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Proceedings of the Workshop on Nuclear-Power-Plant Aging

Bethesda, MD
August 4-5, 1982

Compiled by: Benjamin E. Bader, Lewis A. Hanchey

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PROCEEDINGS OF THE WORKSHOP ON NUCLEAR-POWER-PLANT AGING

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Sponsored by
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Compiled by
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PREFACE

A Workshop on Nuclear Power Plant Aging was held on August 4-5, 1982, in Bethesda, Maryland, sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, and hosted by Sandia National Laboratories, Albuquerque, New Mexico.

The proceedings were divided into four Technical Sessions. Twelve papers were presented in Technical Sessions I and II. The large majority have been reproduced exactly as they were received. A few have been lightly edited and retyped. Technical Session III consisted of oral presentations. As indicated, four presentations have been abstracted from the transcript of the proceedings and in two cases abstracts, as submitted by the authors, are included. A summary of Technical Session IV (Panel Discussion) is presented, also based on the transcript.

All Technical Sessions were chaired by Satish K. Aggarwal, NRC Program Manager.

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SUMMARY

The Nuclear Regulatory Commission (NRC) organized the workshop on Nuclear Power Plant Aging held August 4-5, 1982 in Bethesda, Maryland, to facilitate an exchange of thoughts between the NRC and industry on the subject of time-related degradation and its influence on reactor safety. The workshop objectives were to define the problem, to discuss the state of knowledge on aging phenomena, and to identify future activities necessary to understand the problem. The theme of the workshop was "Where we are today and what should we be doing."

The workshop covered a wide range of topics with much useful information and many interesting thoughts and recommendations being presented. Some of the salient points and recommendations that were made are summarized to provide the reader a perspective on the workshop.

There is a need for a comprehensive program to identify the potential safety problems associated with plant aging. The effects of time-related degradation on the safety of the complete reactor system should be evaluated in terms of the risk to the public. Such a system analysis should narrow the problem to manageable proportions. It should also encompass the multiple causes that have typically been associated with abnormal occurrences. However, because it is individual component failures that create problems, time-related degradation will ultimately have to be addressed in terms of maintenance, monitoring, surveillance, etc., of components.

Numerous speakers expressed the opinion that vast quantities of data exist but that they are not currently in a form useable for application to time-related degradation of reactors. Among potential data sources most frequently mentioned were the licensee event reports (LERs), the military, the National Aeronautics and Space Administration (NASA), and the nuclear industry itself.

The complexity of analyzing time-related degradation was repeatedly emphasized. Among the items noted were the large number of phenomena that can cause failures; detailed lists of parts/materials, including lubricants and other additives, that must be considered; seemingly minor changes in the chemical constituents of a material or in the manufacturing process that can cause significant effects; and changes in the system during operation, such as in water chemistry, that must be understood. However, it was also noted that the same materials and parts are used in many different locations.

Replacement parts were noted as a potential source of problems. The effects of storage on parts and the possibility that new parts may be different from the original ones were mentioned.

There were extensive discussions on the problems with accelerated aging tests. Numerous examples were presented to illustrate the limitations of Arrhenius testing. However, proponents countered that no better technique was available and that with good engineering judgment, the technique was very useful. The need to account for all stress factors, not just thermal, in designing accelerated aging tests

was emphasized, as was the importance of how environments are combined or sequenced. Another related concern was how to correlate aging testing with component ability to function in a design basis event.

The use of naturally aged equipment for test purposes was suggested. Sacrificial replacement of equipment was identified as a source for such equipment. It was also suggested that as much data as possible be obtained from equipment that has failed; however, it was cautioned that in failed equipment, evidence of degradation has often been destroyed.

Maintenance and surveillance in plants and their relationship to time-related degradation were extensively discussed. The use of surveillance in detecting degradation has significant potential; however, it was noted that knowledge of degradation mechanisms and benchmarks of what constitute degradation of concern are needed before an intelligent program of maintenance and surveillance can be defined. Equipment qualification data was suggested as an aid in establishing maintenance intervals. The importance of preventive maintenance was stressed, but caution was voiced that care must be exercised not to damage or degrade equipment during maintenance operations.

Note was taken of industry's understandable concern over the economic ramification of how the problem of time-related degradation is handled.

One question that was raised but unresolved was how to obtain and handle proprietary information that may be needed for evaluating time-related degradation.

Suggested approaches for addressing the problem of time-related degradation included the development of a matrix of aging mechanisms and damage potentials and the involvement of a team representing the various scientific disciplines associated with the degradation process.

INTRODUCTION

S. K. Aggarwal, Program Manager

Good morning. My name is Satish Aggarwal. I am with the United States Nuclear Regulatory Commission.

I am pleased to see all of you here on this bright, sunny morning to participate in the Workshop On Nuclear Power Plant Aging. I am sure that the 2-day workshop will be of great interest to you and help us at the NRC in our research efforts.

I must point out that we are not here to discuss or impose regulatory requirements on nuclear power plants. We are, rather, seeking your active participation in the exchange of technical thoughts between the NRC and the industry experts. The theme of this workshop is "where we are today and what should we be doing."

With this brief introduction, I am pleased to present Bob Minogue, director of the Office of Nuclear Regulatory Research, who will give the welcoming address. Thank you.

WELCOMING ADDRESS

R. B. Minogue, Director
Office of Nuclear Regulatory Research

Good morning. I would like to welcome you on behalf of the Nuclear Regulatory Commission to the Workshop On Nuclear Power Plant Aging. If there is anybody in the room who came in thinking this was a meeting of the Gray Panthers, I am afraid that is down the street.

I am the director of research, and with me as your keynote speaker for the NRC this morning is Harold Denton, who is the director of Nuclear Reactor Regulation. I would like to make a few general remarks of my own, though, before turning the floor over to Harold.

We are in a situation now where the nature of the nuclear regulatory program is changing. As more and more plants are coming on line, as the plants that are already on line are getting older, this raises a whole new set of issues. In the guidance that the Commission has given to us in the Research Office, they have asked us to stress the research issues and safety issues that arise from operating plants.

In an effort to lay out a program for this, we produce each year a long-range plan. The first was last year. I am not suggesting this has been going on for many years. Last year, in commenting on the first draft of the long-range plan, Harold Denton made the following comments. He called for (and I am quoting) "a comprehensive program to identify the potential safety problems associated with plant aging, such as electrical equipment, seals, mechanical equipment, degradation, and material brittleness."

In commenting again this year on this year's plan, he called again for research into aging of plants' structures, systems, and components, including material degradation, valve behavior, flaw detection, maintenance, and in-service inspection.

So this appeal has been reiterated many times. As I see it, there are some key elements to this problem, and I would like to identify them briefly.

First is the very large number of plants that are coming on line all in a rush, so to speak, in the current year and the next few years. The second is the difficulty in applying the experience base obtained from some of the earlier plants both because of increases in plants' complexity and increases in plant size, so that the basis of experience and knowledge that would enable one to predict the aging problem is somewhat limited.

The last is a great breadth of a problem. Many years ago, people recognized some of the issues that were associated with primary coolant pressure boundary integrity. Because of this, there was a lot of work done on neutron embrittlement of pressure vessels, work done looking at specific erosion mechanisms that might lead to intergranular corrosion, work on steam generator problems, and work on piping fatigue.

Of course, all these remain problems today. I am not suggesting that the work resolved all the issues, but at least these were areas that were worked on. What was missing was the broader perspective, the recognition that aging of plants involves many questions of operability of components and the integrity of structures and the operability of electrical control systems.

It was this broader perspective that I am pleased to see is recognized by the program at this meeting.

Let me not ramble on much longer. Harold Denton is the principal speaker, and my main duty this morning was to introduce him. The fact that he is here I think shows the importance that the regulatory staff attaches to this problem. And I think the very large turnout from the industry people reflects that you also attach a great deal of importance to this problem.

I wish you a great deal of success during your endeavors during this meeting. I think this could be a very important step to defining the nature of any research that may be required so that some of the issues may be resolved in technical arenas and not in the context of the licensing questions.

With this preamble, I will turn the podium over to Harold Denton, who is the director of Nuclear Reactor Regulation.

KEYNOTE ADDRESS

H. R. Denton, Director
Office of Nuclear Reactor Regulation

There is a sea change going on among regulators about what we can be concerned about. For 25 years, we have been concerned mainly about design aspects. What we have now come to realize is that most of the reactors that will be in operation during this century are already designed, and the main task ahead is to maintain and operate those plants safely. That requires a different approach than we have used in the past. These plants exist. We have licensed this year Susquehanna Unit 1 and San Onofre Unit 2 for low-power operation, and there are several dozen plants through the next few years that expect to receive licenses.

What we are learning from the performance of these plants is that they do not always behave the way we had expected. I am sure you are all aware that the focus in the early days of the NRC and in the late days of the AEC was on the large-break loss-of-coolant accidents. Today we do not perceive that as a real threat to safety. More likely are small-break loss-of-coolant accidents coming from pump seal failures or steam-generator tube failures, those types of failures.

So I really welcome the turnout today and the opportunity to address you on the importance of aging. Aging of a plant also has a very profound impact on the economics of nuclear power generation. Back in the days when plants were running at a capacity factor of about 60 percent, which was a lot less than utilities had hoped for, performance of the equipment and the people procedures were not what had been expected for a variety of reasons.

So I think there is an economic incentive for the utilities to try to improve the maintainability and operability of the plants. There is a correlation between the reliability of the plant and safety. The fewer the challenges to the safety systems, the safer the plants.

I think that the need for an integrated approach to plant aging is shown by two areas that have come up fairly recently and to which we have given a lot of attention. One, of course, is pressurized thermal shock. If you look into that area, you will find research has been done in almost every aspect of the issue, but there has been little effort to put together the entire story. Neutron embrittlement is an old story in this business. A lot of studies have been done. Fracture toughness is an evolving field. Also involved is analysis of transients and how cold the water might get during various events. Operator behavior during these times has been looked at. However, there has been very little effort to try to put together what all of these various scientific disciplines meant.

If I find one thing missing in our approach to safety analysis today, it is the failure to take the broad look at how the plant systems perform as opposed to taking a narrower look. So I think many of the problems that are related to aging are cutting across the disciplinary lines. They will not just be mechanical or electrical problems. Almost every abnormal occurrence we have had in the last couple of years has had multiple causes.

I am reminded of a story of why this fellow died in a car accident. You will find that he had a fight with his wife. As a result, he went out to a bar, had a few drinks, (a few too many), and got into his car. It was raining and snowing. He was driving at an excessive rate of speed. He had a bald tire. As he went around the curve, he turned over. What was the real cause of that accident? You will find there were multiple causes and each one contributed a narrow part.

I think that is the approach you have got to take when you look at this question of aging. You should not just focus on the tire or just on the radius of the curvature of the road or just on the operator, but you have got to look at the whole system together and what might be going on that would affect the operability of that system.

One area that is of current interest is the performance of auxiliary feedwater systems. This is a very important system in doing PRA studies. We have in our schedule review plan, a requirement that these systems have a failure rate of 10^{-4} or 10^{-5} per demand. The actual experience during the past decade shows that they are not achieving much better than 10^{-3}. It is not basically the design of the system, but it is common-mode failure such as maintenance, operations, tightening bolts too tight on all the valves. It is those kinds of things that occur out in the real world and are not anticipated back in "beautiful downtown design headquarters."

Another issue that is getting a lot of attention today is steam generator performance. That is another one that cuts across a lot of fields. It has got chemistry issues in it, it has got metals issues, it has got operations issues. I think a few years back no one would have believed that over three-quarters of all steam generators in the U.S. today would have extensive denting problems. The industry, through some of its organizations, has attacked that problem and come up with a cure. But just to show what we think would be the cure, or just to show you the magnitude of the problem, it would probably involve putting titanium tubing in the condenser, installing full-flow demineralizers and aerators in the system, and a willingness to shut down and stop any leaks the moment they occur. The whole package might be on the order of $50 million up-front cost to avoid steam generator denting problems. But what is avoided by doing that is a $300 million repair, as a couple of utilities have had to do when the time came to replace the steam generators.

Looking at these problems, I think this workshop is going in the right direction and asking the right questions. I would hope that in getting together with the type of agenda you have, you will identify some of these problems before they occur and enable us to act on them.

There are about 75 operating plants in the U.S. today. We may have as many as 100 by 1990. The majority of these plants, the majority of the big plants, have only been operating about a decade. Thus, the oldest of any real plant of interest today in terms of large size is about 12 years. The challenge will be to try to take what little data we have in the smaller plants and project that to the problems of large plants that will be in operation by 1990.

You might think that some of the first plants going into operation could be lead plants; for example, the LaSalle plant out in Illinois that is under consideration for full power will be the first GE plant licensed since TMI. San Onofre Unit 2 would be the first Combustion Engineering plant licensed since TMI. I think, to some extent, we may be able to look toward plants such as these as lead plants and learn what is going on there and project it to other plants in the same class.

One point that has already been made this morning is that aging does not apply just to physical wearout or just to physical behavior of metals. I think that for a long time in the business, metals were the major concern, and we did not really take a broad look at electrical systems, cables, and valves. So I think it is very important to consider aging broadly and not look at it from a narrow standpoint.

Some of the other issues that are peripheral but probably will be very important will be the availability of replacement parts. Many of the vendors for certain types of equipment probably will not stay in business over the long run if there are no new plant orders. So what about replacing those parts with ones that are not identical; how will this affect the performance of the plant?

There are also other subtle changes going on. What is going to happen to the reliability of the national electric grid over the years? We assume a certain reliability of protecting from loss of off-site power. What will that do? I mention these just as two examples to show that it is not just a design problem.

Bob mentioned the Gray Panthers. This reminded me of a story that maybe you have heard of the two couples who were getting on in years and they decided to retire to a retirement village. The wives were getting bored, and one wife said, "What can we do to liven up this place a bit?" The other wife said, "Why don't we take off our clothes and streak past our husbands?" So they did. And the two fellows were sitting there rocking away. One turned to the other and said, "Wasn't that our wives that went by?" And the other one said, "Yeah. What were they wearing?" "Well, I don't know, but it sure needed pressing."

I think the program fairly outlines all the questions that you need to answer. I think it is an area that you will find is increasingly interesting in the future. Some of the questions that need to be asked probably are ignored today. There are known unknowns and there are unknown unknowns. Today, certainly, we will be able to address some of the known unknowns, and I hope we will hit some of the unknown unknowns that lie ahead. People need to look at the decade or so of operating experience that we have had to see why systems are not performing as well as they should be from a safety standpoint. And I think we can do this and make a major contribution to the plants in the years ahead. Thank you very much.

TECHNICAL SESSION I

THE EFFECT OF AGING ON THE PERFORMANCE OF SAFETY-RELATED EQUIPMENT

James F. Gleason, P.E.
Wyle Laboratories

ABSTRACT

Over the last decade, a concentrated effort has been made to address aging of safety-related equipment in nuclear power plants. Wyle Laboratories has performed research, testing, and sponsored two university studies on the effects of aging on materials and methods to accelerate aging phenomena. Subsequently, hundreds of programs have been performed that included naturally and artifically aged components. In addition, research performed for EPRI has investigated the area of aging/seismic correlation.

Significant results of these efforts have provided the following conclusions:

1. The performance of the majority of safety-related equipment is not significantly degraded by aging.
2. There are some materials utilized in safety-related equipment that deteriorate over time, which causes performance changes.
3. Surveillance activities are modified due to test occurrences.

Research is continuing into the correlation between aging and seismic performance with emphasis on components with potentially age-sensitive materials. Additional research is necessary in the areas of correlation between aging and design basis accidents. The results of an aging program need to be accounted for in the plant surveillance and maintenance activities.

The nuclear power industry has made a concentrated effort to include in the qualification of safety-related equipment the concept of aging. This paper discusses the general philosophy of the inclusion of aging concepts, the application of those concepts into equipment qualification test efforts, and the results of aging on the operability of this equipment.

Safety-related equipment performance is assured via equipment qualification. The industry standard for qualification is IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." The intent of qualification is to demonstrate that safety-related equipment will perform its safety-related function during the design basis event while it is in its worst state of deterioration. Deterioration is postulated to be caused by the adverse effects of all environmental influences. The term "design basis event" refers to a postulated event that could induce common-mode failure. Examples of design basis events are earthquakes and accidents, such as a loss-of-coolant accident. The worst state of deterioration is determined by an analysis of the effects of the environment on the construction materials of the equipment. These environmental deterioration effects are typically referred to as aging effects.

In order for equipment to become qualified, it is necessary that the effect of aging degradation on the safety-related function be considered. This consideration is typically made by reviewing the expected environmental conditions to which the equipment will be exposed during its life in the plant and by performing an analysis to evaluate the potential degradation factors affecting the materials of construction of the equipment. The results of this analysis identify significant aging mechanisms. A representative sample of the safety-related equipment is then placed in an aged condition, naturally or artificially, to simulate all significant aging effects. The applicable design basis events are then performed on the aged sample. Proper operation during the design basis event simulation therefore results in certification of qualification of the equipment.

The knowledge of aging effects has increased steadily for many years. The current state of the art is based on extensive literature reviews, university studies, nuclear power industry studies, and the results of equipment performance during equipment qualification tests. Literature on aging effects has been located in a wide variety of sources. It is typical for many manufacturers to perform research and publish the results in various trade magazines or journals. It is also typical that in order to obtain industrial safety recognition, many materials and components are evaluated for age sensitivity by various industry test programs. The results therefore are documented in the safety certification reports. The U.S. Military and the National Aeronautics and Space Administration (NASA) regularly conduct studies from which valuable information has been obtained. The results of many of these studies can be located and are available from such sources as the National Technical Information Service (NTIS). In addition, university studies continue to be performed on certain aspects of aging. Wyle Laboratories has sponsored two such studies on artificially accelerated aging effects. Additionally, relevant programs are performed by the nuclear power industry itself. Research continues to be performed under NRC sponsorship at many national laboratories. In particular, Sandia National Laboratories has provided much information that has direct bearing on aging effects, synergisms, and accident simulation techniques. The Electric Power Research Institute (EPRI), sponsored by the nuclear power utilities, performs research germane to aging effects. One such program is the research on the correlation of age-sensitivity and seismic qualification. This research is being conducted by Wyle Laboratories for EPRI.

The purpose of this research is to analytically and experimentally evaluate the correlation between aging (thermal and operational cycling) and the ability of selected electronic and electrical components to perform in a seismic environment. To date the program has consisted of postulating hypotheses, based upon knowledge of material characteristics, as to the significance of aging on the performance of safety-related components during a seismic event. A test program was performed in which 1,944 components were subjected to seismic testing, and 1,943 operated without failure. Identical samples of the components had been preconditioned prior to seismic testing. The samples were grouped as to the type of preconditioning. The groups were unaged, cycle aged, thermally aged, and cycle/thermally aged. Both aging and seismic stresses applied to the components were purposely more severe than typically required for qualification of safety-related equipment.

The testing was statistically designed to provide a high level of confidence in the results. No difference was noted in the seismic performance of aged and unaged components for which it had been hypothesized that aging was insignificant or that the aging mechanism was not amplified by the seismic event. In addition, relays had been chosen and hypothesized to have correlation between aging and seismic performance. This hypothesis was rejected for the relays because statistically significant correlation was not demonstrated. Only one out of five types of relays experienced a failure during seismic testing. This research is continuing into other components and equipment types.

Another valuable source of information on aging effects is the equipment qualification programs themselves. Over 1,000 have been conducted, which included natural and/or artificial accelerated aging. The design basis events simulated have been both seismic and accidents. Virtually every type of safety-related equipment has been tested. This includes simple devices such as switches and cables, complex electrical equipment such as control panels and computers, and large machinery such as safety-relief valves and diesel generators. The aging techniques utilized have been state of the art with a continuous increase in sophistication as the latest research results available have been incorporated into the programs.

The results of these equipment qualification programs have demonstrated that the performance of the safety-related functions is not significantly degraded by aging for the majority of safety-related equipment tested. Even when degraded, the performance demonstrated has generally been better than required. For most safety-related equipment the quantity of materials susceptible to significant deterioration due to aging is small. There are several factors that determine a material's susceptibility to aging, such as application, normal environmental conditions, and operability requirement during the design basis event. The same material may or may not be age-sensitive, depending upon the function of the material and its relationship to the safety-related function of the safety-related equipment. If a material does not have a safety-related function and its failure in any mode will not jeopardize the safety-related function of the equipment, then aging would be inconsequential. In another application, deterioration due to aging may have a significant impact on the safety-related performance. Changes in the normal environmental conditions can result in materials being reclassified from not age sensitive to age sensitive. For instance, the high ambient temperatures resulting from electrical energization of coils has caused significant aging degradation in otherwise benign environments. A third factor in the determination of age sensitivity is the operability requirement of the safety-related equipment during the design basis event. Temporary anomalous behavior has been noted during high-stress conditions of design basis event simulations with subsequent proper operation during low-stress conditions. If equipment is necessary only in the low-stress conditions, the age sensitivity during high-stress conditions would be of no consequence.

The results of almost a decade of demonstrating the performance of aged safety-related equipment is summarized in the following conclusions:

- Aging is addressed for safety-related equipment utilizing state-of-the-art techniques.
- Potential aging problems have been discovered and corrected.
- During qualification testing, limitations of materials due to aging are commonly noted. These limitations, consistent with the state of the art, necessitate maintenance and surveillance activities in the plant to provide assurance that aging effects do not become significant.

In general, it is noted that aging is a real concern, but the research and testing performed to date demonstrate that a high degree of confidence exists that it is a manageable concern. It is hoped that continued research will be performed to provide further assurance of manageability and improve the state of the art in quantifying and simulating the effects of aging.

In particular, continued aging research is suggested in accident qualification to address correlation between aging and performance during an accident, interfaces between equipment and various alternative interfacing techniques, and evaluations of the degradation of operability on the system level so that conclusions can better be made of the system degradation by knowing the impact of major components of the system. Continued efforts are recommended to account for the limitations noted in aging programs into the plant maintenance and surveillance activities.

ACCELERATED AGING METHODS OF ELECTRICAL EQUIPMENT AT ELECTRICITE DE FRANCE

Jean Roubault
Electricité de France (E.D.F.)

ABSTRACT

1. Conventional aging methods:
   For 30 years, E.D.F. developed many qualification tests procedures for conventional equipment used in its network and in its power plants. Such procedures are validated by permanent comparison with operating experience; hence, they are accepted for class-1E equipments used in mild environments. For harsh environments, such procedures will replace in the near future a provisional sequence analogous to IEEE-323.

2. Aging and seismic tests:
   Some comparative tests on new and aged small components showed no influence of aging on seismic vulnerability. For such components, E.D.F. tends to admit that aging has no influence if all critical frequencies are over 100 hertz. For larger equipment, an analysis is made for every case and concludes often but not always to the noneffect of aging.

The nuclear safety requirements for class IE electrical equipment imply the simulation of natural aging in qualification test programs. Such requirements quite agree with E.D.F. practice of qualification tests in order to show aptitude to long-life service of conventional equipment for electrical generation and supply.

Such procedures, whose elaboration needs sometimes up to five years, have now the benefit of enough experience to appreciate their effectiveness.

From another standpoint, class IE equipment is often identical or similar to conventional equipment. For these reasons, E.D.F. tended towards the use of its conventional aging procedures or of similar ones.

Such procedures are actually used for equipment that does not have to function during a loss-of-coolant accident (LOCA) or main steam link break (MSLB).

For LOCA or MSLB qualification, we used until today a different sequence analogous to IEEE 323-1974 but intend to replace it by sequences derived from our conventional procedures in the near future.

1. CONVENTIONAL TEST PROCEDURES

1.1 Leading Basis

A qualification procedure includes two main test families:

- Verification of functional characteristics and
- Judgment on on-life behavior.

The first includes the verification of performances under normal and extreme ranges of functional and environmental conditions during normal operation of the plant.

The second is a far more ambitious object since the matter is to try to simulate, in some few hundreds or thousands of hours, up to 20, 30 or 40 years of operation.

For much equipment, such direct correspondance is not possible, and the object of tests is rather to acquire a guarantee on the on-going evolution of the main functional characteristics and on the robustness of the equipment.

To establish this part of qualification tests, E.D.F. proceeds as follows:

1.1.1 Analysis of the Equipment and of Its Calculation

This analysis is done with the manufacturer and consists of the appreciation of the main dimensioning factors and of the strains giving the smaller safety margins.

1.1.2 Analysis of Different Operation Modes

This analysis gives information to estimate the nature and importance of the strains and their cumulative effect in operation.

1.1.3 Research of Accelerated Aging Laws

Some general laws exist on the aging of insulating materials under thermal (ARRHENIUS) or dielectric (WEIBULL) strains. They allow, to a certain extent, to accelerate phenomena, but it must be emphasized that they are limited

- To homogeneous insulating materials and
- To strain ranges without threshold, nonlinearities, or secondary effects.

Such laws must be carefully used and only for equipments mainly constituted of homogeneous materials. It is partly exact for cables but surely not for motors, transformers, etc.

The use of such laws, mainly for thermal aging, may also result from an investigation of new material or equipment if there is not enough time allowed to make a complete analysis.

They are then used provisionally until new tests, better elaborated, are available.

Anyhow, it is judicious and internationally admitted to accept extrapolations only on the basis of experimental, closely checked results and in a limited temperature range.

1.1.4 Operating Experience Results Analysis

Insofar as similar equipment has been operating for some years, the analysis of main failure causes is very useful, especially if it can be related to design weaknesses.

The statistical form of experience results allows also to appreciate the equipment quality and to modify test security according to the required target.

1.1.5 Provisional Test Procedure

From previous analysis, it is obviously impossible to standardize a single test sequence applying to all equipment.

Therefore, the tests are chosen for each type of equipment, taking at best into account:

- The most critical dimensioning factors,
- The operation modes,
- The adequate, accelerated aging laws, if they do exist, and
- The possible failure modes.

This subtle cocktail results in a test sequence, often with combined strains. In such a sequence, every step must be considered in order to secure the best efficiency to the following.

Manufacturers are closely associated with the formulation of the sequence, and the severities are fixed with their agreement for the first investigation campaign.

After this campaign, qualification tests and severities are definitely fixed. A detailed specification is established, taking into account existing national and international standards.

Such a qualification program, in use at E.D.F. for many years, allowed a sensible reduction of in-service failures; it is permanently adaptable to new equipment and is constantly enriched by operating experience.

It gives reproducible results and is not dominated by rigid acceleration laws, which do not meet the complex constitution of most equipment.

Finally, it has recourse to tests sometimes far from actual strains but covering the great variety of equipment used in our work.

1.2 Examples of Application

The above principles have been applied to much different equipment leading, of course, to apparently different procedures.

Among equipment similar to class 1E equipment, we may mention

- Medium voltage squirrel-cage motors whose aging is made by (usually) 1,500 start-stop cycles under normal voltage and load;
- Components such as relays, pressure transmitters, connectors, solenoid valves, etc., whose aging, different for every type, includes a mechanical and environmental sequence followed by an accelerated operation sequence;
- Electrical supply cabinets that endure electrical and mechanical open-close cycles; and
- Cables, rectifiers, inverters, batteries, transformers, etc.

2. CLASS 1E TEST PROGRAM

2.1 Categories

In France, class 1E equipment is divided in three categories:

- K1: Equipment located inside containment and useful during and/or after a LOCA or an MSLB (harsh environment),
- K2: Equipment located inside containment but useless after the beginning of a LOCA or an MSLB, and
- K3: Equipment located outside containment (mild environment).

2.2 K3 Equipment

Such equipment is nearly always of a conventional type. The K3 equipment test program joins up the conventional program and a seismic test.

2.3 K1 Equipment

This equipment is always of a special design, and the accident conditions are very severe.

At the present time, we use a special aging sequence, analogous to the requirements of IEEE 323-1974 standard:

- Thermal aging following a pseudo-Arrhênius law,
- Damp heat test,
- Accelerated operation under normal conditions,
- Radiations (aging part only), and
- Vibrations.

This sequence is, of course, followed by the seismic and accident simulations. It was adopted, some years ago, due to lack of comparative experience and of time.

For the future, E.D.F. and the French Safety Authorities are planning its replacement by aging sequences derived from conventional programs (as used in K3 qualification). As a matter of fact, our experience is now sufficient to allow comparisons between these two sequences.

2.4 K2 Equipments

E.D.F. has two policies:

- If similar K1 equipment does exist, its use for K2 functions is cheaper than the development and the qualification of specific K2 equipment.
- If there is no similar K1 equipment, the K2 test program adds an irradiation test to the K3 program. If possible, we then try to use K3 or near-K3 equipments.

3. AGING BEFORE SEISMIC TESTS OF K3 EQUIPMENT

Formerly, seismic tests of K3 equipment were made in accordance with IEEE 344-1971, without aging. Then, we had many test results that we did not want to do again for the same equipment to be used in new plants.

For this purpose, we made some tests and analyses.

3.1 Components

After conventional qualification tests of components, we keep the aged test samples and nearly always a new identical one for reference.

We were thus able to perform comparative seismic tests on both new and aged samples of some components (electromagnetic relays, electronic time relay, pressure transmitter, solenoid valve).

No significant difference was found.

This result is in accordance with the result of American tests* published last year. We learned recently that another study of Wyle Laboratories will be soon published.

Moreover, the conventional qualification of components (see Paragraph 1.2) always includes a vibration test with measurement of critical frequencies. The analysis of results accumulated in some 15 years led us to think that if critical frequencies are over 100 hertz, they cannot slip by aging down to the seismic range.

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*Carfagno and Heberlein, IEEE Transactions, Power Apparatus and Systems. Vo PAS, 99, 6, Nov/Dec 1981, pp 2272-2280.

In such case, we make the assumption that aging does not affect the seismic vulnerability.

3.2 Larger Equipment

For larger equipment, the analysis must be made for each case. From those made up until today, the conclusion was often, but not always, that aging did not affect the seismic vulnerability.

EQUIPMENT AGING - AN OVERVIEW OF STATUS AND RESEARCH NEEDS

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Abstract The problem of equipment aging in nuclear power generating stations is viewed from the standpoint of the qualification of safety-related equipment. A brief summary of developments related to equipment aging and qualification (in industry, the regulatory arena, and in the research area) is presented to focus on the current status of the problem. Several research topics are suggested for consideration as components of any general plan of research on plant aging. Among these is a topic concerned with an effort to establish the relative merit of different approaches to account for the functional degradation of equipment in terms of their cost and their contribution to assurance of public safety and a topic to investigate ways to improve the monitoring of functional degradation of equipment in service. Some factors to be considered in choosing among research topics and in establishing funding levels are mentioned.

INTRODUCTION

The requirement to account for degradation of the functional capability in the qualification of safety-related electrical equipment for nuclear power applications became generalized by the publication of IEEE Std 323-1974 [1] and its endorsement by Regulatory Guide 1.89 [2]. In 1980, the aging requirement was extended through IEEE Std 627-1980 [3] to the qualification of mechanical equipment. Soon after the publication of IEEE Std 323-1974, it became obvious that implementing what appeared to be a reasonable requirement was by no means straightforward; and to allay some of the industry's concern and confusion, IEEE published a supplement to the foreword of the standard, which stated, "It is expected that known technology will be utilized in any aging program." One of the major difficulties is that of correlating the accelerated aging portion of a qualification program with the calendar period of service simulated, thereby establishing what is called the qualified life of the equipment. Several years after the publication of IEEE Std 323-1974, the IEEE Working Group that produced IEEE Std 381-1977 for the qualification of Class 1E modules concluded that in most cases a quantitative correlation would not be feasible [4, 5]. A note in the aging section of Reference 4 makes the following observation:

> A variety of technical information regarding accelerated life testing of individual components appears readily available. However, considerable uncertainty in the state of the art still clouds one's ability to translate this component information into a scheme for simulating advanced life conditions for many different components assembled into a module.

Appendix B of Reference 4 includes an excellent review of module aging problems.

While much has been learned in the years since 1974, the aging requirement remains a complex issue. This is illustrated by numerous papers on the subject, a selection of which are References 6, 7, 8, and 9. Existing guidelines are general; there is little uniformity among the approaches to satisfying the requirement; and there is inadequate consensus between the regulators and the regulated on acceptable approaches. The situation is one that can clearly benefit by well-chosen research, and an effort is made in this paper to identify research topics that merit consideration.

DEVELOPMENTS IN AGING THEORY, TECHNOLOGY, AND METHODOLOGY

During recent years, there have been numerous developments in the theory, technology, and methodology of equipment aging as applied to the qualification of safety-related equipment. A substantial amount of research into the basic mechanisms of material degradation has been conducted at Sandia Laboratories [10, 11, 12, 13, 14, 15]. This has included research on the influence of oxygen diffusion in conjunction with radiation, radiation dose-rate effects, and the synergistic effects of heat, radiation, and humidity. An important earlier study on the effect of combined environments versus consecutive exposures on insulation life was conducted at the Naval Research Laboratory [16]. A more recent paper [17] gives an excellent analysis and review of radiation damage in organic materials. The Franklin Research Center has conducted studies of the effect of aging on the seismic fragility levels of contact-type devices [18], the natural aging of cables [16], the effect of (so-called) synergistic testing of cables [19], and has also conducted a review of equipment aging theory and technology [9]. Nuclear steam system suppliers have developed and executed qualification programs [21, 22], and utility owners' groups have undertaken cooperative qualification programs. EPRI has supported investigations of the correlation of aging to the seismic capability of electrical components [23] and of the radiation endurance of materials and components [24]. The Atomic Industrial Forum has produced a number of position papers on several topics related to equipment qualification including aging [25, 26]. In addition to the work done in the United States, considerable research and development related to equipment aging has taken place in other countries, principally in Japan, France, and Germany.

At the Nuclear Regulatory Commission, there have likewise been numerous developments relating to equipment qualification encompassing the aging requirement. The DOR Guidelines [27] for review of the environmental qualification of safety-related equipment in operating plants was produced in 1980. A significant concept introduced by the DOR Guidelines was the distinction between mild and harsh environments, acknowledging that qualification of safety-related equipment in harsh environments has higher priority. The Standard Review Plan (NUREG-0800, Section 3.11) defines the qualification criteria for mechanical and electrical equipment in harsh and mild environments. An interim staff position on environmental qualification (NUREG-0588), initially published as a draft, was most recently published as Revision 1 in July 1981 [28]. Recently, a proposed revision of Regulatory Guide 1.89 [29] on environmental qualification was published.

Through its Bulletin 79-01B, the NRC's Office of Inspection and Enforcement required the licensees of all operating reactors to submit documentation of their equipment environmental qualification. A review of the documentation identified qualification deficiencies, many of them concerned with aging, that the licensees were requested to correct; a review of updated documentation submitted by the licensees to the NRC is still under review. In addition to deficiencies in qualification programs, the NRC has also discovered deficiencies in the execution of test programs. Two independent actions have been taken to help correct these problems. An implementation guide is being prepared to clarify the qualification requirements and, particularly, provide guidance for correction of the deficiencies found in qualification programs; the NRC plans to send this guide to each licensee of operating reactors along with its Safety Evaluation Report on environmental qualification. The other action was to initiate an effort toward requiring accreditation of any laboratory performing equipment qualification testing.

CURRENT STATUS

At this juncture - after a substantial amount of research and development, a massive amount of equipment qualification testing and analysis, and several regulatory developments that relate to aging have taken place - there is a widespread recognition of the limitations of aging technology and of the need to place more emphasis on engineering judgment in establishing equipment qualification, particularly with respect to aging.

Several efforts are currently under way at EPRI with the objective of improving the qualification process and increasing the ratio of benefit to cost. One effort is a review of mechanical equipment qualification, another is a survey of equipment surveillance procedures to supplement existing qualification methods, and a third effort is the preparation of a qualification guide that will describe the entire qualification process from the specification of equipment through the preparation of a qualification program, execution of the program, and documentation of the entire qualification program, including the testing and analysis. A number of seminars based on the qualification guide have been tentatively scheduled for presentation in six cities during the first half of 1983. (None of these efforts has reached the stage where reports are available.)

POTENTIAL RESEARCH TOPICS

Evaluation of the Relative Merit of Different Qualification Methodologies

For most of the last 8 years, the aging qualification requirements have not changed in any essential way; the greatest change has been the increased recognition of the limitations of aging technology. The chief dilemma is that of establishing a qualified life for safety-related equipment, which is the period of normal service represented by the accelerated aging part of the qualification program. To consider use of the Arrhenius model an acceptable approach for establishing the qualified life fails to recognize the extent of the limitations of the model for the acceleration of thermal aging and the fact that no practical acceleration model exists for most other aging stresses. If we add to this the fact that our ability to account for the aging of the interfaces and junctions between materials and components and for synergistic effects of combined stresses is primitive by comparison with our limited ability to account for the thermal aging of single materials and components, it becomes all the more obvious that engineering judgment is not only an essential factor in establishing qualified life, it is actually the dominant factor.

There are several undesirable consequences of the limitations of aging technology. Whatever accelerated aging procedure is developed is very likely to produce significantly more or significantly less degradation of functional capability than will take place in the application. In the former case, a specimen may fail a qualification test even when it might perform satisfactorily in service; and in the latter case, a specimen may pass a qualification test but fail to meet its requirements in service. Another undesirable consequence is that the qualified life established is subject to considerable uncertainty. The typical 95% confidence limits for a 40-year thermal life (i.e., excluding all other aging stresses) are of the order of 10 years and 400 years. Thus, a qualified life established exclusively on the basis of the Arrhenius model can easily be too optimistic or too conservative.

In view of the uncertainties associated with existing procedures for taking equipment degradation into account, it is recommended that an investigation to establish the relative merit of alternative aging procedures in terms of their impact on public safety be considered. The primary goal of equipment qualification, and of the aging element in particular, is to provide reasonable assurance that public safety is adequately protected. However, we have no measure of the effect of existing qualification procedures on the probability of risk to the public. The present equipment qualification process demonstrates that a representative specimen <u>can</u> function as specified after the degradation that takes place during the qualified life; the probability that the equipment <u>will</u> function is not a direct output of the qualification process. It is possible that other approaches to aging qualification may be superior in terms of assuring public safety or may be much less costly without compromising public safety. Therefore, a research effort to establish the relative assurance of public safety provided by alternate aging qualification procedures and to evaluate the cost of each alternative merits consideration. The goal of this research would be to increase our assurance of safety with due consideration to cost. This approach is consistent with the recognition that the establishment of quantitative safety goals and objectives might contribute to a determination of priorities and allocation of resources more nearly in proportion to the estimated risk to the public [30, 31].

For purposes of illustration, several possible alternatives to the existing aging qualification procedure will be mentioned. First, the existing aging qualification procedure can be summarized as follows: identify significant aging mechanisms, establish accelerated aging procedures based on existing technology, and establish a qualified life. (In practice, this frequently is translated into a search for the activation energies for the materials and components of a device and deriving a qualified life primarily on the basis of the Arrhenius model for thermal aging.) Alternative procedures that may be included in an initial list of candidates to be investigated for their relative merit include:

1. Conduct qualification testing on new specimens, without any accelerated aging.

2. Conduct qualification testing on new specimens and establish a surveillance program to monitor degradation in service.

3. Follow the existing procedure with the exception that a standard value of activation energy be used (as suggested in Ref. 32) to establish the accelerated thermal aging procedure for the equipment in lieu of the identification of activation energies for each material and component.

4. Establish standard testing procedures patterned after those used in military applications. (This type of test is illustrated by References 33, 34, and 35.)

These few examples serve to illustrate the type of candidate approaches to be evaluated. Some candidates may be eliminated from an initial, exhaustive listing without extensive investigation. It is quite possible that a quantitative determination of the relative merit of the alternative procedures will not prove feasible; engineering judgment may play the major role. Also, it is doubtful that a single approach can be acceptable universally. It is more likely that cost/benefit considerations will recommend different approaches for different classes of equipment and service environments.

A comprehensive study would also compare the importance of qualification procedures with other safety-related activities, such as improvements in operator training, maintenance procedures, and surveillance procedures. The findings could justify reducing the high cost of qualification or directing it to activities with a much greater benefit potential.

Degradation Monitoring

A study of degradation monitoring procedures and how they can be applied by the nuclear utility industry is another topic that merits consideration in any aging research program.

The uncertainties involved in establishing the qualified life of safety-related equipment make it risky to end the qualification process once a qualification program has been executed successfully and it has been accepted. Monitoring of equipment degradation in service is recommended as an essential supplement to existing qualification procedures. Since existing surveillance, inspection, testing, repair, and maintenance procedures do not adequately fulfill the needs of degradation monitoring, there is a need to study this topic.

It is suggested that an investigation be undertaken encompassing the following elements:

- o selection of safety-related equipment items for which degradation monitoring procedures are most likely to produce a significant benefit to public safety
- o identification of existing surveillance procedures that can be applied directly to degradation monitoring
- o identification of monitoring procedures that require development
- o development of selected degradation monitoring procedures
- o investigation of the feasibility of introducing degradation monitoring on an industry-wide basis
- o establishing a mechanism for ongoing evaluation of degradation monitoring data, developing improvements in monitoring procedures, identifying findings of safety significance, and making the data available to the nuclear industry. (This could very likely be accomplished by extension of existing mechanisms developed by EPRI and the nuclear utilities for reporting information of safety significance.)

Installed equipment is subject to all of the aging stresses and synergistic effects and all of the interface influences of the real service environment; therefore, carefully designed degradation monitoring procedures have a high probability of yielding important findings in a reasonable time. It is true that such monitoring is less likely to yield a fundamental explanation of aging and synergistic mechanisms than is possible under controlled laboratory conditions; but it is more likely, for a given expenditure of time and money, to yield practical results contributing to safer plant operation.

Basic Research

Basic research, of the type reported in previously mentioned references [9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 23, 24], on material and component degradation under the conditions of nuclear service and on the correlation of such degradation with the functional capability of safety-related equipment, is worth continuing. Such research increases our understanding of basic aging phenomena and helps us improve qualification methodology; it helps us avoid doing things that are not meaningful and helps prevent our overlooking important aspects of equipment qualification. However, it should be recognized, on the basis of the rate of progress from past research on aging and synergistic effects, that results are obtained slowly and that their practical application is frequently limited. For example, recent research has shown that radiation dose-rate effects, particularly for relatively thin insulating materials in which oxygen diffusion contributes to degradation, are much more important than was previously thought. However, this finding does not yield a practical solution to the accelerated aging problem. Long-term radiation doses will still be simulated by radiation exposures conducted at much higher dose rates than those that occur in a nuclear plant because of the practical limits imposed on the time that can be devoted to equipment testing and the cost that can be accommodated, as well as limitations imposed by the availability of test facilities. Nonetheless, this research finding can be used to improve equipment qualification programs in several ways, such as (1) better recognition of the limitations of radiation endurance data obtained at dose rates much higher than those that exist in the application, (2) modifying the test procedures to the extent feasible using as low a dose rate as practical, avoiding the depletion of oxygen from the surroundings of the specimens, seeking surveillance procedures to monitor the degradation in service, and - for certain materials - conducting radiation aging prior to thermal aging.

Recognizing that the benefits of basic research are of the type illustrated above, and that detailed procedures for equipment aging are not the expected outcome, will help avoid the unjustified extrapolation and application of research findings that sometimes take place.

Testing of Equipment Removed from Service

One of the major distinctions between the demands on safety-related equipment in nuclear service and the demands on similar equipment in ordinary industrial service is the requirement that the equipment be capable not only of functioning satisfactorily over a long period of normal service, but that it also be capable of performing an essential safety function as late as at the end of its qualified life, frequently under stresses much more severe than those existing in normal service; and sometimes the equipment is required to remain functional under severe stress for periods up to the order of a year. Furthermore, in-containment equipment may be required to remain functional under such conditions without the benefit of restorative actions that could be taken if the equipment were accessible.

The foregoing observation identifies another research topic, the investigation of the effect of aging on the ability of equipment removed from service to perform under simulated accident conditions. Analysis of equipment that has undergone accident service (such as equipment at the Three Mile Island plant) to look for correlations between degradation during its service history and its performance during the accident is not a new research idea, but it should be included in any listing of research topics related to plant aging.

Methodology of Accelerated Aging

As with the preceding topic, the study of aging methodology is not altogether a new research topic; however, it merits being included in this listing. As an example, research under this general topic could include an investigation to establish guidelines for the accelerated thermal aging of test specimens. Guidance is needed on matters such as: criteria for the size of an aging oven compared to the space occupied by the specimen; the rate of air change in the oven; the number and positioning of temperature sensors; temperature uniformity requirements in the test chamber; energizing of specimens during accelerated aging; the interspacing of life cycling through the aging period; and whether the cycling should be conducted at the aging temperature or at a temperature within the range of service conditions.

CONSIDERATIONS FOR ESTABLISHING THE SCOPE OF RESEARCH TOPICS AND THEIR RELATIVE PRIORITIES

Correlation with Reduced Risk to the Public

Since the primary purpose of equipment qualification is to protect the public from the potential hazards of operating nuclear power generating stations, a primary consideration in selecting research topics is an evaluation of their potential contribution to enhancing public safety.

Potential for Reducing the Cost of Qualification Without Compromising the Adequacy of Qualification

If more than one approach to a given aging problem is adequate to satisfy the demands of public safety, cost becomes a primary factor of comparison and selection. The relative cost of different acceptable approaches should be evaluated on an integrated basis, i.e., it is not sufficient to calculate only the cost of implementing a particular approach: another essential ingredient is the cost impact over the long term. The cheapest acceptable qualification procedure is not necessarily the most beneficial from the cost standpoint over the long run if it involves lower plant reliability, greater downtime, and the wasted time and cost associated therewith.

Time Required to Achieve Practical Results

Another factor to consider in developing a comprehensive research program is the probable time required to obtain practical results from each research topic. It is necessary to estimate the length of time during which nuclear fission will continue to be a dominant source of energy before a different mechanism such as fusion or solar power becomes dominant. To be of value to the present nuclear industry, practical research results must be obtained within a time that is short by comparison with the probable lifetime of power generation by nuclear fission.

Obviously, research with a potentially fast payback is more attractive than research with a long-range payback; but the more basic the research, the less likely it is to have rapid payback. This consideration tends to reduce the value of long-range, fundamental research. The history of the development of aging theory and technology provides a benchmark by which to estimate the probable payback time of basic aging research. The Arrhenius theory was developed nearly a 100 years ago and the first significant effort to apply it to the deterioration of electrical insulating materials took place not much over 30 years ago [36]. Aging studies over the last 10 to 20 years have enhanced our understanding of the subject, but they have made little progress toward identifying procedures for quantitative simulation of equipment aging. Clearly, basic research on the degradation of materials is not likely to yield practical aging qualification procedures for complex equipment in a reasonable period. And in view of the much greater complexity of synergistic effects by comparison with the effects of single stresses, it appears altogether unjustified to expect significant advances from a reasonable investment of time and funds in research on synergistic effects. However, the importance of basic research in advancing the understanding of aging and synergistic phenomena should not be minimized. The foregoing discussion is intended to give us a realistic picture of the prospects. Basic research should be continued with due recognition of these prospects. If we recognize the time scale of basic research, it will not be required to offer fast turnaround to justify itself. This recognition will also reduce the temptation to make unjustified extrapolation of research findings, which result in their premature and potentially erroneous application.

Distribution of Available Funding

Usually research administrators have less money available than would be needed to fund all the viable research program candidates that fall within their jurisdiction, and they are faced with the problem of selecting the most promising candidates. The considerations already discussed above will help in establishing priorities and in evaluating the relative merit of candidate research programs. An additional factor to consider is the viable magnitude of funding for any given program.

Distributing available funds among so many candidates that none is able to proceed at a healthy rate can result in negligible accomplishment. For example, it has been suggested that the knowledge of synergistic effects would be significantly advanced if each vendor developing a qualification program were to investigate the subject. However, the limit of funding a vendor could invest in studying synergistic effects is probably of the order of $10,000, and the ability of the average vendor's staff to conduct such studies is limited. Therefore, it is unlikely that any vendor could make a significant contribution. No matter how great the number of vendors that might undertake a limited investigation of synergistic effects, the integrated result would still be negligible. At the other extreme, if all available funding were awarded to a single organization, it could lead to too narrow a focus, with consequent loss of effectiveness.

For the types of research suggested in this paper, an effective approach might be to fund initially a short-term/low-cost exploratory study in each case, i.e., programs of approximately 6 months' duration and a cost of about $50,000 to $100,000. Depending on the importance of the topic and the availability of qualified research organizations, more than one exploratory program might be funded in some topics. Once a topic is determined to merit full-scale pursuit, and the qualifications of potential contractors have been established, possibly with the aid of the exploratory programs, a higher level of funding and longer program duration would be indicated. Typically, funding of the order of $500,000 spread over 1 1/2 to 3 years, with periodic interim reviews of progress, would be more conducive to effective research.

CONCLUSION

The status of equipment aging has been reviewed and several aging research topics have been discussed. These topics include fundamental research into aging and synergistic effects aimed toward increasing our understanding of basic aging phenomena and providing guidance for improving our ability to account for aging in the selection of equipment for nuclear power generating stations; an investigation of the relative merit of alternative approaches to the aging element of qualification, measured primarily in terms of their impact on public safety and secondarily in terms of cost; and an investigation of in-plant degradation monitoring, as a supplement to existing qualification procedures, aimed at providing increased assurance of equipment operability and at yielding a practical knowledge of equipment degradation in service. In addition, it is recommended that existing research, such as studies of the functional capability of equipment that has been operated in nuclear plants, the correlation of aging with seismic endurance, and testing methodology, be continued.

While the use of nuclear fission for electrical energy generation has existed long enough to build up the impressive number of approximately 500 reactor-years of service, there have been few occasions (fortunately) to observe the functional capability of aged equipment under accident conditions. On the one hand, this is an indication of the high level of safety built into the design, construction, and operation of nuclear power plants. On the other hand, this is not a basis for complacency regarding the impact of equipment aging on nuclear safety. Therefore, the research topics discussed merit consideration in any effort to review research needs, establish long-range research plans, and recommend funding levels.

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METHODOLOGY FOR ESTIMATING REMAINING LIFE OF COMPONENTS USING MULTIFACTOR ACCELERATED LIFE TESTS

by

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ABSTRACT

Accelerated life tests are recommended for estimating the remaining service life of complex, service-aged components. The multifactor life tests are developed by a select team of scientists with 4 to 6 man-months of effort in accord with a set of agendas described in the paper. Factorial tables, hierarchical trees, and associated mathematical models are used to guide the efforts of the design team in obtaining an experimental design that is scientifically and statistically defensible.

INTRODUCTION

It is sometimes necessary to use accelerated life tests to predict the life of long-life components. However, many accelerated life tests are expensive failures. In the authors' view these failures result largely from the fact that accelerated life tests are often poorly designed, are based on insufficient scientific and statistical expertise, and are not subjected to a sufficiently rigorous review before the design is implemented. The methodology described below has evolved over the past 10 years in an effort to improve the validity of accelerated life testing. The methodology has been applied to the design of accelerated aging tests for spacecraft batteries(1), and to photovoltaic solar cells(2)(3). Current applications are being made to cable insulation, weapons structure, and to radioactive waste packaging.

APPROACH

Assumptions Concerning the Component

The objective of accelerated aging is to increase the rate of aging of the component subject to the constraint that the dominant physical mechanisms responsible for aging do not change from those associated with operation at normal stress. In addition to long life at normal stress, the aging of the component over time is frequently taken to be a nonlinear function of time.

Figure 1 shows a family of hypothetical aging curves, describing the degradation of performance over time. The x-axis is scaled according to a dimensionless measure of time t/t_F, where t denotes the current age of the component and t_F denotes the age of the component at the time of failure. The vertical axis shows the ratio of a measure of performance P at time t/t_F relative to the initial performance at time 0. The equation shown in Figure 1 is similar to an equation used by Simoni to describe the degradation of breakdown strength in high voltage cables.(4)

Interdisciplinary Team

The team that is assigned responsibility for designing an accelerated aging test usually consists of 5 to 10 members. The team members must represent the various scientific disciplines that are associated with the physics of the degradation processes. The team must include a statistician knowledgeable in the area of the statistical design of experiments. It is the responsibility of the team to assure that the resulting experimental design is scientifically and statistically defensible.

In the procedure described below each scientist is required to make quantitative predictions of test measurements that the scientist believes would result if the test were actually run. These predictions are to be based on the scientist's knowledge and assessment of the relevant physics, literature, and data. Candidate team members who are unable or unwilling to make such predictions are excluded from the team.

The team activities are coordinated and controlled by a team manager. The team manager must have sufficient authority to arrange meetings, make assignments, focus discussions, limit debates, and assure that good documentation is obtained.

The team manager is also responsible for assuring that the activities of the team itself are accomplished within time and budgeting allotments. Depending on the detail of the experimental design and the complexity of the component, past efforts have ranged between 4 and 12 man-months.

Desired Graphical Output

Figure 2 shows an example of a graphical display generated for an accelerated aging test for photovoltaic solar cells.(5) The design team is required to present its final design in a form similar to this Figure. The log-log plot of degradation rate versus time includes a solid line passing through nine points with each point representing a separate test. The nine points represent the predicted degradation rates generated by the team. The test conditions associated with these nine points are indicated by the hierarchical tree shown above the line. The inclusion of the explicit extrapolation range serves to emphasize the magnitude of the extrapolation. The inclusion of the predicted points serves to identify the base from which the extrapolation is to be generated. Both the number of points and their relative distribution along this line are clearly important for assessing the suitability of the extrapolation.(6) It may be noted that such a plot clearly shows the absurdity of running a one-point accelerated test.(7)

The line shown in Figure 2 is symbolic in the sense that the points always fall on a line with a slope of -1. The plot is useful for a general overall assessment of the final experimental design; it is not to be construed as a method of data analysis.

TENTATIVE AGENDAS AND SEQUENCE OF ASSIGNMENTS

Preliminary Meeting

A preliminary meeting is required to describe the approach given below to candidate members of a design team. Candidates unwilling to make quantitative predictions are usually identified and eliminated at this stage.

Agenda #1

The first formal meeting of the team is based on the following agenda:

- identify candidate failure mechanisms
- choose one failure mechanism for the first iteration of the design procedure
- identify the associated stresses and the experimental variables for controlling these stresses
- define the dependent variable with appropriate units of measure.

In general, this agenda requires at least one day-long meeting of the team. The discussion of candidate failure mechanisms allows each member to expose his background and views on the degradation processes for the component. The failure mechanism selected for the initial effort is usually the one identified as most likely to be the dominant failure mechanism. The identification of stresses and experimental means for controlling them require knowledge of experimental procedures associated with controlled environments, monitoring, and instrumentation. Initial estimates of experimental costs may be introduced at this first meeting.

Because the team is responsible for the final test design and its ultimate success or failure in predicting remaining life, it is clear that the team must have full responsibility for choosing both the experimental stresses and one or more dependent variables. Typical examples of dependent variables selected by teams include the following: relative "severity" of each test, percentage loss in a performance measure relative to the initial level of performance, the percent change in a performance measure resulting from a percent change in a stress level, or simply the expected life of the component at each test condition.

It is essential that good documentation procedures be set up and implemented during the first meeting of the team. Strict adherence to the procedures should be followed in all subsequent meetings.

Agenda #2

The second meeting of the team is based on the following agenda:

- select a test range for each experimental stress by choosing a "low" and a "high" value for each stress
- make a preliminary assessment of all combinations of low and high stress levels to assure experimental feasibility
- define normal stress in quantitative terms.

Table 1(a) shows a factorial table in a general form for 3 experimental factors. Table 1(b) shows an example of a completed factorial table involving temperature T (50 and 95C), relative humidity RH (60 and 85 percent), and ultraviolet radiation (at 5 and 15 suns). The dependent variable for this example is the expected life, in months, of solar cells operated at the combinations of temperature, radiation, and UV levels shown in Column 2. The lifetimes are typical of individual predictions generated as inputs to the consensus results shown along the diagonal in Figure 2.

In statistical terms Table 1(a) represents a complete factorial design for 3 stresses. A similar table can be generated for any number of stresses. Such stresses typically include thermal, non-thermal, mechanical, and radiation stresses. Cyclic stresses, on-off cycles, storage periods, etc. may all be included in the experimental design if deemed necessary by the team. The low and high levels of each stress are indicated by 0 and 1. Thus, (0, 0, 0) denotes a three factor test condition with each stress at its low level; (1, 1, 1) denotes a test condition with each stress at its high level. The intermediate combinations of low and high levels include all possible combinations of such levels.

Experience to date suggests that not more than 4 or 5 stresses should be considered initially by the design team. The quantitative generation of predicted lifetimes, for example, for the resulting 16 or 32 different combinations of stresses is formidable. It is important to note that each row in Table 1 represents a candidate accelerated life test. The procedure described below determines which of these tests will be included or excluded in the final test design.

Scientists' Assignment

Each team scientist is required to fill out columns 3 and 4 of a table similar to Table 1(a). A time period up to 2 weeks is usually allowed for this purpose and each scientist typically fills out the columns independently of the other team scientists. Column 3 consists of the best numerical predictions that the team scientist can generate for the value of the dependent variable for each of the candidate tests. It is expected that these predictions will be based on references to the literature, laboratory data, computer models, previous experience, arguments based on physical principles, and any other source of relevant information. A written documentation file that is used to support each predicted value of the dependent variable is to be supplied to the team manager by each team scientist and referenced as indicated in Column 4.

Because factorial tables are familiar to statisticians, it is customary for the team statistician to lead the group in explaining how Table 1(b) is to be filled out. Because of a lack of expertise in the relevant physical mechanisms, Table 1(b) is typically not filled out by the team statistician.

The team scientists are responsible, as a team, not only for choosing what stresses should be used in the test design, but also what the low (0) and high (1) levels should be for each stress. The team is instructed to choose the levels so that the highest multifactor stress, indicated by (1, 1, 1) for 3 stresses, will be as stressful as possible under the constraint that the dominant failure modes are expected to be identical to those experienced under normal stress conditions. Similarly, the low stress levels should be chosen so that the (0, 0, 0)-test condition will be expected to yield measureable degradations within a specified time duration for the accelerated test, typically, 6 months to 2 years.

In order to accomplish these objectives it is necessary to define "normal" stress in quantitative terms, together with the associated failure modes under normal operating conditions.

Careful documentation is required in order to assure agreement among all team members concerning the low and high levels selected for each stress, the stress levels associated with normal stress, together with preliminary estimates of the test durations under the (0, 0, 0)-test conditions and the

(1, 1, 1)-test condition. The documentation should include not only the numerical results, but also the arguments and cited data that support these values. It is also desirable to record the failure mechanisms, failure modes, possible stresses, etc., that were considered by the team, but were subsequently excluded from further consideration. The arguments responsible for their exclusion should also be carefully recorded.

Statistician's Assignment

After each scientist completes Table 1(a), the predicted values are subjected to a statistical analysis as though the predicted values were actual measured data. The team statistician is responsible for making these analyses. All main effects and interactions (7) are to be computed by the statistician to identify the anticipated relative importance of the different stresses, the anticipated synergisms, etc.

Figure 3 shows a numerical example of a hierarchical tree that is used to represent the relationships among the experimental factors that are implicit in Table 1(b). This example is based on that shown in the upper left portion of Figure 2. Figure 3 indicates that the most important stress is temperature T, as shown by the fact that it is the first "splitting" variable at the top of the tree. The left and right branches of the tree correspond to temperature at the high (H) and low (L) levels, respectively. The scale at the top of the figure is used to establish the horizontal locations of the boxes of the tree. Large horizontal distances between boxes indicate important stresses. The tree is constructed so that the experimental variable having the largest conditional main effect is used as a splitting variable at each stage. This splitting process continues until the terminal boxes of the tree represent the 8 test conditions defined in Table 1(b). The resulting tree gives a quantitative graphical display of the conditional main effects and interactions(3).

By examination of the graphical tree representation, it is possible for the team scientists to make an assessment of the validity of their predictions as shown in the factorial table. Roughly, the factorial table requires the scientist to consider each test as a combination of stress levels. The combinations (0, 0, 0) and (1, 1, 1) are usually relatively simple combinations to consider and serve to define the low and high limits of the combined stress. The intermediate combinations of stress levels are much more difficult to assess. These intermediate levels require the scientist to make predictions for all possible "trade-offs" among the stress levels. The generation of the predictions for these intermediate trade-off conditions has proved to be the most difficult part of the design process.

The hierarchical tree contains exactly the same information as that given in the factorial table. However, in the tree the stresses are displayed, not as combinations of levels, but in terms of their relative overall importance (vertical position in the tree) and their average effect on the dependent variable (the horizontal spacing associated with boxes defined by successive splits). By considering both kinds of information the team scientists can detect and remove inconsistencies.

As a final test for consistency the scientists are required to describe their predicted results in mathematical form. These mathematical models must be shown to yield acceptable extrapolations to normal stress levels using the predicted values as a basis for the calculations. As part of this effort the team statistician usually fits polynomial forms to the data to permit assessment of the relative numerical coefficients. More appropriate models, including mathematical forms based on the Arrhenius and Eyring relations, are usually fitted to the hypothetical data by the team scientists.

The objective of these efforts is to obtain assurance, by several iterations if necessary, that the factorial table, the hierarchical tree, and the fitted mathematical models correctly represent each scientist's judgments concerning the anticipated relationships among the multifactor stresses and the predicted value of y. It is further required that the mathematical models yield defensible extrapolated degradation rates at normal stress conditions.

Agenda #3

The individually generated factorial tables, hierarchical trees, and mathematical equations are exposed for mutual review and criticism as part of Agenda #3. The trees form a basis for identifying areas of agreement and disagreement. In general, considerable disagreement can be expected. Wherever disagreements are identified, the supporting reference files associated with the factorial tables are then compared. This procedure exposes for criticism the arguments, data, and models that were used as a basis for generating the predictions. In general, the team is charged with identifying and adopting the supporting evidence that is agreed to be superior.

