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Qualification Testing Evaluation Program Light Water Reactor Safety Research Quarterly Report

October – December 1978

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QUALIFICATION TESTING EVALUATION PROGRAM LIGHT WATER REACTOR SAFETY RESEARCH QUARTERLY REPORT

OCTOBER-DECEMBER 1978

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SUMMARY

The October-December 1978 quarter can be characterized as a period of formal reporting and continuing effort in the Qualification Testing Evaluation (QTE) Program.

The national and international interest in the program remains very high. Numerous requests for information and reports were processed. Teams of French and Japanese nationals visited Sandia Laboratories; in addition, a separate visit was made by the chairman of the Institute of Electrical Engineers of Japan committee devoted to wire and cable environmental qualification.

A major effort of the quarter was an extended visit to English, French, Swedish, and Finnish facilities engaged in various aspects of safety-related equipment qualification in Europe; several experimental facilities used in this qualification effort were toured. These visits were made in conjunction with participation at the International Topical Meeting on Nuclear Power Reactor Safety in Brussels, 16 to 19 October. Eight Sandia-authored or coauthored papers were presented at a session dedicated to "Environmental Equipment Qualification." The complete texts of these papers are included in the appendix of this report.

The balance of the quarterly effort centered on continuation of the ongoing projects. Within the methodologies task, the test facility upgrade and the Commission-requested connector tests received primary emphasis. Within the radiation source task, effort continued on the development of a best-estimate LOCA-radiation signature. Within the accelerated aging task, the primary emphasis was on fire-retardant aging, continued combined environments testing of electrical cable materials, and the evaluation of ambient-aged electrical cable samples.

CONTENTS

TABLES

QUALIFICATION TESTING EVALUATION PROGRAM
LIGHT WATER REACTOR SAFETY RESEARCH
QUARTERLY REPORT

OCTOBER-DECEMBER 1978

1. PROGRAMMATIC OVERVIEW

Programs were initiated in late 1974 to evaluate the significance of synergistic effects in post-loss-of-coolant-accident (LOCA) testing of Class 1E safety-related equipment. As a result of these activities, two 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

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 to 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 signatures, the development of guidelines and rationale for use of radiation simulators in typetesting Class 1E components, and the definition of the LOCA-radiation signature based on "best estimates" of the accident progression and fission product release sequences.

Typetesting requires a component 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 FY79 effort under this task is concentrated in seven broad areas and numerous related subtasks:

1. The High Intensity Adjustable Cobalt Array (HIACA) test facility upgrade is to be completed to (a) accommodate large 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.

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

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

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

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

6. Initial methodology testing will begin based on the test plan(s) development and approval; these tests could include synergistic effects tests on larger/other components (e.g., valve operators, sensors, penetrations, cable assemblies) or investigations into additional qualification concerns (e.g., superheated steam, pressure-rate phenomena, pressure/temperature magnitude effects, dose-rate effects, component configuration influences). Some effort may be devoted to developing

requalification test methods using naturally-aged components; similarly, other specific confirmatory tests may be performed under this subtask. An initial attempt will be made to develop typetesting methodologies and standard laboratory techniques to assure comprehensive and repeatable test sequencing; this could include development of screening procedures (tests) to determine rank-order component vulnerability and appropriate failure criteria for components.

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

1.2 Task 2--Radiation Qualification Source Evaluation

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

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

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

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

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

1.3 Task 3--Accelerated Aging Study

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

1. Single-environment aging tests will be continued 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. Similarly, aging tests in combined radiation and temperature environments will be continued in order to determine the importance of synergisms and to test the method postulated for combined-environment accelerated aging.[1.10]

2. The study of the effects of aging with regard to the retention of flame-retardant additives in cable materials (begun in FY78) will be continued and

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

3. Alternate indicators of damage will continue to be investigated under this task; examples of such indicators are voltage withstand, mandrel bend tests, dissipation factor, or equivalent series/parallel resistance. These tests more closely parallel current industry failure criteria which require "functionability" of electrical cable. The evaluation of alternate real-time damage indicators (to complement elongation) will continue, aimed toward the development of a parameter indicative of age (e.g., dissipation factor) which can be evaluated on-line.

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

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

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

1.4 Quarterly Programmatic and Common-Task Activities

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

Program Reviews with NRC Staff -- A special program review was held in Silver Spring, Maryland, on 1 December 1978 for Operating Reactors, Systems Safety, and Reactor Safety Research Division Directors, and other NRC staff. The primary issue discussed was the Commission-requested connector tests.

A brief program review was held for R. Feit at the National Bureau of Standards on 7 November 1978 to discuss (1) moisture leakage through single-conductor electric cable, (2) several Japanese reports supplied by Dr. Yahagi, and (3) other general programmatic developments.

Program Formalization -- The final Schedule 189 for FY79 was formally submitted to DOE/ALO on 18 October 1978. The DOE/ALO office formally transmitted the proposal to the Office of Nuclear Regulatory Research by letter dated 17 November. Notice of funding for FY79 was received on 12 December through order No. 60-79-012.

A Buff-Book submittal was prepared and submitted on 1 November. This submittal represented routine input, except that several potential programmatic difficulties were identified which could affect the milestone schedules for selected subtasks.

The routine reporting of program activities is done through formal quarterly reports issued 4 to 6 months after the close of the reported quarter. A draft of the third quarterly (April-June 1978)[1.11] was completed on 23 August, and it completed internal Sandia-required review and signoff on 15 November; it is scheduled for distribution in early January 1979. A draft of the fourth quarterly (July-September 1978)[1.12] was completed on 19 December and is undergoing review and signoff.

Meetings and Conferences Participation -- D. Dugan attended a short course on "Qualification of Safety-Related Equipment for Nuclear Power Generating Stations" held in San Francisco on 25 to 27 October. The course, sponsored jointly by Drexel University and the IEEE, was designed to provide a better understanding and clarification of equipment qualification through interpretation of IEEE Std 323-1974.

L. Bonzon, R. Luna, and K. Gillen participated in the International Meeting on Nuclear Power Reactor Safety held 16 to 19 October 1978 in Brussels, Belgium. Eight Sandia-authored or coauthored papers describing various aspects of the QTE Program were presented and/or published in the Proceedings at a session dedicated to "Environmental Equipment Qualification." Attendance, discussion, and requests for program reports indicated the timeliness of, and interest in, the topic, especially within the European nuclear community, where equipment qualification is just emerging as an issue. (The complete texts of these papers are included in the appendix of this report.)

K. Gillen attended the 1978 Conference on Electrical Insulation and Dielectric Phenomena, held at Pocono Manor, Pennsylvania, 29 October to 2 November. He presented a paper describing the combined-environments accelerated aging method.[1.13]

Review of Quality Assurance (QA) in Sandia Research Programs -- A 2-day review headed by G. Bennett was held on 29 and 30 November for NRC, DOE, and ALO staff to discuss quality assurance practices within USNRC-sponsored research programs. L. Bonzon presented the QTE Program QA considerations; various other Sandia staff presented particular aspects of the issue also. Along with the formal presentations, various test facilities were toured by the group.

Equipment Qualification Programs in Europe -- In conjunction with attendance and paper presentations at the European Nuclear Society/American Nuclear Society

Meeting on Reactor Safety, Brussels, 16 to 19 October, L. Bonzon and K. Gillen participated in discussions with several European organizations concerning safety-related Class 1E equipment qualification practices in Europe. These contacts were made with English, French, Swedish, and Finnish counterparts. In England, discussions were held with staff of the Fire Research Station at Borehamwood. In France, discussions were held with Électricité de France, Framatome, and Commissariat a L'Énergie Atomique personnel at four locations: (1) Framatome offices in Paris, (2) Saclay irradiation and test facilities, (3) Cadarache test (autoclave) facilities, and (4) Bugey (pressurized water reactor) nuclear power station. In Sweden, discussions were held with ASEA-ATOM and ASEA-KABEL personnel at Vasteras and at the Forsmark (boiling water reactor) nuclear power station. In Finland, discussions were held with the staff at the Technical Research Centre of Finland.

While all of these centers are doing applicable research and testing, that work is generally reported only in internal memoranda. A major purpose of the visits was to stimulate the flow of information, possibly through initiation of formal exchange agreements.

Sixth WRSR Information Meeting Presentation -- L. Bonzon presented an overview paper[1.14] on the QTE Program at the USNRC Sixth Water Reactor Safety Research Information Meeting at the National Bureau of Standards on 7 November. The paper, "Status of the Qualification Testing Evaluation (QTE) Program," described the past year's achievements in the program and highlighted the planned activities for FY79.

Foreign Interest -- On 20 October, a team of French nationals from CEA and EDF and headed by G. Gaussens (CEA) visited Sandia to discuss LOCA environments, qualification testing, and aging. They were hosted by E. Salazar and F. Thome. Besides general discussion of the QTE Program, a number of reports detailing the program were provided at their request.

A team of Japanese Atomic Industrial Forum members headed by Y. Shinozaki visited Sandia to discuss the QTE Program on 27 October; they were hosted by E. Salazar, L. Klamerus and F. Thome. Specific discussions were held with Dr. Y. Oshima and Dr. S. Machi, both of JAERI, with specific emphasis on aging and qualification techniques. Several reports were provided to them.

As a result of the Brussels Meeting presentations, a number of requests for reports were received along with requests to be included in future reports distributions. In early November, copies of SAND reports 78-0067, 78-0091, 78-0341, and 78-0799 were sent to Westinghouse Europe (Brussels), Traction & Electricite (Brussels), Euratom (Ispra), Rhein-Westf TÜV (Essen), Battelle-Institute V (Frankfurt), and Studsvik Energiteknik AB (Nyköping).

Dr. K. Yahagi (Electrical Engineering Department, Waseda University, Tokyo) visited Sandia Laboratories on 10 November 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. Discussions with L. Bonzon concerned recent developments in the QTE Program, the Brussels meeting papers, European approaches to equipment qualification, specific questions on the Sixth WRSR paper, status and plans for the upgraded test facility, and the Japanese (simultaneous) test facility for wire and cable currently under construction. Several reports were also provided at Dr. Yahagi's request. He also requested and received Draft 9 of IEEE P627 in early December.

Several individual foreign requests for reports were also separately received. SAND78-0067 was sent to Dr. S. Machi, Japan, on 6 November. Copies of SAND78-0091 were sent to Dr. K. Nowicki, Poland, on 6 October and 6 November.

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

TABLE 1.1

Industry Liaison

		Requests		
Date	Company	Prior Reports	Mailing List	Other
27 October	GE-Bridgeport			Submit samples for testing
6 November	SRP-duPont	SAND78-0067		
6 November	ITT	SAND78-0067		
6 November	Square D	SAND78-0067		
6 November	AEP Services	SAND78-0067		
7 November	Westinghouse			Concerning depth-dose calculations and 6th WRSR paper
8 November	Okonite	SAND78-0067		
9 November	Bendix-Sidney			Status of Commission-requested connector tests
10 November	Bechtel-San Francisco	QL Test VI QL Test VII SAND78-0067		
28-29 November	DuPont-Wilmington	SAND76-0715 SAND78-0341 SAND78-0799	X	Discussions on aging methods and results
30 November and 13 December	Nuclear Services Corporation			Unsolicited proposal to examine/evaluate the qualification testing of electrical penetrations
7 December	Duke Power			Availability of ambient aged cable and Tefzel® aging
8 December	Washington Public Power Supply System			Requested supplier of Tefzel® samples and its aging characteristics

TABLE 1.1 (Continued)

Industry Liaison

Date	Company	Requests: Prior Reports	Requests: Mailing List	Requests: Other
13 December	Stone and Webster			Range of work on LWR safety being done at Sandia
13 December	Gulf & Western	SAND78-0067 SAND78-0799		Details on connectors and their suppliers. Experience with connectors in hot-cell environs
21 December	SAI-La Jolla			Information on future work

1.5 Publications/Presentations

The following is an inclusive list of formal publications and presentations for FY79 which detail aspects of the QTE Program. The first eight papers were presented at and/or published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16 to 19 October 1978 and can be found in the appendix.

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

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

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

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

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

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

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

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

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

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

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

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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, and Section 2.1 highlights the various technical activities. To provide a complete background on the program, the eight formal papers presented at the International Meeting on Nuclear Power Reactor Safety, Brussels, 16 to 19 October, are included as the appendix of this quarterly report. Their development and presentation were a significant effort and reporting milestone during the reported quarter.

2.1 Task 1--Technical Activities Summary

Publications and Presentations -- Two of the papers[2.1,2.2] given at the Brussels meeting are programmatic and/or summary in nature and pertain to all three major tasks of the QTE Program. A third[2.3] dealt specifically with Task 1--the results of the nine (sequential or simultaneous) synergistic typetests conducted on a variety of Class 1 equipment, but specifically on electrical cable, cable field-splices, and cable connector assemblies. The complete test program is summarized in SAND78-0067.[2.4]

UEC Subcontract -- United Engineers and Constructors (UEC) has effectively completed a four-concurrent-phase subcontract to assemble comprehensive data packages for all in-containment Class 1E equipment in a contemporary PWR nuclear power plant. The contract remains in force through FY79 to allow the necessary revisions and updates to be made in a timely manner.

During this quarter, no specific input was made to the safety-related equipment list or data packages. UEC staff assisted in the preparation of a paper[2.5] presented at the Brussels meeting and directly participated in that meeting and in a portion of the visits made to European facilities.

Franklin Institute Research Laboratories (FIRL) Subcontract -- This contract continues through FY79 for the FIRL staff to provide general assistance to Sandia in the evaluation of Class 1E equipment data packages, in the development and critique of test plans, and in other general matters pertaining to the QTE Program.

In early October, FIRL staff provided information on their electrical cable LOCA-simulation testing which summarized the testing on same-type cables tested to different levels of gamma radiation. FIRL staff also assisted in the preparation of a paper presented at the Brussels meeting (see Reference 2.2).

Autoclave Seal Systems Tests -- In early July, NRC/RES staff requested that Sandia establish a test capability to evaluate the migration of moisture through single-conductor electric cable. Based on this request, two accompanying informal NRC reports, and discussions with RES staff, Sandia conducted three tests during August to develop an autoclave sealing system to withstand 70 psig differential pressures without excessive pinching of exiting single-conductor electric cables. The three tests evaluated a two-part RTV epoxy (Test I), a non-pinching pseudo-compression fitting using elastomeric rubber stoppers (Test II), and a one-part epoxy (Test III). In an attempt to detect the extent of moisture migration in single-conductor cables, an intense dye was used, and in some cases the cable was sectioned after testing to inspect for dye/moisture migration.

The results of these tests were informally submitted to R. Feit in early October by letter dated 29 September; subsequently the tests and results were formally reported.[2.6] The following paragraph summarizes these results.

In these seal systems tests, the rubber stopper/Swagelok system performed best. It is easily adaptable to any cable size. The test results (amount of moisture leakage) were not significantly different from the results using the two-part epoxy, and post-test examination of the cable indicated no excessive cable pinching. As to the moisture leakage, it was concluded that

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

Facility Upgrade -- The existing test capability at the Sandia Gamma Irradiation Facility (GIF) is being upgraded to accommodate larger Class 1E test items. Along with size, the principle capability improvement is the addition of the high intensity adjustable cobalt array. The HIACA will have the ability to select radiation dose rate while holding the spatial gradient stationary. The upgrade also includes (1) new radiation sources and source positioning apparatus, (2) a new source elevator system for storing HIACA under water, (3) an electro-hydraulic control system to select irradiation rates and to interlock for safety, (4) irradiation cell modifications, (5) test chambers, and (6) diagnostic/test equipment. The major efforts this quarter are detailed below.

The cobalt order from Neutron Products, Inc., was delivered and unloaded into the GIF facility on 25 October 1978. The total source consists of 315,989 contained curies (288,576 effective curies) as of 21 October 1978 and is divided between 32 pencils. The active length of each pencil is 0.61 metre (24 inches), and the specific activity is tailored to provide a nearly flat distribution over an active length of 1.3 metres (50 inches) for stacked pencils.

The HIACA positioning apparatus contract was let with D-Velco Manufacturing Company of Phoenix, Arizona, in November. They will fabricate a preliminary test fixture consisting of four telescoping tube assemblies to test the concept and to assure raising and lowering operations. The test fixture will also be used to adjust clearances to optimize free operation and low water leakage. The HIACA itself will consist of 128 tube assembly positions on four radii that permit manual adjustment of the source radius. Thirty-two tubes/pencils will normally be utilized in eight ganged groups of four to adjust the dose rate. The three stages of telescoping tubes will extend to a height about 3.7 metres (12 feet) and when stored will be nested in the fixture with an overall height of about 1.7 metres (5-1/2 feet). Delivery of the final apparatus is expected sometime in March 1979.

A major elevator redesign was started during this quarter. The elevator is used to raise the HIACA positioning apparatus to its operating position prior to extending the telescoping tubes. The elevator also lowers the apparatus to the bottom of the pool for safe storage. Initially it was determined that only minor changes would be needed to the existing system, but on further analysis it was determined that the rail system could not support the increased load. It would be even less able to support the loads in the normal operating condition in the event of elevator free fall following separation of the lifting cables. Although the probability of cables breaking is extremely small, it is felt that since the damage would be so severe, a dashpot arrangement under the elevator is necessary to cushion a possible fall. The drop time for a free fall would be about 1 second. Along with complete rail replacement, the elevator roller bearing system requires redesign to support the new load. All of these designs were started this quarter, as was initiation of the purchase for a commercially available winch to haul the cables for positioning the elevator.

An electro-hydraulic control system initial design was completed but cannot be finalized until the HIACA test fixture is evaluated. This information is necessary to determine the need for a hydraulic system that will assist in lowering the cobalt tubes. All of this control equipment must interface with the existing elevator and cell door controls on the GIF for safe operation.

Additional cell modifications were completed this quarter. The depleted uranium shield on the wall that separates the two GIF cells was installed, completing the wall shields. Ventilation shaft shielding is near completion. The cell manipulators were removed and returned to the factory for refurbishing and shield additions; this will prevent radiation streaming from the new source, which is much taller than any of those previously used in the cell. Three more holes are needed for penetration into the GIF cell. Two large holes were core drilled the last quarter for steam pipe and cable access. The order was placed for the additional holes to provide an access hole for installation of the new elevator rails. Two smaller holes will be drilled for the larger elevator lifting cables.

The Sandia Non-Reactor Safety Review Committee specifically reviewed and approved the elevator rail design, the elevator winch system, and the test chamber support and positioning apparatus.

The LOCA steam supply system, test chamber, and other support equipment has been received or nears completion. The steam accumulators and steam pressure regulators were received. The environmental chamber was scheduled to be delivered in

December, but the manufacturer experienced some difficulty in getting spun heads of the correct material thickness, thus delaying shipment of that item until some time in January 1979. The stainless-steel, 2.8 m^3 (750-gallon), chemical mixing tank was shipped and received along with the large mixing motor needed to prepare the chemical spray solution. The electric boiler (steam generator) and the condensate return tank arrived in December, thereby completing the major environmental equipment requirements for the system.

In addition to the environment producing equipment, significant work was done relating to other mechanical equipment requirements. These include the handling and supporting equipment for the environmental test chamber and other major modifications of the cobalt source handling items. In this report period, the design was completed and procurement initiated for the support assembly that will position the environmental chamber in the radiation facility. The support assembly also provides a means to raise and lower the test chamber to allow inspection of test items. Two more needed items for which design has not yet begun are a trunnion stand and the test chamber overhead rail and hoist system. The trunnion stand is needed to hold, move, and rotate the test chamber during test preparation, and the rail and hoist system is required to handle and position the test chamber inside the GIF.

<u>Connector Tests</u> -- The effort this quarter directly followed up on the efforts of the previous quarter with specific regard to acquisition of the Browns Ferry Unit 3 (BF3) Bendix connectors and Conax modules.

Following internal NRC concurrence, six Conax modules (7590-100000-18), identical to those used at Arkansas Nuclear One, Unit 2, were ordered on 7 November with delivery expected in early January 1979. While not technically a connector, they are a type of routinely used penetration system.

The balance of the Bendix connector order was received on 1 November; six each of two connector types have now been received:

- Type I 10-214628-51S
  10-214028-51P
- Type II 10-214636-78S
  10-214036-78P

On 7 November, a review meeting was held with W. Rutherford of NRC/IE in Bethesda, Maryland. Discussions included the status of the Bendix order, plans for their assembly at Browns Ferry, comments on the August test plan, plans for the Conax modules, and investigations of other connector assemblies as potential test candidates.

On 27 and 28 November, W. Rutherford and F. Jablonski of NRC/IE visited Sandia to discuss details of the BF3-Bendix tests. Order information and Scotchcast No. 9 product literature were provided and discussed. Plans for the assembly of the connectors by BF3 staff on 11 and 12 December were discussed with emphasis on verification of assemblies adequacy. (The 11 and 12 December visit to BF3 was later indefinitely delayed at the request of BF3 staff.)

2.2 European Meeting Presentations

A significant effort this quarter was the formal documentation of the QTE Program through papers presented at and/or published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16 to 19 October 1978. These complete-text papers are presented in the Appendix to this report.

It should be kept in mind that the papers are presented serially in the appendix as they were at the meeting. The opening paper provides an overview of methodology assessment in general and the QTE Program specifically. This is followed by several specific papers on various aspects of the Program. Finally, the summary paper outlines some other issues in equipment qualification.

2.3 References

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

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

2

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. To provide a complete background on the program, the eight formal papers presented at the International Meeting on Nuclear Power Reactor Safety, Brussels, 16 to 19 October, are included as the appendix of this quarterly report. Their development and presentation were a significant effort and reporting milestone during the reported quarter.

3.1 Task 2--Technical Activities Summary

Publications and Presentations -- Two of the papers[3.1,3.2] given at the Brussels meeting are programmatic and/or summary in nature and pertain to all three major tasks of the QTE Program. Two others[3.3,3.4] deal specifically with the Task 2 effort. The first describes the radiation signature, as derived from the applicable Regulatory Guides, for the LOCA accident. The second describes an approach to derive a "best-estimate" LOCA-radiation signature based on available literature for fission product releases and accident time sequencing and progression.

IRT Subcontract -- The major effort on this contract is to complete the "best-estimate" LOCA-radiation signature study. The Phase 1 report[3.5] was completed in June; further work on the study was continued this quarter.

IRT staff assisted in the preparation of two papers (see References 3.3 and 3.4) for the Brussels meeting and participated in that meeting and in a portion of the visits made to European facilities.

3.2 European Meeting Presentations

A significant effort this quarter was the formal documentation of the QTE Program through papers presented at and/or published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16 to 19 October 1978. These complete-text papers are presented in the appendix to this report.

It should be kept in mind that the papers are presented serially in the appendix as they were at the meeting. The opening paper provides an overview of methodology assessment in general and the QTE Program specifically. This is followed by several specific papers on various aspects of the Program. Finally, the summary paper outlines some other issues in equipment qualification.

3.3 References

3.1 R. E. Luna and L. L. Bonzon, Methodology Assessment: An Overview of the Qualification Testing Evaluation (QTE) Program, SAND78-0342 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16-19 October 1978.

3.2 L. L. Bonzon, R. E. Luna, and S. P. Carfagno, Qualification Issues: The Rest of the Iceberg, SAND78-0350 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16-19 October 1978.

