radiation damage assessment of electric cables and an assessment of radiation simulator adequacy to be based on the previous work of References 3.1, 3.5, and 3.8. This additional work is limited in scope and is to be completed by mid-August. Sandia personnel are performing a concurrent complementary assessment effort.

Generic Containment LOCA-Radiation Profiles: NRC staff (specifically NRR and Standards) are developing a user's requirement letter for a proposed study to estimate generic in-containment LOCA radiation doses and dose rates to Class 1E equipment. At a May 12 meeting with NRC staff, L. Bonzon outlined the applicable work under NRC sponsorship completed, or in progress. It is expected that a funded study to be programmed by RSR will be requested by NRR/Standards in the near future.

Expressions of (unsolicited) subcontractor interest in performing this work have been received from IRT, SAI, and Stone & Webster. Meetings with these companies were held on June 20 and 21 in San Diego to discuss the implementation of this new effort. The work may be directed towards defining some generic plant areas for which dose and rate calculations could be made; these would include open plant areas, shielded (labyrinth) rooms, drywell, congested piping area, nearby structure (walls) area, etc. Computational methodology, as well as typical calculations, could then be defined for each area.

3.2 Calculations to Support Radiation Simulators Adequacy Assessments*[3.4,3.5]

The LOCA radiation sources appropriate for Class 1E equipment qualification testing have been specified in Regulatory Guide 1.89.[3.2] These sources are divided into two source strengths (one for containment heat-removal systems, etc., and one for other safety-related electric systems) and three spatial distribution categories: (1) airborne

*This section prepared by L. Bonzon, Division 5432.

(uniformly dispersed in the containment atmosphere); (2) plate-out (plated-out on surfaces exposed to the containment atmosphere); and (3) waterborne (initimately mixed with coolant water). (See Table 3.1.)

TABLE 3.1

Guide Sources Specifications[3.2]

Source 1 (Containment heat-removal systems, etc.)	Airborne	100% Noble gases, 25% Iodines
	Plate-Out	25% Iodines, 1% Solids
	Waterborne	50% Halogens, 1% Solids
Source 2 (Safety-related electrical systems)	Airborne	10% Noble gases (except Kr-85), 30% Kr-85, 5% Iodines
	Plate-Out	5% Iodines
	Waterborne	10% Halogens

These sources have been redefined in terms of energy release rate and particle/energy spectra as functions of time and have been reported in References 3.1, 3.3, and 3.4.

With the LOCA sources signatures established, energy and charge deposition profiles can be calculated for the sources and for common radiation simulators. These calculations can then be used to support the assessments of simulators adequacy. Initial calculations have been completed for a generic electric cable and for Co-60 and Cs-137 simulators by IRT Corporation; the results of these calculations are summarized in the succeeding subsections.

3.2.1 Cable Model and Methods of Calculation

Many types and brands of electric cable are used in nuclear power plants but a typical Class 1E type is a 600-V power cable consisting of a copper core with elastomer insulator and jacket (Figure 3.1). Typically, the dimensions and materials vary widely; thus, the model selected may be considered as typical, but not exactly like any one manufacturer's brand in particular. In general any organic, CH_2-dominated materials would yield similar calculational results if material thickness and density are appropriately considered.

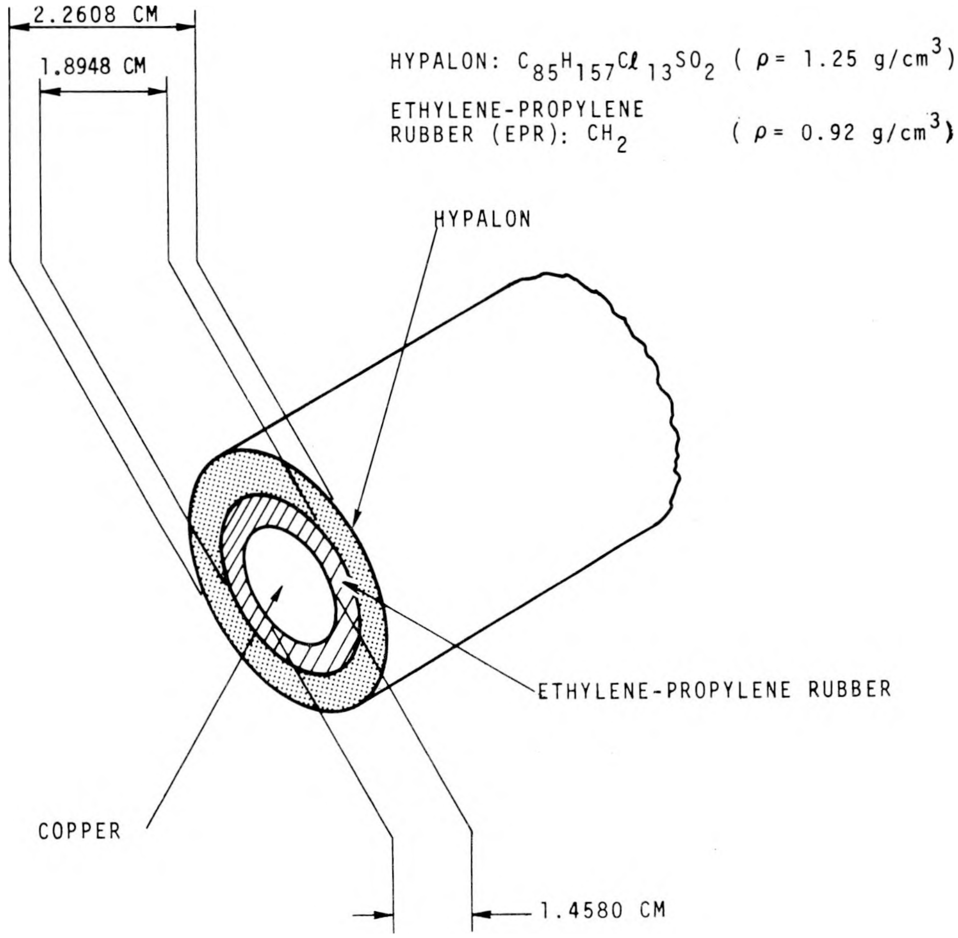

Figure 3.1. Model of the power cable

For the LOCA-sources calculations, the cable was assumed to be centered in the sources volume (for the waterborne and airborne sources) and the plate-out source was assumed to be uniformly deposited on the cable jacket surface. Furthermore, the airborne source was taken to consist of a uniform distribution of point sources in 70-psig saturated steam. The spatially extended source distributions are not tractable to direct brute-force Monte Carlo transport methods. The reason for this is that the cable subtends a very small solid angle; in other words, the probability that a source particle sampled from the air will hit the cable is too small to get statistically meaningful results in a reasonable time. Consequently, an approximation technique was used to map the distributed airborne and waterborne sources into a practical configuration.

The simulator sources were assumed to be isotropic emitters on the surface of a cylinder radius of 12 in. (30.48 cm), coaxial with the cable. Air at standard temperature and pressure (STP) with a density of 0.001293 g/cm^3 filled the space between source and cable. A test chamber is typically used to contain the test piece and its environment. The effect of a 1/4-in. (0.64-cm) steel test chamber with an inside radius of 10 in. (25.4 cm), also coaxial with the cable, was examined; the test chamber has little effect. For Co-60, the average photon energy penetrating the test chamber is 1.156 MeV compared to 1.253 MeV without the chamber.

The calculations of energy and charged-particle deposition were carried out using SANDYL, a general and widely recognized coupled photon-electron transport code. It uses the physical approach which consists of random sampling and simulation of individual histories which are used to construct the solution to the physical problem.

3.2.2 Calculations and Results

Selections from the source spectra reported in References 3.1 and 3.3 were made for the depth-dose studies in the model cable. The results for an equilibrium irradiation for Sources 1 and 2, which included all daughters but did not include capture and depletion effects, were used. Cooling times producing the hardest (i.e., highest average energy) and softest spectra were selected in order to study the bounding conditions. The hardest spectrum occurs at the shortest cooling time. As a matter of practicality, the very short cooling times are not expected to be accurate since a significant number of nuclides which contribute to the energy release at short times do not have spectral data in the library. Therefore, the one-minute cooling time data were selected for the hard spectrum. The softest spectrum (i.e., lowest average energy) occurs between 1 and 10 days depending on the type of source; the 4-day results were selected. For each source and each distribution category, depth-dose calculations were carried out at two times for betas and gammas.