The objective of this agenda is to obtain a consensus factorial table with an associated hierarchical tree, and a mathematical model that generates the predicted values of the dependent variable and is supported by documented consensus arguments, data and calculations. To date it has been found that the process of obtaining a consensus can be difficult. However, the process is aided by the fact that most scientists seem to enjoy comparing arguments, data, and models with a view toward identifying the best overall compromise. No formal procedure has been identified to assure that a consensus is obtained. The manager of the team is taken to be responsible for assuring that an acceptable compromise is obtained. In some instances, several sessions have been required to complete this Agenda.

Agenda #4

The next step in developing the experimental design involves "pruning" the tree. It is usually found that the consensus hierarchical tree calls for an excessive number of tests. Moreover, for a variety of reasons, some tests may be declared unnessary by the team. Some tests may have been run already. It may be that, under certain conditions in certain branches of the tree, the effect of changing a stress from its low to high level would be expected to cause insignificant changes in the value of the dependent variable. Consequently, all splits in the hierarchical tree that are associated with relatively small horizontal distances are re-examined. These correspond to conditional main effects that are expected to be small, and can possibly be eliminated from the final test design.

The elimination of tests must be accompanied by careful documentation that explains the basis for the elimination. It is frequently desirable to take some initial account of the budget constraints at this time. If the tree cannot be pruned to a level consistent with budget constraints, the arguments of the team may be construed to indicate that either no accelerated test should be implemented or that a budget increase is necessary.

The Statistician's Second Assignment

A complete factorial design can be severly degraded by pruning. It may no longer be possible to obtain satisfactory estimates of certain main effects and interactions. For this reason the team statistician is next required to assess the statistical properties of the test design associated with the pruned hierarchical tree. In the likely situation that the design is statistically unacceptable, the statistician is charged with adding-back a minimal number of tests to achieve an acceptable level of validity. The objective is to obtain a subset of a factorial design that barely meets statistical requirements.(9) The benefits to be gained from adding-back specified tests must be documented by the statistician for review by the team scientists as a part of Agenda #5.

Agenda #5

The fifth agenda calls for the team scientists to accomplish the following:

- evaluate the statistician's augmented design
- arrive at a possibly revised consensus design
- review the overall design and insert additional levels of stress between the low and high levels to account for anticipated nonlinear relationships
- re-examine the final tests to assure that when implemented the test data will be sufficient to estimate all parameters in the fitted mathematical models
- re-examine the extrapolations of degradation rates to normal stress conditions
- with the statistician's participation, identify the number of components to be tested at each test condition
- identify the test instrumentation and measurement procedures, schedules, tear down analyses, etc. that are required for a fully specified test design
- document all choices made with the reasons given that support each choice.

Agenda #6

The last agenda consists of a team review and approval of a write-up that documents the final test design. The final design is to be expressed in a form similar to that given in Figure 2. The documentation should include the consensus factorial table, the hierarchical tree, and mathematical models used to fit the predicted values and to make extrapolations to normal stress levels.

Team Follow-Up

Ideally, the design team is also responsible for making concurrent analyses of the actual test data as the data become available. It is the authors' conviction that some of the validity of the accelerated life tests must be established by generating a convincing "track-record" during the course of the experiment. The track record can be obtained by requiring the team scientists to make real time predictions of two kinds. One kind of prediction occurs "within" stress levels. In this case the data obtained to date from a particular test condition are used to predict the measurement values to be expected at a specified future date, say a week or month hence. A second kind of prediction involves predictions "across" stress levels. In this case the data obtained at a high stress level are used to predict the measurement values at lower stress levels at future specified dates. This second kind of prediction is especially relevant to accelerated life testing. Acceptable predictions must be made using real-time extrapolations from high to lower stress levels during an accelerated life test in order to demonstrate the possibility for making convincing extrapolations from high stress levels to those associated with normal stress.

The description given above omits many details. For example, it may be necessary to repeat the procedure for different competing failure mechanisms; to run "side tests" to obtain data that cannot be obtained through monitoring measurements in test chambers; to put sufficient numbers of components on test to allow periodic removals for destructive tear-down analysis. All such matters are to be decided by the team scientists with full documentation of the related arguments.

CONCLUSION

The agendas described above, together with the factorial tables, hierarchical trees, and associated mathematical models, provide a useful formal procedure for extracting and synthesizing the expert knowledge that is required to improve the scientific and statistical validity of accelerated life testing.

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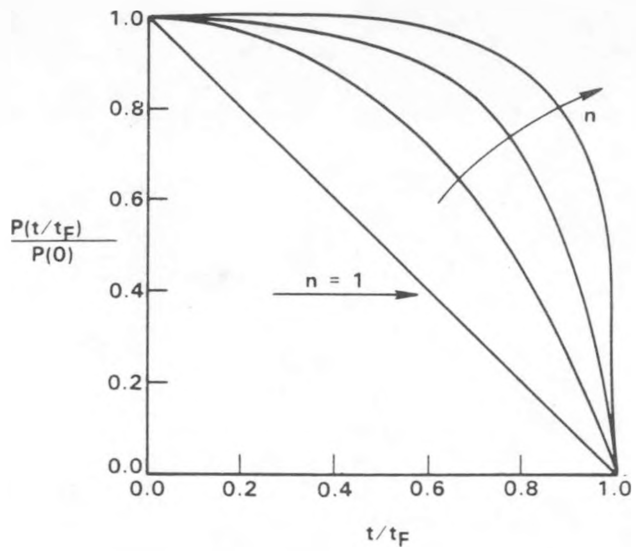

$$P(t/t_F)/P(O) = [1 - (t/t_F)]^{1/n}$$

Figure 1. Degradation of Performance

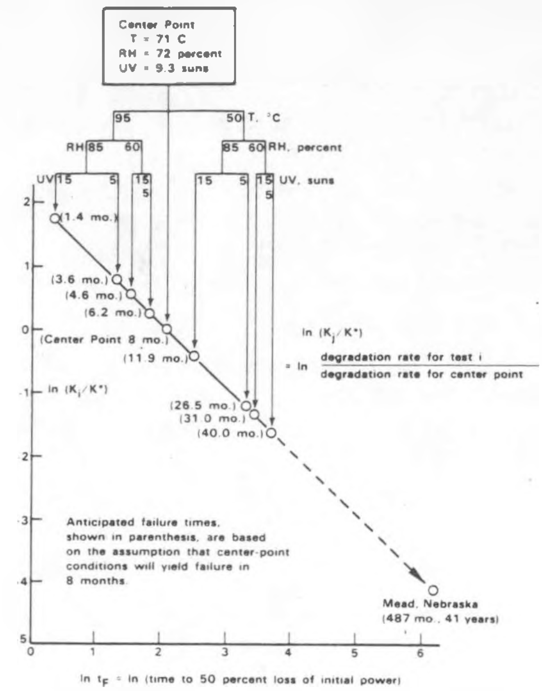

Figure 2. Illustration of Idealized Test PL Plan

Table 1(a) Factorial Table to be Completed Independently by Each Scientist on the Test Design Team (General Form for 3 Factors)

TEST NUMBER	TEST COMBINATION OF STRESS LEVELS(1) (S_1, S_2, S_3)	PREDICTED VALUE OF Y	DOCUMENTATION SUPPORTING THE PREDICTED VALUE
1	(0, 0, 0)	Y_1	(1)
2	(0, 0, 1)	Y_2	(2)
3	(0, 1, 0)	Y_3	(3)
4	(0, 1, 1)	Y_4	(4)
5	(1, 0, 0)	Y_5	(5)
6	(1, 0, 1)	Y_6	(6)
7	(1, 1, 0)	Y_7	(7)
8	(1, 1, 1)	Y_8	(8)

(1) 0, 1 DENOTE LOW AND HIGH LEVELS OF STRESSES S_1, S_2, S_3.

Table 1(b) Numerical Sample of a Completed Factorial Table

TEST NUMBER	TEST COMBINATION OF STRESS LEVELS (T C, RH %, UV SUNS)	PREDICTED LIFE AT TEST CONDITION(1), MONTHS
1	(50, 60, 5)	40
2	(50, 60, 15)	30
3	(50, 85, 5)	24
4	(50, 85, 15)	12
5	(95, 60, 5)	6
6	(95, 60, 15)	4
7	(95, 85, 5)	3
8	(95, 85, 15)	1

(1) VALUES SHOWN ARE TYPICAL OF INDIVIDUAL PREDICTIONS GENERATED AS INPUTS TO THE CONSENSUS RESULTS SHOWN ALONG THE DIAGONAL IN FIGURE 2.

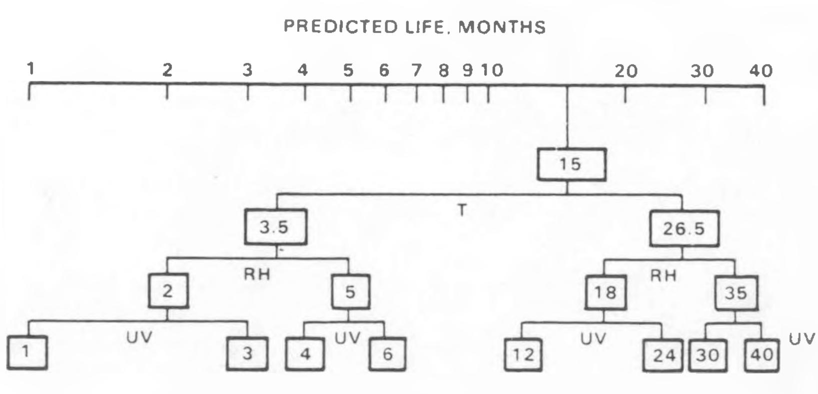

Figure 3. Numerical Example of a Hierarchical Tree Representation of a Factorial Experiment

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USNRC/Sandia
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Time-Related Degradation, A Key Issue in Nuclear Plant Safety Evaluations*

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ABSTRACT

Sandia National Laboratories is conducting a number of programs under NRC sponsorship which deal with safety-related equipment qualification issues, including the important aspect of "aging." Among these is the Qualification Testing Evaluation (QTE) program which was probably the first to devote significant effort towards aging research and was one of the primary motivators leading to the Workshop. The thrust of the QTE aging efforts has been on elastomeric materials, typically used in electrical cables, seals, gaskets, and the like; currently, efforts are being pursued on plant ambient environments measurements, aging of electronics, and aging of motors. A brief status report is presented in this paper.

SYNOPSIS

We also share some insights into the aging (1) as an issue, (2) what we know, and (3) what we should do.

Is "aging" an issue? Clearly it is! The information from worldwide research, operational experiences, and regulatory mandate says so. Is "aging" a problem? We will only know with research, understanding, and/or through proven predictive or surveillance techniques.

What do we know about "aging"? Aside from technical details, here are a few thoughts:

- sufficient evidence exists to convince us that aging is a concern.
- sufficient understanding exists that good engineering judgements are possible.
- aging is a "sensitive" subject; it must be approached carefully.
- there is a reluctance to achieve and defend the state-of-the-art.

*Key Sandia contributors are Larry D. Bustard, Roger L. Clough, and Kenneth T. Gillen.

What do we do about "aging"? Clearly we attack and solve the "problem"! Again a few thoughts.

- coordinate the effort; the Workshop is a good start.
- stop writing words on "need to be safe," and concentrate on the prerequisite research, evaluation, and thinking necessary.
- share a common pool of knowledge; eliminate "proprietary" barriers.
- eliminate the artificial "electrical," "mechanical," "structural" distinctions.
- consider a buffer period (say, five years), free from premature standards and regulation.

Our involvement with "aging," through NRC-sponsored research, has provided certain insights into the historical, present, and future perspectives. We share these in this Workshop paper.

EXPERIENCES

The discussion in this paper is based, in large part, on the author's association, as principal investigator, with the Qualification Testing Evaluation (QTE) Program being conducted at Sandia for the Electrical Engineering Branch, Division of Engineering Technology, USNRC. A brief review of that program is in order.

The objectives of the Qualification Testing Evaluation (QTE) Program are: to obtain data needed for confirmation of the suitability of current standards and regulatory guides for Class 1, safety-related, equipment; to obtain data that will provide improved technical bases for modifications of these standards and guides where appropriate; to establish data-based and standardized test methodologies for equipment qualification programs; and to support the NRC licensing process with qualification expertise and test capabilities.

Currently, these objectives are embodied into research activities covering seven major activities.

TASK 1: Provide assessments of accident (environmental) testing methodologies. Representative issues pertinent to these methodologies include combined environments synergism, dose-rate effects, superheat, post-DBE acceleration, humidity, thermal shock, and oxygen depletion.

TASK 2: Establish the accident radiation signature, evaluate the adequacy of radiation simulators, and establish calculational and testing methodologies.

TASK 3: Evaluate the aging processes of safety-related materials, components, and equipment, and establish accelerated aging methodologies.

TASK 4: Evaluate the vulnerability of safety-related equipment based on the experiences from the Three-Mile Island Unit 2 accident.

TASK 5: Establish contact with international research groups, coordinate with other research efforts, and pursue mutual-benefit programs and exchanges.

TASK 6: Define the long-term need for research programs and test facilities and prepare design drawings and facility equipment specifications.

TASK 7: Evaluate the adequacy of separate/sequential seismic testing with regard to overall equipment qualification.

The program has generated numerous reports over the past several years, and some 10-12 are currently in progress. The interested reader can contact the author for specific information.

The program also relates to the Workshop scope and programmatic interests in that (1) research is already underway on several aging issues and methodologies and (2) aging is being addressed, to the extent the state-of-the-art will allow, in various tests of materials and equipment. The tasks and subtasks listed below illustrate some instances where aging "impacts" are felt in the QTE program.

TASK 1 ACCIDENT METHODOLOGIES ASSESSMENTS

A) Facility Capability Improvements
B) Components/Materials Methodology Tests
C) Equipment Methodology Tests
D) Joint US-French Tests
E) Follow-on Testing, Interconnections (EPA)

TASK 3 AGING METHODOLOGIES ASSESSMENTS

A) Chemistry/Extrapolation-of-Data of Elastomeric Materials Research
B) Humidity Aging Acceleration
C) Fire-Retardant Aging Research
D) Multi-Photon Ionization Laser Technique
E) Ambient-Aged Equipment/Materials Acquisition and Research
F) Material-Age/Voltage-Breakdown Correlation Study
G) Develop Aging Methodology Handbooks

H) Extend Research to Other Materials/Stresses
I) Thermally-Stimulated Current Technique
K) Plant Ambient Environment Study
L) Electronics Aging Evaluation and Initial Research
M) Aging of Motors Study

THEME

In keeping with the goal of the Workshop, a broad view is appropriate. It is unnecessary to reference a litany of technical reports which describe aging phenomena and methods. Most of us are familiar with these already, or can get access to them if desired.

Rather, then, we need to discuss the programs and options to arrive at an understanding of "aging" in the broadest sense. Even the word "aging" should be eliminated from this discussion (but I will not either), because it carries a narrow meaning for most of us. "Time-related degradation" is the phrase I propose as a substitute. "Aging" has been too long associated with safety-related equipment qualification issues, and that is a very small part of the overall concern for nuclear plant safety. Think of the issue this way: "will the plant operate safely when called upon to do so, given that the plant and all of its equipment is growing continually older?"

So we will concentrate on "aging" in the broadest sense: (1) as an issue, (2) what we know, (3) what we should do.

AGING: AS AN ISSUE

We would all agree that everyone and everything "gets older." "Gets better?" That is a separate issue!

We also need to distinguish between "issue" and "problem". The rhetorical question is, "is aging an issue?" The answer is, "clearly it is." I offer these observations as proofs:

- worldwide research is underway.
- operational experiences (e.g., licensee event reports) indicate "failures" even in "normal" operation.
- regulatory and industry mandates call for addressing the issue, including:

    1. Regulatory Guide 1.89
    2. NUREG-0588
    3. Several Rulemakings (in progress)
    4. Daily licensing decisions
    5. IEEE Standards

If you accept these proofs, then aging is an issue. The rhetorical question them becomes, "is aging a <u>problem</u>?" The answer is not so clear; but it is probably safe to say, "yes <u>and</u> no." I offer these observations.

Prudent individuals could conclude that aging is a "problem", just from the limited research available from which you must conclude that we do not fully understand. A lack of thorough understanding generally implies a problem.

At least for some important plant equipment, the concern for "common-mode" failure is an overriding uncertainty. It is not sufficient to know that a piece of equipment works (with some reliability) in normal environments. What must be determined is whether a (common) degradation has occurred to such an extent that a superimposed, non-normal, environment is sufficient to fail the equipment. Thus reliability data, taken in normal environments, is largely unusable for a superimposed accident situation; it amounts to a conditional probability of failure, for which data does not usually exist.

A further uncertainty exists because aging cannot be (usefully) investigated in real time; you cannot wait! That implies an "acceleration" of age, which, of itself, clouds the results. The concern is now doubled. Is aging actually occurring in real time, and is the acceleration actually realistically producing real-time aging?

Is aging a "problem?" We will only know with research, understanding, and/or through proven predictive or surveillance techniques.

<u>AGING: WHAT WE KNOW</u>

The available evidence suggests a concern. A myriad of examples exist: dose rate effects, synergistic effects, non-Arrhenius behavior, non-linear behavior, embrittlement, corrosion, and so on. But always, the "research" results must be tempered by good "engineering judgement": test environment applicability, range of variables, overdesign, redundancy, and so on.

I submit that good engineering judgements: (1) are possible, and (2) are being made. Industry is testing, the NRC is licensing, and plants are operating. QED.

But there are problems we need to come to grips with:

- the rapid evolution of the industry (e.g., new, improved products) <u>implies</u> a difficulty in learning from past history. The difficulty in extrapolating the data is real.

- data already exists, more research will produce more results. Be assured that this data must influence future approaches, decisions and directions. It is a fact of life; we adjust based on the latest information.

- in many cases we do not seem to know the actual plant environments and actual plant usage or installation. How reliable can an "aging" estimate be then.

- (over) conservatism is a means for compensating for uncertainty. But more emphasis on realism in equipment/environmental specifications is needed, both conservative and non-conservative.

Aging is a sensitive, "black-magic", political subject. Consider these additional observations, if you will.

- The world equates "ignorance" or "uncertainty" with "problem".

- Simple in concept, very difficult in practice. A real-time event, but you cannot wait to observe in real time. "Acceleration" techniques must be used.

- Everyone must understand that research results must impact the currently operating plants.

- Research "results" are sometimes applied concurrently with their evolution in licensing decision, without benefit of extensive confirmatory research and peer review.

- A little of this, a little of that (corollary to the "Eye of newt, and toe of frog, wool of bat, and tongue of dog," with apologies to Macbeth, IV, i, 14, 15).

  What are <u>all</u> the environments, how do you accelerate <u>all</u> the environments, how are <u>all</u> the environments coupled?

- Someone will suffer; some manufacturers will be driven out of the market. Because the methodologies developed can be used as "screening" tools to eliminate poorer equipment.

It is also apparent, to me at least, that a large problem with the industry and regulation is a general reluctance to state and defend the state-of-the-art as it exists at any given point in time. It takes these forms and, no doubt, others:

- Not admitting to the issue somehow implies to many that the issue doesn't exist; a "hide your head in the sand" approach.

- "Weak-link" engineering seems to be the rule, not quality products based on precise and realistic design specifications.

- Regulatory "persuasion" and retrofits are real.

- The NRC pushes too hard and fast
  and
  not hard or fast enough.
  What is the right balance?

- Cost is a concern.

- A philosophy of "don't fix it if it ain't broke" permeates, but doesn't apply in common-mode failure issues.

AGING: WHAT WE SHOULD DO

This a summary section, although a conclusion section follows and hints of "what we should do" were scattered throughout the previous sections. Aging is a large multifaceted issue. No one has tried to come to grips with it on a global plant scale yet. Since the problem is large, the solutions are probably radical. I don't have the ultimate answer, but consider the thoughts in this section.

What do we do about "aging?" Clearly, we attack and solve the "issue" or "problem", whichever it is. That is not meant to be flip or dramatic. It is just an admonishment that we quit "talking" the problem and "attack" it. The available data is insufficient to produce absolutely defendable solutions in all cases. If it were, we would use it and "aging" would be solved.

Here are a few thoughts:

- Fragmented research efforts do not produce best results. Coordinate the effort. The workshop and program are good starts.

- We need to eliminate the adversary approach and have all affected groups cooperate.

- Narrow the effort by selectively prioritizing the issues.

- Need multi-pronged approach

  1) research on basic understanding
  2) research on engineering understanding
  3) collection/evaluation of actual-use data
  4) develop surveillance procedures, to provide long-term assurance of qualification and functionability (after all, we will never know a priori for sure)
  5) sacrificial/replacement items in plants; plan ahead

I would also ask you to consider these (more radical) thoughts:

- Stop writing words on "need to be safe", and concentrate on the prerequisite research, evaluation, and thinking necessary.

- Share a common pool of knowledge; eliminate "proprietary" barriers. What good is market advantage, if the market disappears?

- Eliminate the artificial distinctions between "electrical", "mechanical", "structural". The aging issues criss-cross these politicized boundaries.

- Bring together the worldwide research effort and reactor engineering efforts (even at the possible expense of "competition").

- Consider an "aging casebook" approach in equipment qualification to provide a legitimate, unbiased, peer review process.

- Consider a buffer period (say, 5 years?), free from premature standards and regulation. Give research and engineering a chance in a brick house, not a glass house.

CONCLUSION

Several areas of investigation have been mentioned. But it is clear that a broader view is necessary. If there is no other immediate result, I would recommend a comprehensive look at aging. Perhaps a matrix approach with equipment, environments, knowns, unknowns, degradation mechanisms, etc. From this table, we would be able to comprehensively recognize the issues and necessary work areas. We can then proceed to prioritize, perform cost/benefit evaluations, and proceed. There is nothing profound there, just the usual engineering logic approach.

CONDITION MONITORING

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ABSTRACT

This paper presents an alternative method for addressing aging of equipment in nuclear power plants. Condition monitoring is the term used for an empirical surveillance technique by which the damage caused by environmental stress imposed on equipment is periodically assessed. The assessment of existing damage versus the undamaged and failed conditions allows for extrapolation of remaining life. A discussion of currently used Arrhenius and statistical techniques used to calculate useful life and their relationship to condition monitoring is given in the paper. A simple example of the condition monitoring method and the advantages of the method are discussed.

Condition monitoring is a means of quantifying the natural aging of equipment. It is applicable for mechanical as well as electrical equipment. Although the technique is not an equipment qualification method, its use will provide technical justification for any aging assumptions applied during equipment type testing.

The condition monitoring technique is a new method based on measuring degradation of the component or components with the most limiting lifetime as determined by Arrhenius, i.e. activation energy, methodology for the equipment under consideration. Equipment located in the worst case environment considering all stresses (radiation, temperature, humidity) is selected for monitoring. The critical components are removed, and new components installed. The used components are tested for degradation effects and the results graphed. A remaining lifetime for the critical component is calculated and a new equipment surveillance period established based on the remaining lifetime. Cumulative information is gathered for each successive condition monitoring test and a degradation line drawn which represents a worst case degradation for any similar component in the plant since the component environment for the equipment under consideration was defined initially as worst case.

INTRODUCTION

The ability of equipment in nuclear power plants to operate during normal, abnormal, and accident conditions is demonstrated by equipment qualification. Qualification methods generally consist of 1) tests which subject equipment to the most severe adverse environments expected to occur at the installed equipment location and/or 2) analyses which consider the mechanisms and values of hazardous environments and their potential for causing unacceptable failures of equipment.

Typical adverse environmental parameters specified in testing or analysis include: temperature, pressure, humidity, radiation, and chemical spray. In addition, normal degradation of equipment functional characteristics are evaluated by means of "aging" tests or analyses.

A major problem with assessments of equipment in operating plants and plants near completion is lack of "aging" testing or data to support the ability of old equipment to operate for its design life. The method used to evaluate the equipment consists of calculating the ability of organic materials to function in a given environment over a period of time. Resistance to ionizing radiation, thermal rating, and corrosion resistance of materials are often evaluated to determine the material status after exposure to normal and adverse environments. The evaluation may be difficult since exact materials of construction are often unknown to both the utility performing the assessments and the manufacturers of the equipment.

A further difficulty exists in the lack of adequate environmental parameter data. The environments calculated may be averaged or given as limiting conditions. The relationship between the calculated and actual values is not well-defined. However, this actual environmental data is critical to accurate testing or analysis of equipment degradation over time since these parameters are used as input to qualification determinations.

BACKGROUND

There is a great deal of speculation about the validity of the methodologies used in accelerated age testing and in aging analyses. Arrhenius and other techniques are, at best, gauges for stress degradation and do not predict accurate lifetimes. Studies have shown that the time periods for a good approximation of stress degradation as determined by accelerated techniques is on the order of a few years[3].

Condition monitoring is an attempt to address these difficulties by looking at the bases for aging assessments, tracking the validity of aging techniques over time, restating the degradation equation based on empirical data, and forecasting the ability of equipment to perform at dates in the future through evaluation of present degradation as compared to past history. Condition monitoring is a synthesis of accelerated aging analysis, age testing, reliability and maintainability, and surveillance.

Current monitoring techniques are related to reliability and maintainability (R&M), preventive maintenance (PM), and accelerated qualification testing based on the Arrhenius model. Examples of such techniques are vibration monitoring to detect rotating machinery bearing wear and partial discharge testing of large motors. These techniques detect wear and forecast incipient failure of components. Condition monitoring (CM) is merely an enhancement of existing techniques which can be incorporated into present nuclear plant surveillance and preventive maintenance procedures.

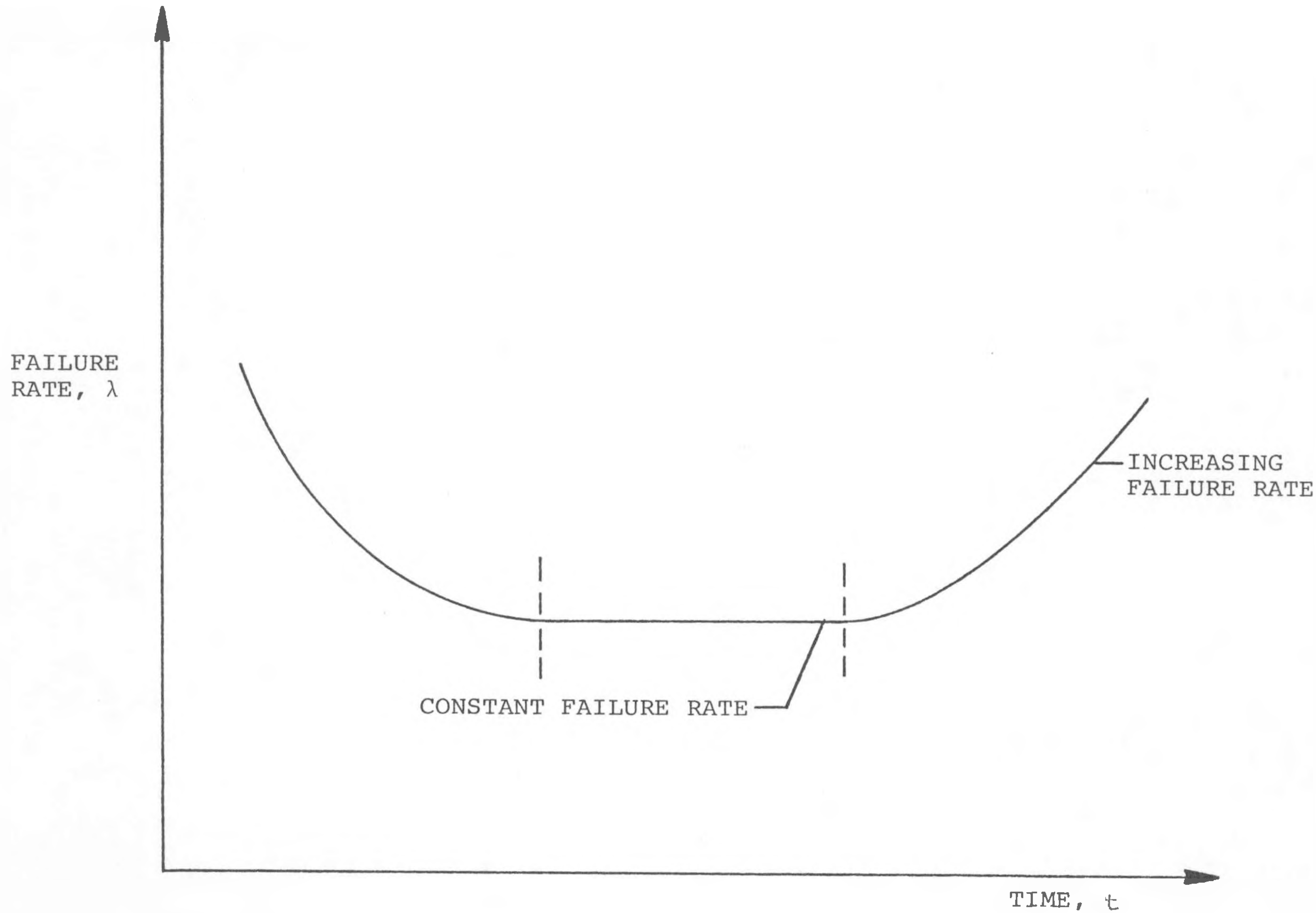
The goal of an integrated condition monitoring program is to characterize the degradation of components over time and extrapolate that degradation into the future to predict the failure point of the component. This is accomplished by periodic measurements and tests of weak-link degrading components or by statistical evaluations of constant failure rate components.

The following paragraphs give background on present R&M, PM and Arrhenius methodologies and their relationship to condition monitoring.

Reliability

Present programs which use R&M as a basis, are designed to ensure operability by means of redundancy of systems or components. A reliability goal is established and failure rates for components are determined. An appropriate number of components is specified in order to achieve the reliability goal with statistical confidence.

The constant failure rate portion of a reliability curve is determined for each component by evaluation of historical failure data. The reliability curve, also known as the bathtub curve, has three portions - decreasing failure rate, constant failure rate, and increasing failure rate (See Figure 1). Burn-in testing is used to eliminate the infant mortality portion (decreasing failure rate) of a population of components. Once the constant failure rate regime is reached, the device or system is presumed to function for its design lifetime. Replacement or maintenance is then typically performed according to a schedule recommended by the component or device manufacturer. It is presumed that the increasing failure rate portion of the curve is never reached. The replacement or maintenance schedules are determined statistically for a specified environment based on the mean time to failures (MTTF), mean time to repair (MTTR), degree of redundancy, and the reliability goal.



"BATHTUB" CURVE OF FAILURE RATE

FIGURE 1

The major drawbacks for CM based on R&M alone are: 1) a large data base of knowledge must exist on component and device failure rates and 2) the environmental conditions under which failure rates have been established must be well known. A large amount of data does exist for electronic components, most of it in a form which recognizes increasing failure rates as a function of increasing temperature. Data on electromechanical devices is very sparse, however, and it is doubtful that an adequate degree of confidence can be gained from such data. Environments in nuclear plants may include radiation and temperature stresses which may not be accounted for in failure rate data. If these stresses are significant, published failure rates may be invalid and lifetimes could be shorter. R&M models, however, are very useful for devices having electronic components, for devices which cannot be easily tested for degradation, and for devices which tend to fail catastrophically rather than degrading over time.

Preventive Maintenance (PM)

Preventive maintenance models for precluding failures may be classified as statistical (R&M), quasi-statistical, or surveillance. The R&M model was discussed above. Quasi-statistical models are replacement schedule models based on non-technical, non-quantifiable assumptions, e.g. rules-of- thumb, operating experience, prior practice with similar equipment, etc. Surveillance models include degradation testing, in-situ observation, on-line monitoring, etc. True surveillance models include analog measurements and degradation trend assessments. Determination of function alone by means of proof or periodic testing cannot predict function in the future and should not be considered surveillance. Functional trend testing can be used to predict degradation only if quantitative, time-variant values are recorded in addition to pass/fail data.

Quasi-statistical models, although conservative (due to their relationship to manufacturer guarantees), have questionable technical bases. These models usually take the form of recommendations and are often merely protection for manufacturers in the form of guarantees of performance. It is often extremely difficult to find any testing, statistics, or other quantifiable justifications for quasi-statistical models and, consequently, their use should be minimized.

Surveillance models require long-term committments by users and accurate recording of test results and values. Rather strict schedules need to be established for surveillance and results must be evaluated periodically to determine the status of the equipment. On-line or in-situ testing in support of surveillance may not be rigorous enough to detect small changes in critical parameters and often do not account for differences in equipment and in-situ test environments.

Use of surveillance is especially appropriate for devices which have no statistical data base for failure rates. Surveillance is also indicated for equipment which degrades over time and fails at a given degraded value.

Arrhenius Model

The Arrhenius equation[3] is given as:

$$L = Be^{\phi/kT}$$

where
L = time to reach a specified endpoint or lifetime
B = constant (usually determined experimentally)
ϕ = activation energy (eV)
k = Boltzmann's constant (0.8617×10^{-4} eV/K)
T = absolute temperature (K)

A relationship can be developed which shows that degradation at a future time can be determined from data gathered in accelerated testing. The above equation can be modified to give:

$$\ln(t_s/t_a) = \frac{\phi}{k}\left(\frac{1}{T_s} - \frac{1}{T_a}\right)$$

where
t_s = service time
t_a = accelerated testing time
T_s = service temperature
T_a = accelerated testing temperature

This equation is referred to as the life line equation and may be represented by graphing time versus temperature. The line when graphed on semi-log paper is representative of the various time/temperature combinations which produce failure. The slope of the line is ϕ.

It should be noted that the Arrhenius equation is an empirical formula based on a single, endothermic reaction. It is used to predict the degradation of material and components because of its simplicity and the fact that a logarithmic, thermal degradation of material properties has been observed in experiments with both materials and components[3,4].

Another form of the Arrhenius equation[3] which may be used to characterize the fraction of useable resource remaining at time t when subjected to temperature T is:

$$\ell n\left(\frac{c}{c_o}\right) = A\ t\ e^{-\phi/kT}$$

where
c_o = initial concentration of reacting material (resource)

c = endpoint concentration of reacting material (resource)

This form of the Arrhenius equation states that degradation is characterized by a time and temperature relationship which results in a loss of a critical functional resource. The goal of accelerated testing is to induce the failure value of critical resource in a relatively short time when compared to actual time to failure. Figures 2 and 3 show the relationship of the forms of the Arrhenius equation. It should be noted that Figure 2 is locus of failure points shown on Figure 3.

Other stress/rate functions which include additional environmental stresses besides temperature have also been developed[17,20].

The inherent assumptions in the Arrhenius model make it useful as an approximate measure of life, at best. The initial choice of an end of life point is based on the premise that the material or component is useless when an estimated parameter reaches a limiting value. The choice of parameter may or may not be indicative of the primary function of the item. For example, partial loss of elongation is often specified as the limiting parameter for organic materials used as electrical insulation, seals and gaskets, etc. The value chosen as a failure point may vary from 50% loss to 70% loss, depending on the judgment of the engineer specifying failure criteria. Besides the variation in endpoint specification, a parameter such as loss of elongation, may or may not be directly related to functional (e.g. electrical) performance. The crucial aspect is that the failure point and mechanism be selected carefully and be related to equipment function.

Also inherent in the Arrhenius method is the assumption of a single-reaction, logarithmic degradation model, which can be graphed as a straight line on semi-logarithmic paper. This is a questionable approximation of actual degradation and causes gross errors when lifetimes are extended far into the future. For complex materials, more than one reaction occurs with differing specific heats resulting in complex activation energies for the chemical changes which occur in the material or component.

The Arrhenius method is often used to establish a calculated lifetime for equipment used in nuclear plants. Where observed activation energies are unavailable for the specific item under consideration, a conservative value of activation energy will be chosen. In most cases, a temperature corresponding to the maximum temperature expected to occur during the equipment service life is established as the service temperature for the material or component. It is unlikely that the equipment will have either an activation energy or service temperature which is accurate. However, the Arrhenius technique does serve to identify weak link components when considering an assembly of components, and can be used to predict minimum surveillance times for ensuring the continued integrity of the assembly.

ARRHENIUS MODEL

CLASSICAL & LIFELINE

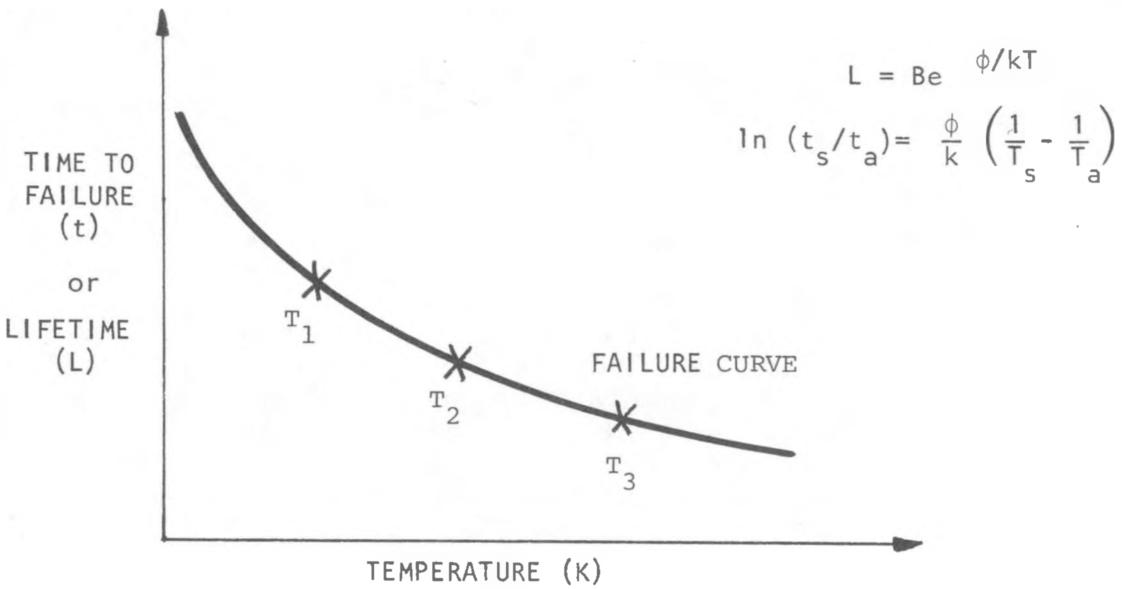


FIGURE 2

ARRHENIUS MODEL

RESOURCE CONCENTRATION

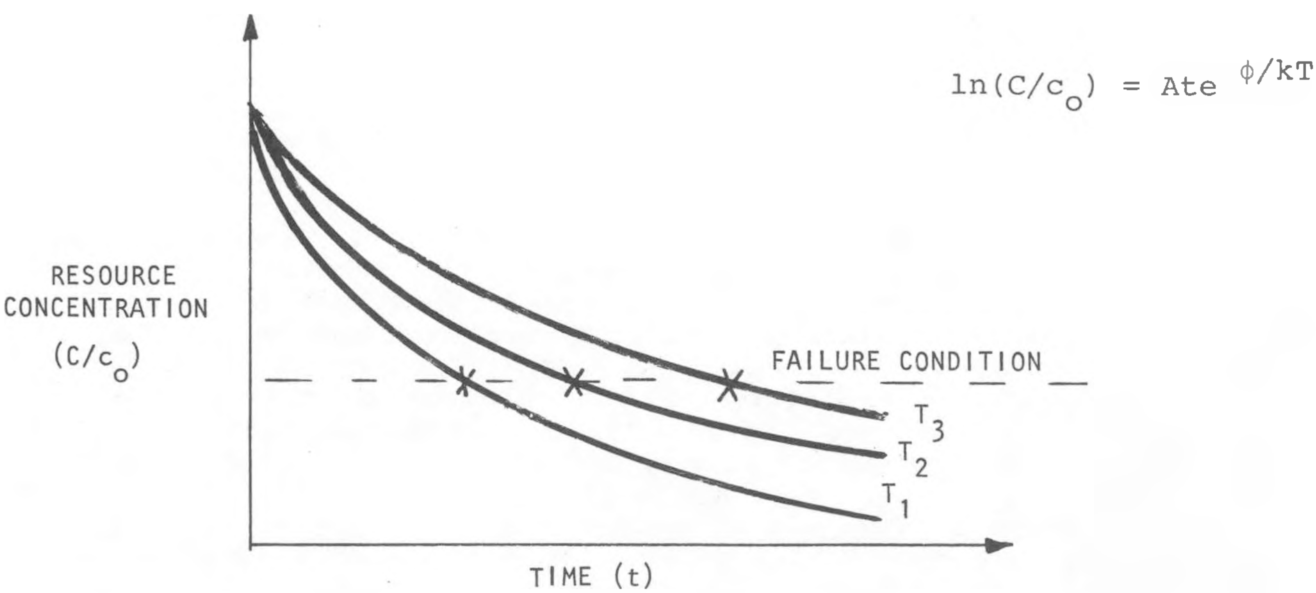


FIGURE 3

CONDITION MONITORING (CM)

Condition monitoring is an integrated program which uses the best features of the models above to track and predict the ability of equipment to function. Figure 4 shows three models - simple wear, Arrhenius (logarithmic), and statistical - which are basic to the condition monitoring program. The dotted failure line indicates the point at which the critical parameter causes failure of the equipment.

Logarithmic models (e.g. Arrhenius) track critical parameter degradation by testing at elevated conditions for a specified period of time to reach an end-of-life condition.

For example, a loss of elongation of 50% may be specified for an end of life condition for an insulating material. A temperature of 100°C may be selected as the aging temperature. The resulting time to effect a 50% loss at 100°C is recorded. Different temperatures are then specified and the time to reach the same endpoint (50% loss of elongation) is recorded. The resulting values are graphed. A regression line (on semi-logarithmic paper) is drawn to approximate the degradation versus time. The slope of the degradation line is the activation energy of the material.

Simple wear models are a limiting case of the logarithmic model which are linear in time. Simple wear models are more conservative than logarithmic models, since the slope of logarithmic models decreases with time.

The statistical model is based on the bathtub curve with the decreasing failure rate portion eliminated. An unacceptable component failure rate is presumed to occur at a point corresponding to the device reliability goal (with other component failure rates presumed to be in the constant failure rate region). This assumption leads to a tracking of the weak link component failure rate.

Stress History

The models discussed above are used to describe the life cycle of materials and components. Each material or component which exhibits degradation over time exhibits a history of all imposed stresses. The effects of synergisms and unknown aging effects will be included in this stress history of the materials or components assessed. Redefinition of environments merely causes a different stress history to be imposed on the materials or components. Surveillance periods and testing can and should be defined based on this actual stress history. Only the initial surveillance period should be determined by the assumed stress history of accelerated testing.

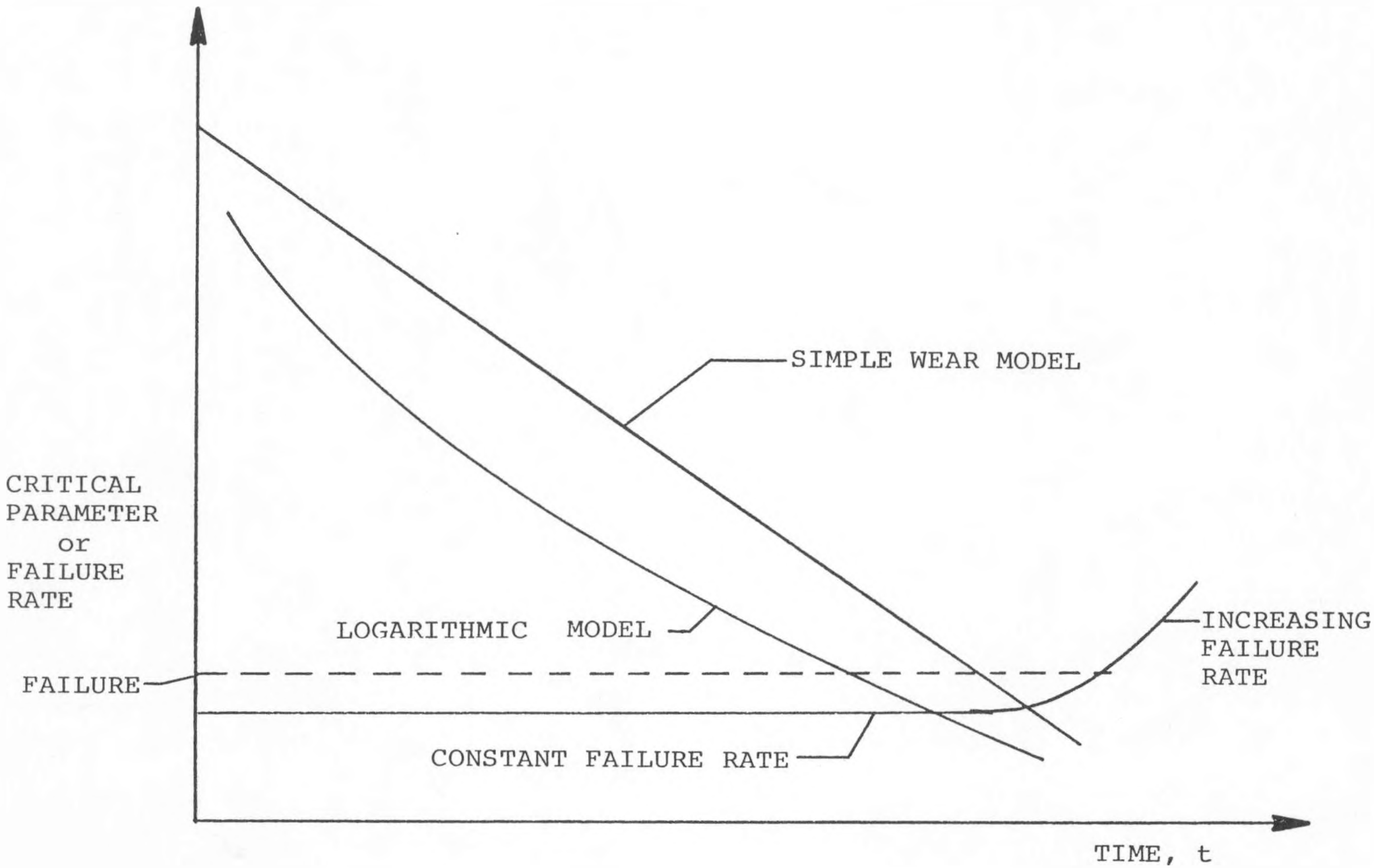

SIMPLE WEAR MODEL
CRITICAL PARAMETER or FAILURE RATE
LOGARITHMIC MODEL
INCREASING FAILURE RATE
FAILURE
CONSTANT FAILURE RATE
TIME, t

FIGURE 4

CM Program Implementation

A. Identification of Degrading Resource

The initial step in any CM program is the definition of the material or component property to be measured. This property should degrade over time and be related to performance, although not necessarily a direct performance property. Both R&M and Arrhenius models use failure relating to a single degraded resource (failure mode) as the basis for determining the end-of-life for the material or component.[6] Statistical models generally measure go or no go conditions. Logarithmic models measure weaknesses in a particular material or component. Use of any other trending model or predictive technique also requires that a failure mode or modes be established.

Specification of a degrading resource is the most important aspect of condition monitoring. The choice of degrading resource can be made by assessing the function of the material or component and looking at the capability of detecting the reduction of the critical resource which produces failure. Often a degrading resource which contributes to loss of function rather than the loss of function itself is specified. For example, elongation of electrical insulation as a percent of original elongation is frequently specified as the degrading resource. Although loss of elongation may not correspond directly to electrical failure, it is a measure of the condition of the aging insulation and can be used to predict incipient failure.

B. Rate Assessment for Degradation Models

If the logarithmic model is selected, the failure point value is approached at a rate which can be derived from the activation energy of material or component. The activation energy can be determined by so-called step-stress testing.*[3,4] This method shows that an equivalent or lesser percentage of deterioration occurs over identical time periods for increases in stress (temperature) indicating that the continuous degradation model can be used to estimate the available lifetime remaining by either a simple wear or exponential decay model. (It is important to note here that the simple wear model is the worst case logarithmic degradation versus time to an eventual failure point.) In other words, deterioration may be characterized as a constant or decreasing rate process.

If a statistical model is selected, increasing failure (IFR) rate can be approximated from published data on component lifetimes. Some degradation rate assessment testing may also be performed at spaced intervals to test the validity of lifetime specification for components.

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*Other techniques such as Thermogravimetric Analysis (TGA) testing are used to calculate activation energies for materials but they are of doubtful use. TGA measures weight loss caused by controlled heating to determine activation energy. However, some failure criteria must be used to establish the initial test point.

The mean time to failures (MTTF) can be calculated from historical data as it exists in industry data banks, e.g. NPRDS (Nuclear Power Reliability Data System, GIDEP (Government Industry Data Exchange Program) etc. MTTF can also be determined from plant maintenance records if the records are comprehensive enough. A simple statistical test of recorded failures can be used to determine whether the device has reached the increasing failure rate region. The statistical measure can be stated in the form of,

$$\frac{x}{t} = P$$

where x = number of failures
t = surveillance period
P = failure rate

The calculated statistical measure can then be compared to the manufacturer's stated constant failure rate and the significance of the measure determined. If the increasing failure rate is significant, replacement or further degradation testing should be considered. Other tests for IFR have been suggested which may be used instead of the simple relationship above.[3] Both degradation rate and IFR assessments require degradation testing to determine applicable degradation data. IFR assessments do not require surveillance revision, but will require replacement of degraded components at the earliest possible time. No further definition of surveillance period or determination of remaining life is required.

C. Surveillance Period Determination

Based on degradation rate evaluations or the simple wear model, a surveillance period can be established. Normally, this period should be between the 50% point of failure deterioration and the 80% point in order to allow for inadequacies in the Arrhenius technique. In general, maximum defined surveillance periods should be on the order of 10 years where accelerated aging has determined useful life greater than 10 years. It is doubtful whether valid extrapolations can be made to lifetimes greater than 10 to 20 years.[3]

The Arrhenius model is based on an exponential degradation assumption and predicts the shape of the degradation curve even though the accuracy of the assumptions may be in question. Although catastrophic failure may follow an exponential degradation, it is precluded from consideration by structuring condition monitoring to track degradation to a point which preceeds catastrophic failures. For instance, the 80% point of failure parameter maybe chosen to allow sufficient margin between the replacement degradation value and catastrophic failure value.

D. Determination of Remaining Life

Once the degrading resource and degradation rate or MTTF have been established, it is a simple matter to estimate life. A duplication of the degraded resource measurement as determined in accelerated testing, a recording of results, and a comparison with the initial value and failure value gives a percentage life remaining for testable devices. When the percentage of life used is compared to the surveillance period, a simple wear (or exponential decay) model can be used to assess remaining life in the material or component. The surveillance period can be shortened if more than a given percentage of life has been expended, the existing surveillance period can be maintained if the predicted percentage of life remains, or the surveillance period can be extended if degradation is less than expected.

E. Testing

At the specified surveillance time, devices and components are evaluated for degradation. Components which are to be statistically tested are compared against the MTTF to determine wearout status, i.e. whether the increasing failure rate region has been reached. If necessary, the components are replaced or physically tested for degradation. If wearout appears imminent, then near term replacement should be considered.

Equipment located in worst case environments are selected for degradation testing to ensure maximum degradation. For example, if a nuclear plant has a large number of similarly specified valve operators, the weak link component in a single valve operator in the worst case environment should be removed. All components in the remaining valve operators are, by definition, less degraded than the weak link item in the worst case environment.

Components to be functionally tested are removed. These components are previously-defined weak-link components whose degradation has been shown to progress at a faster rate than other components within the device. The weak-link components are replaced with new spare components. It is important that the removal procedure be performed on working rather than failed components since catastrophic failures often destroy evidence of degradation. Replacements are minimal since only one or two samples of like items are removed for physical testing.

After removal, the weak link component is tested for degradation. The test is a duplication of the original test used to establish the failure point of the item. (The test may be destructive or non-destructive in nature.) The test results are then recorded and compared to specified initial values and failure point values. The test results should be graphed versus time so that an empirical degradation curve may be determined. (The results may also be compared to accelerated aging assumptions to determine the validity of the initial assumptions.)

F. Life Evaluation

The percentage of remaining life is checked versus the prior surveillance period. An assessment of the amount of life remaining is made and the surveillance period confirmed or a new one established. If the testing indicates an unacceptable level of degradation, replacement of the component or device will be considered and assessments of other similar components or devices specified. A further assessment of near weak link components having slightly longer lifetimes than weak link components may also be performed to further substantiate the amount of degradation measured in the weak links and to aid in the determination of an optimum surveillance period.

Example (See Figure 5)

A nuclear plant has 10 type B transmitters installed in the plant, 5 inside containment and 5 outside containment. All component parts in the transmitters have been ascertained to be essentially equal (function is identical). It has been found through Arrhenius evaluation that an apparent activation energy of the limiting component is .7 eV and that the calculated life at 45°C is 6.69 years.

From a literature search, it is determined that degradation failure modes for electronics (IC's, resistors, diodes, etc.) are characterized by drift from a specified value (The manufacturer-specified initial value is used as the specified value). The manufacturer of the specific component has also stated that 10% drift is the maximum allowable for circuit function to be maintained.

A surveillance period of 5 years (or equivalent in number of outages) is chosen. After the 5 year period, the transmitter printed circuit board (PCB) which contains the electronics is removed and a spare PCB installed. The naturally aged printed circuit board is tested for component drift. (Since all the transmitter electronics are contained on the PCB, several components may be tested for drift.) It is found that the maximum drift of the component from the specified initial value is 6%.

Since the initial value was given as ± 2%, a maximum possible drift of 8% has occurred. If one assumes simple wear, then the portion of useable life remaining is:

$$\frac{12\%\ \text{(maximum range)} - 8\%\ \text{(maximum drift)}}{12\%} = 33\%$$

The new expected life is equal to the surveillance period plus one third of the initial lifetime determination or 7.23 years. A new surveillance period of one year or one outage could be established or, since simple wear is the most conservative approximation, a new surveillance of two years may be appropriate.

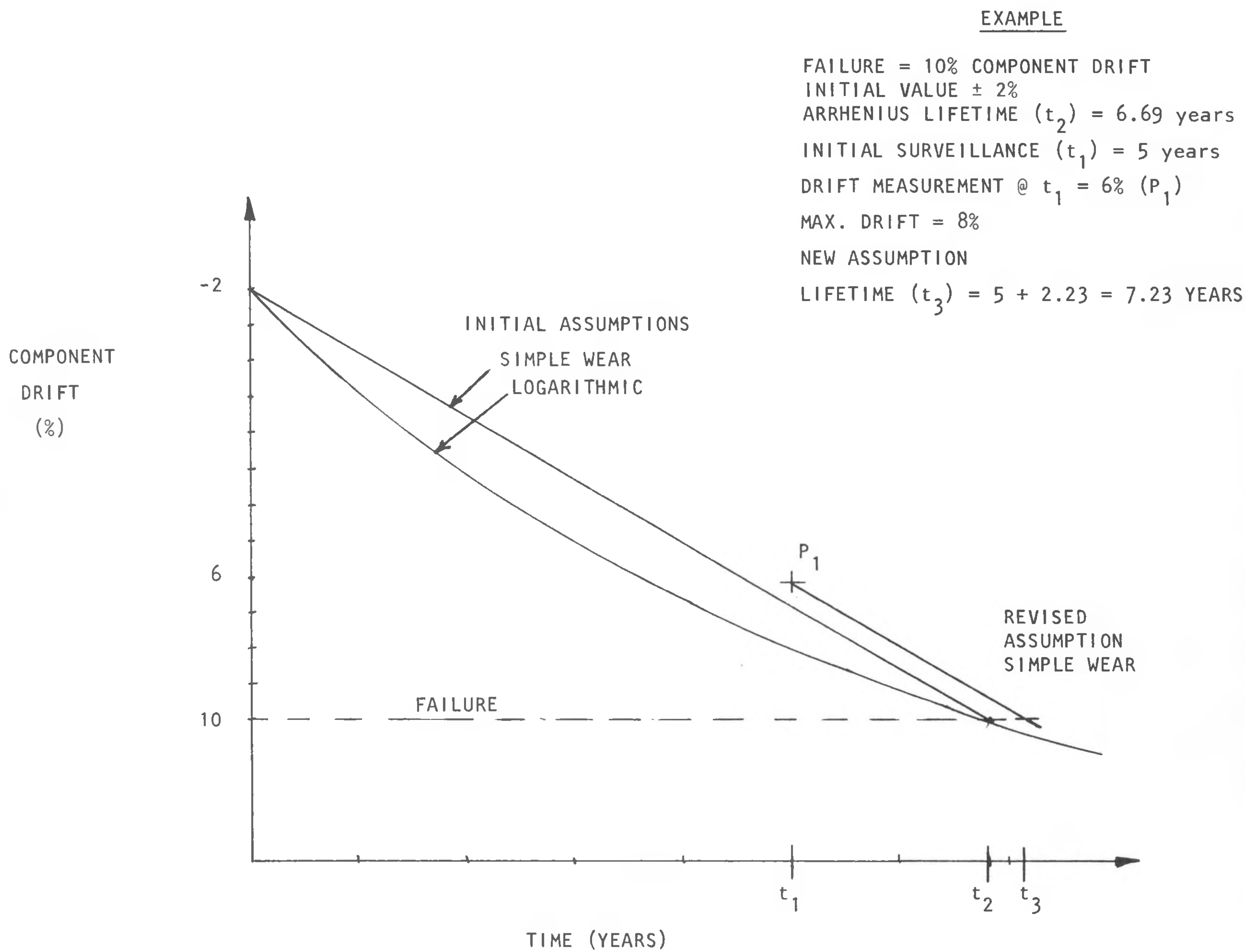
EXAMPLE
FAILURE = 10% COMPONENT DRIFT
INITIAL VALUE ± 2%
ARRHENIUS LIFETIME (t_2) = 6.69 years
INITIAL SURVEILLANCE (t_1) = 5 years
DRIFT MEASUREMENT @ t_1 = 6% (P_1)
MAX. DRIFT = 8%
NEW ASSUMPTION
LIFETIME (t_3) = 5 + 2.23 = 7.23 YEARS
INITIAL ASSUMPTIONS
SIMPLE WEAR
LOGARITHMIC
COMPONENT DRIFT (%)
-2
6
10
FAILURE
P_1
REVISED ASSUMPTION SIMPLE WEAR
t_1
t_2
t_3
TIME (YEARS)

FIGURE 5

The above example is the most conservative approximation of remaining life since the measurement taken at the 5 year surveillance point gave a value of drift which was less than expected. The revised estimate line was drawn parallel to the initial simple wear assumption line. Equally valid assumptions of the new degradation curve may be shown by redrawing the simple wear line through the 5 year surveillance point and the initial value of drift (-2%) or redrawing a logarithmic curve to approximate the data. These latter two assumptions are less conservative in this example but become more justifiable over time as long term trends become obvious within a few surveillance periods.

If, however, the initial surveillance measurements fall below that expected, one of the latter assumptions should be used to predict lifetime since, in this case, they give more conservative answers than assuming that the initial simple wear slope is correct.

Advantages of CM

1. Activation energy is used only to determine initial periods not for life determination.

2. Replacement times should be at least as long as calculated by Arrhenius techniques and probably much longer.

3. All components do not have to be tracked.

4. Only worst case normal environments must be defined.

5. Aging data is estimated conservatively based on actual measurements without requiring unnecessary replacement.

6. As periodic replacements of weak link components are performed, aging diversity is introduced into the plant.

7. Although the technique is based on thermal degradation, any other degradation mechanism effects will be identified through failure mode testing. Capability to survive until the next period will be demonstrated since the CM method is based on the effects of actual degradation rather than assumed causes of the degradation.

8. Poor and outstanding performers for certain equipment types will be identified. This information can be used for purchase of new equipment and can also be employed to encourage manufacturers to improve their products.

9. As more data is developed, failures and successes can be viewed from both an equipment performance and reliability standpoint.

Disadvantages of CM

1. CM may identify problems in the existing surveillance techniques requiring replacement of equipment at an earlier date than previously envisioned.

2. CM will require periodic removal and replacement of weak link components rather than in-situ proof testing.

Conclusion

A well-structured CM program enhances utility knowledge of equipment operational status. It provides a technical basis for preventive maintenance and surveillance. It can eliminate unnecessary replacement costs. CM also allows aging assessments and approximations of useful life which are independent of stress environments since stress history is measured rather than assumed. CM is a life cycle program which can provide continuing assurance of equipment operability over long time periods.

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TECHNICAL SESSION II

FUNCTIONAL QUALIFICATION OF MECHANICAL COMPONENTS AND CONSIDERATIONS FOR AGING PROBLEMS

by

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This paper provides a review of functional qualification and considerations for aging for pumps and valves used in commercial nuclear power plants. This multipurpose paper provides an overview of the status of qualification standards for pumps and valves and how aging is being considered; provides a discussion of qualification approaches; gives examples of aging of valves; and offers some suggestions for progress in the future. It should be noted that the views presented are those of the individual author except those portions pertaining to aging in the AEOD paper (Reference 1) which is an office position and the INPO paper (Reference 2). However, the paper does not represent a USNRC position.

There have been several items of interest presented at this morning's session. I would like you to keep in mind a few of those items which I believe pertain to a "realistic approach" and are as follows:

- Some equipment has degraded and some has not degraded;
- What is qualified life?
- Do we know actual plant conditions?
- A five-year buffer period is needed from premature standards;
- How can data bases such as LER, NPRDS, etc., be used? and
- There is a need for cooperative effort.

Figure 1 illustrates the various aspects pertaining to mechanical components that are covered by this paper. The general aspects are (1) who has responsibility (ASME/IEEE agreement); (2) differing philosophies of how to implement qualification and aging; (3) what is being done (proposed B16.41 on valves and pump standards); (4) examples of aging (AEOD/C203 and INPO paper); and (5) suggestions for progress. My intent is to explain what exists based on my understanding and participation in the development or review of the standards under consideration.