3.3 N. A. Lurie and L. L. Bonzon, The Hypothesized LOCA Radiation Signature and the Problem of Simulator Adequacy, SAND78-0348/IRT 8167-005 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16-19 October 1978.

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3.5 N. A. Lurie, "Best-Estimate LOCA Radiation Signature: Phase 1, Suggested Accident Scenario and Source Definition," IRT 0056-001 (Prepared for Sandia Laboratories, June 1978, and intended for internal Sandia use only; this report will be published in a full Sandia topical report).

3

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. To provide a complete background on the program, the eight formal papers presented at the International Meeting on Nuclear Power Reactor Safety, Brussels, 16 to 19 October, are included as the appendix of this quarterly report. Their development and presentation were a significant effort and reporting milestone during the reported quarter.

4.1 Task 3 - Technical Activities Summary

Publications and Presentations -- Two of the papers[4.1,4.2] given at the Brussels meeting are programmatic and/or summary in nature and pertain to all three major tasks of the QTE Program. Two others[4.3,4.4] dealt specifically with the Task 3 effort. The first summarizes the method for accelerated aging of materials in combined environments and illustrates that method using data for neoprene. The second describes the effects of dose rate, dose, and humidity on room temperature aging of a variety of electrical cable insulation/jacket materials.

K. Gillen presented results[4.5] of the accelerated aging study at the 1978 Conference on Electrical Insulation and Dielectric Phenomena. Specifically, the paper dealt with the combined environments method applied to a neoprene jacketing material.

Similar results and information were presented by K. Gillen[4.6] at the 6th WRSR Information Meeting at the National Bureau of Standards on 7 November.

Consulting and Related Efforts -- During October, R. Clough attended the workshop on Electrical Aging of Cable Insulation Material at Battelle Laboratories, Columbus, Ohio, where he served as a consultant for the aging program being undertaken by Battelle. The objective of the program there, which is sponsored by DOE, is to try to understand the rates and mechanisms of cable insulation breakdown due to factors such as voltage cycling and mechanical stress. Suggestions for the Battelle program were made both at the conference and in a subsequent written evaluation.

K. Gillen presented an invited seminar to staff at Bendix Corporation, Kansas City, on 6 November; the seminar was devoted to "Accelerated Aging--Principles and Techniques."

Embrittled Polyethylene Cable Evaluation -- E. Salazar visited staff of the Savannah River Plant (SRP) in October to review the results of the cable insulation degradation study conducted at Sandia. Discussions with members of the Reactor Technology and Equipment Engineering groups centered on the test methods and results, with particular emphasis on the strong synergistic effects observed in the polyvinyl chloride-jacketed, polyethylene-insulated (old-style) cable. Also discussed were SRP's plans to establish a facility for conducting combined thermal/radiation tests.

Fire-Retardant Aging Program -- The study to investigate the effects of aging on fire retardants in electric cable continued. Two formulations of EPR and Hypalon are being evaluated, one containing the full range of additives and fillers excepting fire retardants, the other with the same formulation but including an SB_2O_3-halocarbon flame retardant. Oven aging of test specimens continued, and the radiation and the combined radiation/temperature aging tests were initiated. Combustion data on unaged samples will be used for comparison with data on samples having varying degrees of accelerated aging. Chemical-analytical data on flame retardant retention will also be obtained on the aged samples as they become available, using techniques which will include mass spectroscopy.

Cable Material Aging Experiments -- The general single- and combined-environments aging experiments, on a variety of modern cable insulation and jacketing materials, continued this quarter to support the accelerated aging method analyses (see References 4.3 to 4.6). These experiments were performed at the Sandia low-intensity cobalt array (LICA) facility and (under subcontract) at the Naval Research Laboratories.

4.2 European Meeting Presentations

A significant effort this quarter was the formal documentation of the QTE Program through papers presented at and/or published in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, 16 to 19 October 1978. These complete-text papers are presented in the appendix to this report.

It should be kept in mind that the papers are presented serially in the appendix as they were at the meeting. The opening paper provides an overview of methodology assessment in general and the QTE Program specifically. This is followed by several specific papers on various aspects of the Program. Finally, the summary paper outlines some other issues in equipment qualification.

4.3 References

4.1 R. E. Luna and L. L. Bonzon, Methodology Assessment: An Overview of the Qualification Testing Evaluation (QTE) Program, SAND78-0342 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, 16-19 October 1978.

4.2 L. L. Bonzon, R. E. Luna, and S. P. Carfagno, Qualification Issues: The Rest of the Iceberg, SAND78-0350 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor safety, Brussels, 16-19 October 1978.

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4.4 K. T. Gillen and E. A. Salazar, Aging of Nuclear Power Plant Safety Cables, SAND78-0344 (Albuquerque: Sandia Laboratories, October 1978). Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, 16-19 October 1978.

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

APPENDIX

SUPPLEMENTARY PAPERS

The Appendix comprises eight papers presented at and/or included in the Proceedings of the International Topical Meeting on Nuclear Power Reactor Safety, 6 to 19 October 1978, Brussels, Belgium.

CONTENTS

SAND78-0342

A1. METHODOLOGY ASSESSMENT: AN OVERVIEW OF THE QUALIFICATION TESTING EVALUATION (QTE) PROGRAM*

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ABSTRACT

Methodology assessment is required to assure that the qualification testing applied to nuclear power plant safety systems is both realistic and conservative. This task is carried on throughout the nuclear industry, but NRC, through contracts with independent laboratories, has the principal role. Evaluating test methodologies poses problems for the laboratory because testing Class 1E equipment in new situations may produce component failures. Such failures have potential economic, regulatory, and political ramifications for the nuclear industry and, therefore, careful consideration of program policy is required.

An example of a methodology assessment program is the Qualification Testing Evaluation (QTE) Program supported by NRC at Sandia Laboratories. Other papers in this session provide detailed results from specific portions of the QTE program, but the complete program's scope encompasses (1) assessment of LOCA testing methodologies including synergistic effects of combined environments and effects of test procedure; (2) determination of the radiation environment from accidents and its impact on the adequacy of radiation simulators; and (3) modeling of natural aging processes and their simulation by accelerated aging methods including synergistic effects of combined environments.

Introduction

The nuclear power industry is required to demonstrate that certain safety-related equipment (i.e., Class 1E equipment) is "qualified" and will function even in the event of a severe reactor accident. In the process of achieving qualification the equipment can be subjected to searching technical

*This report documents part of the Qualification Testing Evaluation (QTE) Program (A1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

analysis, documentation to establish links to equipment of known capability, direct testing methods, or a combination of all three [1]. Of the three, qualification testing is of particular importance, for it is the technique which demonstrates that a component will work in a simulated environment and thus (with high confidence) will work in the real case. Moreover, it is really the basis for the other two techniques since, at some point in analysis or in linking to components with established qualification, those techniques rest on the results of a qualification test. As a result it is judicious for the U. S. Nuclear Regulatory Commission (NRC) to assure that qualification testing reliably demonstrates operability of safety systems in emergency situations. This is the impetus for establishment of a methodology assessment activity.

Methodology assessment, as discussed here, means no more than "testing the test." This activity really consists of two principal parts, one relating to validating the test environments and the second to validating the test procedure. The first is related to how realistically a qualification test environment stresses a component in comparison to that which would be experienced in predicted accident situations for which it is designed to survive. The appropriate severity of a testing environment may be especially difficult to achieve because the predicted environments in which a component must operate are both variable and extreme. Additionally, the optimum environment may be costly relative to reasonable budgets to be allotted to a qualification test program. As a result of these difficulties, NRC has specified separate environments and the standards writing groups, IEEE in particular, have spent considerable effort to produce a set of qualification standards which address the techniques of simulating nuclear reactor accident environments in the most realistic and achievable manner. These test protocols are used by equipment manufacturers and testing laboratories to verify the design and construction features of various items used in nuclear safety systems.

These qualification standards are the second target of the methodology assessment activity. Here the aim is to examine the procedures by which a test is done (given the basic environment specification) to determine if the procedure makes a difference in the result or whether more faithful portrayal of actual environments is required.

But who will undertake the task of methodology assessment? Will it be the nuclear utilities, nuclear component or system suppliers, NRC, testing laboratories, or, perhaps, the independent laboratory? The correct answer to this rhetorical question is seemingly biased if one considers the authors' affiliation, but, in truth, the answer is "all of the above". All participate in methodology assessment by working in concert with the standards writing groups and by specifying and/or performing the testing function.

While there is no conclusive data to suggest that the qualification testing function as practiced in the nuclear industry is not conservative, the U.S. Nuclear Regulatory Commission has sought to assure itself that the testing is conservative. However, the legislation that created NRC from the AEC did not provide NRC with its own laboratories. Thus, an "independent laboratory" enters into the picture to assist the NRC in fulfilling its role relative to the nuclear industry.

An independent laboratory should be one which is neither involved in producing equipment subject to regulation by the NRC nor involved in routine qualification testing of equipment for manufacturers. The laboratory must have the capability to perform such testing and a technical staff knowledgeable in the supporting scientific endeavors. While other organizations exist which also meet these criteria, Sandia Laboratories is such an entity; it is one of eight multi-program laboratories of the Department of Energy and is operated by the Western Electric Company, a component of American Telephone and Telegraph Co., Inc.

This paper describes some aspects of the role of the independent laboratory in conducting methodology assessments and provides an overview of the Qualification Testing Evaluation (QTE) Program as an example of a methodology assessment program. This three-task, multi-faceted, program addresses three aspects of equipment testing methodology: accelerated aging techniques, radiation signature and simulator adequacy, and accident-simulation techniques.

Some General Problems

The easily articulated goal of methodology assessment is somewhat more difficult to translate into practice. There have been both hardware and administrative problems to be overcome in launching this relatively new program. Hardware problems are particularly severe when new test methodologies, for which there is little test experience, are being investigated. Multiple equipment breakdowns in early LOCA tests made one despair of ever getting data which might reflect methodology problems. Because numerous tests have now been completed, these kinds of testing problems are now largely in the past [2].

Administrative problems are also mitigated by experience, but continue to cause difficulty because of the highly charged public debate on the future role of nuclear power and the nuclear industry in the USA. While this condition is one arguement for involvement of an independent laboratory, any results produced by an independent laboratory and appearing to question the safety of nuclear power can be exploited to dispatch nuclear power. The sort of program underway at Sandia Laboratories is particularly vulnerable since methodology assessment must necessarily deal with real components or materials exposed to extreme environments sometimes exceeding the conditions of predicted accidents. Thus, the spectre of a component failure during methodology tests arises and that poses problems for the nuclear industry, NRC, and ultimately for the independent laboratory. Because the component under study may be substantially the same as Class 1E equipment, a failure presents a quandry for the industry because they use or make the component, for the NRC because they licensed its use in a safety system, and for the laboratory because they "tested it" too severely and it failed. Since this component failure scenario has already taken place (see discussion of the QTE program later in this paper), the results chronicled above are not at all speculative. Of course, one can protest that the laboratory did not really test the component, but, rather, used it in a methodology test. However, that distinction is lost in the arena of nuclear debate.

To be able to respond to the issues in a methodology assessment program where actual Class 1E hardware could be involved means:

(1) Evaluating whether a test procedure can use materials, rather than actual components themselves. In most cases the answer is negative.

(2) Evaluating a test procedure, but somehow assuring that any resultant failures have no safety or licensing significance.

(3) Evaluating a test without preferentially qualifying (or failing to qualify) the components of one manufacturer over another.

(4) Assuring that a component selected for use in test evaluation is not subject to a failure-producing design or material fault peripheral to the objective of evaluating the test methodology.

(5) Establishing whether failure of a qualified component when it is subjected to a test profile more stringent than that

used for its prior nuclear qualification (but still within the "envelope" of all nuclear reactor qualification environments) necessarily means that the component is unfit for nuclear use.

Each of these issues is the burden of the independent laboratory in methodology assessment. The questions posed by these issues must be resolved within the laboratory and between it and its sponsor before any testing should go forward or any components are procured for use in the evaluation process. No matter how each issue is satisfied, there will be penalties which stem, not from the research program itself, but rather from the sometimes conflicting interests of regulators, utilities, and intervenors who have, not coincidently, pledged themselves to the same general goal of ensuring nuclear safety. These points of conflict are the "thorny" life of the independent laboratory. Nonetheless, the independent laboratory has to maintain an impartial scientific demeanor throughout and provide relevant results for use by NRC and, in the longer run, by the nuclear industry.

In summarizing the role played by an independent laboratory, as exemplified by Sandia Laboratories, it appears that by scrupulous observation of policies which (1) eschew direct public disclosure of specific component performance data, (2) provide for open discussion and mutual agreement of test plans and test priorities, (3) allow no dealings with utilities and/or suppliers which suggest conflict of interest, and (4) provide for unbiased reporting of results, it is expected that such independence can be preserved while still providing valuable information for establishing data-based qualification test methods.

An Example, the QTE Program

The principal guidance for accomplishing safety-related equipment "qualification" is contained in an industry standard, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations" [1] and its associated daughter standards, and in Regulatory Guide 1.89, "Qualification of Safety-Related Electric Equipment for Nuclear Power Plants" [3]. While these provide general guidance, it is recognized [4] that they lack the specific procedures necessary to develop a consistent and defensible qualification program.

A program was initiated by the U. S. Nuclear Regulatory Commission at Sandia Laboratories in late 1974 to evaluate the significance of synergistic effects in loss-of-coolant accident (LOCA) testing of Class 1E equipment. As a result of these activities, two complementary tasks were identified and initiated (for further definition) 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 [3]. In late 1976, these two (and the original) tasks were included in a broader program, Qualification Testing Evaluation (QTE), which had as its goals the evaluation of overall adequacy of the qualification testing of safety-related equipment and to resolve specific anomalies and uncertainties associated with qualification testing as broadly outlined in IEEE-323-1974 [1].

The objectives of the QTE Program, now under the sponsorship of the Office of Water Reactor Safety Research, USNRC, 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. The specific major tasks of the research are:

1. Provide <u>assessments of LOCA testing methodologies</u> including a qualitative assessment of the synergistic effects resulting

from the combined environments testing of representative Class 1E equipment,

2. Determine the nuclear source term signature for the design-basis LOCA and evaluate the adequacy of radiation simulators, and

3. Provide a model that can be used to simulate the natural aging process of representative Class 1E materials by accelerated aging methods.

Each of these tasks has the potential for significant impact on the nuclear power industry.

The QTE Program

The overall program has long term objectives and, in many cases, final results are not yet available. The several companion papers at this meeting describe specific aspects of the program; a purpose of this paper is to provide a continuity and overview to serve as an aid to understanding the complex and interactive program dimension. Tables 1-3 summarize the three major tasks and subtasks; both the principal activities and the near-term objectives are included.

Under Task 1 (Table 1), the recent major activities are: (1) the completion of the current LOCA-synergistic test series, (2) the test facility designs, approvals, and acquisitions, (3) the completion of the Class 1E equipment lists and data packages, and (4) the initiation of the Commission-requested connector assembly tests. The LOCA-synergistic test series and the Class 1E equipment survey are reported at this meeting. Several near-term objectives under Task 1 (Table 1) require amplification:

In the course of the hearings conducted on the Union of Concerned Scientists' Petition dated November 4, 1977, (prompted by failure of connectors in LOCA tests [2] conducted by Sandia Laboratories as part of the QTE Program) the Commission expressed a desire to conduct LOCA tests on connectors known to be qualified to IEEE-323 standards. Specifically, the memorandum 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." As currently proposed, connector assemblies from several manufacturers will be tested. The connector assembly types will be identical to qualified models that are currently installed in nuclear power plants that are either operating or planned for operation in the near future. The connector assembly will include the cable used in the plant so as to duplicate the connector-to-cable interface.

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

Some initial effort will be directed toward the problem of "statistical" qualification. The concept of typetesting, in fact, does not include consideration of statistical confidence; qualification of a single specimen is accepted for the qualification of an equipment type, within the bounds of quality assurance and quality control. Margin (the amount that the test environment exceeds the actual environment) is the only tool used ". . . to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance" [2]. In most cases, testing the number of specimens necessary to achieve a high level of confidence involves prohibitive costs. But perhaps alternate concepts can be discovered; for example, it may be possible to mathematically relate overtest results to numerical statistics at the (lower) desired test level.

Under Task 2 (Table 2), the recent major activities are: (1) the hypothesized LOCA-radiation signature definition, (2) the preliminary evaluation of simulator adequacy for electric cable, and (3) initial definition of the "best-estimate" LOCA-radiation signature. All of these activities are reported (in part) at this meeting. Of particular interest is the subtask to define a "best-estimate" LOCA-radiation signature based on realistic accident-time-release sequencing [5]. The resultant signature should be the most appropriate for the qualification of Class 1E equipment; its adoption by industry and regulatory agencies will move radiation qualification toward a logical and consistent basis.

The recent major activities under Task 3 include: (1) electric cable aging experiments and combined-environments aging model verification, (2) the aging and evaluation of PE/PVC cable, (3) the initiation of fire-retardant aging tests, and (4) the development of a polymer computer model. Several near-term objectives under Task 3 (Table 3) require amplification:

- The effect of <u>aging</u> on the retention <u>of fire-retardant additives</u> in electrical cable materials will be investigated. Test specimens will be made of the common polymer materials with known fire-retardant additives; these will be subjected to accelerated aging and undergo quantitative testing to determine change in flammability with age. Preliminary testing has begun and first results will be reported shortly; extension of these methods to fire-retardant coatings is possible.

5

- As an alternate to an accelerated aging model, 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. Development of the appropriate tests and test schedules is the main preliminary objective.

- A computer program has been written which replicates the wide range of structural changes that occur during irradiation of polyethylene. The program solves the simultaneous kinetic equations for the network of chemical reaction pathways involved as the basic structure of the polymer continuously alters. Output data include numerous variables such as net scission of polymer chains, net crosslinking between adjacent chains, and net evolution of volatile gas. The input for the reaction conditions can be varied in terms of temperature, irradiation rate, and details of the initial composition of the polyethylene. Computer results are being compared with experimental data, including spectroscopic and mechanical measurements, and satisfactory correlations are being obtained. The computer approach will be applied to questions fundamental to the aging program, including dose rate effects, annealing effects, and combined environment synergistic effects.

Summary

In summary, adequate Class 1E equipment qualification is a concern of industry and the USNRC; the QTE program is aimed at resolving specific issues in qualification typetesting. While final results are not yet available in all aspects of the program, the preliminary findings are significant and the extent of the program activities, as presented in this paper, is obviously far-reaching. The various papers in this session will elaborate on specific results of the QTE program.

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

4. G. T. Dowd, D. A. Hansen, and S. G. Kasturi, "Qualification of Class 1E Equipment--An Approach," IEEE 1976 Nuclear Power Systems Symposium, New Orleans, October 21-23, 1976.

5. N. A. Lurie, "Best-Estimate LOCA Radiation Signature, Phase 1, Suggested Accident Scenario and Source Definition," IRT 0056-001, prepared for Sandia Laboratories, June 1978.

Table 1

QTE Program
Qualification Methodologies Assessments, Task 1

FY78 Activities

- Completed/documented current LOCA synergistic test series.
- Test facility upgrade; designs, approvals, acquisition.
- Completed comprehensive Class 1E equipment lists and data packages (by subcontract).
- Initiated Commission-requested connector assembly tests.

FY79 Near-Term Objectives

- Complete test facility upgrade, make operational.
- Complete/document Commission-requested connector tests.
- Provide specific NRC-requested confirmatory testing.
- Complete short- and long-range test plans, project descriptions.
- Conduct initial methodology tests.
- Perform Class 1E equipment "vulnerability" evaluation.
- Acquire off-site data; conduct off-site testing.
- Develop requalification tests for naturally-aged equipment.
- Evaluate "statistical" qualification methods.

Table 2

QTE Program
Radiation Qualification Source Evaluation, Task 2

FY78 Activities

- Completed Regulatory Guide 1.89 source signature definition.
- Completed preliminary evaluation of simulator "adequacy".
- Initiated "best-estimate" LOCA-radiation signature definition.

FY79 Near-Term Objectives

- Extend simulator "adequacy" evaluation to other Class lE equipment.
- Complete/continue "best-estimate" signature definition and Class lE equipment response calculations.
- Initiate simulator "tailoring" studies; devise benchmark calculations.
- Develop dose and dose-rate estimates for a generic containment structure (separate funding).

Table 3

QTE Program
Accelerated Aging Study, Task 3

FY78 Activities

- Completed low-level radiation, aging, facilities.
- Continued aging experiments, model verification, for typical electric cable insulations/jackets.
- Performed extensive evaluation of old-style polyethylene/polyvinylcholoride cable.
- Initiated fire-retardant aging studies.
- Examined alternate damage indicators for electric cable.
- Developed computer polymer model.

FY79 Near-Term Objectives

- Continue/complete aging experiments and model verification on electric cable.
- Complete fire-retardant aging, initiate coatings aging evaluation.
- Continue alternate damage indicators evaluations.
- Expand computer polymer model and application.
- Extend methodology to other equipment, other environments.
- Evaluate alternate methods of aging.
- Evaluate naturally-aged cable samples when available.

A2. A METHOD FOR COMBINED ENVIRONMENT ACCELERATED AGING*

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ABSTRACT

An accelerated aging method which can be used to simulate aging in combined stress environment situations is described. It is shown how the assumptions of the method can be tested experimentally. Aging data for a chloroprene cable jacketing material in single and combined radiation and temperature environments are analyzed and it is shown that these data, as well as literature data on a biological system, offer evidence for the validity of the method.

Introduction

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

The present paper describes some results of a continuing program at Sandia Laboratories in which the main goal is to develop more reliable accelerated aging techniques.[1] One result of these studies was the development of a method potentially applicable to combined environment aging when synergism is

*This report documents part of the Qualification Testing Evaluation (QTE) Program (A1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

important.[1,2] The purpose of this paper is to describe this method, indicate how it can be used to carry out accelerated aging in combined environment situations, and, using literature data and data generated in the program, show how the assumptions of the method can be tested.

Single Environment Accelerated Aging

A brief review of some of the principles of single environment accelerated aging is necessary before describing the combined environment method. For both single and combined environments, we restrict ourselves to the constant overstress technique of accelerated aging. In this technique the environmental variable or variables of interest are raised above their ambient values to a constant level and the aging is followed with time. Figure 1 summarizes the method by which single environment accelerated aging is normally carried out. At each of the constant overstress states, S_i, S_j, S_k, ... , the normalized degradation of a material or component, D_S, is followed versus time. D_S could represent, for example, the fraction of good components remaining or the fraction of some material variable remaining. In many instances, constant acceleration factors will relate the decay in one overstress state to the decay in a second. When this occurs, the decays at the various overstress conditions will be superimposable by horizontal shifts on the log time axis (see Fig. 1). Where S is a thermal stress, this corresponds to so-called time-temperature superposition used in numerous polymeric systems. The functional relationship between time and stress will define the accelerating function, $A_S(S)$. If the functional form of $A_S(S)$ can be extrapolated to the use condition, S_u, degradation predictions can be made at the use environment, as shown in Figure 1.