The charged particle distribution is shown in Table 3.1. Net charge deposition in each layer is shown for Co-60 and for each of the sources investigated in this work. Charge deposition is interpreted as the instantaneous charge deposited for the various identified fission product sources. No mechanisms for leakage or other forms of dispersal of the charges have been included. The gamma-ray sources (including Co-60) consistently produce a positive charge on the outer layer of the jacket (Hypalon), but no regular pattern is apparent on the interior of the cable. On the other hand, the beta sources produce a much larger negative charge on the outermost region of the cable; this falls off with decreasing radius in a much more regular pattern. For the case of the cable surrounded by water, the charge buildup is about two orders of magnitude smaller. Of course, in a combined beta-gamma radiation field, there will be some compensation of the opposite signed charge depositions. Interpretation of these results and their importance insofar as failure of the cable is concerned must await analysis of the mechanisms of damage in the cable.

TABLE 3.2

Charge Deposition in the Cable

Net Electrons Deposited per 10^3 Source Gamma Rays

Material	Zone Outer Radius (mm)	^{60}Co	Airborne Source 1 1 min	Airborne Source 1 4 days	Airborne Source 2 1 min	Airborne Source 2 4 days	Plate-Out Source 1 1 min	Plate-Out Source 1 4 days	Plate-Out Source 2 1 min	Plate-Out Source 2 4 days	Waterborne Source 1 1 min	Waterborne Source 1 4 days	Waterborne Source 2 1 min	Waterborne Source 2 4 days
Hypalon	11.304	-5.10	-1.30	-0.30	-1.70	-0.45	-9.15	-8.15	-8.15	-5.00	-0.95	-0.85	-1.60	-0.75
"	10.847	-2.35	-0.45	0.10	-0.80	0.15	-0.55	0.25	-1.95	-0.80	-0.40	-0.15	0.35	0.05
"	10.390	0.45	0.40	-0.05	0.20	-0.05	1.25	0.50	0.15	-0.35	0.60	0.20	-0.15	0.15
"	9.932	0.70	-0.65	0.30	0.35	0.15	0.25	0.80	-0.90	1.55	-0.30	0.10	0.15	-0.20
EPR	9.474	0.10	0.25	-0.05	-0.10	0.15	-0.65	-1.20	0.75	-1.25	-0.10	0.10	0.25	0.35
"	8.928	-0.35	0	-0.25	0.15	-0.15	-1.05	0.75	-0.35	0.90	0.50	-0.10	-0.05	-0.25
"	8.382	-0.30	-0.15	0.25	-0.15	-0.20	-1.05	0.25	0.05	-0.70	-0.20	0.05	-0.25	-0.10
"	7.836	0	-0.50	-0.10	-0.40	0	0.95	-0.60	0.35	-0.35	-0.25	-0.25	-0.20	0.10
Copper	7.290	-0.60	0	0.10	0.25	0.20	0.20	0	-0.25	0.45	-0.15	-0.15	0.25	0.05
"	5.832	-0.25	0.25	-0.10	-0.10	-0.20	0.55	0.60	0.60	0.30	0.35	0.10	0.10	0.05
"	4.374	1.50	0.15	0.05	-0.05	0.05	-0.30	-0.30	0.10	-0.10	-0.25	-0.05	0	-0.05
"	2.916	-0.30	-0.15	-0.05	0.20	-0.05	0	0.15	-0.40	0	0.05	0.05	0.05	0
"	1.458	-0.15	0.10	0	-0.05	0.05	0	-0.05	0.25	0	-0.05	0	0	0

Net Electrons Deposited per 10^3 Source Betas

Material	Zone Outer Radius (mm)	^{60}Co	Airborne Source 1 1 min	Airborne Source 1 4 days	Airborne Source 2 1 min	Airborne Source 2 4 days	Plate-Out Source 1 1 min	Plate-Out Source 1 4 days	Plate-Out Source 2 1 min	Plate-Out Source 2 4 days	Waterborne Source 1 1 min	Waterborne Source 1 4 days	Waterborne Source 2 1 min	Waterborne Source 2 4 days
Hypalon	11.304	-5.10	60.90	175.80	75.60	169.80	350.20	363.30	357.30	535.80	2.94	2.01	2.76	8.09
"	10.847	-2.35	29.00	10.70	29.30	11.90	114.50	116.70	119.60	65.50	1.34	0.40	0.88	1.08
"	10.390	0.45	22.60	0.60	22.70	2.00	60.50	61.33	56.50	11.50	0.84	0.15	0.40	0.27
"	9.932	0.70	17.30	0	17.10	0.20	34.90	39.00	34.00	3.40	0.46	0.05	0.24	0.02
EPR	9.474	0.10	14.20	0	14.90	0.10	19.80	17.33	17.90	0.60	0.33	0.06	0.11	0.01
"	8.928	-0.35	12.30	0	9.80	0	9.70	12.00	9.30	0.20	0.19	-0.01	0.03	0.01
"	8.382	-0.30	9.10	0	7.20	0	7.10	4.33	7.70	-0.10	0.11	0.02	0.03	0
"	7.836	0	8.20	0	5.20	0	4.80	3.33	3.20	0.10	0.13	0.02	0.02	0
Copper	7.290	-0.60	26.40	0	16.10	0	9.10	10.33	8.20	0	0.26	0	0.01	0
"	5.832	-0.25	0.50	0	0	0	0.10	0	0.10	0	0	0	0	0
"	4.374	1.50	0	0	0	0	0.10	0	0	0	0	0	0	0
"	2.916	-0.30	0	0	0	0	-0.10	0	0	0	0	0	0	0
"	1.458	-0.15	0	0	0	0	0	0	0	0	0	0	0	0

Results of the depth-dose calculations for the Source 1 and simulator spectra are shown in Figures 3.2 through 3.7. Since the shapes of the depth-dose profiles are of primary interest here, the simulator results were normalized to the LOCA calculations to give the same dose in the outer layer of the jacket. Clearly, the gamma (LOCA) sources are reasonably uniform across the insulation and jacket. Just as clearly, the limited penetration power of the betas results in steep depth-dose gradients.

3.2.3 Discussion

If equivalence of these profiles is to be used as the only basis for judging simulator adequacy, then it must be concluded that these isotopic sources are inadequate; the most serious discrepancy is with the beta sources, as expected. The penetrability and energy loss characteristics of beta particles are very different from those of gamma rays, and thus it is not surprising that mono-energetic gamma sources provide a poor simulation of betas.

However, before judgments on simulator adequacy can be made, it is necessary to understand and evaluate the damage and failure mechanisms due to radiation for each piece of Class 1E equipment. This is the subject of a continuing effort and one on which simulator adequacy will be ultimately judged; the calculations represented in this section are only part of the basis for adequacy assessment.

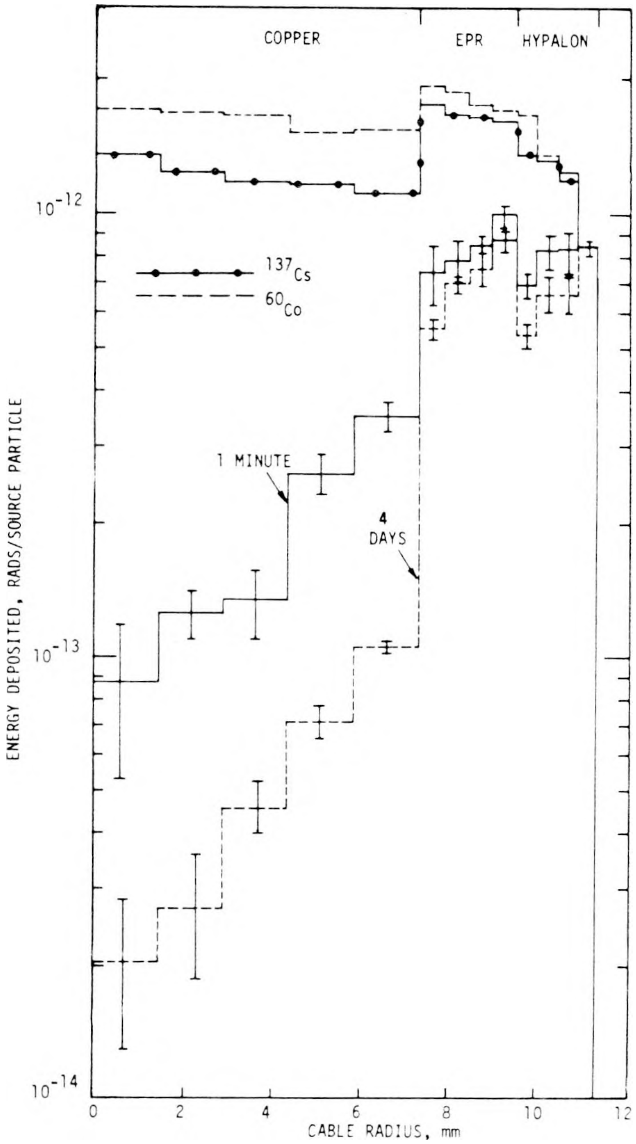

Figure 3.2

Comparison of depth-dose for Co-60 and Cs-137 with airborne gamma rays (Source 1). The isotopic source results have been normalized at the outer zone (surface) of Hypalon