The ASME/IEEE agreement was reached in 1977. It was decided that IEEE was to prepare the generic qualification standard that was issued as IEEE STD 627-1980. This is a general standard for qualification of all types of safety systems equipment, mechanical and instrumentation as well as electrical. We will see later that this approach poses a dilemma for preparation of qualification standards for mechanical equipment.

A brief description of specific aspects of "design qualification" as used in IEEE STD 627-1980 is beneficial to understand the approach. Figure 2 shows the definition of design qualification and the qualification principles. The essential feature is to confirm the adequacy of equipment to perform its safety functions over the expected range of normal, abnormal, design basis event, post design basis event, and inservice test conditions. This procedure gives design life in real time, number of operating cycles or other performance intervals appropriate to the conditions.

The IEEE qualification principles consist of mandatory requirements (criteria, program development, evidence of completion, and a file), fundamental requirement (demonstrate equipment can perform), determination of aging (an essential part of the qualification process), determination of qualified life if aging is significant, and margin requirements. The approach and goal are certainly desirable. However, such a goal is difficult to implement for mechanical components.

The basic dilemma encountered with mechanical equipment is that aging mechanisms or parameters are generally not known or not understood. Additionally, it is not clear what phenomena actually constitute aging. For example, should aging include such aspects as wear, manufacturing tolerances, maintenance requirements or procedures, and effects of internal fluid environment? Therefore, those involved in the development of qualification requirements for mechanical components have adopted the approach shown in Figure 3. The approach utilizes a realistic two-step procedure that requires demonstration that equipment can operate under a set of severe conditions (as built with components representative of production procedures and equipment) and that accommodate aging by imposing additional operational testing and maintenance and inservice detection and evaluation requirements. I would also like to suggest a third step which should include maintenance procedures to ensure that operational maintenance work is conducted in a manner that does not obviate the original qualification program.

Functional qualification standards for valves and pumps have been under development by the ASME for several years. Figure 4 illustrates the approach used for the proposed B16.41 standard on functional qualification of power-operated active valve assemblies. This proposed standard was issued in June 1982 and it has taken nine years to reach this stage. The general approach is to require preparation of a functional specification in accordance with ANSI N278.1, demonstrate by test that the valve assembly can operate under a set of severe conditions, and the valve assembly must be representative of production valves.

It is recognized that this approach does not address aging effects directly. However, as shown in Figure 4, the standard addresses environmental and aging effects by relying on metal to provide assurance against significant loss of operability from environmental and aging effects (such as radiation, elevated environmental temperature and impinging chemical sprays), requires aging programs for plastic and elastomeric materials in accordance with IEEE-382, and states that periodic testing in accordance with Section XI of the ASME Code will assess effects of normal plant service conditions on valve assembly operation. In addition, some of these valve assemblies will be subject to the standards being developed by the Operations and Maintenance Committee of ASME.

The efforts related to pump standards are shown in Figure 5. There are six distinct but related standards planned and in draft form, but none have been issued. These standards will be identified as QNPE-1 (and successively higher numbers) as contrasted with the former N551.1 sequence. Although environmental and aging effects are generally not specifically addressed, there are some provisions that offer protection in this area. For example, the general requirements standard, QNPE-1, requires preparation of a functional specification for service and accident conditions and requires that maintenance requirements be provided when aging is known to be a problem. Some other requirements are on materials for protection against corrosion; bearings on hours of service life; wearing rings material and clearance; type testing to include erosion, corrosion and fatigue related to turbine drives; and rules for materials, design, fabrication, examination, and QA for intervening elements of reactor coolant pump motor frames. More detail is provided in Figure 5 and the information is provided without judgment concerning effectiveness or adequacy.

Examples of changes in valve assembly operating characteristics (aging) are shown in Figures 6 and 7. Figure 6 is a brief synopsis of an AEOD report, AEOD/C203, and as such represents the view of the Office for Analysis and Evaluation of Operational Data, but is not a USNRC position. The survey used licensee event reports (LERs) submitted to the NRC for 1978, 1979, and 1980.

The findings were: (1) torque switches represent approximately 25% of all motor operator events; (2) torque switches do not appear to be the dominant or root cause of inoperability; (3) repetitive problems occur on the same valve or similar valves; (4) plant staff efforts are directed toward return to operational status rather than finding the root cause; and (5) there were several motor burnout problems in the HPCI and RCIC systems of BWR plants. It is important to note that only five plants had more than three torque switch events in the three-year period. This tends to indicate that it may be difficult for individual plants to observe trends and patterns and indicates how important it is to cooperate and share information.

There were four major recommendations in the report. One was to reassess Regulatory Guide 1.106 because motor burnout appears related to bypassing the motor thermal overload protective devices. A second was to improve methods and procedures for setting torque switch positions. A third was to develop signature tracing techniques (such as voltage and current readings or actual stem thrust) to provide indication of deterioration or inadequate maintenance and also provide indication of margin for operation under accident conditions. The fourth recommendation was to have USNRC staff take positive steps to assure that aging is accommodated in the qualification process.

Additional evidence of valve aging is illustrated in Figure 7 and Reference 2 which is a synopsis of a study conducted by the Institute of Nuclear Power Operations (INPO). The paper was presented at the 1982 Engineering Conference on Reliability for the Electric Power Industry. This study was conducted for Limitorque valve operators and used a more extensive data base than the AEOD study. It concluded there was evidence of mechanical wear out, dirty contacts, motor failures and electrical failures. There were also some miscellaneous problems related to manufacturing errors, plant personnel error, grease-related failures, and limit switch failures. The results of the INPO study are similar to the AEOD report.

Figure 8 is a matrix representation of three broad aspects of qualification and aging related to valve assemblies. These broad aspects are definition of functional characteristics, provision to achieve the desired function, and quality assurance and reliability. The matrix has been filled in with various standards that apply to the valve assembly, body, operator, or components. The asterisks (*) indicate items that are being developed by standards writing groups. Although not shown, the standards developed by the Operations and Maintenance Committee could also be listed. The approach is useful to identify which standards have been completed and to aid in establishing priority for future work.

In summary, I have attempted to illustrate the current status and what needs to be done for qualification and aging with valves and pumps. I would encourage that we adopt a "realistic approach" that accommodates the lack of knowledge about aging. This could be done as follows:

1. Utilize the qualification approach to demonstrate components can operate under a set of severe conditions;

2. Impose additional requirements for operational testing and maintenance and inservice detection and evaluation;

3. Complete the draft standards for valves and pumps to provide basic qualification guidance (a five-year buffer period is not needed for these);

4. Use evidence from operating plants to identify aging mechanisms;

5. Question all operating experience, cooperate and share the information.

This approach should be benficial to all interested in accommodating aging problems in mechanical equipment.

References:

1. Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980, AEOD/C203 by Earl J. Brown and Frank S. Ashe.

2. A Safety/Availability Impact Study Using Data Combination Techniques, by J. M. Huzdovich, INPO, presented at 1982 Engineering Conference on Reliability for the Electric Power Industry.

Figure 1

I. Equipment Qualification

ASME/IEEE Agreement

II. Approaches to Qualification

A. Design Qualification
IEEE 627-1980
B. Other Qualification Approaches

III. ASME Developed Qualification Standards for Valves and Pumps

A. Proposed B16.41 - Valves
B. Proposed Pump Standards

IV. Examples of Changes in Valve Assembly Operating Characteristics (Aging)

A. AEOD Report (AEOD/C203)
B. INPO Paper (1982 Engineering Conference on Reliability for the Electric Power Industry)

Figure 2

II.A. Design Qualification (IEEE 627-1980)

1. Design Qualification - Confirm the adequacy of equipment to perform its safety functions over the expected range of normal, abnormal, design basis event, post design-basis event, and inservice test conditions.

2. Qualification Principles

A. Mandatory Requirements
B. Fundamental Requirement
Demonstrate equipment can perform the safety function(s) required, when operational and environmental loads are imposed.
C. Aging
Essential part of qualification process is to determine if aging has a significant effect on operability. If identified, aging program shall be developed.
D. Qualified Life
Required if aging is significant.
E. Margin
Specific qualification standards should provide specific guidelines.

Figure 3

II.B. Other Qualification Approaches

1. Demonstrate equipment can operate and perform its safety function under a prescribed set of severe conditions.

   A. As built

   B. Representative of production equipment

2. Impose additional operational testing and maintenance and inservice detection and evaluation.

3. Requirements for procedures to maintain qualified status.

Figure 4

III.A. Proposed B16.41 (Functional Qualification for Power-Operated Active Valve Assemblies)

1. General Approach (requirements)

   A. Functional specification shall be prepared

   B. Demonstrate by test that valve assembly can operate under specified conditions

   C. Representative of production valves

2. Environmental and Aging Effects

   A. Metal provides assurance against significant loss of operability

   B. Radiation aging required for plastic and elastometric materials (IEEE-382)

   C. Periodic inservice testing, ASME, Section XI, to assess effects of normal plant service conditions on valve assembly operability

Figure 5

III.B. Proposed Pump Qualification Standards

1. General Requirements (QNPE-1, formerly N551.1)

   A. Prepare specification for service and accident conditions

   B. Maintenance requirements for aging

2. Pump Assembly Qualification (QNPE-2, formerly N551.2)

   A. Requirement on materials for corrosion

   B. Bearings for hours of service life

   C. Wearing rings material and clearance

3. Shaft Seal Assemblies (QNPE-3, formerly N551.3)

   A. Production unit

   B. Material specification required.

   C. No environmental and aging requirements

4. Motor Drives (QNPE-4, formerly N551.4)

5. Turbine Drives (QNPE-5, formerly N551.5)

   A. Specification to include aging mechanisms

   B. Type test to include erosion, corrosion, and fatigue

6. Intervening Element (QNPE-7, formerly N551.7) (RCP Motor Frames)

   A. Similar to ASME Code Section III

   B. Rules for materials, design, fabrication, examination and QA

Figure 6

IV.A. AEOD Report (AEOD/C203) May 1982

1. Survey of valve operator events during 1978, 1979, and 1980 (LERs)

2. 444 valve operator events, 193 motor operator events

3. Findings

   A. Torque switch 25%
   B. Torque switch not dominant cause
   C. Repetitive problems
   D. Plant staff return to operational status rather than find root cause
   E. Motor burnout HPCI and RCIC

4. Recommendations

   A. Reassess Regulatory Guide 1.106 on bypass of motor thermal overload devices
   B. Improve methods and procedures for setting torque switches
   C. Develop signature tracing techniques
   D. NRC staff take positive steps to assure aging is accommodated

Figure 7

IV.B. INPO Paper (A Safety/Availability Impact Study Using Data Combination Techniques by J. M.Huzdovich)

1. Data Base - LERs, NPRDS, SEE-IN, OPEC

2. Study of Limitorque valve operators

3. Conclusions

   A. Mechanical failure
      Torque switch/limit switch adjustment
      Peaks in 1st, 4th, 6th, and 8th years
      Suggests wearout
   B. Dirty contacts
      1st, 5th, 8th, + years imply four-year cycle
   C. Motor failures
      2nd and 5th year. Following torque switch problems in 1st and 3rd year
   D. Electrical failures
      Exponential decline stabilizing at eight years
   E. Miscellanous
      Manufacturer errors
      Plant personnel error (maintenance)
      Grease-related failures
      Limit switch failures

Figure 8

	A. Valve Assembly	1. Valve Body (Pressure Boundary)	2. Operators a. Electric	b. Others (Hydraulic, Mechanical, Pneumatic)	3. Valve Components a. Internal and External
I. Define Functional Characteristics					
A. Function					
B. Functional Specification	N278.1				
C. Design Specifications		NCA 3250			
D. Loading Combinations					
II. Provision to Achieve Function					
A. Analysis, Design, & Construction					
1. Pressure Boundary		NB,NC,ND			CC 1621
2. Operability					
3. Environmental Compatibility					
B. Operation and Maintenance					
1. Operating Sequence					
2. Limits Inherent in Design					
3. Component Removal					
C. Misc. Requirements					
1. Redundant components systems					
2. Design, process, or configuration requirements					
III. Quality Assurance and Reliability					
A. Design, Material QA	NQA-1	NCA-4000			
B. Component or System Qualification					
1. Functional Qualification					
a. Qualification Tests	B16.41		IEEE 382	N41.6	N41.6
b. Analytical Evaluation	*				
c. Reliability	*				
2. Preoperational Testing					
C. Product Control					
1. Production Testing	*				
2. Non-Destructive Examination					
D. Construction Control					
1. Fabrication					
2. Materials					
3. Assembly					
4. Installation					
E. Operation Control					
1. Preservice Testing					
2. Operating Tests					
3. Inservice Testing	Sec.XI				
4. Maintenance					

Operation Testing and Maintenance Approach to Aging

Problems in Mechanical Components

by

Donald Beatty, P.E.

Manager of Nuclear Construction and Maintenance

Burns and Roe

ABSTRACT

The ASME Committee on Operation and Maintenance of Nuclear Power Plant Components recognizes the importance of aging and degradation to mechanical equipment which must operate to perform a safety function to nuclear power plants. The approach taken by the Committee in developing standards is primarily one of inservice testing and preventative maintenance coupled with qualification and production testing of components.

Since approximations to true service conditions must be made in qualification and production test programs, it is understood that some aging and degradation mechanisms may not be adequately understood and simulated. Likewise, the aging mechanisms may not be adequately circumvented in the design of the components. However, baseline performance data may be obtained in these preservice tests which assists in monitoring performance for tests performed inservice. Also for those situations where the aging problems are reasonably well understood, preventative maintenance programs can assist in assuring that individual mechanical components will be able to operate to perform their safety functions when needed. This paper describes the approach taken by the ASME Committee on Operation and Maintenance of Nuclear Power Plant Components in developing requirements to detect, evaluate and control aging problems in mechanical components.

Gentlemen:

Thank you for this opportunity to speak to you today.

I am speaking to you as an individual volunteer member of the ASME Committee on Operation and Maintenance of Nuclear Power Plants. I believe that I can reflect my own professional thoughts, and the philosophy that most of us on the Committee adher to.

However, in light of several recent court decisions, I must stress that these are individual committee member thoughts, and are not presented with "apparent authority" of the ASME. I am sure you are aware of the recent court decision regarding "apparent authority" of volunteer workers on Code Committees.

Perhaps I can take advantage of another recent court decision and plead "Innocent due to Temporary Insanity" or innocent because I have not yet passed a "Psychological Impact Study".

In any event, the assigned scope of this Committee is:

> To develop, maintain, and review Codes and Standards that are considered necessary for the safe and efficient operation and maintenance of Nuclear Power Plants, particularly as they relate to structural and functional adequacy.
>
> (a) Once a Code or Standard has been determined to be necessary or desirable, the Committee may then establish Subcommittees, Special Committees, or Working Groups to develop and maintain the Code or Standard or to refer them to already established committees.
>
> (b) This scope excludes administrative, Quality Assurance, security, and plant personnel qualification Standards.
>
> Codes and Standards developed by this Committee will be supervised by the ASME Nuclear Codes and Standards Committee.
>
> This Committee will establish recommendations for ASME representation on committees of organizations other than ASME developing interfacing Codes or Standards.
>
> This Committee will develop recommendations for ASME positions on interfacing or referenced Codes or Standards.
>
> This Committee will develop Codes and Standards in accordance with the Committee Procedures for Nuclear Projects approved by the American National Standards Institute under the Accredited Organization Method.

The committee is composed of approximately 25 members, representing -

Owners/Operators
Manufacturers
Designers/Constructors
Regulatory Agencies
Insurance/Inspections
General Interest

with not more than one-third of the membership from any of the above categories, thus giving our commitee industry concensus status.

We try, and have been successful, in keeping the owner/operator category at their full one-third membership.

Our original chairman was Wendell Johnson, Vice Presidnet, Yankee Atomic Electric. Our present chairman is Ed Williamson, Vice President, Southern Company Services.

Our underlying philosophy, since the formation of the committee, has been to develop a family of standards that will assure not only safe, but note the word efficient in our scope. We attempt, as much as possible to devise standards that can be applied without undue interference with the nuclear plants' purpose in life of producing power.

We have attempted to issue standards that allow operational testing that dovetails into refueling outages, where possible.

The standards that we have issued to date include:

OM-01	"Requirements for Inservice Peformance Testing of Pressure Relief Devices"
OM-02	"Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water Systems"
OM-03	"Requirements for Preoperational and Initial Start-up Testing of Nuclear Power Plant Piping Systems"
OM-04	"Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)"
OM-05	"Inservice Monitoring of Core Support Axial Preload in Pressurized Water Reactors"
OM-06	"Performance Testing of Pumps"
OM-07	"Requirements for Preoperational and Initial Start-up Thermal Expansion Testing of Nuclear Power Plant Piping Systems"
OM-08	"Periodic and Performance Testing of Motor-Operated Valves (MOV's)"
OM-09	"Inservice and Performance Testing of Cranes"
OM-10	"Performance Testing of Valves"
OM-11	"Vibration Monitoring of Heat Exchangers"
OM-12	"Vibration Monitoring and Diagnostics of Light Water Reactor Loose Parts"

Some of these are still in final review and comment stage, so are not as yet "formally" available.

Note that several of these, for example OM-06 - "Performance Testing of Pumps" and OM-08 and OM-10 - "Performance Testing of Valves" will be lifted from ASME Section XI by reference. This is an attempt to purify Section XI to ISI only, not operational testing.

The committee is presently feeling our way on early drafts on the following:

OM-XX	"On Line Monitoring of Rotating Equipment"
OM-XX	"Vibration Monitoring of Transient Events"
OM-XX	"Vibration Monitoring of Valves"
OM-XX	"Requirements for Periodic Performance Testing and Monitoring of Power Operated Relief Valves and Actuating Systems"
OM-XX	"Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems"
OM-XX	"Short Duration Leak Rate Testing of Containment Systems"

I believe that you will see much more action in the future toward On Line monitoring, including Acoustic and Mechanical Vibration Testing, as a diagnostic tool to detect component changes.

You will also see a strong orientation toward preventive maintenance tests and/or inspections to monitor changes from the original base line data.

The old adage of "don't take a pump apart if it is running good" may well be the safest approach if diagnostic tools, such as vibration monitoring, tell you that no changes are occurring in the acoustic or mechanical vibration signature. Periodic disassembly, measuring, testing, and reassembly is not always the safest or most efficient method of ensuring good, reliable operation. I am sure that all of us can point to instances where this has created problems that did not exist before the disassembly and reassembly.

All of us on the ASME OM Committee are dedicated to safe, efficient Nuclear Plant Operation. We recognize that Loss Prevention due to malfunctions of equipment or components is of paramount importance. We also believe that periodic inservice testing and diagnostic monitoring can go a long way toward achieving this end while keeping the Nuclear Plant operating efficiently.

We hopefully will be issuing industry consensus standards that achieve both safety and efficient plant operation. We might even find a few components that do not agree that they degrade in safety within their design lifetime.

Thank you.

Donald Beatty, P.E.

BNL-NUREG-31864

The Degradation of Steam Generator Tubing and Components by Operation of Pressurized Water Reactors*

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Experience in operating pressurized water reactors (PWR) has shown a number of materials degradation processes to have occurred in their steam generators. These include stress corrosion cracking (SCC), intergranular attack, generalized dissolution, and pitting attack on steam generator tubes; mechanical damage to steam generator tubes; extensive corrosion of tubing support plates (denting); and cracking of feedwater lines and steam generator vessels. The current status of our understanding of the causes of each of these phenomena is reviewed with emphasis on their possible significance to reactor safety and directions the nuclear industry and the NRC should be taking to reduce the rate of degradation of steam generator components.

INTRODUCTION

Operation of PWR steam generators has led to a number of degradation processes on the steam generator tubing, the support plates, the vessels, and the piping. These are listed in Table 1.

Table 1
Steam Generator Degradation Processes

A. Tubing (Inconel-600)
   - Primary Side:
     - SCC
     - Intergranular Attack
   - Secondary Side:
     - SCC and Intergranular Attack
     - Localized Wastage and Pitting
     - Mechanical Damage

B. Tube Support Plate
   - Corrosion
   - Deformation

C. Piping and Vessel
   - Fatigue
   - Corrosion

The steam generators in most PWR's that have operated for more than a year have experienced one or more of these phenomena. The interests of the NRC in these phenomena are shown in Table 2, as we understand them to be.

*Research carried out under the auspices of the U. S. Nuclear Regulatory Commission.

Table 2
NRC Interests in Steam Generator Degradation

1. Is it safe to return a degraded steam generator to service?
2. Do we understand the degradation mechanisms and proposed remedial actions well enough to determine it is safe to operate the unit?
3. For how long?
4. How reliable are inservice inspection techniques for identifying the nature and extent of the defects?
5. Is the degradation that occurred likely to be affecting other units? If so, what actions should be required of other licensees to ensure their continued safe operation?
6. What actions should be required of applicants for licenses to minimize the risk to safety, based on our understanding of past experience?

The purpose of this paper is to describe the problems that have occurred in terms of these interests, indicating where possible, what research is being (or needs to be) performed to meet these objectives.

In the design and licensing of PWR's, minor steam generator tube leakage was anticipated, leak detection and tube inspection were provided for, and such leakage was not considered a major safety concern in itself. However, widespread degradation that could lead to major leakage developing from failure of a number of tubes during a design basis accident, resulting in a significant release of radioactivity to the environment, is a potential safety concern. As nuclear plants get older and degradation processes occur, the NRC needs to assure itself that this latter condition has not been reached. We are, of course, frequently in the gray area between these extreme cases, so that, when degradation and/or leakage develops, careful reviews of the facts and proposed remedial actions need to be made on an ad hoc basis to provide answers to questions such as those listed in Table 2.

Steam generator degradation and suggested remedial actions have been the subject of several recent review articles (1,2,3), including a topical conference sponsored by the American Nuclear Society (4), a draft NUREG (5), and an NRC press release in February 1982 (6). For the purposes of this workshop, therefore, the following summary is intended only to provide a basis for discussing possible research needs for improving performance and safety of PWR steam generators.

TYPES OF STEAM GENERATOR DEGRADATION

Materials of construction of PWR steam generators currently in service in the United States are listed in Table 3. Steam generators currently under construction by Westinghouse and Combustion Engineering have, in addition to design changes, changes in materials, as shown in the table. Since operating experience with nuclear steam generators has, on occasion, shown that a treatment designed to alleviate one problem has introduced a different one, we believe it is appropriate to discuss anticipated concerns over operating steam generators with the newer materials as well as the experience to date.

Attack on Inconel 600 Tubing

As shown in Table 1, degradation of Inconel 600 tubing has occurred from both the primary and secondary sides of the PWR steam generators.

Primary Side Stress Corrosion - Stress corrosion cracking of Inconel 600 tubing originating from the primary side of the steam generator has occurred in service in areas where high residual stresses were present from fabricating the steam generator and in areas where in-service deformation occurred as a result of secondary side corrosion processes (denting). Typical locations of cracks from residual stresses have been short radius U-bends of the innermost rows of tubes in a steam generator, and where the transition zone occurs on a tube that has been partially expanded into the tube sheet hole. The U-bend cracking has primarily occurred on tubes fabricated by Westinghouse over a several year period, although it has also been observed on European plants. It appears to be associated with the details of the bending process. Cracks appear to initiate at a point of maximum gradients in residual stresses. Stress corrosion cracking where tubes are expanded into the tube sheet crevice has occurred primarily in overseas PWR's of the present time. Figure 1(a) and (b) shows typical areas where these cracks have occurred. Where in-service deformation has resulted from denting, there has been an accelerated tendency toward primary side SCC. The most significant leak developing from this cause occurred at Surry, as a result of ovalization of a short-radius U-bend, resulting from denting, as sketched in Figure 2.

Figure 3 is an often repeated sketch (7) showing the synergistic interplay of environment, material composition, and stress that underly all SCC phenomena. An NRC sponsored research program is underway at BNL (8), the purpose of which is to develop a sufficient understanding of the degradation mechanisms and causative factors to permit an estimation of the lifetime of a steam generator tube. Results to date have been encouraging that these goals can be met, the principal observations are summarized in Table 4. It is anticipated that by the end of 1983, the program will be completed to a point where predictions of the life time of tubing under a variety of operating conditions can be made, including the situations where active deformation of the metal continues, where deformation has occurred but has been arrested, and where abnormal water chemistry might be encountered with or without active deformation.

With regard to solutions to the problem, returning again to Figure 3, the environment, at least on the primary side of a PWR steam generator, probably cannot be changed at the present time. The material can be changed only in new plants; for all SG's manufactured after approximately 1980, the Inconel 600 tubing has been given a heat treatment to improve its resistance to primary side SCC. On an existing generator, the only variable that can conceivably be modified is the stress pattern, and research is underway at the present time, under the auspices of the Electric Power Research Institute Steam Generators Owners Group (EPRI-SGOG), to develop techniques for in situ stress relief, particularly in those areas where the tube is expanded into the tube sheet.

TABLE 3

STEAM GENERATOR MATERIALS

SECONDARY SIDE	
Vessel	Carbon Steel
Tube Sheet	Carbon Steel
Tubes	Stainless Steel
	Inconel-600
	Incoloy-800
(Tube Support Plates)	Carbon Steel
	Ferritic Stainless Steel
PRIMARY SIDE	Inconel-600
	Stainless Steel

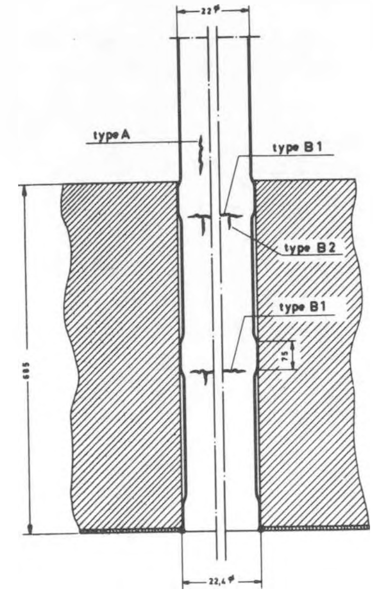

FIGURE 1A: SCHEMATIC OF TUBE-TUBE SHEET AT OBRIGHEIM, SHOWING LOCATION OF PRIMARY SIDE SCC (TYPE B) AND SECONDARY SIDE CAUSTIC SCC (TYPE A)

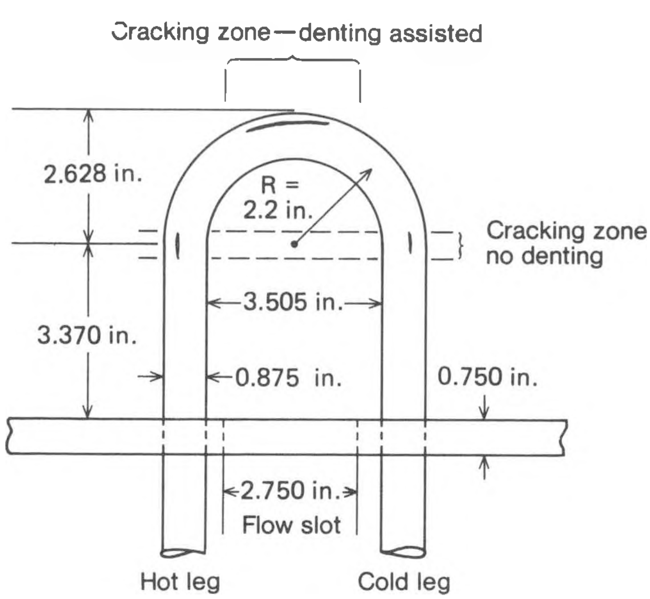

FIGURE 1B: SCHEMATIC OF U-BEND, SHOWING WHERE PRIMARY SIDE SCC HAS OCCURRED

TABLE 4

RESULTS OF BNL INCONEL-600 PROGRAM

1. Wide heat-to-heat variation in susceptibilit
2. Strain-rate effects
3. Accelerating effects of cold work
4. Coupling with C steel
5. Effect of H_2
6. Effect of Alloy composition

Intergranular Attack from the Primary Side has occurred in one unit during an extensive layup period in which the upper portion of the (once-through) steam generators were vented to the atmosphere, and small amounts of sulfur were present in the solution, presumably originating from traces of thiosulfate present in the containment sprays of the affected unit. This intergranular attack appears to be accelerated by stress. Research is presently underway under the auspices of the affected utility to determine whether or not it is safe to return the unit to service with tubes in it that have been exposed to this environment. Bench tests performed at BNL have shown that the phenomenon can be reproduced in extremely dilute thiosulfate solutions in the presence of oxygen, and that increasing the amount of Li_2O, reducing the thiosulfate in the solution and eliminating oxygen in the primary coolant can inhibit further attack. The main solution to this problem in terms of Figure 3, therefore appears to be in adjusting the environment. The situation, however, for the affected utility is still under active review by the NRC regulatory staff and their consultants.

Secondary Side Corrosion - Most of the damage to Inconel steam generator tubing from service exposure has occurred on the secondary side, and three of the four largest primary to secondary inservice leaks have originated from secondary side degradation processes. The need to boil water to produce steam creates a mechanism for concentration of impurities against a hot steaming surface and also creates a highly agitated environment in which mechanical damage from foreign objects is a distinct possibility. The need to support the tubing to prevent vibration also inevitably creates creviced areas where impurities can concentrate due to the boiling process; the concentrated impurities may precipitate as boiler scale on the tubes, accumulate on the tube sheet as a pile of sludge, or, where highly soluble, redissolve when the heat flux is reduced.

Impurities that have caused corrosion of the Inconel tubing in these creviced areas fall into several types: first, caustic (NaOH, KOH), which can develop by hydrolysis of carbonates and nitrates of sodium or potasium that leak in through the condenser or by hydrolysis of sodium phosphates intentionally added to buffer condenser inleakage; second, acid phosphates (typically $Na_xH_yPO_4$) where the Na/PO_4 molar ratio is less than 2.3) can where concentrated in crevice areas react with the Inconel to produce complex sodium-nickel phosphates; third, chloride ions inleaking through the condenser (especially when accompanied by oxygen and a cation such as magnesium that hydrolyzes to produce an acid chloride solution) can concentrate in crevice areas to produce local corrosion (pitting) of the Inconel or local corrosion of the tube support plate; and fourth, acid sulfates, which can be produced by thermal decomposition of cation resin beads in the steam generator.

Caustic Stress Corrosion Cracking - The majority of instances of caustic SCC have occurred in the sludge pile area, above the tube sheet, on plants that used a phosphate treatment, where hydrolysis reactions between the phosphate and corrosion product oxides led to the development of free sodium hydroxide in the sludge pile area. This type of attack can be minimized by appropriate water chemistry controls to eliminate the build up of free caustic in the sludge pile. Phenomenologically, caustic stress corrosion appears to

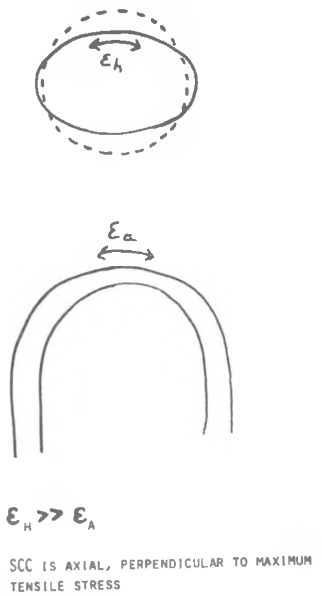

FIGURE 2: SKETCH OF OVALIZATION AT TOP OF U-BEND THAT OCCURS WHEN MOTION OF TUBE SUPPORT PLATE, RESULTING FROM DENTING, DECREASES THE DISTANCE BETWEEN THE HOT AND COLD LEGS

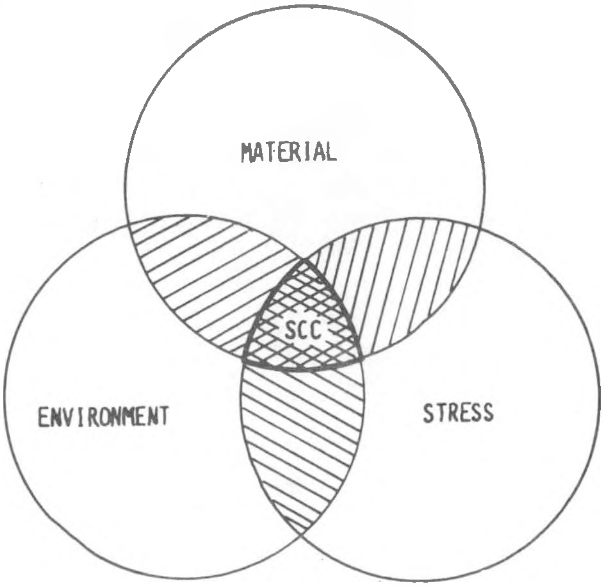

FIGURE 3: SKETCH OF THE INTERPLAY OF STRESS, MATERIAL, AND ENVIRONMENTAL FACTORS THAT CAUSE SCC

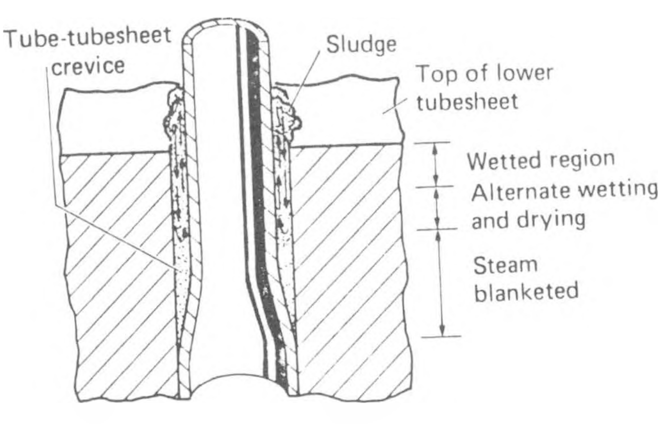

FIGURE 4: TUBE-TUBE SHEET CREVICE

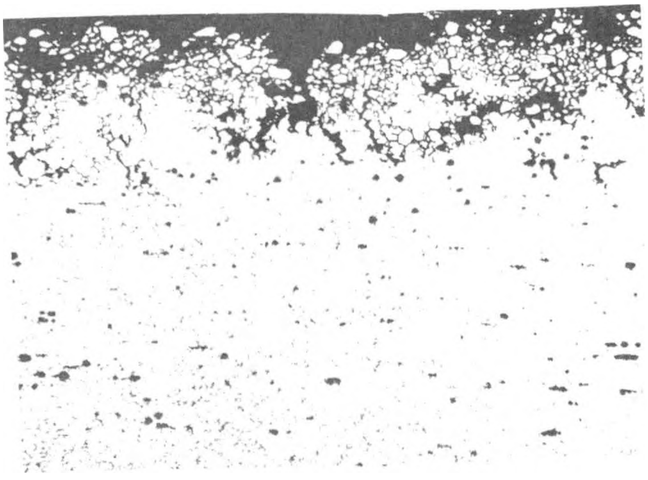

FIGURE 5: NATURE OF IGA ON SECONDARY SIDE OF TUBES IN THE TUBE-TUBE SHEET CREVICE

be quite similar to the primary side (pure water) stress corrosion, in that the same conditions of the Inconel tubing that make it susceptible to one type of attack make it susceptible to the other. The use of thermally treated Inconel tubing to minimize primary side SCC should also minimize secondary side caustic SCC.

Intergranular Attack - In many older operating PWRs, the tubing is only expanded partially to the tube sheet crevice, as sketched in Figure 4. In some instances, the development of a caustic-forming environment in these crevices has led to a general intergranular attack on the Inconel tubing. A typical example is shown in Figure 5. This phenomenon has been observed in units that utilized a phosphate water chemistry (San Onofre-1), that developed a caustic environment following conversion of phosphates to an avt chemistry (Point Beach units 1 and 2), or in units utilizing an avt from startup, following leakage of caustic forming impurities (Beznau-1 in 1968). In terms of the NRC interests, itemized in Table 2, this has been one of the most difficult forms of degradation to evaluate. Research under EPRI sponsorship is just now beginning to develop an understanding of the degradation mechanism. From a safety point of view, any leaks that develop in the tube to tubesheet crevice area are likely to be small and detectable. The crevices themselves are tightly packed with corrosion product oxides, which tend to limit the rate of leakage. Further, the tube sheet crevice will restrain the tube and prevent either a fish-mouth type opening or a double-ended tube rupture. To date this type of degradation has not been observed to have progressed above the tube sheet into the sludge pile area, where the tubes (or leakage) would not be restrained. The situation in affected units, however, bears watching and requires frequent inspections to ensure that an unsafe situation has not been reached. Based on laboratory tests, Inconel tubing that has been thermally treated to improve its stress corrosion resistance to primary coolant and secondary side caustic SCC also appears to have improved resistance to intergranular attack. Several utilities have performed an extensive sleeving of the affected areas with a more corrosion resistant material in order to minimize the primary to secondary leakage from the source and the chance that an unsafe situation might develop. The newer units utilize the thermally treated Inconel tubing and have eliminated by design the crevices in which this attack has occurred.

Wastage - General corrosion of the Inconel tubes in creviced areas by acid phosphates, which was a matter of some concern in the early to mid seventies, has largely been eliminated by the abandonment of phosphate water chemistry on all but two operating PWR's. Figure 6 shows the general nature of this attack. Complete removal of the phosphates however, has been difficult to achieve, and phosphate residues in the blow-down from some affected units were observed for several years following abandonment of this treatment. On the two units that continue to utilize the phosphate treatment, careful control of sodium to phosphate ratio has prevented significant further instances of either caustic corrosion cracking or rapid wastage. However, some units still in operation have many previously-plugged tubes containing defects similar to that shown in the figure. One unit (Palisades) attempted a program of sleeving to increase the residual strength of the tubing in the wasted areas.

Pitting - Pitting of Inconel tubes has occurred on several units where a combination of oxygen or air inleakage through the condenser, chloride inleakage, and copper ions has set up local corrosion cells. Typically, the pits are observed as rows of small diameter penetrations. Mechanical tests suggest that these pits, while they may lead to primary to secondary leakage, cannot seriously weaken the rupture strength of the tube. They can however, be a significant operational problem and, since any primary to secondary leakage is matter of some concern to the NRC, they may require frequent shutdowns of the affected unit. Pitting by chlorides can be controlled by improved condenser performance both for the elimination of the condenser as a source of chloride, and (perhaps even more importantly) as a source of oxygen to the system.

Pits of the type described have occurred in operating units in an area away from the tube support plates. Where the same species that cause pitting on a free tube surface get concentrated in the tube to tube support plate crevice, denting type reactions, i.e., corrosion of the carbon steel support plates, have predominated. In these areas the carbon steel plate probably provides galvanic protection against pitting of the Inconel. In units currently being designed with stainless steel tube support plates, the denting type reactions are prevented, and pitting type reactions will probably prevail in these crevices, should the same combinations of impurities enter the steam generator.

Pitting also has been observed where condensate demineralization was used to protect the steam generators from intrusion of impurities from the condenser. Resin fines are known to hydrolize to form acid sulfate impurities at steam generator temperatures. Should these resin fines concentrate or get trapped in a tube to tube support plate crevice, a localized pitting of the Inconel will occur.

Mechanical Damage - Mechanical damage to the secondary side of steam generator tubes can arise from several sources. Tube to tube-support plate impact has not been widespread to date , although vibrating flow baffles have caused some wear of Inconel tubing in the preheater sections of one or two of the newer units. As design changes are made to steam generators to minimize areas where corrosive impurities can concentrate, we may find greater instances of vibration induced degradation. In several of the earlier PWR's, fretting between the tubes and anti-vibration bars was observed. This problem was resolved by changing the design and material of the anti-vibration bars on subseqent units. Mechanical damage (wear) has also been observed near the uppermost tube support plates in once-through steam generators (3). The abrasive agent is presumed to be corrosion product oxides carried in suspension by the high velocity steam.

The largest leaks from mechanical damage have occurred as a result of foreign objects inadvertently left in the steam generators. At one unit, a large spring from a sludge lancing tool was inadvertently left in the steam generator and vibrated against the tube, producing a massive leak. In a second unit (9) a foreign object vibrating against the outermost row of tubes produced a series of small leaks over a several year period. Following the plugging of the affected tubes, the foreign object continued to damage the

FIGURE 6: Nature of Wastage of Inconel in the Vicinity of Tubing Supports in the Palisades Reactor. The Defect Shown is 35 mils Deep. Flow was from Left to Right.

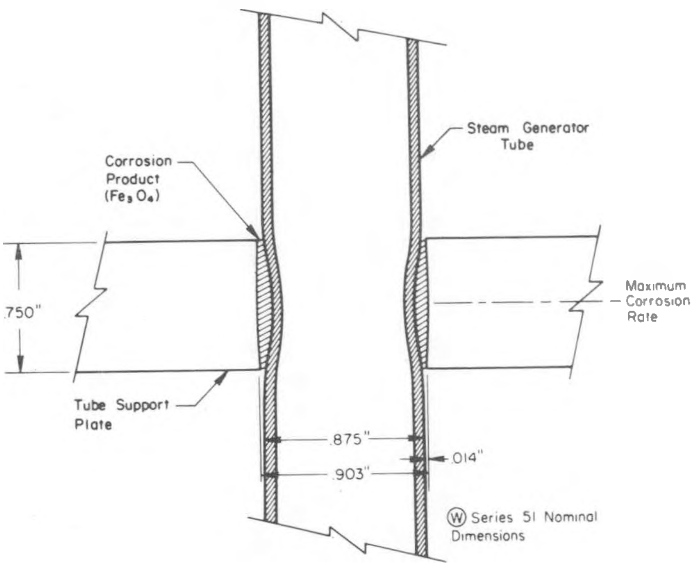

FIGURE 7: SCHEMATIC OF DENTING AT TUBE SUPPORT PLATE CREVICES

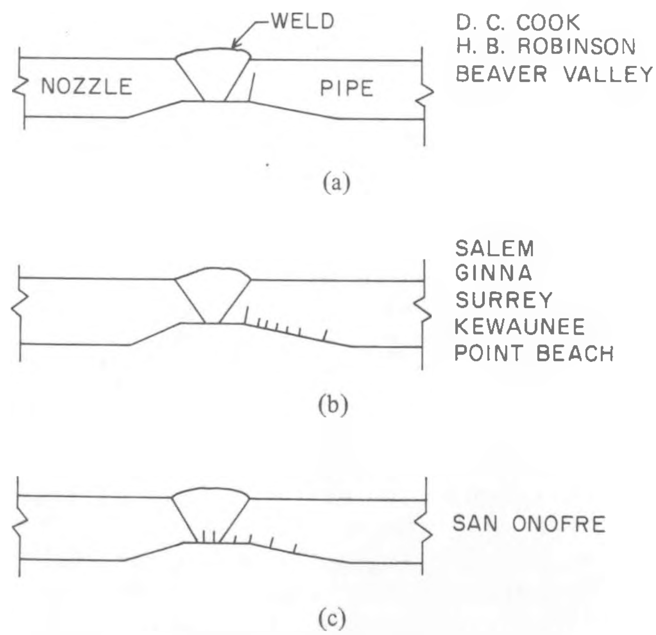

FIGURE 8: Schematic diagram of crack morphology by type and plant: (a) single large crack at root of the notch, (b) several small cracks in counterbore region, and (c) shallow cracks in weld and counterbore region.

previously plugged tubes resulting in their eventual rupture, leaving these tubes free to vibrate against the adjacent tubes in the bundle. Eventually, one of these broken previously-plugged tubes rubbed against a good tube, producing a general loss of thickness over an extended length of that tube, resulting in a fish-mouth type rupture. In terms of the questions in Table 2, the mechanism of this degradation process is, we think, sufficiently understood that it is safe to operate the unit with the foreign objects removed. But the incident does raise the question of whether or not damage might be occurring to previously plugged tubes in other operating units that could lead to these tubes breaking during operation and therefore rubbing and wearing against adjacent tubes. For example, tubes with severe phosphate wastage or through-wall stress corrosion cracks may be continuing to degrade. This is an area which research to date has not specifically addressed, either funded by the NRC or by the industry.

Tube Support Plate Corrosion (Denting)

Denting of steam generator tubes by runaway corrosion of tube support plates was first observed in the middle 1970's and is continuing to the present in some units. Widespread denting and the related problems of inservice strain-induced primary side cracking has led to the replacement of all steam generators at both of the Surry and both of the Turkey Point units. Figure 7 shows a schematic of the denting type reactions. The nonprotective magnetite growth on the carbon steel support plate is known to be triggered by the presence of chlorides, an acid environment, and oxidizing ions such as copper or nickel. In addition to producing dents and deformation in the Inconel tubes, leading to primary side SCC, denting processes result in considerable distortion to and cracking of the tube support plate as well. To date, foreign objects that are capable of damaging tubes have not broken loose from support plates by this mechanism, but the situation bears watching in the future. Although in some units the low pH has been attributed to hydrolysis of residual phosphates, the majority of the acidity is produced by hydrolysis in the steam generator of substances such as magnesium chloride contained in sea water. Thus, plants with condensers cooled by sea water or brackish water are both by experience and prediction the more likely to be degraded by denting processes. The copper or nickel ions that are known to trigger or accelerate denting will not be produced by reaction of these metals with water. Therefore, proper oxygen control or deaeration in the condenser and feedwater lines, as well as in the steam generator, together with careful monitoring of and response to condenser leakage appear to be the best defense against denting reactions. EPRI funded research has shown that chemical treatments involving a mixture of ammonia and boric acid have a beneficial effect on reducing the rate of denting in an affected unit. The effects of this treatment on other components of the system, however, are not clear at the present time.

All secondary side degradation processes (except for mechanical damage) can be controlled or minimized by meticulous attention to water chemistry on the secondary side of the steam generators. The NRC Office of Regulation has issued a Draft Branch Technical Position 5-3 indicating what it believes (given the present state of knowledge) to be acceptable controls for minimizing these degradation processes. The EPRI Steam Generator Owner's

Group has likewise prepared draft recommendations on secondary water chemistry. Were it possible for an operating unit to remain at all times within very tight water chemistry controls, most of these problems could be minimized. Where research is needed, however, is in attempting to develop a sufficient understanding of the interrelations of impurities with crevices in steam generators to permit the placing of acceptable limits on time at which predefined concentrations of impurities can be present before significant damage can be initiated or before crevice corrosion reactions will be triggered that are difficult to stop.

Each design change in either water chemistry or materials that has been made (or is being proposed to be made) needs to be carefully examined in terms of side effects of this change on other aspects of the overall secondary coolant system. For example, lowering the temperature slightly to reduce the rate of secondary side corrosion could throw more impurities into the turbines and potentially increase corrosion problems there, or introduction of demineralizers to protect the system from a leaky condenser has at least in one instance thrown corrosive impurities into the steam generator in the form of resin beads. An older example was the onset of denting type reactions following the abandonment of the phosphate water chemistry treatment. The authors believe that the overall key into future improvement in steam generator performance must come from the proper understanding of the secondary side degradation processes combined with improved design and operational maintainence of condensers to eliminate inleakage of caustic or acid-forming impurities, chlorides, and air into the steam generator and feedwater lines.

Piping and Vessel Cracks

Thermal fatigue processes, possibly accelerated by environmental exposure, have produced cracks and leaks in the feedwater lines to the secondary side of a number of PWR steam generators and in the steam generator vessels themselves at one unit. Figure 8 sketches the type of degradation that were observed on specimens from feedwater lines of a number of units examined in the authors' laboratory[10]. The thermal fatigue mechanism arises from the introduction to the steam generator of ambient temperature water from the auxiliary feed system at various stages during startup and shutdown of the steam generators. The obvious solution to this problem comes from proper mixing of the auxiliary and main feedwater to prevent further stratification. Quite recently a similar phenomenon has been observed in the vicinity of the cone to upper cylinder weld on the steam generators at one operating unit. The situation is under investigation at the present time, but phenomenologically the cracks are similar in appearance to those observed in the feedwater lines at several units, especially at San Onofre 1, where environmental effects appear to have been a contributory cause. It is premature at this time to conclude whether residual stresses adjacent to this weld or the thermal stresses from the introduction of cold auxiliary feed water or environmental effects (this unit has suffered considerable denting and pitting and has used a boric acid treatment to retard the progression of denting) or all three of these factors contributed to the cracking and leakage. Research into the relative importance of these three factors on causing vessel damage needs, in the authors' opinion, to be performed to minimize future occurrences of this phenomenon in steam generators of other operating units.

SUMMARY AND CONCLUSIONS

There are a number of items which the authors believe need to be addressed in order to improve both the operational reliability and safety of nuclear steam generators. Many (but not all) of these are currently being addressed in research programs funded by the NRC and by the industry through EPRI. NRC funded research is currently addressing primary side SCC from the point of view of developing sufficient understanding of the mechanism to determine the time that is safe to operate an affected unit. A program at Oak Ridge (and an earlier program at Battelle-Columbus) is being performed to determine the reliability of eddy current inspection techniques to determine the presence and extent of degradation from the various processes. The NRC also has a program to determine the mechanical properties of severely degraded tubes in order to set safe limits on the amount of degradation that a tube can undergo and still withstand the stresses anticipated during a design basis accident with a sufficient margin of safety. NRC is also funding a failure analysis on one of the severely degraded units removed from the Surry Plant, which will provide samples for confirmatory analysis for both the effects of inservice exposure on the SCC process, and for checking on a real unit the reliability of inservice inspection techniques, possible proposed chemical cleaning techniques, and the mechanical properties of service-degraded tubes. The long range NRC research plan calls for initiation of a program on water chemistry controls for the secondary side of steam generators. As noted above, additional research is needed to answer several questions listed below.

As a plant with known degradation ages, we need to know to ensure its safe continued operation (to answer the questions in Table 2) the following information:

1. How is degradation progressing following remedial actions?

2. Does the degradation stop on tubes that have been plugged or will continued degradation on plugged tubes progress until one breaks, perhaps to reinitiate the Ginna chain of events? Although this has not yet happened in service, we have no steam generators with Inconel tubes that have operated for even half of their design 40 year life.

3. Have remedial actions taken to correct one type of degradation introduced possibly a second one? For example, if we replaced the carbon steel supports with stainless steel tube support plates, as on new units, will we lose the galvanic protection of the carbon steel and therefore, see more pitting of Inconel tubes? If we reduce the area of support to minimize the areas in which impurities can concentrate, will we see more fretting and wear type damage on steam generator tubes? If we rely on demineralizers to improve the water chemistry, will we be introducing more resin fine pitting?

4. What water chemistry controls really are satisfactory to ensure reliability and safety? Careful maintainence of water chemistry in the steam generator alone may not be sufficient, in that careful monitoring of the condensers for chloride and air inleakage, and careful deaeration of all sources of water to the steam generator, including the condenser, the auxiliary feedwater, and the condensate storage tank may be necessary.

ACKNOWLEDGEMENT

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STRESS CORROSION CRACKING IN BOILING WATER REACTOR (BWR) PIPING[†]

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ABSTRACT

Results are presented from recent Electric Power Research Institute projects on stress corrosion cracking in 304 stainless steel Boiling Water Reactor (BWR) piping. Repair rates have been calculated for stress corrosion cracking incidents grouped by piping systems, BWR generations, and individual plants, using non-parametric order statistics techniques. From these results, the original data groups are lumped into statistically-justified populations for further analysis. The rates for different plants or BWR generations are not found to be significantly different, but the rates for different piping systems are. Consequently, the number of repairs per plant per year is calculated separately for several piping systems with all plants lumped together. The repair rate is increasing with plant age in most systems, indicating a time-dependent wear-out phenomenon. The cost-effectiveness of countermeasures such as enhanced inspection, startup deaeration, material replacement, etc., is analyzed with a breakeven approach.

A simple analytical model is under development for predicting intergranular stress corrosion crack growth as a function of material sensitization, oxygen concentration, temperature, and stress or strain rate. The model has been partially calibrated with available laboratory test data from constant extension rate and constant load tests. The preliminary model predictions are encouraging, exhibiting appropriate functional shape in all terms.

INTRODUCTION

Intergranular stress corrosion cracking (IGSCC) of stainless steel piping in Boiling Water Reactors (BWRs) has resulted in significant losses in plant

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[†]The projects described in this paper were funded by the Electric Power Research Institute, Contracts RP700-6 and RP2006-4.

availability and costly repairs. To reduce the incidence of IGSCC, a number of countermeasures have been proposed (1). This paper looks closely at the cost-effectiveness of these countermeasures, and presents preliminary results for a predictive model that is under development to estimate time to failure under field conditions (2,3).

COST-EFFECTIVENESS OF COUNTERMEASURES

A decision analysis was performed for a general class of countermeasures: those that reduce the incidence of IGSCC by an equal percentage in all piping systems. Examples of this class include deaeration, hydrogen water chemistry, complete material replacement and uniform inspection. In developing a computer-searchable data base for evaluating these countermeasures, an intensive effort was made to collect and evaluate all relevant data on IGSCC incidents, including crack description, material description, dates and operating data, repair details, piping systems and pipe size, and post weld treatment.

The information in the data base was compiled from, and checked against, several significant sources. Early cracking incidents (pre-April 1975) were summarized in NEDO 21000 (4). Post-July 1975 incidents were obtained from several sources, the two most important being the monthly Operating Unit Status Reports (5) and a computer search for cracking incidents performed by the NRC on their Licensee Event Report data base (6), which also provided an additional check on the incidents described in the NEDO 21000 study. The detailed NRC Licensee Event Reports (7) were used to supplement the summary information provided by these sources. Additional information was provided by cognizant EPRI and GE personnel, a listing from NUS Corporation (8), which included some of the recent cracking incidents, and the NUS annual summary (9). Discrepancies and details on the extent of repair (e.g., individual weld ground out and repaired, or entire line replaced) were resolved via telephone calls to operating personnel at the particular nuclear plant.

If all BWR piping welds were the same age, analysis of failure rate would be relatively simple. A Weibull analysis approach could be used, as described in (10). Unfortunately, the IGSCC data have complexities that cannot be handled by a simple Weibull approach. The primary difficulty is that the number of welds at risk decreases with plant age. That is, all operating BWR's are at least two years old, a few reactors are 10 years old, but only one reactor is over 20 years old (Dresden 1). The welds in all plants are considered a single population, so the analysis technique must account for the variable number of welds in the population. The Failure Analysis Associates computer program SAFECC does this by incrementally constructing the cumulative fraction failed, using order-statistics techniques. The code also considers the fact that the welds are replaced as they are repaired, and the possibility that the failure process begins after $t = 0$. The latter complexity produces a curve on

Weibull paper and requires a three-parameter Weibull distribution for accurate fitting. SAFECC calculates a mean estimate of the failure rate and confidence limits that are $\pm 2\sigma$ from the mean.

The definition of failure for this project is any individual repair that is the direct result of either definite IGSCC cracks or ultrasonic indications that are believed to be IGSCC. Because of this definition, many cracks in one weld, or more than one cracked weld in a pipe, would count as only one failure if they were corrected by a single repair. "Repair" and "failure" are used interchangeably below. The reason for choosing "single repair" rather than "single crack" for the failure definition is that the only economic impact is the cost of the repair, regardless of how many cracks exist. In many cases, multiple cracks can be repaired for about the same cost as single cracks.

The failure rate used in this study is the number of repairs per day divided by the effective number of welds at risk, using the GE weld counts (4). The weld counts in (4) only include the welds inside containment, and recent information indicates that the number of welds in small (<4") diameter piping may be underestimated. The exposure variable is calendar time in days. In previous studies (4), the number of long shutdowns (>24 hours) was used as the exposure variable, based on the argument that the failure rate fits a Weibull curve better with this definition. As demonstrated in Figure 1, the number of shutdowns and the age of the plant are highly correlated, so it doesn't matter which variable is used. Calendar time was chosen because it is natural for economic analysis, convenient, available, and historically correlated with the natural physical exposure variables such as number of temperature and stress cycles and time at temperature.

The data was divided according to type of plant (BWR 1, 2, 3, or 4) and for each of these, different piping lines were separately considered. These included recirculation bypass; core spray; control rod drive; reactor water cleanup lines; and all welds that were not in these systems, designated as "other." Some individual plants were considered separately, such as Dresden 1, to determine whether the data would support allegations that the plant is different in some way.

Using the failure rate calculations from SAFECC to determine statistically sound data consolidations, it was discovered that there is no significant difference between the different BWR generations. The only exception is that the recirculation bypass failure rates demonstrated two failure modes, as shown in Figure 2. The earlier failures (hollow circles) are all from BWR 4 reactors. The three latest points from BWR 4s appear to fit in with the failure points from other generations, though much more data is required to state this conclusively. One failure mode therefore appears to operate in the early years, followed by a second mode after approximately the third year.

The early mode may be nothing more than enhanced inspection of the first operating years of BWR 4s relative to the previous generations. Whatever its cause, this early failure mode does not affect the economic analysis, since zero time for the economic analysis was taken as the fifth year of plant operation. Most current plants have at least 5 years of operating experience, so this approach analyzes a fairly typical plant.

Figure 3 indicates that the mean repair rate estimates for Dresden 1 and all plants except Dresden 1 are well within 2 standard deviations of each other and can therefore be lumped together. In each piping system the Dresden 1 failure rates, rather than being different, appear to be only farther along on the same curve (as befits the plant's age). Note that this conclusion differs from the failure rate studies in NEDO 21000, where Dresden 1 was separated from the other plants. The primary reason for the different conclusion is probably the fact that the comparisons made here are for each piping system, so differing rates between piping systems do not confound the comparison. The results show significant differences in failure rates (4σ and more) between different piping systems, so it is essential that they be separately considered. Figure 4 compares cumulative repair distributions for two piping systems that are clearly failing at different rates. The recirculation bypass system, in particular, has a higher failure rate than all other system failure rates combined.

As can be seen from Table 1, the repair rate appears to increase with age for most systems. Where the shape factor, α, of the Weibull curve is equal to one, as for the core spray and control rod drive systems (combined), the repair rate is constant. If $\alpha > 1$, the rate is increasing with plant age. Although Table 1 indicates there <u>is</u> an aging problem, this conclusion must be tempered by two observations. First, these are repair rates, not cracking rates, and it is not clear that the shape factors are the same. New curves are currently being generated for failure defined as a leak and failure defined as an UT indication, but these are not available yet. Secondly, since 1975 the industry has been inspecting more welds and inspecting more closely than prior to that time. Hence, not only it is likely that more cracked welds are being found, but inspections are also more likely to find small crack indications, and such indications force an early repair that would have been delayed with less rigorous inspection.

An economic and probabilistic model was developed for use in the sensitivity analysis (Figure 5). The purpose of the economic model is to define the breakeven cost, or the cost a utility should willingly pay now for a countermeasure that reduces the expected future repair cost. The maximum justifiable cost is equal to the expected reduction in cost of future repairs, using discounted cash flow analysis. The cost reduction in turn depends on the reduction in repair rate, or effectiveness of the countermeasure. The details of this analysis are in (2).

Table 1

Trends in Repair Rate by System

PIPING SYSTEM	FITTED WEIBULL SLOPE, α	REPAIR RATE TREND WITH TIME(TO 1/80)
RECIRCULATION BYPASS	2.1	INCREASING
CORE SPRAY AND CONTROL ROD DRIVE	1.0	CONSTANT
REACTOR WATER CLEANUP	1.8	INCREASING
ALL OTHERS LUMPED TOGETHER	1.9	INCREASING

The cost of a pipe crack repair is the sum of direct repair costs and unavailability. Unavailability is defined as the cost of obtaining replacement power when the nuclear plant is not operating, and it is the single greatest cost factor. To estimate repair time for the model, historic repair times were identified in the data base by piping systems and by extent of repairs. Three basic repair modes were identified: (a) grind out the cracked weld and re-weld; (b) cut out a section of pipe and weld in a new section; or (c) replace or remove a significant portion of the piping system. Forty-four repair incidents were examined in detail, using shutdown times and descriptions from NUREG 0020 (5), Licensee Events Reports (6), and direct contact with the cognizant plant personnel.

Based on trends from these examinations, the following pipe repair model was developed. All repairs are assumed to be quick repairs (grind-and-weld or replace section), and an average repair time of 6.4 days is used for all sizes of pipe in all piping systems and for both forced outages and repairs during scheduled down time. System replacement is considered a countermeasure, not a repair, even though it may be implemented in response to a repair need. These assumptions ignore the very real effects of inaccessibility, contamination, and increasing difficulty of repair with pipe size; they are believed to lead to underestimates of the true repair cost.

A very important observation is the relatively low frequency of forced outages. In most piping systems, only 13% of the repairs are performed during forced outages. The Reactor Water Cleanup (RWCU) System differs significantly (<0.05 level) from that pattern, with 43% of the repairs in that system performed during forced outages. One possible explanation for this result is that much of the RWCU System is outside of the containment, where it may be inspected during operation and where leaks are visible. The other systems are mainly inside containment, and leaks or cracks are found only when containment is entered during refueling or other major outages.

The life cycle repair costs for each piping system are shown in Figure 6, based on the mean failure rate. The two most expensive systems, in terms of future repairs, are recirculation bypass and reactor water cleanup. Since many reactors have removed or replaced the recirculation bypass system by now, the results for that system are primarily of historic interest. Core spray and control rod drive together account for less than 5% of the repair cost. This is partly because of differences in failure rates, but a major factor in the expected repair cost is the fraction of forced outages. The reactor water cleanup system accounts for 32% of the repair cost but only 14% of the repair actions, because a higher percentage of cleanup system repairs cause forced outages.