Combined Environment Accelerated Aging Method

Suppose one would like to simulate the ambient aging of a component in a combined stress environment. For a nuclear power plant component, the ambient (i.e., use) environment might include, for example, a temperature, T_u, and a radiation dose rate, R_u. To accelerate the combined environment degradation by a factor X, the proposed method suggests the use of a temperature, T_X, which accelerates the thermal degradation by the factor X, simultaneously with a radiation dose rate, R_X, which accelerates the radiation degradation by the same factor X. The appropriate values of T_X and R_X are obtained from knowledge of the single environment acceleration functions, $A_T(T)$ and $A_R(R)$, which are obtained from single environment aging studies. This relatively simple concept is similar to the idea used by Paloniemi[3] for accelerating all thermal reactions equally through control of the gaseous environment. It is next assumed that any synergistic reactions are accelerated or scaled by the same factor X; this is the key assumption underlying the approach. The set (T_X, R_X) is called a matched set to the use conditions (T_u, R_u). By carrying out combined environment aging under a number of matched set conditions, the validity of the scaling assumption can be ascertained. For example, Figure 2 shows the results of two hypothetical matched set experiments carried out using acceleration factors of X = 20 and X = 80. The combined environment degradation, D_c, is plotted versus log time with the solid curves representing the accelerated degradations. By multiplying the times in the X = 20 and X = 80 accelerated degradations by factors of 20 and 80, respectively, predictions under the use conditions will result. If these and similar extrapolated predictions from other matched set experiments agree, good evidence exists for the validity of the scaling assumption. In other words, the method predicts consistency of matched set conditions. This implies that when the method is appropriate, the time dependent degradation of a material carried out under a given set of combined stress variables can be used to predict the time dependent degradation of that material under any set of stress conditions which is a matched set to the experimental conditions. Some experimental data will be used in the next section to show how the consistency

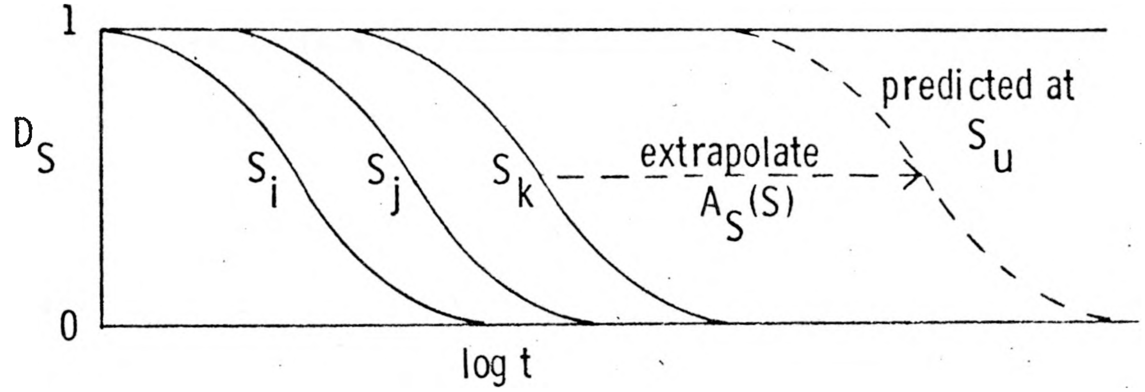

Figure 1. Conventional accelerated aging; plot of normalized degradation D_S versus log time at three constant stress levels and extrapolation to use conditions.

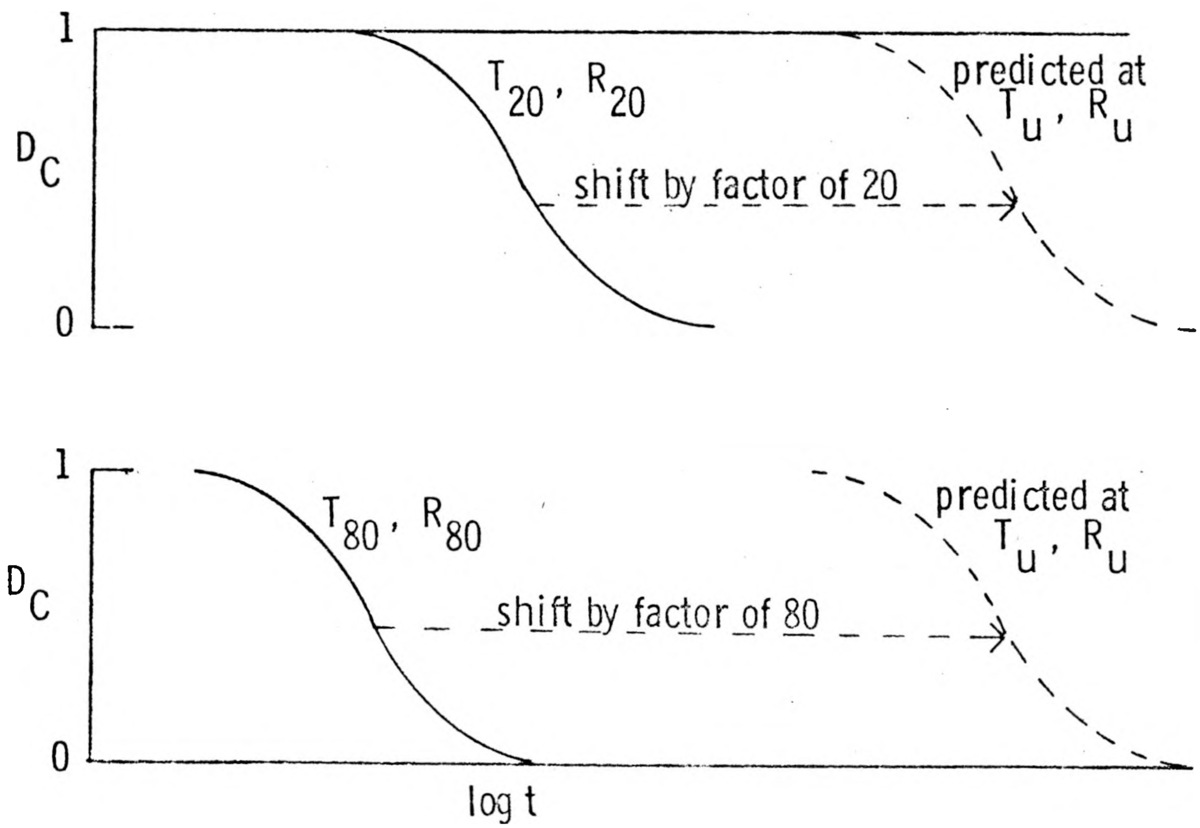

Figure 2. Combined environment accelerated aging approach; normalized degradation D_C versus log time using 20 times (above) and 80 times (below) matched set conditions to use conditions.

6

of matched set conditions can be conveniently checked in order to verify the scaling assumption.

Analysis of Cable Aging Data

An extensive experimental aging program is currently being run on cable jacketing and insulation materials stripped from low voltage electrical cables which are used for safety applications in nuclear power plants.[1,4] Aging is being carried out in single and combined radiation and thermal environments with the degradation being monitored by following the ultimate tensile elongation versus time. Results for thermal aging of a chloroprene jacketing material at temperatures ranging from 363°K to 413°K indicate that constant acceleration factors relate the decays under various constant temperature conditions and that the thermal acceleration function, $A_T(T)$, has an Arrhenius form with an activation energy of 21 kcal/mole. As shown in a companion paper,[4] room temperature radiation aging results for the chloroprene material indicate that the degradation depends only on the integrated radiation dose. The radiation acceleration function, $A_R(R)$, is, therefore, proportional to the radiation dose rate. Combined environment (temperature simultaneous with radiation) experiments are being carried out in Sandia's radiation facility,[5] as well as at the Naval Research Laboratories' facility.[6] Some typical results for the chloroprene material in a combined environment comprising 361°K and 95 krad/hour are shown in Figure 3. The filled circles represent the experimental fractions of ultimate tensile elongation remaining versus time for the combined 361°K plus 95 krad/hour environment. The solid curves, marked D_T and D_R, represent the fractions of elongation remaining versus time for the single thermal environment of 361°K and the "single" radiation environment (low temperature) of 95 krad/hour, respectively. The dashed line approximates the expected decay in the combined environment in the absence of synergistic effects. Comparing this result with the experimental combined environmental data indicates that synergistic effects are found for the chloroprene material.

Using the chloroprene data, Figure 4 shows how the single and combined environment studies can be analyzed according to the formalism of the proposed method. Contours of the time required (0.1 year and 1 year) for the elongation to decrease an arbitrary amount (to 50 percent of initial for this figure) are plotted versus the log of the radiation dose rate on one axis and inverse temperature on the other axis. The horizontal portions of the contours are in regions (low temperature) where the radiation environment dominates the degradation. From the room temperature radiation results,[4] it takes 0.1 year for the elongation to decrease to 50 percent of initial at 55 krad/hour. This determines the horizontal portion of the 0.1 year curve. Since $A_R(R)$ is proportional to the radiation dose rate, R, the horizontal portion of the 1 year curve will occur at 5.5 krad/hour. At low radiation dose rates and sufficiently high temperatures, the thermal environments dominate the degradation; thus, the single environment thermal results lead to the vertical portions of the contours. For example, it takes 0.1 year at 363°K for the elongation to decrease to 50 percent of initial. From the result that $A_T(T)$ is Arrhenius with an activation energy of 21 kcal/mole, one can locate the vertical portion of the 1 year curve at 336°K. To connect the two single-environment-dominated extremes, experiments must be carried out in the region where both radiation and temperature are important. From Figure 3 approximately 194 hours are required in the combined environment of 95 krad/hour and 361°K for the elongation to decay to 50 percent of original. This point is marked on Figure 4 with a cross and the line through it represents the locus of all matched sets (T, R) to the experimental condition of 95 krad/hour and 361°K. It should be noted that in this instance the locus of matched sets is a straight line because $A_T(T)$ is Arrhenius (linear on abscissa scale) and $A_R(R)$ is proportional to dose rate (linear on ordinate

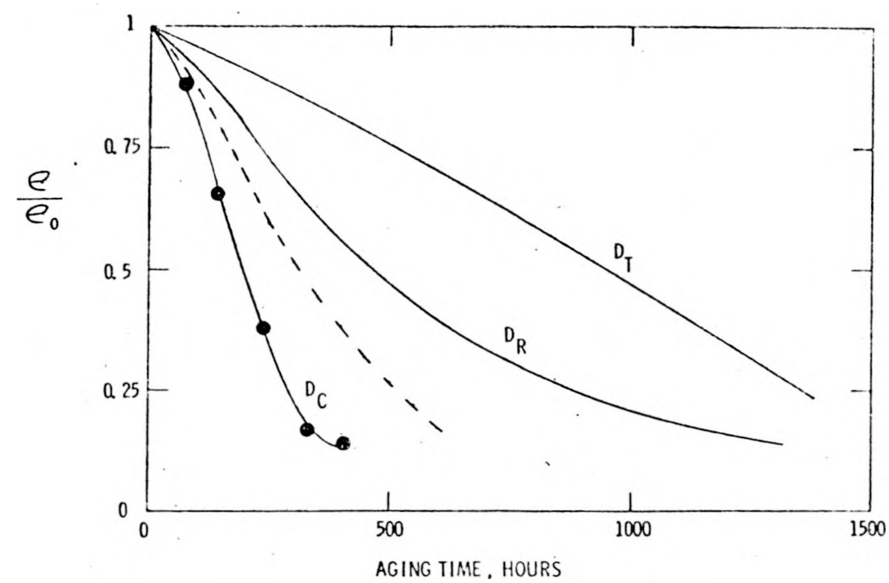

Figure 3. Aging of chloroprene: D_T, 361°K; D_R, 95 krad/hour; D_C, 361°K combined with 95 krad/hour; dashed curve, prediction without synergism.

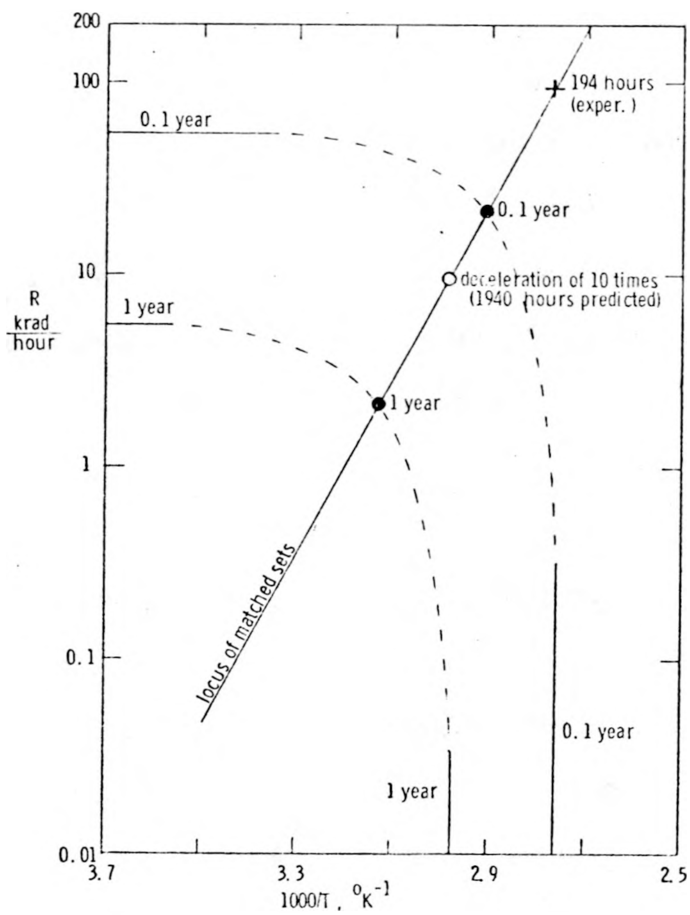

Figure 4. Analysis of chloroprene aging data using proposed model; contours of time required for $e/e_0 = 0.5$.

scale); in the general case where dose rate effects and/or non-Arrhenius behavior exists, the locus of matched sets will not be a straight line. For the chloroprene material the matched set with a deceleration factor of 10 with respect to the experimental conditions (361°K, 95 krad/hour) is given by (334.7°K, 9.5 krad/hour); this result is obtained from $A_T(T)$ and $A_R(R)$ and its location is denoted by the open circle in Figure 4. The proposed method predicts that the time corresponding to this point would be 10 times 194 hours, or 1940 hours. In the same way, predictions (solid circles) corresponding to 0.1 year, 1 year, or any other specified time, can be determined. By carrying out other combined environment experiments under conditions in which both environments are important, the remainder of the time contours (dashed curves) can be generated and the method assumptions checked. For instance, Figure 5 summarizes the results of 11 different combined environment experiments run on the chloroprene material, where again the time contours for decay to 50 percent of original elongation are plotted. The locations of the crosses and the numbers near them denote the combined environment conditions and the corresponding experimental times required for the elongation to decrease to 50 percent of original. The circles are the 0.1 year points predicted using the approach outlined in Figure 4. The large number of experiments scattered throughout the region where both environments are important allow the complete contour to be constructed. The temperature and radiation conditions for 9 of these experiments were chosen such that 3 groups of approximately matched set experiments were carried out, each group comprises 3 matched set conditions which are connected by dashed lines in Figure 5. The superposition within experimental uncertainty ($\sim \pm$ 10%) of the 0.1 year predictions (filled circles) derived from the matched set conditions verifies the scaling assumption underlying the proposed method. As pointed out earlier, this verification implies that experimental results under a set of specified radiation and temperature conditions can be successfully used to predict results under conditions which are a matched set to the original conditions. It should be emphasized, however, that in general, it is not necessary or practical to carry out exact matched set experiments. One need only carry out a number of combined environment experiments at enough properly chosen conditions to generate the contours and at enough different accelerations to verify that the predicted contours remain unchanged as the aging conditions become less severe. It should also be noted that Figure 5 actually represents a slice through a three-dimensional plot at a constant value of e/e_o (0.5). Similar figures can be constructed for other values of e/e_o; in the case of the chloroprene material, analysis of data at other values of e/e_o also confirms the scaling assumption. Once contours similar to those in Figure 4 and 5 have been constructed and confirmed for a given material, they can be used to predict damage to the material under various radiation and temperature conditions. In addition, the matched set approach can be used to accelerate by a chosen factor any ambient aging conditions.

A second test of the proposed method utilizes literature data[7] on the thermoradiation sterilization of a biological material. For this system, decay in the various environments was always first order so a contour of the predicted first order rate constant ($k = 0.1$ hour^{-1}) is plotted versus radiation and temperature in Figure 6. The experimental combined environment points are denoted by crosses and the points predicted for $k = 0.1$ hour^{-1}, again using the approach outlined in Figure 4, are denoted by solid circles. Synergism of radiation and temperature was important for this biological system; the self-consistency of the predicted results offers convincing evidence for the validity of the proposed method. As pointed out earlier, exact matched set experiments are neither practical nor necessary to confirming the scaling assumptions when the data is analyzed in the above manner.

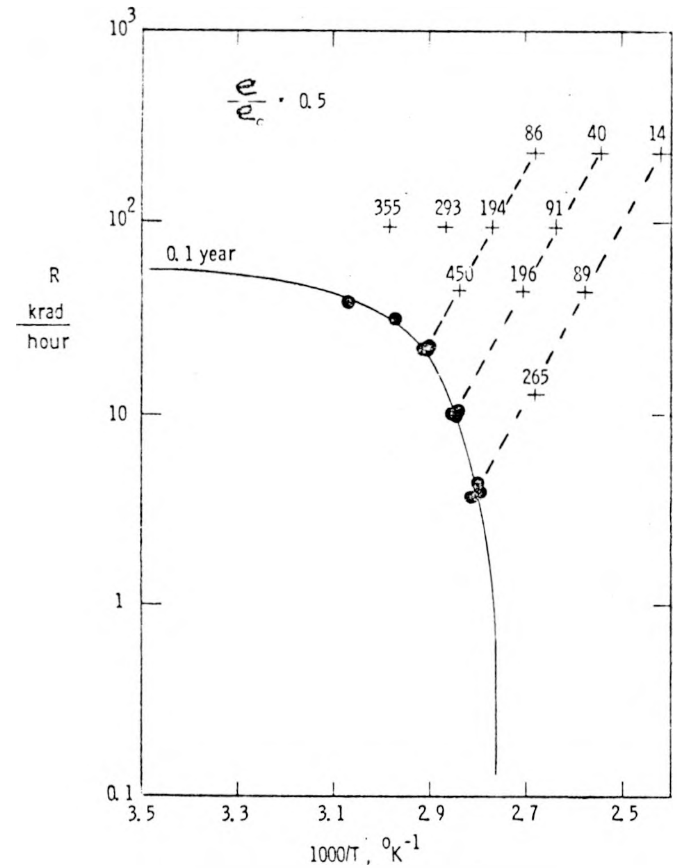

Figure 5. Analysis of chloroprene data using proposed method. The crosses represent the experimental radiation and temperature coordinates. The numbers above them represent the time (hours) required for e/e_o to reach 0.5. The filled circles are the predictions for the 0.1 year contour using the approach outlined in Figure 4. The crosses connected by dashed lines are approximately matched sets to each other.

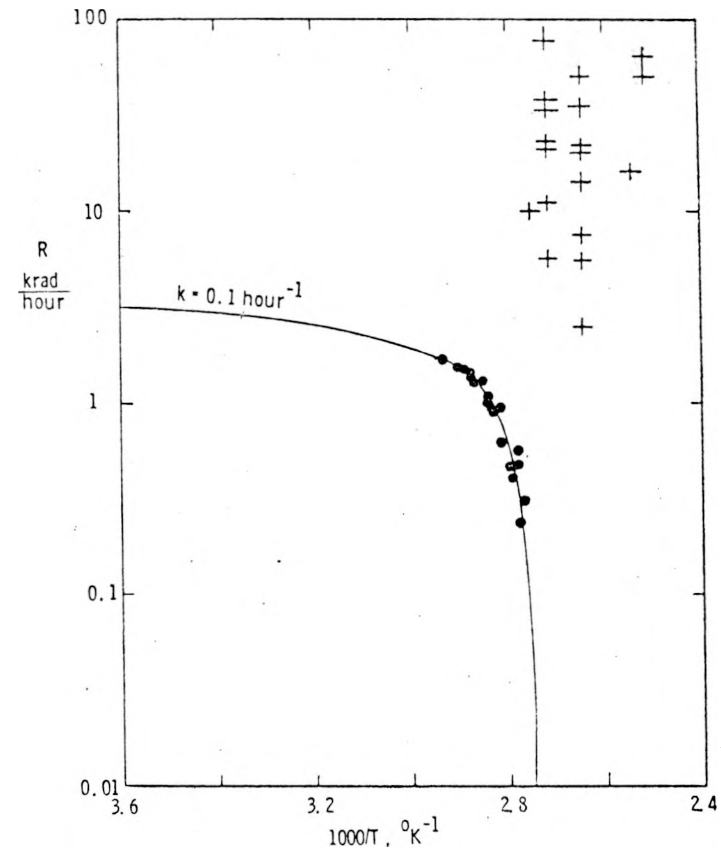

Figure 6. Analysis of literature data using the proposed combined environment accelerated aging method. The crosses represent the experimental radiation and temperature coordinates. The filled circles are the predictions for $k = 0.1\ \text{hour}^{-1}$ using the approach outlined in Figure 4.

Summary and Conclusions

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

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SAND78-0344

A3. AGING OF NUCLEAR POWER PLANT SAFETY CABLES*

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ABSTRACT

Results from an extensive aging program on polymeric materials stripped from unused nuclear reactor safety cables are described. Mechanical damage was monitored after room temperature aging in a Co-60 gamma radiation source at various humidities and radiation dose rates ranging from 1.2 Mrad/hr to 2 Krad/hr. For chloroprene, chlorosulfonated polyethylene, and silicone materials, the mechanical degradation was found to depend only on the total integrated radiation dose, implying that radiation dose rate effects are small. On the other hand, strong evidence for radiation dose rate effects were found for an ethylene propylene rubber material and a cross-linked polyolefin material. Humidity effects were determined to be insignificant for all the materials studied.

Introduction

One aspect of the qualification of safety-related components for the nuclear power industry is that consideration be given to placing the component in an "aged" condition prior to subjecting it to a simulated test of a design basis event, such as a loss of coolant accident. Since aging to the equivalent of 40 years at ambient conditions may be desired, time constraints necessitate the use of accelerated aging techniques. For safety-related components the worst aging environments normally occur within the reactor containment. In this region a number of environmental stresses, including temperature, radiation, and humidity, can contribute to component deterioration. Accelerated aging in thermal environments has been

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†A U.S. DOE facility.

extensively studied and modeled; analysis is usually carried out using the so-called Arrhenius approach. Much less work has been done in radiation environments, where one of the primary concerns is the effect of radiation dose rate on component deterioration. In addition, much work remains to be done on the effect of humidity on aging. Finally, in combined environments the potential importance of synergistic effects on material degradation needs to be determined and methods of accelerated aging for situations where synergism is important must be developed.