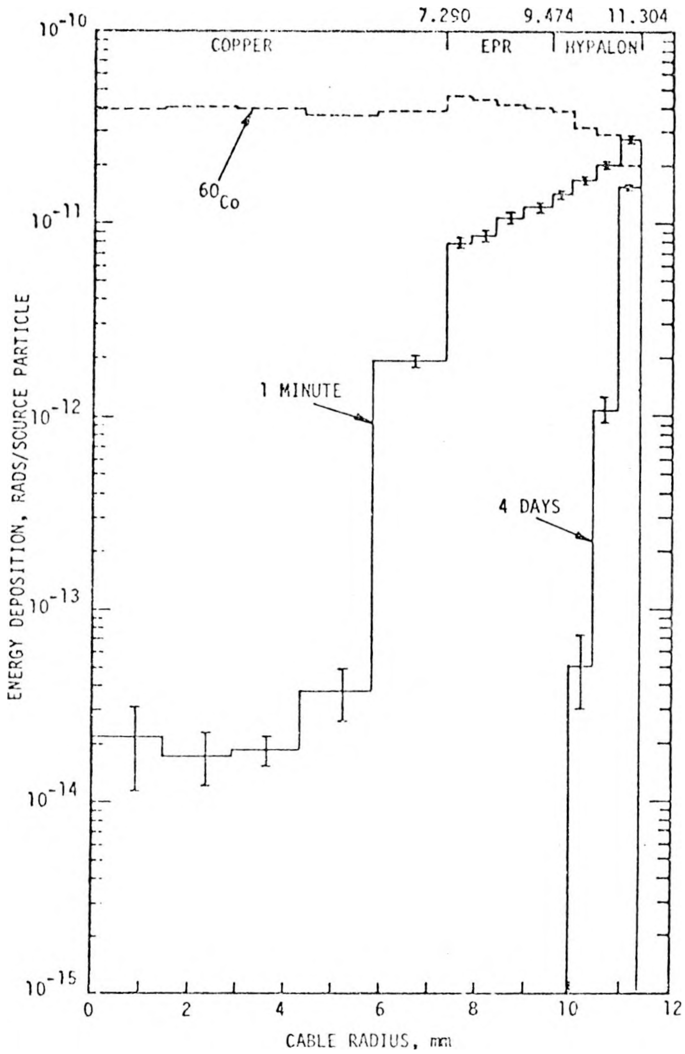

Figure 3.3

Comparison of depth-dose for Co-60 with airborne betas (Source 1); Co-60 result normalized to the average dose at the surface

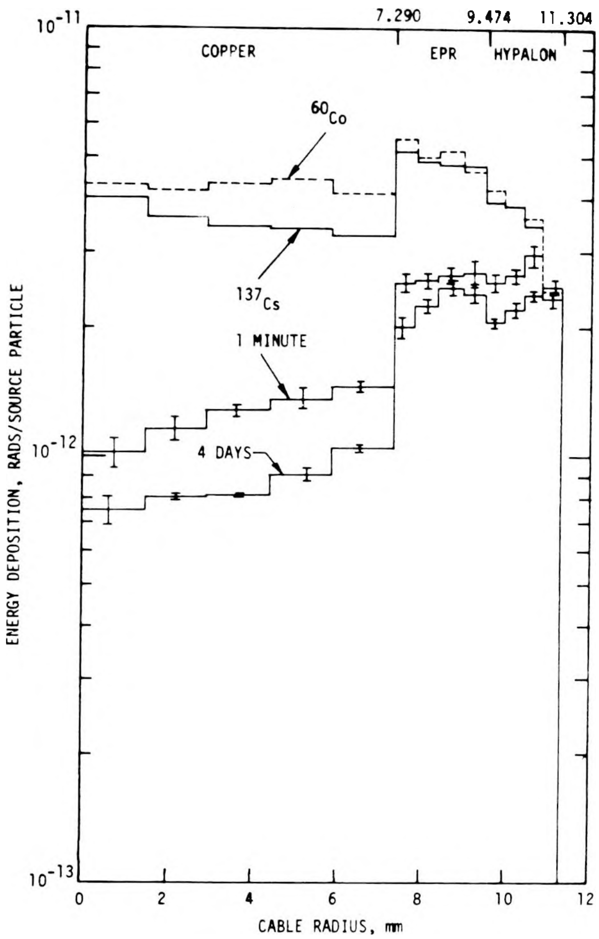

Figure 3.4

Comparison of depth-dose for Co-60 and Cs-137 with plate-out gamma rays (Source 1). Both the Co-60 and Cs-137 results have been normalized to the average surface dose

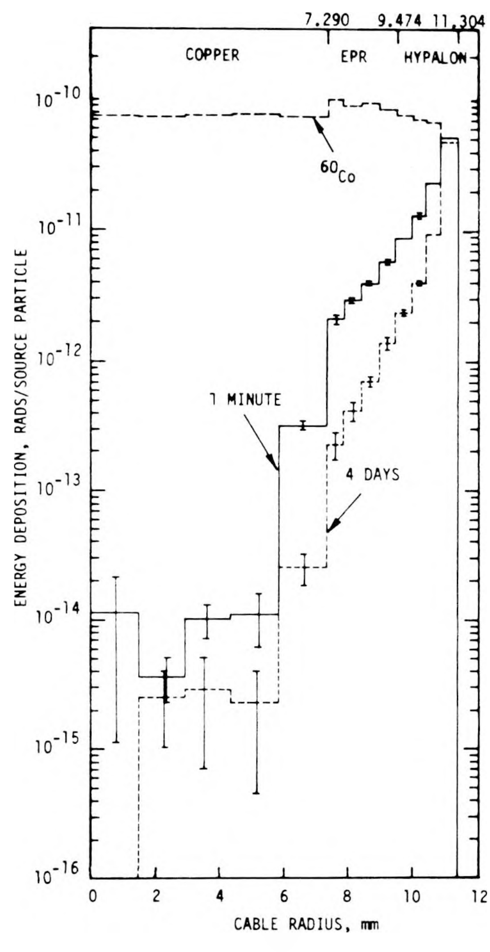

Figure 3.5

Comparison of depth-dose for Co-60 (normalized at average dose in surface zone) with plate-out betas (Source 1)

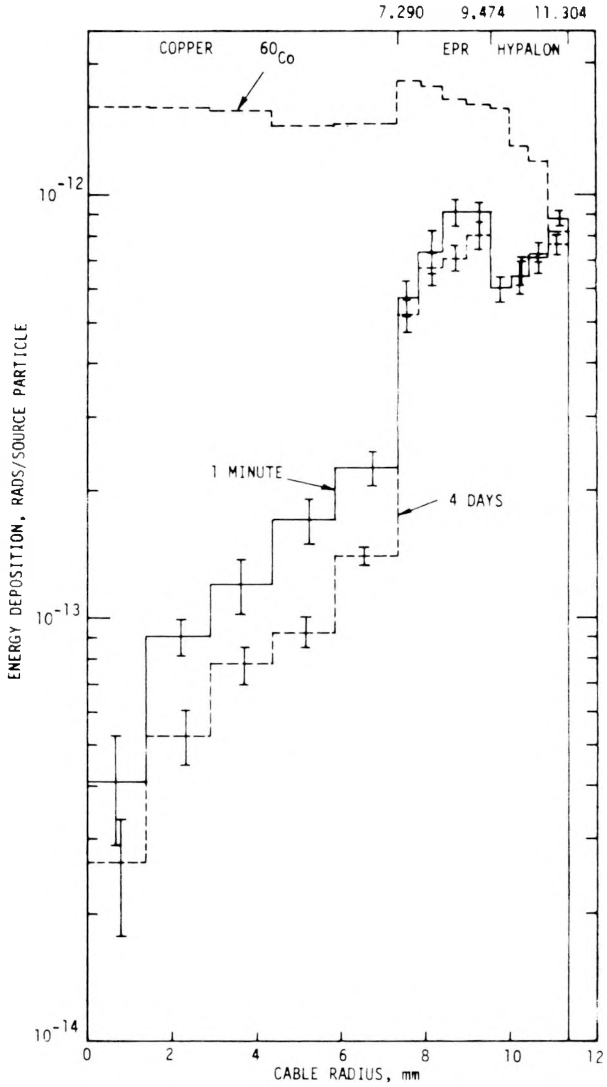

Figure 3. 6

Comparison of depth-dose for Co-60 (normalized to average dose in surface zone) with waterborne gamma rays (Source 1)