General Electric has proposed several countermeasures, or "target line" programs, that involve removal or rerouting of particular lines, replacement of core spray piping with less sensitive material, and improved welding techniques and post-weld treatments. If the target line program is 100% effective, eliminating all future repairs in the affected lines, the value of subsequent countermeasures is substantially reduced. A realistic expectation is that the target lines program would virtually eliminate repairs in recirculation bypass piping (by removal), and would cut the repairs in core spray and control rod drive systems to half of their current (already low) incidence. The reactor water cleanup system is not included in the current GE program, though Figure 6 suggests that it should be. If there is no improvement in reactor water cleanup repair rates, elimination of IGSCC in bypass piping, and 50% improvement in other target lines, the resulting program is approximately 50% effective overall. Subsequent countermeasures would have to be 100% effective to reduce the cost of future repairs by the same dollar amount. The cost saving at other possible levels of effectiveness, given that the target lines program is implemented first, is shown in Figure 7. The baseline is believed to be an underestimate, and the upper and lower bounds illustrate the combined effects of different costs of replacement power, economic assumptions, repair rate and time uncertainties, fraction of incidents forcing outages, and the uncertain effectiveness of the target lines program.

The primary conclusions of the cost effectiveness study are that the rate of repair increases with plant age in most piping systems, the repair rates differ significantly for different piping systems but not for different plants or BWR generations, the reactor water cleanup system is expected to be the primary source of future repairs, and fairly expensive countermeasures may be cost effective. In addition, past arguments for using the number of long shutdowns as the exposure variable are defeated by the observation that shutdowns are correlated with plant age, so it doesn't matter which variable is used. Other conclusions are discussed in (2), including the fact that the uncertainty in this analysis is large, and that the in-service inspection program appears to have cut the cost of repairs by 75% from a theoretical non-inspection case.

IGSCC MODELING PROJECT

For a more detailed analysis of particular piping, to support a decision whether to replace it in advance of cracking, the cost-effectiveness model described above is too general. For these detailed studies, a predictive model is being developed to estimate the time to failure by IGSCC in 304 stainless steel under field conditions of stress, material condition, and oxygen/temperature environment.

The purpose of the work is to modify and calibrate available stress corrosion models to obtain a single model for predicting crack growth rate under any combination of the three primary cracking factors: load, environment, and material sensitization. Available models include an oxygen and temperature model by Eason (11) and the stress/time/strain rate models by Garud and Gerber (12), and (13). Each model is incomplete, treating in detail only one of the three primary factors.

A multiplicative model has been developed which uses crack length (rather than failure time) as the fitted variable. There is logical support for a multiplicative model; it has been known for some time that eliminating any one of the three major factors would essentially stop the stress corrosion process. In an additive model, eliminating a factor would merely reduce the rate of attack by the ratio of that factor to the sum. Another justification for the multiplicative model is simply the success with which it fits the data.

The preliminary form of the model is

$$a = c_1 P_a^{\,c_2} [1 - c_3 \exp(c_4 O_2)]\ [2T + c_5 (T-50)^2 + c_6 (T-50)^3]\dot{\varepsilon}^{\,c_7} t \qquad (1)$$

where a is the final intergranular crack length, c_i are numerical fitting constants, P_a is the EPR material sensitization measurement, O_2 is the bulk oxygen concentration, T is the bulk temperature, $\dot{\varepsilon}$ is the bulk strain rate, and t is time. An Arrhenius temperature term $\exp(c_5/T)$ has also been tried, producing a slightly inferior fit. For constant extension rate test data, the strain rate is taken to be the extension rate divided by gage length. For constant stress tests, strain rate is estimated by a creep rule in the form

$$\dot{\varepsilon} = c_{10}\, \sigma^{c_8}\, t^{c_9} \tag{2}$$

where σ is the initial applied stress. Full details and preliminary values for all constants are given in (3).

The preliminary model has been calibrated over a broad range of conditions in constant load and constant extension rate laboratory tests. While nearly 300 data points were found in the literature, many of the observations were incomplete (e.g., missing material sensitization data or final crack length). A reasonable fit was obtained by fitting to only the complete observations (117 data points) and imposing constraints on the predicted crack length where the actual length is unknown. The author's computer code SURFIT was used for the nonlinear least squares surface fit.

Preliminary results indicate that the fitted function is capable of predicting IGSCC crack length to within an order of magnitude over a range of several orders of magnitude in material, environment, and loading variables. This is illustrated by Figure 8, which shows a plot of residuals vs. measured crack length over the range of data.

The points labeled as Berry's CERT data and the point at 0.2 inch crack length are believed to be outliers because of uncertainties in material sensitization and stress level. The other data are fitted as well as can be expected, considering the inherent variability in stress corrosion cracking results, even with duplicate specimens under identical conditions.

Selected term-by-term comparisons between data and the preliminary model are shown in Figures 9 and 10. The data in these figures were not used to determine the shape of the terms in the model. Consequently, Figures 9 and 10 provide partially independent verification that the model has an appropriate shape in each term. Other figures in (3) show that the shape of the other model terms agrees with the fitting data. So far no calibration or verification has been attempted using field data, but work is continuing on this project.

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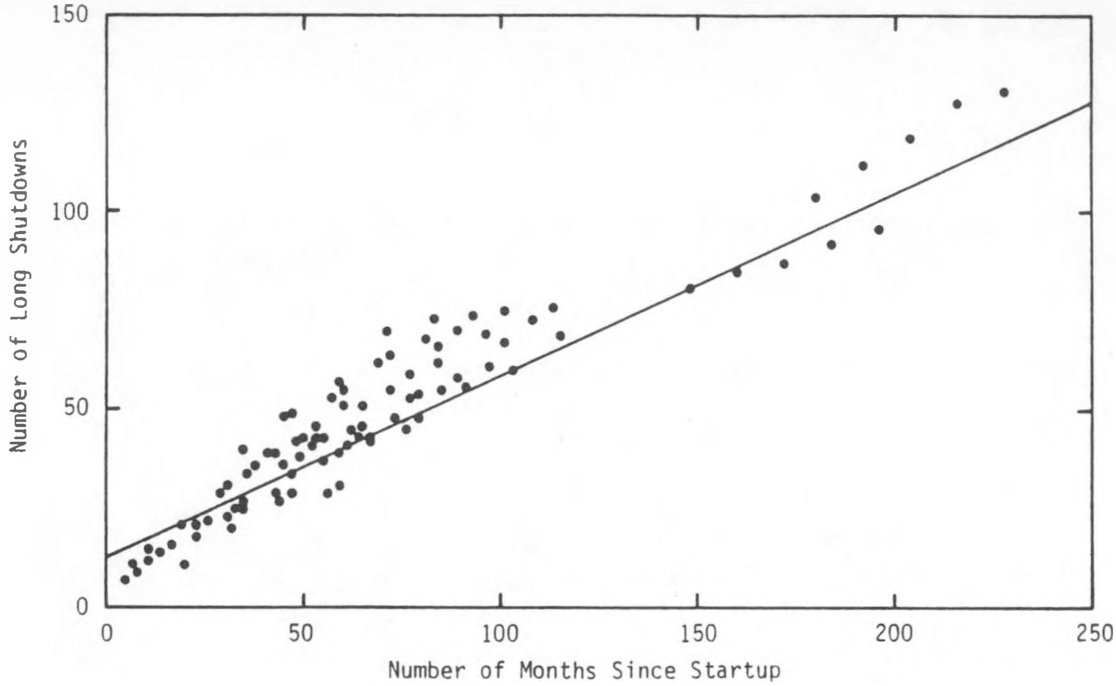

Figure 1 - Correlation Between Long Shutdowns and Calendar Time for 18 Plants Between 1975 and 1979.

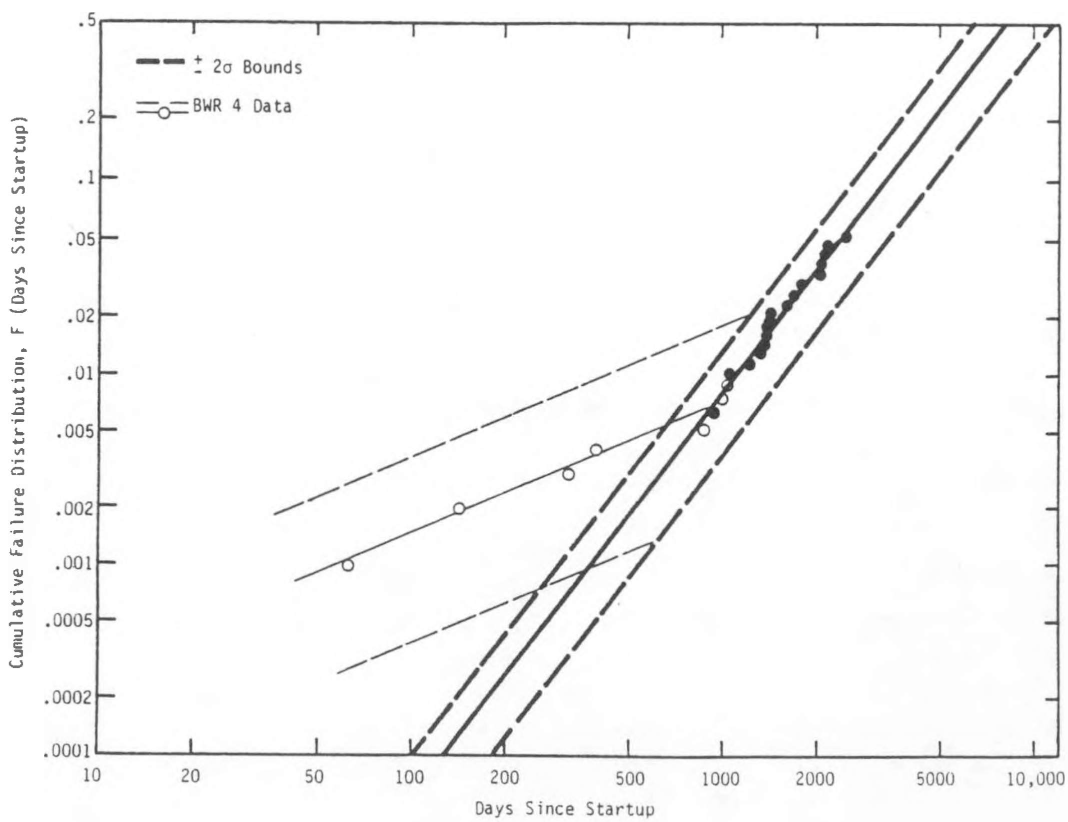

Figure 2 - Recirculation Bypass Failures, All Plants.

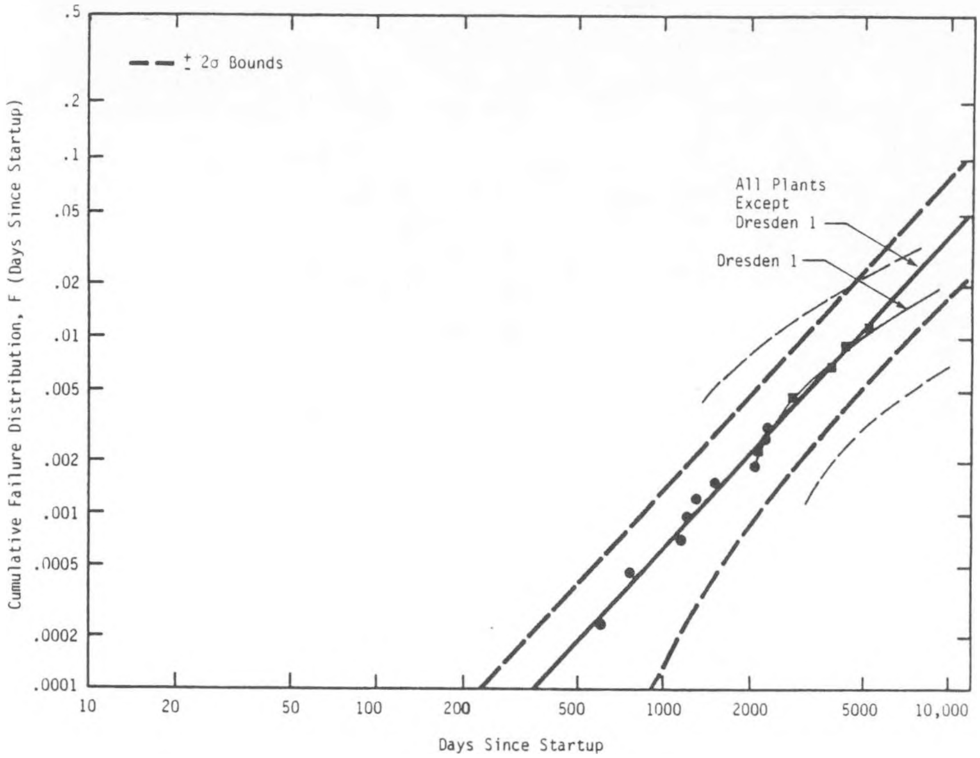

Figure 3 - Comparison of "Other System Failure Rates for Dresden 1 and All Plants Except Dresden 1.

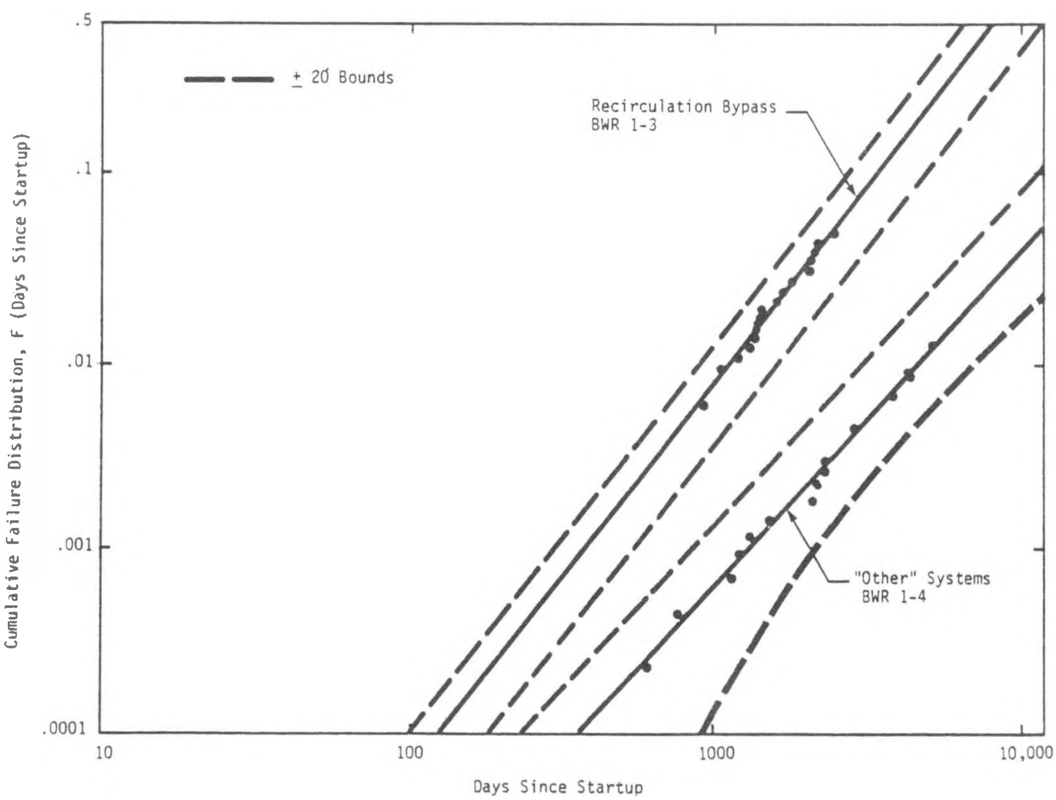

Figure 4 - Comparison of System Failure Rates.

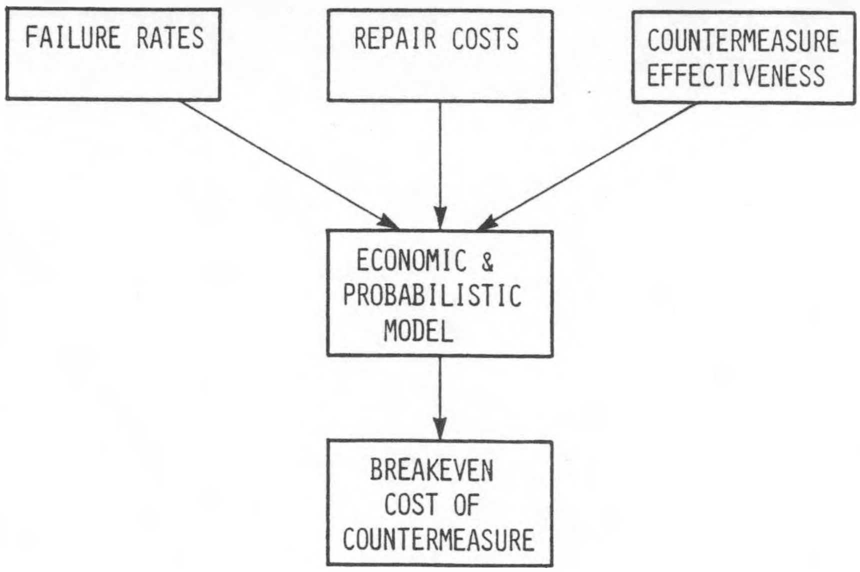

Figure 5 - Sensitivity Model.

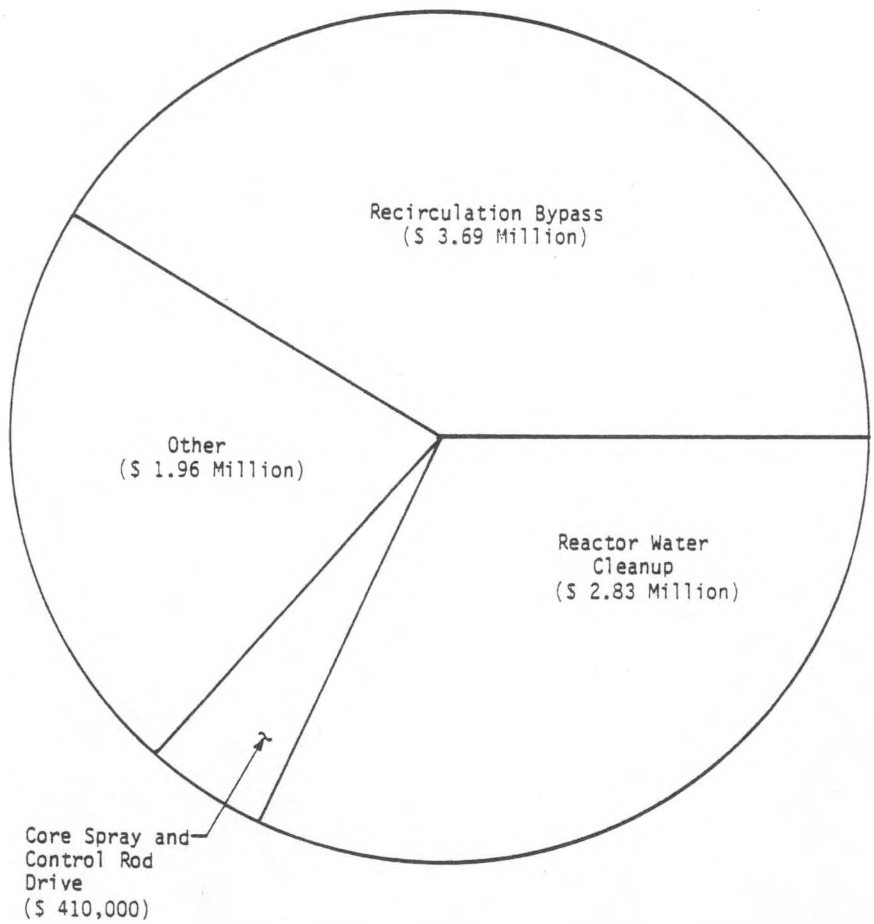

Figure 6 - Present Value of Expected Repair Costs by Piping System. Baseline Economics, Mean Failure Rates. Costs are Believed to Underestimate the Cost for a 1,000 MW Plant with Replacement Power at $218,00 Per Day.

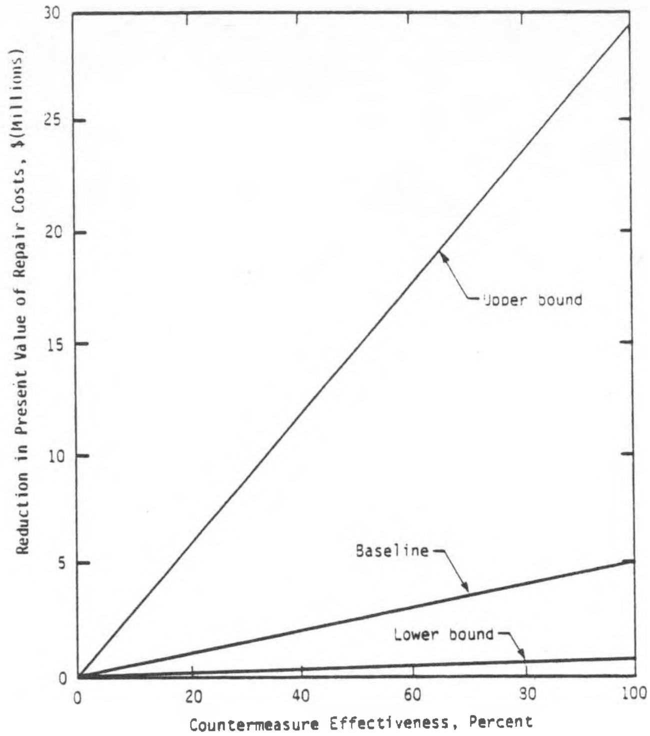

Figure 7 - Value of a General Countermeasure, Given That a 50% Effective Target Lines Program is in Place (Baseline Assumption). Probable Upper and Lower Bounds for the Combined Effect of All Uncertainties.

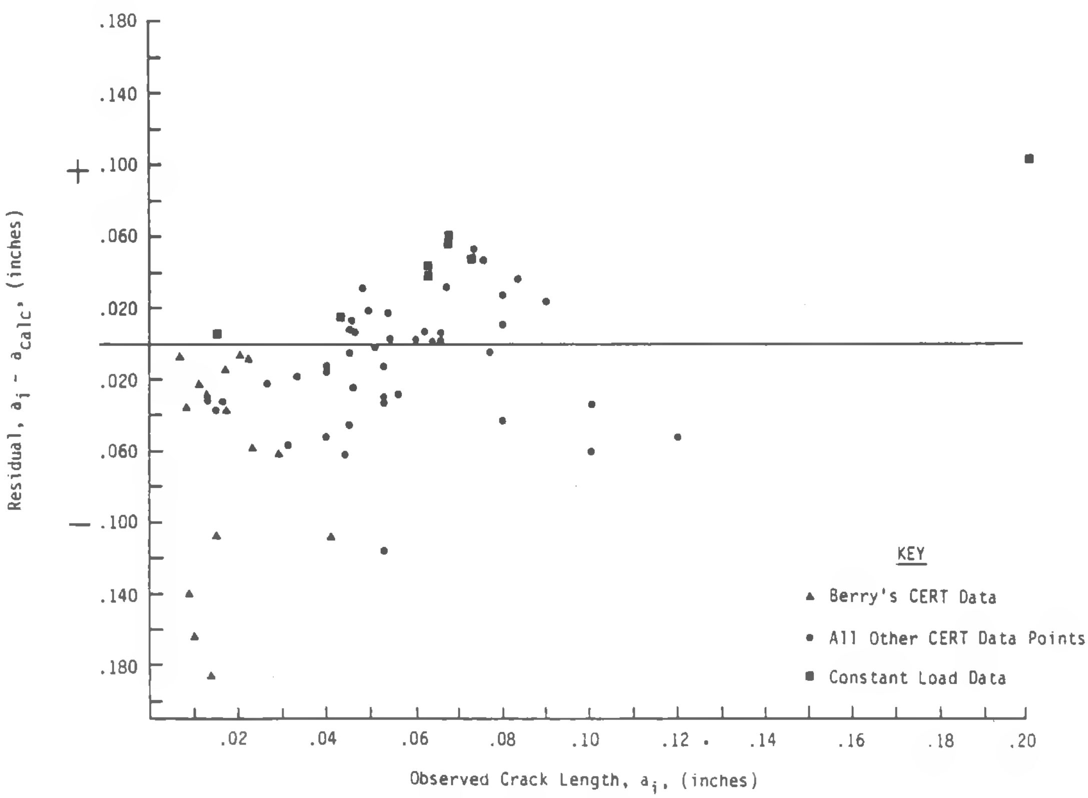

Figure 8 - Error in Model Prediction vs. Observed IGSCC Crack Length at Failure.

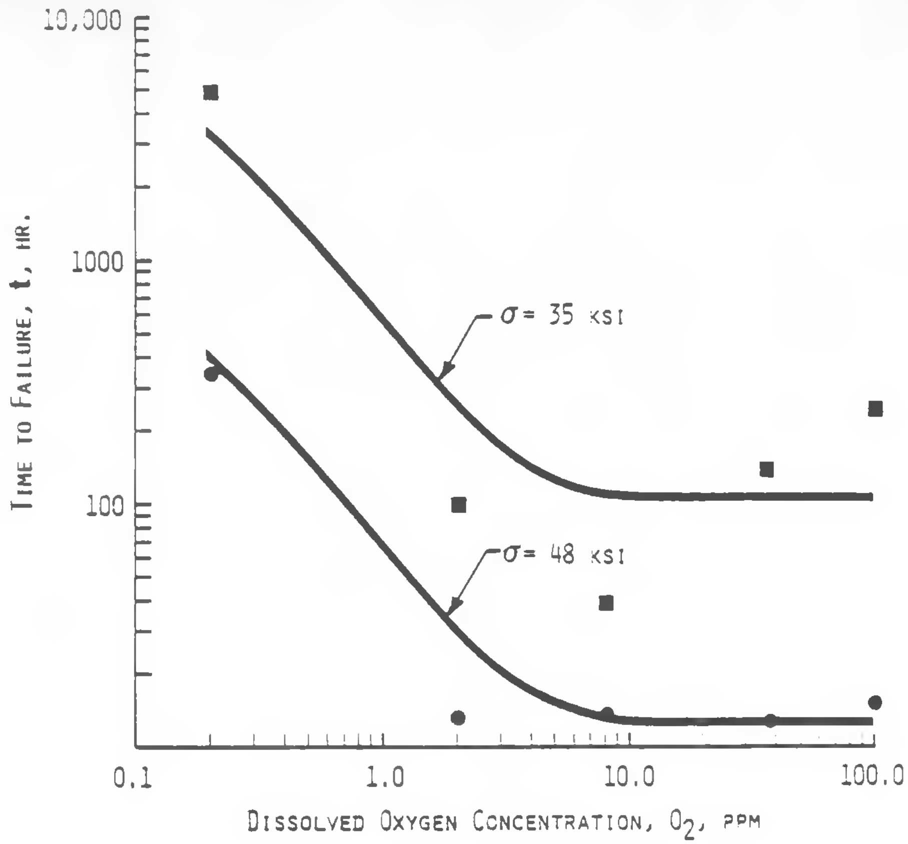

Figure 9 - Comparison of Oxygen Term and Constant Load Data at 288°C, P_a = 40 C/cm^2, a = 0.01 Assumed.

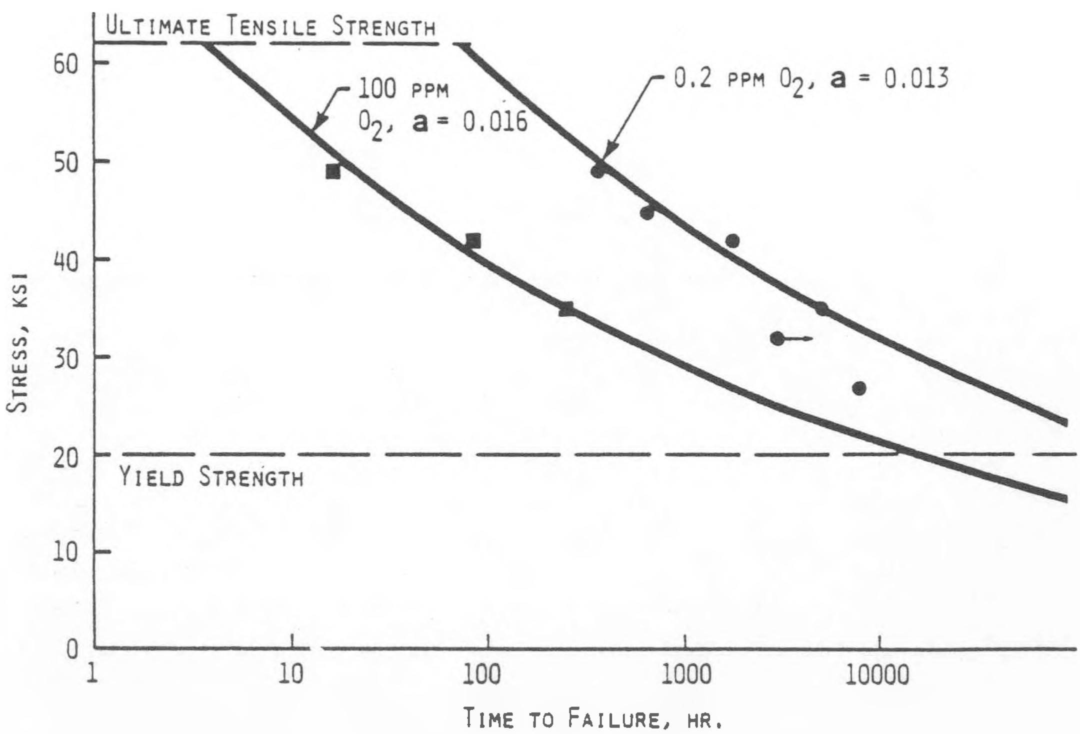

Figure 10 - Comparison of Stress Term with Furnace-Sensitized Constant Load Data, P_a = 40, T = 288°C.

OVERVIEW OF SYNERGISTIC AGING EFFECTS

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ABSTRACT

Proper, technically defensible qualification of materials and equipment for nuclear power facilities requires that the effects of combined environment exposures be addressed. The full significance of synergistic effects resulting from combined stresses still remains largely an "unknown" to be provided for by use of conservatisms, allowing a sizeable margin in test programs and analyses to account for possible combined effects. However, these margins, when applied to sequential aging tests, may under- or over-estimate the "qualified life" of the material or equipment. Experimentation with radiation dose-rate effects, simultaneous vs. sequential ordered exposures, and other combined environment testing are highlighted in this paper to provide an overview of the current state-of-knowledge concerning synergistic effects and their significance to qualification programs.

INTRODUCTION

Synergistic effects are those that result from two or more stresses (in the broad sense of the term) acting together, as distinguished from the effects of the same stresses applied separately. In many cases, an adequate understanding of the individual stresses has yet to be established. As such, it is difficult to provide a rigorous quantification of synergistic effects in the wide variety of materials that are used in safety-related equipment.

During the period between the day when the material was first produced and the day when a particular design basis event (severe earthquake or accident) is postulated to occur, the various materials in an item of equipment -- and the equipment itself -- are said to "age" as a consequence of exposure to environmental conditions and of operation. Examples of aging degradation include embrittlement and loss of tensile strength of nonmetallic materials as the result of exposure over time to ambient temperature, nuclear radiations, moisture and humidity, and chemicals in the surrounding atmosphere; loss of insulation resistance as the result of the same exposures just listed, plus voltage gradients;

electrical contacts becoming pitted, motor bearings becoming worn, vibrations causing connections to loosen, and high currents causing overheating.

Although some environmental exposures may result in temporary improvements of physical properties, the long-term consequences of environmental and operational phenomena is degradation of performance characteristics. The key questions are, of course, 1) Does the amount of degradation become significant before the licensed life of the power plant is reached?; and 2) How can the actual aging phenomena be simulated in a qualification program to produce the same amount of degradation in the equipment as it would actually experience? While neither of these questions are answered in this paper, they provide a framework by which to measure aging degradation caused by single and combined environmental stresses.

BACKGROUND

Traditionally, synergistic effects have been considered to be "second order" phenomena and thus have not been a primary focus of attention. The major emphasis has been to establish the qualified life of organic materials used in nuclear applications through a series of essentially single-environment accelerated aging tests. These test programs attempt to consider each stress factor in isolation by exposing the material to a specific environment stress at a level commensurate to the exposure received throughout the installed life of the material in the nuclear facility. This ideal is seldom achieved in practice, however, because the material being exposed is at some temperature above absolute zero (usually room temperature or above), and usually there is air and some water vapor surrounding it. Single-environment accelerated aging programs do not simulate the actual operating conditions of the material/component because no item in a nuclear installation is kept in one environment, isolated from simultaneous stresses. For this reason, the qualified life of a material calculated via this method may overestimate or underestimate the actual capacity of material to tolerate certain stresses. Accelerated aging programs must account for all significant stress factors and calculate the degradation caused by combined environments either through simultaneous or sequential exposure to two or more stress-producing environments, vibration, etc.

Since definite knowledge of synergistic effects for all organic materials is still evolving and no comprehensive aging program to account for them has been developed to date, synergisms are referred to as "potential" in most documents on equipment qualification. NUREG-0588 -- "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"[1/] -- provides for the inclusion of synergisms in an aging program for plants qualified according to IEEE Std 323-1974 only if evidence of known synergistic effects exists.

Otherwise, thermal aging evaluations conducted on pre-irradiated or concurrently irradiated samples are suggested. Sequential testing with adequate margins is considered sufficient to account for synergistic effects.

IEEE Std 323-1974 allows for the addition of radiation to other known degrading influences in testing and specifies that total radiation test equivalent must account for oxidation gas-diffusion effects. Formulas for calculating equivalent doses were originally promulgated by IEEE Std 278-1967 (since withdrawn) and ASTM D2953-71 (limited applicability). Where oxidation occurs in thin insulating materials, accelerated radiation testing will not adequately simulate the total degradation occurring over many years at a lower rate.

In the NRC's current rulemaking on qualification, the specific wording of the requirements concerning synergisms have been changed a number of times in the various drafts prepared by the staff for consideration by the Commission and the ACRS. (As of this writing, the final text of the rule has not been established.)

CONSIDERATION OF DOSE RATE EFFECTS

From early experiments with the irradiation of organic polymeric samples either in a vacuum or under inert atmosphere, the notion grew that material damage was determined solely by the total integrated dose, ignoring the effects of dose rate. Much evidence was generated to support this idea and it appeared generally true for all known cases of material irradiation in the absence of oxygen. Integrated radiation dose levels found to be tolerable for various materials were published based on the assumption that material degradation under environments of low level radiation over long time periods could be simulated under high dose rate conditions by simply equating the integrated dose. Dose rate effects were ignored and this was widely accepted as part of the philosophy of aging test design for the nuclear industry.

It was not until recently that experimentation revealed the common occurence of dose rate (or time) effects promoted by the presence of oxygen. Subsequent experiments at Sandia National Laboratories, the Japan Atomic Energy Research Institute and the Naval Research Laboratory confirmed the existence of the synergism known as "dose rate effect," but which in reality may be a "time effect". A sample of these studies is provided in Table 1.

Aging is generally complicated by 1) multiple reactions important to degradation, 2) diffusion effects (such as oxidation), 3) sorption effects - both permanent and condensable gases. Sandia National Laboratories promotes use of the kinetic rate approach to describe synergisms since it accounts for all three issues.[10/] Multiple degradation reactions above and below a transition (e.g., a crystalline melting point or glass transition temperature of a polymer) can be quite distinct

Table 1

DOSE RATE EFFECTS STUDIES

INSTITUTION	MATERIALS	TEST TYPE*	FINDINGS
Sandia National Laboratories[2]	• Chloroprene Chlorosulfonated polyethylene Silicone • Ethylene propylene Cross-linked polyolefin	SIM	• Small dose rate effects; total integrated dose more significant • Strong dose rate effects
Sandia National Laboratories[3]	• Polyvinyl Chloride	SIM SEQ	• Varied results SIM test approximates actual 12 yr aging
Sandia National Laboratories[4]	• Low-Density polyethylene • Polyvinyl chloride	SIM SEQ	• Strong dose rate effect; significant synergisms
Sandia National Laboratories[5]	• Ethylene propylene Cross-linked polyolefin • Chlorosulfonated polyethylene Chloroprene	SIM	• Dose rate effects important for all
Sandia National Laboratories[6]	• Ethylene propylene	SIM SEQ	• Some sensitivity to ordering of SEQ tests; significant SIM test results
Naval Research Laboratory[7]	• Polytetrafluoroethylene • Polyamide Polyvinyl formal Polysiloxane	SIM SEQ	• Strong synergisms in SIM test • SEQ test more severe; SIM test produced "beneficial" crosslinking
Japan Atomic Energy Research Institute/Hitachi Cable Co.[8]	• Cross-linked polyethylene Ethylene-propylene terpolymer Chlorine (polymeric)	SIM	• Varying but significant dose rate effects
Japan Atomic Energy Research Institute[9]	• Polyethylene	SIM	• Oxidative degradation proportional to total dose

*SEQ = sequential; SIM = simultaneous.

and extrapolation through a transition point may be invalid. Diffusion effects are best recognized when the degradation rate varies with sample thickness; reducing sample thickness minimizes diffusion effects. Diffusion effects are indicated by a low activation energy or by an activation energy which decreases as temperature is raised. Sorption concentrations are temperature dependent -- lower temperatures retard sorption of gaseous substances. Since sorption effects under accelerated and ambient conditions are seldom known, accelerated experiments use permanent gas concentrations representative of ambient conditions. Sorption effects of condensable gases are even more complex since they are dependent upon partial pressure, as seen in polyurethane experiments at various temperature-humidity combinations.

Dose-rate and oxygen effects in polymer degradation confirmed the basic theory that inhibited gel formation (i.e., polymer degradation) varied directly with the square root of environment oxygen pressure and in an inverse manner with the square foot of the dose rate for given polymer thickness and integrated radiation dose (constant). The studies cited in Table 1 examine this relationship as well as the ordering of sequential radiation-thermal tests and the results of simultaneous vs. sequential testing.

SIMULTANEOUS VS. SEQUENTIAL STRESS EXPOSURES

Campbell pioneered work in comparing simultaneous vs. sequential testing of 5 polymer-based magnet wire insulations at Naval Research Laboratory.[7] With one exception (namely, polytetrafluoroethylene) sequential radiation-thermal exposures proved more damaging than simultaneous exposures. This was attributed to the "beneficial" cross-linking in polymers which initially occurs during irradiation. Subsequent experimentation at Okonite[11] revealed that while cross-linking initially occurs, chain scission follows as radiation exposure continues, resulting in total degradation. Okonite tested 13 elastomer-based insulation-jacket combinations, determining the threshold of damage and highest dose rate sustainable for each specimen.

Radiation-thermal accelerated aging tests at Sandia have determined the proper procedure for sequential exposure of polymers -- radiation followed by heat -- in order to approximate the simultaneous exposure experienced during normal operation.[4] By this method, Clough and Gillen found tolerable radiation doses for PE and PVC to be lower than those prescribed by single-environment aging. Real-time aging and simultaneous accelerated aging tests reveal even lower radiation tolerance doses. For this reason, combined-environment aging approximated by sequential exposure tests must allow for potential synergistic effects with adequate margins.

Additional experimentation with ethylene propylene rubber (EPR) insulation materials at Sandia National Laboratories was recently published by Larry Bustard.[6] He aged six unidentified EPR materials at elevated temperature and radiation stress exposures common to cable LOCA qualification tests. The ordering of sequential tests established for PE and PVC held for certain EPR samples and the severity of simultaneous over sequential testing was proven as well. Table 2 provides the results of 6 tests applied to one sample as measured by ultimate tensile strength and elongation. This study corroborated the findings of the Japan Atomic Energy Research Institute (JAERI) and Hitachi Cable Co., Ltd., in their dose rate experiments with EPR, cross-linked polyethylene (XLPE) and some chlorine containing polymeric materials.[8/] JAERI has developed a model to evaluate the extent of cross-linking and scission in polyethylene and plans to apply it to other polymers.

OTHER COMBINED-ENVIRONMENT TESTING

Accelerated aging tests are investigating the synergistic effects caused by other combined environments -- humidity, vibration, steam, chemical spray, etc. -- and have yet to determine these for all organic materials. For example, Franklin Research Center (FRC) under contract to Bell Laboratories tested 24 electric switching devices to determine their vulnerability to malfunction due to vibratory stresses in the range of seismic frequencies and acceleration amplitudes.[12/] The study included a vibration test before and after an accelerated aging program designed to simulate 40 years of service outside containment of a nuclear plant. Gamma irradiation, thermal aging combined with near-100% humidity, electrical/mechanical cycling and simulation of DBE were included in the accelerated aging program. Specimens passed all inspections and functional tests with minor exceptions occuring after gamma irradiation: discolored plastic materials, and three inoperative devices due to fracture of Celcon and Delrin as a consequence of crystallization and embrittlement.

Extensive testing of elastomeric materials at various temperature-humidity-radiation combinations has been done at Sandia National Laboratories, using ultimate elongation as the damage indicator.[13/] Humidity effects were determined to be insignificant; however, combined thermal and radiation environments caused synergisms of various magnitudes for certain polymer materials. This was not the case with thermal aging experiments performed in Finland on polyester imide and amide-imide overcoated polyester enamels used on copper and aluminum wires.[14/] Paloniemi and Lindstrom found large humidity effects for the samples; however, since no humidity control was included in the experiments, no conclusions may be made.

Sandia National Laboratories is testing a number of materials with and without fire retardants to determine changes in flammability after combined-environment exposure. Sandia has concentrated most of its effort in evaluating possible synergisms occuring during the aging of extruded plastics (PE and PVC), silicone, polyolefin, and EPR. Sandia is planning to test seals, gaskets and connector assemblies for material degradation later this year.[15/]

Table 2

RELATIVE TENSILE PROPERTIES OF EPR-1483 AFTER AGING

Aging Method	Center of Chamber Dose Rate in EPR (krd/hr)	Center of Chamber Total Dose in EPR (Mrd)	Ultimate Tensile Strength T/T_o	Ultimate Tensile Elongation e/e_o
1. Unaged	0	0	1.00 ± .05 (10.6 ± 0.5 MPa)	1.00 ± .09 (340 ± 30%)
2. Simultaneous 30 day radiation and thermal exposures	60 ± 4	43 ± 3	0.79 ± .07	0.41 ± .10
3. Sequential 28 day thermal then radiation exposures	65 ± 5	44 ± 3	0.98 ± .07	0.47 ± .10
4. Sequential 28 day radiation then thermal exposures	65 ± 5	44 ± 3	1.01 ± .10	0.41 ± .05
5. Sequential 28 day thermal then 55 hour radiation exposures	850 ± 60	47 ± 3	0.97 ± .08	0.35 ± .04
6. Sequential 55 hour radiation then 28 day thermal exposures	850 ± 60	47 ± 3	0.93 ± .06	0.32 ± .04
7. Simultaneous 7 day radiation and thermal exposures	290 ± 20	49 ± 3	0.83 ± .06	0.41 ± .05

NOTES: (1) Errors reflect one standard deviation of three measurements.
(2) Insulation thickness is nominally 2.8 mm.

Source: Reference 6.

JAERI has developed test facility at its Takasaki Radiation Chemistry Research Establishment in which cables and small electrical components can be simultaneous and sequentially exposed to radiation, steam and chemical spray.[9] Preliminary experiments on insulating materials found sequential testing (radiation followed by steam and chemical spray) for more damaging than simultaneous testing. Further experimentation with sequential ordering may provide different results.

Outside of the nuclear field, a serious reduction in the life of polyethylene (PE) and cross-linked polyethylene (XLPE) cable has been observed.[16] Effects of accelerated aging exposure time and moisture are highly pronounced, and after more than 3 years of research the fundamental cause of the problem has still not been identified.

CONCLUSIONS

The findings and results of combined-environment aging tests summarized above do not represent a comprehensive treatment of the subject of synergisms. All test programs have not been included since much experimentation is still in the preliminary stages and a number of U.S. firms and French organizations consider their work in this realm to be proprietary in nature. What we have attempted to show is simply an overview of the current state-of-knowledge concerning synergistic effects.

Combined-environment accelerated aging programs are currently being developed based on the results of experiments performed and reported by electrical equipment manufacturers (such as Okonite and Hitachi) and research institutions (such as Bell Laboratories, Naval Research Laboratories, Sandia National Laboratories, and the Japan Atomic Energy Research Institute). Their work leave no doubt that for some materials the combined effects are significantly more damaging than the separate effects of environmental parameters.

For those materials and/or equipment in which synergistic effects appear significant, research must be continued to quantify the extent of combined-environment stresses and to then incorporate these parameters into qualification programs. Synergisms, in and of themselves, are not likely to be a major concern for the nuclear industry; rather, they constitute an aspect of the overall qualification procedures that must be addressed. The NRC, DOE and nuclear industry should increase the level of funding for research by a few orders of magnitude. In the interim, qualification programs should be conservative by using appropriately large margin values when estimating the "qualified" life of materials and equipment that are important to safety.

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A STATE REGULATOR VIEWS THE AGING OF NUCLEAR POWER PLANTS

by

JAMES B. KEATING

ABSTRACT

This report provides the author's views of the current status of operating nuclear power plants as they are affected by replacement or major repairs of large components. Opinions expressed in this report are subjective, based on the author's experience with regulation of utility companies in New York State. Costs associated with long outages required for replacement of major components are mentioned, as well as areas where attention should be given to help achieve forty years of design life for nuclear plants.

Why would a person responsible for utility company matters be concerned with specific information pertaining to aging of nuclear plants?

New York State law requires that every electric corporation under the jurisdiction of the Public Service Commission furnish safe and adequate service at reasonable charges.

The law additionally states that every unjust or unreasonable charge made or demanded for....electricity or any service charge or in connection therewith or in excess of that allowed by law or by order of the Commission is prohibited. The Commission must therefore be knowledgable about the actual conditions which exist in the electric plants in order that all necessary provisions be taken to satisfy the law and to provide protection for the electric customers of New York State. In addition, a utility company is permitted a return on its investment as long as the equipment is in its rate base. The expected life of a nuclear plant is forty years. It is highly desirable that a nuclear power plant continue to function at full capacity during the entire period of expected life.

Will nuclear power plants be capable of supplying full capacity for the expected forty years of design life? This question is being given attention these days, especially since it now takes over ten years to design and construct a nuclear generating plant.

A nuclear plant, like any thermal electric system, can supply power only as long as its component parts operate in a proper manner. In other words, a plant must function in accordance with actual design criteria. Since every plant is designed with some limitations depending on the capability of a specific component in the system, any degradation of plant equipment may limit generation output as well as affect overall efficiency of operations. Any reduction in plant output translates into additional costs which must be absorbed by electric customers.

Longevity of various components that are part of a nuclear plant is currently receiving a great deal of attention; and this has been clearly demonstrated at this workshop. In an era when many nuclear power plants in the United States are approaching their first ten years of operations, there are active programs of replacing major equipment items and parts that have become defective. More specifically, there are utility companies that have nuclear plants currently involved in replacing steam generators in pressurized water reactors, replacement of large bore recirculation system piping in boiling water reactor plants and parts of secondary systems, namely, tubes of feedwater heaters and main condensers. Replacement of the steam generators of the Virginia Electric and Power Company's Surry units and Florida Power and Light Company's Turkey Point units and more recently replacement of reactor recirculating piping at Niagara Mohawk's Nine Mile One, are examples of major refurbishing projects costing many millions of dollars and requiring outages in excess of nine months for each plant. It is estimated that replacement of the steam generators at both Turkey Point units will cost $160 million, with costs for backup power in excess of $700,000 per day.

Since such repairs and maintenance activities require extensive planning, preparation of procedures and following such plans very carefully, it would benefit owners of similar nuclear plants to monitor activities very attentively so that outage periods and associated costs might be reduced if similar projects are necessary to be undertaken in the future.

The radioactive environment at some parts of a reactor plant provide a hostile atmosphere for personnel performing their necessary tasks. A radioactive environment requires decontamination of parts of the reactor plant where personnel will be working. Detailed plans and schedules often must be prepared on short notice. This approach to plant maintenance should be revised. Procedures that are required to perform the necessary tasks should be carefully prepared and checked out on full-scale mockups of the parts being maintained or replaced well in advance of the time that actual work is being done. Unique tools must be checked out and personnel trained in the use of such tools and devices outside of containment, since misuse of some special tools has caused serious problems requiring major corrective action.

Supervision of tasks associated with replacement parts of reactor plants cannot be completely delegated to subcontractors. There are too many instances of disappointing performance by outside contractors. Some less than desirable performances have been attributed to the use of poorly designed tools, spare parts on hand having wrong dimensions, the taking of wrong measurements, wrong electrical connections, undercutting of critical metal areas, and so on. Regardless of such poor performance, it is still the plant owner who retains full responsibility for plant operations and who must therefore provide control over all activities concerning tasks affecting the equipment.

Quality control procedures must consist of much more than initialling appropriate forms. It must begin by drafting precise specifications, being knowledgable about the source of supply of materials, and carefully inspecting and measuring spare parts received from manufacturers. It is difficult to ascertain whether spare parts are fabricated correctly unless the owner of a plant is supplied with detailed information, usually drawings of the parts in question. Manufacturers have been reluctant to supply such information, claiming that it is proprietary.

The reason for replacement of major components in nuclear reactor plants currently in operation is primarily the corrosion of metal surfaces, resulting in reduction in thickness of pressure barriers between primary and secondary systems or between the primary system and the surrounding environment in containment. Therefore, we are mainly changing out equipment due to corrosion and have not yet felt any strong impact from aging of materials. Because of this type of degradation, the condition of steam generators of pressurized water nuclear plants located in New York State was recently evaluated. Corrosive action has caused the tubes of the steam generators to leak, requiring plugging and more recently, sleeving. Plugging removes the available tube surface from future use, while sleeving permits using defective tubes for future operations, thereby restoring some longevity to the life of the steam generators. It was somewhat surprising that the condition of the steam generators for each of the three plants evaluated was substantially different, in spite of the fact that two of the plants are supplied with cooling water from the same source.

An evaluation did provide the following recommendations, which should lead to longer life of the steam generators of pressurized water nuclear plants:

1. Techniques used in measuring the condition of steam generator tubes that prolong the life of the equipment are desirable. Any attempt to prolong the life of a steam generator tube rather than follow a program of preventive plugging is also highly desirable. Use of a profilometer when performing eddy current testing is superior to using a fixed diameter probe coil. In addition, sleeving of defective tubes will tend to prolong the longevity of a steam generator in contrast to plugging defective tubes, especially prior to detecting leaks.

2. Since corrosion from impurities in feedwater systems is one of the major contributors to equipment failures, all efforts should be made to reduce any possible leakage of impurities into the secondary system. This means operating main steam condensers with tight constraints--that is, with a minimum amount of tube leaks. Of course, zero inleakage is the ideal goal.

3. Feedwater that comes in contact with the various metal surfaces of the secondary system must be chemically controlled so that corrosive action is minimized. Therefore, continuous monitoring of ever increasing stringent requirements must be followed as a routine program.

4. Since operating experience continues to uncover information on materials currently in plant systems that contribute to corrosive action, long-range programs should be established to replace parts of equipment with parts fabricated from more corrosion-resistant materials.

5. Since the existance of oxygen in secondary systems has significantly contributed to corrosion activity, reduction of the oxygen in feedwater by mechanical means should be given every consideration. New pressurized water nuclear plants should consider utilizing separate deaerating heaters while existing nuclear plants should consider the costs/benefits of changing out a surface closed type feedwater heater and replacing it with a direct contact deaerating heater.

6. When a major component in a reactor system is replaced, the new equipment should provide for ease of inspection and maintenance. A steam generator that could be fabricated with flanges rather than having the parts welded together could provide for ease of disassembly for purposes of inspection and repairs.

At this stage in the development of nuclear plants, the effects of corrosion are well documented. The consequential loss of components due to aging of materials are not as well known. However, the results of aging of materials could affect nuclear plants in a similar manner as failures due to corrosive action.

Because of the large capacity of most nuclear plants, the loss of generating capacity or unavailability of a unit is indeed significant. Charges that may be assessed to an outage of a nuclear plant consists of fixed charges on the capital investment which continue regardless of whether or not the plant is operating, the costs for replacement power, usually from a source using fossil fuel, and, of course, costs for actual work performed during the outage. Eventually all of these costs must be borne by the ratepayer.

The capital investment in nuclear plants today requires over ten times the costs for such plants ten years ago, with corresponding fixed charges; and at the same time, the costs for replacement power from a fossil fuel generating station are over ten times what they were ten years ago. Since most utilities have provisions in their tariff to use a fuel adjustment clause, the higher costs of fossil fuel for backup power are passed through to electric customers

When a utility company is faced with spending over $4,000 per kilowatt for installed nuclear capacity and must pay out replacement power costs of close to $1 million per day, there are economic incentives to reduce outage times.

Manhours spent in preparing for possible outages along with expenditures on useful research and testing to better understand the effects of corrosion and/or aging of materials, are clearly justified.

The concern being raised here today regarding aging of materials is a legitimate one and brings to mind the federal regulations requiring annual inspections of ships' propulsion systems. For over fifty years, the overriding question has been, what are the effects of wear, erosion, corrosion, and perhaps even aging of materials? Annual inspections still cover the status of pressure containing components and piping. Prior to the development of non-destructive testing, it was necessary to drill into the wall of piping systems to learn the actual thickness of the pipe after twenty years of ship operation. I would not recommend this same course of action today with nuclear plants since there are many available methods for learning of material conditions without cutting into the system or even disassembling the piping or particular component. However, once a system or component is disassembled, careful inspections with documentation of observations and tests may prove to be extremely beneficial to the particular plant and to the entire nuclear industry.

Our understanding of aging of materials is better today than it was in the past. Sophisticated testing methods as well as methods of analyzing the results of such tests should not only extend the life of materials but also make it easier to predict the behavior of materials.

We have observed, during the last decade of nuclear plant development, an industry capable of withstanding environmental constraints and enduring financial constraints. If aging of materials used in the the various components becomes a serious obstacle to operation of nuclear plants, we will see mass replacement of existing parts, some of which may or may not be defective. Knowledge of the behaviour of materials used in reactor systems will prevent development of such an obstacle.

Through the combined efforts of the Nuclear Regulatory Commission and the work of many high technology laboratories, it is my belief that any potential problems relating to the aging of materials will be resolved, and the general public will continue to share in the benefits of less costly electric power from nuclear plants.

TECHNICAL SESSION III

Oral Presentations

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SYSTEMS ENGINEERING APPROACH TOWARD AGING

by

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(Abstracted from the Workshop Transcript)

Industry is concerned about meeting NRC requirements for aging. There are good general criteria, but guidance is needed on specifically how industry should expend its limited resources to show that the criteria are met. A systems approach that looks at the total problem from design through operations is suggested. Process control, reliability analyses, acceptance tests, control of spares, in-service inspection and maintenance, failed equipment, data collection, and material behavior are areas that should be addressed in a systems evaluation.

WESTINGHOUSE COMPONENT AGING PROGRAM

by

R. B. Miller
Westinghouse Electric Corporation

(Abstracted from the Workshop Transcript)

The goal of the Westinghouse Component Aging program is to show whether or not there would be common mode failures of mild environment components during seismic events. The data produced is used in the licensing process. An evaluation of Westinghouse's major systems revealed that there were 4,000 to 5,000 components of interest. The program was reduced to a manageable level of 300 components by selecting representative samples from each vendor. The components are thermally and, where necessary, mechanically aged and then subjected to seismic tests. There have been no failures in the program. One hundred components have been qualified for a 10-year life. Work will be completed on 200 more components next year. It is hoped that eventually the components can be qualified for a 25-year life. The information can be used by program subscribers to establish the qualified life of components, to show there are no age-related problems, and in procuring spare parts.

APPLICATION OF THE ARRHENIUS EQUATION

by

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(Abstracted from the Workshop Transcript)

Large errors may result if in application of the Arrhenius equation the following restrictions are not observed: the equation applies only to homogeneous single-phase reactions; the plot of the equation is linear and applicable only over a narrow temperature range; and the line may not be extrapolated outside the temperature range of the test points. Extrapolation of experimental data to predict design life has consistently resulted in predictions of shorter life than that which has been observed for materials in actual service. Insistence on too much conservatism in establishing safety margins may mean that there is no equipment that can be used. There is a need for a research project on application of the Arrhenius equation and also on synergistic effects.

MAINTAINING QUALIFIED LIFE OF EQUIPMENT AND PARTS IN NUCLEAR POWER PLANTS

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Abstract

The following issues are presented in the paper:

1. Should the storage time period be a part of the total "qualified life" or should the qualified life commence after installation of the equipment or the parts?

2. Can the equipment or the parts be periodically "reformed" to arrest degradation? ("Reforming" is defined as a procedure to be followed to prolong the serviceability of the parts during storage.)

Department of Defense practice as described in the military standards, is introduced. The importance of retrieving available technical information from various industry standards and from industry's past equipment qualification related experience is emphasized.

Introduction

This paper describes certain approaches which may extend the available "qualified life" of an item (the term "item" is used to designate an item of equipment or a part of an equipment item) during storage or during in plant service. These approaches include:

- Reduction of the time period between manufacturing an item and supplying it to the plant, and

- "Reforming" of items in storage.

The use of existing information available in the industry standards and sharing of results of qualification test efforts among various users are emphasized. The purpose of this paper is to invite industry's attention to the need for coordination in the existing qualification test and analysis efforts. The paper aims at initiating additional research efforts needed to provide directions to ensure maintenance of qualified life.

Reduction of Time Period Between an Item's Manufacture and its Supply to the Plant

Expeditious delivery of an item to the plant will allow the purchaser to use the maximum available qualified life of an item. Control of the delivery time period will help to prevent the supplying of items containing materials which could have significantly degraded. Guidelines available in military standards to control this delivery time period for certain specific materials are of interest.

Military Standard MIL-STD-1523, "Age Control of Age Sensitive Elastomeric Materials," issued in 1973, requires the manufacturers:

- To mark the date of manufacture on certain elastomeric materials, and
- To deliver these parts within a specified time period to their procurement authorities.

Certain guidelines provided in this standard are useful to the nuclear power industry.

Standard 1523 applies to specific classes of synthetic elastomers. This standard requires marking of "cure dates" on the elastomeric materials. When the cure date can not be determined, elastomeric material controlled by this standard shall be rejected. "Cure date" is defined as the date when the compounded, uncured high molecular weight elastomer is crosslinked to produce an elastomeric product.

This standard also requires the manufacturer or the distributor to maintain records on the complete identity of the part, including information on the manufacturer's name, cure date, assembly date, and specifications on cure-dated material.

The maximum time period allowed between the date of "cure" of the material and the date of delivery to the procurement authority is 12 quarters (three years) for the specified materials in the standard.

"Reforming" of an Item in Storage

"Reforming" may be applied to improve the electrical insulation characteristics of degradable materials in items which are in storage or which are used in applications where the equipment is normally de-energized.

Military Standard MIL-STD-1131B, "Storage Shelf Life and Reforming

Procedures for Aluminum Electrolytic Fixed Capacitors," issued in 1979, includes procedural guidelines to be followed in determining and prolonging the serviceability of aluminum electrolytic fixed capacitors during storage. This standard applies to specific types of capacitors which are procured in accordance with the requirements of the following standards:

- MIL-C-62, "General Specification for Fixed, Electrolytic, Capacitors (D C Aluminum, Dry Electrolyte, Polarized)"

- MIL-C-39018, "General Specification for Fixed Electrolytic Capacitors (Aluminum Oxide)"

The reforming procedure is performed on capacitors which are found to have out-of-tolerance dc leakage characteristics. Capacitors whose leakage current cannot be improved to values within the specified limits are discarded. The standard also discusses an example for establishing a "reforming" test set-up which includes a variable dc power supply, an ammeter, a voltmeter, a series lamp, and a bank of capacitors to be reformed.

Reliance on Other Industry Standards and Specifications

The know-how available in numerous existing industry standards and specifications may be beneficially applied to the efforts needed to arrest aging degradation of materials and equipment. This know-how may be factored into the nuclear power industry's research programs and into the development of new industry standards related to the "aging" phenomenon. A few of the research programs which may benefit from the available technical know-how are presently being sponsored by the U.S. Nuclear Regulatory Commission, nuclear power utilities, and private industries.

Most of the existing industry standards related to equipment qualification have been developed by the Institute of Electrical and Electronics Engineers (IEEE); and a few standards have also been developed by the American Society of Mechanical Engineers (ASME). These organizations are actively involved in developing new standards related to equipment qualification.

The industry standards which may contribute useful information to research efforts or which may help the development of new standards can be divided into two categories. The first category is the standards and specifications related to the Department of Defense; the second category of standards consists of non-defense industry standards. All unclassified Department of Defense standards are listed in "The Department of Defense Index of Specifications and Standards" (DODIS); whereas, various non-defense industry standards are listed in the indexes published by the respective organizations. A few of the organizations whose standards may be of use are:

AFBMA	Anti-Friction Bearing Manufacturers Association, Inc.
ANS	American Nuclear Society
API	American Petroleum Institute
ASLE	American Society of Lubricating Engineers
ASTM	American Society for Testing and Materials
EIA	Electronic Industries Association
ISA	Instrument Society of America
NEMA	National Electrical Manufacturers Association
UL	Underwriters Laboratories

Sharing Industry's Experience

Utilities, manufacturers, the U.S. Nuclear Regulatory Commission, and other organizations in the United States and abroad have sponsored equipment qualification tests and analysis in the past few years. Presently, the outcome of these qualification tests is shared among various users through data banks or through the efforts of owner groups. The data banks are maintained and operated collectively or individually by several organizations. The owner groups are formed by utility members to share the results and expenses for qualifying certain pieces of equipment. However, the information exchange between the segment of the industry which has successfully completed certain qualification tests (or analyses) and the other segment, which could use the available information, is not very effective. Renewed efforts are needed to facilitate this information exchange. Initiative on the part of organizations which have completed certain tests successfully is required to apprise the interested parties of the information availability. It is recognized that contractual arrangements will be needed between the involved parties to share the expenses and to define requisite terms and conditions.

Conclusion

Efforts to answer the following questions should be made through the U.S. Nuclear Regulatory Commission or through industry research:

Question 1

Should the storage time period be a part of the total qualified life or should the qualified life commence only after installation of the equipment or the parts?