This paper describes some results from a continuing program at Sandia Laboratories whose main goal is to develop more reliable accelerated aging techniques.[1,2] One aspect of these studies is to develop methods appropriate to combined environment aging when synergism is important. A description of a proposed method and data on two systems which offer evidence for the validity of this method are described in a companion paper.[3] The present paper describes some experimental aging results on typical modern electrical cable jacketing and insulation materials stripped before aging from commercial low-voltage safety cables. Electrical cables were chosen as the first subject of experimental study because they are an important and relatively simple safety component. Most low-voltage safety cables have low electrical duty cycles during aging so that the most important potential environmental stresses during aging are temperature, radiation, and humidity. Aging is, therefore, being carried out in various combinations of these environments. To minimize the potential pitfalls associated with large data extrapolations, relatively long term accelerated aging experiments are being run (e.g., from approximately one week to several years). Thus, many of the aging experiments, especially the long term ones, have not been completed. Sufficient data have been generated, however, to allow some interesting preliminary observations and conclusions to be made. This paper will describe some of the results obtained to date for room temperature radiation aging environments.

Experimental

Samples of modern Class 1E (qualified for nuclear power plant safety applications) low-voltage electrical cables were obtained from a number of manufacturers. To facilitate testing of the insulation and jacketing materials after aging, the various materials were carefully stripped from the cables prior to aging. The materials, with their nominal wall thickness in parenthesis, included cross-linked polyolefin (0.8 mm), ethylene propylene rubber (1.0 mm), Tefzel® (0.4 mm), and chlorosulfonated polyethylene (1.4 mm) insulations, a chloroprene jacket (1.5 mm), and two chlorosulfonated polyethylene jackets (0.4 mm and 1.2 mm) from two different manufacturers. In addition, a silicone insulation material (1.2 mm) was obtained from one manufacturer as an extruded tube, thus eliminating the necessity of stripping this material. All the insulation materials were aged as tubes; the thin chlorosulfonated polyethylene jacket was aged as a slit tube. Rectangular samples approximately 5.5 mm wide by 150 mm long were cut from the other jacket materials prior to aging.

The radiation exposures were carried out using the Naval Research Laboratories' ^{60}Co gamma source.[4] The facility was modified such that continuous air flow could be supplied to the aging chambers. This modification allowed one to observe the effects of different gaseous atmospheres and relative humidities on radiation aging. The various cable materials were aged at as many as 5 dose rates ranging from 2 krad/hr to 1.2 Mrad/hr. Two aging chambers were used at each of the 4 lower dose rates; the aging conditions in the two chambers were identical except that dry air (0% relative humidity)

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was circulated through one and humid air (approximately 70% relative humidity) through the other. At 1.2 Mrad/hr, aging was conducted only under humid conditions. For both humid and dry air conditions, air flows of approximately 30 cc/min were used. Since the volume of the aging chambers is approximately 1 liter, two "complete" air changes occur every hour. Aging experiments at the two lowest dose rates (2 and 10 krad/hr) are still continuing. Radiation dose rates were obtained from the extensive mapping data generated by Naval Research Laboratory personnel. Because gradients in radiation dose rates were present in the aging chambers, the chambers were periodically rotated to assure that all samples received the same average dose rate. One to four samples of each material were extracted periodically from a given aging chamber to test for deterioration.

For low-voltage electrical cables extensive evidence indicates that mechanical degradation (embrittlement) of the insulation and jacketing materials may eventually lead to cracking of these materials and subsequently to electrical failure during safety equipment operation.[1] Since decreases in the ultimate tensile elongation of a material are intimately related to material embrittlement, the ultimate tensile elongation (e) was used to follow the aging of the cable materials. Tensile testing was accomplished with an Instron Testing Machine Model 1020. Samples were gripped using pneumatic jaws with an air pressure of approximately 3×10^5 Pa. This gripping method seemed to minimize both slippage of samples in the grips and problems associated with tensile failure occurring at the jaws. Initial jaw separation was 50 mm and samples were strained at 125 mm/min. An Instron electrical tape extensometer clamped to the sample was used to follow the strain. Complete stress/strain curves were recorded, so that other tensile parameters, besides the ultimate tensile elongation, were available. Most of these other parameters, such as the ultimate tensile strength, are not nearly as sensitive to the degradation as the ultimate tensile elongation. In addition, the large variations in cross section that often occur for materials stripped from cables lead to enhanced scatter in such parameters. Therefore, only elongation data will be discussed.

Results and Discussion

Figures 1 through 7 summarize the results to date of the extensive room temperature radiation aging experiments on the materials described earlier. For each material, the ultimate tensile elongation, e, measured after aging in the various environments, is plotted against the total integrated radiation dose. Each point in a figure typically represents the average of between 2 and 4 tensile tests, although in a few instances, aging chamber space limitations allowed only one tensile sample to be tested for a given total dose. When the values of elongation drop below approximately 20 percent, the uncertainty in the measured values can become significant. This is especially true for insulation materials in the form of tubes. After significant aging, the pneumatic jaws of the Instron can cause the gripped area of the samples to crack, leading to preferential failure at the jaws during tensile testing.

The first observation apparent from the results of Figures 1-7 is that e usually decreases smoothly and monotonically towards 0% elongation. The chlorosulfonated polyethylene insulation behaves somewhat differently in that its elongation drops precipitously at first (from 600% to ~200%) followed by a steady slow decline. All the materials become quite brittle as their elongation approaches 0%. This mechanical brittleness is a prelude to the development of cracks in the material, which in turn could lead to electrical failure during cable operation. For these reasons, the elongation appears

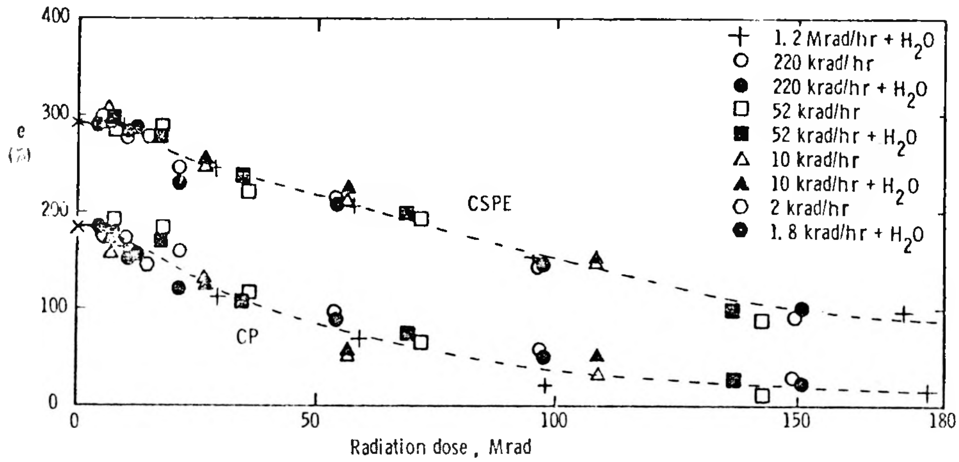

Figure 1. Ultimate tensile elongation (e) vs. total radiation dose for chloroprene (CP) and chlorosulfonated polyethylene (CSPE) jacket materials at various dose rates under dry air and 70 % relative humidity (H_2O) conditions as indicated in the figure.

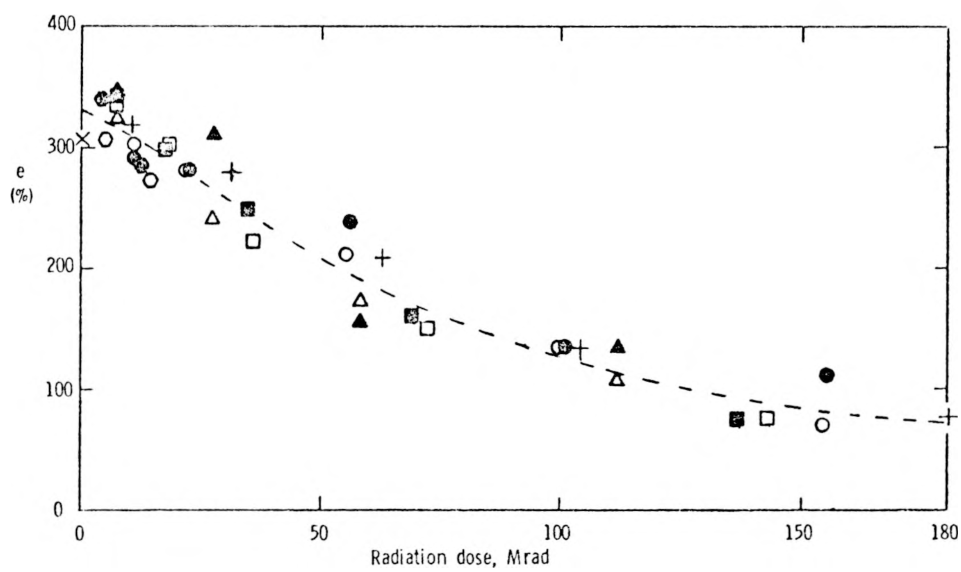

Figure 2. Ultimate tensile elongation (e) vs. total radiation dose for a chlorosulfonated polyethylene jacket (different manufacturer than material in Figure 1). See Figure 1 for explanation of symbols.

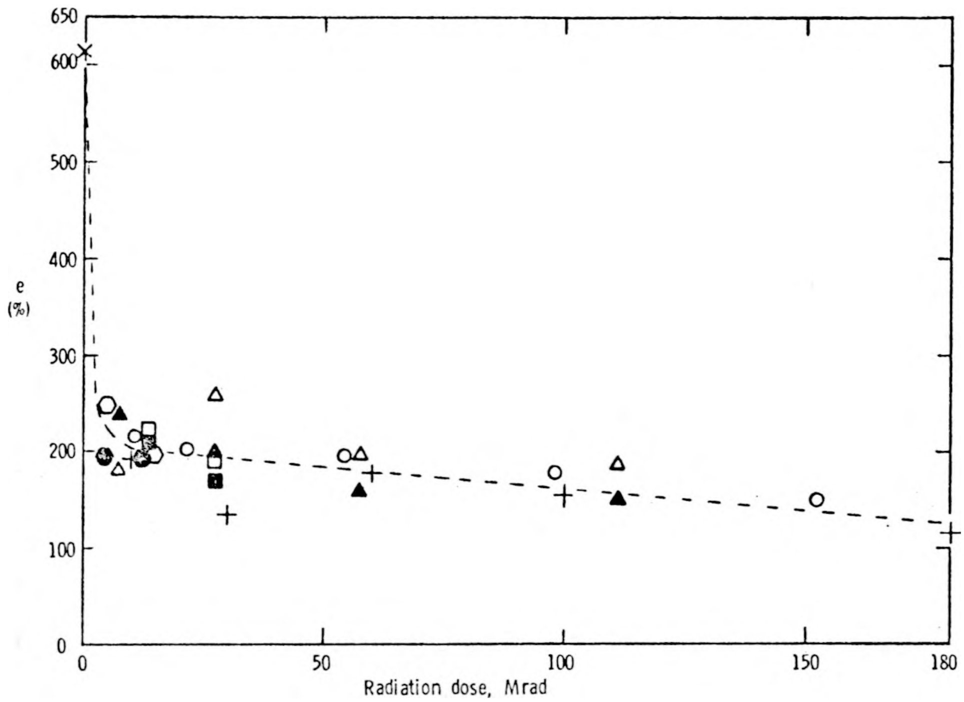

Figure 3. Ultimate tensile elongation (e) vs. total radiation dose for chlorosulfonated polyethylene insulation material. See Figure 1 for meaning of symbols.

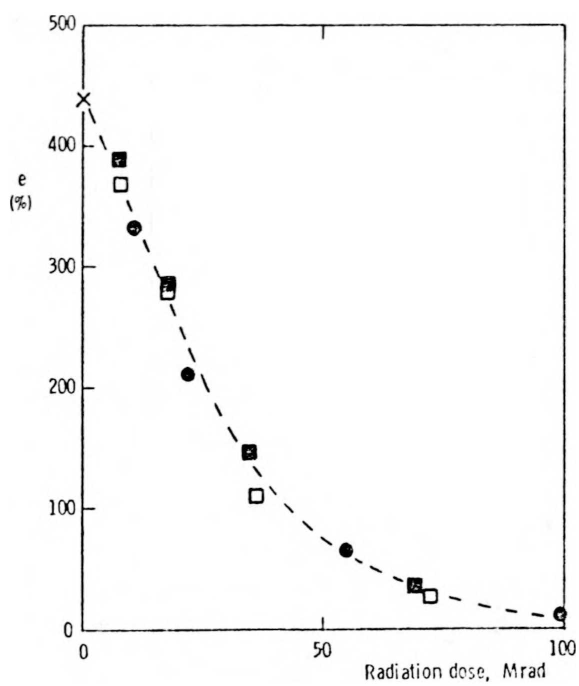

Figure 4. Elongation vs. total radiation dose for silicone insulation. See Figure 1 for meaning of symbols.

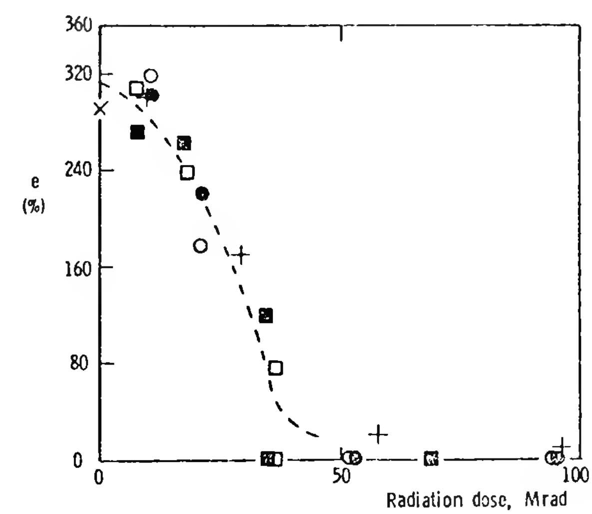

Figure 5. Elongation vs. total radiation dose for Tefzel® insulation. See Figure 1 for meaning of symbols.

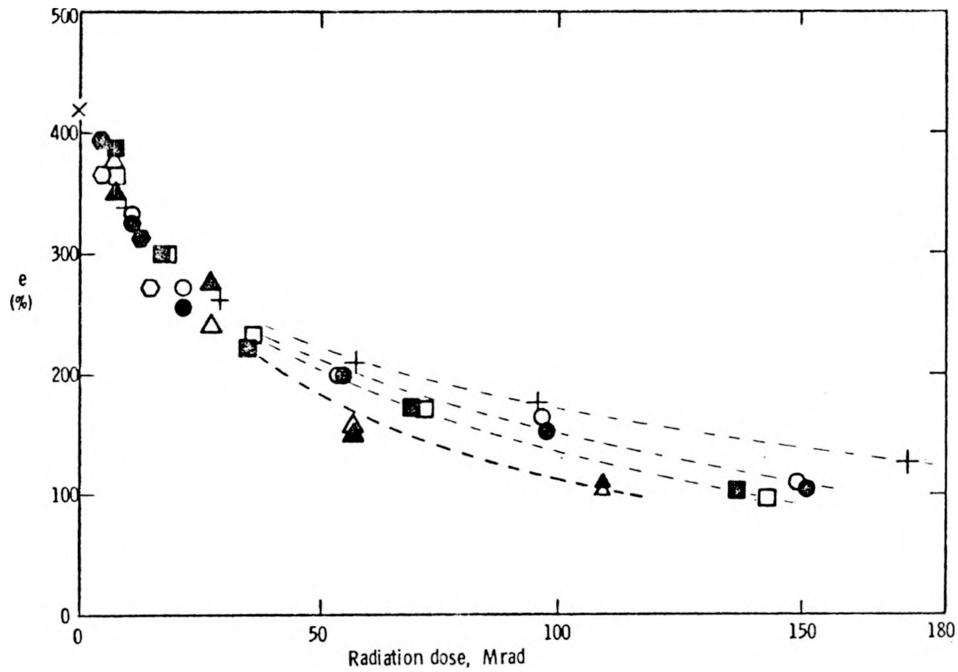

Figure 6. Ultimate tensile elongation (e) vs. total radiation dose for ethylene propylene rubber insulation. See Figure 1 for meaning of symbols.

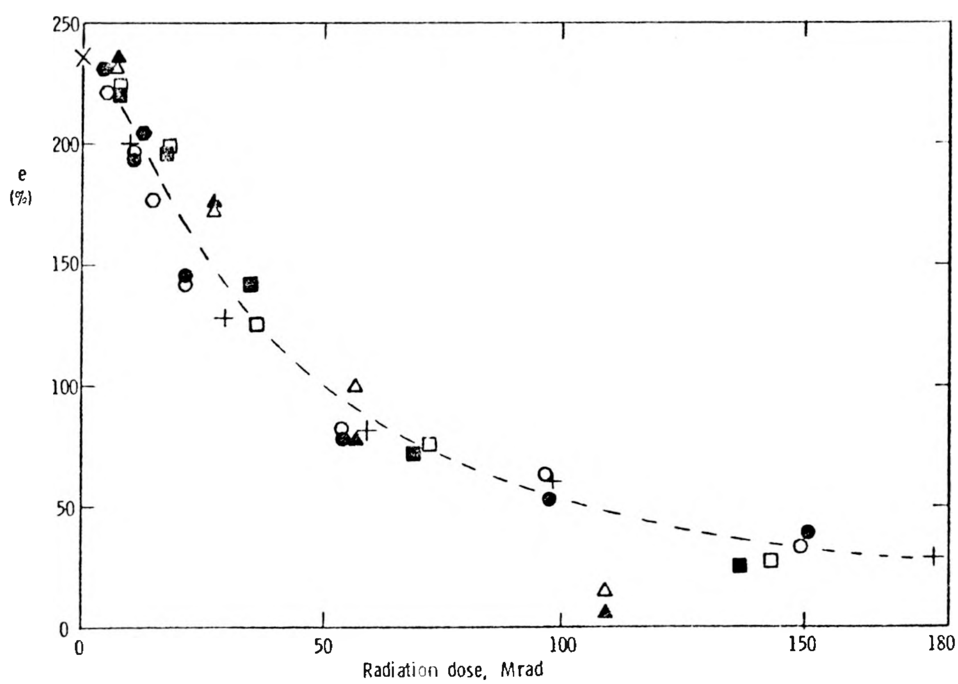

Figure 7. Ultimate tensile elongation (e) vs. total radiation dose for cross-linked polyethylene insulation. See Figure 1 for meaning of symbols.

8

to be an ideal variable for following the deterioration of cable materials. A second interesting observation is that humidity appears to have no noticeable effect on the room temperature mechanical degradation of these materials within the experimental scatter.

The data on radiation dose rate effects are particularly interesting. Figures 1-4 for the chloroprene, chlorosulfonated polyethylene and silicone materials show little indication of any radiation dose rate effects; the degradation appears to be independent of the dose rate and dependent only on the total integrated radiation dose.

The results for Tefzel® in Figure 5 are more difficult to interpret. For this material the scatter in the measured values of elongation tended to be much larger than for the other materials in the study. At times the scatter was so severe that averages of the data were meaningless and individual data points had to be placed on the figures. For example, at 52 Krad/hour in a humid air environment, the measured values of elongation ranged from 5% to 120% after 30 Mrad total dose. The large scatter in the data coupled with the rather rapid deterioration of the mechanical properties tend to make conclusions concerning dose rate effects difficult. Some slight indication of a dose rate effect was noticed at high dose rates, since some elongation (and therefore flexibility) remained after total doses ranging from 50 Mrad to greater than 150 Mrad given at 1.2 Mrad/hour whereas the same doses given at lower dose rates led to a much more brittle (and essentially untestable) material.

The data of Figure 6 show that radiation dose rate effects are important for the ethylene propylene rubber material. For equivalent total radiation doses, deterioration becomes more severe as the dose rate is lowered. This indicates that accelerated simulations which use total integrated dose as a means of estimating deterioration under ambient conditions will underestimate the ambient damage. One mechanism potentially responsible for dose rate effects involves oxygen diffusion. If oxygen is important to the reactions leading to mechanical deterioration in radiation environments and if the oxygen sorbed in a material is used up faster than it can be replenished from the surrounding ambient atmosphere, then the rate determining step in the deterioration will be oxygen diffusion into the material. In such cases, for the same total radiation dose, more deterioration would be predicted at lower dose rates. However, since equilibrium for oxygen sorption would be expected in a few hours for the 1.0 mm thick ethylene propylene rubber material, it is difficult to believe that oxygen diffusion would be important for experiments approaching two years duration. It is anticipated that continuing experiments being run on this material will be helpful in elucidating the reasons for the interesting dose rate effects.

The data for the cross linked polyolefin material is shown in Figure 7. Except for a surprising set of data points obtained at 10 krad/hour after 110 Mrad total dose, there is little indication of radiation dose rate effects. We believe, however, that the 10 krad/hour data is real since consistent results were obtained in the two aging cells (0% and 70% relative humidity). Whether these data indicate that damage will occur at much lower total doses for low dose rates will become clearer after the accumulation of higher total doses has occurred in the 2 krad/hour aging chambers.

Summary and Conclusions

In summary, an extensive aging program is being carried out on polymeric materials stripped from commercial nuclear reactor safety cables. The results obtained so far indicate that humidity is not a significant environmental stress. In addition, for room temperature radiation environments, the mechanical degradation of a number of the materials studied depended only on the total integrated radiation dose. However, dose rate effects were found to exist for the ethylene propylene rubber and some evidence of the existence of similar effects was noted for the Tefzel® and cross-linked polyethylene materials. For materials in which dose rates and humidity effects do not exist, the potential problems associated with accelerated aging are greatly reduced.

It should be emphasized that the studies described in this section pertain to the room temperature radiation degradation of the eight materials. The effects of elevated ambient temperatures and the possible complicating effects of synergism in combined temperature and radiation environments were not included. The choice of a material for an application in a complex combined stress environment must address these problems by attempting to simulate the combined environment degradation of the candidate materials. With this problem in mind, combined environment studies are currently being carried out; some results from these studies are reported in a companion paper.[3]

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SAND78-0346

A4. AN EXPERIMENTAL INVESTIGATION OF SYNERGISMS IN CLASS 1E COMPONENTS*

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ABSTRACT

Nine generic LOCA typetests have been conducted on single- and multi-conductor electrical cables, cable connector assemblies or cable splice assemblies to evaluate synergistic effects due to the (simulated) LOCA radiation environment. While no functional synergisms have been observed, significant material damage and equipment failures have occurred. The tests have demonstrated the complexity and the precision necessary in conducting the tests to avoid biasing the results.

A complementary activity to secure evidence of possible synergistic effects in earlier test programs recognized that qualification of safety-related equipment has been conducted by the nuclear power industry for more than 10 years. Franklin Institute Research Laboratories (FIRL), under contract to Sandia Laboratories, conducted a review of their historical LOCA-typetest data to perform an independent synergism evaluation for Class 1E electric cables. In the conduct of these (66) tests FIRL did not note any obvious correlation between cable performance and whether the typetest was sequential or simultaneous.

Introduction

The qualification of Class 1E equipment [1] for use in safety-related systems of nuclear power generating stations consists primarily of two parts: accelerated aging of the components to put them into a state of advanced life and testing under conditions simulating those that exist inside the containment of a nuclear plant during, and subsequent to, a loss-of-coolant accident (LOCA). Many qualification tests consist of the following sequence: thermal

*This report documents part of the Qualification Testing Evaluation (QTE) Program (A1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

aging; gamma irradiation (simulating the effect of the lifetime ambient radiation as well as that released by the LOCA); and steam/chemical-spray exposure (simulating the non-radiation factors of a LOCA). Such tests are referred to as sequential-exposure tests, or simply "sequential" tests. In another type of test, the environmental exposures are combined in a way that simulates the plant conditions more realistically than is done in a sequential test. A typical "simultaneous" test consists of: combined thermal and radiation aging; and combined radiation, steam and chemical-spray exposure simulating a LOCA.