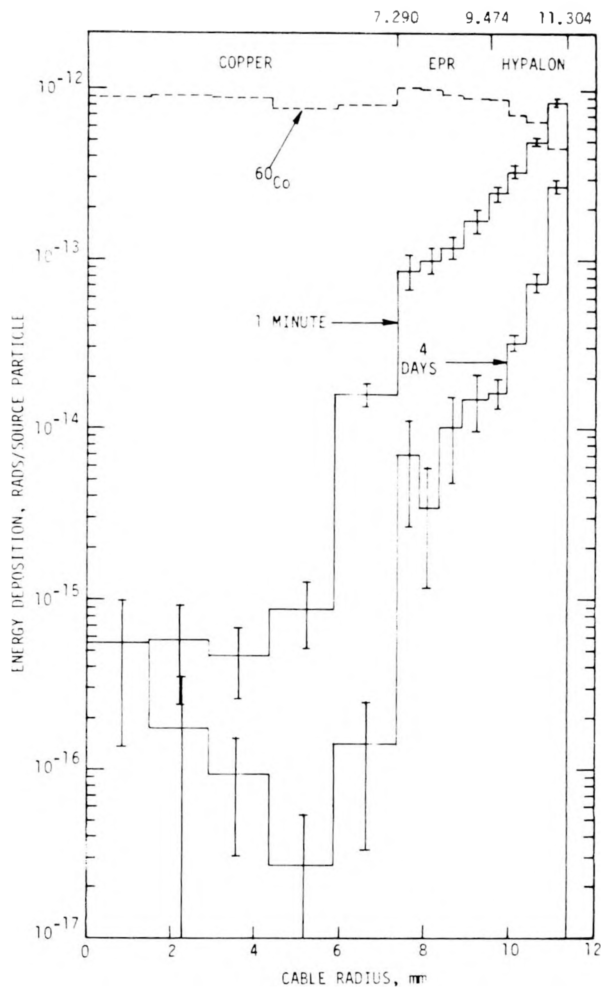

Figure 3. 7

Comparison of depth-dose for Co-60 (normalized to average dose in surface zone) with waterborne betas (Source 1)

3.3 References

3.1 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/IRT 8167-004, Sandia Laboratories, Albuquerque, NM, May 1978.

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

3.3 L. L. Bonzon, N. A. Lurie, D. H. Houston, and J. A. Naber, "Definition of Loss-of-Coolant Accident Radiation Source," SAND78-0090/IRT 8167-002, Sandia Laboratories, Albuquerque, NM, February 1978.

3.4 N. A. Lurie, J. A. Naber, and L. L. Bonzon, "Adequacy of Radiation Sources for Qualification of Class 1E Reactor Components," IRT 8167-003/SAND78-016A, Transactions of the American Nuclear Society, Vol 28, 1978.

3.5 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.)

3.6 "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.

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

3.8 L. L. Bonzon, "Radiation Signature Following the Hypothesized LOCA," SAND76-0740/NUREG 76-6521, Sandia Laboratories, Albuquerque, NM, October 1977 (Revised).

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 Summary

Publications and Presentations: A paper entitled "Computer Modeling of Polymer Radiation Chemistry,"[4.1] was presented by R. Clough at the Fifth International Symposium on the Chemistry of the Organic Solid State, Brandeis University in June; it will also be published in the proceedings of the symposium. The computer program duplicates the structural changes that occur during irradiation of polyethylene with input variable in terms of temperature, irradiation rate, and details of the initial material composition.

A full paper entitled, "A Model for Combined Environment Accelerated Aging Applied to a Neoprene Cable Jacketing Material,"[4.2] was submitted in June to the 1978 Conference on Electrical Insulation and Dielectric Phenomena, to be held October 29-November 2, 1978, in Pocono Manor. In the paper, a model potentially applicable to combined environment aging and methods for carrying out accelerated aging in combined environment situations is discussed; data on single and combined environment aging of neoprene are presented and analyzed using the proposed model.

Alternate Damage Indicators: Two methods to evaluate electric cable "age" using active on-line electric measurements were initially evaluated during this quarter.

Based on LOCA-simulation testing,[4.3] it was found that the dissipation factor changes as cable is aged. (The dissipation factor is the ratio of loss current to charging current in the dielectric and indicates the quality of the "capacitor.") Following this test, arrangements were made with an equipment supplier to supply test equipment for evaluation. Preliminary testing of cables in ambient- and elevated-temperature environments indicated that the method has potential but requires further refinement and automation of the test equipment.

A separate technique to measure dielectric conductivity (the product of angular frequency and loss factor) was also evaluated.[4.4] Using digital processing techniques and fast Fourier transform analysis, the technique was evaluated using test equipment supplied by an equipment supplier. Loss could be determined in the cable, but not dispersion of the loss, because of equipment and computer program limitations. Further refinement of the test apparatus is underway by the supplier and the technique will be reevaluated.

Fire-Retardant Aging Program: The study to investigate the effects of aging on fire retardants in electric cable is progressing. Two polymers, ethylene propylene terpolymer and chlorosulfonated polyethylene, commonly used in cables for nuclear power generating stations, have been master-batch compounded into representative rubber formulations. Four fire retardants are being evaluated. Materials with and without fire retardants are being aged in thermal, radiation, and combined thermal-radiation environments. A contract has been let to the Smithers Laboratory to perform combustion tests using the Ohio State Rate Release Calorimeter. Combustion tests are scheduled to begin in late August. Simultaneous analytical/chemical tests will be made on the samples to measure fire-retardant retention.

Embrittled Polyethylene Cable Evaluation: Accelerated aging experiments on cables obtained from the Savannah River Plant reactor building have been essentially completed. 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. Although the results are still being evaluated, preliminary analysis indicates that strong synergistic effects exist both for the polyethylene insulation and for the polyvinyl chloride jacket.

NRL Subcontract: Since April 1977, the Naval Research Laboratories (NRL) has been subcontracted to provide radiation services related to the accelerated aging study. A number of modern cable insulation and jacketing materials have been aged in various combinations of radiation, temperature, and humidity. The subcontract was extended through this fiscal year to provide a test facility complementary to the Sandia (LICA) irradiation facility. An additional series of dose rate, humidity, thermal, and combined aging experiments were completed this quarter and are currently being evaluated.

Computer Calculations on Polyethylene Radiation Degradation: 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. A presentation describing the methodology of the program and the results obtained to date was given at the Fifth International Symposium on the Chemistry of the Organic Solid State at Brandeis University, June 12-16.[4.1]

Low Intensity Cobalt Array (LICA) Facility: During the past quarter, a number of aging cells have been constructed and tested in the new radiation aging facility (a more complete description of this facility is in Section 4.2). Room temperature radiation aging exposures have been applied to (1) a PVC jacketed, polyethylene insulated cable (from Savannah River) at two different dose rates; (2) various commercial cable material (EPR, Hypalon, Tefzel®); and (3) sheets of specially formulated EPR and Hypalon materials which will be used to investigate fire-retardant aging. Combined environment aging exposures have been applied to the Savannah River cables at 80°C and 4 krad/hr in both air and inert (nitrogen or argon) environments. In addition, a neoprene material has been aged in eight different combined environments in an attempt to generate sufficient data to make an early determination of the utility of the proposed combined environment accelerated aging method.