Question 2

Can the equipment or the parts be periodically "reformed" to arrest degradation?

The answer to the above questions may vary depending on the type of materials used in the item and the nature of the environment where the item is located. In the absence of specific guidelines, the following recommendations may be helpful:

- Loss of qualified life during storage of safety-related items located in the "mild environment" of the plant should not be a concern unless the manufacturer's instructions or results of research on items located in mild environments require otherwise. "Surveillance," "maintenance" and "reforming" of items, if required, should be performed in accordance with the manufacturer's instructions.

- For items located in the "harsh environment" of the plant, evaluations should be made to determine if the loss of "qualified life" of the part during storage has any effect on the required in-service qualified life of the part. Adjustments in the in-service qualified life should be made if required. This practice will provide the needed conservatism until the industry's research results are available.

All efforts should be made to utilize the technical information available in various industry standards and the need for cooperation to develop the qualification technology should be realized by the nuclear power industry.

References

1. MIL-STD-1523, "Age Control of Age Sensitive Elastomeric Materials", September 1973.

2. MIL-STD-1131B, "Storage Shelf Life and Reforming Procedures for Aluminum Electrolytic Fixed Capacitors", June 1979.

3. The Department of Defense Index of Specifications and Standards (DODIS).

IMPACT OF PRE-CONDITIONING ON THE
QUALIFICATION OF SAFETY-RELATED EQUIPMENT

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August, 1982

Abstract

This paper shares some recent experiences on the effects of pre-conditioning on the qualification of safety-related equipment not located in a harsh environment. Environmental and seismic qualification testing programs were conducted following the guidelines of IEEE 323-1974, IEEE 344-1975 and appropriate IEEE "daughter" standards, where available. The examples that follow will illustrate the degree of pre-conditioning of safety-related equipment qualified to the requirements of IEEE-323-1974, and its effect on the outcome of the qualification program.

Definitions

Safety-Related Electrical Equipment - Electrical equipment required to achieve and maintain emergency reactor shutdown, containment isolation, reactor core cooling, containment and reactor heat removal, and prevention of significant release of radioactive material to the environment following a design basis event.

"Mild" Environments - An environment that would at no time be more severe than the environment that would occur during normal power plant operation or during anticipated operational occurrences. (Re: Proposed Revision 1 To Regulatory Guide 1.89, Feb. 1982)

"Harsh" Environments - Environments that may change significantly from the normal expected environment or anticipated operational occurrences in a sudden or prolonged manner due to the direct effects of a design basis event (i.e. Loss of Coolant Accident (LOCA) or High Energy Line Break (HELB) Accident).

Introduction

For the past five years, the author has reviewed various vendors and laboratories environmental and seismic qualification plans/ reports for various types of Class 1E equipment for nuclear power plants. Some of the Class 1E equipment types reviewed are listed below. This equipment was located both in "harsh" and "mild" environments.

1. Medium and Low Voltage Switchgear/Transformers

2. Low Voltage Motor Control Centers

3. Batteries and Battery Chargers

4. Static Uninterruptible Power Supply System Inverters & Bypass Transformers.

Introduction

5. Low Voltage AC & DC Switchboards and Distribution Panel-boards.
6. Low Voltage Terminal Blocks
7. Medium and Low Voltage Motors (for various pumps, fans, etc.)
8. Low Voltage Lighting & Miscellaneous Power Transformers
9. Electric Penetration Assemblies
10. Medium and Low Voltage Power Cable; Control & Instrumentation Cable.
11. Protective Relay Boards & Racks
12. Conduit/Cable Environmental Seal Assemblies
13. Main Steam Isolation Valve Actuators (Stored Energy Type)
14. Main Steam Bypass Valve Actuators (Stored Energy Type)
15. Motorized Valve Actuators
16. Limit Switches (for dampers & valves)
17. Diesel-Generators, Associated Panels & Auxiliaries
18. Feedwater Valve Actuators (Pneumatic-Hydraulic Type)
19. Electro-Dynamic I/P Transducer
20. Chiller Electrical Control Components
21. Atmospheric Cleanup Trains Electrical Components
22. Solenoid Pilot Valve Operators
23. Control Room A/C Units Electrical Components
24. Containment Hydrogen Analyzer Equipment
25. Remote Multiplexer Equipment
26. Process Solenoid Valves
27. Main Control Board Electrical & Instrumentation Equipment
28. Electronic Transmitters

Introduction - Cont'd.

29. Analog Control System

30. Indicating Differential Pressure Switches

31. Level Switches

32. Thermocouple & RTD Assemblies

33. Containment Water Level Instrumentation System

Discussion

The tabulation in Appendix A lists the pre-conditioning imposed upon some of the above equipment prior to seismic qualification testing. This listing only includes equipment located in a "mild" environment.

As can be seen from this listing, the comprehensive pre-conditioning programs imposed on these Class 1E equipment types had little or no effect on the equipment's ability to perform its safety-related function during or after the postulated seismic events. It should be noted, the sequential application of accelerated thermal, radiation, and mechanical pre-conditioning produces more severe stresses on the equipment component/materials than would be experienced during the normal aging process of operation in a mild environment of a nuclear power plant. Many of the components in these Class 1E equipment types were thermally overaged due to the use of "state-of-the-art" Arrhenius methodology in determining the "weak-link" component, or that which had the lowest activation energy level. All components with materials having higher activation energy levels were thermally aged well beyond the "weak-link" component material's qualified life determination.

Discussion - Cont'd.

One of the reasons why safety-related electrical equipment in a mild environment is virtually unaffected by pre-conditioning is the strict design, manufacturing and testing standards the equipment is constructed in accordance with, irrespective of its use in nuclear or non-nuclear power plant applications. These stringent standards were developed by ANSI, IEEE, NEMA, ICEA, ISA and UL. These standards impose upon the electrical equipment manufacturer rigid performance, type and rating tests. Many of these tests are at purposely elevated levels relative to normal usage to search out material and design defects. Satisfactory passage of such tests tends to demonstrate the margins inherent in electrical equipment design. Many years of test and operating experience have shown that such equipment has a satisfactory design life in normal applications. The "mild" environment definition differs little from these historic normal service conditions. This ability of electrical equipment to attain its design life without meaningful structural, mechanical or electrical degradation provides capability to perform satisfactorily during seismic events. In addition, conservative application of safety-related electrical equipment by the nuclear plant designer (A/E) provides additional margin. Further, accepted industry practices such as application of equipment below their rated values, provides even further margin.

Perhaps the only parameter not addressed in the above cited electrial standards for safety-related equipment testing is that of radiation. This is the only parameter in a nuclear power plant "mild" environment that must also be addressed in addition to those parameters for a non-nuclear plant. However, it should be noted an Electric Power Research Institute report, "Radiation Effects on Organic Materials in Nuclear Plants", (EPRI-NP-2129, Nov. 1981) presents the results of a literature search conducted by Georgia Tech for data on the radiation resistance of organic materials.

An important finding of the report is that a total dose of less than 10^5 rads produces no significant degradation of mechanical or electrical properties. (Notable exceptions are equipment that contain Teflon or semiconductor devices). Information presented in this report concerning organic materials used in nuclear plant equipment suggests than an exclusion from testing or further analysis should be allowed for non-electronic equipment subject to 10^4 rads or less. Non-electronic equipment that does not contain

Teflon and is subjected to less than 10^5 rads should likewise be excluded.

The results of this latest report (EPRI-NP-2129), as described above, agree with Section 7.1.9 of EPRI-NP-1558, "A Review of Equipment Aging Theory and Technology", issued in Sept. of 1980. Also, from the examples of pre-conditioning listed in Appendix A, total doses of up to 10^6 rads had no detrimental effects on safety-related equipment located in a "mild" environment.

Conclusions

Generally, it was found that pre-conditioning prior to seismic testing had no discernible effect on the equipment's ability to perform its safety-related function during and after both the plant Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE), when this equipment was located in a "mild" environment of a Nuclear Power Plant. Where failures during or after seismic testing were discovered, they were in most cases, due to the structural inadequacy of the supporting structure, which required additional bracing, etc., in order to meet the plant Required Response Spectra (RRS). The component failures were not the result of or related to pre-conditioning.

Other component failures, attributed to test procedures, were found which were usually due to elevated temperatures used during accelerated thermal aging, in order to reduce the amount of pre-conditioning time. When these component types were subjected to lower accelerated aging temperatures, they successfully passed the thermal aging phase of the pre-conditioning program, and all subsequent required testing, including seismic.

Based on the experience of qualifying all the balance-of-plant (BOP) safety-related electrical equipment for a complete nuclear power plant, it is the author's opinion that preconditioning of such equipment for a "mild" environment is not a cost effective approach, and should not be a general requirement.

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1.	EPRI NP-1558 Sept. 1980	"A Review of Equipment Aging Theory and Technology"
2.	EPRI NP-2129 Nov. 1981	"Radiation Effects on Organic Materials in Nuclear Plants"
3.	AIF Position Paper July 2, 1981	"Environmental Qualification of Safety Related Electrical Equipment Subjected Only To Mild Environments"
4.	AIF Position Paper	"Concerns For Seismic Qualification of Safety Related Electrical Equipment Subjected Only To Mild Environments" (Supplement to Reference 3 above, approximate date August 12, 1981)
5.	Proposed Revision 1 To Regulatory Guide 1.89 Feb. 1982	"Environmental Qualification of Electric Equipment For Nuclear Power Plants"
6.	IEEE Standard 323-1974	"IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"

Acknowledgement

The author wishes to thank the following Gibbs & Hill, Inc. fellow employees for their contributions, comments and advice to this paper:

W.C. Dumper - Chief Electrical Engineer

T.R. Vardaro - Assistant Chief Electrical Engineer

M.P. McBride - Instrument & Controls Engineer

"APPENDIX A"

MILD ENVIRONMENT EQUIPMENT PRE-CONDITIONING

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
1.	7.2-kv Indoor Metal Clad Switchgear Assemblies	40 Days @ 145°C	1 x 10^5 RADS (gamma)	10,000 Operations	Test results show pre-conditioning had no detrimental effects on breaker performance.
2.	480 Volt Switchgear	4,000 Operations @ max. rated continuous current.	3 x 10^5 RADS (gamma)	8,000 Operations @ No Load	Test results show no significant degradation of the breakers occurred due to the preconditioning. Switchgear structural mods. required to meet plant RRS.
3.	480 Volt AC Motor Control Centers	81 Days @ 115°C	1 x 10^6 RADS (gamma)	1) Circuit Breakers: 6000 cycles @ rated Current; and 4000 cycles @ no load. 2) Contactors/ Starters: 19,000 Operations making 6X rated current & breaking rated current; and 30,000 Operations @ no load. 3) Relays, Selector Switch & Push-buttons: 100,000 Oper. @ no load & 20,000 Oper. @ rated load (make + break)	During early seismic testing, seismic braces were found to be inadequate and were redesigned. Further testing indicated the new braces were satisfactory. Test results show preconditioning had no detrimental effects on breaker and starter performance.

		PRE-CONDITIONING PRIOR TO SEISMIC TESTING			
ITEM NUMBER	SAFETY-RELATED EQUIPMENT	THERMAL AGING	RADIATION AGING	MECH. CYCLING	REMARKS
4.	Battery Chargers and Accessories	1) Magnetics: 230°C for 750 hours. 2) DC Electrolytic Capacitors: 95°C. @ full rated DC voltage for 500 hrs. (10 yr. life). 3) Circuit & Alarm Boards Control Transformers: 160°C for 500 hrs.	1.4×10^{3} RADS	1) Circuit breakers: 260 Operations @ full rated load. 2) Float-Equalize Switch: 660 Operations @ full load.	Since the test results show the specimens demonstrated sufficient integrity to withstand, without compromise of structures or electrical functions, the plant prescribed seismic environment, preconditioning had no adverse effect.
5.	Static Uninterruptible Power Supply System, Inverter and By-Pass Transformer. (SUPS)	1)Circuit breakers: 150°C for 150 hrs. 2)Term. & Fuse Blocks: 140°C for 194 hrs. 3)Magnetics: 207°C for 400 hrs. (38 yrs. life) 4)Fuses: 150°C for 159 hrs. 5)Connectors: 150°C for 221 hrs. (29 yr. life) 6)Relays: 150°C for 168 hrs.	1×10^{4} RADS	1)Circuit Breakers: 500 cycles under load. 2) N/A 3) N/A 4) N/A 5) N/A 6) Relays: 500 cycles under load.	Visual inspection of the test specimen upon completion of each portion of the seismic test revealed no structural damage to the specimen or change in performance due to the seismic test. Therefore, preconditioning had no detrimental effect.

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
6.	118, 120 & 208/120V.AC Panelboards and 125V D.C. Switchboards & Panelboards	1) Switches: 34 days @ 130°C, 2) Molded Case Circuit Breakers 42 days @ 130°C,	1×10^5 RADS	1) Switches: 6000 Operations @ rated load & 4000 Operations @ no load. 2) Molded Case Circuit Breakers: 4,000 Operations @ rated load & 4,000 Operations @ no load.	There were no anomolies attributable to the switches or to the breakers during the entire seismic testing segment. This proves that pre-conditioning had no adverse effect on these components.
7.	Terminal Blocks	168 hrs. @ 130°C	220×10^6 RADS	1,000 Open/Close Cycles	There was no evidence that the environmental exposures of this test program had any effect on the functional performance of the specimens.
8.	Valve Actuators (Class B Insulated Motors - for Outside Containment Use)	165°F. for 200 hrs. (@ 100% R.H.)	2.04×10^8 RADS (gamma)	176 cycles during thermal aging & 1817 cycles @ room ambient.	Since no detrimental effects due to aging were discernable, it is concluded that preconditioning had no adverse effects prior to seismic testing.

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
9.	Limit Switches	120°C for 400 hrs.	2.04×10^{8} RADS (gamma)	100,540 on-off actuations with load applied to the contacts.	Test results show that closed contacts did not open for more than 2 milliseconds during seismic testing. This proves that pre-conditioning had no detrimental effect on the limit switch components.
10.	Electro-Hydraulic Valve Actuator on M.S. By-Pass Valve.	11 Days @ 284°F	1.92×10^{8} RADS	2622 Cycles	Visual inspection of the test specimen revealed no structural damage had occurred during the OBE & SSE tests. No change in the performance of the test specimen was noted upon completion of each test, proving that pre-conditioning had no detrimental effect on the specimen components.
11.	Stored Energy Valve Actuator on M.S. Valve	200°F for 200 hrs. (See "Mech. Cycling" Col. for additional thermal aging during cycling).	5.4×10^{6} RADS (gamma)	1275 cycles @ 150°F ambient (485 hrs. approx.)	The test specimen operated before, during and after the seismic tests, proving that pre-conditioning had no adverse effects.

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
12.	Local Control Station	413 hrs. @ 130°C	1 x 10^6 RADS	2816 Cycles under load	The seismic test results demonstrated that the specimen possessed sufficient integrity to withstand, without compromise of structure or electrical function, the prescribed seismic environment. Therefore, preconditioning had no detrimental effects.
13.	Atmospheric Filter Units & Control Room Air Cond. Units Electrical Components	1) Contactors: 100°C. for 2821 hrs. 2) Aux. Contactors 100°C. for 3786 hrs. 3) Terminal Blocks: 100°C for 5011 hrs. 4) Temp. Switches & Solenoid Valve: 100°C for 3936 hrs. 5) Transformer: 170°C for 3141 hrs. (20 yr. life) 6) Disconnect Switch 100°C for 4332 hrs. 7) Circuit Breaker: 100°C for 3409 hrs. (15 yr. life) 8) PB & Sel. Switch 100°C for 4599 hrs.	4 x 10^4 RADS for atmos. filters and 1 x 10^3 RADS for Control Room A/C Units.	1) Contactors, PB, Relays & Switches: 32,120 Actuations. 2) Solenoid Valve (filter units): 40 cycles. 3) Solenoid Valve (A/C Units): 320,000 cycles.	The filter unit components did not experience any failures throughout the test program, therefore, preconditioning had no detrimental effects on the components. The air cond. units did experience (2) component failures, however, it was demonstrated that these failures are not significant to the operation of the air cond. unit. A retest is planned for these (2) failed components to verify that these failures are random in nature and do not represent common mode failures. In the event that a common mode failure is demonstrated, either a reduced qualified life will be established or a design change implemented.

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
13. (cont'd)	Control Room Air Cond. Units Electrical Components (Cont'd	9)Control Relay and Pressure Switch: 100°C for 2114 hrs. 10)Timer: 100°C for 1400 hrs. 11)Thermostat, Temp. Controller, Power Supply, Press. Controller & Transmitter, & Flow Control Valve: 100°C for 1044 hrs.	1×10^3 RADS	(See previous sheet)	(See previous sheet)
14.	HVAC Dampers & Valves - Solenoid Valves	268°F for 12 Days	50×10^6 RADS (gamma)	Electrically cycled 40,000 times @ max. oper. pressure differential.	All the valves successfully completed the seismic simulation tests, proving that preconditioning had no detrimental effects on valve materials or performance.

ITEM NUMBER	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
15.	HVAC Dampers & Valves - Limit Switches (Outside Containment Use)	120°C for 421 hrs.	2.04×10^8 RADS (gamma)	104,700 Actuations @ 125V DC, 5A	Same as Item 9 above)

ITEM	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
16	I/P Transducer	220°F for 25 Hrs.	1.05×10^{7} Rads Gamma	27,000 cycles	Small zero shift in valve travel due to testing above design temp. of 150°F.
17	Process Solenoid Valves	318°F for 172 Hrs.	2×10^{8} Rads. Gamma	7500 cycles	Test results within accept. limits.
18	Pressure Transmitters	203°F for 47 Days.	2.21×10^{7} Rads Gamma	8062 cycles	Test results within accept. limits.
19	Timing Relay	158°F for 130 Days	4.4×10^{4} Rads	9350 cycles	Test results show no significant degradation due to pre-conditioning.
CONTROL BOARDS & COMPONENTS	Fuse	212°F for 122 Days	4.4×10^{4} Rads	N/A	
	Wattmeter	158°F for 130 Days	4.4×10^{4} Rads	6600 cycles	High aging temp. caused face plate to warp.
	Voltmeter	212°F for 122 Days	4.4×10^{4} Rads	18,700 cycles	Test results show no significant degradation due to pre-conditioning.
	Transformer	158°F for 130 Days	4.4×10^{4} Rads	N/A	Test results show no significant degradation due to preconditioning.
	Transducer	158°F for 130 Days	4.4×10^{4} Rads	N/A	Set point drift during thermal aging. Recommend annual calibration.
	Terminal Block	212°F for 122 Days	4.4×10^{4} Rads	N/A	Test results show no significant degradation due to pre-conditioning.

ITEM	SAFETY-RELATED EQUIPMENT	PRE-CONDITIONING PRIOR TO SEISMIC TESTING			REMARKS
		THERMAL AGING	RADIATION AGING	MECH. CYCLING	
19 -CONT'D					
CONTROL BOARDS & COMPONENTS	Fuse Block	212°F for 122 Days	$4.4x10^4$ Rads	N/A	Test results show no significant degradation due to pre-conditioning.
	Counter	158°F for 72 Days	$4.4x10^4$ Rads	9350 cycles	
	Minalite	212°F for 122 Days	$4.4x10^4$ Rads	N/A	
	Relay	158°F for 130 Days	$4.4x10^4$ Rads	4675 cycles	
	Test Switch	212°F for 122 Days	$4.4x10^4$ Rads	N/A	
	Meter (0-10VDC Input)	158°F for 104 Days	$4.4x10^4$ Rads	9350 cycles	
	Meter (0-1 MADC Input)	158°F for 104 Days	$4.4x10^4$ Rads	4675 cycles	
	Meter (4-20 MADC Input)	158°F for 104 Days	$4.4x10^4$ Rads	4675 cycles	
	Receptacles	212°F for 122 Days	$4.4x10^4$ Rads	N/A	
	Selector Switch Module	212°F for 122 Days	$4.4x10^4$ Rads	19,000 cycles	2 switches failed wear aging test. Qualified life was set at 5 years.
	Pushbutton Module	212°F for 122 Days	$4.4x10^4$ Rads	18,900 cycles	Test results show no no significant degradation due to pre-conditioning.
	Microswitch Module	212°F for 122 Days	$4.4x10^4$ Rads	N/A Exempted based on test data.	
	Control Switch Module (6 Stages)	212°F for 122 Days	$4.4x10^4$ Rads	N/A Exempted based on test data.	
	Control Switch Module (2 stages)	212°F for 122 Days	$4.4x10^4$ Rads	18,700 cycles	Switch handle broke during seismic test. Replacement interval determined. Otherwise no significant degradation due to pre-conditioning.

		PRE-CONDITIONING PRIOR TO SEISMIC TESTING			
ITEM	SAFETY-RELATED EQUIPMENT	THERMAL AGING	RADIATION AGING	MECH. CYCLING	REMARKS
19 - CONT'D					
CONTROL BOARDS & COMPONENTS	Annunciator Light Box	N/A	N/A	N/A	Component must maintain structural integrity only, non-safety related.
	Meter (0-5A Input)	158°F for 104 Days	$4.4x10^4$ Rads	468 cycles	Due to low activation energy & time constraints these instr. were qualified for 1 year only. Due to be replaced by longer qualified life meters.
	Meter (0-150VAC Input)	158°F for 104 Days	$4.4x10^4$ Rads	468 cycles	
	Monitor Light Box	212°F for 122 Days	$4.4x10^4$ Rads	Exempted based on test data.	Test results show no significant degradation due to pre-conditioning.
	Push-to-Test Light	158°F for 130 Days	$4.4x10^4$ Rads	6600 cycles	
	Meter (±.5 MADC Input)	158°F for 104 Days	$4.4x10^4$ Rads	N/A	Same comment as (0-5A Input) Meter above.
	Transformer	158°F for 130 Days	$4.4x10^4$ Rads	N/A	Test results show no significant degradation due to pre-conditioning.
	Frequency Transducer	158°F for 130 Days	$4.4x10^4$ Rads	N/A	

EQUIPMENT QUALIFICATION MAINTENANCE

Roy M. Scates, Wyle Laboratories

INTRODUCTION

Over the past few years, the nuclear power industry has devoted a great deal of attention to achieving equipment qualification. As this goal approaches, the attention of prudent utilities is now being directed toward maintaining that qualification throughout the useful life of the equipment. This paper presents a Qualification Maintenance Program (QMP) which is designed to meet the Nuclear Regulatory Commission requirements for ongoing qualification maintenance of safety-related equipment.

This paper will describe this program by answering the following four questions:

1) What is QMP?

2) Why is a QMP necessary?

3) What are the benefits of a QMP?

4) How can QMP best be achieved?

Following the discussion of the above questions, an example is presented which illustrates a typical application of QMP in defining plant equipment maintenance requirements.

WHAT IS QMP?

A Qualification Maintenance Program is a methodical process which provides an auditable link between the equipment qualification central file and those plant maintenance and inventory control activities which affect qualification status.

There are a number of activities that could affect qualification status. They include:

- o Start-up and testing functions associated with safety-related equipment
- o Corrective and preventive maintenance actions performed on safety-related equipment
- o Procurement of qualified replacement or spare parts
- o Inventory control of qualified replacement parts and spare equipment
- o Storage and retrieval of Equipment Qualification Central File documentation
- o Surveillance activities which monitor equipment performance/status

The QMP, when fully implemented, will provide the utility with:

- o An auditable method which provides information and instructions necessary to maintain equipment qualification status throughout the plant's life
- o A specialized program which is integrated into existing:

- equipment qualification document storage and retrieval systems
- inventory control systems
- procurement control systems
- quality assurance systems
- maintenance and preventive maintenance scheduling systems
- start-up and testing programs
- repair/refurbishment programs
- maintenance and surveillance history of each safety-related equipment item from which trend data can be established.

WHY IS QMP NECESSARY?

The requirement for establishing a Qualification Maintenance Program is expressed in NRC IE Bulletin 79-01B and in current Safety Evaluation Reports. The need for such a program was also addressed at the joint NRC/Utility Workshop held in Bethesda, Maryland, July 7-10, 1981. Utilities are required to maintain auditable documentation files in a central location on the qualification status of safety-related equipment in licensed nuclear power plants and to ensure that equipment qualification integrity is maintained throughout the plant's life. An effective Qualification Maintenance Program provides the vehicle for meeting those objectives in a cost-efficient manner and must take into consideration the following:

- o Equipment qualified for time periods less than the lifetime of the plant must be replaced or refurbished periodically.
- o Spare parts for safety-related equipment must comply with NRC guidelines on equipment qualification.
- o High cost of plant outages necessitates coordination of equipment maintenance with scheduled down times.
- o Immediate availability of qualified spare parts for both planned and unplanned outages ensures maximum plant availability.
- o State-of-the-art qualification practices necessitate a surveillance system to detect and correct unanticipated deterioration.

WHAT ARE THE BENEFITS DERIVED FROM A QMP?

Once QMP has been fully implemented into the existing management network, several short and long-term benefits will be realized. Besides satisfying licensing requirements and providing the means to ensure that the qualification status of all safety-related equipment is continually maintained, QMP will:

- o Help coordinate maintenance and procurement activities for outages, thus minimizing outage time. This is particularly important for unexpected outages which do not have the benefit of preplanning.

- Improve overall availability of spare parts, including identification of potential unavailability of spare/replacement parts and lead times required for identifying and procuring qualified substitutes. A realistic output would be reduction of spare parts inventories.

- Improve equipment surveillance procedures and record keeping, thus potentially increasing plant safety, reliability, and availability.

- Identify an overall cost-effective method for procuring safety-related equipment.

- Help detect unexpected deterioration trends.

All these benefits tend to improve overall system reliability and increase plant availability.

HOW CAN QMP BEST BE ACHIEVED?

Every utility has at least some elements of a Qualification Maintenance Program in place. A practical approach is to build upon existing utility management systems which deal with replacement procedure, inventory control, maintenance, quality assurance, etc. This approach should also make use of existing equipment qualification files and integrate all these systems into an effective Qualification Maintenance Program. The nature and extent of adapting a Qualification Maintenance Program to present systems naturally depends on the breadth and complexity of existing procedures.

The practical approach to QMP should take full advantage of existing procedures. These procedures must be integrated and, if necessary, supplemented to facilitate incorporation into the QMP system. The system should consider the existence of a management system for plant operation and maintenance, as well as equipment qualification data files. Figure 1 illustrates that in an operating plant, present baseline systems may be maintained independently without the required interaction for an effective and operational Qualification Maintenance Program.

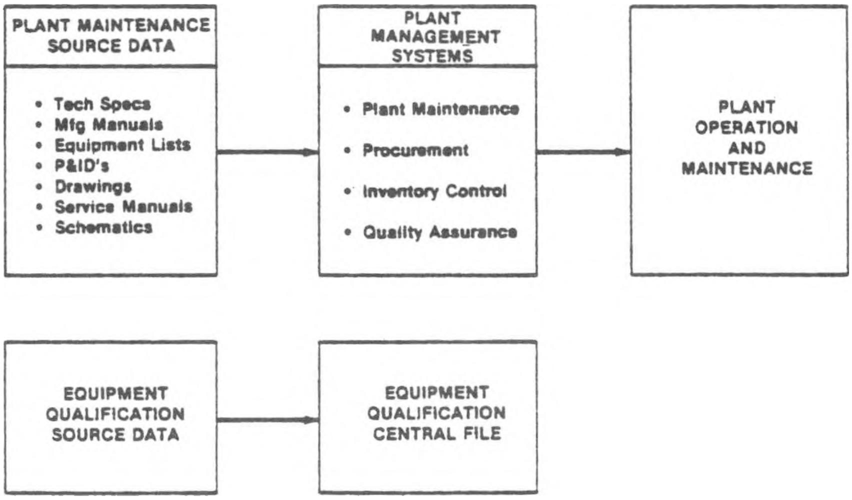

Figure 1. Utilities Baseline Management Systems

The building block approach inherent in the QMP concept is described in the following illustrated discussions and may be implemented in three phases.

Phase I

The first step in developing a QMP (Phase I) begins with a review of the Equipment Qualification Central File (EQCF) and existing procedures and systems that includes, but is not limited to, plant management, maintenance, quality assurance, procurement, and inventory. The primary purpose of the EQCF review is to identify all qualification conditions which must be maintained throughout the equipment's installed life. This review can be systemized by the development of a series of data sheets and instructions for the reviewing engineers. A typical Equipment Qualification Summary Sheet, illustrated later on, can be part of the first activity of the EQCF review.

Some examples of possible qualification conditions which must be considered are:

- <u>Mounting Position</u> - Vertical position required. Repositioning in a horizontal position would nullify the qualification.

- <u>Interfaces</u> - A motor was qualified with a specific class of leads. Reconnection with another class of leads nullifies the qualification. (Note: This example was the subject of IE Circular No. 80-10, dated April 29, 1980.)

- <u>Periodic Maintenance</u> - A fan is qualified for 40 years, provided it is lubricated in accordance with the manufacturer's recommendations every six months. Failure to do so would nullify the qualification.

- <u>Parts Replacement</u> - A valve is qualified for 40 years, provided the O-ring is replaced with a qualified O-ring every four years. Failure to do so, or replacement with an O-ring of another material, would nullify the qualification.

- <u>Environments</u> - Most equipment is qualified for use in specific environmental conditions. Use of the equipment in other environments or a change in environment may invalidate the qualification.

Once all conditions which form the basis for qualification have been identified and summarized, the next step in the systems review process is to similarly review, summarize, and define interaction of the existing utility management systems. Other forms similar to the Maintenance Requirement Summary Sheet, illustrated subsequently, can be used to summarize this information. The purpose of this review is to identify procedures and methods for generating the following information:

- Replacement parts list (including alternative parts supplier)

- Replacement and refurbishment interval

- Preventive maintenance schedule and instructions

- Surveillance and inspection schedule and instructions

- Corrective maintenance instructions

- Replacement parts traceability requirements
- Parts on hand, bin locations, shelf life restrictions
- Reordering schedule
- Purchasing instructions
- Receiving instructions
- ASME Section XI inspections that impact QM

The completion of the review, summarization, and interaction definition tasks initiates the next step in the Phase I effort. This task uses results of previous efforts to formally assess the compatibility of existing plant systems and procedures with the equipment qualification files and develops the actions necessary to complete an integration of the entire system.

Figure 2 illustrates this intermediate step and, as shown, this integration and compatibility action is a two-way flow of information and requirements between the management systems and the equipment qualification files to:

- Ensure maintenance of qualification integrity during performance of all normal plant procedures
- Maintain auditable records of plant qualification status
- Provide a method of recognizing unanticipated equipment degradation

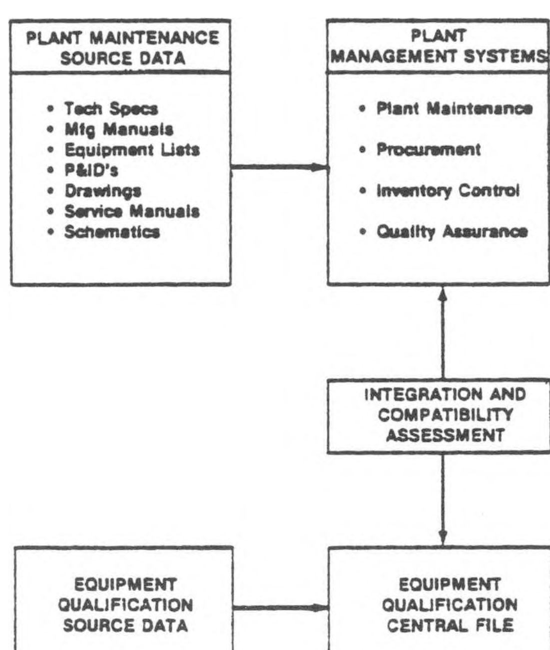

Figure 2. Integration of Plant Management Procedures and Equipment Qualification Requirements

After the compatibility study is complete, a basic decision will be possible and one of the three possible courses of action is selected:

1) Existing management systems are adequate and no further action is required.

2) Existing management systems are adequate with minor changes or additions of data.

3) Existing management systems are not adequate and a new integrated Qualification Maintenance System is recommended.

The appropriate recommended action should be substantiated, as required, by the following:

- The qualification conditions found in the sample equipment items

- Recommendations for updating and modifying the EQCF, if inadequacies are found

- An analysis of all areas where existing management systems do not adequately support the maintenance of the qualification conditions identified in the review of the EQCF

- Recommended actions for filling informational voids in the present system(s) or recommendations for formatting a new integrated system specifically to support the maintenance of equipment qualification

Phase II

The Phase II QMP effort consists of developing compatible systems and procedures based on Phase I activities and results. The options resulting from Phase I are illustrated in Figure 3. Specifically, if the existing management systems are adequate or require only minor changes or additions, the Phase II effort will be a relatively straightforward process. However, if the existing management systems are not adequate to support the QMP, it will be necessary to establish the requirements and, subsequently, develop a new QMP system to include, at a minimum, maintenance procedures, QA procedures, spare parts specifications, and procurement procedures. An additional consideration for the Phase II effort would be any required updating or modification of the EQCF to assure compatibility with the preventive maintenance (PM), inventory control (IC), and QMP systems.

Phase III

Phase III formally implements the QMP into the plant operation cycle and can be characterized as closing the loop between all phases of the QMP with a resultant ongoing working tool that will support and enhance the efforts of the plant maintenance and operation activities. Figure 4 illustrates this "closing of the loop." The elements of the systematic updating of the QMP data bank, as showed by the numbered interactions (1, 2a, 2b, and 3) are described as follows:

1. Plant maintenance source data and equipment qualification source data are continually updated to account for new qualified equipment which may be added to the plant and new qualification data which may become available

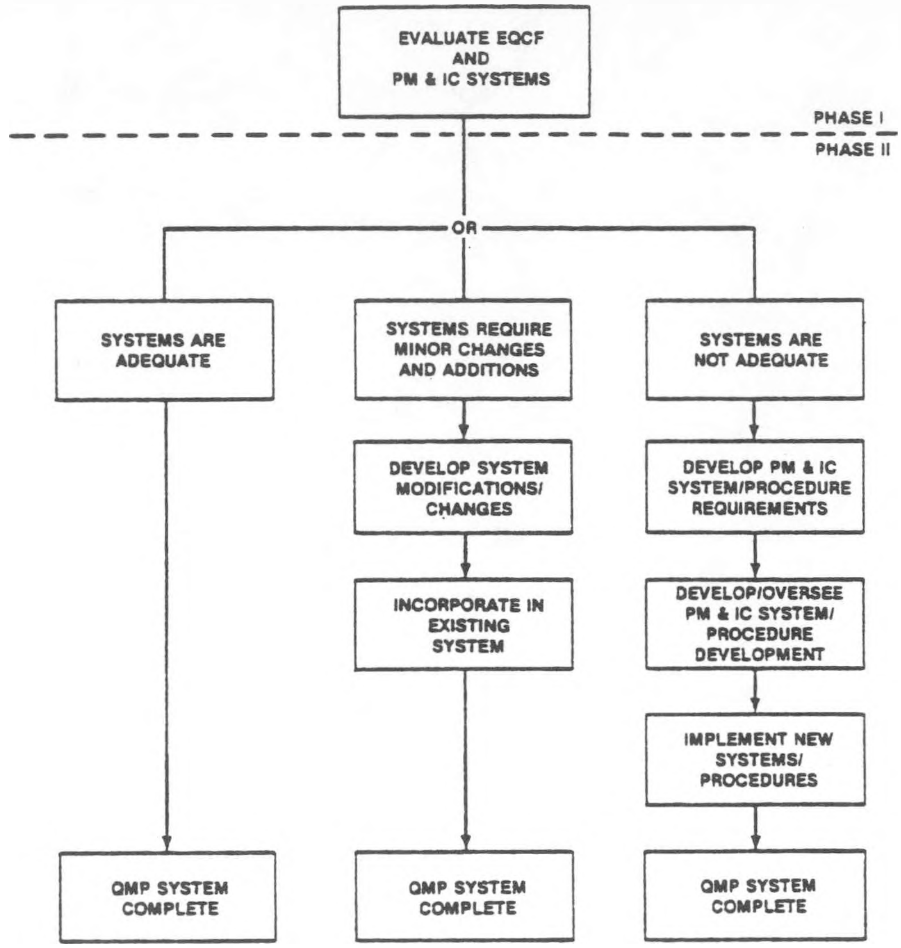

Figure 3. Phase II Options

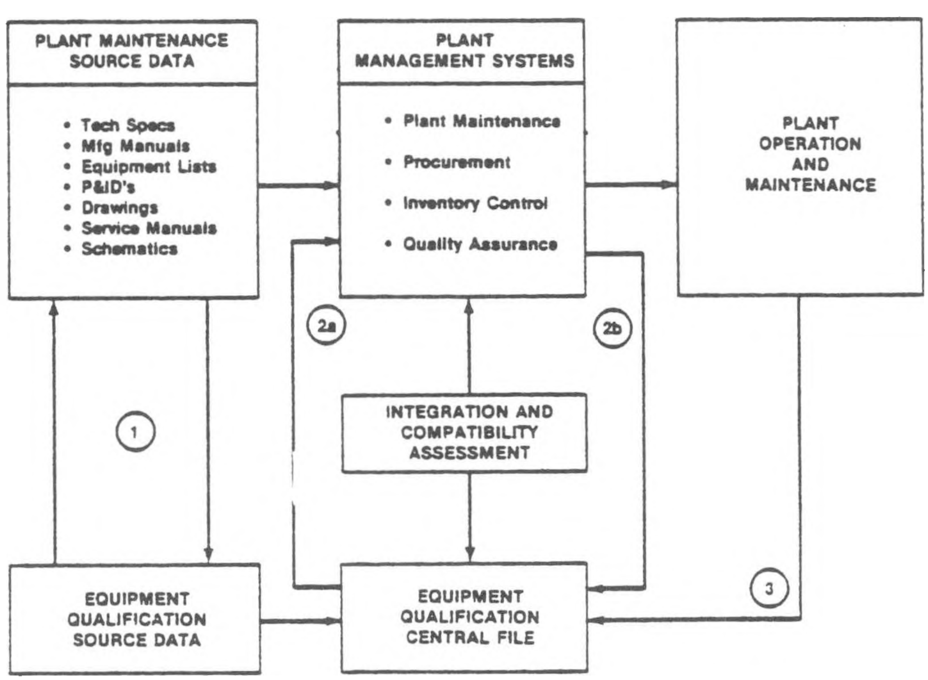

Figure 4. Closing the Loop or Maintaining Equipment Qualification Integrity

(2) The management systems and applicable procedures are continually updated to ensure:

(2a) that the procedures are consistent with the equipment source data and all conditions for maintaining qualification are included

(2b) that the correct procedures are followed in performing plant maintenance and that the Equipment Qualification Central File is systematically updated to reflect changes in the procedures

(3) Maintenance and surveillance records of action performed in the plant are fed back to the Equipment Qualification Central File to indicate equipment status and that the required maintenance and surveillance has been performed. It may also be useful to detect unexpected trends.

QUALIFICATION MAINTENANCE PROGRAM EXAMPLE

An example of how this critical compatibility study portion of Phase I of the QMP might be carried out follows for two identical solenoid valves found in different plant locations.

The EQCF is reviewed for each of these solenoid valves and the pertinent data is summarized on Equipment Qualification Summary Sheets (EQSS), as noted in Figures 5 and 6, for plant ID numbers SV007 and SV008, respectively. Qualification conditions as determined from the EQCF review are listed in Section 4.0 of the EQSS. For Figure 5, it is noted that qualification is dependent on vertical mounting, electrical connections being routed through conduit, periodic seal replacement, and the definition of replacement parts. For Figure 6, it is noted that qualification is dependent on vertical mounting, electrical connections being routed through conduit, and the replacement of parts. It should be noted that no periodic replacements are mandatory because the valve is normally not energized and the qualification documentation demonstrated that the solenoid valve materials are not age sensitive in this application.

Next, the maintenance requirements which the plant personnel are already performing are reviewed for each ID number. The results of these reviews are shown in the Maintenance Requirement Summary Sheets (MRSS), Figures 7 and 8. It should be noted that, for each of these valves, the seals and coils are currently being replaced at four-year intervals. However, neither the seal nor the coil part numbers are listed so the possibility exists for replacement with materials that are not the same as those qualified.

The data on the EQSS is compared with the data on the MRSS and Section 5.0 of the MRSS, "Recommended Modification to Maintenance Procedures," is completed.

For Figure 7, the following actions result:

- o Increase the relacement interval from four to ten years for the seals
- o Delete replacement requirements for the coils
- o Add note requiring vertical mounting and conduit connection
- o Identify replacement part numbers

EQUIPMENT QUALIFICATION SUMMARY SHEET

EQSS No. 001

COMPILED BY: J. Gleason DATE: 11-18-81

VERIFIED BY: R. Scates DATE: 11-19-81

1.0 ITEM DESCRIPTION

1.1 ITEM: Solenoid Valve 1.2 MFG: XYZ Corp.

1.3 MODEL NO(s): AH 123 1.4 PLANT ID NO(s): SV007

1.5 OTHER (Series, Mfg Dates, Option, Etc.)

1.6 QUALIFIED LIFE: 40 yrs 1.7 QUALIFIED FOR PLANT ZONES: H5

2.0 SAFETY RELATED FUNCTION

Open upon loss of 120 VAC to coil against 100 ± 20 PSI Differential Pressure

3.0 QUALIFICATION REFERENCES

Assessment Report No. 5678

4.0 QUALIFICATION CONDITIONS

4.1 MOUNTING & INTERFACE: Vertical Mounting
Connect Via Conduit

4.2 PERIODIC MAINTENANCE: None Specified

4.3 PARTS REPLACEMENT: EPR Seals P/N 123A Every 10 yrs
Nomex Coil P/N H7A >40 yrs

4.4 OTHER

WH 1222

Figure 5. Wyle Equipment Qualification Summary Sheet for a Normally Energized Solenoid Valve

EQUIPMENT QUALIFICATION SUMMARY SHEET

EQSS No. 002

COMPILED BY: J. Gleason DATE: 11-18-81

VERIFIED BY: R. Scates DATE: 11-19-81

1.0 ITEM DESCRIPTION

1.1 ITEM: Solenoid Valve 1.2 MFG: XYZ Corp.

1.3 MODEL NO(s): AH123 1.4 PLANT ID NO(s): SV008

1.5 OTHER (Series, Mfg Dates, Option, Etc.)

1.6 QUALIFIED LIFE: 40 yrs 1.7 QUALIFIED FOR PLANT ZONES: H4

2.0 SAFETY RELATED FUNCTION

Open Upon Application of 120 VAC to coil against 100 ± 20 PSI Differential Pressure

3.0 QUALIFICATION REFERENCES

Assessment Report No. 5678

4.0 QUALIFICATION CONDITIONS

4.1 MOUNTING & INTERFACE: Vertical Mounting
Connect via Conduit

4.2 PERIODIC MAINTENANCE: None Specified

4.3 PARTS REPLACEMENT: EPR Seals P/N 123A Greater than 40 years
Nomex Coil P/N H7A Greater than 40 years

4.4 OTHER

WH 1222

Figure 6. Wyle Equipment Qualification Summary Sheet for a Normally Deenergized Solenoid Valve

MAINTENANCE REQUIREMENT SUMMARY SHEET

MRSS No. 638

COMPILED BY: R. SCATES DATE: 11-19-81

VERIFIED BY: J. GLEASON DATE: 11-19-81

1.0 ITEM DESCRIPTION

1.1 ITEM: SOLENOID VALVE 1.2 MFG: XYZ CORP.

1.3 MODEL NO(s): AH123 1.4 PLANT ID NO(s): SV007

1.5 OTHER (Series, Mfg Dates, Option, Etc.)

1.6 EQSS REF. NO: 001

2.0 MAINTENANCE REFERENCES

XYZ MAINTENANCE MANUAL FOR MODEL AH123 SOLENOID VALVES

3.0 MAINTENANCE REQUIREMENTS

REPLACE SEALS & COILS EVERY 4 YEARS. VISUALLY INSPECT AND CHECK PROPER OPERATION EVERY 18 MONTHS.

4.0 REPLACEMENT PARTS

NO SEAL PART NUMBER SPECIFIED.
COIL RATED AT 180°C - NO OTHER INFORMATION SPECIFIED.

5.0 RECOMMENDED MODIFICATIONS TO MAINTENANCE PROCEDURES

- SEALS QUALIFIED FOR 10 YEARS; CHANGE REPLACEMENT INTERVALS FROM 4 YEARS TO 10 YEARS - SPECIFY EPR SEALS & SHELF LIFE.
- COIL IS QUALIFIED FOR >40 YEARS; DELETE REQUIREMENT FOR COIL REPLACEMENT - SPECIFY NOMEX COIL P/N IS H7A.
- ADD NOTE REQUIRING VERTICAL MOUNTING & CONNECTION VIA CONDUIT.
- DOCUMENT THE RESULTS OF VISUAL INSPECTIONS & OPERATIONAL TESTS IN SURVEILLANCE FILE.

6.0 OTHER RECOMMENDED ACTIONS

VERIFY SPARE PARTS SPECIFICATION. IF NONE EXISTS, PREPARE SPECIFICATION.

WH 1323

Figure 7. Wyle Maintenance Requirement Summary Sheet for a Normally Energized Solenoid Valve

MAINTENANCE REQUIREMENT SUMMARY SHEET

MRSS No. 633

COMPILED BY: R. SCATES DATE: 11-19-81

VERIFIED BY: J. GLEASON DATE: 11-19-81

1.0 ITEM DESCRIPTION

1.1 ITEM: SOLENOID VALVE 1.2 MFG: XYZ CORP.

1.3 MODEL NO(s): AH123 1.4 PLANT ID NO(s): SV008

1.5 OTHER (Series, Mfg Dates, Option, Etc.)

1.6 EQSS REF. NO: 002

2.0 MAINTENANCE REFERENCES

XYZ MAINTENANCE MANUAL FOR MODEL AH123 SOLENOID VALVES

3.0 MAINTENANCE REQUIREMENTS

REPLACE SEALS & COIL EVERY 4 YEARS. VISUALLY INSPECT AND CHECK PROPER OPERATION EVERY 18 MONTHS

4.0 REPLACEMENT PARTS

NO SEAL PART NUMBER SPECIFIED.
COIL RATED AT 180°C - NO OTHER INFORMATION SPECIFIED.

5.0 RECOMMENDED MODIFICATIONS TO MAINTENANCE PROCEDURES

- SEALS & COIL ARE QUALIFIED FOR GREATER THAN 40 YEARS. THEREFORE, DELETE REQUIREMENT FOR SEALS & COIL REPLACEMENT.
- THE EPR SEALS PART NUMBER IS 123A AND THE NOMEX COIL IS P/N H7A.
- DOCUMENT THE RESULTS OF VISUAL INSPECTIONS & OPERATIONAL TESTS IN SURVEILLANCE FILE.
- ADD NOTE REQUIRING VERTICAL MOUNTING & CONNECTION VIA CONDUIT.

6.0 OTHER RECOMMENDED ACTIONS

VERIFY SPARE PARTS SPECIFICATION. IF NONE EXISTS, PREPARE SPECIFICATION

WH 1323

Figure 8. Wyle Maintenance Requirement Summary Sheet for a Normally Deenergized Solenoid Valve

- Document results of visual inspections and operational tests in service

- Verify spare parts specification

For Figure 8, the following actions result:

- Delete replacement requirement for the seals and the coils

- Add note requiring vertical mounting and conduit connection

- Identify replacement part numbers

- Document results of visual inspections and operational tests in surveillance files

- Verify spare parts specifications

Of major significance is the fact that the qualification documentation required no periodic parts replacement for ID No. SV008 so that the current four-year replacement cycle can be eliminated. For ID No. SV007, the periodic replacement cycle is limited to only the seals at ten-year intervals instead of the current four-year replacement cycle.

Some utilities may continue to perform four-year relacements at their discretion when outage schedules allow, but in order to maintain qualification, the above recommendations are all that is necessary.

Although this example is a real situation existing in virtually all plants, for other items and/or more severe environments, the qualification documentation may require shorter maintenance and replacement intervals than is current maintenance practice. A prime objective of QMP is to assure that the preventive maintenance actions are in concert with the qualification documentation. Therefore, identification of replacement parts is also necessary.

Because state-of-the-art procedures were followed in the qualification, uncertainties exist with respect to the definition of the service environmental conditions and events can occur which will cause the service environmental conditions to vary with time. Therefore, a surveillance system needs to be established. For the SV007 and SV008 in the demonstration, parts replacement and periodic functional tests are currently being performed. The surveillance system needs to document the results of these periodic replacements and functional tests.

During the periodic replacement, the visual condition of replacement items should be noted and records kept describing the items replaced. Thus, the various action items resulting from completion of the MRSS would be implemented. The implementation would necessitate coordinating existing systems associated with quality assurance, procurement, inventory control, and surveillance system. As trends are noted, the maintenance procedures would be adjusted to account for improved information.

The key to the development of a Qualification Maintenance Program is familiarity with equipment qualification. The review of the Equipment Qualification Central File to determine the qualification conditions that must be maintained throughout the life of the equipment is the single most important function of the entire QMP development process. If errors are made here, the program has failed. The organization chartered with this review must, therefore, possess the widest range of qualifications.

METHODS FOR PREDICTING STRAIN IN DENTED STEAM GENERATOR TUBES FROM PROFILOMETRY DATA

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Failure Analysis Associates, 2225 E. Bayshore Road,
Palo Alto, California 94303, (415)856-9400

ABSTRACT

Steam generator tubes become dented due to the growth of oxide between the tube and its support structure. When the denting becomes severe, the tube must be plugged to prevent the possibility of primary water leakage. A number of factors, including tube strain, determine the optimal time for plugging. Tube strains cannot be measured directly, and thus, tube strain must be predicted from profilometry measurements. The prediction of tube strain requires two steps: definition of tube shape from radial displacements measured by a probe, and determination of strain from the defined tube shape. Procedures for both of these steps are presented and compared. Recommendations for the use of these methods and future probe developments are given.

INTRODUCTION

Steam generator tube defects are responsible for significant losses in nuclear power plants. The accepted method of guarding against leakage or loss of integrity because of these defects is to remove tubes from service by plugging. Plugging criteria depend on the type, growth rate, and limits of the defect.

Assuming strain to be the dominant variable controlling tube cracking, a reasonable plugging standard requires the determination of (1) actual strains, and (2) strain limits. Results of extensive research on actual strain measurements using profilometry data are presented. This effort provides an understanding of the relative accuracies of possible methods for strain calculation and the identification of improvements necessary in the more successful methods.

The area of tube strain limits is in great need of refinement. Today the primary method for establishing limits within the field is failure of a particular diameter probe to pass through a tube, which is presumed to correlate with a specific strain. With profilometry becoming an accepted practice for estimating strains, the need for also refining strain limits and correctly determining strains from discrete displacement values measured by the probe, is clear; otherwise, most of the advantages of profilometry are lost.

Several techniques used to approximate the shape of the deformed tube and compute strains from discretely measured inner radii of the tube will be discussed along with their relative merits and limitations. Nonlinear kinematic strain-displacement relations are shown to be necessary for accurate strain determination.

CURRENT BASIS FOR TUBE PLUGGING

Several techniques are currently used to determine whether or not steam generator tubes should be plugged. Eddy current probes are used to determine wall thinning due to fretting or wastage. Depending upon the specific location and plant, indications from 10% to 40% of the wall thickness can justify tube plugging. This decision is typically based on the rate of thinning before the next scheduled outage and the necessity of meeting the required Nuclear Regulatory Guidelines. Eddy current techniques are also used to determine stress corrosion cracking in or near U-bends, tube-sheet, or dented regions. Again, the general guideline has been to plug any tube with a 40% through-wall defect. In these cases, a more quantitative understanding of mechanisms involved in the defect formation and growth and/or improved non-destructive testing procedures would permit improved decisions regarding generator tube plugging.

Ball gauging is a second method used to measure tube defect sizes. The passage of various ball sizes is used to determine the minimum diameter (maximum dent) at the tube support levels. The decision to plug is based on previous strain calculations for a given dent size combined with observed field cracks and leaks. However, the exact dent configuration is unknown, therefore, only a weak strain correlation is obtainable. Figure 1 shows an example of two dented tubes where ball gauging is incorrect. In tube A the dent is localized but not deep. Accordingly, this tube will pass the ball gauging tests in spite of the severity of strain at the local dent. On the other hand, tube B is regularly dented and under uniform compression, but the ball will not pass through. A more accurate method for determining dent configuration, though not without uncertainties, is by profilometry. Given a complex dent shape, present eddy current profilometry produces a few radial measurements from which tube strains can be calculated. The uncertainties of measurement on analysis affect the accuracy of calculated strains.

TUBE STRAIN CALCULATION

Determination of steam generator tube strains from probe measurements is an involved process. It requires two basic steps:

1) Definition of the shape of the deformed tube from the discrete radial measurements at probes.

2) Computation of strains from the approximated (interpolated) deformed tube shape.

Various strain calculation methods classified as either linear or nonlinear are available for each of these steps. Any of the methods from Step 1) can be combined with the alternative methods from Step 2).

As mentioned earlier, several techniques have been used in the past for approximating the shape of the deformed tube from the tube's discretely measured inner radii. These are: 1) the Fourier series, 2) cubic spline, 3) finite difference, and 4) finite element techniques.

FOURIER SERIES METHOD

In this method, the shape of the deformed tube is approximated by a finite Fourier series. The total displacement $w(\theta)$ is given by:

$$w(\theta) = \frac{a_0}{2} + \sum_{n=1}^{H} (a_n \cos n\theta + b_n \sin n\theta) \tag{1}$$

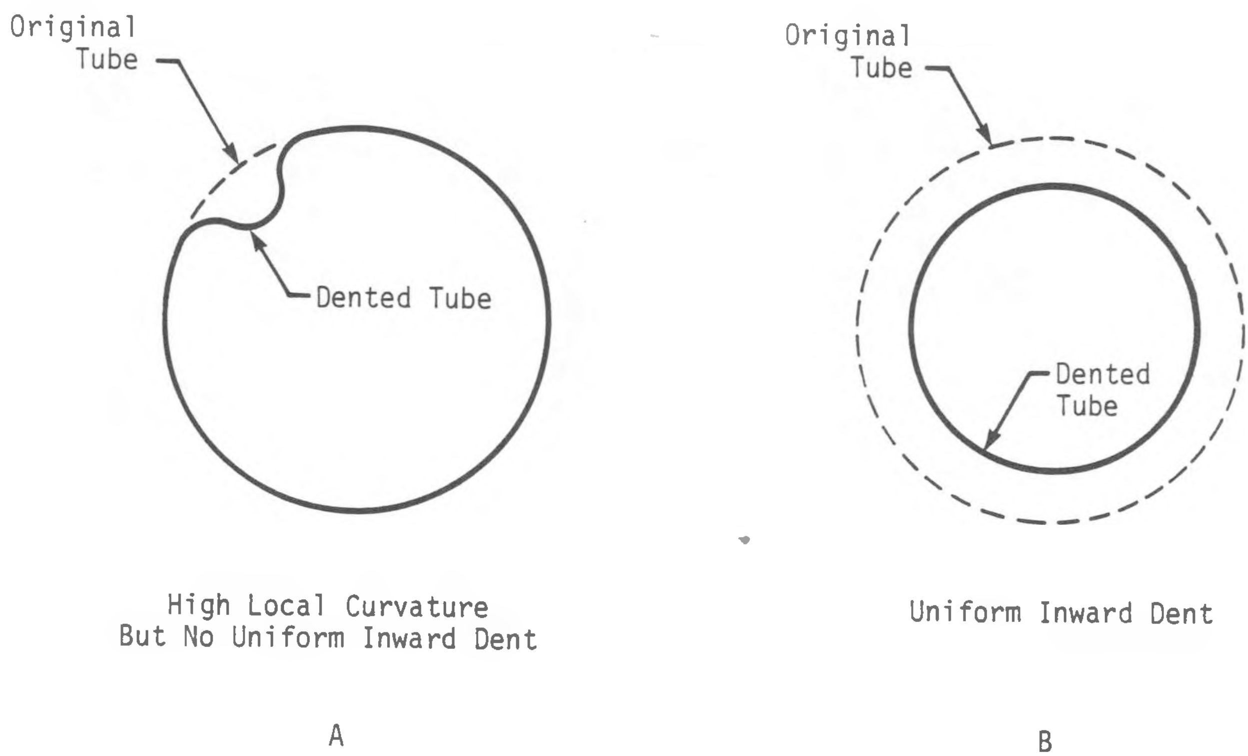

Figure 1 - Examples of Dented Tube Shapes that the Ball Gauging Method will Assess Incorrectly.

The coefficients a_n and b_n are obtained by requiring the usual minimizing of the square of the error.* The coefficients are given by

$$a_n = \frac{2}{N} \sum_{i=1}^{N} w_i \cos n\theta_i \qquad 0 \leq n < \frac{N}{2} \tag{2}$$

$$a_n = \frac{1}{N} \sum_{i=1}^{N} w_i \cos n\theta_i \qquad n = \frac{N}{2} \tag{3}$$

$$b_n = \frac{2}{N} \sum_{i=1}^{N} w_i \sin n\theta_i \qquad 1 \leq n \leq \frac{N}{2} \tag{4}$$

where N is the number of radial measurements, w_i is the radial displacement at angle θ_i, and H is the number of Fourier harmonics chosen in the analysis. It has been experienced that for an eight-point probe tube measurement technique, four harmonics is the most appropriate number to calculate strains. For sixteen and higher probe measurements, six Fourier harmonics should be used.

Generally, very high Fourier harmonics generate unrealistically large bending strain because curvature κ, related to bending strain by $\varepsilon_b = \kappa \frac{t}{2}$, is proportional to the square of n.† The number of radial measurements also affects accuracy. Efforts of Failure Analysis Associates (FAA) have quantified to some extent the improvement of sixteen-point measurement over eight-point measurement. The variation of calculated strains as a function of the probe's angular orientation is far less for the sixteen-point probe compared to the eight-point probe. Figure 2 shows the variation of bending strain between using eight- and sixteen-point probes.

PERIODIC CUBIC SPLINE METHOD

In the Cubic Spline method, the perimeter of the deformed tube cross-section is approximated with a set of cubic polynomials (one between each probe location). These polynomials and their first and second derivatives are forced to be continuous at the probe locations connecting adjacent polynomials. A periodic spline has been used by FAA where end points are joined and the resulting closed loop is described. The spline assumes the shape, which minimizes its potential energy using

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*The Fourier coefficients of a finite series are such that the square of the error is minimized (see Kreyszig (1)).

†Linear theory is assumed here. For nonlinear theory a more complicated relation between curvature and the harmonic exists. Nevertheless, curvature is strongly dependent on n^2.

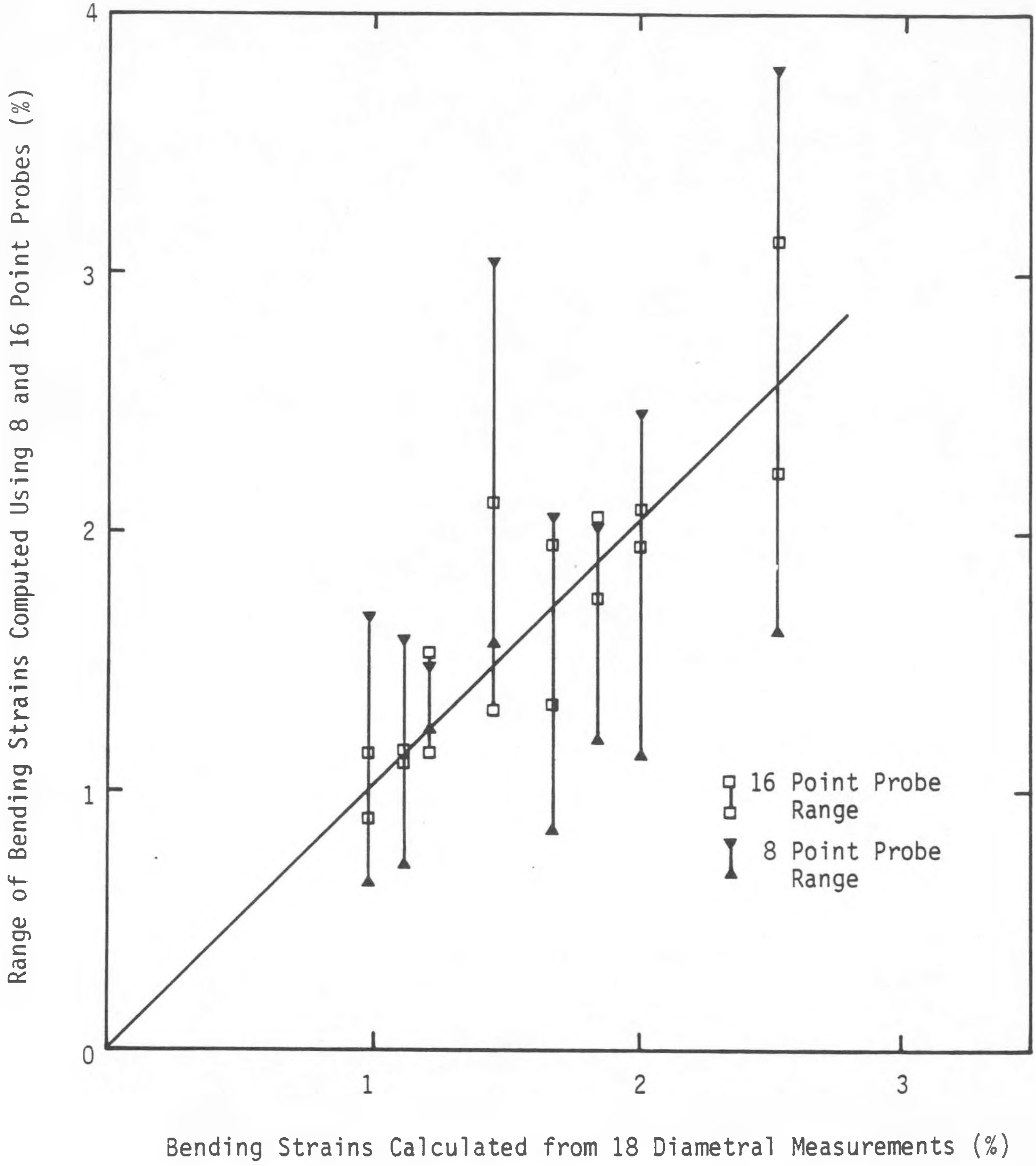

Figure 2 - Accuracy of Maximum Tube Bending Strains Calculated Using 8 and 16 Point Probes.

linear theory.* For a detailed description of splines, see Forsythe, et al. (2) and for computer codes utilizing splines, see Derbalian (3,4).