While the sequential test is not as "realistic" as a simultaneous test, intuitive consensus is that it is a conservative test. This results primarily because the components are thought to be in a more degraded state at the point where they are exposed to the severe thermal transients and high temperatures at which the steam/chemical-spray exposure is initiated, than they are at the comparable point in a simultaneous test (i.e., at the start of the combined radiation/steam/chemical-spray exposure). In fact, IEEE-323 [1] recommends, without supporting justification, that equipment qualification be accomplished in the sequential mode. However, arguing against that rationale is the possibility that there might be significant synergistic effects associated with a simultaneous test that would not be revealed in a sequential test.

To investigate the potential for synergistic effects in Class 1E equipment during typetests, the U. S. Nuclear Regulatory Commission, Division of Reactor Safety Research, has funded the Qualification Testing Evaluation (QTE) Program at Sandia Laboratories and specifically the experimental investigations described in this paper. Two aspects of these investigations are reported. First, nine generic-LOCA typetests have been conducted at Sandia

Laboratories on single- and multi-conductor electric cables, cable connector assemblies, and cable splice assemblies to evaluate for synergistic effects; results of these tests are presented. Secondly, as a complementary effort, the Franklin Institute Research Laboratories (FIRL) performed an independent evaluation of synergistic effects in electric cable, using historical data obtained in over 10 years of testing for the nuclear industry; their study, its development, and results are also presented in this paper.

Conduct of Tests

The objective of this study is to systematically perform sequential and simultaneous tests on a variety of (identical) Class 1 components to evaluate for potential synergistic effects. The testing and, particularly, the analyses have been concentrated on Class 1E (i.e., electrical) equipment. Two important considerations in this study were that "typical" "equipment" be tested. "Typical" meant that the items were purchased from the manufacturers as standard items rather than one-of-a-kind items; all such purchases were stated and understood to be Class 1E equipment per IEEE-323. "Equipment" meant that testing was to be conducted on actual equipment and not its constituent components or materials; this was vital since synergisms associated with functional performance of equipment (not just extensive material damage) are of first-order importance and the principal concern in qualification.

Nine LOCA typetests [2] have been conducted using the generic simultaneous profile shown in Figure 1 or its complementary sequential profile, in which the total radiation dose is sequenced between thermal aging and the other LOCA environments simulation. These profiles can be considered as combined PWR/BWR "worst case" profiles and are abstracted from IEEE-323-1974, Appendix A [1]. It is important to note that the LOCA profiles were intended to be identical in the sequential and simultaneous tests except for the application of the radiation profile; any observed synergistic effects would then only be a result of the radiation application. Again "functional synergisms", not material degradation or damage, is the important consideration.

Test Results

Three generic types of Class 1E equipment have been subjected to these tests: single- and multi-conductor electrical cables (four manufacturers coded as A, B, C, D); cable connector assemblies (three connector manufacturers coded as R, S, T and assembled by the manufacturer on two multi-conductor cable types, A and B); and cable splice assemblies (only one qualified manufacturer on types A and B single- and multi-conductor cables). All test items were monitored during the test for electrical property degradation (e.g., insulation resistance, capacitance and dissipation factor) and all test items were loaded with their rated current/voltage loads to simulate use conditions.

In general, the electrical cables showed no synergistic effects in either the electrical or material property (i.e., elongation) data. For some cable types, the material damage was visually significant as shown in Figure 2; qualitatively, there also appears to be a definite material damage synergism, but the difference in total radiation exposure is an important and significant variable. The type C cable suffered extensive damage as characterized by circumferential cracks in the cable jacket, Figure 3. No multi-conductor cable failed (in the sense of failing to carry rated current/voltage loads) in any tests, but at least one single-conductor cable of both types A and C did fail; these failed because of severe cracks in the cables' insulation/jacket and resulting electrical shorts, but, while troublesome, the numerical statistics are too minimal to make any definitive conclusions.

Although no synergistic effects were observed in the cable connector assemblies data, most assemblies that were tested failed during the saturated-steam profile phase of the tests. Even though the data is sparse, certain of the data indicate (1) there is a greater failure rate for the sequential test and (2) the failures may be a function of the assembly, not just the connector. That is, the mating interface of the connector/cable may be the "weak link" of the assembly; therefore, it may be necessary to demand qualification of each proposed connector/cable combination. Damage to a cable connector assembly is shown in Figure 4. Although severely damaged, this assembly did not fail, but increasing degradation was evidenced by a constant rise in capacitance over the test duration. In those assemblies which did fail, there was pin-to-pin shorting, caused by the inability of the elastomeric O-ring to prevent entry of moisture to the connector faces.

All cable splice assemblies have functioned satisfactorily and have shown no evidence of synergistic effects. The only damage observed was some surface roughness of the splice sleeve (primarily in the bend area of the splice) and of the extruded splice adhesive. See Figure 5.

Data on cable material damage was also available from the cable and insulation tensile specimens included in these tests. Table 1 summarizes the tensile specimen results; two compounds of typical radiation-crosslinked, highly-flame-retarded polyolefins (designated as A and B in the table) were tested. To evaluate the effect of "adequate" sample age, samples of each were given a "preage" of 168 hours at 175°C in a circulating air oven. For Compound A-IX and preaged A-IX, and Compound B-VIII and preaged B-VIII, the data are logically self-consistent; i.e., the preaged samples are always more severely degraded. While anomalies exist in the other two data sets, the data illustrate the potential importance of accurate and adequate aging. It could be postulated, for example, that inadequate aging might mask synergistic effects in LOCA typetesting.

A preliminary evaluation of synergistic effects can be made using this data. The data for groups 5 and F are roughly comparable, and a preliminary survey would indicate no synergisms exist. Similarly, groups 4 and D are comparable; no synergisms exist for Compound A, but the significance and differences in the Compound B data may be real and suggest very minor synergistic effects.

FIRL Historical Typetest Data Review

The Franklin Institute Research Laboratories conducted a study [3] of their own 1969-1977 historical data to determine whether or not synergistic effects resulted from simultaneous applications of radiation, steam, and chemical spray during qualification testing of Class 1E electrical cable. They completed 66 cable qualification programs in those eight years. Of these, 9 were conducted with the cables exposed simultaneously to environments of gamma radiation, steam, and chemical spray. In the remaining 57 sequential tests, the gamma irradiation was conducted separately from the steam/chemical-spray (S/C) exposure, either in one continuous dose preceding the S/C exposure or in two doses, one preceding and one following thermal aging. Because the test descriptions and results are proprietary, it was necessary to obtain permission from the companies that funded the tests to use the data in this study. The descriptions of 43 sequential tests and 6 simultaneous tests were made available for the evaluation.

Apart from the question of synergistic effects, the historical evolution of qualification testing and test methodology is represented in these test data. A few pertinent statistics may serve to illustrate this point:

- A No. 12 AWG wire was the most popular conductor size, being included in slightly more than half of the tests.

- Crosslinked polyethylene was the most frequently tested conductor insulation, being included in about one-fourth of the tests.

- A gamma radiation level of 200 Mrad or more was used in approximately half of the tests.

- A peak temperature of 340°F (171°C) or more was used in the steam/chemical-spray exposure of approximately half of the tests. The first three 340°F (171°C) tests were conducted in 1970; after 1974, almost all tests included a 340°F (171°C) peak temperature.

- Durations of the steam/chemical-spray exposures were as follows:

Less than 1 day:	7 tests
1 day to 8 days:	11 tests
9 days to 28 days:	14 tests
30 days to 33 days:	9 tests
100 days to 110 days:	4 tests

  Most of the short-duration tests were conducted in the early years.

However, among the 49 test programs examined, only a single pair of tests (one sequential and one simultaneous test) was found to meet certain basic requirements for synergistic-effects comparison; i.e., that the cable specimens and the test profiles be essentially the same in both tests. The comparative analysis indicated a number of similarities and dissimilarities between these two programs. The two tests analyzed involved similar cables, and both included thermal aging at 250°F (121°C) for 168 hours (with some cables receiving additional thermal aging), 200 Mrad of gamma radiation, and a 30-day exposure to steam and chemical spray with two 3-hour dwells at a temperature of 346°F (174°C). The cables were multi-conductor #12 and #14 AWG 600 VAC control cables with primary insulations of flame-resistant crosslinked polyethylene, ethylene propylene rubber, and silicone rubber. The outer jackets consisted of flame-resistant neoprene and silicone-saturated asbestos.

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

Discussion and Summary

The experimental work represents the first known systematic attempt to specifically evaluate for synergistic effects in Class 1E equipment resulting from LOCA-typetesting methodology influences. No functional synergisms have been observed in the testing to date, but equipment "failures" have occurred and the test results have suggested test methodology deficiencies and areas requiring additional experimental investigations.

With regard to the FIRL study, after a comprehensive review of 43 sequential and 6 simultaneous electric cable LOCA typetests, it was concluded that none of the tests contained sufficient data and similarities to permit definitive evaluation of synergistic effects. But in the conduct of these tests, FIRL has not noted any obvious correlation between cable performance and whether the test was of the sequential or simultaneous type.

In summary, the experimental investigation and the data review demonstrate the complexity and the precision necessary in the test conduct to detect synergistic effects in the equipment response.

References

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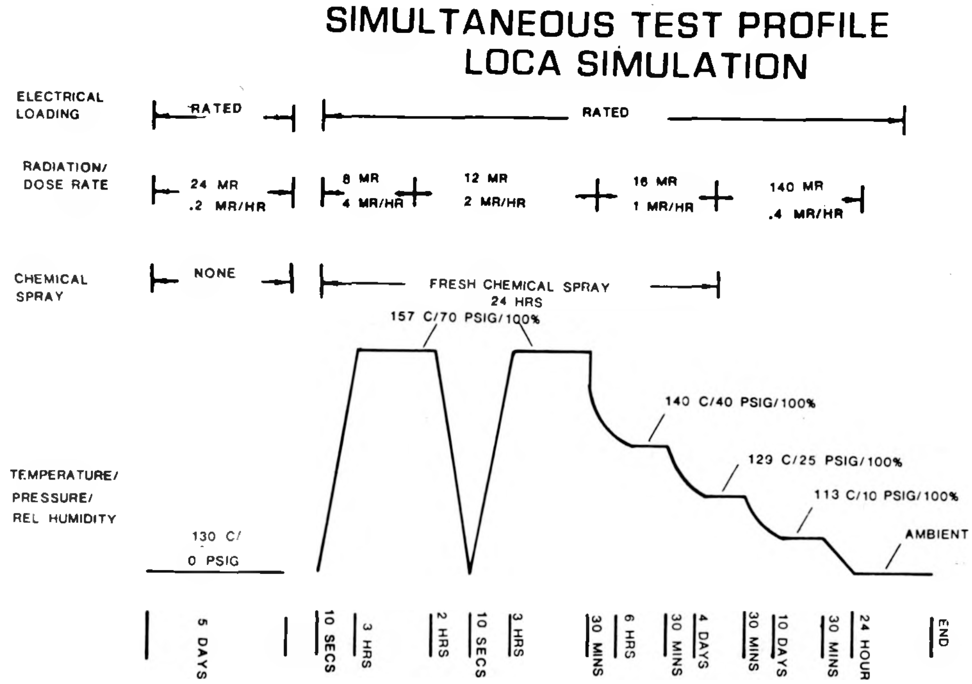

Figure 1. The simultaneous test profile abstracted from Reference 1.

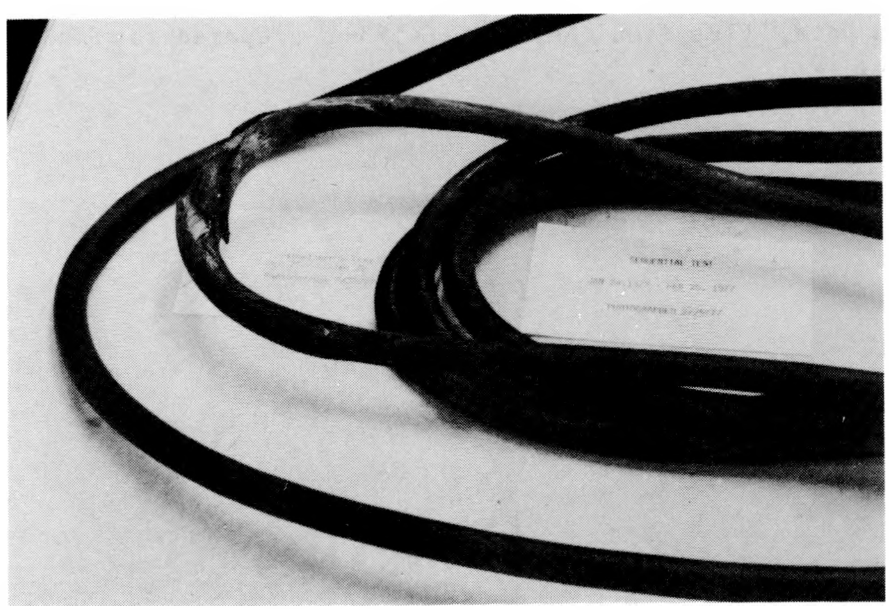

Figure 2. A comparison of type A multi-conductor cable damage. The single cable at the left is an untested cable; the single cable at the center is from a simultaneous test; the three at the right are from a sequential test. Total radiation dose, at the cable bend, was 425 megarads for the simultaneous test and 325 megarads for the sequential test.

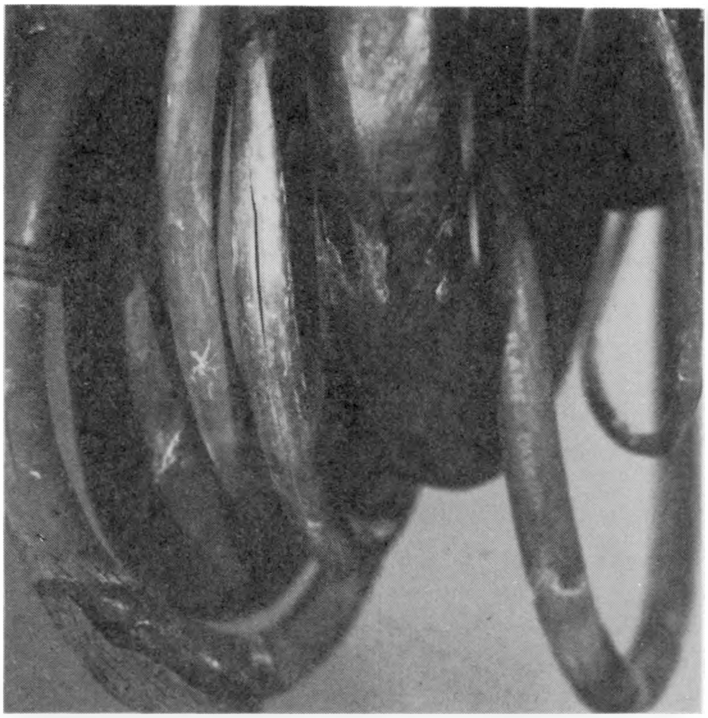

Figure 3. Simultaneous posttest (Test 9) view of the cable bend area: type A/straight splice (lower left); type B cable to the left of the type C cable (with the large circumferential crack); type A cable (lower right, no cracks). Total radiation dose in the bend area was about 250 Megarads.

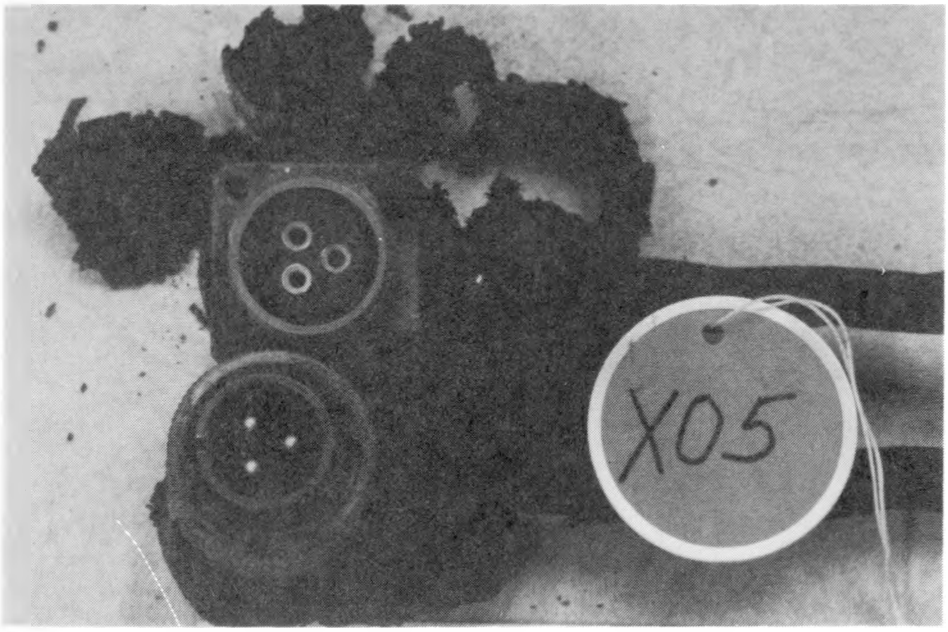

Figure 4. Damage to cable connector assembly R/B, R-5, after a simultaneous test; the connector received approximately 360 Megarads total dose.

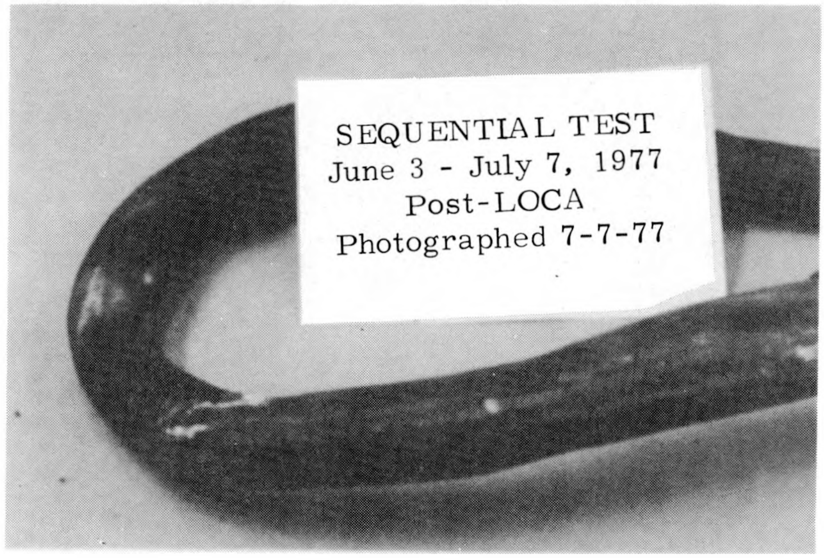

Figure 5. The type A/bend splice after a sequential test and approximately 360 megarads dose. Some minor surface roughness is visible in the bend area.

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TABLE 2.5

Insulation Tensile-Specimens
Physical Properties Comparisons

Group ID	Conditioning Sequence	Radiation Dose Range (Megarads)	Percent Ultimate Elongation (ϵ/ϵ_o) A	Preaged A	B	Preaged B
			Test VIII--Sequential			
A	Aging	-	208 (0.39)	310 (0.58)	442 (0.94)	278 (0.59)
B	Aging + LOCA	-	315 (0.59)	248 (0.47)	188 (0.40)	87 (0.19)
C	Aging + Radiation	38 - 95	85 (0.16)	72 (0.14)	213 (0.46)	147 (0.31)
D	Aging + Radiation + LOCA	38 - 95	158 (0.30)	125 (0.24)	128 (0.27)	62 (0.13)
E	Radiation	12 - 31	457 (0.86)	255 (0.48)	430 (0.92)	222 (0.47)
F	Radiation + LOCA	12 - 31	293 (0.55)	170 (0.32)	178 (0.38)	85 (0.18)
G	LOCA	-	427 (0.81)	207 (0.39)	208 (0.44)	125 (0.27)
			Test IX--Simultaneous			
1	Aging	5 - 8.5	310 (0.58)	300 (0.57)	430 (0.92)	273 (0.58)
2	Aging + LOCA	20.5 - 38.5	243 (0.46)	203 (0.38)	123 (0.26)	140 (0.30)
3	Aging	10.5 - 18.5	370 (0.70)	230 (0.43)	390 (0.83)	273 (0.58)
4	Aging + LOCA	45.5 - 78.5	160 (0.30)	140 (0.26)	100 (0.21)	113 (0.24)
5	LOCA	15.5 - 30	280 (0.53)	190 (0.36)	140 (0.30)	100 (0.21)
6	LOCA	35 - 60	203 (0.38)	140 (0.26)	130 (0.28)	110 (0.24)

NOTES:
- Initial percent ultimate elongation for A and B was 530 and 468 respectively.
- Radiation dose varied from top (1st figure) to bottom (2nd figure) of sample.
- All values are the average of three sample tests.
- Pre-aged samples received additional thermal aging of 168 hours at 175°C by manufacturer.
- Numbers in parenthesis are fractional elongation of initial values.
- All data measurements were performed by the material manufacturer; the data were supplied by private communication.

A5. QUALIFICATION OF CLASS IE EQUIPMENT:

THE ROLE OF THE UTILITY AND ARCHITECT-ENGINEER*

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Introduction: Key principles and procedures to be addressed as part of the evolution and implementation of an acceptable qualification program, as described in IEEE-323 [1], include:

(a) determination and demonstration of a qualified life for all Class IE equipment, and/or

(b) establishment of an ongoing qualification program.

In fulfilling this obligation the nuclear industry is currently being challenged to come up with an interim solution to qualify equipment to meet their immediate plant licensing and operating schedules. They must also influence the long-term technology development efforts currently underway in this area to fill the "state-of-the-art-gap" that exists between conceptually complete qualification programs (e.g., Reference 1) and available, proven techniques (e.g., for simulating aging, synergistic effects). Considerations involved in meeting this challenge include:

(a) gradual consolidation of the competitive forces in the market place,

(b) restrictions on product innovation,

(c) availability of replacement parts over a long time horizon to assure continued plant operation with minimum downtime,

(d) risks involved in commitments made on technology yet to be fully accepted in the scientific and engineering communities; and

(e) increases in initial investment as well as operating costs.

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

This dictates that the users*, the vendors, and the test laboratories, as well as other segments of the industry, work together toward reaching their common goal of achieving acceptable equipment qualification. This implies that some of the traditional economic relationships that exist between these various segments may need to be detached. A new process of equipment selection has evolved resulting from the current state-of-the-art in equipment qualification; now the user is required to make a cost-effective judgment based on the potential capability of the vendor to qualify the equipment and not necessarily on equipment the vendor has actually and previously qualified.

This setting requires a definition of the organizational objectives of the different factions of the industry and an examination and redefinition, if necessary, of the roles played by these different organizations as well as their inter-relationships.

In this paper the authors discuss:

(a) the roles to be assumed by the utilities and the architect-engineers,

(b) the objectives behind this role definition, and

(c) the development of a Class IE equipment data file by United Engineers & Constructors Inc. (UE&C) under subcontract with Sandia which provides a working example of these roles. Yankee Atomic Electric Company (YAEC), as the agent for the plant owner, participated in this effort through UE&C.