Thermal Aging Facility: Since the completion of this new facility (see Section 4.2), a number of materials have been or are being aged at various temperatures ranging from 363°K to 423°K. The materials include a chloroprene jacket, three chlorosulfonated polyethylene materials, a CLPO insulation, and an EPR insulation. In addition, unused Savannah River cable has been aged at 80°C in an air-circulating oven.

4.2 Aging Facilities*

In order to carry out certain aspects of the extensive experimental work proposed for the aging program, two versatile aging facilities have been constructed. The first is a low-intensity radiation-aging facility (LICA) which allows combined environment accelerated aging to be carried out under selected conditions of radiation dose rate, temperature and gaseous atmosphere. The second facility is useful for carrying out

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*This section prepared by K. Gillen, L. Jones, and E. Salazar, Division 5813.

thermal aging under various gaseous environments and gas flow conditions. Detailed descriptions of these facilities and their capabilities will be given in this section.

4.2.1 Low Intensity Radiation Aging Facility

Figure 4.1 shows an artist's rendition of the radiation facility. Approximately 9000 curies of Co-60 is positioned at the bottom of a water-filled concrete tank whose dimensions are 4 ft wide by 8 ft long by 15 ft deep. Radiation aging is carried out in water-tight test cells by lowering the cells to the bottom of the tank. Thirteen feet of water, separating the Co-60 from experimenters at the top of the tank, provides radiation shielding. A water level control system and various radiation level monitors connected to an alarm system provide backup safety.

Details of the radiation aging portion of the facility are shown in Figure 4.2. Two parallel channels support the Co-60 holder, air tanks, and the test cell holders. The Co-60 holder, configured similar to a linear test tube rack, has 58 equally spaced holes each capable of containing a single 0.45 in. diameter pencil of Co-60. Currently, twenty five 12-in. long Co-60 pencils (approximately 9000 total curies) are fairly evenly distributed along the holder. The extra 33 holes allow flexibility in minimizing dose rate gradients parallel to the cobalt holder and could be used as high-dose-rate locations for small samples. Test cell holders, each containing 4 cylindrical holes, are oriented parallel to the linear cobalt holder and are located on both sides of the source at various distances from it. Test cells placed in any of the holes of a given holder receive comparable radiation dose rates. The particular dose rate is governed by the distance from the cobalt array and by the shielding between the test cell and the array. Gradients in dose rates occur in the individual test cells; for example, there is a dropoff in dose rates between the parts of the cell closest to, and furthest from, the cobalt. Minimizing these radial gradients requires moving the cell as far from the cobalt as possible. At the same time, it is desirable to minimize the amount of Co-60 required for a given dose rate at a given distance; this necessitates a minimum of shielding between the test cells and the cobalt. These two requirements were satisfied by using air tanks for filling any space between the Co-60 holder and the test cell holders and by designing air-filled (water-tight) test cell holders. This arrangement not only minimizes gradients across the test cells for a given dose rate, but also minimizes the steps in dose rate occurring among adjacent test cell holders.

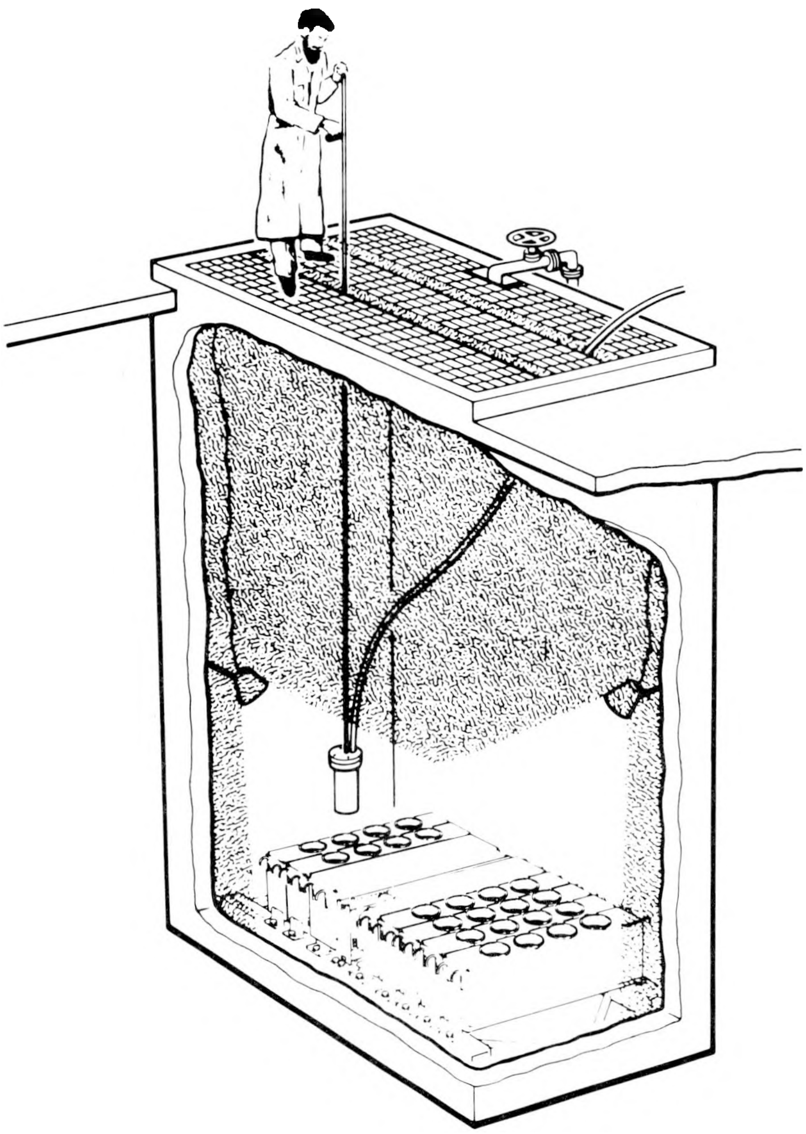

Figure 4.1. Low-intensity cobalt array (LICA) test facility

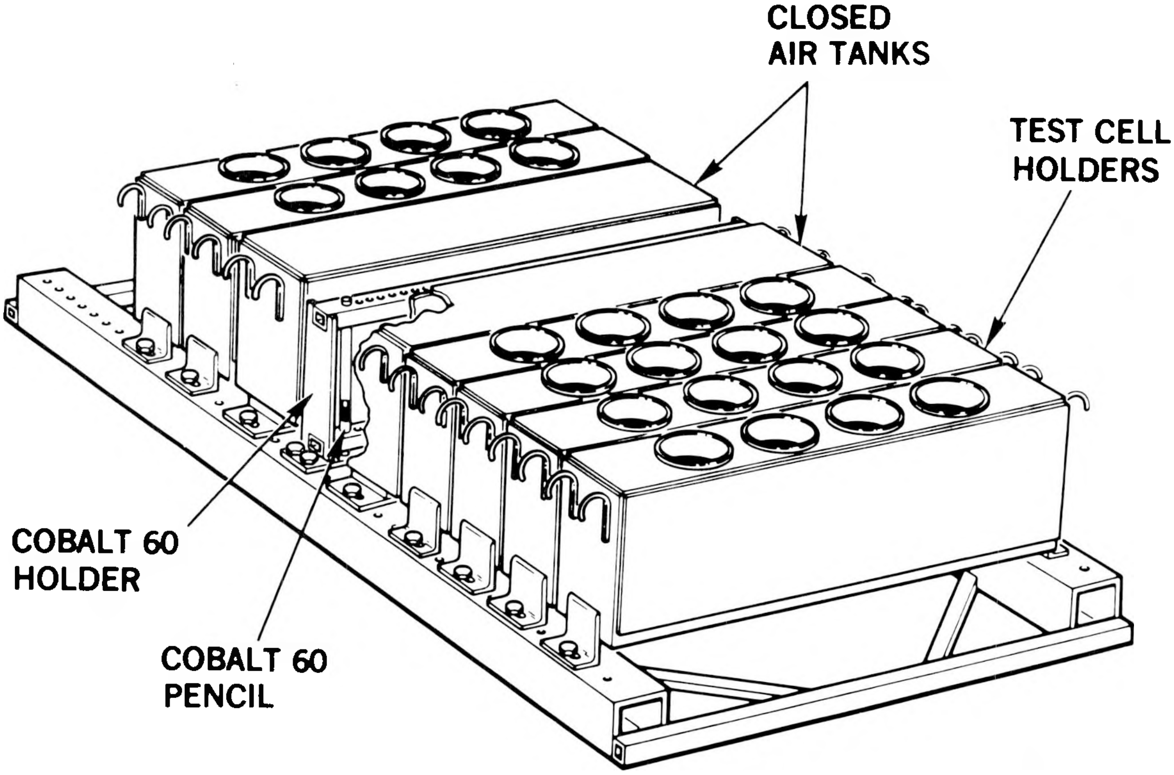

Figure 4.2. Linear array radiation source and test chamber arrangement

Another feature of the design is the flexibility to reposition the test cell holders along the track to select appropriate dose rates for a given experiment. This also implies that new test cell holders capable of holding bigger size test cells could be easily incorporated into the apparatus.