Choosing the number of probe points also affects the spline solution. Too many data points (probe data) would generate unrealistically high bending strains oscillating around the circumference with large amplitudes of strain and high frequency. The eight-point probe is known to produce stable results, although it may be inadequate to properly describe the dented shape (Figure 2). A large number (e.g., 32) of probe points has been observed to produce overly constrained results with splines generating large but totally non-existent strain oscillations. Figure 3 shows good agreement between cubic spline and Fourier series results for the eight point probe.

FINITE DIFFERENCE METHOD

Three-point and five-point central difference schemes (5) are also employed to compute tube strains. Errors are introduced on the order of $(\Delta\theta)^4$ and $(\Delta\theta)^6$ in the three-point and five-point schemes, respectively. When few radial measurements are available, the finite difference method (FDM) has the disadvantage that strains can be computed only at the discrete locations where radii are measured (i.e., probe points). More recently, it has been shown by FAA that strains must be examined at intervals as frequent as 1° to determine the maximum value of bending strain. Since the Fourier series and spline techniques fit smooth curves to the data, results may be systematically obtained at any circumferential location.

When many radial measurements are available, the FDM suffers from extreme sensitivity to small errors in the radial measurements. In light of these difficulties, the FDM should not be used in practice.

FINITE ELEMENT METHOD

The Finite Element Method (FEM) is a very general tool, although, in practice, this method may be prohibitively expensive and impractical for analyzing large numbers of dented tubes. The technique requires a lengthy elastic-plastic analysis of a tube so that its final shape will be duplicated in the deformed tube. The finite element method, applied to tube strain calculations, has the same limitation as other methods discussed in that the history of loading is not known and only the final deformed shape is available. For example, FEM model techniques can be used to crudely simulate a peanut-shaped dented tube (as observed in Turkey Point) by applying two opposite point loads to a tube and constraining the displacements at 90° from the point of load application (Figure 4). The point load approach does not seem to provide a good representation of the actual shape of the peanut-shaped tube seen in Turkey Point.

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*For linear theory the second derivative of a function approximates the curvature and the quantity

$$\int_{x_1}^{x_2} [f''(x)]^2 \, dx$$

is minimized where (x_1, x_2) represents the domain of one of the spline polynomials.

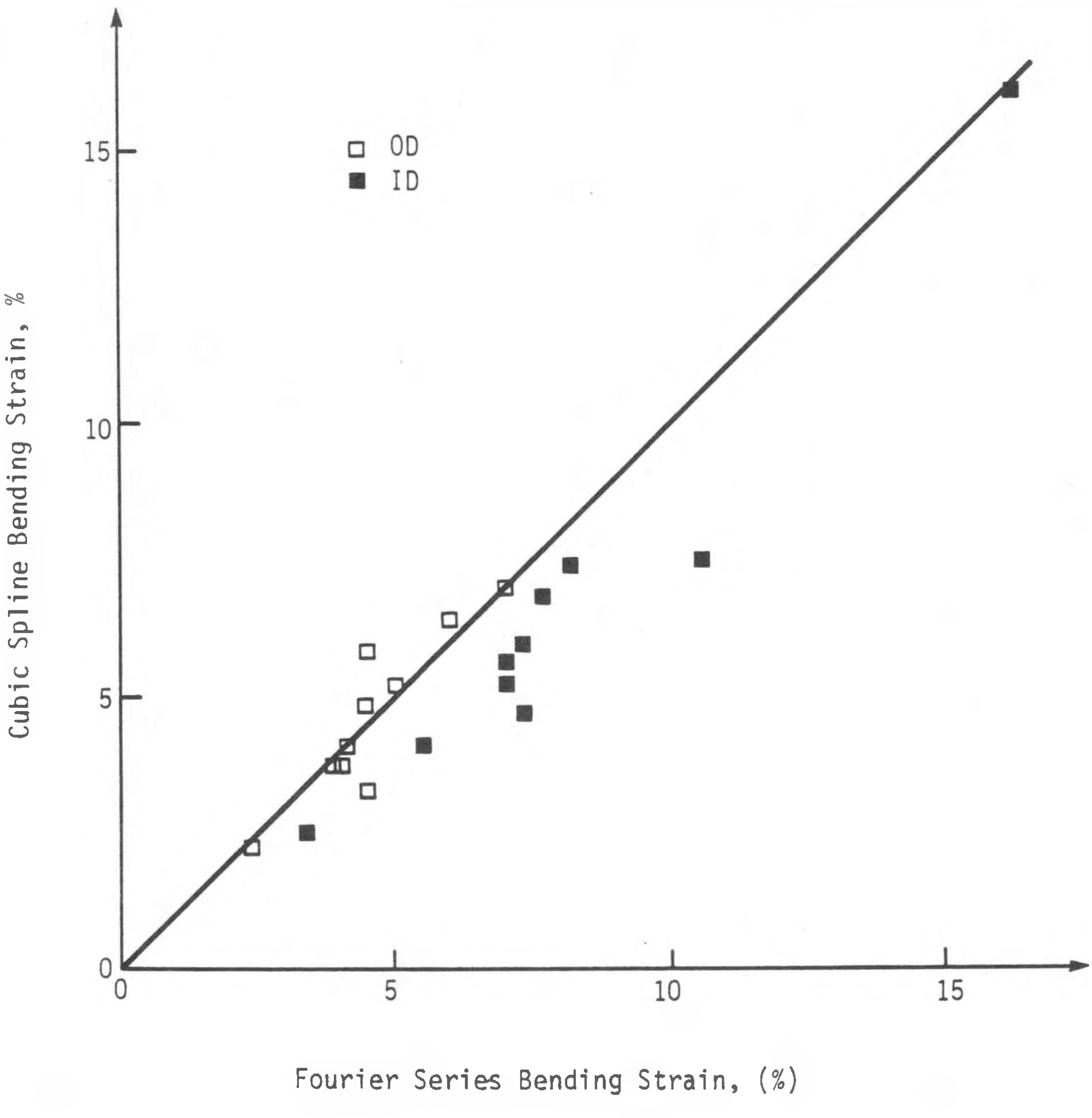

Figure 3 - Comparison of Nonlinear Bending Strain Calculation Methods for 8 Point Probe Data.

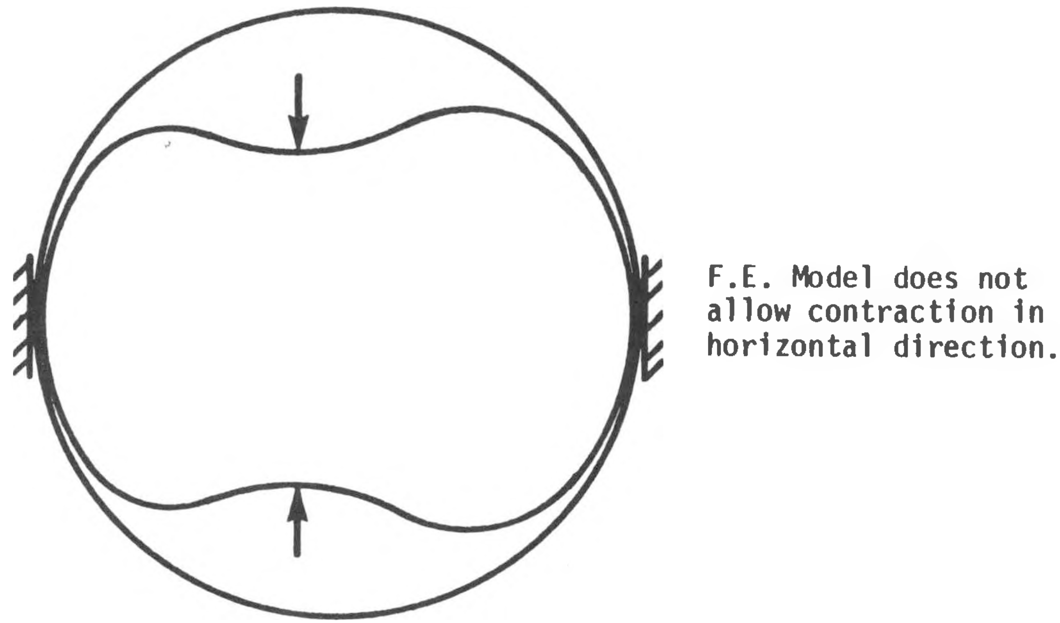

Figure 4 - Finite Element Model Simulating a Peanut Shaped Dented Tube.

It has been found that the finite element method suffers from most of the same difficulties as in the finite difference method. In particular, if displacement inputs are attempted, oscillating strains can result. This, combined with expensive computer time, probably makes FEM impractical as a profilometry processor.

TUBE STRAIN CALCULATION METHODOLOGIES

In this paper, strain calculations are described using approximated Fourier and spline shapes. An exact strain calculation technique would require the complete history of deformation of the tubes. Since such information is not available and only the final deformed tube shape is known or, rather, estimated by using one of the interpolation techniques discussed above, certain hypotheses are needed to allow determination of strain. Strains are assumed to be caused by radial displacements only; that is, each point originally on the circumference of a tube can deform either inward or outward. The total circumferential strain is divided into membrane and bending components. The membrane strain is computed by comparing the length of the deformed tube with the circumference of the original tube. Because of large deformations during tube denting, the nonlinear expression for curvature is recommended and has been shown to be superior to linear theory. FAA has developed computer programs (4, 6, 7) which use exact nonlinear strain calculations. Table 1 shows comparison of strains for Turkey Point Tube 2-49. Results indicate that the differences between linear and nonlinear theory can be on the order of 100%. Also, the cubic spline and non-linear Fourier series methods produce essentially identical results.

To refine analysis methods still further, several important issues must be addressed:

1) Membrane strain is assumed to be uniform along the entire circumference of the tube. In reality, because of complicated external loads and bonding of the oxide with the tube, membrane strains need not be uniform; in fact, in a dented region, they could be tensile. However, the average membrane strain, which is the quantity computed at present, is generally compressive.

2) The tube is generally assumed to be a thin shell. This assumption leads to the conclusion that plane sections remain planar. Thickness effects may need to be considered.

3) Radial offset of the probe from the tube centerline and the change in the orientation of the probe are known to affect the calculated strain. These effects should be evaluated.

Formulae which define the general nonlinear bending and membrane strains are derived here. These expressions are used for both the nonlinear Fourier series and cubic spline methods. The basic assumptions made in computing strains are that the thin shell theory is applicable and that strains are only caused by radial displacements. These radial displacements are the only information available from profilometer probes.

Membrane strain is computed by comparing the length of the deformed tubes perimeter with the original circumference of the tube. The perimeter of the deformed tube is computed by finding the total arc length:

Table 1

COMPARISON OF STRAINS FOR TURKEY POINT TUBE 2-49

	Membrane Strain	Max. Bending OD Strain %	Max. Bending ID Strain %
Nonlinear Fourier	-16.4	8.8	31.4
Cubic Spline	-17.1	7.4	32.2
Linear Fourier	-22.2	10.9	16.4

$$L = \int_0^{2\pi} \sqrt{\left(\frac{dx}{d\theta}\right)^2 + \left(\frac{dy}{d\theta}\right)^2} \; d\theta = \int_0^{2\pi} \sqrt{r^2 + \left(\frac{dr}{d\theta}\right)^2} \; d\theta \tag{5}$$

where r is the mid-surface radial coordinate, x and y are the rectangular Cartesian coordinates given by $x = r \cos\theta$ and $y = r \sin\theta$, and θ is the circumferential angle in radians.

The average engineering membrane strain is computed by:

$$\varepsilon_m = \frac{L - 2\pi R}{2\pi R} = \frac{1}{2\pi R} \int_0^{2\pi} \sqrt{r^2 + \left(\frac{dr}{d\theta}\right)^2} \; d\theta - 1 \tag{6}$$

where R is the nominal (mid-surface) or original tube radius. The intergral is numerically evaluated using an order two Gaussian integration (8) (exact for a cubic polynomial).

The maximum bending strain is given in terms of the curvature* (1) by the following expression:

$$\varepsilon_b = \frac{t}{2} \kappa \tag{7}$$

where κ is the curvature change of the tube due to bending alone and t is tube thickness.

It is important to recognize that during bending no stretch occurs. This is commonly described as inextensional deformation. Therefore, the curvature is computed from a deformed position of the tube which does not include the membrane stretch.

Here, a second radial distance parameter $\bar{r} = \bar{r}(\theta)$ is defined which accounts for only the pure inextensional deformation, i.e.,

$$\bar{r} = R + \tilde{w}(\theta) \tag{8}$$

where $\tilde{w}$ is the radial displacement due to bending only.

The total radial displacement is given by:

$$r = \bar{r} + w_{av} = R + w = R + \tilde{w} + w_{av} \tag{9}$$

where w_{av} is the average membrane displacement, and

$$w_{av} = R\varepsilon_m \tag{10}$$

The total curvature change in a tube is computed by:

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*Plane sections are assumed to remain planar.

$$\kappa = \frac{\dfrac{d\bar{x}}{d\theta}\dfrac{d^2\bar{y}}{d\theta^2} - \dfrac{d\bar{y}}{d\theta}\dfrac{d^2\bar{x}}{d\theta^2}}{\left[\left(\dfrac{d\bar{x}}{d\theta}\right)^2 + \left(\dfrac{d\bar{y}}{d\theta}\right)^2\right]^{3/2}} - \frac{1}{R} \tag{11}$$

where

$$\begin{aligned} \bar{x} &= \bar{r} \cos \theta \\ \bar{y} &= \bar{r} \sin \theta \end{aligned} \tag{12}$$

Substituting for $\bar{x}$ and $\bar{y}$ the above relation gives the bending strain:

$$\varepsilon_b = \frac{t}{2}\left[\frac{2\left(\dfrac{d\bar{r}}{d\theta}\right)^2 + \bar{r}^2 - \bar{r}\dfrac{d^2\bar{r}}{d\theta^2}}{\left[\bar{r}^2 + \left(\dfrac{d\bar{r}}{d\theta}\right)^2\right]^{3/2}} - \frac{1}{R}\right] \tag{13}$$

Note that:

$$\frac{d\bar{r}}{d\theta} = \frac{dr}{d\theta} = \frac{dw}{d\theta} = \frac{d\tilde{w}}{d\theta} \tag{14}$$

Computer programs have been written at FAA implementing the above techniques for the Fourier series and the spline method (4, 6, 7). Generally, for the eight-point probe, a close correlation between the spline and Fourier results has been experienced. However, as discussed earlier, the Fourier series has shown better results for higher probe points.

The effect of radial and circumferential displacements of the probe on strain measurements as well as the accuracy of the solution as a function of the number of probe points could be important. The effect of probe orientation is already familiar to FAA. As shown in earlier investigations (9), varying the orientation of the probe when measuring strains of a known deformed shape can result in large differences in computed maximum bending strain for the eight-point probe. Far less strain variation occurs for the sixteen-point probe.

Varying the probe orientation in 1° increments and calculating the maximum strains for known deformed tubes can establish a distribution of maximum computed strains. From this data it is possible to describe the probability during a random probe measurement (i.e., any orientation) of the calculated strain characterization as a given percent of the maximum. Indeed, for a sixteen-point probe, it is expected that the random calculated maximum strains will be closer to the actual maximum than an eight-point probe result.

Movement of the probe away from the center of a tube may also influence the calculated strains. Preliminary analysis without a change of orientation, however, shows that a radial displacement of the probe without a change of orientation will have only a small effect on the strain.

CONCLUSIONS AND RECOMMENDATIONS

Results of this work show that nonlinear analysis methods should be used in calculating tube strains from profilometry measurement of dents and that linear theory can be unacceptable. Both the Fourier series and periodic cubic spline methods are shown to be viable methods for accurately computing strains. However, care should be given in selecting parameters, such as the number of Fourier harmonics and discrete radial dent depths used in these methods. Too many Fourier harmonics will cause oscillating and diverging results; too many radial dent depth values will similarly cause an oscillatory solution with the spline method. The eight-point eddy current probe is limited in accuracy. Predicted strains can significantly vary due to probe orientation. Eight points simply do not give a good picture of the dented shape. A sixteen-point probe would be significantly more accurate. In light of this, the optical probe seems very promising. Even if a selected number of points are chosen from the dented profile of the tube, the strains can be determined accurately and tested by varying the probe orientation (i.e., the selection of points).

The method of ball gauging is shown to be misleading. There can be dented tubes under essentially uniform compression with hardly any tensile strain, which will fail the ball test. Even more critical are those tubes with local singular and lobe dents that will permit the ball to pass through but be under high tensile strain due to the severe local curvature.

A tube-plugging criterion is needed to correlate failure to tube strain, strain rate, and temperature. Such a program is currently underway at Failure Analysis Associates with encouraging preliminary results.

ACKNOWLEDGEMENTS

The authors wish to acknowledge the work of their colleagues at Failure Analysis Associates, in particular, Dr. Jerrell Thomas and Mr. Scott Rau, who have greatly contributed to the analysis of steam generator tube denting and the development of a tube-plugging criterion.

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COMMENTS ON THE PREDICTION OF SERVICE LIVES OF POLYMERIC MATERIALS BY THE ARRHENIUS METHOD

Dr. Morton Brown, E. I. Du Pont de Nemours Co.

Background

In recent years the Arrhenius equation has been incorporated into certain important performance specifications in order to predict the long-term performance characteristics of polymeric materials. Furthermore, Underwriters' Laboratory has used the technique for many years to determine a so-called "temperature rating" for polymerics. This discussion is intended to clarify the theoretical and experimental bases for the Arrhenius equation and to discuss its applicability to the question of polymer aging.

Discussion

In 1889 Sven Agustus Arrhenius published his landmark research work on the temperature dependency of reaction rate. In this investigation of the gas-phase reaction

$$H_2(gas) + I_2(gas) \longrightarrow 2\,HI(gas) \qquad (1)$$

he found that the measured rate of the reaction (measured either by the disappearance of the reactants or by the appearance of the product) could be related to the temperature by the exponential curve of Figure 1. This figure corresponds to the common statement that a reaction rate increases by a factor of 2-3 for each ten-degree increase in temperature. Rate curves of similar shape came to be known subsequently as examples of "Arrhenius Behavior".

Discussion (Continued)

Rate curves such as illustrated in Figures 2-5 were discovered in subsequent years and are known as non-Arrhenius curves. Figure 2 is characteristic of an explosive reaction. Figure 3 is observed in catalytic hydrogenations and in enzyme reactions. Figure 4 is observed in the oxidation of carbon and 5 is observed in the nitric oxide-oxygen reaction.

Arrhenius found that the rate of the hydrogen-iodine reaction could be expressed as:

$$\text{Rate} = k \text{ (conc. of hydrogen X conc. of iodine)} \tag{2}$$

where k is the so-called rate constant. His choice of a homogeneous (single phase) reaction was a conscious one since he recognized that mass transfer across phase boundaries could, in certain circumstances, represent the rate-controlling factor. For homogeneous reactions he found that a plot of ln k (or log k) vs. 1/T is linear over small temperature ranges and has a negative slope. This result is equivalent to the Arrhenius equation (3).

$$d \ln k/dt = E_a/RT^2 \tag{3}$$

If E_A, the activation energy is constant with temperature, then integration results in

$$\ln k = - E_A/RT + \text{constant} \tag{4}$$

or

$$k = Ze^{-E_a/RT} \tag{5}$$

In practice Arrhenius found that E_A was constant within experimental error over a temperature range which did not exceed 10°C.

Discussion (Continued)

The following points are important limitations in the applicability of the Arrhenius equation:

a) It applies only to a homogeneous (single phase) reaction.

b) The plot of ln k vs. 1/T is linear, and therefore applicable, over a narrow temperature range.

c) The line may not be extrapolated outside the temperature range of the data points since the extrapolation implies that Ea is constant throughout the range, and that the reaction mechanism and the rate-controlling step (if a sequential reaction) are also the same throughout the range.

Practical Aspects of the Aging of Polymeric Materials

The most common application of the Arrhenius equation to the aging of polymeric materials is contained in UL Standard 746B, "Polymeric Materials - Long-Term Property Evaluations". In all cases UL uses a failure criterion of 50% of the unaged value. It applies this criterion to a number of physical, electrical and flame properties.

In the case of electrical cables, cable engineers measure the decay of ultimate elongation with time at several aging temperatures. Their criteria of failure, however, are not uniform. Some companies rely on the time to 50% retention of the original elongation, while others use the time to decay to an absolute value of 100% as the criterion. These values are plotted on a semi-log scale vs. 1/T and are extrapolated to 90°C to determine a service life.

It has already been noted that Arrhenius recognized that his equation could not be applied to a heterogeneous system because of mass transfer considerations. Figures 6 and 7 illustrate this point. Figure 6 summarizes an EPR insulation aged at four temperatures (150°C, 136°C, 125°C, and 121°C) for two thicknesses of test dumbbells. The thicknesses chosen were at the extremes of the values permitted by ASTM D-412-76, namely .060 inches (1.52 mm) and .120 inches (3.04 mm). The chosen test end point was 100% absolute elongation.

The data form two distinct simple regression lines with the thinner dumbbells producing shorter aging periods and having a steeper slope than that derived from the thick dumbbells. Although the data indicate that the factor of mass transfer is of greater importance at high aging temperatures and decreases in importance as the aging temperature decreases, the important point to note is that extrapolation of the lines to 90°C gives widely differing service lives.

Figure 7 shows the same EPR compound aged in an oven meeting the requirements of ASTM D-2436-68 (100-200 air changes per hour) vs. aging in a tube by the procedure of ASTM D-865-62. This latter procedure ages the samples in large borosilicate glass test tubes, loosely stoppered, equipped with vent tubes in the stopper, and heated by immersion in an oil bath. This procedure differs from the former in that the samples are exposed to very low levels of air flow and exchange in the tubes. As in the previous example, the simple regression lines converge, indicating that air flow rate is a less-important factor at the lower aging temperatures. However, the extrapolated service lines at 90°C are considerably different.

Finally, Figure 8 indicates the effect on the simple regression line of three different antioxidants. Again the lines tend to converge at the lower aging temperatures, indicating that the antioxidants are becoming more equal at these temperatures. But the key point is that the extrapolated 90°C service lines are longer for the inferior antioxidant!

A further criticism of the applicability of the Arrhenius equation to the aging of polymeric materials is that even if the rate of chain scission could be related truly to 1/T, this rate is only related distantly and indirectly to the ultimate elongation. The latter is much more a function of compounding variables, as well as the type of rubber, its molecular weight and its MW distribution. (In fact, the elongation of some rubbers, notably butyl rubber, do not decrease with aging, but rather increase.) To illustrate this effect, Figure 9 shows the relation of elongation at break with aging time at 150°C for two different EP rubbers. Both rubbers have the same E/P ratio, but one has a slight crystalline content, the result of polymerization of the ethylene into "blocks" which crystallize, rather than in a random fashion. It is precisely this type of aging data which is used to develop an Arrhenius plot, but depending on the failure point chosen - 50% retained elongation or 100% absolute elongation - a different EPR is preferred. Similar criticisms can be developed if other failure criteria are employed.

Theoretical Background of Simple Linear Regression Analysis

The relationship of two variables, one independent (x) and one dependent (y) is often treated statistically by simple linear regression analysis. The term "simple" is used if y is a function of only variable, x, and the term "linear" if the expression $y = ax + b$ is chosen as the simplest relation for the two variables. However, it is must be emphasized that the best straight line representing $y = ax + b$ is simply the line wherein the square root of the sum of the squares of the standard deviations is at a minimum. It does not imply that the true function linking x and y is linear - in fact, it does not even imply a "cause-and-effect" relationship between x and y.

It follows that the linear regression line cannot be extrapolated to a region outside the area where data do not exist. If we do not know the actual relationship between x and y in the area where data exist, we certainly can't assume a relationship outside this area.

The fallacy inherent in this extrapolation can be further illustrated in Figure 10 in which AB represents the simple linear regression of x and y, while the segment BC is the extrapolated portion of the line. It is possible to define the 90% confidence limits for every point on the line AB, shown as the dogbone-shaped area. Within the contained area every value of y for a given x has at least a 90% probability of being the correct value for $y = f(x)$. It can be seen that this area widens geometrically at the extremities of AB (generally because the available data at the extremities are too sparse to define adequately the 90% limits) and this area in effect increases to infinity in the extrapolated region.

<u>Theoretical Background of Simple Linear Regression Analysis</u> (Continued)

It is unfortunate that such flawed statistical methods are used to predict service lines of polymeric materials, since the method implies an unwarranted degree of scientific reliability. Perhaps it is fortunate that, in those few cases in which the predictions can be compared with actual data obtained from polymers in service, the predicted values appear to be rather conservative. We can't always rely on this fortuitous result, however. Perhaps it is necessary to rely upon continuous upgrading of these polymerics in the light of the most recent in-service performance, rather like the method used in the aircraft industry.

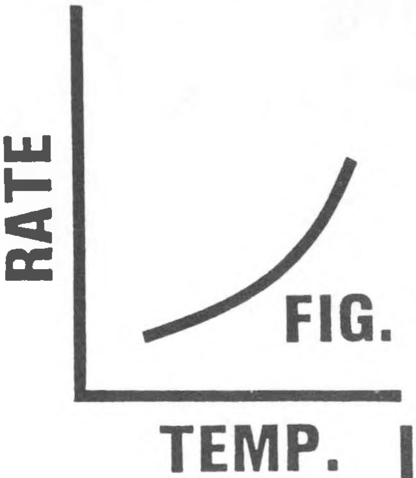


FIG. 1

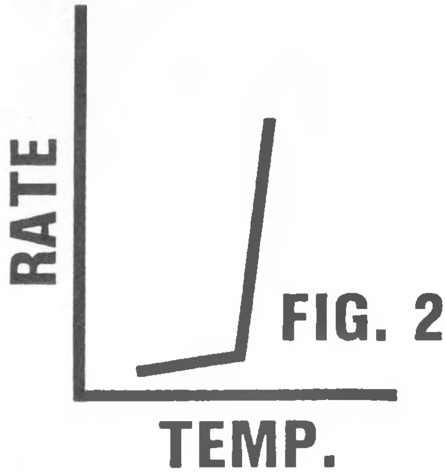


FIG. 2

FIG. 3

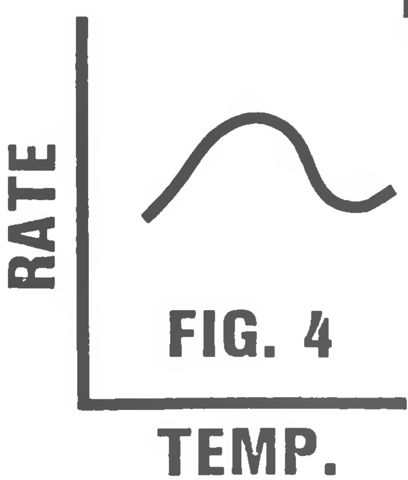


FIG. 4

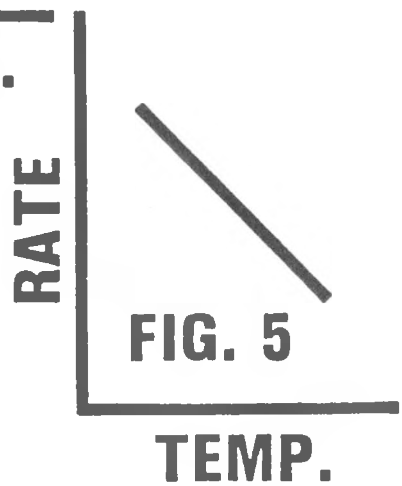


FIG. 5

Figure 6

TIME TO 100% ELONGATION
VS.
TEMPERATURE
EPR AT 1.5mm AND 3.0mm

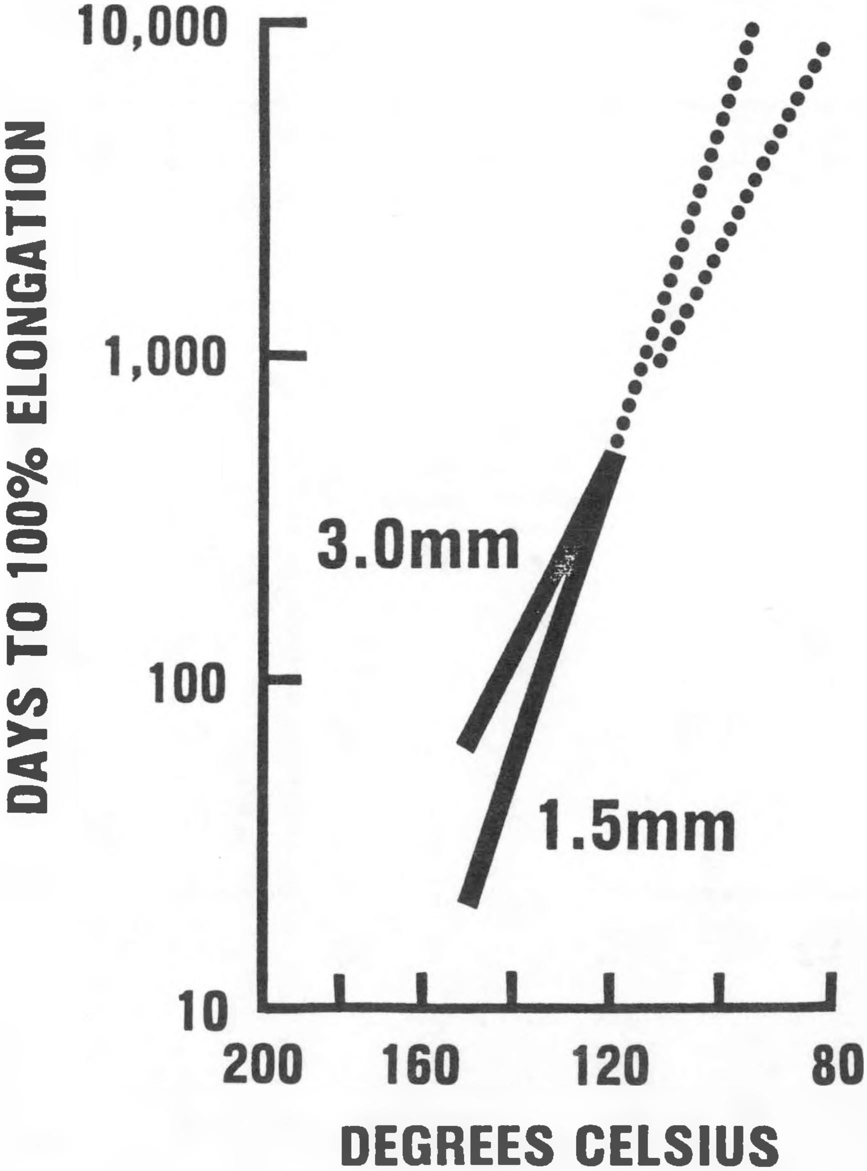


Figure 7

TIME TO 100% ELONGATION
vs.
TEMPERATURE
EPR IN TUBE vs. OVEN AGING

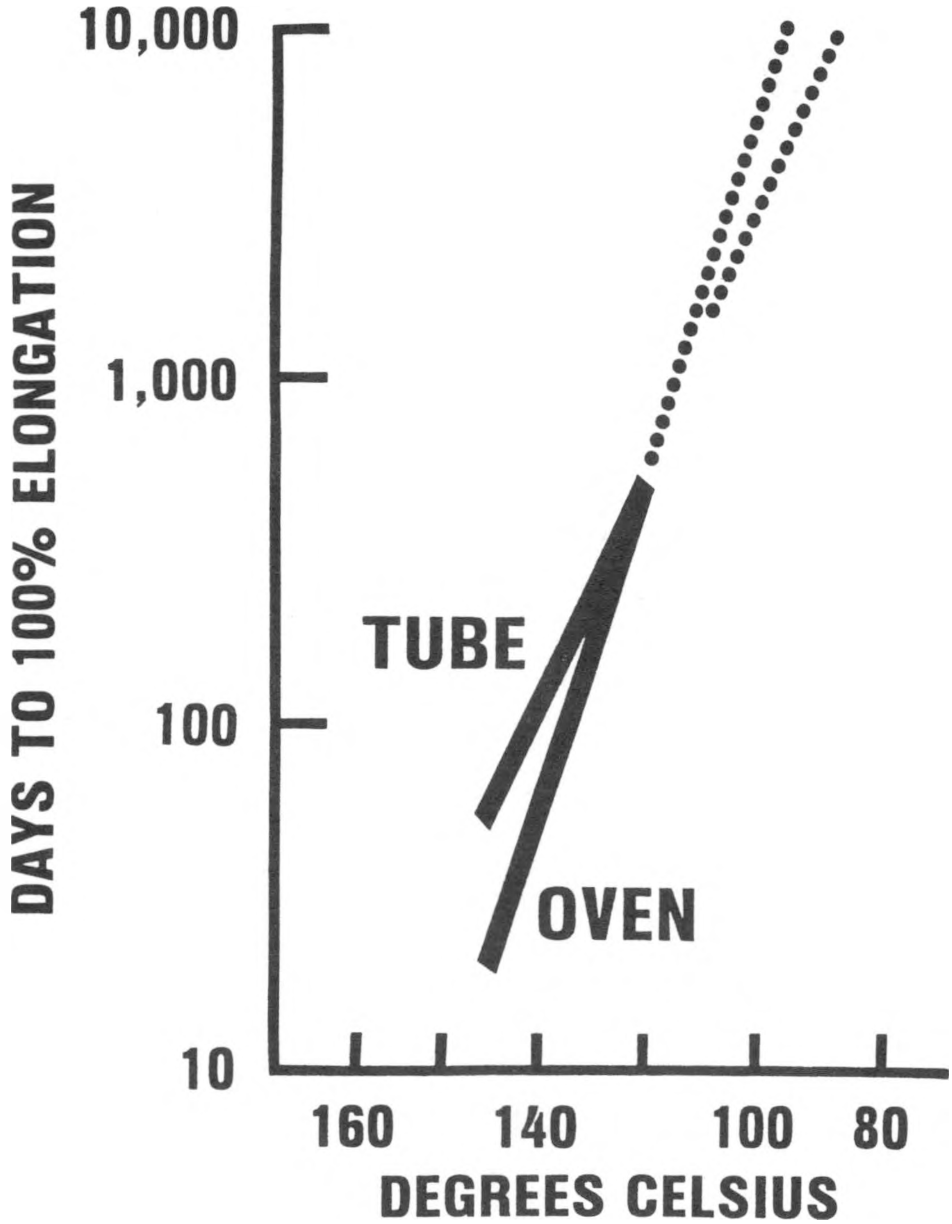


Figure 8

TIME TO 100% ELONGATION
VS.
TEMPERATURE
EFFECT OF ANTIOXIDANT IN EPR

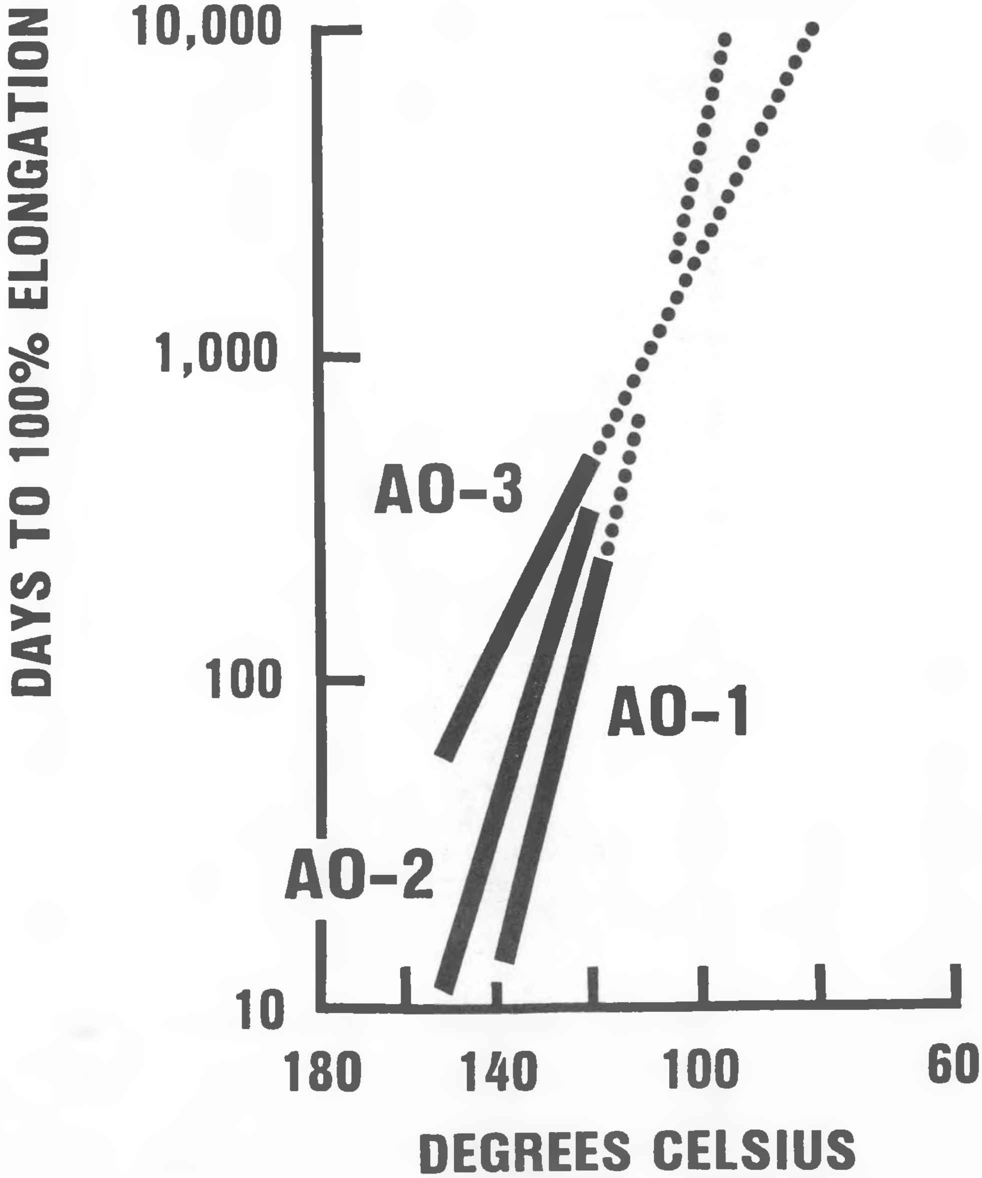


Figure 9

ELONGATION vs. DAYS AT 150°C. SEMI-CRYSTALLINE vs. AMORPHOUS

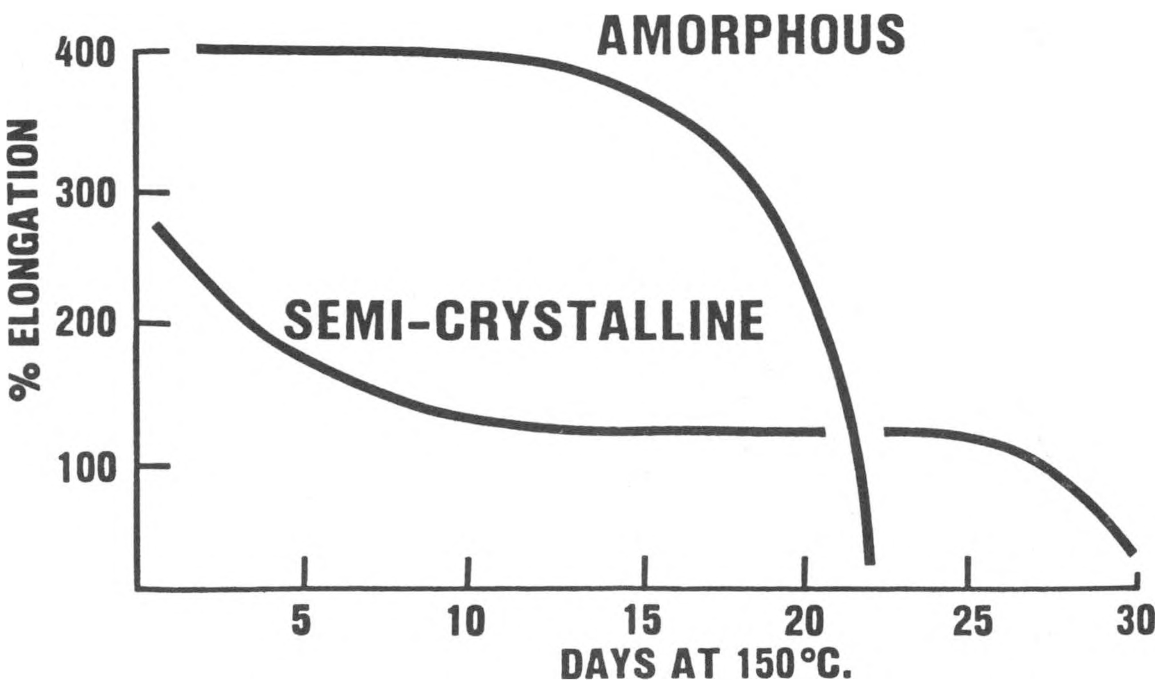
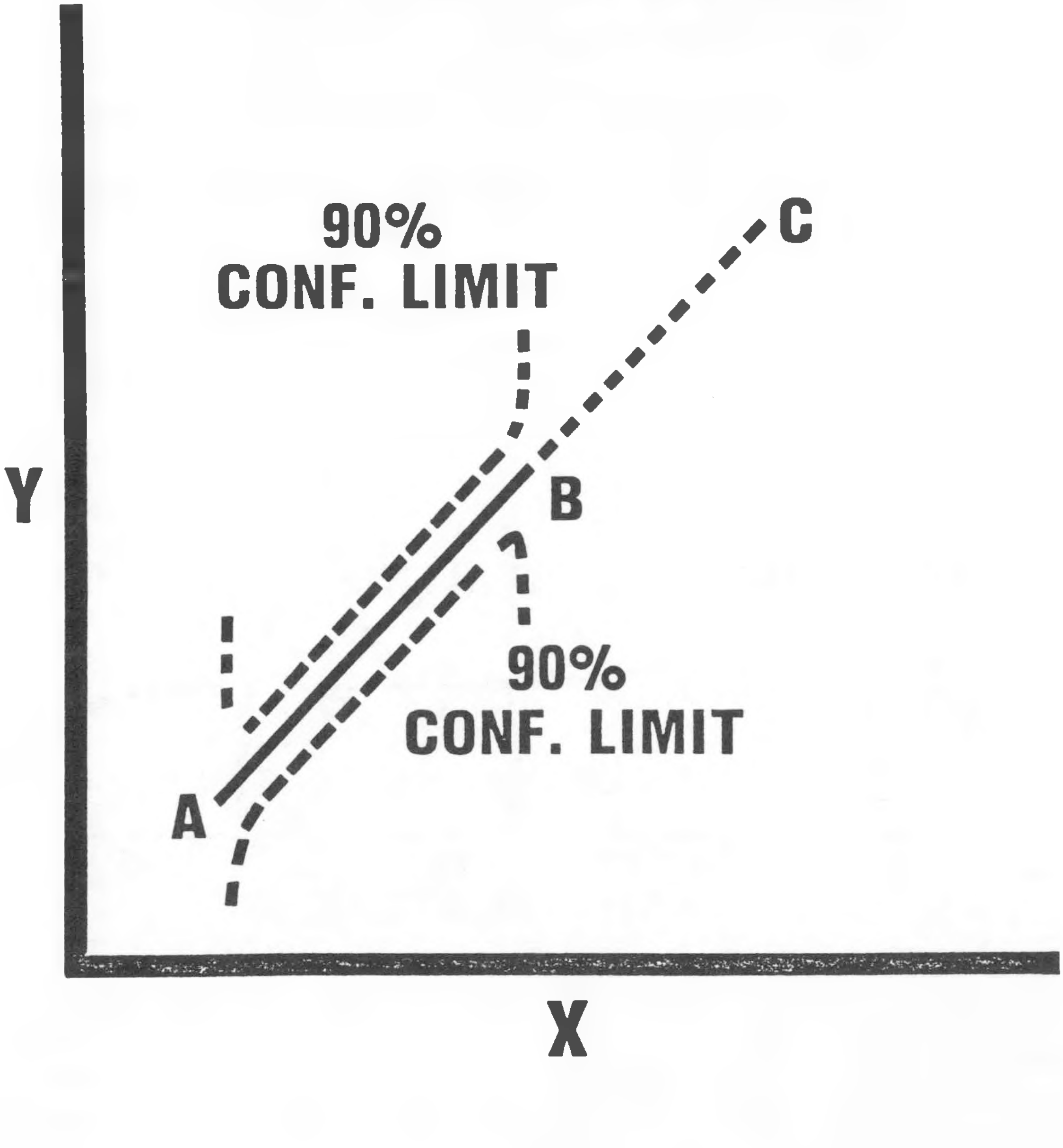


Figure 10

SIMPLE LINEAR REGRESSION



A BRIEF OVERVIEW AND COMPARISON OF THERMAL AGING MODELS USED FOR POWER PLANT EQUIPMENT QUALIFICATION

Martin F. Chamow
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ABSTRACT

Some of the confusion that often accompanies conversions or comparisons among the various thermal aging models used for power plant equipment qualification evaluations is alleviated by the methods described in this paper. Specifically, an Arrhenius-based model is examined and compared with relatively more approximate models, such as the so-called 'ten-degree rule' and other representations of aging acceleration found in test reports and technical literature. Based on the comparisons performed, useful conversions are developed, and are presented as formulas, tables, and graphs.

INTRODUCTION

In 1889, Svante August Arrhenius, then a promising and sometimes controversial young Swedish chemist who eventually was to become a Nobel Laureate, published a proposal of a means to account for the influence of temperature on the rate of a particular type of chemical process.[1,2] Since then, methods based on his formulations have been developed and used for predicting the life expectancies of various materials under storage and operating temperature conditions, in terms of accelerated testing results (i.e., relatively short term tests at relatively high temperatures).

Arrhenius' proposal, although initiated from empirical evidence, has been confirmed to have a strong theoretical foundation, and is thus potentially very accurate. Other methods used for similar purposes involve temperature coefficient rules. For example, the so-called 'ten-degree rule' is used, in which life expectancy doubles for every temperature reduction of ten degrees C. These methods are also empirical but are not theoretically founded, and are more approximate and limited in range of usefulness than are Arrhenius-based methods.

The Arrhenius methods are viewed by some as being approximate and limited in applicability. This paper does not debate this issue, but instead assumes the appropriateness of the Arrhenius-type formulations in situations for which there is only one (or one extremely dominant) degradation process and corresponding activation energy. In any case, the temperature coefficient rules are more approximate than the Arrhenius relationships.

Inasmuch as the Arrhenius and the temperature coefficient methods are both widely used, the ability to perform comparisons between them is often important. Carfagno and Gibson have already made strides in this direction;[3] this paper can be considered as a limited extension of their results. Also, in a broader sense, this paper presents some results of an investigation into practical methods and manipulations that might be useful to equipment qualification engineers.

All of the work reported in this paper is in terms of degrees C and degrees K. A parallel development in terms of degrees F and degrees R was not felt to be justified at this time, since conversions can be easily and readily performed by the user.

PRELIMINARIES

Arrhenius Relationship

Typically, the Arrhenius-type formulation for relating lifetimes at two temperatures is written as:

$$\frac{L_2}{L_1} = \exp\left[\frac{\emptyset}{k}\left(\frac{1}{T_2} - \frac{1}{T_1}\right)\right] \tag{1}$$

or

$$\frac{L_2}{L_1} = \exp\left[\frac{\emptyset}{k} \cdot \frac{(T_1 - T_2)}{T_1 T_2}\right] \tag{2}$$

where:

T_1 = Test temperature, degrees K
T_2 = Service temperature, degrees K
L_1 = Test time at temperature T_1
L_2 = Expected life at temperature T_2
$\emptyset$ = Activation energy, eV
k = Boltzmann's constant (0.8617×10^{-4} eV per degree K)

The units for L_1 and L_2 are consistent, in hours, days, years, etc.

The graphical plot of the Arrhenius relationship in Equations (1) and (2) is linear for logarithmic time vs. reciprocal absolute temperature coordinates. The straight line so derived is sometimes referred to as a 'life line.' It is inherent in the Arrhenius method that 'end-point' or 'failure' definitions are the same for all pairs of coordinates along this line, for the region in which the line is valid.

Temperature Coefficient Rules

A general expression for the temperature coefficient (TC) rules is:

$$\frac{L_2}{L_1} = m^{(T_1 - T_2)/d} \tag{3}$$

or

$$\frac{L_2}{L_1} = \exp\left[(T_1 - T_2)\ \frac{\ln m}{d}\right] \tag{4}$$

where T_1, T_2, L_1, and L_2 are as above, and m is the multiplier applied to the expected liftime for a temperature decrease of d degrees C (or K).

TC rules are usually expressed as 'm times per d degrees,' for which a shorthand expression 'm*/dC' is used in this paper. The effective range found in the literature appears to extend from about 1.5*/10C to a little more than 4*/10C.

As mentioned earlier, a very familiar TC rule is the so-called 'ten-degree rule,' under which life expectancy doubles for each ten degrees C temperature decrease. In the shorthand notation of this paper this becomes 2*/10C. Mathematically, the 2*/10C rule is expressed, in accordance with Equation (3), as:

$$\frac{L_2}{L_1} = 2^{(T_1 - T_2)/10} \tag{5}$$

COMPARISON OF METHODS

In the interest of comparing methods, and developing conditions for equivalence between the Arrhenius model, Equation (2), and the temperature coefficient model, Equation (4), these two equations are set equal to obtain:

$$\exp\left[\frac{\emptyset}{k} \cdot \frac{(T_1 - T_2)}{T_1 T_2}\right] = \exp\left[(T_1 - T_2)\ \frac{\ln m}{d}\right] \tag{6}$$

Taking natural logarithms, or simply equating the exponential arguments, produces:

$$\frac{\emptyset}{k} \cdot \frac{(T_1 - T_2)}{T_1 T_2} = (T_1 - T_2)\ \frac{\ln m}{d} \tag{7}$$

Cancellation, rearrangement, and definition of new terms results in:

$$\phi_I = \frac{k\ T_G^2\ \ln m}{d} \tag{8}$$

where, under the equivalence conditions:

ϕ_I = Implied activation energy, eV
T_G = Geometric mean of T_1 and T_2, degrees K

Further arrangement and definition yields:

$$\phi_I = \left[\frac{T_G}{T_I}\right]^2 \tag{9}$$

for which:

$$T_I = \left[\frac{d}{k\ \ln m}\right]^{1/2} \tag{10}$$

and where:

T_I = An implied quantity, related to temperature and activation energy, for which $\phi_I = 1$

It is convenient to treat T_I as a temperature although this is not true dimensionally. (The dimensionally correct units for T_I are degrees K per $\sqrt{eV}$.)

It is important to note that the equivalence between the TC and Arrhenius models is a function of the geometric mean, T_G, of the end-point temperatures T_1 and T_2, and not the temperature difference $\Delta T = T_1 - T_2$.

In other words, the equivalence relationships are invariant for all possible combinations of ΔT, T_1, and T_2 values corresponding to a given T_G.

Some of the significance of Equations (8) through (10) can be demonstrated by reference to Figures 1 through 4.

Figure 1 is a plot of Equation (9), and is a normalized representation of the equivalence relationships. The curve, or equivalence locus, separates two important regions as identified in the figure. Above the curve, the TC model is more conservative than the Arrhenius. This means that, for a given short term test, the TC model would lead one to expect a shorter service life than would be determined by the Arrhenius. Below the curve, the opposite is true.

The importance of the regions of Figure 1 lies in the fact that they allow one to distinguish between suitable and unsuitable applications of the TC models. Strictly, the use of TC models should be considered only in the absence of actual activation energy information.

In essence, the following rules apply:

1. For a given T_G/T_I ratio, if the assumed (or actual) activation energy is greater than the $\emptyset_I$ value of the curve, or as calculated by Equations (8) or (9), then it is appropriate to employ a TC model, which would be conservative.

2. Otherwise, the Arrhenius model should be used.

3. For coordinates on the curve, results of both methods are equivalent.

4. To change the dictates of this rule, a different assumed activation energy can be selected, if it is reasonable to do so.

Referring now to Figure 2, the presentation of Figure 1 is expanded to explicitly depict a number of specific TC rules. The abscissa is no longer normalized and is expressed as a T_G scale. (The parenthetical b values are explained below.)

Figures 3 and 4 repeat the information of Figures 1 and 2, respectively, but with logarithmic coordinates introduced to produce linear plots.

Attention is now directed to Figure 5 in which some equivalences among TC rules are depicted. All pairs of coordinates along a given line in Figure 5 define equivalent m*/dC rules. For example, the bottom line (b = 0.0347) shows that 2*/20C = 4*/40C, etc. Also, the middle line (b = 0.0693) shows that 2*/10C = 4*/20C, etc.

The defining equation for Figure 5 is derived by rearranging Equation (3) to yield:

$$\frac{L_2}{L_1} = m^{(T_1 - T_2)/d} = \left[m^{1/d}\right]^{\Delta T} \qquad (11)$$

Setting the L_2/L_1 ratio of Equation (11) equal to a constant, normalizing to a unit value for ΔT, and taking natural logarithms, results in:

$$\ln m = bd \qquad (12)$$

from which:

$$b = \frac{\ln m}{d} \qquad (13)$$

By graphing Equation (12) such that the abscissa is d and the ordinate is the natural logarithm of m, as in Figure 5, b is the slope of the straight line so produced. Each line is designated by its b-value.

The quantity b, as defined by Equation (13), is important in that it is indicative of the severity of a TC rule. The value of b increases with the degree of severity. That is, 4*/10C, for which b = 0.1386, is obviously more severe than 2*/10C, for which b = 0.0693.

Another measure of interest derived from Equation (11) is:

$$a = m^{1/d} \tag{14}$$

which is merely the paranthetical term of Equation (11). The quantity a is useful in calculations in which the effect of ΔT is of direct interest, in that:

$$\frac{L_2}{L_1} = a^{\Delta T} \tag{15}$$

Table 1 presents, for a number of TC rules, values for a, b, and T_I, as defined by Equations (14), (13), and (10), respectively. Equation (10) can be modified to reflect the definition of b to obtain:

$$T_I = \left[\frac{d}{kb}\right]^{1/2} \tag{16}$$

Also, conversions between a and b are accomplished by:

$$a = e^b$$

and

$$b = \ln a$$

ARRHENIUS ACTIVATION ENERGY

We now turn to one final topic that is familiar to all who work with the Arrhenius model, namely the extraction of the underlying activation energy from the Arrhenius relationship. This topic is felt to be necessary for completeness of coverage since much of the discussion in this paper is dependent on the knowledge of the actual activation energy.

An expression for the equivalent, or underlying Arrhenius activation energy, $\emptyset_E$, is shown in Table 2, along with the corresponding $\emptyset_I$ formulation. It is interesting to note that the denominator of the $\emptyset_E$ expression is linear in $\Delta X = X_2 - X_1$, where X is reciprocal absolute temperature (degrees K). Thus, if necessary, a parametric plot can be drawn involving $\emptyset_E$, $\ln(t_2/t_1)$, and ΔX, for the purpose of extracting $\emptyset_E$, and would be especially useful if the Arrhenius equation itself is presented graphically.

SUMMARY OF RESULTS

Two results of this paper are felt to be primarily significant, namely:

1. Comparisons between the Arrhenius and TC models, including tests to determine the appropriateness of using a TC model.

2. Comparisons among the various TC rules themselves.

These results, along with other details in the paper, should contribute some flexibility to the art of using the various models discussed.