Objectives and Roles: Traditionally, users have objectives of purchasing and installing equipment that meets or exceeds the performance requirements and results in lowest evaluated cost. This implies evaluating the demonstrated capabilities of the equipment versus the cost on a comparative basis.

The suppliers on the other hand have the objective of developing and marketing equipment that meets the demands of the user while remaining competitive. This means spreading the cost of research and development, and marketing over a volume of sales. However, the nuclear power industry's share (especially now) of the instrumentation equipment market is smaller in size

* The utility and the architect-engineer (AE) are referred to as the users throughout this document.

compared to other industries, such as petro-chemical and process industries. Yet demands of the nuclear industry in terms of demonstrated[+] capabilities are more stringent than those of other industries.

In this situation suppliers' responses, such as the ones quoted below, come as no surprise.

- "Considering that the size of the market is less than 5% of our company's overall sales volume it becomes unattractive to stay in the nuclear market," or

- "We will be happy to quote on any specific tests or qualification requirements which may be determined to be necessary to obtain approval of the responsible regulatory agencies.", or

- In the overall qualification program a "preventative maintenance program, including critical components, will be defined and identified to the plant owner via equipment manuals. Periodic testing is required for safety-related Class 1E equipment .... It is expected that the downtime during the testing would essentially be the same as experienced with the present periodic test program. Procedures already developed to implement the plant periodic test program would still apply and the additional test and inspection most likely can be performed within the same time frame. Only manpower levels will be affected."

The quotes above clearly illustrate suppliers' attitudes which have developed and these provide the bases for the users' concerns listed below:

(a) Increasing fear of driving competition away from the market place.

(b) Money commitments over an uncertain time frame on undefined programs with uncertain results.

(c) Implied commitments made for him by his suppliers (over the plant life) for certain required maintenance programs to maintain qualification which could significantly affect the downtime and manpower levels required to handle maintenance, thus severely impacting the plant operating cost.

\+ Requires "generation and maintenance of evidence to assure that the equipment will operate on demand to meet the system performance requirements, under the specified service conditions and during its installed life", [1].

This situation calls for a risk management approach to decision making. In this setting an effective strategy would be for the users to assume leadership roles to:

I participate in industry standards formulation groups and provide appropriate input at the working level,

II form user and industry task groups to pool industries' funds and technical resources in an attempt to standardize the approaches to qualification,

III identify the areas where additional information and studies are required to resolve long term technological problems and participate in the initiation of the necessary programs to solve them, and

IV develop guidelines for interim solutions consistent with the state-of-the-art.

In the paragraphs below we will review these items in detail.

I Participate in industry standards formulation groups and provide appropriate input at the working level.

The resulting benefits of participating in these efforts are:

(a) published standards that establish an industry norm for assuring a degree of consistency in design and testing practices

(b) exchange of technical ideas and discussion of problems in specific areas, thus broadening the perspectives of the individual and his organization. In some cases this could help forewarn of potential problem areas.

In the United States, professional societies, such as the Institute of Electrical and Electronic Engineers (IEEE), the American Society of Mechanical Engineers (ASME), the Instrument Society of America (ISA), and the American Nuclear Society (ANS), involved in the writing of standards seek voluntary participation. These standards are written and published by a consensus approach. Yankee Atomic Electric Company, United Engineers & Constructors, and Sandia Laboratories have participated on the committees for many years and contributed to the formulation of several of the qualification standards used in the industry today.

II Form user and industry task groups to pool industries' funds and technical resources in an attempt to standardize the approaches to qualification.

The Public Service Company of New Hampshire (PSNH), Yankee Atomic Electric Company (YAEC), and United Engineers & Constructors (UE&C) have, as part of their effort to qualify Class 1E equipment for the Seabrook Project, initiated a systematic task force [2] approach to finding answers to the problems involved, and as a result, have gained insight into the complexities involved.

As depicted in Figure 1, the testing hierarchy usually encountered involves the Nuclear Steam System (NSS) supplier who has the responsibility for qualification of a substantial number of Class 1E systems. It is, therefore, imperative to develop a coordinated approach to qualification programs for the NSS and the balance-of-plant Class 1E equipment. To address this, Westinghouse, one of the NSS suppliers in the U.S.A., formed an industry task group early in 1978.

PSNH, YAEC, UE&C and a number of other utilities and AE's are involved in this task group. In the meetings to date, the group has addressed some of the generic issues pertaining to the Westinghouse topical report [3] on this subject. Westinghouse also has utilized these meetings as a forum for updating all involved utilities and AE's on the current status of their testing programs and R&D efforts in the area of qualification. An important subject yet to be addressed by this group is that of developing acceptable ongoing qualification programs and pooling resources to minimize the redundancy in programs on similar equipment, thus minimizing the cost to the users.

III Identify the areas where additional information and studies are required to resolve long term technological problems and participate in the initiation of the necessary programs to solve them.

Through the use of task groups, areas requiring further research and development efforts to gain better insight into qualification techniques must be identified and prioritized. Some of the areas already identified include:

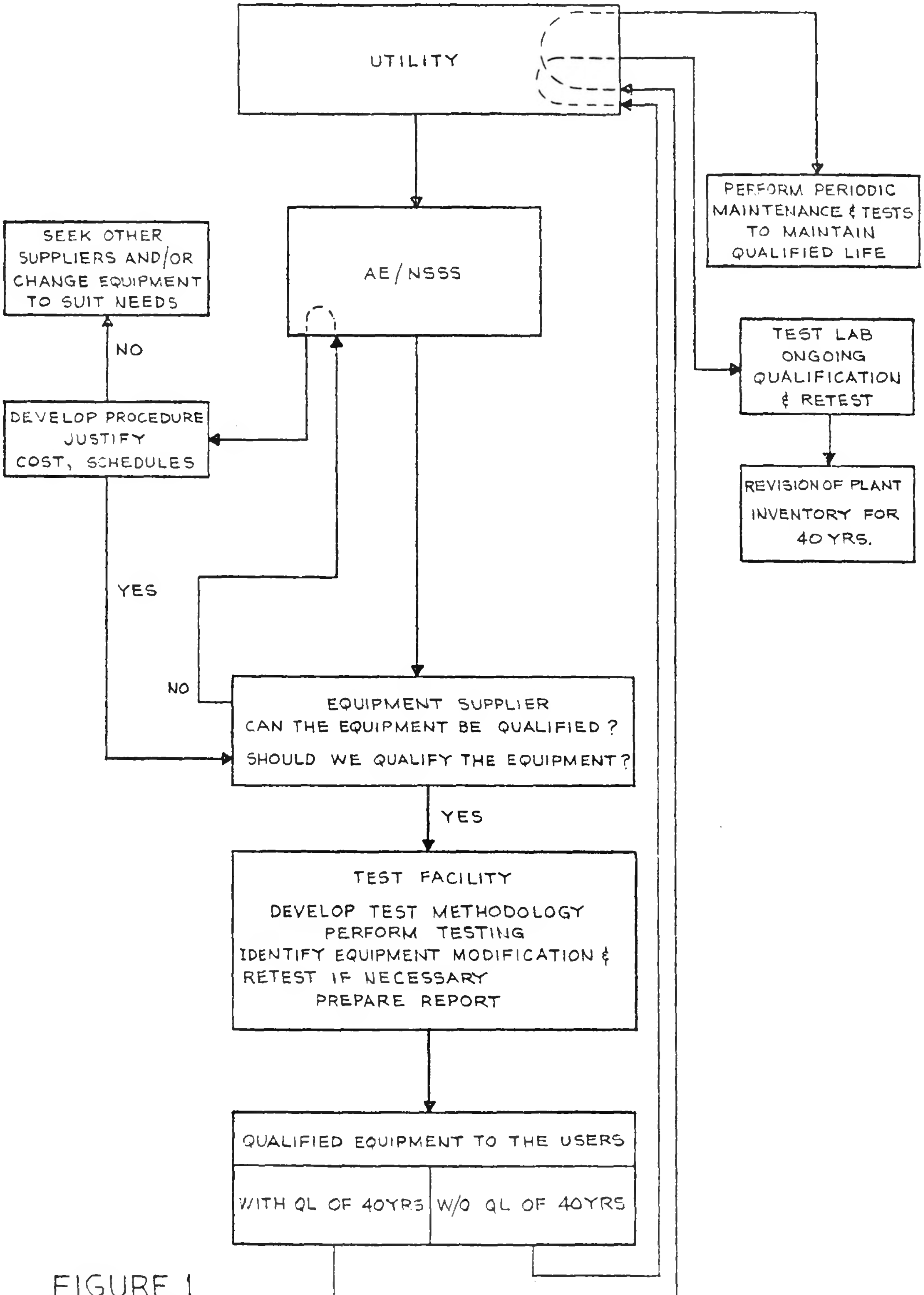
UTILITY
PERFORM PERIODIC MAINTENANCE & TESTS TO MAINTAIN QUALIFIED LIFE
AE/NSSS
SEEK OTHER SUPPLIERS AND/OR CHANGE EQUIPMENT TO SUIT NEEDS
NO
DEVELOP PROCEDURE JUSTIFY COST, SCHEDULES
TEST LAB ONGOING QUALIFICATION & RETEST
REVISION OF PLANT INVENTORY FOR 40 YRS.
YES
NO
EQUIPMENT SUPPLIER
CAN THE EQUIPMENT BE QUALIFIED?
SHOULD WE QUALIFY THE EQUIPMENT?
YES
TEST FACILITY
DEVELOP TEST METHODOLOGY
PERFORM TESTING
IDENTIFY EQUIPMENT MODIFICATION & RETEST IF NECESSARY
PREPARE REPORT
QUALIFIED EQUIPMENT TO THE USERS
WITH QL OF 40YRS
W/O QL OF 40YRS

FIGURE 1

(a) synergistic effects of simultaneous steam, chemical and radiation environments in the different materials/subassemblies used in the construction of Class 1E systems

(b) determination of whether simultaneous or sequential testing results in conservative testing of Class 1E systems

(c) development of acceptable aging simulation methodologies for the various Class 1E equipment

Having identified these areas, the users must then initiate the necessary research programs through organizations like the Electric Power Research Institute (EPRI) and test laboratories.

From time to time YAEC and PSNH have identified to EPRI areas where specific R&D is required and have supported programs aimed at focusing on these issues. The formation of an EPRI adhoc committee (in which YAEC participates) is an example of the type of effort recommended.

Utilities and AE's must also cooperate in R&D programs initiated by the governmental agencies through national laboratories such as Sandia Laboratories. In this case, users' cooperation is required in supporting the laboratories with necessary application oriented expertise. UE&C and YAEC involvement in the current Sandia program is an example of such an effort.

IV Development of guidelines for interim solutions consistent with the state-of-the-art.

For each individual equipment item, the users must review its design, construction, performance requirements for the application intended, and the current state-of-the-art for its testing and qualification to develop an acceptable qualification program for that equipment. More specifically, it is the utility and architect-engineer's responsibility to:

(a) <u>identify the safety-related function of each piece of Class 1E equipment and develop acceptance criteria through performance requirements and allowable tolerances</u>

For example, the ventilation stack radiation monitor is a Class 1E system. Its safety function is to monitor the gaseous discharges from the plant. It consists of a filter tape system and detectors for particulate monitoring and detectors for gaseous monitoring. The portion of the equipment used for particulate monitoring can be excluded from qualification testing if it can be demonstrated that its functioning or malfunctioning will not affect the safety-related portion of the system.

(b) establish a realistic qualified life that can be achieved for each type of equipment based on cost, product-life-cycle, operating and maintenance experiences in other plants, and the vendor's own design experience

For example, if an instrument costs $300 today with an initial qualified life of 5 years and if it will cost $70,000 in current dollars for a continuing qualification program to achieve a qualified life of 40 years, it would be economical* to consider replacement every five years rather than commit to the long-term expenditure now. This also has the advantage of providing the flexibility required to factor in product-life-cycle influences in later equipment selection decisions.

(c) define an acceptable maintenance and testing program for each type of equipment that should be factored into establishing and demonstrating a qualified life

If an equipment item has a qualified life of 5 years which is contingent upon performing maintenance, repair or replacement, then it is necessary to review these contingency requirements and ensure that they are compatible with the scheduled downtime of the plant and available maintenance manpower. For example, if the qualified life of a motor is contingent upon stripping down and inspecting the winding every year, this would require a commitment of a number of people to perform this task, and/or an increase in the plant downtime. More importantly, it may increase the likelihood of equipment failure due to probability of a defective reassembly. It is, therefore, important to develop acceptable maintenance programs and review the contingency maintenance provisions for the qualified life.

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* This computation takes into account effects of escalation at 8% and discount rate at 8% on the money invested.

(d) develop guidelines for a complementary continuing qualification program that will result in cost savings

All equipment with qualified life less than the design life of the plant will require ongoing qualification programs. For example, if three utilities utilize transmitters of the same manufacture and type it would be prudent to establish a joint ongoing qualification program to achieve a uniform approach at a minimal cost.

Appendix A which describes the qualification program for panel-mounted indicators, is an example of the type of effort being recommended. This respresents a "best available" program in the industry for this type of equipment.

The Seabrook Project task force has successfully initiated several such qualification programs. They have been working through the suppliers, and (where practical) utilities, to establish consistency in approach as well as minimize redundancy in ongoing programs.

Test Laboratory Interface: The Seabrook Project task force determined that the expertise for deciding on the basic test methodology based on available test facilities, equipment geometry, design and operating characteristics is concentrated largely in the various test laboratories. It was also reasonable to believe that a research of the type test data already available at some of the major laboratories would preclude the need for some of the routine testing and analyses. The Seabrook task force held discussions with some of the major test laboratories (Franklin Institute, Wyle Laboratories, and Acton Laboratories) in regard to providing this assistance as required. It was during this Seabrook task force effort that Sandia Laboratories engaged United Engineers & Constructors to assist in the Qualification Testing Evaluation (QTE) Program [4]. YAEC participated in this effort through UE&C. The Seabrook task force viewed this as another opportunity to work with the different agencies that comprise the nuclear industry, in this case a national test laboratory engaged in assisting the U. S. Nuclear Regulatory Commission in testing and assessment programs.

Sandia Program: The QTE Program required the establishment of a data file on the various Class 1E equipment, their service and operating conditions, normal and accident environmental conditions, safety functions, etc. The Seabrook Project was selected as the reference plant to compile this data bank based, in part, on the fact that the Project had an established, ongoing, data compilation activity. The schedule and scope of the activity was modified somewhat to suit specific requirements imposed by the QTE Program.

The data file development was organized into multiple activity phases. Initially, a comprehensive generic list of Class 1E equipment for Seabrook was compiled. From this list a priority ranking was formulated on the bases of equipment location, supplier and order status, and potential for the equipment to be subjected to the "worst" environmental conditions. The principal activity concentrated on in-containment equipment which was originally identified as follows:

- -Electronic Transmitters
- -Motor-Operated Valve Actuators
- -Limit Switches
- -Differential Pressure Switches
- -Area Radiation Monitors
- -5 kV Cable
- -600 Volt Power Cable
- -600 Volt Control CAble
- -Terminal Blocks
- -Connectors
- -Penetrations
- -Thermocouple Cable
- -Instrumentation Cable
- -Penetrations
- -0 - 100 Hp, 460 V Motors
- -125 - 250 Hp, 460 V Motors
- -Pneumatic-Operated Valve Actuators
- -Solenoid Valves
- -Thermocouples
- -Neutron Monitors
- -Resistance Temperature Detectors
- -Level Switches

The second phase of this effort, to complete the data file compilation, had two major parts: to prepare a complete Class 1E equipment list, and to compile a complete data package for each generic type of equipment. The equipment list included information on:

- -Equipment Identification
- -Service Legend
- -Supplier
- -Equipment Location
- -Applicable Accidents
- -Safety Function Performed
- -Start Time of Function
- -Duration of Function
- -Normal Operating Environments
- -Accident Environmental Conditions
- -Qualification Parameters

In support of the list, descriptions of the systems and system functions, the environmental design criteria, and the design basis accident profile and duration were developed.

For each type of generic equipment a complete data package has also been developed. The packages include information on at least each of these topics:

- -Function and Operating Principle
- -Description and Interfaces
- -Operating and Environmental Conditions
- -Performance Specifications
- -Materials of Construction
- -Schematic/Circuit Diagrams
- -Maintenance Procedures
- -Calibration Procedures
- -Vulnerability Details
- -Manufacturer's Technical Bulletins
- -Test Reports

The data accumulated is being used by Sandia to assess and evaluate the available qualification methodologies and to evaluate the "LOCA-sensitivity" of the various generic equipment. The results of these evaluations may also dictate a prioritized methodology testing schedule.

Certain insights and auxiliary benefits were directly derived from the study. As a result of accumulating this data, it was discovered that the industry uses the worst case design-basis more than is required. For example, the post-accident time duration is generally conservatively estimated at one year, whereas in actuality, there is little basis for such a qualification number (i.e., the real "accident" may be much shorter). Similarly, enveloping all environmental parameters to minimize the number of qualification profiles may result in unwarranted conservatism for some generic equipment. Another benefit of this data collection program was the results of an additional state-of-the-art survey of the industry. It was noted that since the last survey was done in early 1976 [2], many more manufacturers have either performed, or committed themselves to perform, qualification testing by taking a "best available alternative approach" to simulating aging. In addition, increasing acceptance of the use of natural aging* as the preferred technique to validate the simulated aging experiments has been observed.

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* Appendix A provides an example of the use of natural aging as validation techniques.

Conclusion: In summary, the utility and the architect-engineer, as the users, must assume leadership and bring to bear their risk management skills in solving the complex problems of equipment qualification. A disciplined approach to this problem in developing the qualification program and their supporting bases requires close cooperation of the different factions of the nuclear industry. Adoption of the roles recommended in this paper by the different organizations will lead to the closer working relationships required, thus ensuring success in meeting the challenge ahead. The generic data file developed under Sandia sponsorship by UE&C with YAEC involvement is another example of exercising this leadership role and of an opportunity for significant impact by the ultimate users of Class 1E equipment.

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3. W. G. Jordan, D. G. Lorentz, and R. G. Miller, "Methodology for Qualifying Westinghouse PWR-SD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Revision 1, September, 1977.

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APPENDIX A

An adequate qualification program for panel mounted indicators shall consist of the following:

1. A design proof test shall be performed on a set of old indicators. (Indicators that have been in service in some plants.) Efforts shall be made to obtain reasonable documentation or certification to the fact that they have not been tampered with or that major undocumented maintenance has been performed, and that they have been in service in a control room environment for all the years of service. The purpose of this test will be to establish the design constants for these indicators.

2. A set of new indicators shall be aged to stress them to a level equivalent to that experienced due to shipping and storage. These indicators shall then be subjected to the same design proof test as the old indicators. The purpose of this would be to generate a set of design constants on new indicators, and compare them with those generated in Item 1 above.

3. A set of performance tests such as sensitivity, accuracy, and repeatability shall be run on old and new indicators and comparisons made.

4. Comparison of the design constants and performance parameters made in Items 2 and 3 above is expected to prove that age does not have any effect on the indicators, or if it does, will provide a clue to the relationships that may be present.

5. Seismic tests shall be run both on the new and the old indicators to verify the seismic capability using the random motion input test procedure. This seismic test will only be run on indicators singly mounted.

6. A performance test will again be run to establish the change in performance due to a seismic test.

7. The indicators will then be subjected to a performance test at a design temperature of $120^{o}F$ and 95% humidity.

8. A life expectancy, or expected qualified life, will now be generated using the data from Items 1 thru 7 above. It is expected that a five year qualified life will be possible.

9. An ongoing qualification program will be established to extend the five year qualified life. The ongoing qualification program will consist of:

a) installing a set of six old and twelve new indicators in conditions similar to that in control rooms and keep them energized

b) performing periodic monitoring and maintenance on those indicators similar to the ones that will be performed in a plant

c) maintain full documentation regarding items a and b above

d) at periodic intervals (5 and 15 years for old indicators and 5, 15 and 30 years for the new indicators) seismically test them to demonstrate their seismic capability

This program is expected to result in extension of the 5 year qualified life to 40 years.

12

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A6. THE HYPOTHESIZED LOCA RADIATION SIGNATURE AND THE PROBLEM OF SIMULATOR ADEQUACY*

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ABSTRACT

The hypothesized LOCA radiation source specifications have been translated into energy release rates and spectra for a wide range of operating conditions. The results will be used as a basis for evaluating the suitability of radiation sources, such as Cobalt-60, to simulate the damage in typical Class 1E equipment.

INTRODUCTION

Qualification of certain safety-related nuclear power reactor equipment is necessary to ensure that its functionability will not be compromised or limited by the environments expected under postulated accident or design-lifetime conditions. Radiation qualification is but one aspect of a larger program which must consider the effects of all potential environments. Proper radiation qualification requires a precise knowledge of the actual radiation environment signature, as well as an understanding of the physical mechanisms of radiation effects for the specific equipment being qualified. In this paper the translation of the (Regulatory) accident radiation source specifications into energy release rates and and spectra is discussed and the problem of selecting a radiation source that properly "simulates" the accident source is examined.

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

SOURCE SPECIFICATIONS

The hypothesized loss-of-coolant accident (LOCA) radiation source for a light water reactor has been specified by the U.S. Nuclear Regulatory Commission in its Regulatory Guide 1.89 [1] for purposes of qualification testing of Class 1E electrical equipment in commercial U.S. power reactors. The approach taken in the Guide is to specify certain fractions of fission products by categories that are assumed to be released from the reactor core and distributed within the containment structure. The Guide presently calls for two radiation source terms (for containment heat removal systems and for other safety-related electrical systems) and three distribution categories for each source (airborne, waterborne and plate-out). The table summarizes the specifications of fission products.

SOURCE TYPES AND DISTRIBUTION CATEGORIES [1]

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

It is necessary for the user to translate these source specifications into energy release rates and energy spectra for gamma rays and beta particles as a function of time after the accident. Such calculations involve following the build-up and decay of the fission products in the core for some prescribed operating conditions of the reactor. For testing purposes it is highly desirable to minimize the number of sources that need to be simulated. Thus, it is useful to examine the variation in energy release rates and spectra that result from different operating conditions or other choices of parameters not specific in the Guide.

ENERGY RELEASE RATES AND SPECTRA

Energy release rates have been calculated for a wide variety of reactor operating parameters including fuel composition, power level, duration of operation, and treatment of progeny [2]. The method used was the RIBD-II code [3] with the ENDF/B-IV fission product data library [4,5]. However, it has been demonstrated that the choice of code is not significant, whereas the data library is important. The results of these extensive energy release rate calculations are presented in Reference 2. An example of the results is shown in Figure 1. The beta and gamma-ray energy release rates are plotted as a function of time after LOCA for the case of 200-day continuous operation at 4000 MW(t).