A detailed sketch of a test cell is shown in Figure 4.3. The cylindrical sample aging region (4-in. diameter and 7 in. long) is located inside a brass can, which is in turn suspended from the lid of a double-walled stainless-steel can. (The double-walled design provides some thermal insulation.) When the stainless-steel lid is lifted free of the stainless-steel can, a brass lid at the bottom of the brass can is easily removed, allowing access to the sample aging region. The sample region is heated using an insulated nichrome wire wrapped around the brass can and around a pancake heater which lies between the brass can and the stainless-steel lid. The auxiliary pancake heater was found to be useful for reducing rather substantial temperature gradients at the top of the sample aging region caused by the heat-sink effect of the massive top of the stainless-steel can. To accurately control and monitor the temperature in the sample region, two resistance temperature devices (RTDs) were incorporated in the design. A control RTD is directly clamped to the inside wall of the brass can and a monitor RTD is positioned close to the center of the sample aging region. A 1-1/8 in. OD plastic tube runs from the top of the stainless-steel can up through the water tank. Lead wires for the heater and the two RTDs feed through this tube. In addition, a small tube inside the outer plastic tube is used to circulate air or other gaseous environments past the sample region at controlled flow rates. The temperature capabilities of the test cells are room temperature to $423^{o}K$; with slight modification it should be possible to reach $523^{o}K$. At $423^{o}K$, long term stability and accuracy of the temperature was found to be better than $\pm 0.3^{o}K$. Measured temperature gradients across the sample aging region were less than $\pm 1.0^{o}K$.

Figure 4.4 shows the current arrangement of the test cell holders as viewed from the top of the tank. Three adjacent test cell holders are located as close as possible to one side of the cobalt source. On the other side, a 9-in. wide air tank separates three adjacent test cell holders from the cobalt. The radiation doses for the various test cell locations in this arrangement were measured using TLD-400 dosimeter chips. During the mapping, test cells filled all 24 aging locations, a situation similar to that anticipated during actual aging exposures. Analysis after exposure of a single TLD chip leads to an uncertainty in the estimated dose rate of ± 10%. However, by using numerous chips distributed throughout a given test cell, dose rates can be determined with greater precision. A large number of readings in a given cell allow estimates to be made of dose rate gradients in the sample aging region.

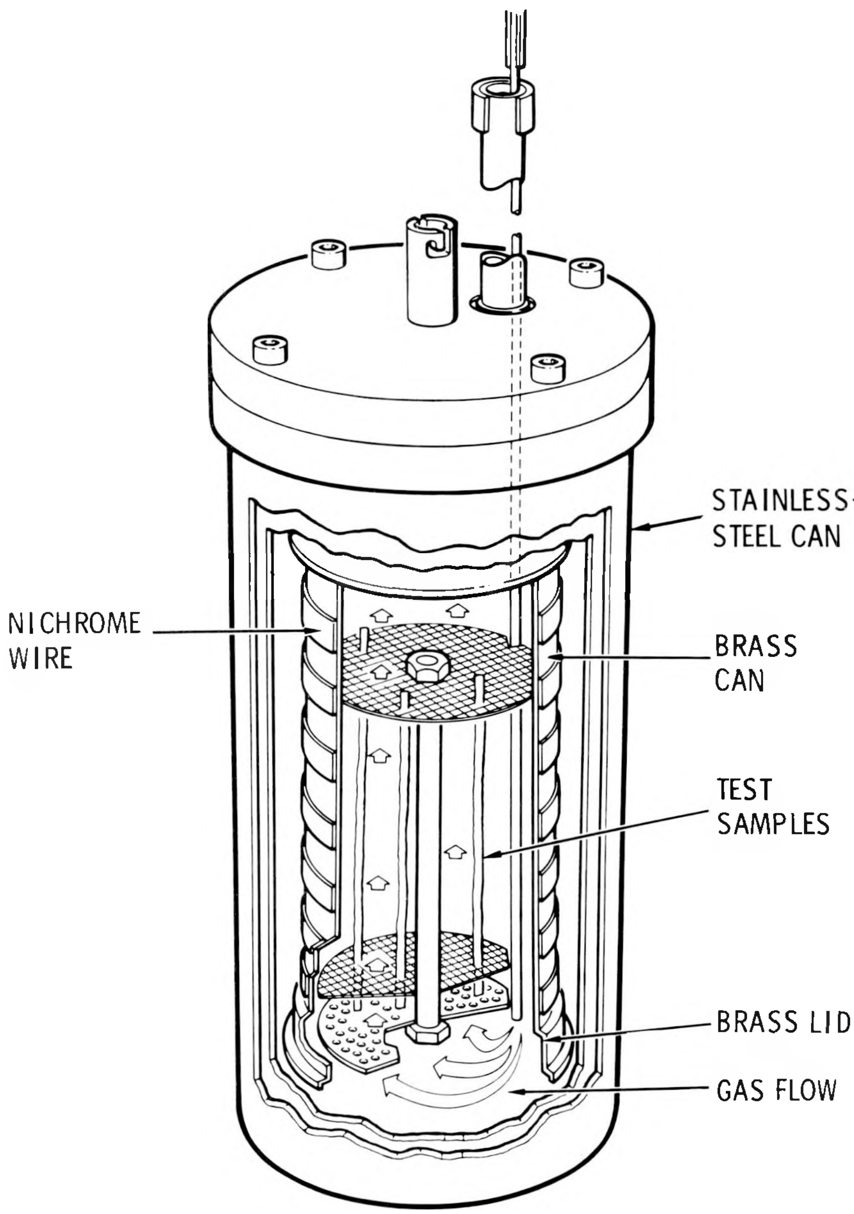

Figure 4.3. Typical heater test cell

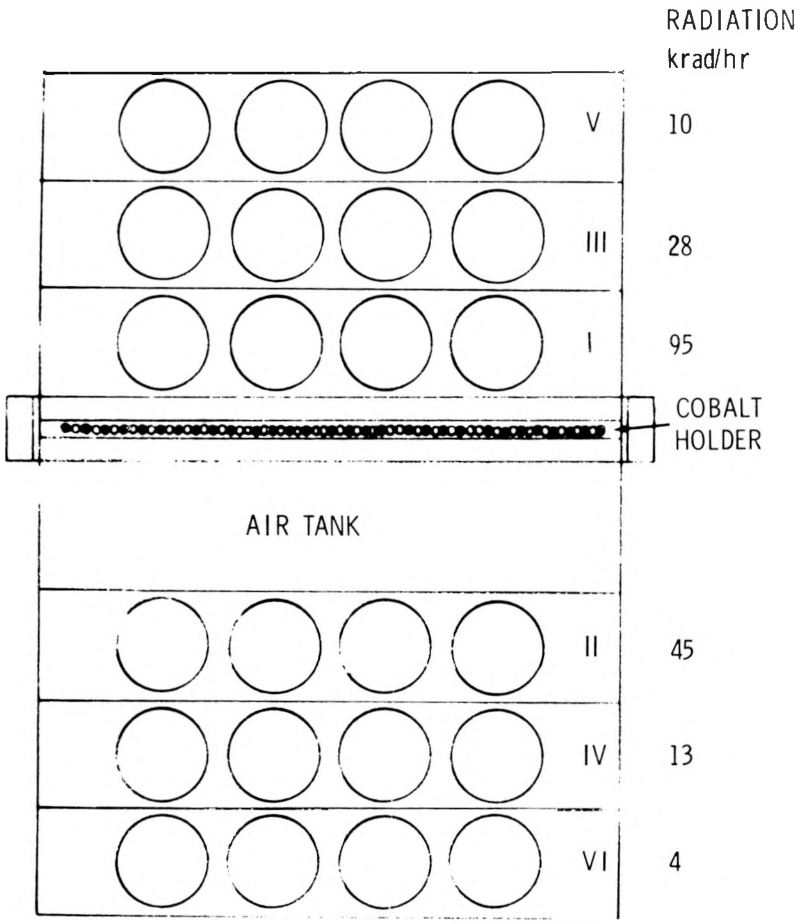

Figure 4.4. Current LICA arrangement with approximate centerline dose rates noted for each test cell holder