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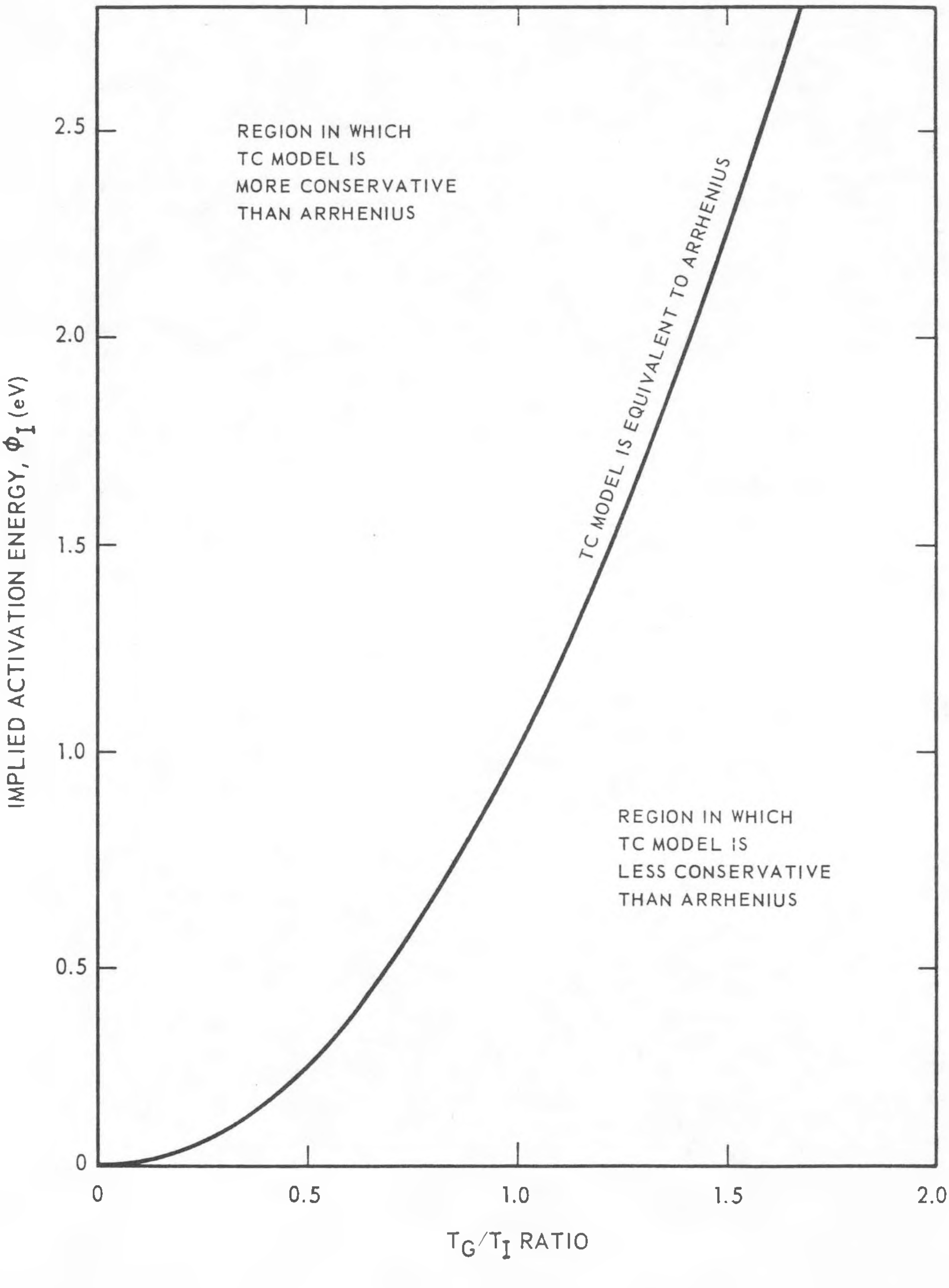
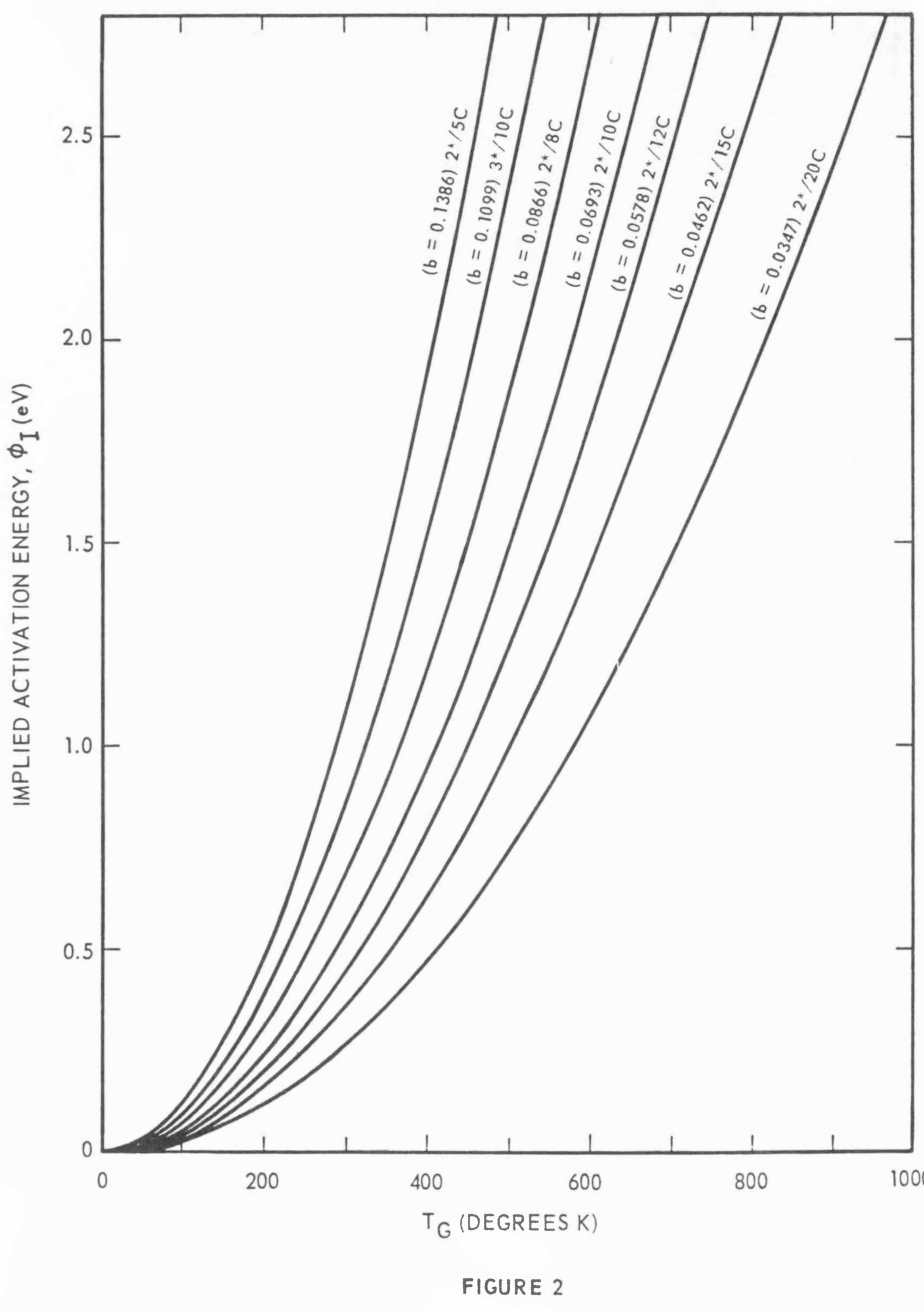


IMPLIED ACTIVATION ENERGY, ϕ_I (eV)
2.5
2.0
1.5
1.0
0.5
0
0
0.5
1.0
1.5
2.0
T_G/T_I RATIO
REGION IN WHICH TC MODEL IS MORE CONSERVATIVE THAN ARRHENIUS
TC MODEL IS EQUIVALENT TO ARRHENIUS
REGION IN WHICH TC MODEL IS LESS CONSERVATIVE THAN ARRHENIUS

FIGURE 1



IMPLIED ACTIVATION ENERGY, ϕ_I (eV)
2.5
2.0
1.5
1.0
0.5
0
0
200
400
600
800
1000
T_G (DEGREES K)
(b = 0.1386) 2*/5C
(b = 0.1099) 3*/10C
(b = 0.0866) 2*/8C
(b = 0.0693) 2*/10C
(b = 0.0578) 2*/12C
(b = 0.0462) 2*/15C
(b = 0.0347) 2*/20C

FIGURE 2

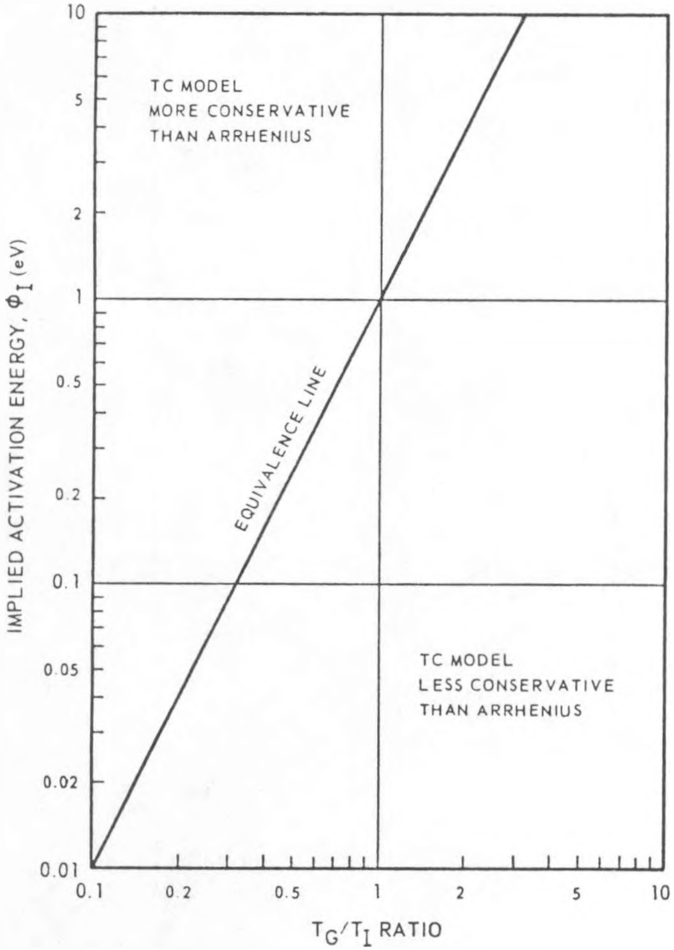


FIGURE 3

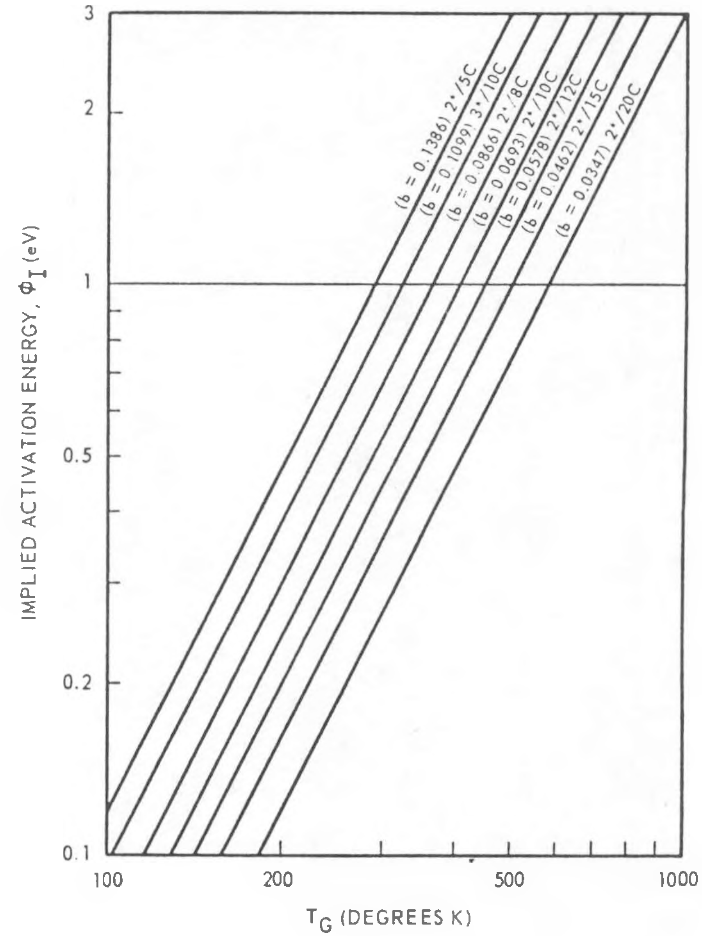


FIGURE 4

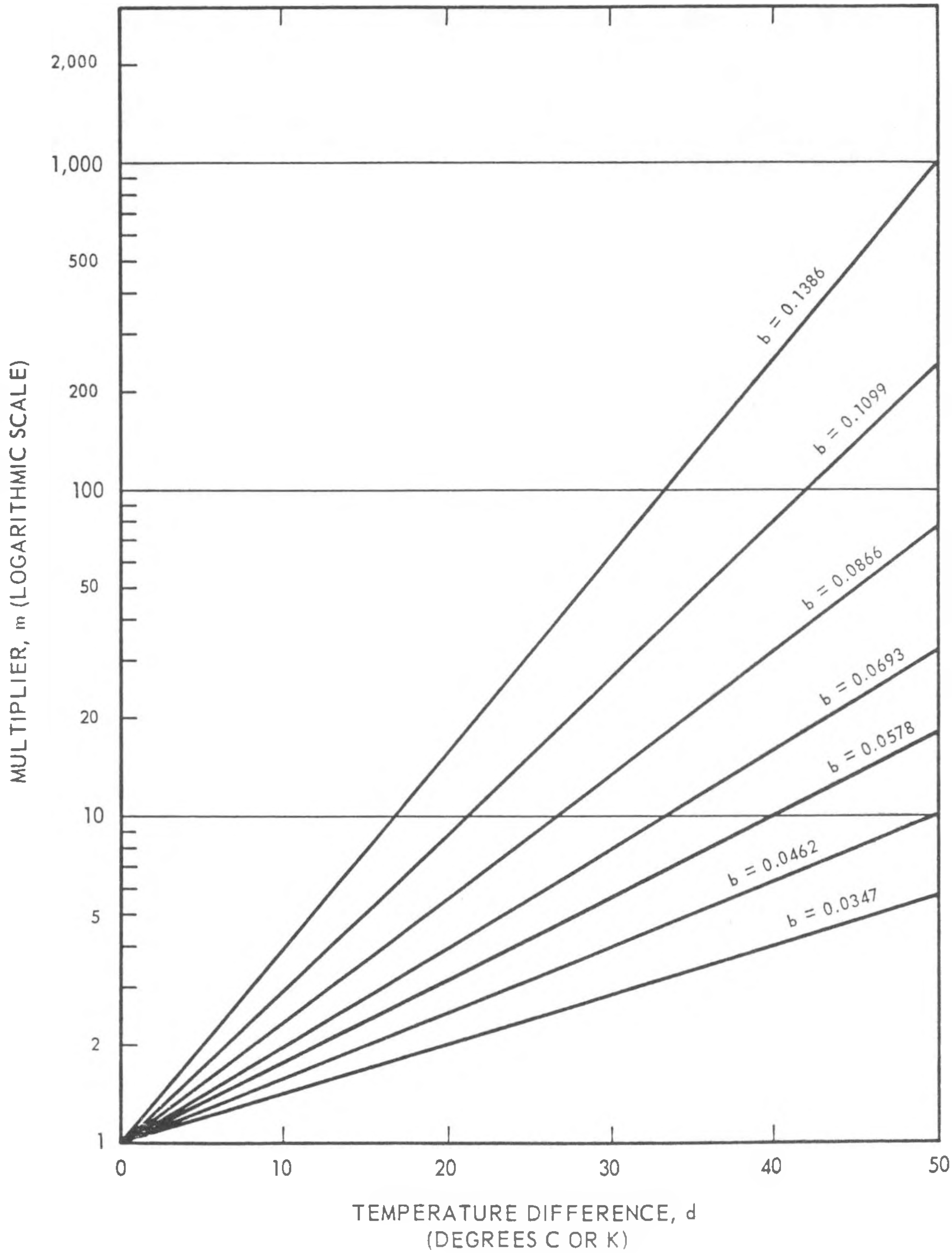

MULTIPLIER, m (LOGARITHMIC SCALE)
2,000
1,000
500
200
100
50
20
10
5
2
1
b = 0.1386
b = 0.1099
b = 0.0866
b = 0.0693
b = 0.0578
b = 0.0462
b = 0.0347
0
10
20
30
40
50
TEMPERATURE DIFFERENCE, d
(DEGREES C OR K)

FIGURE 5

TABLE 1
TYPICAL VALUES

a	b	T_I	SOME CORRESPONDING TEMPERATURE COEFFICIENTS
1.1487	0.1386	289.3	2*/5C, 4*/10C
1.1161	0.1099	325.0	3*/10C, 2*/6.31C
1.0905	0.0866	366.0	2*/8C, 2.38*/10C
1.0718	0.0693	409.2	2*/10C
1.0595	0.0578	448.2	2*/12C, 1.78*/10C
1.0473	0.0462	501.1	2*/15C, 1.59*/10C
1.0353	0.0347	578.7	2*/20C, 1.41*/10C

TABLE 2
EFFECTIVE ACTIVATION ENERGIES

DESCRIPTION	EQUATION
EQUIVALENT FROM ARRHENIUS MODEL	$\phi_E = \dfrac{k \ln (t_2/t_1)}{X_2 - X_1}$ WHERE: $X_1 = 1/T_1$, $X_2 = 1/T_2$
IMPLIED BY ARRHENIUS/ TC EQUIVALENCE	$\phi_I = T_G^2 \left(\dfrac{k \ln m}{d}\right) = \left(\dfrac{T_G}{T_I}\right)^2$ WHERE: $T_I = \sqrt{\dfrac{d}{k \ln m}}$

EPRI - Sponsored Correlation of Age-Sensitivity and Seismic Qualification

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Wyle Laboratories

INTRODUCTION

The Electric Power Research Institute (EPRI) has sponsored research by Wyle Laboratories into the correlation between age-sensitivity and the seismic performance of safety-related equipment. The purpose of the research is to analytically and experimentally evaluate the correlation between the effects of thermal aging and operational cycling on the ability of selected electronic and electrical components to perform in a seismic environment. The results of this research provide the technical justification for exempting from aging, prior to seismic testing those component types for which it has been demonstrated that no significant aging and seismic coupling exists.

This paper discusses the aging mechanisms which were simulated, the components which were tested, the seismic testing performed and the conclusions of this series of testing.

AGING MECHANISMS

The research focussed on safety-related equipment which is normally located in areas of nuclear power generating stations which would not experience adverse environmental conditions caused by Design Basis Accidents. The most common aging mechanisms for the equipment located in these benign areas are time-temperature effects and operational cycling effects. Therefore, the research simulated these aging mechanisms artifically on the selected components utilized.

TEST PROCEDURE

The basis research was designed to ascertain if a difference in the seismic performance exists for aged and unaged safety-related components.

Therefore, components were selected for testing which were known to be utilized in safety-related equipment located in mild environments of nuclear power plants. The majority of components were selected to test the hypothesis that no correlation exists. The components selected for this hypothesis were resistors (wire-wound, carbon-composition and metal film), diodes, integrated circuits, transistors, optical couplers, capacitors (tantalum and ceramic), terminal blocks, P.C. boards, and sockets (transistor and integrated circuit). An additional group was selected to test the hypothesis that correlation exists. The items in this group were relays.

The program consisted of subjecting assemblies of the components, mounted in a cabinet to a severe seismic test series. The seismic test levels were purposely selected as representing a worst case condition. Some of the components had been preconditioned prior to seismic testing. Therefore, identical samples of the components were unaged, thermally aged, cyclically aged and thermally/cyclically aged.

The total number of components tested was 1944 and consisted of 60 different manufacturers. For those samples which were aged, the aging represented the equivalent of a nominal 50 years.

TEST RESULTS

For those components, for which it was predicted that there would be no correlation, there was no difference noted in the seismic performance. All samples in all of the aged and unaged categories were tested to the worst case seismic response spectrum without malfunction.

For the relays, which were chosen as a counter example to test the hypothesis that all relays would have a correlation between aging and seismic performance, a difference was noted in only one (1) of five (5) types of relays. One relay experienced contact chatter, which in some instances, constitutes failure of a relay. The relay had been thermally and cyclically aged. Another identical relay from the same manufacturer, which also had been thermally and cyclically aged did not experience any malfunction.

Because of these results, the hypothesis of no correlation was accepted for those items hypothesised to have no difference in seismic performance. For the relays, the hypothesis that aged and unaged relays have a difference in seismic performance was rejected.

Future work in this research investigates additional items with emphasis on identification of additional components which do not have a correlation between aging and seismic performance as well as identification of those components with significant correlation.

The Case for Non-Material Specific Thermal Aging

by
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ABSTRACT

The state-of-the-art model for accelerated thermal aging of components prior to seismic testing is the Arrhenius Model. The most pertinent independent variable in the equation is the minimum activation energy constant characterizing the component aging. With minor exceptions, existing measured values of the activation energy constant are inadequate as input to the model where a material specific aging acceleration factor is to be determined, for reasons described. The model itself is not very accurate. A case is made for a statistically justified minimum activation energy constant which is not material specific. The advantages of this are assessed. The major advantage is that this would provide the industry with a practical and uniform aging method that is consistent with the accuracy of the model.

INTRODUCTION

There is currently no deterministic way of extrapolating the effects of thermal aging at short periods of time in the laboratory to those which would have occurred in a safety related component after a service life in a nuclear plant which may be as long as 40 years. The state-of-the-art model for thermal aging is the Arrhenius model. It is widely used because it has been found "acceptable" by the NRC (Section 4. (4) Reference 1) and because it is the only model of broad application available. Since this model has represented the state-of-the-art for about 100 years now without any major advances in aging technology, it can be assumed that despite its significant defects it will remain the state-of-the-art for some time to come. Suppliers of safety related components to the nuclear power industry are constrained to qualify their products which usually necessitates type testing including thermal aging (Section 6.3.3, Reference 2; Section 4, Reference 1). A recent comprehensive review of equipment aging theory and technology (Reference 3) on the other hand, concluded that "the dominant picture that results from the study is that there is no comprehensive scientifically rigorous solution to the problem of accelerating the aging of equipment. Aging that can be accelerated in ways that yield verifiable correlation between real and simulated aging is an exception rather than the rule."

The problem that suppliers face is, to qualify components now, as required by existing regulations, in an atmosphere where the proper method of qualification is uncertain. Clearly, suppliers need a well defined and uniform "best method" of qualification immediately, while research is sponsored to advance the state-of-the-art of thermal aging if possible. Bearing in mind that the "best method" cannot hope to be deterministic at this time, it should seek to incorporate as much of what is known as possible without putting too much emphasis on areas which may give only the illusion of determinism. In particular, the current practice of attempting to associate an activation energy constant with each constituent material in a component to be qualified, through the performance of a lengthy and expensive so-called "aging analysis", is unwarranted relative to any expected increased accuracy in predictions using the Arrhenius model. Instead it would be beneficial to the industry to adopt a minimum activation energy constant which is not material specific and therefore can be uniformly applied to all components without regard to material constituency. As discussed below, it appears a reasonably good statistical basis can be found for such a minimum activation energy constant and that its major drawback would probably be conservatism, i.e., too long aging of a component in the laboratory to simulate the equivalent 40 year service life. Since it has currently been demonstrated by others (see Proceedings of this workshop) that for a wide variety of components only rarely do thermal aging failures occur or in any way effect the outcome of an overall qualification test program, it seems likely that some excessive conservatism in aging would be acceptable to the industry in exchange for a consistent method of qualification.

Based on our laboratory experience through the years and a review of existing aging data, what is described below is a case for a practical method of thermal aging for current use. This method does not preclude the possibility for refinement or exception as the state-of-the-art thermal aging is advanced. Relative to the current state-of-the-art however, it is as adequate and certainly more practical than some other methods in current use.

DEFINITIONS

T_S - Service life temperature (average over service life), °K.

T_A - Aging temperature, °K.

E - Activation energy constant, eV.

CURRENT THERMAL AGING METHODS IN USE

Until recent years a practical model for thermal aging was the so-called 10°C Rule (page 952, Reference 4) which is equivalent to the application of the Arrhenius model where $E/(T_A x T_S) = 6 \times 10^{-6}$ eV/°K^2. Use of this model has fallen into disfavor because it primarily applies to the reactions of salts in solution and therefore did not seem to apply to elastopolymeric materials, which are the primary constituents of the type of components qualified for nuclear service. This method had the advantage, however, of not being material specific in its application. Furthermore, as $T_A \times T_S$ decreases, the E of the equivalent Arrhenius model also decreases, thus the 10°C Rule tends to greater conservativism relative to the Arrhenius model as normally applied for the lower range of aging and service life temperatures.

Another method of aging may be called the statistical failure method. Using this method large numbers of the actual components to be qualified are aged at different temperatures for various lengths of time until equivalent age related failures occur. A lifeline plot as a function of time and temperature can then be generated and extrapolated to the end of service lifetime (usually 40 years). If the lifeline temperature at the end of service life is greater than T_S, survival of the component under in-service aging conditions has been established. Usually this method is impractical however, because of the long times associated with aging components to failure and because of the economic aspects of aging large numbers of components.

With minor exceptions the aging method used today results from application of the Arrhenius model and an "aging analysis". This latter consists of the following:

1. Each component or subcomponent of a system to be qualified is analyzed for identification of all of its constituent materials.

2. From existing data on material specific activation energy constants, activation energy constants are associated with each of the constituent materials.

3. The minimum activation energy constant in the list is used in the model to predict the appropriate time-temperature relationship for aging in the laboratory equivalent to the in-service conditions specified; aging of test items is performed accordingly.

THE PROBLEMS WITH THE CURRENT METHOD OF THERMAL AGING

Without belaboring the problems with application of the Arrhenius model itself for thermal aging, there are certain other problems with the current method of thermal aging which should be addressed. In particular, those associated with the choice of a material specific value of E. Some of the assumptions behind this choice are as follows:

1. E can be accurately measured for a specific material.

2. All component constituent materials are identifiable.

3. E is related to a component specific failure mode.

4. E measured on one material is transferable to other materials of the same name.

5. The measured E is a minimum for the material.

6. E, usually measured at high temperature, is appropriate for T_S.

The first three of these assumptions are particularly a problem. Figure 1 is an example of data obtained from the literature for the E of a material used for a relatively simple function (wiring insulation, page 173, Figure 8A, Reference 5). This data was obtained by the conventional method of monitoring a functional parameter while aging the material samples at various temperatures. The E value for this material was obtained from the slope of a best fit regression line (solid line in Figure 1). I have added the dashed lines which indicate the spread in values of slope which could have been obtained based upon the spread in the data. Note that E values between .5 and 1.39 eV for this material could have been obtained from this graph. Figure 2 shows data obtained on another insulation material using thermal gravitational analysis (Page 3, Figure 1, Reference 6). The dashed lines (my analysis) show that similar to conventional means of measurement, the TGA measurement also produces a wide spread in values for E. Figure 3 (Page 3, Figure 2, Reference 6) is the distribution as given in the reference itself. It is not surprisingly Gaussian in form. To the extent this data typifies the measurement of material specific E values, it can be seen that these values are based on a statistical or regression line fitting to measured values with a considerable spread in distribution. Since E is in the exponent of the Arrhenius equation, the spread in distribution of time-temperature relationships predicted by the model using these material specific E values can be expected to be even greater.

A second problem with the Arrhenius methodology as currently applied is the assumption that all constituent materials of a component are identifiable. This is hardly the case for complex electrical components containing circuit boards with many integrated circuits and other small subcomponents. In general, suppliers buy subcomponents in lots on the basis of function not material constituency. Thus two different circuit boards with the same function may have subcomponents supplied by different manufacturers and of different material constituency. Since it is impossible to directly measure E values for all materials in a given component to be tested, E values must be taken from the literature on the assumption that the material from which they were measured is identical to the material identified in the component. Chemical identity of the materials is not likely because manufacturing materials are not chemically pure to begin with. Chemical identity is, in fact, assumed however when a material specific E is an input to the Arrhenius model.

Perhaps the biggest problem with use of material specific E is the assumption that data on measured values as obtained from the literature is pertinent to the specific failure modes of the component to be tested. The material specific E measured by detecting percent loss in elongation, for example, does not necessarily apply if the material is used for insulation in the component and the relevant failure mode is loss in bulk resistivity. Thus, it really isn't enough to obtain material specific E values from the literature and relate them to constituent materials of the component to be tested. In addition, the property of the material monitored while obtaining the material specific E value must be associated with the relevant failure mode of the component to be tested. In present application of the Arrhenius methodology this is almost never the case because insufficient data exist for material specific E values.

Because of material homogeneity and the differences between the "same material" from lot to lot, measured material specific E values from the literature are not really directly transferable to materials in the component to be tested. Furthermore, the minimum E obtained from a list based on material constituency of the component is not really a minimum (i.e., conservative) because as we have previously seen, it is the result of a regression curve fitting and thus an average value, not a minimum. It would not be possible to obtain the relevant minimum E for a component in any case unless all potential failure modes of the component were known and could be associated with specific materials which are constituents of the component. Finally, it is not clear that measured material specific E values obtained from the literature are applicable for the range of T_S for normal service conditions. This is because the measured values must usually be obtained from tests at much higher temperatures than would normally occur in a nuclear plant, in order for testing to failure time to be minimized. E is related to specific chemical reactions within the material. The reactions dominant at the normal service temperatures of a nuclear plant in general may not be those dominant at higher tamperatures where the E measured values were obtained.

THERMAL AGING USING A NON-MATERIAL SPECIFIC E

Figure 4 shows the frequency distribution of activation energy constants for a large sample of data for many common component constituent materials (Appendix B, Reference 3). This distribution approximates a Gaussian (random) distribution with a mean of 1.11 eV and a standard deviation of 0.48 eV (see Table 1). On the basis of this data, a reasonable assumption for a minimum activation energy constant that is not material specific might be the mean minus one standard deviation or 0.63 eV. If this value is selected, 87% of all measured activation energy constants will fall above this value assuming the world population of E values is random and that our sample of 207 points is adequate to describe the world population. This does not seem unrealistic since the measured values are for a wide variety of materials and were obtained by many different investigators using different material failure mode criteria.

THERMAL AGING USING A NON-MATERIAL SPECIFIC MINIMUM ACTIVATION ENERGY CONSTANT

The arbitrary selection of what appears to be a reasonably conservative activation energy constant (0.63 eV) based on the statistical distribution of existing measured activation energy constant data is certainly not deterministic, nor does it remove the possibility of a large error in the calculated aging time-temperature relation where the Arrhenius equation is used as the model. This is because the Arrhenius model has significant limitations itself. However, use of material specific activation energy constants guarantee no greater accuracy even for the aging of components of relatively simple material constituency and function such as wiring, as is shown by the insulation data already discussed. On the other hand, use of a minimum activation energy constant that is not material specific has the following advantages.

1. The same aging procedures can now be used uniformly for all components to be aged.

2. "Aging analysis" of marginal value will not be needed. In particular, the problems of identifying component material constituency and association of failure modes with method of measurement of the material specific activation energy constant are eliminated.

3. The accuracy of the minimum activation energy constant used is consistent with the accuracy obtained in application of the Arrhenius model.

4. Selection of the minimum E is on an expressed statistical basis. This is consistent with the measured material specific values of E which are statistical in nature (even if only implied) since they are based on regression analysis.

5. Some confidence limits or statistical justification of the minimum activation energy constant can be established which is at least verifiable.

A potential drawback of selecting an activation energy constant so low as 0.63 eV as a non-material specific minimum value is the fact that this tends to rather conservative aging time-temperature relationships compared to those currently used for many components. For example, a component with a T_S equal to 40°C and a T_A equal to 125°C would have to be aged for approximately 100 days to obtain the equivalent 40-year service life aging. If this component was a terminal block whose only ageable constituent material was phenolic, a material specific activation energy constant as high as 0.96 eV might be justifiable, which would give an aging time of approximately 7.3 days for the same T_S and T_A. Since current work shows that thermal aging failures are rare even for very conservative assumptions, (see Westinghouse presentation at this workshop in which E = 0.5 eV was assumed) this tendency to conservative aging does not seem to be a prohibitive disadvantage.

CONCLUSIONS

There is a need for a practical and uniformly applied thermal aging methodology that is consistent with the accuracy inherent in applying the Arrhenius model. Use of this model with a non-material specific activation energy constant based on a review of published and well-documented activation energy constant data, and a statistical analysis, has been shown to be a useful way. To the extent that the minimum activation energy constant selected (such as 0.63 eV as shown in this paper) is arbitrary, it should be pointed out that widely used environmental testing of a similar nature, notably much of the military specification testing, is based on equally arbitrary assumptions. In this case, at least, a review of existing knowledge has been made and the method proposed is as accurate as the state-of-the-art. Since data from the type of materials that are actually the constituent materials of many components being qualified is considered, this method is an improvement over the previous application of the 10°C Rule. The method also employs the Arrhenius model which has current NRC acceptance.

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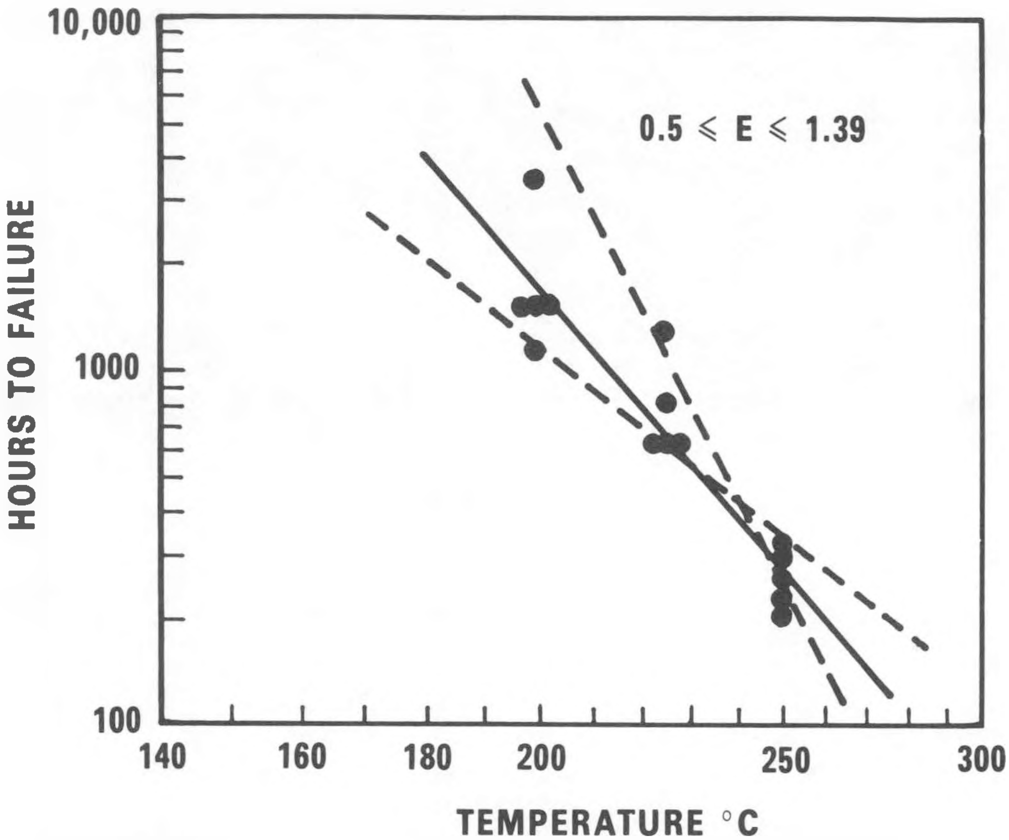

FIGURE 1. CONVENTIONAL DATA ON CLASS H INSULATION LIFE

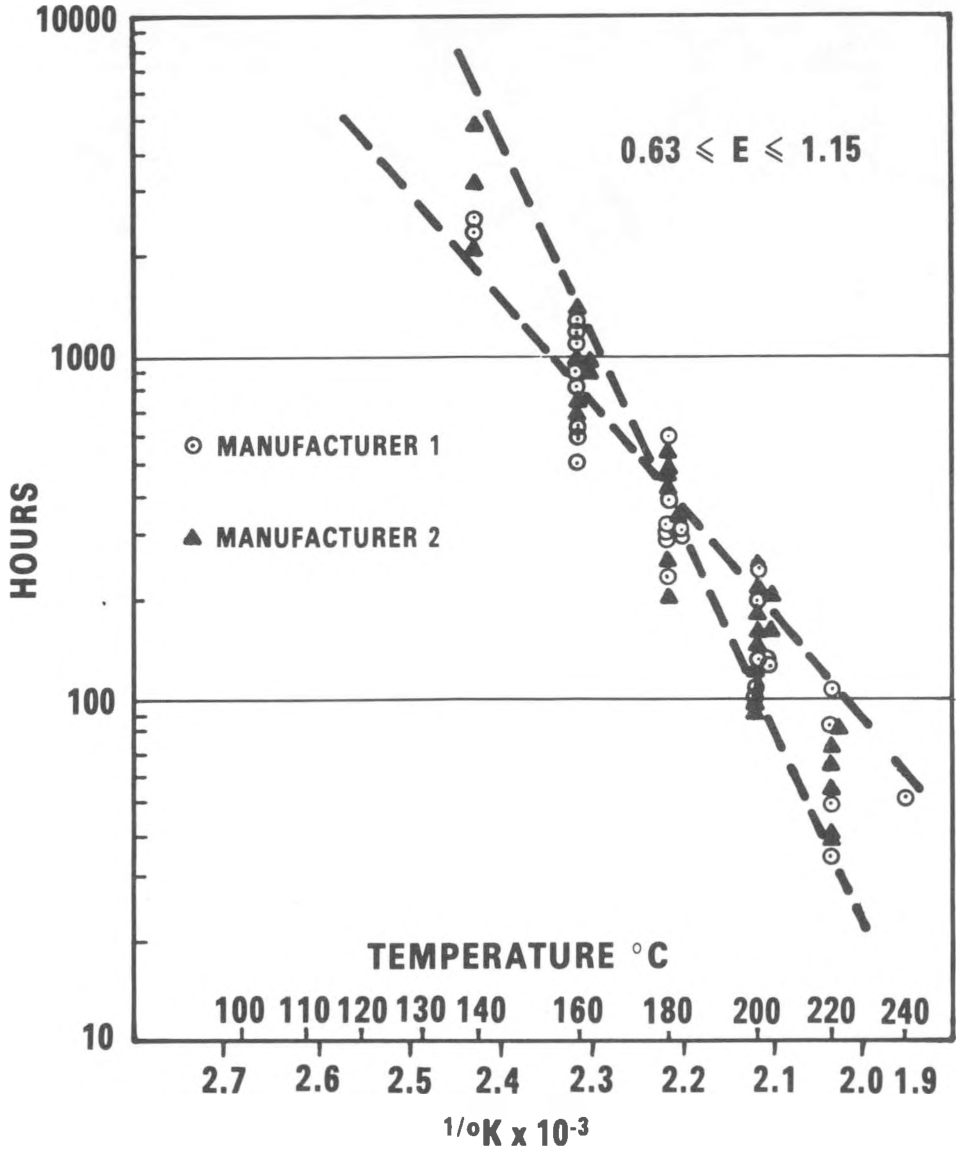

FIGURE 2. TGA DATA ON FORMVAR INSULATION LIFE

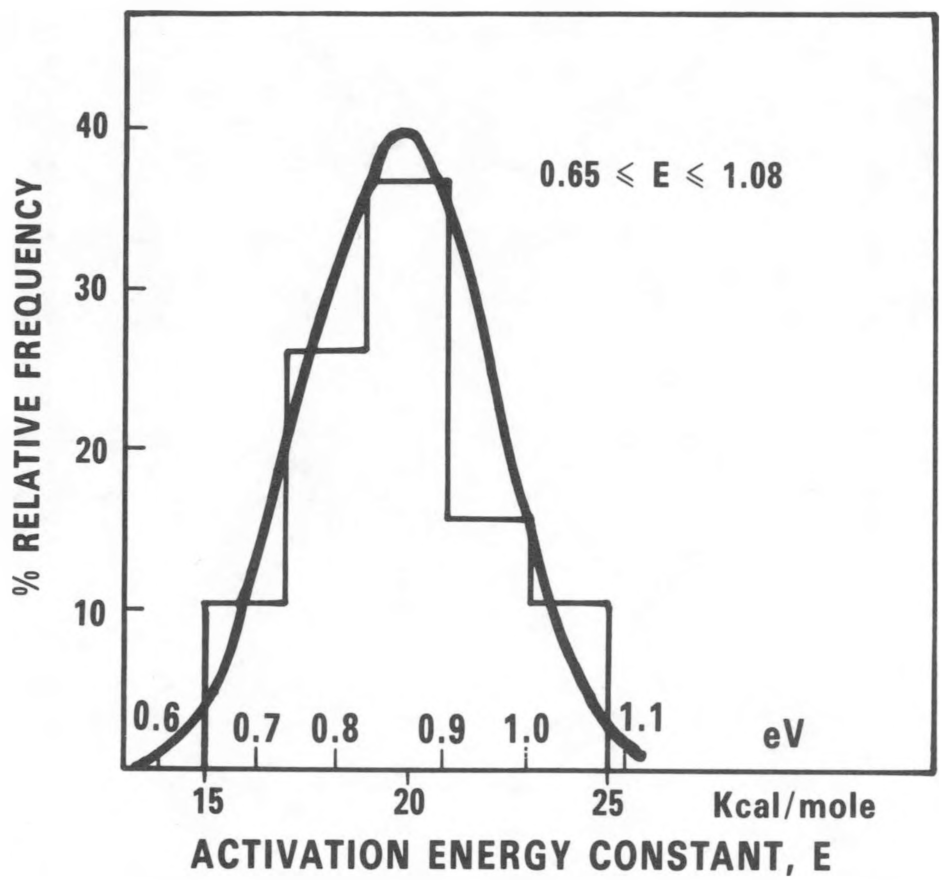

FIGURE 3. TGA DATA ON FORMVAR, DISTRIBUTION IN SLOPE

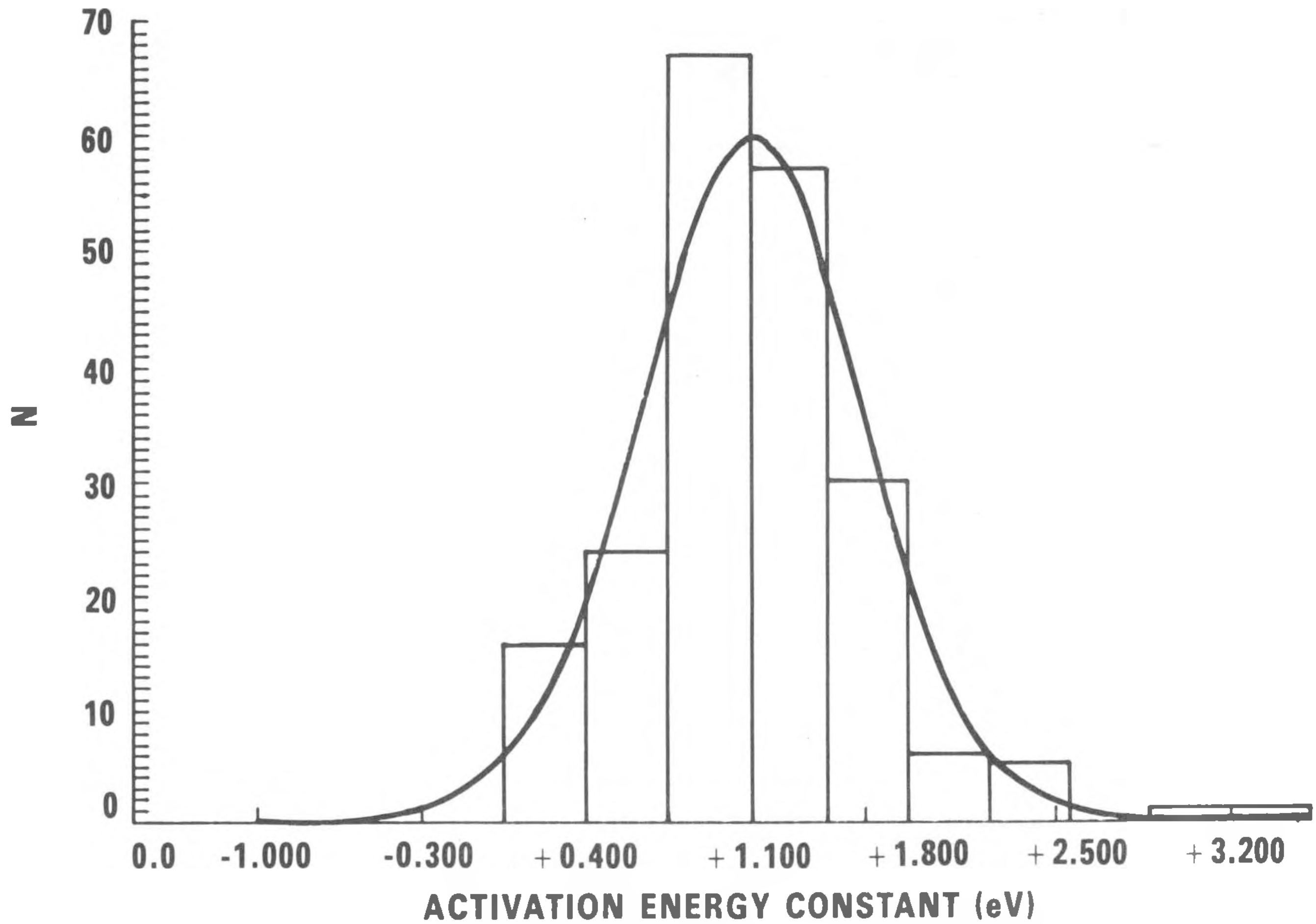

FIGURE 4. DISTRIBUTION OF ACTIVATION ENERGY CONSTANTS OBTAINED FROM THE LITERATURE

TABLE 1. DESCRIPTIVE STATISTICS OF ACTIVATION ENERGY DATA DISTRIBUTION

N =	207
MEAN =	1.110 eV
VARIANCE =	0.2320 eV
STD DEV =	0.4817 eV
DATA MIN =	0.09 eV
DATA MAX =	3.29 eV
DATA RANGE =	3.2 eV
STANDARD ERR OF MEAN =	0.0335 eV
COEFFICIENT OF VARIATION =	43.38
SKEWNESS =	0.9445
KURTOSIS =	5.862

TREND EVALUATION

by

Harold Cobb
HAFA International

(Abstracted from the Workshop Transcript)

There is a need for information on the previous performance or history of components. Measurements of component parameters through numerous cycles can provide data for trend evaluations. The information obtained can provide the rationale for code relief request and guidance on the need for in-service inspection and testing.

EXAMINATION OF BOLTING MATERIAL DEGRADATION IN PRESSURIZED WATER REACTORS*

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ABSTRACT (as submitted by authors)

An increasing amount of nuclear industry reports have indicated corrosion of ferritic and austenitic steel components by various mechanisms. Ferritic bolting materials have suffered failures induced by a boric acid corrosion as well as a stress corrosion cracking mechanism. Austenitic stainless-steel bolts have failed transgranularly during service in reactor coolant pumps. Investigation of these failures performed by Brookhaven National Laboratory have indicated that chloride is responsible for the failures of the austenitic stainless material and that the nonjudicious use of lubricants is the probable cause of the cracking of ferritic alloy bolts. Additionally, corrosion of ferritic steel bolts by leaking boric acid solutions from PWR primary coolant will be discussed.

* Research carried out under the auspices of the U.S. Nuclear Regulatory Commission.

CONSIDERATIONS IN THE THERMAL AGING ANALYSIS OF SAFETY-RELATED POWER EQUIPMENT

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ABSTRACT

There are several approaches to aging analysis in the qualification of nuclear power plant equipment in accord with IEEE and NRC standards and regulations. The importance of analysis as a technique in the aging step of nuclear environmental qualification results from the fact that industry is frequently faced with limited time to qualify a nuclear component or system, inadequate budgeted funding to perform an elaborate test program, and equipment that is already in place and in service that is required to meet new regulatory requirements. Full utilization of the analysis approaches to aging will assist the nuclear power industry in solving many of its problems in meeting these new requirements.

1. INTRODUCTION

The accepted model for accelerated thermal aging testing is the Arrhenius model. It is important to review the limitations of the Arrhenius model in order to modify the model to more accurately describe the thermal aging process. Although a rationale is presented to include more phenomenology, there is still doubt that thermal aging tests always cause only deleterious effects. This paper discusses these aspects of accelerated aging tests.

2. ACCELERATED THERMAL AGING

In developing the qualified life evaluation for an item of electrical equipment which has a safety-related function in a nuclear reactor, the thermal aging environment is considered by means of the listed Arrhenius activation energy (E_A) for each of the materials comprising the equipment. A test data point (temperature and duration) from an oven test, the selection of a value of E_A that represents the item of

*The authors are indebted to Dr. Alex Samsonov of TRW and Mr. Roc Fleishman of National Technical Systems for their contributions and helpful comments.

equipment, and the expected service temperature are used to develop the qualified life. This paper discusses the considerations that must be included during the selection of the E_A value.

It is vital to stay in touch with the original requirement for accelerated thermal aging, which is to simulate all the degradation mechanisms critical to the functioning of the equipment to be qualified for a particular service temperature. Care must be given that the degradation mechanisms at the service temperature are indeed simulated at the higher temperature of the oven tests.

Not all degradation mechanisms are of interest. Only those degradation mechanisms associated with a critical function of the equipment need be considered. Furthermore, the presence of a degradation mechanism and a weak material in a critical location still does not include whether the material is under severe or under mild stress.

The specific application is also vital. Consider the case of material being used in a vital function which indeed is weakened by a severe stress, and it becomes brittle. If the equipment experiences only a static load, the embrittlement may not be vital. But this same loss of physical property would be critical for dynamic application where vibration and shock must be withstood.

These preliminary comments indicate the need to stay in touch with the relevant phenomenology during the course of applying the Arrhenius activation energy model.

3. ARRHENIUS MODEL

The Arrhenius model indicates that a material undergoes chemical reaction at a rate (r) given by:

Absolute Temperature, T
Activation energy, E_A
Boltzmann constant, k

where the rate is described by the relation,

$$r \sim \exp\left(-\frac{E_A}{kT}\right)$$

Thus the reaction rate r, at temperature T, is related to the reaction rate r_2 at temperature T_2 by the relation

$$\frac{r_1}{r_2} = \exp\left(\frac{E_A}{kT_1} - \frac{E_A}{kT_2}\right)$$

Thus, a reaction rate is accelerated by the use of high temperatures. The chemical reaction occurring over a long time t_1 and low temperature T_1, will be identical to the chemical reaction occurring over a short time t_2 and high temperature T_2, provided

$$\frac{t_1}{t_2} = \exp\left(\frac{E_A}{kT_1} - \frac{E_A}{kT_2}\right)$$

The above expression has been empirically demonstrated in oven tests of coupon size materials, where specific measure of the material degradation was used. For example, tensile elongation at a fixed stress of 25 percent, or of 50 percent, has been used as measures of the material degradation.

The activation energy derived from fundamental tests of coupon size specimens for each non-metallic material is listed in the open literature. EPRI has made a list, and several testing companies have created their own data bank. Subtle differences in the value of an activation energy from different test agencies results in strongly different projected lifetimes for the same material. This is the result of the exponential relation in the Arrhenius equation. Figure 1 shows that altering E_A from 1.0 to 1.1 eV (10 percent) results in a qualified life change from 2.3 to 4.9 x 10^5 hours (>110 percent).

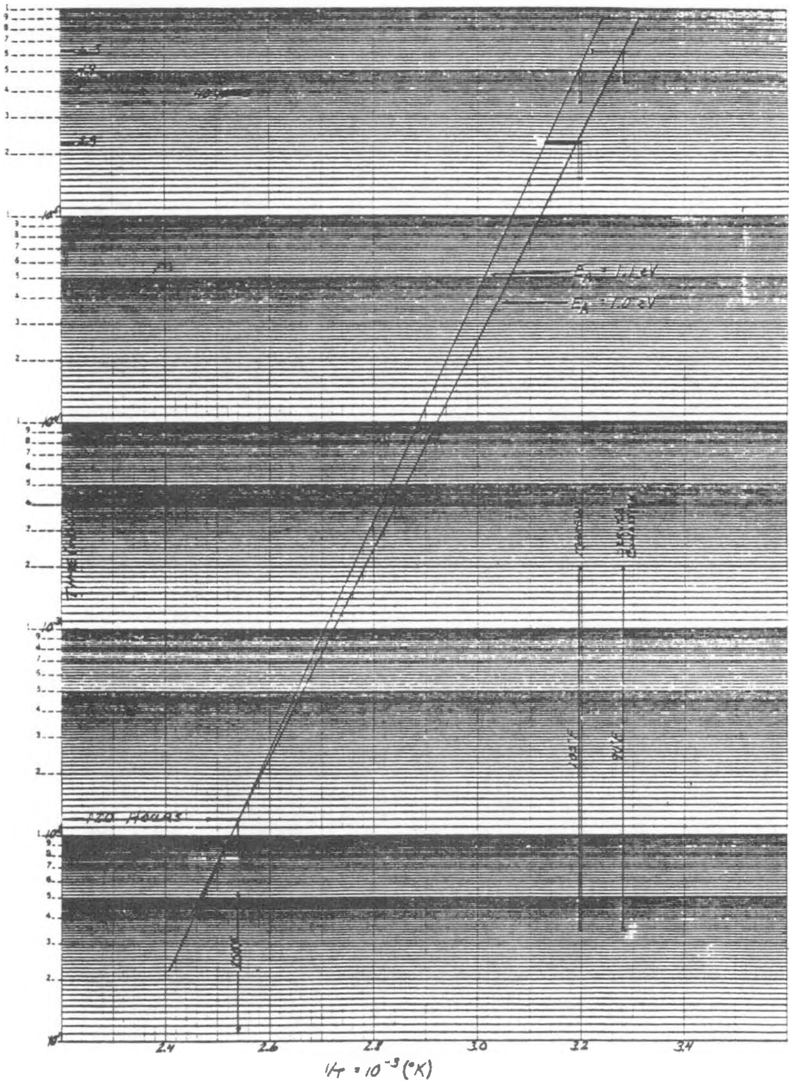

Figure 1. Sensitivity of Arrhenius Model to Small Changes in E_A

A small error band in the measurement of time to achieve a specified level of strength degradation in an oven test at high temperature T_1, results in an extremely large error band at long time at lower temperature T_0, as shown in Figure 2. This occurs because of the same exponential relationship dictated by the Arrhenius model. However, Figure 2 indicates that some of this error can be reduced by scheduled inspection and recalculation of predicted lifetimes of equipment after the inspection.

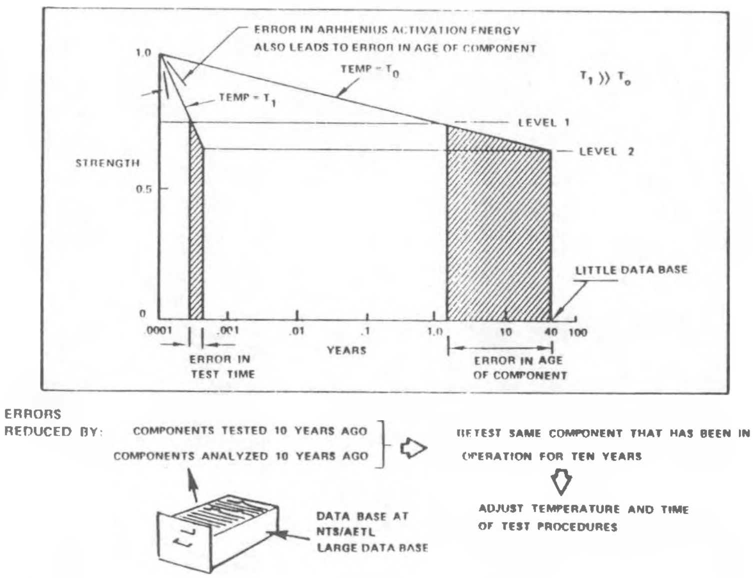

Figure 2. Critique Test Specification

4. ARRHENIUS MODEL LIMITATIONS

The activation energy for a particular material is derived for a small specimen of the material, with air circulating through the oven and around the material. This environment has the entire specimen at a uniform temperature. Also, typical tests are in the order of about 120 hours.

The overall chemical reaction rate for the small test sample is also dictated by the surface area of the sample. For small test specimens, the surface to volume ratio is extremely large compared to large specimens. Furthermore, the surface exposure of the full scale specimen will be more complex.

In the full scale item to be aged thermally, these boundary conditions are altered. A key material within the test item may be completely contained between other solid materials so that oxygen and humidity cannot interact with the key material in the same way as in the test oven. Even under boundary conditions where there are free surfaces, the test item may outgas other chemicals from the diverse materials in the test item which interact with the free surface of the key material; whereas in the test oven containing a coupon size sample of only the key material, only the effects of the oxygen and humidity present in the oven will be experienced.

The effects of temperature only on degradation of material properties are well known. But the effect of temperature and humidity are known to be even more dramatic (Figure 3). Thus, it is vital to consider the special effects of tests in an oven where the humidity will have effects at the higher temperature, compared to the same humidity at room temperature where even condensation droplets can form and induce special damage effects. For printed circuit boards, the impact of voltage and impurities need also be considered.

Equipment Type	Percentage of Failures classified by environment						
	Temperature and humidity	Dust	Humidity	Radiation	Salt spray	High temperature	Low temperature
Electronic and electrical equipment	8.2	1.4	4.8	0.5		6.7	6.7
Lubricants, fuels, and hydraulic fluids		0.5				3.8	1.9
Metals	4.8		4.3		12.5	3.8	2.4
Optical instruments and photo equipment	2.4	0.5	1.4				
Packaging and storage	4.3		4.3				
Textiles and cordage	6.7		2.4	5.3		1.4	
Wood and paper	5.8	0.5	1.9			1.0	
TOTAL	32.2	2.9	19.1	5.8	12.5	16.7	11.0

Figure 3. Distribution of Field Failures

There are a number of reactions that do not scale according to the Arrhenius expression. For instance, the diffusion of oxygen into a material over the course of several years cannot be accelerated and still realistically describe the material degradation. There are a number of effects that do not scale just because of their size. A coupon size material may achieve a uniform temperature distribution, whereas a large piece of varied thickness material pinned at several points and sandwiched between other kinds of materials, will experience thermal gradients and stress gradients that are more severe.

There are a number of transient effects which do not scale well. When power is turned on or turned off, the power transient causes special thermal effects at junctions. These are scaled differently than what is indicated by the Arrhenius equation.

There are special end effects that are critical. The overall coaxial cable may well be able to have a long qualified life. However, it is important to place the end-coupling of the cable in the test oven in order to properly rate the cable. The degradation at the end-coupling may prove to be the critical degradation mechanism.

Thus it is recognized that although it is economical to run a thermal aging test in an extremely short time, that this economy is not always achievable when one is required to simulate degradation mechanisms. Figure 4 shows examples of temperatures that should not be exceeded for specific materials. These limitations must be kept in mind in order to properly use the Arrhenius equation in accelerated thermal aging tests.

Type of rubber	Highest usable temperature	
	°C	°F
Silicone	260	500
Polyacrylic	177	350
Buna-N	171	340
Neoprene	157	315
Butyl	149	300
Buna-S	138	200
Natural	127	260
Thiokol	121	250

Figure 4. Degradation of Rubber of High Temperatures

5. ACTIVATION ENERGY DEVIATIONS

The activation energy for a particular material will be different each time the same test is performed. This is the normal spread of values associated with test data. There is also the difference in mean value of activation energies reported by one test agency as compared to another. These differences can be used to give an extremely wide spread in qualified life for the same thermal test data.

There is yet another source of error. In the case of integrated circuits, the accelerated thermal tests have shown other problems in extrapolating thermal test data. At temperature T_1, failure mechanism I may be excited among the population of integrated circuits (Figure 5). At temperature T_2, failure mechanisms, I, II and III may be excited, where failure mechanism I dominates. By increasing the temperature further and studying the failed integrated circuits under a microscope, other failure mechanisms will have been detected and earlier failure mechanisms will have disappeared. These observations of failure mechanisms as a function of temperature are described by the Arrhenius activation energy being a function of temperature. This is a serious consideration when one requires the extrapolation of test data recorded at high temperature to predict performance at the lower service temperatures.

6. ADJUSTED ACTIVATION ENERGIES

The earlier discussion showed some of the limitations of the Arrhenius model of thermal aging. To improve the model it is necessary to modify the Arrhenius activation energy by such factors as:

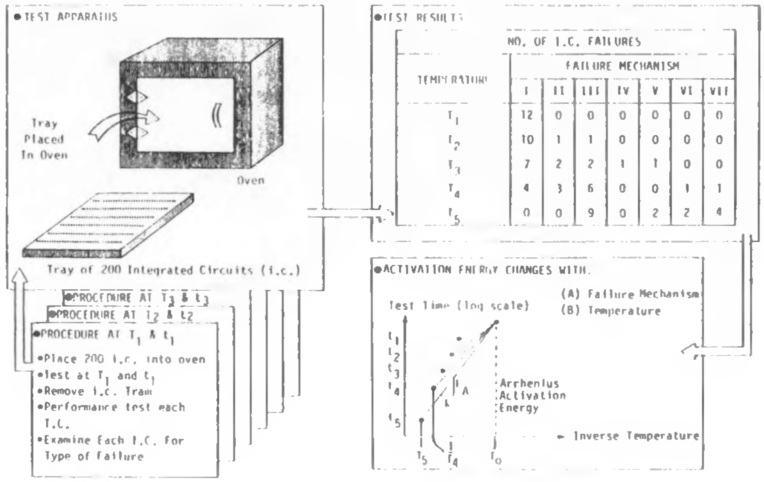

Figure 5. Arrhenius Model Can Be Temperature Dependent

- Stress level of the material compared to its allowable stress level (ψ)
- Criticality of the component to the equipment operation ($\emptyset$)
- The number of times the component appears in the equipment circuitry, assuming a series connectivity (N)

These adjustments will not compensate for all the Arrhenius model limitations, but attempt to introduce more realism.

There are two areas where the E_A adjustments are of interest. First, there is the decision of whether or not an item of test equipment should be tested for qualification. Second, there is the need to predict the qualified life.

6.1 Qualification Testing

There are extreme cases where a material analysis shows that the activation energies are so high or so low that the decision as to whether or not to test is obvious. When the activation energies of all the materials in a device are so high that 40 years qualified life is definitely assured, the qualification can be made on the basis of analysis.

Where the activation energies of several of the materials are so low that only a few years of qualified life is assured, the short qualified life can be assigned along with the recommendation for early inspection with a view to replacement of specific parts.

For the intermediate cases a rationale for recommending or not recommending test can be developed as follows.

It is well known that o-rings made of low activation energy material can suffer great deformation and even have cuts through much of its cross-section, and still function in its role to prevent leakage. NASA tests have demonstrated just this phenomenon in its aging tests of o-rings. Thus an o-ring may have a low activation energy (E_A < 1.0 eV) material, but have a high safety margin (Ψ).

The product of these two effects, where $\Psi \gg 1$ results in an effective activation energy greater than unity, ("E_A" > 1 eV).

Should the particular o-rings be part of an extremely vital portion of the device, the criticality number ($\emptyset$) must be applied which reduces the effective activation energy. Thus the product of the low activation energy (E_A < 1.0 eV) and the low value of criticality number ($\emptyset$ < 1) results in an effective activation energy less than unity ("E_A < 1 eV).

In order to consider all the effects together where "E_A" = $\Psi \emptyset E_A$, a rationale must be developed to give particular values to Ψ and $\emptyset$. Figure 6 shows one approach.

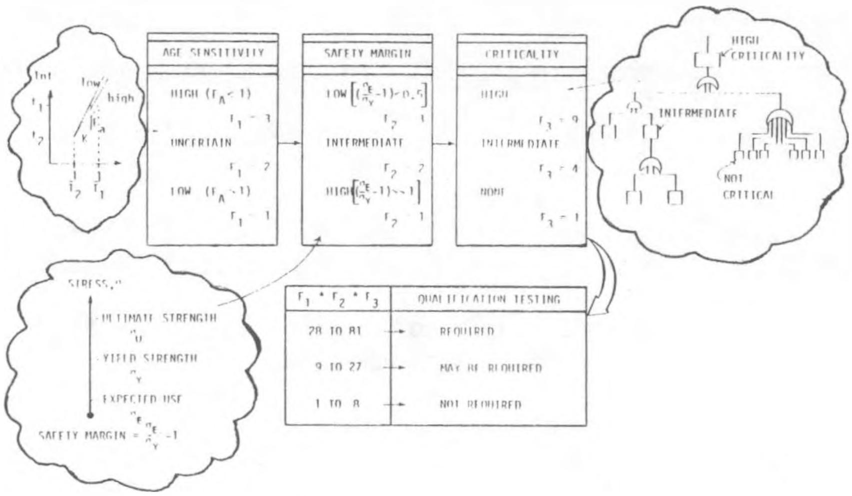

Figure 6. Assessment Methodology for Qualification Testing

The values of $(E_A)_i$ are obtained for the most susceptible materials in the device. Consideration is given to the several susceptible materials as to how severely they are each stressed $(\Psi)_i$. Consideration is also given to the several susceptible materials as to how critical they are to the operation of the device $(\emptyset)_i$. The products of $\Psi \emptyset E_A)_i$ are formed and the lowest product dictates the choice of test or no-test recommendation. Figure 6 suggests values to assign to Ψ and $\emptyset$.

The safety margin Ψ may be based on a mechanical stress (σ) or upon an electrical rating. One approach is to use $F_2 = 1$, $F_2 = 2$ and $F_3 = 3$ categories for values of Ψ.

The criticality $\emptyset$ may be obtained from a fault tree analysis as depicted in Figure 6. A component that has great redundancy can be assigned $F_3 = 1$, whereas a component that is situated at a key location can be assigned $F_3 = 3$. For the intermediate case $F_3 = \Psi$. These values of F_3, which are categories of $\emptyset$, are made larger because more importance is assigned to $\emptyset$ than Ψ.

The use of a fault tree must also be used with discretion. Thermal aging is a common mode failure such that all components are uniformly degraded in this environment. Thus, not all the benefits are really available from redundancy. However, the fault tree does give a rationale for ranking the criticality of each component. But the ranking is artificial because of the common mode failure mechanism. Thus the use of the three values of F_3 is sufficiently coarse to include the limitations placed on the fault tree analysis.

To use Figure 6, take the example of an o-ring with high age sensitivity material ($F_1 = 3$), but with a high safety margin ($F_2 = 1$) and in a noncritical location ($F_3 = 1$). The product of $F_1 \cdot F_2 \cdot F_3 = 3 \cdot 1 \cdot 1 = 3$ yields a value between 1 and 8 so that testing is not required for that device. Actually, other products of $F_1 \cdot F_2 \cdot F_3$ must also be considered for other pieces of the part being considered for test. In all cases, $F_1 \cdot F_2 \cdot F_3 < 8$ must be shown to maintain the decision of no test.