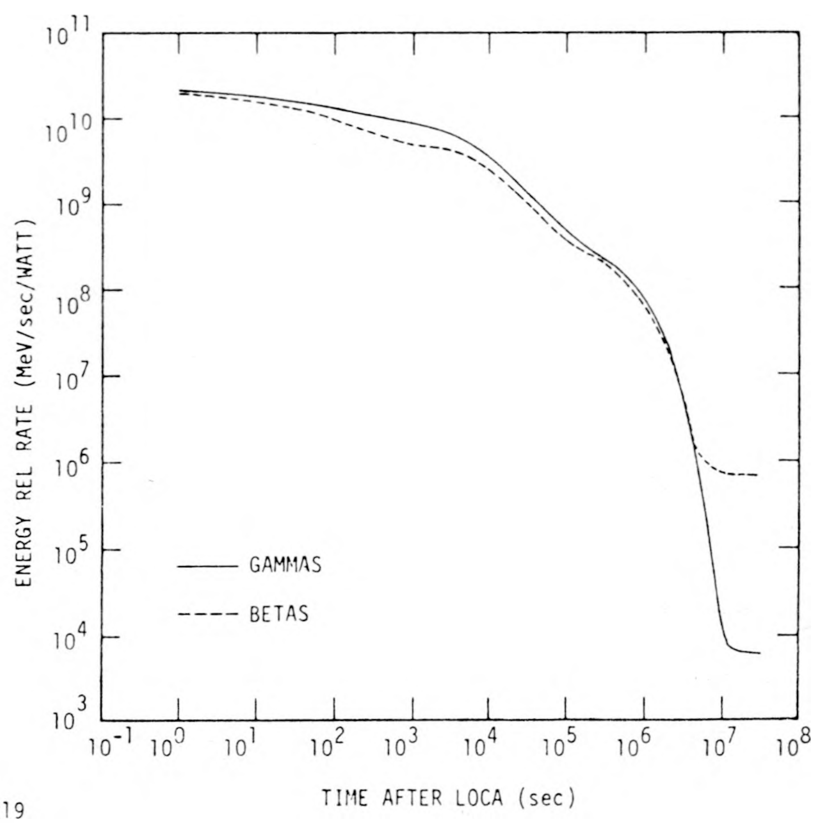

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Figure 1. Energy release rates for gamma rays and betas, source 1 (airborne), 4000 MW and realistic fuel loading.

Several conclusions can be drawn from the energy release rate calculations:

(1) The length of irradiation prior to LOCA (in the range 200 days to equilibrium) does not have a significant influence on the energy release rates until post-accident times greater than about one day; after that, differences can be significant.

(2) The reactor operating power affects the fission product inventory by neutron capture transmutation of the fission products and fuel, and by depletion of the fissile isotope. For the post-accident time range of interest here, and for realistic power levels and irradiation histories, these effects are not significant.

(3) Several methods for treating daughters of the specified fission product types were investigated. Substantial differences in energy release rates can be obtained depending on whether or not daughters are included.

In addition to energy release rates, energy spectra of the fission products were also calculated at selected times after LOCA. The method of calculation involved folding the activities of each of the fission product isotopes at the cooling time of interest with the individual beta or gamma-ray spectrum of that nuclide and summing over all nuclides [2,6]. The accuracy of this method is limited by the library of isotope decay spectra. The library presently includes spectral data for 180 fission product nuclides which account for 90 to 98 percent of the total decay energy for times greater than about one minute after assumed LOCA initiation [7]. At shorter times the library is not sufficently complete to give accurate spectra. The results of the spectra calculations for all sources and distribution categories are given in Reference 2.

An interesting result obtained from the calculations is the behavior of the average particle energy as a function of cooling time. An example of these results (for Source 1) is shown in Figure 2. The spectra "soften" (i.e., average energy decreases) with time, reaching a minimum in the neighborhood of 1 to 10 days, then "harden" at longer times. This behavior occurs for both gamma-ray and beta spectra. This variation in average

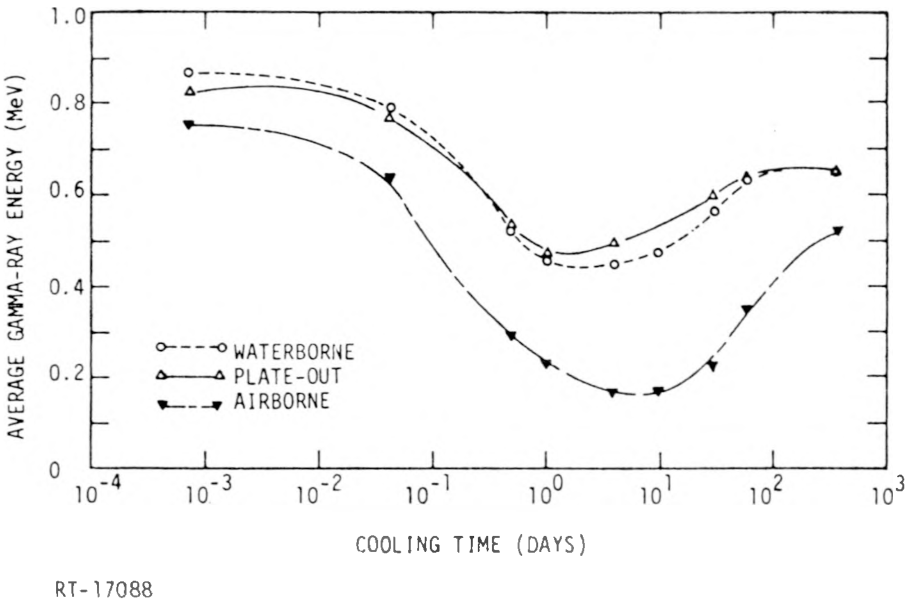

Figure 2. Example of cooling time dependence of the average gamma-ray energy for source 1 for pure ^{235}U at 1 watt; all daughters have been included. Betas show a similar behavior. This quantity is very nearly independent of the length of irradiation.

energy is not unexpected since it is generally true that radioactive emissions from short-lived nuclides are higher in energy than those from long-lived nuclides.

The spectral shape has potential for impact on radiation qualification testing and thus the implication of changing spectrum hardness must be addressed. This is particularly true when radiation simulators which have fixed or monoenergetic spectra, like Cobalt-60, are used in qualification testing.

DEPTH-DOSE STUDIES

A knowledge of the accident signature makes it possible to evaluate the performance of radiation simulators used for qualification testing. An important consideration for judging the adequacy of a radiation simulator is the "damage" mechanism for the equipment or component of interest. In other words, the source used for radiation qualification testing must simulate those damage mechanisms which cause failure in the equipment.

In an effort to evaluate the performance of simulators compared to the LOCA sources, a series of calculations of energy deposition in a modeled electric power cable were carried out [8]. The model used for the calulations, shown in Figure 3, consists of a solid copper conductor surrounded by an elastomer insulator (ethylene-propylene rubber) and jacket (Hypalon). Using a coupled photon-electron transport technique based on Monte Carlo methods [9], the energy deposition as a function of radial depth in the cable was calculated for the LOCA sources and Cobalt-60. In order to establish bounds on the spectra, the depth-dose calculations were limited to the extremes of the hardest spectra (one minute cooling) and the softest spectra (four days). Care was also taken to distribute the source spatially to approximate an extended isotropic distribution for the airborne and waterborne cases.

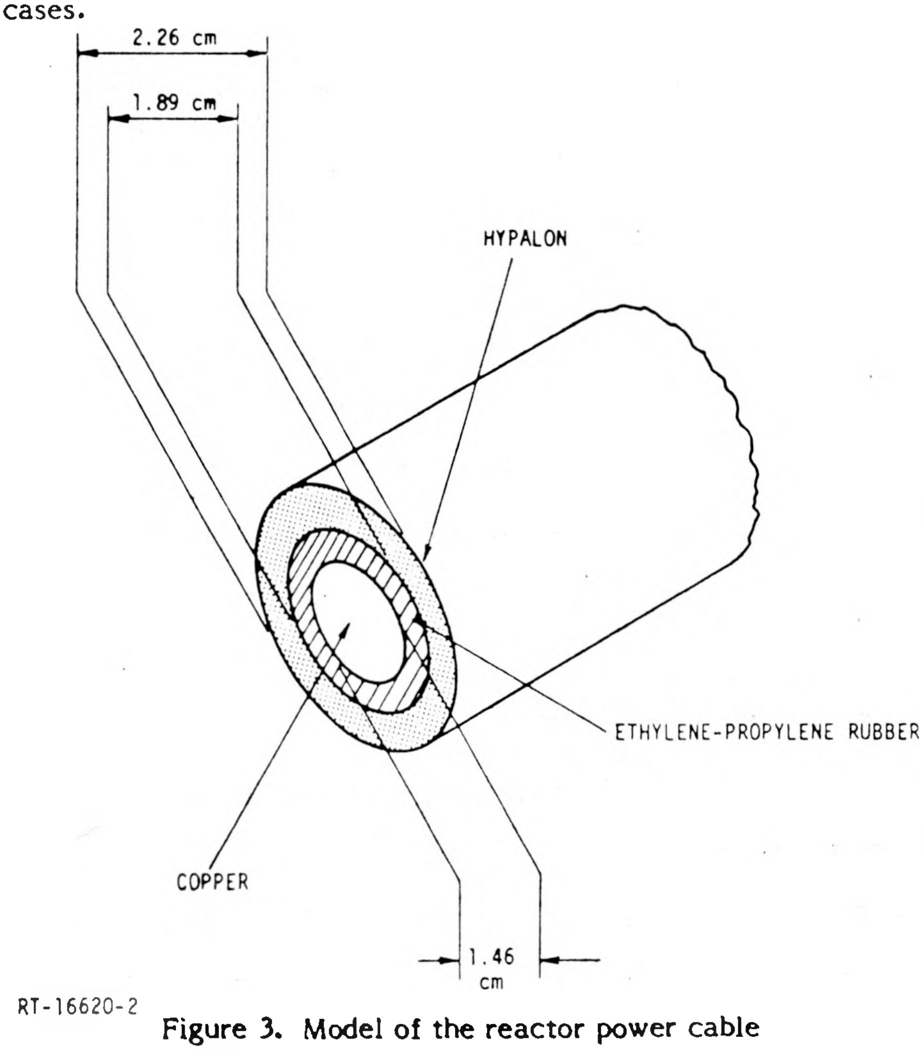

Figure 3. Model of the reactor power cable

An example of the results of the depth-dose calculations is shown in Figure 4 for the case of the LOCA plate-out sources at one minute after LOCA compared to Cobalt-60. The profiles are compared relative to the energy deposition in the outer zone of the jacket. Clearly, for this case, the energy deposition profiles for Cobalt-60 and the LOCA sources are quite different. In particular, the beta calculation shows a much steeper dose gradient. Similar results are found for the other sources investigated. For the case of the softer spectra (four days) the differences between Cobalt-60 and the LOCA profiles are even greater.

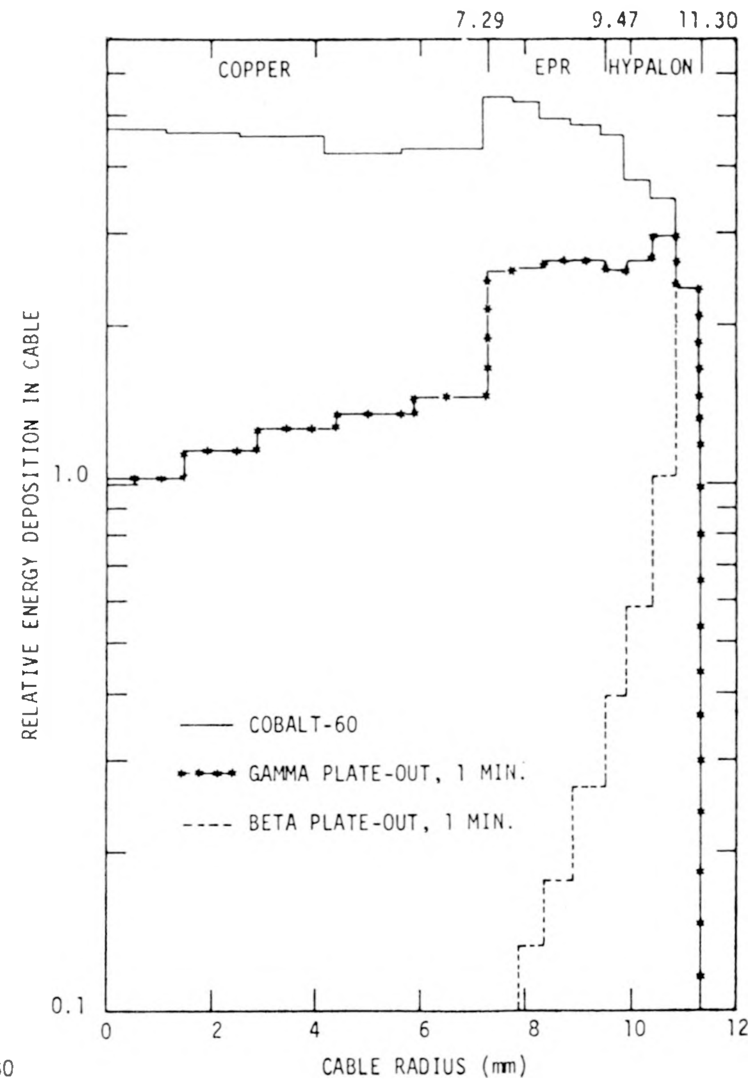

Figure 4. Calculations of relative depth-dose profiles in a typical electrical power cable for Co-60 compared to the beta and gamma-ray plate-out sources at one minute.

DISCUSSION

Energy release rates and spectra have been generated for the regulatory Guide 1.89 source specifications. In doing so the importance of the various parameters relevant to the fission product inventory was investigated. Detailed results of these calculations are given in Reference 2.

The interpretation of the depth-dose calculations in terms of establishing simulator adequacy is not straightforward. Reasonable matching of the dose profile in the cable by the simulator source would be sufficient to guarantee its adequacy. However, a failure to produce equivalent depth versus dose does not necessarily mean that the simulator is inadequate. It is necessary to determine the mechanism of failure of the cable in the LOCA radiation field, and then ascertain whether or not this damage mechanism is equivalently stressed by the simulator. For example, in the case of the cable, if failure is governed by total dose in the insulator, then differences in dose profiles are not significant as long as equivalent total doses are produced. On the other hand, if some other damage mechanism not simply related to total dose governs failure of the cable, then it will be necessary to demonstrate whether or not the simulator produces equivalent damage. Efforts along these lines are currently in progress. Depth-dose calculations of the type reported here provide a basis for evaluating simulator performance, and can be extended to other Class 1E equipment.

13

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A7. THE BEST-ESTIMATE LOCA RADIATION SIGNATURE: WHAT IT MEANS TO EQUIPMENT QUALIFICATION*†

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ABSTRACT

The available fission product release data from damaged reactor fuel elements form the basis for this "best-estimate" LOCA radiation signature. In particular, WASH-1400 provides a composite time-sequence fission-product-release schedule which describes the "actual" post-LOCA events. The resultant radiation signature should be the most appropriate bases for the qualification of Class 1E equipment.

INTRODUCTION

In a companion paper [1] the loss-of-coolant accident (LOCA) radiation source specifications [2] presently required as the basis for radiation qualification of Class 1E equipment were translated into energy release rates and spectra [3] for use in evaluating simulator "adequacy". In that work, the sources specified by the U.S. Nuclear Regulatory Commission in Regulatory Guide 1.89 [2] were accepted without evaluation or justification. However, it is useful to evaluate whether a significantly different source will obtain if the most recent experimental and analytical results for fission product release in a LOCA [4,5] are utilized.

The Regulatory Guide assumptions are questionable in two principal areas. First, the fission product in-containment release magnitudes are generally based on nuclear power plant siting considerations rather than expectable damage of fuel elements

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*This paper documents part of the Qualification Testing Evaluation (QTE) Progra (A1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

†The LOCA signature discussed in this paper is actually for an "unterminated" accident condition. The signature is used as a data base to allow the reader to understand the contribution of each accident phase to the total release signature (note added 20 April 1979).

during a LOCA. Secondly, the Guide assumptions of instantaneous releases and uniform dispersion or deposition result in unrealizably high initial dose rates which must be considered in equipment qualification.

In this paper we define "best-estimate" LOCA radiation signatures based upon realistic estimates of fission product release and time progression of a postulated accident. Energy release rates and spectra are discussed and compared to the Regulatory Guide quantities presented in Reference 1.

ACCIDENT SCENARIO AND TIMING

The postulated accident is defined as an unterminated LOCA that leads to full meltdown and subsequent vaporization of the core. Fission product release is expected to occur more or less continuously until the system cools. During this period, release rates should vary over wide limits depending on the properties of the fission products, system temperatures, and surface-to-volume ratio of the molten material. The time sequence of the accident is adopted from the Reactor Safety Study [4]. Four phases of the accident are defined during which major driving forces for release exist. The release components associated with these phases are: gap release (fission product release which occurs when claddings experience initial rupture, consisting mostly of activity that was released to void spaces within the fuel rods during normal operation); meltdown release (fission product release which occurs from the fuel while it first heats to melting and becomes molten); vaporization release (fission product release which occurs after large amounts of molten core material fall into the reactor cavity from the pressure vessel); and oxidation release (release which occurs just after and as a result of a steam explosion event in which finely divided fuel material is scattered into an oxygen atmosphere, undergoes extensive oxidation and liberates specific fission products).

The four release components are expected to occur sequentially during an unterminated LOCA. The gap release would occur first, followed by the meltdown component, and then by the vaporization component. However, steam explosions could potentially occur any time (if they occur at all) after appreciable amounts of the core have melted. Thus, the oxidation component may be somewhat randomly located in time.

Using Reference 4 as a guide, the release sequence shown in Figure 1 was adopted for the "best-estimate" source. The oxidation release term was conservatively placed at the early end of its range, and assumed to be instantaneous. Each of the other releases follows a time behavior that is conservatively adopted from the expected accident sequence [4,5].

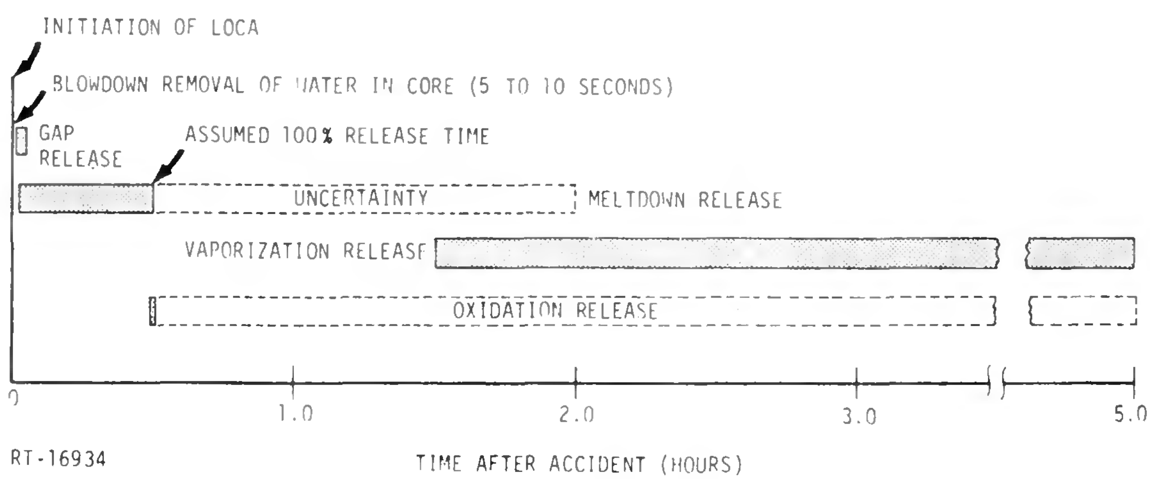

Figure 1. Time sequence of fission product release for an unterminated LOCA without emergency core cooling.

The time dependence of the gap release was taken to follow a linear fuel rod failure rate starting at 30 seconds after LOCA initiation and reaching 100 percent failure at 150 seconds. The meltdown release was also assumed to follow a linear time dependence starting at one minute and reaching 100 percent release at 30 minutes. The latter is a conservative estimate for complete melting of the core; the Reactor Safety Study [4] estimates that it could take up to two hours. About an hour after most of the core is molten, the pressure vessel will fail, giving rise to the vaporization release term. A simple exponential expression for the time dependence of the vaporization release was assumed with a characteristic release half-time of 30 minutes [4].

"BEST-ESTIMATE" SOURCE DEFINITION

The fractions of fission products in the inventory at the time of the postulated accident that are released during each phase of the accident were derived chiefly from the Reactor Safety Study [4]. Work reported subsequent to that study was also reviewed, but although there is some new work in progress [7,8], none of the newer results were judged sufficiently well established to include at this time.

The results of the review are presented in Table 1 showing the contributions of each of the four release components. The numbers given are fractions of the shutdown inventory. A more complete discussion of these results, including their justification and uncertainty is given in Reference 6.

It is believed that these values are still rather conservative. Recent preliminary results of gap release fractions measured at Oak Ridge National Laboratory [7] indicate that the quoted gap release fractions may be an order of magnitude or more too large.

It is interesting to compare the total release fractions (integrated over the accident) for each of the elemental groups of the "best-estimate" source with the two sources specified in Regulatory Guide 1.89 [2]. It should be noted that the best-estimate source calls for time-dependent releases during the evolution of the accident,

Table 1. Best Estimate LOCA Fission Product Release Fractions

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

X = Fraction of core involved in steam explosion

Y = Fraction of inventory remaining for release by oxidation

whereas the Guide sources are assumed to be instantly released at the time of LOCA initiation. Table 2 shows such a comparison. For the Regulatory Guide results the contributions from the three distribution categories have been summed.

The best-estimate source gives a much larger integrated contribution from the alkali metals, tellurium group, noble metals and alkaline earths, but a smaller contribution from the rare earth and refractory species compared to Source 1. Another major difference between the best-estimate and Source 1 is in the elements in the low-yield wings and middle of the fission product mass distribution (labeled "others" in the table). Because of the very small yields of this latter group, they contribute very little to the energy release rate.

Table 2. Comparison of Best-Estimate and RG-1.89 Sources

Fission Product Species	Fractional Release		
		Regulatory Guide	
	Best-Estimate	Source 1	Source 2
Noble Gases (Xe, Kr)	1.0	1.0	0.10[a]
Halogens (I, Br)	1.0	1.0	0.20
Alkali Metals (Cs, Rb)	1.0	0.02	0
Tellurium Group (Te, Se, Sb)	1.0	0.02	0
Noble Metals (Ru, Rh, Pd, Mo, Tc)	0.908	0.02	0
Alkaline Earths (Sr, Ba)	0.109	0.02	0
Rare Earths (Y, La, Ce, Nd, Pr, Eu, Pm, Sm, Np, Pu)	0.013	0.02	0
Refractories (Zr, Nb)	0.003	0.02	0
Others	0	0.02	0

[a]Except ^{85}Kr which is 0.30.

ENERGY RELEASE RATES AND SPECTRA

Beta and gamma-ray energy release rates and spectra corresponding to the "best-estimate" source have been calculated using methods identical to those used for the Regulatory Guide sources [3]. Figures 2 and 3 show examples of gamma-ray and beta energy release rates as a function of time after the LOCA for the case of an equilibrium irradiation (10^{13} seconds) of pure ^{235}U at a power of 1 watt. Each of the individual release components is shown as well as the total. Also shown in Figures 2 and 3 are the corresponding energy release ra'es for the larger of the Regulatory Guide sources (Source 1 - for containment heat removal systems). The latter includes all three distribution categories (airborne, waterborne and plate-out) summed together.