Some of the mapping results are summarized in Figures 4.4 through 4.7. Figure 4.4 shows the approximate dose rates found for the 6 test cell locations. The dose rates range from 95 krad/hr to 4 krad/hr in roughly five equal steps. The test cell holders are labeled with Roman numerals I through VI in order of their decreasing dose rates. Figure 4.5 shows dose rate results for dosimeters located on a line parallel to the cobalt source and through the centers of the sample regions in the four test cells of radiation level I. The results imply that dose rate gradients across a given cell are not severe. Figure 4.6 gives the vertical dose profile along the center of an outside cell in level I. These results show that vertical gradients are also small for a 6-in.-long sample. On the other hand, radial gradients (gradients along the direction perpendicular to the plane of the cobalt) are significant for all test cell locations. The largest radial gradients occur in the cells of level I; representative data are shown in Figure 4.7 for a line through the center of the sample aging region of an outside level I cell. The dose rate drops from approximately 120 krad/hr at the wall nearest the cobalt to 75 krad/hr at the farthest wall. If a large component is being aged, periodic rotations of the test cell during aging can be used to achieve approximately homogeneous total radiation doses throughout the component. Rotations may not be necessary for smaller components. (The radiation facility is currently being used to age electric cables and cable materials.) The samples are 5 in. long with relatively small cross sections. Since vertical radiation dose gradients are small, the samples can be aged standing up in sample baskets which are compartmentalized (Figure 4.7C). By noting the particular compartment used for aging a given sample, the experimental dose rate can be estimated to $\pm 5\%$.

In summary, the recently completed radiation aging facility will be useful for carrying out long term aging experiments at controlled temperatures (maximum of $423^{\circ}K$), radiation dose rates (maximum of 100 krad/hr) and atmospheric conditions (dry and humid air, inert gas, etc.). The flexibility of the design will allow larger aging chambers to be easily constructed if necessary. It will also be possible to modify the chambers so that electrical power can be supplied to a component during aging.

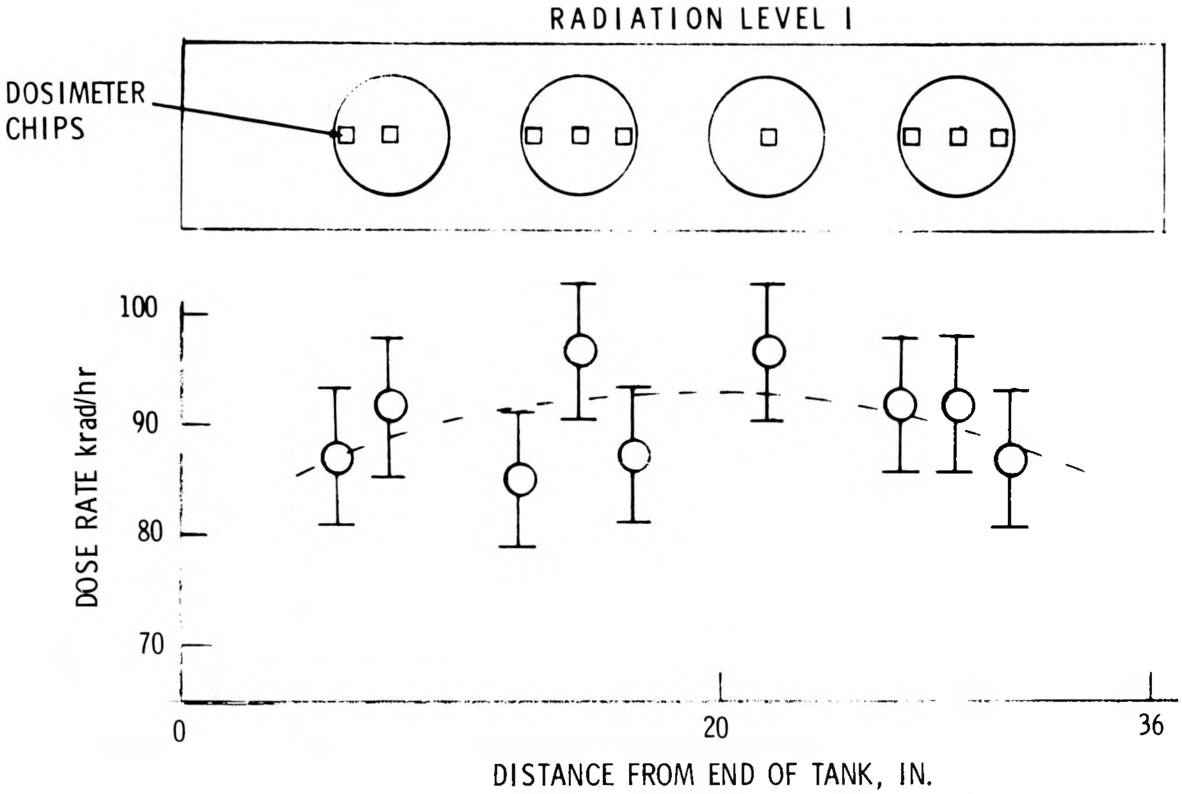

Figure 4.5. Dose rate parallel to linear cobalt array in closest test chambers, level I

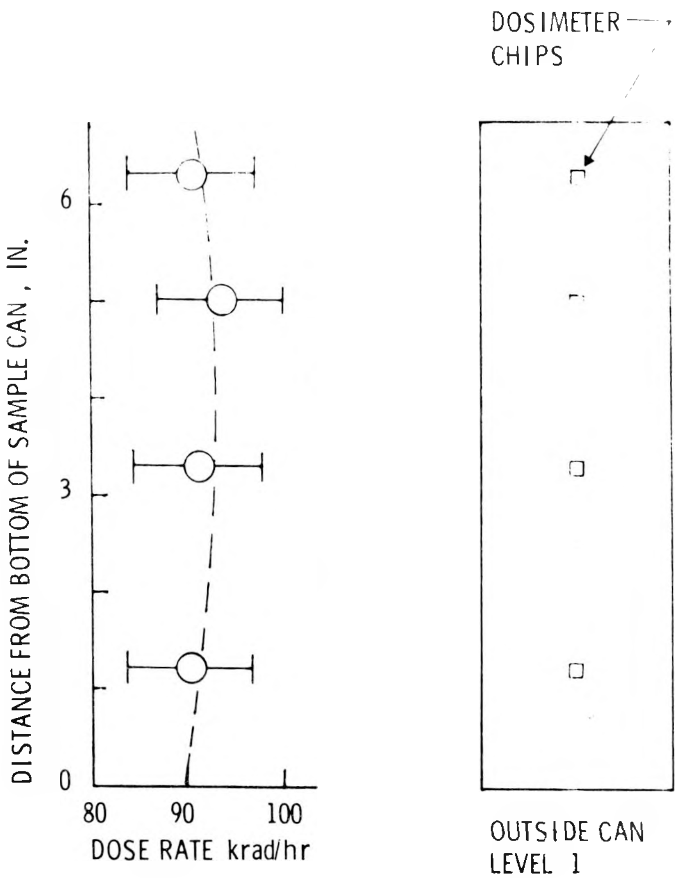

Figure 4.6. Vertical dose rate profile along center of level I (outside) can

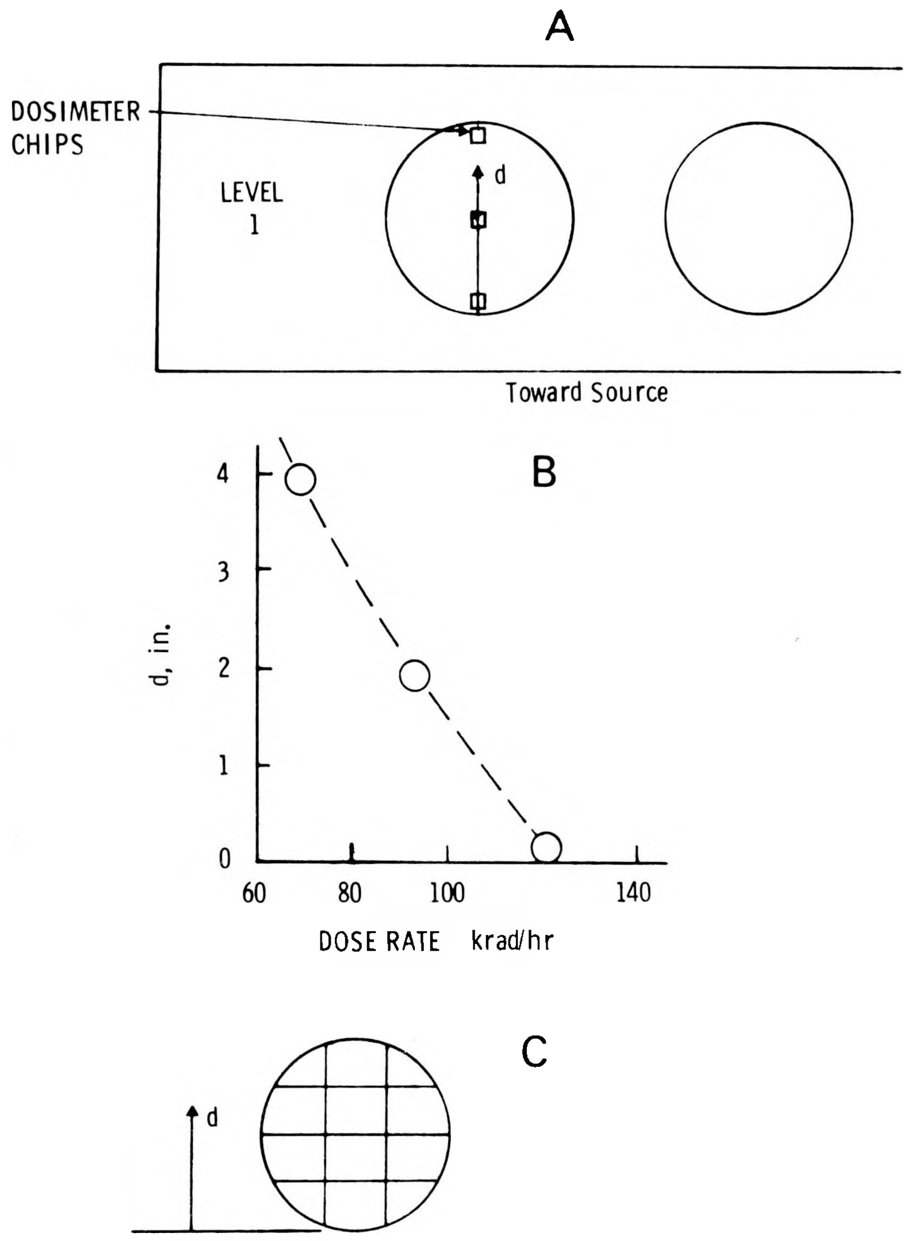

Figure 4.7. A. Dosimeter chip location for radial mapping;
B. Radial dose rate results;
C. Sample basket configuration

4.2.2 Thermal Aging Facility

Thermal aging is normally carried out in air-circulating ovens. There are numerous disadvantages and potential problems associated with this arrangement. Large temperature gradients occur in many ovens, and rapid movement of air past samples will often hasten the removal of volatile components (plasticizers, anti-oxidants), leading to unrealistically accelerated degradations. If two or more materials are aged in the same oven there is the danger that outgassing or volatile degradation products from one material will affect the deterioration of another material. If, for this reason, only one material is aged in a given oven, the number of ovens required for a long-term aging program can be prohibitive.

With the above problems in mind, a unique and versatile thermal aging facility was constructed. The facility uses air-circulating ovens, each modified to accomodate a number of self-contained aging cells. A detailed sketch of one of these aging cells is shown in Figure 4.8. It consists of a bell jar glass chamber which rests on a silicone gasket which in turn rests on an aluminum stand. A gas inlet line enters the sample aging region from the bottom through holes in the aluminum base and the silicone gasket. A small hole at the top of the glass chamber serves as a gas exit and as a means of introducing a permanently positioned thermocouple into the center of the sample region. Lead and aluminum collars help to insure an air-tight seal around the base of the glass chamber. These seals also improve the thermal stability and lessen temperature gradients in the sample aging region. A perforated stainless steel sample basket sits inside of the glass chamber. To gain access to the samples, the glass chamber is lifted and tipped sideways to allow the sample basket to slip free.

The aging cells were constructed in two sizes; smaller cells hold 2 in. ID by 5 in.-long sample baskets, while larger cells accomodate 3 in. ID by 5 in.-long baskets. Three of the smaller aging cells have been incorporated in 1 ft^3 air-circulating oven; as many as six of the large cells have been placed in bigger ovens.

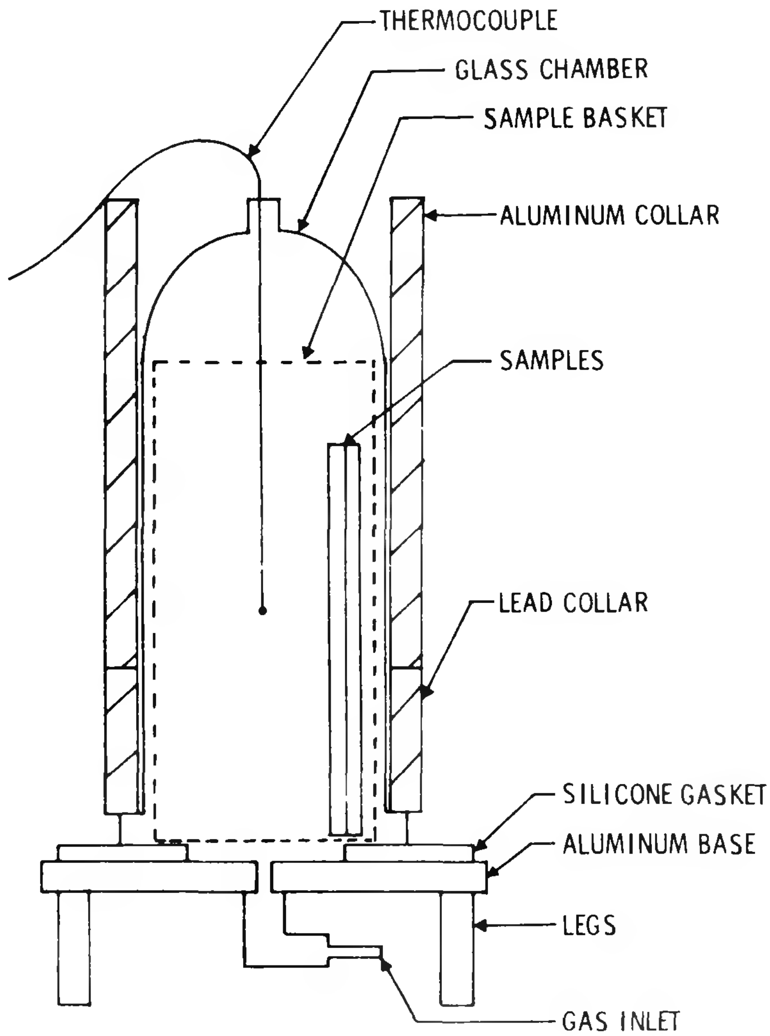

Figure 4.8. Section view of temperature aging chamber

The advantages of this aging facility over a typical air-circulating oven are numerous. One oven equipped with from three to six aging cells can be used to age simultaneously a number of different materials without interactions among materials. In addition, the gas inlet system allows aging to be carried out under various atmospheres (air, inert gas) and controlled gas-flow conditions. Thermocouples permanently positioned in each aging cell offer a convenient means of monitoring the temperature near the center of the samples. The thermal mass of the metal collars offers other advantages; extensive temperature mapping inside the cells indicates that variations in temperature throughout the sample region are less than $1^{o}K$ at $423^{o}K$, and there is much less short-term cyclic temperature fluctuation inside a cell than in the surrounding oven.

This facility is currently being used to carry out thermal oxidative aging on various Class 1E qualified electric cables and cable materials in order to obtain acceleration functions appropriate to single environment thermal aging.

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