Additional calculations were made to investigate the sensitivity of the results to the values of the release fractions and the assumed time dependence. Upper and lower

14

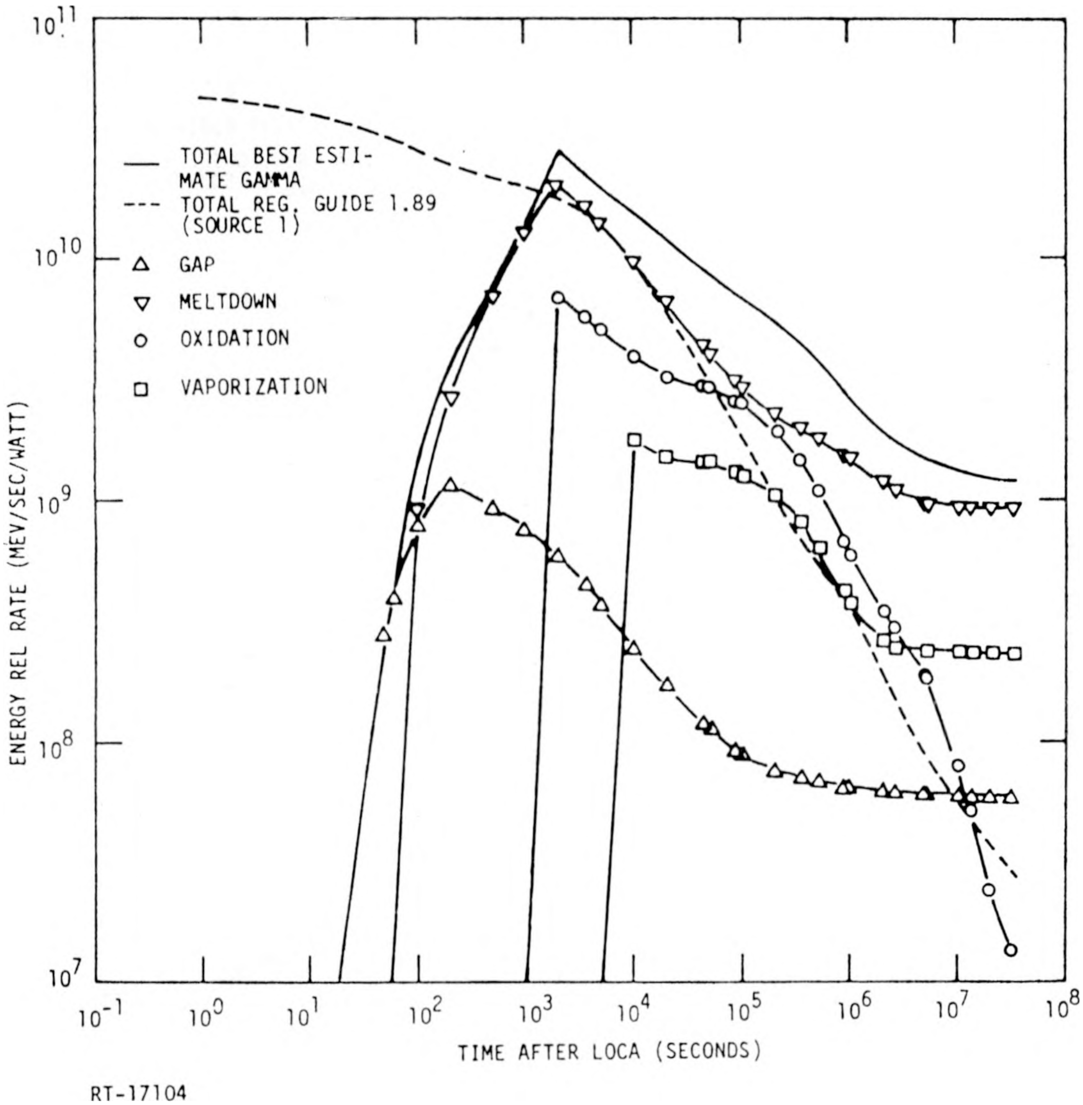

Figure 2. Gamma-ray energy release rate for "best-estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result.

bounds were calculated by using the upper and lower limits of the specified ranges of release fractions and the limits on the times of release. Such variations can lead to an uncertainty band for energy release rates of as much as a factor of 10.

The most significant feature of the best-estimate calculations is the reduced release rate at early times compared to the Regulatory Guide source. This translates to a lower dose rate specification for the equipment and perhaps a lower total dose as well.

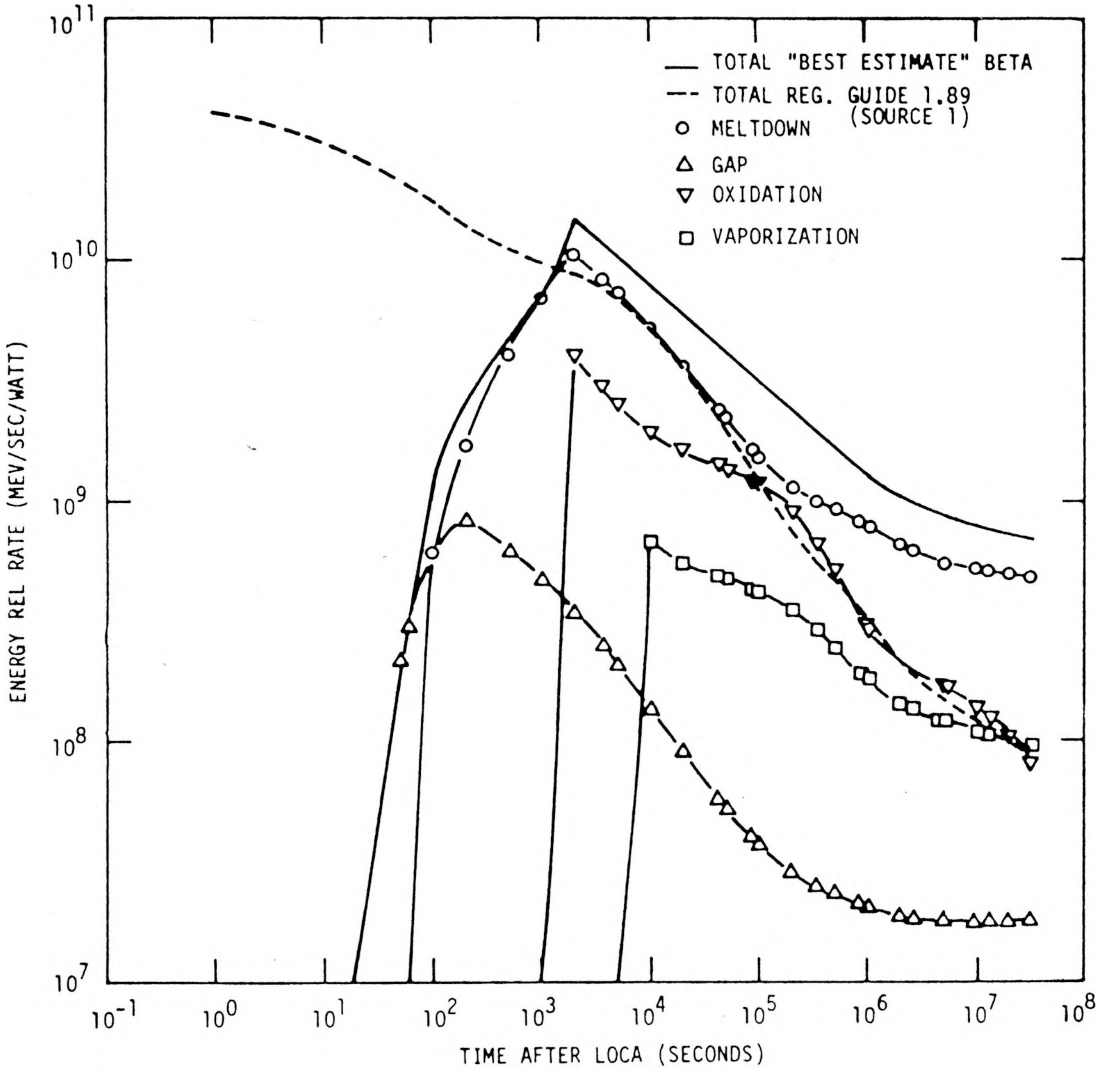

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Figure 3. Beta energy release rate for "best-estimate" source showing the contributions of the constituents, and compared to the total Regulatory Guide 1.89 source result.

Spectra corresponding to the best-estimate sources have also been generated using the previous procedures [3]. There is not much difference in the spectra for the best-estimate sources compared to the Regulatory Guide sources. The former tend to be slightly harder at shorter times, and slightly softer at long cooling times.

DISCUSSION

The best-estimate LOCA radiation signature provides a more realistic basis for radiation qualification of nuclear power plant equipment than the sources presently specified in Regulatory Guide 1.89. The most significant difference being in the early time dependence of the energy release rates. Because of the time dependence of the best-estimate release, it is expected that the dose and dose rate requirements will be reduced compared to the Regulatory Guide requirements.

An additional problem not addressed in this work is the partitioning of the best-estimate source and the resultant local dose to equipment. It is necessary to partition and transport the radiation within the containment to arrive at doses and dose rates suitable for qualification testing.

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A8. QUALIFICATION ISSUES: THE REST OF THE ICEBERG*

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ABSTRACT

Reactor safety has been a primary concern since the beginning of commercial power reactor development. But only since the IEEE became actively involved (circa-1970), through IEEE-323-1971 and its predecessor drafts, has the qualification of selected Class 1 equipment been formalized and intensified. Currently EPRI and the USNRC are funding significant qualification research programs. However, these current research topics address only a few of the issues and there are more to come. A review of recent achievements and a discussion of some issues still to be encountered are presented.

Introduction

Other papers at this session described activities addressing specific issues in Class 1E equipment qualification. While the IEEE standards activities, the EPRI-sponsored Aging Methodology Program (AMP), and the NRC-sponsored Qualification Testing Evaluation (QTE) Program are significant and constitute a formal (albeit, preliminary) recognition of issues in equipment qualification, these programs are not intended to be exhaustive

*This report documents part of the Qualification Testing Evaluation (QTE) Program (A1051-8) being conducted by Sandia Laboratories for the United States Nuclear Regulatory Commission under DOE Contract AT(29-1)-789.

in their scope. In reality, qualification and qualification issues will remain a part of a progressive commercial nuclear power industry. As equipment, fabrication methods, designs, data bases, and accident analysis techniques evolve, so must qualification testing methodology.

The current programs suggest that there are additional unidentified qualification issues beneath the surface like the symbolic iceberg. Many of these issues may not be "problems" at all and may disappear when a reasoned analysis and logic are applied to them. In this paper we try to anticipate some of those issues, while realizing that others may not yet be obvious because their technology base currently does not exist. But by their early discussion it may be possible to distinguish between real icebergs and mirages.

Why Problems

The difficulties encountered in the qualification of safety-related equipment are a consequence of a number of factors including: (1) the fact that the development of standards lags behind the need for them; (2) the inherent limitations of state-of-the-art testing; (3) the time required to conduct research and acquire experience necessary to provide the bases for qualification tests; (4) the existing variety of, and newly evolving, equipment that requires qualification; (5) the fact that significant technological changes and requirements can occur over times that are short compared to the projected life of nuclear plants; and (6) the fact that regulatory agencies necessarily specify their requirements in terms of general performance standards which must then be specifically interpreted for each equipment type, equipment function, and power plant.

The early history of nuclear power standards development can be best characterized as "responding" to, rather than as "anticipating" problems. Both regulators and industry were limited to "responding" because it was a developing technology and a rapidly developing industry. Without time to develop a data base specific to a problem, the regulatory agencies, naturally and correctly, used, adopted, and modified the existing standards, guides, and regulations. Some of these were too general or only marginally applicable, compelling the applicability to be determined on a case-by-case basis. The nuclear industry, too, has promulgated certain standards which were sometimes detrimental to the users as a whole. Two examples are:

1. The tendency toward consensus standards and the adopting of historical problem solutions, at the expense (perhaps) of the more normal evolution consisting of precise problem statement followed by state-of-the-art problem resolution.
2. The too rapid extension of concept standardization (a prime example may be accelerated aging) beyond the supportive data base.

The regulated and the regulator alike have no other choice but to remain within the state-of-the-art to allow orderly decision-making and to assure a viable industry. Expanding testing capability and understanding can be tedious and expensive and lack immediate payoff. At the same time, it must also be recognized that the state-of-the-art must not stagnate into an unacceptable status quo and that money and time must be devoted to extending the state-of-the-art.

The number of equipment types is substantial and this presents a qualification problem of some magnitude. Typically, 30-50 generic Class 1E components may exist for any nuclear plant with 1 to 8 suppliers of each, and several types from each supplier. Clearly, typetesting this number of components is an immense task and, in addition, "improved" and changing products are continually being made in any viable industry. This strains the economic capability of industry to fully qualify each item for use.

Significant changes in qualification technology requirements are occurring. Between 1967 and 1974, IEEE-279-1968 and IEEE-323-1971 were utilized as qualification requirements and bases at the construction permit (CP) stage. 34 of the plants in construction on the bases of these standards have not yet reached the operating license stage; they range from 0% - 99.5% complete, with 14 of the plants less than 50% complete [1]. Their qualification programs are based on IEEE-323-1971 [2], even though IEEE-323-1974 [3] is currently recognized as the Class 1E qualification standard. Environmental qualification of safety-related electrical equipment for plants granted CPs after July 1974, reflect the more comprehensive guidelines specified in IEEE-323-1974; these plants range from 0% to 57% complete, with most less than 25% complete. The commitment by industry to IEEE-323-1974 is causing concern and problems as the industry trys to meet the changing and more stringent requirements [4]. But even as the licensees work with IEEE-323-1974, a new IEEE project, P627 [5], is being prepared which is likely to supersede it. Even though its foreward suggests that P627 presents no new requirements beyond IEEE-323-1974, it is another in a series of standards and actions which must be accommodated.

While the concept of nuclear plant standardization would allow uniformity in many aspects of plant design, and specifically in equipment qualification, its promise has not reached fruition. Because many qualification requirements are specified in terms of quasi-legal general performance standards rather than specifics, they are subject to individual plant interpretation and case-by-case review and approval. This compromises the simplification and time-savings implied by plant standardization. In addition, changes in qualification requirements means that the "standardized" equipment qualification may have to be repeated with each new issue of the standards and thus defeat the goal of standardization.

Programs and Achievements

To provide a common understanding of the scope of the existing programs, it is useful to review them briefly; the table summarizes the activities of the EPRI and NRC technology-advancement programs.

EPRI-AMP Program	
Aging Methodology	- Define accelerated aging programs for selected Class 1E equipment consistent with the state-of-the-art.
NRC-QTE Program	
Accident Methodology	- Provide assessments of post-LOCA qualification testing methodologies and their influence on Class 1E equipment degradation, including an evaluation of synergistic effects from combined environmental testing.
Radiation Source	- Determine the radiation signature from the nuclear source term for a design basis LOCA and evaluate the adequacy of radiation simulators.
Accelerated Aging	- Provide a method, and method verification, that can be used to simulate the natural aging process of Class 1E equipment (materials) by accelerated aging methods.

These are ambitious programs with potential for eliminating some of the current questions in qualification, but the results to date are only sufficient to indicate that potential. The aging methodology (EPRI) program will be effectively limited by the state-of-the-art, consistent with its stated objectives. The QTE program has produced significant achievements which represent firm bases for proceeding to new qualification issues. The LOCA methodology task has concentrated on synergistic effects in relatively "simple" systems, i.e., cables, connectors, and splices; the extension to more complex systems, such as transmitters, valve operators, and penetrations, remains to be completed. The radiation source and simulator adequacy task is in its early stages of completion; the common simulators do not appear to "adequately" simulate all aspects of the exposures, but it may not be necessary to do so and still achieve the required equipment degradation. A final conclusion is not yet available. The accelerated aging task has been concentrated on electrical cable materials using elongation as a measure of degradation and has progressed well. However, accelerated aging of active equipment, as a prerequisite to LOCA and seismic testing, is expected to be more complex and remains to be addressed. Specific aspects of the QTE program are briefly discussed below.

The LOCA - accident methodology experiments have been definitive for cables and cable splice assemblies [6]. No significant functional, or even material, synergisms appear to exist for these very important Class 1E equipment items. But the singularly most important aspect of the task efforts has been the increased emphasis on equipment qualification and quality in general which has resulted from the "failures" of cable connector assemblies [7]. When connector assemblies were observed to fail in the tests, it was clear that the results of these methodology tests were being inappropriately interpreted as pass/fail indicators of existing in-plant equipment. This was not the correct interpretation; rather, the test results were indicative of a potential generic problem. After a somewhat labored initiation, recent equipment evaluations by NRC have been made in an attempt to discover other potentially generic problems. The in-progress SEP (Systematic Evaluation Program) effort exemplifies this recent tack [8]. An objective of the SEP is to evaluate the degree to which the mechanical and Class 1E electrical equipment of safety-related systems have been qualified for the environments associated with design basis events. As such, the SEP will be directed toward the determination of existing safety margins and the evaluation of the adequacy of such safety margins to determine if any backfitting or facility upgrading is necessary.

The radiation source task has been concentrated on source signature definition. The earliest results [9] were significant in three areas: (1) a definition of the LOCA radiation sources energy release rate and spectra as a function of time following the hypothesized releases, with both the gamma and beta spectra exhibiting a changing energy dependence with time; (2) a (formal) definition of expectable magnitudes of the gamma and beta dose and dose rate for a typical containment, using the applicable Regulatory Guides; and (3) a (formal) recognition that the beta dose and rate are significantly greater than the corresponding gamma values. More recently, the source signature study was expanded [10,11]. That work is significant as the most detailed definition of the hypothesized LOCA-radiation signature to date. The parametric evaluation of signature sensitivity indicates some parameters which are individually significant, but relatively little effect from varying these and other parameters over conservative or expectable ranges. The work suggested the following:

(1) The range of parameters influencing the radiation signature suggests the need for greater precision in their specification;

(2) The greatest signature-influencing factor appears to be the original nuclide fractionation specifications and it would be useful to support these with experimental data bases;

(3) It may be prudent to adopt a uniform source signature (from the work), as a replacement for nuclide fractionation specifications;

(4) Early assessment of beta radiation significance is important in evaluating radiation qualification programs;

(5) With a generic data base established, the plant-specific dose and dose rate values can be accurately established within the constraints of plant-layout assumptions;

(6) The evaluation of simulator "adequacy" can proceed from this generic base;

(7) An even more realistic representation of the LOCA radiation source could be constructed based on a prescribed accident sequencing as suggested in WASH-1400 [12] and supported by ongoing experimental programs.

The accelerated aging task results have been particularly encouraging. The proposed combined environments model has been verified for several cable insulation and jacketing materials [13]. The data resulting from analyses of combined environments dose rate and humidity experiments on electric cable materials alone is extremely significant [14]. Another major aspect of the task has been the evaluation of damage in naturally-aged polyethylene/ PVC cable; that work demonstrates the general utility of the test facilities

16

and the aging model. It is an excellent example of the program's ability to address timely questions in equipment qualification.

More Problems, Reality and Illusion

Even in these established programs, not all results are final and work continues. But beyond these immediate programs and their goals, other qualification issues can be identified. To stimulate discussion, the table below lists several areas of inquiry that might be considered by the nuclear industry. The act of making and presenting a list of such issues should not be viewed in the analogy of opening Pandora's box; in most cases no "problem" will remain when a reasoned analysis is applied. The distinction between reality and illusion must be made. It is always possible to require unrealistic qualification based on poor or imprecise assumptions and data bases. By concentrating on establishing realistic accident scenarios, realistic qualification programs can result and the number of real problems can be minimized.

The table is divided into technical and administrative issues. While the distinction may be subtle and disputable for certain of the table entries, the concept behind this separation is sound and deserves elaboration. The technical issues can be resolved by straightforward application of the scientific method; the problem statement can be made precise and its resolution can proceed. The administrative issues are of two types. The first involves the further clarification of an issue so that it can become a technically tractable issue. Secondly, some questions require permanent administrative attention and resolution by their inherent nature. Some of the table entries deserve additional brief comment.

Qualification Issues

Administrative	Technical
- Accident definition	- Accelerated aging
- Backfitting	- Failure experience data
- Failure criteria	- Interfaces between components
- Interpretation of standards	- Radiation simulator(s) adequacy
- Maintenance of Class 1E status during design life	- Seismic qualification
- Margin and envelopes	- Standardized test procedures
- Specific or envelope qualification	- Statistics
- Superheated steam (MSLB)	- Synergistic effects (LOCA)
- Surveillance methods for early detection of problems	- Test method influences

Accurate accident definition is prerequisite to specifying qualification programs. Often, accident definition is subordinated by "regulatory acceptability;" instead of plant specific definitions, the utility or architect-engineer may accept environment calculations developed for previously licensed plants to avoid licensing delays. These are generally conservative. The qualification problem is then compounded further with additional conservatism, enveloping all extremes of environmental conditions, and margin. The result can be a qualification program unnecessarily requiring safety-related equipment that is beyond the state-of-the-art. (Kasturi, et al., alluded to this problem in a previous paper in this session.) But, at the same time, accidents should be carefully defined initially to avoid any (potential) requirement for plant backfitting or equipment requalifi cation; the "discovery" of the potential for superheated steam from main-steam line breaks (MSLB) in 1976 is a case in point.

Interfaces between Class 1E equipment are themselves Class 1E. In some cases, this has not been recognized in the plant design and in many cases it has not been recognized in qualification testing. The recent controversy over connector qualification [7] emphasizes this point. It seems clear that interfaces have the highest potential for being the weak link in a system of individually qualified components and therefore must be considered in the test program and in the selection of components for test. Junction-box/cable and connector/cable interfaces are excellent examples which also lead to questions of generic qualification. For example, does a qualification of a specific connector and cable assembly suffice for any combination of that connector type with any other cable?

Statistics in qualification testing is generally not addressed. The concept of typetesting, in fact, does not include consideration of statistical confidence. The emphasis in the applicable standards [3, 5] is toward design and type tests. Qualification of a single specimen is accepted for the qualification of an equipment type, within the bounds of quality assurance and quality control. Margin is the tool used ". . . to account for normal variations in commercial production of equipment and reasonable errors in defining satisfactory performance" [1]. In most cases, testing the number of specimens necessary to achieve a high statistical confidence involves prohibitive costs. Perhaps alternate concepts can be discovered; for example, it may be possible to mathematically relate overtest results to statistical confidence at the lower, nominal, test level.

Superheated steam may represent an imagined problem in two respects. First, to achieve a superheated-steam accident situation may be unrealistic and only be a function of many pyramided conservative assumptions. But even if "real", it may be unnecessary to duplicate this short-lived phenomenon in qualification testing for all equipment. It may only represent an additional thermal "spike" which may be simulated by an equivalent thermal exposure produced with saturated steam. It is premature to require superheated-steam tests for the sake of "realism" without first verifying that the mechanism for damage is directly related to the superheated steam and not just to temperature effects.

Discussion and Summary

Active programs are underway to address specific Class 1E equipment qualification questions. These programs and their recent achievements are very encouraging, but they are only the initial thrust of exploratory programs into unknown areas.

In outlining some other issues, we have attempted to promote early consideration of these by the commercial nuclear power industry. The outline is certain to be incomplete--there will be more yet to be identified.

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