6.2 Qualified Life Prediction

Given that a device was tested in an oven at the high temperature T_2 for the test duration t_2, it is necessary to use a value of activation energy to determine the qualified life t_1 at the service temperature T_1. Figure 7 depicts a procedure for selecting E_A. As in the above discussion, the concept of $\emptyset$ and Ψ are introduced. Furthermore, the concept that there are a number of elements in series (N) need to be considered.

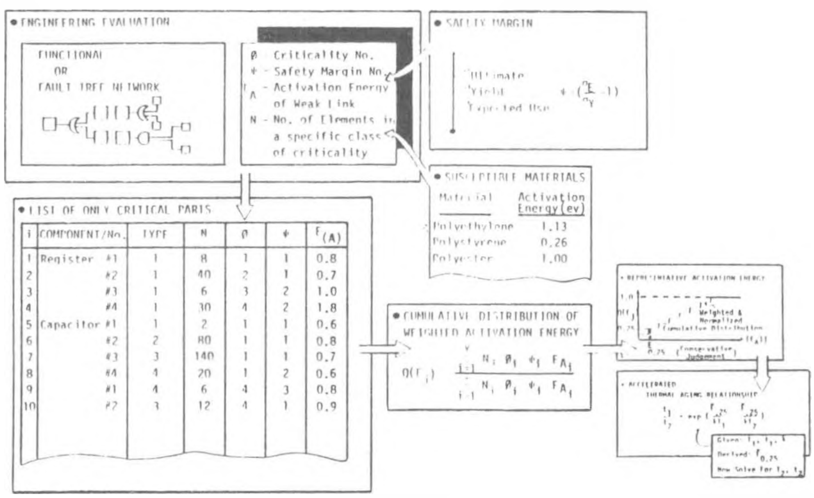

Figure 7. Device Thermal Aging (t_2 - T_2) Methodology Using a Weighted Activation Energy

If a device has N_1 parts of activation energy $(E_A)_1$ and has N_2 parts of activation energy $(E_A)_2$, where $N_1 > N_2$, this effect should be included in the overall estimate of the average $\bar{E}_A$. Suppose that $E_{A1} = 1.2$ eV and $E_{A2} = 1.0$ eV. The straightforward approach is to assign $\bar{E}_A = E_{A1} = 1.0$ eV, in order to be conservative as to the overall device behavior in thermal aging. However, we do anticipate that in actual devices that true answer for $\bar{E}_A$ lines between 1.0 and 1.2 eV, and probably closer to 1.2 eV because of dominance of N_2 over N_1. This is based on all the limitations presented earlier with regard to the Arrhenius model.

The value of E_A must also be modified by Ψ and $\emptyset$, as discussed previously. Thus the activation energy is represented by the product $N_i \Psi_i \emptyset_i E_{Ai}$. Figure 7 shows how a table is constructed where the products $N_i \Psi_i \emptyset_i E_{Ai}$ are formed. There are four categories of resistors that have been gathered together in this example based upon their actual versus allowed power propagation ($\emptyset$). Within each category thereafter, the highest value of critically Ψ and lowest value of activation energy E_A is selected to represent that resistor class. This procedure is followed for all the resistors, capacitors, etc. Ultimately, the cumulative distribution of weighted activation energies (Q) which is a function of activation energy (E_A),

$$Q(E_j) = \frac{\sum_{i=1}^{j} N_i \emptyset_i \Psi_i E_{Ai}}{\sum_{i=1}^{\infty} N_i \emptyset_i \Psi_i E_{Ai}}$$

The cumulative distribution of weighted activation energies includes all the separate effects of the number of each item, the margin of safety of each item (determines class), the highest criticality number of each item class, the lowest value of activation energy of each item class. It is then necessary to assign a criteria, say Q = 0.5, to obtain an average value E_j. To maintain a conservation approach, say Q = 0.25. This would lead to a value of $(E_A)_{Q=0.25}$, which would be used in the Arrhenius model, i.e.,

$$\frac{t_1}{t_2} = \exp\left[\frac{(E_A)_{Q=0.25}}{k}\left(\frac{1}{T_1} - \frac{1}{T_2}\right)\right]$$

This approach attempts to circumvent the usual practice of just using the lowest E_A value of a device because of the excessive conservatism. The proposed model uses more specific data concerning the actual design of the device and how it is used in order to safely reduce the excessive conservatism.

The more complex model is not designed for use when the use of the lowest value of E_A still delivers the 40 year qualified life overwhelmingly. But is is designed for use on a device which is assigned an extremely short qualified life by a low value of E_A, but where it is intuitively clear that the device has a much longer lifetime considering the specifics of its operation and intended application.

The motivation for reducing the conversatism in the use of the Arrhenius model is certainly made clear in the important test results of S. P. Carfagno, in his 1979 study of the effect of aging on the operation of switching devices. For most of the twenty-four devices he showed that the fragility level was approximately the same before and after aging. The accelerated aging included not only thermal aging at high relative humidity, but also gamma radiation and electrical/mechanical life cycling. His results questioned the need for the requirement that seismic qualification be conducted with aged specimens.

7. SUMMARY

The requirement for accelerated thermal aging is that the oven test duration and temperature simulate all the thermal degradation mechanisms present in the long duration (40 years) service temperature. Figure 8 shows the factors which cause the Arrhenius model to predict excessively short qualified life for equipment. The most important factor is the selection of the smallest value of activation energy among the list of activation energies of the thermally susceptible materials. The factors which predict excessively long qualified life for equipment are shown in a separate block, as well as the factors that could go in either the excessively conservative block or in the excessively non-conservative block. Figure 8 shows that E_A should be increased to more closely match test experience.

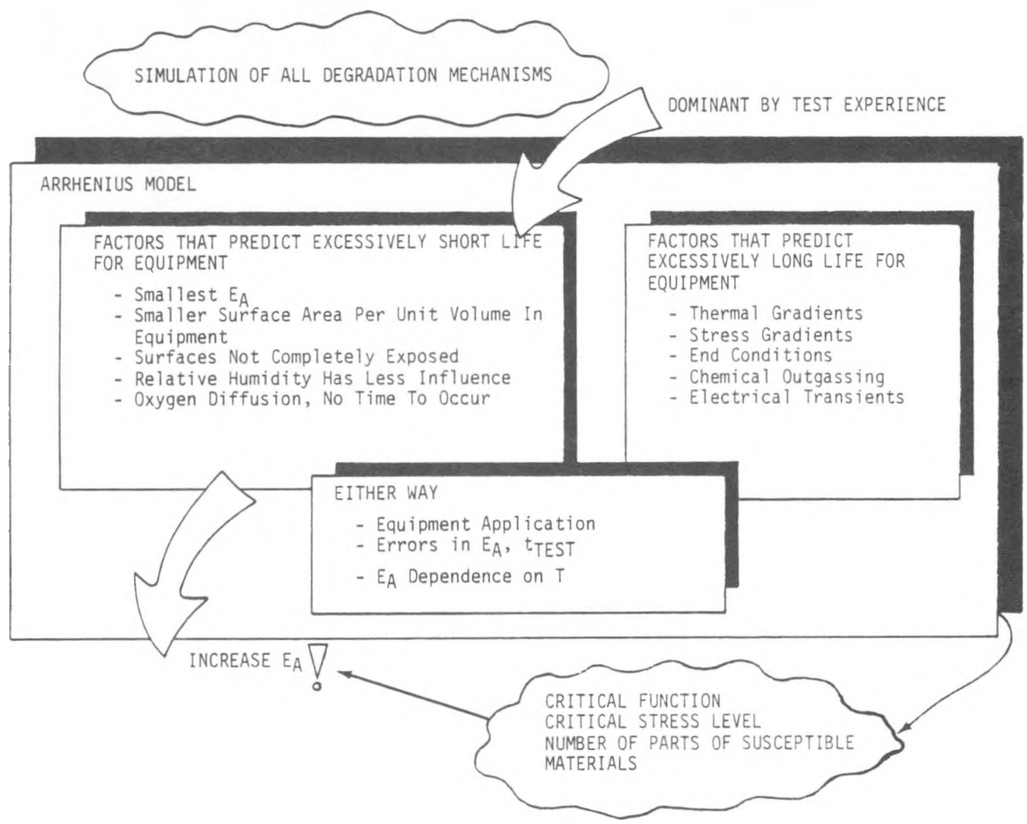

Figure 8. Accelerated Thermal Aging, Summary

This paper indicates an approach wherein

- critical function
- critical stress
- number of components that are in the critical function,

can all be combined to yield an average $\bar{E}_A$ that is not excessively conservative and is consistent with test experience.

The problem is always to predict the lifetime of a particular item of equipment based upon its performance in an accelerated thermal aging test. The item is typically composed of several different non-metallic materials, each having different activation energies. The goal is to select the proper activation energy that would realistically represent the "item" behavior. One approach is to list the components vital to the operation of the item and consider only those activation energies. To assure conservative predictions, the lowest activation energy in this group can be used to extrapolate the thermal age test data to late time (40 years) and service temperatures. Further conservatism is added by assuming that the service temperature will be a shade higher than planned so that predicted lifetime will be shorter.

The obvious case where there is, say, a small projection on a bobbin that is readily damaged, and is constructed of a susceptible material, is that this elaborate procedure should be bypassed in favor of the decision to replace the bobbin. Once the bobbin is replaced, the procedure may again become appropriate for the new piece of equipment with the redesigned bobbin.

INDICATION OF AGING FROM
IN-PLANT MAINTENANCE RECORDS*

by

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ABSTRACT

The In-Plant Reliability Data (IPRD) program[1,2] personnel have reviewed 120,000 plant maintenance records covering 24 unit-years of commercial operation. From these records, 24,000 corrective maintenance records have been extracted in hard-copy form. Of the extracted records, 4000 records of pump and pump-related corrective maintenance actions have been encoded and entered into the data base.[3] By the end of 1982, 5500 additional records related to valves will be encoded. These records include documentation of the corrective maintenance performed on components following catastrophic component failure as well as degraded and incipient failures. The IPRD system includes all components of a particular type, for example, both safety and non-safety related pumps. The completeness of this data provides substantial insight into the effects of actual plant operating environment. Some of these insights on aging effects may be relevant to the expected performance of aged components during an abnormal event. These effects may also impact the simulated conditions required to provide for adequate qualifications[4] testing. A sample of the preliminary indications which have been gleaned from the encoded records are discussed.

References

1. J. P. Drago, R. J. Borkowski, D. H. Pike, F. F. Goldberg - "The In-Plant Reliability Data Base for Nuclear Plant Components: Data Collection and Methodology Report," NUREG-CR-2641, July 1982.

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*Work sponsored by USNRC Office of Nuclear Regulatory Research, Division of Risk Analysis, under Dr. James W. Johnson, Project Manager. This work is being performed under an interagency agreement with Oak Ridge National Laboratory. Science Applications, Inc. is under contract with ORNL.

2. J. P. Drago and J. R. Fragola - "The In-Plant Reliability Data System - History, Status, and Future Effort," ANS International Meeting on Thermal Nuclear Reactor Safety, Chicago, IL, August 1982.

3. J. P. Drago, R. J. Borkowski, J. R. Fragola, J. W. Johnson - "The In-Plant Reliability Data Base for Nuclear Plant Components: Data Report - The Pump Component," NUREG/CR-2886 (to be published).

4. J. R. Fragola - "Standards Activities Related to Class 1E Equipment Qualification" Proceedings of ANS/ENS Meeting, August 1978, Brussels, Belgium.

STATISTICAL PERFORMANCE ANALYSIS OF SALT WATER COOLED PRESSURIZED WATER REACTORS*

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ABSTRACT

Pressurized water reactors (PWRs) using salt or brackish[1] cooling water are found to have significant long-term decreases in capacity factors and increases in forced outage rates (FORs) due to equipment failures and maintenance needs. Larger units have significantly lower capacity factors and higher FORs. The significance and magnitude of this correlation increases with unit age. Post-Three Mile Island effects are controlled for.

INTRODUCTION

Crucial decisions regarding the design, construction, and operation of nuclear power plants are being made today without the benefit of important information latent in accumulated operational experience. Today there are over 400 reactor-years of commercial operation, but no study has analyzed the long-term behavior of nuclear unit capacity factors,[2] the long-term stability of forced outage rates,[3] or squarely addressed the impact of aging on plant operation. And despite the growing technical literature of the special engineering problems of PWRs[4] and salt water cooled units,[5] there has been no systematic analysis of the impact of salt water cooling on PWR performance.

The NRC and the nuclear industry have reached a general consensus on the nature of the problem. Salt water is known to corrode many metals including those used in power plant condensers and service water systems[6] leading to the eventual infiltration of chlorides into the secondary cooling circuit of PWR units (and the primary circuit of BWRs). Reactor alloys, including Inconel 600, the material of choice for steam generator tubing, and Type 304 austenitic stainless steel, which is used extensively in pipes and components, have been found to be vulnerable to chloride induced stress corrosion cracking and other degenerative processes.[7] This syndrome is believed to be a principal cause of steam generator degradation in PWR units.[8]

It is hoped that the problem can be kept under control through the careful monitoring of condensers for leaks, the protection of damaged condenser tubes with titanium sleeves, the plugging and replacement of leaking tubes, and the construction of new units with titanium condenser tubing.[9] According to the NRC such measures assure that these problems "will not present an undue risk to the health and safety of the public."[10] Utility operators are similarly confident in the future productivity of such units.

These measures may not be adequate. As recently as 1975, chlorine levels of 75 parts per million were thought safe. But 50 parts per billion (ppb) chlorine in secondary circuit water are now thought sufficient to cause stress corrossion cracking in reactor environments. Further, as of 1979, detection equipment was unable "by a factor of ten" to detect concentrations at this low level.[11] Extensive laboratory tests have found that while titanium alloys demonstrate the best corrosion resistance of all alloys tested, galvanic corrosion could still result from seawater induced electro-potential differences between different condenser materials.[12] Titanium condensers may also be vulnerable to other problems, such as as significant wall thinning[13] and to catastrophic failure[14] even though they are highly corrosion resistant.

Chlorine injection into cooling water as a biocide is becoming more widely used. Biofouling of titanium condenser tubes accelerates over time and significantly reduces heat transfer.[15] Continuous chlorination is a favored method of biofouling control. At least one operator is planning to inject the maximum legally permissible amounts.[16] Even if condensers can withstand high-level chlorination, service water systems are not made of titanium; their corrosion has in some instances created the entry-path for chlorides in plants with chronic problems.[17]

There is thus little reason at present to conclude that the corrosion problem in salt water cooled reactors is under control. Meanwhile, there are no studies of the impact on actual plant safety and productivity. Existing research, statistical and otherwise, tends to be oriented toward solving problems, and away from evaluation of their ongoing impact. The present work seeks to analyze accumulated experience as reflected in capacity factors and forced outage rates.

DESCRIPTIVE STATISTICS

Average capacity factors of PWR and BWR units, further divided into those cooled by salt water and those which are not, are presented in Figure 1 for the years 1975 to 1981.[18] The long-term decline of the average capacity factor for salt water cooled (SWC) PWRs is immediately apparent. From 1975 to 1977 SWC and non-SWC PWRs had comparable averages, but the gap widened over the years 1978 to 1981, amounting to about 7 percentage points in 1981.

Simple trend analysis of these averages is shown in Table 1. The SWC PWR average is estimated to decline 3.7 points a year, while the non-SWC PWR average is estimated to decline at only 1.2 points a year. The t-statistic for the SWC PWR rate is significant at the highest (99.8%+) confidence level, while the non-SWC rate is significant at the 75% level.[19] Performance is similar for BWR units, except that non-SWC BWRs display marked maturation in capacity factors and SWC BWR performance is the most erratic.

Further indication of poor SWC PWR performance is evident in an examination of average outage hour trends in four outage categories. These are forced and scheduled outages due to equipment failures and maintenance needs. The data were derived from the NRC Grey Book database, exclude data from the Three Mile Island accident, and do not include NRC mandated shutdowns.[20] Table 2 presents trend analyses of each outage type by each reactor type. The percentages represent the estimated average annual escalation rates. The figure in parenthesis is the t-statistic. Significance levels for reported t-values are shown at the bottom of the table.

Significant sharp rates of increase for SWC PWRs can be seen in forced equipment failure, scheduled equipment failure, and forced maintenance outages. The only significant aggregate trend for non-SWC PWRs is the decline in the rate of scheduled maintenance outages. Trends for BWR units are not as clear.

The rapid increase in scheduled equipment failure outages reflects the increasing number of outages for steam generator replacement and major condenser overhaul. Significantly, almost all such work is being performed on SWC PWR units (Surry units 1 and 2, Turkey Point units 1 and 2, and San Onofre unit 1). And despite these outages being segregated from the forced outage data, those rates are escalating as well.

Thus far, it has been demonstrated that the aggregate trends show very disturbing tendencies. Trend analysis may not, however, tell us much about what to expect from individual units; and what effects unit size, age, steam system supplier, and recent regulatory changes have had on performance. The regression analyses that follow are designed to take these factors into account.

REGRESSION ANALYSIS

The regression model used divides the reactor's operational experience into four periods: the first calendar year of operation, the second to fifth years, the fifth to tenth years, and the tenth year onward. For each period there is a magnitude term for the indicator under study (capacity factor or FOR). And for every period except the first there is a rate of change term. Also, for every period there is a term that tests for the impact of unit size on the magnitude term, with the exception of the last period for SWC PWRs. This term is excluded because there is only one SWC PWR (San Onofre) in operation for ten years. Additionally, there are binary variables to test for the overall impact of steam system supplier and "post-TMI" operation on results.

Forced outage hours are converted to forced outage rates (FORs) by the following relation:

$$\text{FOR} = \text{outage hours} \,/\, ((8760 * \text{Frac}) - \text{Static Outage Hours}),$$

where Frac is the fraction of the year that the plant has been in commercial operation, and Static Outage Hours is the sum of refueling, operator training, and NRC mandated outage hours for each unit-year. Capacity factors are annualized for first calendar years of operation.

Because annual observations are not based on equal numbers of operational hours, observations are weighted using the generalized least squares (GLS) regression technique.[21] GLS weights observations by use of a estimated variance-covariance matrix of the observed data.[22] For capacity factor analysis, the variance of an observation was assumed to be proportional to the square of the inverse of the fraction of the year the unit was on line. In FOR analysis, the variance was assumed to be:

$$\text{Var} = 1/(\,(8760 * \text{Frac}) - \text{Static Outage Hours})^2.$$

Because of the relative complexity of this procedure, results are not presented BWR units and scheduled outages. BWR units could be analyzed on a similar basis. Aggregate trend analysis gives an adequate picture of scheduled outages.

The regression equation, together with capacity factor and FOR results is shown in Table 3. The break (BRx) terms are binary variables that are 1 until year of operation x, when they become 0. Thus, (BR9-BR4) has value 1 during the fifth to ninth years of operation, and is 0 otherwise. Age is a linear trend which starts at 1 for the first calendar year and increments by one every year. MDC is the unit's maximum dependable capacity. WESTM is a binary variable that is 1 for Westinghouse steam system units, BWSTM is a binary variable that is 1 for Babcock and Wilcox (B&W) steam system units. Estimates for Combustion Engineering (C-E) units need no correction and serve as the baseline. GETTOUGH is a binary variable that is 1 during the years 1979-81 and is 0 otherwise. All M- coefficients refer to magnitude, R- coefficients to rate of change, and S- coefficients to size effects. The 1, 2, 3, or 4 suffix to these coefficients refers to the period of effect.

Lines 1 through 14 show the estimated coefficients reported for each case. The t-statistics are reported in parenthesis below the reported coefficients. The R^2 statistics are presented on line 15, the standard error of the regression on line 16, the equation F-statistic on line 17, Cond(x)[23] on line 18, and the number of observations on line 19.

Non-Salt Water Cooled PWRs

Examination of non-SWC PWR results shows a maturation period estimated as the first four years, and a period of stability over the next five years, after which capacity factors, on average, decline. Additionally, coefficients S1 to S4 show that there is a significant negative impact of increasing unit size, estimated to grow from a 1.4 point decrease per 100 MW of MDC in the first year of operation to 4 points after the ninth year of operation. During the second through ninth years the decline of about 2.8 points per 100 MW is stable and significant at the highest confidence level.

There was an estimated 4 point average decline in capacity factors during the 1979-81 period, which is presumably due to NRC shutdowns and more stringent regulation. B&W reactors are estimated to have 7.6 points higher average capacity factors (TMI 1 and 2 shutdowns excluded) than C-E units, while Westinghouse units are estimated to perform 12.7 points better.

Equipment failure FORs for non-SWC PWRs reach an early peak in the second year, followed by a significant decline until the fifth year and a small average decline until the tenth year, followed by a strong 2.7 point per year escalation rate in succeeding years. There are only weak correlations with unit size until the latest period of plant operation. B&W units are estimated to have a 3.7 point higher average forced outage rate than C-E units; Westinghouse units are not distinguishable from C-E's.

Maintenance FORs for non-SWC PWRs show low levels, and little correlation with unit size except a small, but significant, positive correlation for the first year of operation. There is an estimated .5 point increase in the outage rate in the period 1979-81. A 1.5 point relative decrease is predicted for B&W and Westinghouse reactors.

Salt Water Cooled PWRs

The overall regression statistics --- the R^2, standard error, and F-statistic --- indicate higher explanatory power of the regression model for SWC units. The regression for capacity factor indicates accelerating rates of decline with increasing unit age; only the results of the period from years two through four are of questionable statistical significance. The size effect is estimated to be about twice as great as for non-SWC PWRs, as indicated by coefficients S1 to S3. The size effect increases with age, as does its statistical significance. There was no detectable "post-TMI effect" on capacity factors and both B&W and Westinghouse units are estimated to have sharply poorer performance compared to C-E units.

Equipment failure forced outage rates are estimated to be very high for the first year, to decline sharply in the second year, and then to increase at an average rate of 2 points a year until the fifth year, and then to increase at a rate of about .5 point per year until the tenth year, when an average growth of 9 points a year ensues. The early period increase rate is significant at the 85% level, the mid-period increase rate is significant at only the 75% level. In the first year of operation, size is negatively correlated with outages. The effect reverses, at a modest level, until the fifth year. From the fifth through ninth years the size effect is estimated to be 4 points per 100 MW of reactor size and is significant at the highest confidence level.

B&W units are estimated to have a 13.7 point higher equipment failure FOR than C-E units; Westinghouse units are estimated to have rates 4 points higher. In the case of B&W, this reflects the poor performance of Crystal River 3. The regression estimates that overall, equipment failure FORs declined by 2.9 points over the 1979-81 period, an effect significant at only the 75% confidence level.

The regression for maintenance FORs also shows distinct behavior from non-SWC PWRs. There is a very strong correlation between first year FORs and unit size. There is a weak but positive correlation between size and FOR during years two through nine. SWC PWR Maintenance FORs are stable over time (as compared to declining levels for non-SWC PWRs). There is a sharp rise in the ten year plus period because of the San Onofre condenser failure. Forced maintenance outage rates are estimated to have a 1 point decline over the 1979-81 period. Overall B&W units are estimated to have a 1.7 point higher rate than Westinghouse and C-E units.

DISCUSSION

Consideration of these results shows that the average capacity factors and forced outage hours summarized in Figure 1 and Tables 1 and 2 are the result of fundamentally different performance patterns between SWC and non-SWC units, at least in the case of PWRs. A consistent pattern is observable over all years of operation for SWC units: larger units have poorer performance and performance of all units is expected to decrease with time. Non-SWC units show marked decline after the ninth year of operation, and strong correlations by the mid-period of plant age, but in all cases the effects are smaller.

The modest decline in equipment failure FORs over the 1979-81 period is cause for some hope that there is some stabilization of this indicator.

An unexpected finding of this study has been that both B&W and Westinghouse steam systems outperform C-E units in non-SWC environments, but perform much more poorly with salt water cooling. This result could be due to C-E's early choice of AVT chemistry over phosphate control or to the "egg-crate" design of C-E tube supports.

Estimated and projected average capacity factors for salt and non-saltwater cooled Westinghouse units are presented in figure 2. The circles represent non-SWC units, while the triangles are for SWC units. Projections are made for 600, 800, and 1100 MW units. Improving performance until the tenth year of operation is predicted for non-SWC units, while SWC units are estimated to have sharp and continuous declines. Smaller units are uniformly expected to outperform larger units in either environment. The expected difference is much greater for SWC units. Similar estimates for equipment failure FORs are shown in Figure 3. The strength of the negative impact of increasing unit size on SWC performance contrasts sharply with results for non-SWC units, and is probably the most significant finding presented.

CONCLUSION

This study shows that reactor aging is already a serious problem for salt water cooled PWRs. GLS analysis of capacity factors and FORs reveals significantly increasing forced outage rates and declining capacity factors, strongly correlated with size, for SWC units. Variations are found between reactor manufacturers, including higher equipment failure FORs for B&W units, excluding TMI data. It is likely that existing data are capable of yielding yet more precise information. Given the uniformly negative results for salt water cooled units, it seems imperative that further analysis be done because of potential safety and economic implications.

Further, the results presented here underscore the need for careful statistical analyses of other measures of accumulated operational experience. As one example, work performed by this author on trends in operations and maintenance (O&M) costs found similar differences between SWC and non-SWC units in overall O&M costs, their rate of increase, and their correlation with unit size.[24]

Existing research has emphasized problem solving while failing to adequately study the performance impacts of problems that resist solution. This approach is faulty on two grounds: (1) The practical economic significance of factors such as declining plant productivity are ignored. And (2), chronic operational problems should be recognized as serious safety problems now that the potential consequences of synergetic interactions have been acknowledged.

The adoption of IEEE-627 and the extension of aging standards from structural components to all safety related equipment is a vital step to insure nuclear safety. Presentations at this conference, however, have pointed out in great detail the limitations of accelerated aging testing.[25] There is no effective way to know that a component maintains its functional integrity, except through operational experience. There is a great need to expand the limited analyses of operational failure rates to study the effect of aging on component failure rates.[26] There is no excuse for not using all the information available.

NOTES

* Revised version of paper prepared for the NRC Conference on the Aging of Nuclear Power Plant Components, Bethesda, Md., August 4-5, 1982. Acknowledgement is made of the use of outage data developed with the Energy Systems Research Group during the course of capacity factor analysis. (See ESRG 82-12, "Report on Maine Yankee ...," recently released, and a forthcoming report on the Indian Point units.)

[1] Water from rivers which are tidal estuaries at the point of water intake or polluted seawater.

[2] See, for example, R. G. Easterling, "Statistical Analysis of Power Plant Capacity Factors Through 1979," NUREG/CR-1881 (April 1981). Summarized in Energy vol. 7 No. 3 (1982), pp. 253-58. Also C. Komanoff, "Nuclear Plant Performance Update 2," (June 1978) Koumanoff Energy Associates, 475 Park Avenue South, N.Y., N.Y.

[3] There are outage studies. See, for example, Nuclear and Large Fossil Unit Operating Experience, S. M. Stoller Corp., EPRI NP-1191 (Sept. 1979). Such studies typically identify outages causes but omit serious time-series or impact analysis.

[4] See, for example, D. G. Eisenhut, B. D. Law, and J. Strosnider, Summary of Operating Experience with Recirculating Steam Generators, NUREG-0523, US NRC, (Jan. 1979); Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors, NUREG-0691, US NRC (Sept. 1980) and W. J. Shack, et. al., Environmentally Assisted Cracking in Light Water Reactors: Critical Issues and Recommended Research, NUREG/CR-2541 US NRC and ANL-82-2 (Argone National Laboratory), (Feb. 1982).

[5] See the above and Corrosion Related Failures in Power Plant Condensers, Battelle Columbus Labs, EPRI NP-1468, (Aug. 1980).

[6] See Corrosion Related Failures ... .

[7] See Environmental Assisted Cracking ..., p. 2.1 and ff. and B. M. Gordon, "The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature," Mat. Perfor., 19(4), 29-38 (1980).

[8] See Summary of Operating Experience ..., pps. 3 and 42.

[9] See Summary of Operating Experience ..., sections 4 and 6. Cu-Ni alloys are also under consideration for some units although their corrosion resistance is inferior to titanium.

[10] See Summary of Operating Experience ..., pps. 48 and 73.

[11] See comments of L. J. Martel, Workshop Proceedings: Outage Planning and Maintenance Management, EPRI WS-78-94 (June 1979), p. 13-9.

[12] See Corrosion Related Failures ..., section 5. The cathodic protection necessary to prevent prevent galvanic corrosion can embrittle titaniam tube ends through hydriding. There has been minimal experience with use of titanium in 1100 MW class reactors. See also, Assesment of Condenser Leakage Problems, MRP Associates, EPRI NP-1467 (Aug. 1980).

[13] See D. G. Tipton, Effect of Mechanical Cleaning on Seawater Corrosion of Candidate OTEC Heat Exchanger Materials. Part 2., Argonne Nat. Lab W-31-109-ENG-38 (June 1981).

[14] Central Maine Power's almost-new Wyman 4 unit suffered catastrophic condenser failure leading to a prolonged outage. There is a docket presently before the Maine PUC on this matter.

[15] See J. S. Nickels, et. al., "Effect of Manual Brush Cleaning on Biom-

ass and Community Structure of Microfouling Film Formed on Aluminum and Titanium Surfaces Exposed to Rapidly Flowing Seawater," in App. and Env. Microbio., Vol. 41, No.6 (June 1981), pp. 1442-53.

[16] Public Service of New Hampshire, constructor of the Seabrook units. PSNh intends to maintain chlorine levels of .2 mg/L in the intake water by the injection of 848 pounds of chlorine equivalent per hour for at least half the year. See Draft Environmental Statement Relating to the Operation of Seabrook Station Units 1 and 2, NUREG-0895, US NRC (May 1982). See also comments of R. L. Kaufman to L. W. Wheeler (NRC Project Manager), 20 June 1982. Kaufman estimates that this will result in the injection of as much as 3.7 million pounds of chlorine per year. General environmental implications have not been carefully studied either.

[17] Surry and Indian Point 2 units. Re: Surry, see Investigation and Evaluation..., p. 2.6, and "Stress Corrosion Cracking of Recirculation Spray Piping," Virg. El. & Pwr. Co. Rpt. USRE-52-76-15, USNRC Docket 50-280, 281 (Nov. 1976). Regarding Indian Point, see the testimony of H. E. Sheets, in In the Matter of the Proceeding on the Motion of the Commission to Investigate the Outage at the Indian Point No. 2 Nuclear Generating Plant, NY PSC Docket 27869 (April, 1981), p. 26, ff. Catastrophic service water system failure led to the flooding of the Indian Point 2 containment building to the 46 foot level of the reactor vessel.

[18] Date for Dresden unit 1 is not included because it has been removed from the NRC Grey Book data base. Data for some new plants coming on line was excluded because of the time-series nature of the regression analysis presented.

[19] The confidence level refers the the probability that the true value of the estimated coefficient is actually greater than 0, and is thus a meausre of the precision of the estimate. Typically 95% is regarded as adequate to completely rule out chance association. See P. G. Hoel, et. al., Introduction to Statistical Theory, Houghton Mifflin (1971), p. 85, ff., for more information.

[20] Also excluded was data from the Browns Ferry incident, as it had such a large effect on the forced maintenance outage rate, that it was necessary to know what the trend would be without it. The data was tabulated from an NRC computer tape by a program written by Jon Wallach of ESRG for this purpose.

[21] For a description of GLS, see Kmenta, Elements of Econometrics, p. 499, ff.

[22] Normal regression assumes that all variances are equal and that all covariances are equal to 0.

[23] Cond(x) is a measure of multicollinearity. Values over 100 indicate that serious multicollinearity may be present. This condition means that the regression procedure cannot separate adequately the contributing causes to an effect, leading to instability in estimates. For more information, see Kementa, supra, pp. 380-91.

[24] An early version of my O&M work presented in Appendix B of ESRG 78-14, an analysis of the Limerick construction project performed for the Pennsylvania PUC, an updated study will be available from NTMR by Dec. 1982.

[25] See the papers by M. Brown, S. P. Carfagno, and T. H. Ling in these proceedings. See also, Review of Equipment Aging Theory, Franklin Research Institute, EPRI NP-1558.

[26] There is significant work in progress at the present time developing methods for operational failure rate analysis, a difficult undertaking. No work, however, is known to be analyzing the effects of aging.

Fig. 1

Average Capacity Factors

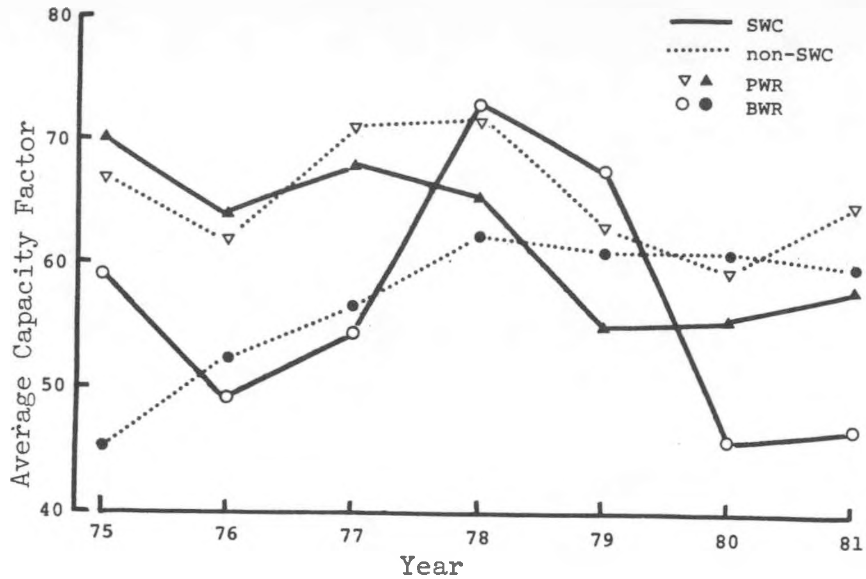

Fig. 2

Predicted Capacity Factors for Westinghouse Units

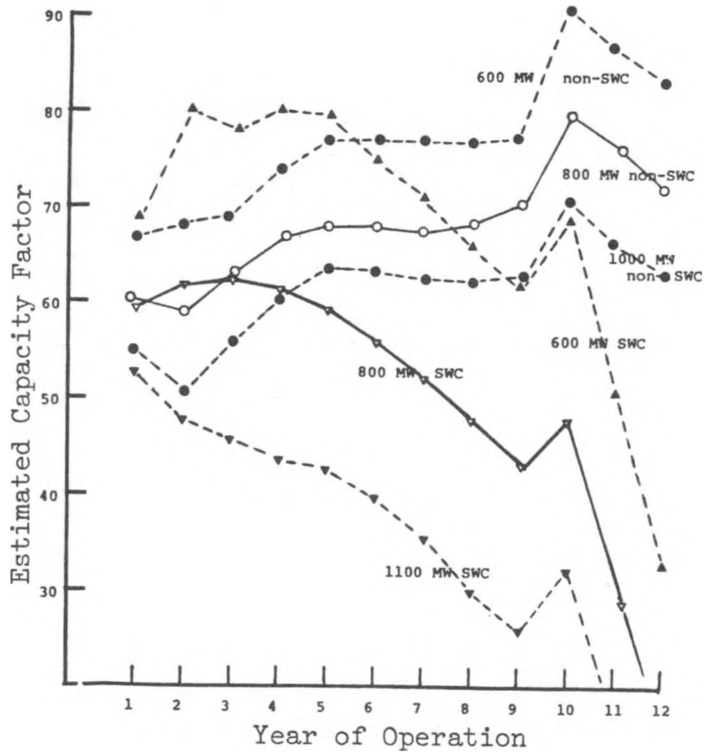

Fig. 3

Predicted Equipment Failure FORs -- Westinghouse PWR Units

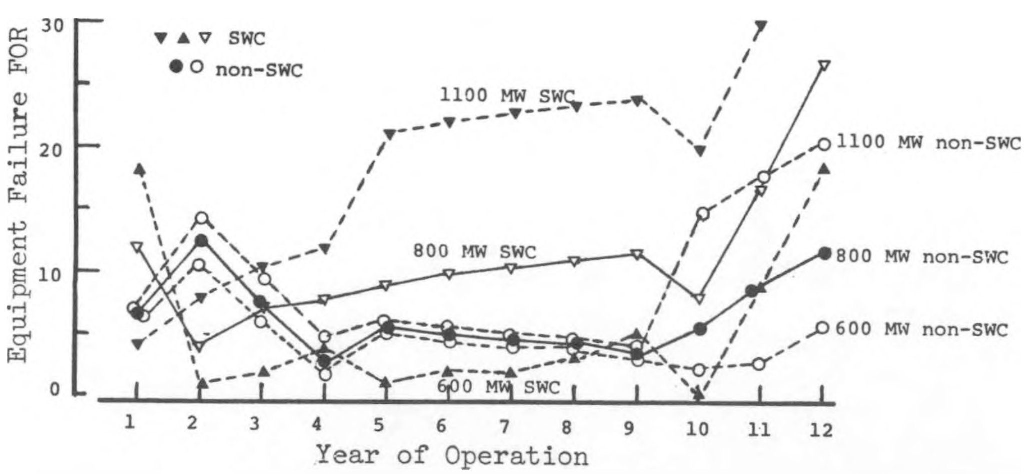

TABLE 1: Annual Capacity Factor Trends

SWC PWR	non-SWC PWR	SWC BWR	non-SWC BWR
-3.7%	-1.2%	-2.25%	4.4%
(3.30)	(.92)	(.60)	(3.18)

TABLE 2: Outage Rate Trends

	SWC PWR	non-SWC PWR	SWC BWR	non-SWC BWR
Equipment Failure Forced Outages	15.86% (1.86)	-.09% (0.13)	-1.85% (0.13)	12.81% (1.49)
Maintenance Forced Outages	124.79% (3.99)	-.12% (0.23)	49.57% (1.84)	-33.80% (1.85)
Equipment Failure Scheduled Outages	52.58% (2.66)	-7.12% (0.37)	----- ---	-15.31% (0.93)
Maintenance Scheduled Outages	-6.41% (0.39)	-14.25% (3.91)	7.30% (0.31)	-9.26% (0.76)

(Levels in parenthesis are t-statistics. Significance levels are as follows: .72, 75%; 1.27, 87.5%; 1.94, 95%; 2.57, 97.5%.)

Average Outage Hours by Type and Year.

	1975	1976	1977	1978	1979	1980	1981
Equipment failure Forced Outage Hours:							
SWC PWR	166.61	512.42	398.59	456.82	333.94	371.06	865.74
non-SWC PWR	784.51	516.50	367.21	346.57	350.29	537.74	711.54
SWC RWR	502.44	427.92	503.47	171.44	264.50	156.44	1025.44
non-SWC BWR	206.99	157.26	166.07	391.46	560.29	341.21	253.63
Maintenance related Forced Outage Hours:							
SWC PWR	2.87	2.08	60.41	150.91	44.24	215.56	284.12
non-SWC PWR	176.65	43.50	4.86	29.67	116.23	52.54	53.47
SWC BWR	112.65	6.86	151.51	64.40	158.14	150.12	608.60
non-SWC BWR	76.85	26.63	80.26	68.58	6.41	13.02	18.94
Equipment Failure Scheduled Outage Hours:							
SWC PWR	9.07	133.50	85.77	57.88	63.54	250.00	340.36
non-SWC PWR	11.91	120.18	50.99	34.13	8.02	27.89	30.75
SWC BWR	93.94	0.00	34.41	106.82	19.16	86.64	134.40
non-SWC BWR	5.94	19.64	27.90	8.38	19.25	10.02	2.78
Maintenance related Scheduled Outage Hours:							
SWC PWR	394.87	265.42	517.67	89.69	65.53	254.93	443.97
non-SWC PWR	278.47	305.21	188.38	184.52	222.69	165.61	114.13
SWC BWR	120.48	578.42	99.46	186.84	31.42	143.84	871.94
non-SWC BWR	341.10	224.52	81.66	82.51	84.18	115.74	229.70
Unit-years of operation:							
SWC PWR	7.66	9.36	13.22	14.00	14.00	14.00	14.00
non-SWC PWR	19.05	20.83	22.08	24.57	25.25	25.81	27.00
SWC BWR	4.16	5.00	4.79	5.00	5.00	5.00	5.00
non-SWC BWR	16.18	16.00	18.83	19.00	19.32	20.00	20.00

SWC = salt water cooled, 1975 and 1976 data omit Browns Ferry units 1 and 2, Humboldt Bay excluded after 1976, TMI-2 entirely excluded, TMI-1 excluded after shutdown.

TABLE 3: Regression Results

FACTOR = (M1 + S1*MDC) * BR1 + (M2 + S2*MDC + R2*AGE) * (BR4-BR1)+
(M3 + S3*MDC + R3*AGE) * (BR9-BR4) + (M4 + S4*MDC + R4*AGE)
* (1-BR9) + K*GETTOUGH + Y*BWSTM + Z*WESTM

		non-salt water cooled PWR			salt water cooled PWR		
		cap factor	equipment	maint.	cap factor	equipment	maint.
1	M1	.6482 (4.59)	.6450 (0.66)	-.0045 (0.19)	.9831 (2.51)	.3192 (1.42)	-.4985 (6.48)
2	M2	.6273 (4.65)	.1559 (1.73)	.0285 (1.30)	1.3463 (5.30)	-.1730 (1.19)	-.0263 (0.53)
3	M3	.8296 (8.24)	.0691 (1.03)	.0388 (2.40)	1.5531 (6.89)	-.3132 (2.42)	-.0155 (0.35)
4	M4	1.4153 (4.24)	-.4541 (2.14)	-.0327 (0.63)	3.0683 (5.35)	-1.2187 (3.70)	-1.7990 (16.01)
5	R2	.0438 (1.71)	-.0462 (2.70)	-.0021 (0.50)	-.0232 (0.75)	.0201 (1.14)	.0011 (0.18)
6	R3	-.0004 (0.04)	-.0048 (0.72)	-.0024 (1.49)	-.0432 (2.58)	.0070 (0.72)	.0001 (0.26)
7	R4	-.0366 (2.23)	.0277 (2.65)	.0012 (0.47)	-.1831 (4.01)	.0932 (3.56)	.1599 (17.89)
8	S1	-1.41 E-4 (0.86)	-2.11 E-6 (0.01)	4.91 E-5 (1.80)	-3.23 E-4 (0.71)	-2.96 E-4 (1.14)	6.01 E-4 (6.78)
9	S2	-2.82 E-4 (2.59)	7.14 E-5 (0.96)	-4.53 E-6 (0.25)	-6.56 E-4 (2.50)	1.59 E-4 (1.06)	3.53 E-5 (0.68)
10	S3	-2.79 E-4 (3.50)	6.72 E-6 (0.13)	-6.27 E-6 (0.50)	-7.30 E-4 (3.54)	4.10 E-4 (3.47)	3.48 E-5 (0.86)
11	S4	-4.04 E-4 (1.40)	2.98 E-4 (1.62)	3.62 E-5 (0.81)	---- --	---- --	---- --
12	K	-.0402 (1.59)	.0092 (0.56)	.0049 (1.23)	.0185 (0.39)	-.0291 (1.07)	-.0115 (1.24)
13	Y	.0766 (1.69)	.0368 (1.26)	-.0167 (2.35)	-.2432 (3.70)	.1370 (3.63)	.0170 (1.32)
14	Z	.1276 (3.11)	.0002 (0.01)	-.0143 (2.38)	-.1076 (2.92)	.0412 (1.95)	-.0027 (0.37)
15	R^2	.3031	.1465	.1378	.4518	.3502	.8813
16	F-stat.	5.19	2.046	1.906	5.29	3.46	47.62
17	Std. error	.1131	.0788	.0192	.1161	.0666	.0228
18	Cond(x)	36.63	36.87	36.87	36.24	36.24	36.24
19	Observations	169	169	169	90	90	90

SIGNIFICANCE LEVELS: (t-statistics are in parenthesis) .678, 75%; 1.16, 87%;
1.67, 95%; 2.00, 97.5%; 2.86, 99.8%

Effects of Simultaneous Exposures to Heat and Radiation on Insulation Life

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ABSTRACT

The capability for predicting service life of an insulating material from test procedures based upon thermal aging alone becomes much more complex when the aging is conducted in an environment of heat and radiation combined. The Arrhenius equation has been found to be applicable to most organic materials for thermal aging, in that the plot of log life versus reciprocal absolute temperature is a straight line. However, the life of an insulating material which will be used in radiation environment cannot be predicted by the usual thermal-aging methods; neither can it be predicted from experiments in which thermal aging follows a pre-exposure to radiation at room temperature, or vice versa. To have any reliable significance, the experiment must be conducted in a combined environment of both heat and radiation. At the Naval Research Laboratory an apparatus has been designed and used to achieve this exposure condition. Initial studies with this facility have demonstrated that in environments combining radiation with high temperatures, the life of magnet wire insulating materials is far different from that obtained in individual environments or from sequential exposures to each environment (1). Each material is affected differently, with some showing an accelerated degradation while others have a longer life in some environmental combinations. For example, the simultaneous aging at high temperatures in gamma-ray radiation of a polyimide, a polyvinyl formal, and a polysiloxane produced considerably longer lifetimes when compared at the same temperature to thermal aging along. On the other hand, polytetrafluorethylene deteriorated much more rapidly under the simultaneous action of heat and radiation. The observed increases in life under some conditions is probably due to a balancing of the chain-scission and crosslinking mechanisms in some polymers in which the rates of the two reactions vary with the temperature and dose rate conditions, particularly when there is a crossover of the glass transition temperature.

INTRODUCTION

A standard procedure for determining the thermal life characteristics of magnet wire insulation now in practice for some time, has been very useful for classifying each new insulating material according to its specific maximum operating temperature (2). Short-time high-temperature tests provide data points which can be extrapolated (by application of the Arrhenius relationship) to predict the aging life of the insulation at the rated operating temperature of the equipment in which the wire will be used.

Statistical treatment of experimental life determinations under applied service stresses has demonstrated that the effective life of wire insulations at a controlled temperature is a function of deterioration rate and that this rate is temperature dependent. Since this deterioration is the result of a chemical reaction such as oxidation, this rate can then be expressed by a mathematical formula derived from a combination of the first order kinetic equation and the Arrhenius equation which relates the thermodynamic equilibrium constant of a reacting system to the exposure temperature (3). This derivation in final form is expressed as follows:

$$\text{Log } L = \log A + B/T$$

where L = hours of life to a specified end-point while aging at a temperature T

T = absolute or Kelvin scale temperature (°C + 273)

A and B = constants of the intercept and the slope, respectively, and are related to the entropy of the system and the activation energy of the degradation process

This equation yields a straight-line curve which is easily interpreted when plotted on semi-log graph paper which has been specially prepared to present the hours of life on a logarithmic scale as the ordinate and the aging temperature in degrees Celsius on the abscissa with a scale that has been graduated to the reciprocal of the corresponding absolute temperature.

The life at any temperature can then be predicted from only three or four experimental points by extrapolating the curve of the experimental data of life versus temperature following a regression analysis. Figure 1 shows a typical curve plotted from experimental data points. The reliability of the prediction will then depend upon the accuracy of the equation constants selected to describe the curve. Since deterioration is a function of chemical changes occurring in the insulation material, extrapolations are limited to the temperature region in which there are no transitions either in the physical structure of the plastic, such as melting, or in the chemical mechanisms of deterioration, such as a change from free-radical chain scission to rapid oxidation or combustion.

EXPOSURE FACILITIES

The radiation sources used in this study consisted of a 7000-Curie and a 1500-Curie ^{60}Co field. These units are submerged in approximately 12 feet of water which affords adequate shielding for the operators. Each unit is arranged so that sealed stainless steel containers housing the experiments can be accurately positioned to receive a specified exposure rate. The maximum exposure rate attainable in the center position of the large source was 1.8×10^6 Roentgens per hour. Eighteen other positions are available about each source with exposure rates ranging to a minimum of about 10^3 Roentgens per hour. A view of this arrangement as seen from the top of the pool with several of the chambers in place for this experiment is shown in Figure 2. The containers used in this study have been constructed with a double wall to reduce heat transfer losses from the specimen area. An easily removeable lid is sealed against water leakage with a neoprene rubber O-ring.

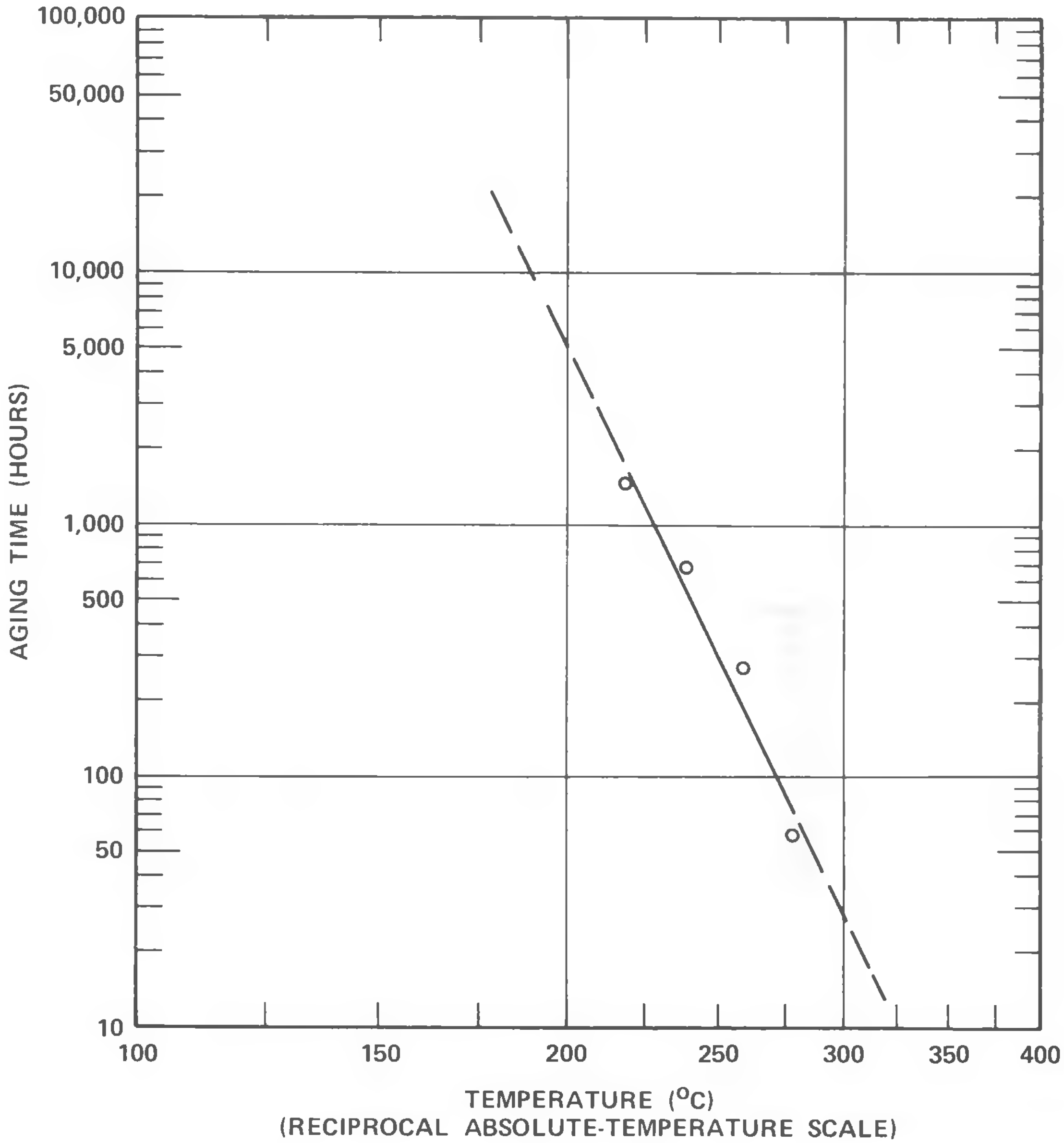

Fig. 1 — Typical arrhenius curve

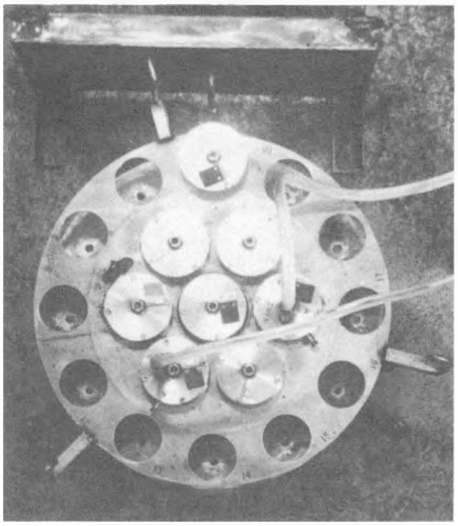

Fig. 2 — ^{60}Co radiation exposure unit

In order to conduct exposures to the combined environments of gamma radiation and heat, a unique assembly consisting of a tiny heating chamber was designed and constructed to fit snugly into the double wall container. This consisted of a thin-wall copper cylinder wound with No. 30 nichrome wire and electrically insulated with braided fiberglass sleeving and asbestos tape. The power lead wires and thermocouple wires extended through a short stainless steel tubing welded to the container lid and then through a Tygon (plasticized polyvinyl chloride) tube to the top of the water pool and into the control instruments. A view of the exposure assembly is shown in Figure 3. Temperature measurement and control instruments for each unit consist of an indicating-controlling pyrometer and a Variac, controlling power to the heating element. With this unit, which employs the proportioning-control system of sensing heat requirements, the temperature fluctuation at the center point of the oven was observed to be $\pm 1°C$ at 300°C. By means of several thermocouples placed throughout the oven a contour of temperature gradient was measured and found to be constant within 3°C from top to bottom of the chamber.

EXPERIMENTAL

Wires insulated with various enamels and enamel-varnish combinations which were found to be compatible in thermal environment studies were selected to represent a variety of polymer classes. These are: polyvinyl formal, polyester, silicone, aromatic polyimide, and fluorocarbon. From the thermal — life curves available on these wires, aging temperatures and the approximate exposure times per test cycle were selected. Ten specimens of the twisted pair configuration, having the standard pitch, as specified in reference 2, but slightly shorter in length (3.25 inches compared to the standard 4.75 inches) so as to fit into the heating chamber, were tested for each data point. A test for determining the end of aging life was applied after each exposure period. The specimens were removed from the radiation source, immediately transferred to room temperature, and when cool were given an electric strength proof test of 1000 volts for 1 second. If a sample failed under this test, its aging life was recorded. If it passed this test, it was returned to the environment for the next aging period and tested after each cycle until it did fail. The average of the 10 specimens was then reported as the aging life data point.

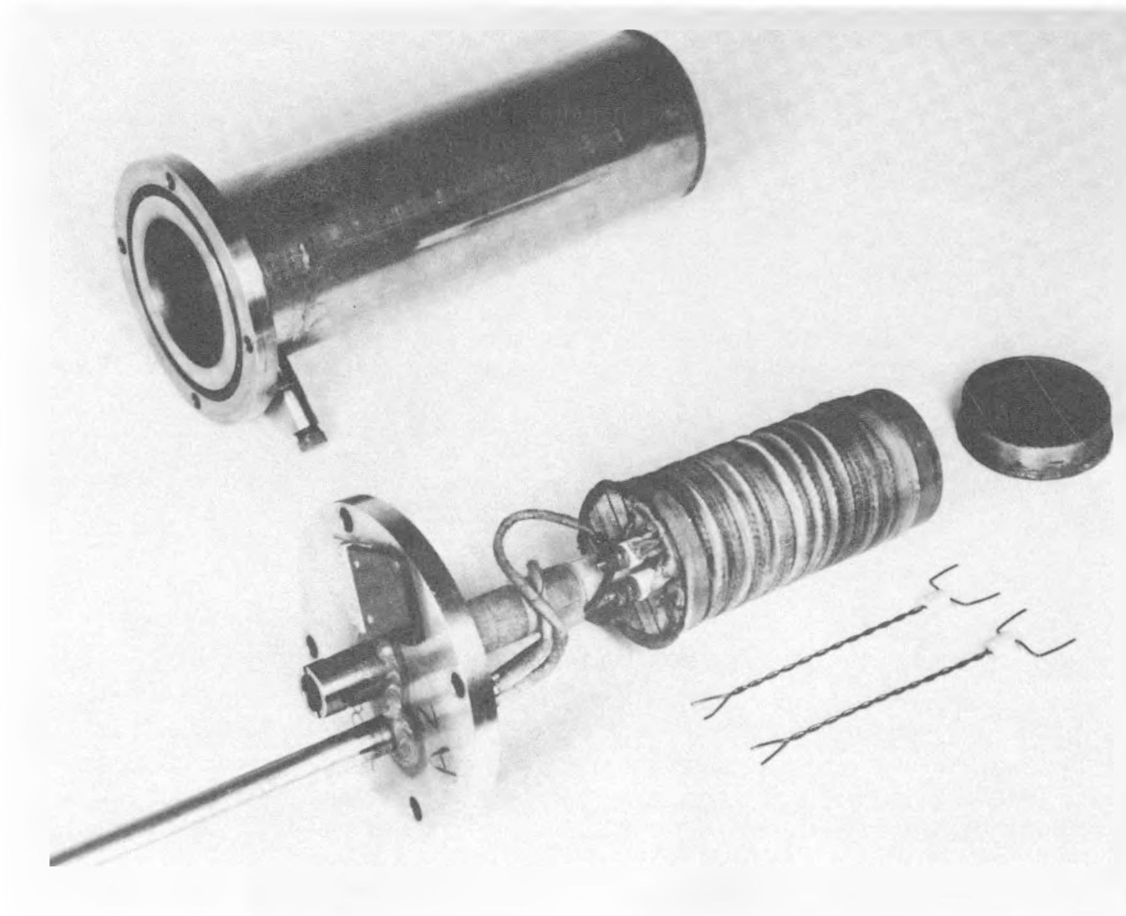

Fig. 3 — Radiation-thermal exposure assembly

RESULTS

A comparison of wire aging lives for the various insulation materials is presented in Table 1. The first section gives the lives obtained by thermal aging of unirradiated specimens, and the second section gives those obtained with the combined environment.

As can be noted by comparing the column headed "Percent of Thermal Life," the combined environment can generate an entirely different aging life characteristic to the insulated wire from that obtained by thermal aging alone. For example, it can be noted that in the combined exposures the normal thermal lives of several of these insulations are greatly enhanced. At 300°C combined with the gamma exposure rate of 0.37 MR/h, the life of the polyimide insulation is 820 percent greater than it is at this temperature alone. The thermal life of a polyester insulation aged at 200°C combined with a gamma exposure rate of 0.4 MR/h is also enhanced to 200 percent of normal, thermal aging life.

Polyvinyl formal was studied at several combined environment conditions. In a direct comparison of the two modes of exposure studies, at 180°C combined with a gamma exposure rate of 0.5 MR/h, a 200 percent increase was found; and, as shown in Table I, other combined environment exposures of polyvinyl formal produced even greater enhancements of the thermal lives. Of particular note is the indication that infinite life may be obtained at or near the combination of 180°C and 0.09 MR/h. Aging in this environment was discontinued at 10,000 hours, which is more than 3500 percent of the normal life at this temperature.

Table I

EFFECT OF COMBINED ENVIRONMENT AGING ON THERMAL LIFE OF MAGNET WIRE INSULATION

INSULATION MATERIAL	THERMAL AGING		SIMULTANEOUS RADIATION AND THERMAL AGING			
	AGING TEMP. (^{o}C)	LIFE (H)	EXPOSURE RATE (MR/H)	TOTAL EXPOSURE (MR)	LIFE (H)	PERCENT OF THERMAL LIFE
POLYIMIDE	300	940	0.37	2900	7750	820
POLYESTER	200	3160	0.40	2540	6350	200
	200	3160	0.02	25	1260	40
SILICONE	240	350	0.50	290	570	160
MODIFIED SILICONE	240	500	0.40	124	310	60
POLYVINYL FORMAL	160	630	0.40	2200	5510	870
	180	280	0.50	280	560	200
	200	90	0.40	120	300	330
	180	280	0.09	900	>10,000	>3500
POLYTETRAFLUOROETHYLENE	180	>10,000	0.014	1.5	105	<1
	270	>10,000	0.015	0.75	50	<0.5
	13	∞	0.015	34	2280	–
	13	∞	0.32	240	760	–

The several experiments conducted on polytetrafluoroethylene show that its life is very short at temperatures in the 180°C to 280°C range when combined with a gamma exposure rate of 0.015 MR/h. From the results of the two gamma exposures at 13°C, the temperature of the pool water, it is evident that radiation exposure life is rate dependent, with 700 percent greater life obtained at 0.35 MR/h than at the 0.015 MR/h exposure rate.

The life dependency on exposure rate exhibited here by the experimental data on polytetrafluoroethylene and polyvinyl formal lends support to the necessity for duplicating the complete environment when comparing durabilities of similar materials.

DISCUSSION

Through a consideration of the mechanism of chemical reactions occurring in the polymeric materials of organic insulators, it is possible to provide a general explanation of why each class of material is affected differently by the same exposure conditions. Basically, it is understood that as energy is added to a chemical structure such as a polymer chain, excited states, bond ruptures, and free radicals are generated. Again, depending upon the complete structure of the molecule, the recombination of these free radicals will occur in different formations. Thus it is commonly known that materials will either become crosslinked or degraded (by chain scission leading to many shorter chain fragments). In several references delineating these mechanisms (4,5) it was also explained that these reactions will occur simultaneously in many structures. Thus, it is the net predominance of one over the other that is ultimately observed in the property changes of these materials. The predominant reactions occurring in many various polymers have been observed and tabulated in the above and many other references. These observations were made primarily on room temperature radiation exposures, however. It could be reasonably predicted that for some materials in the presence of both heat and radiation the kinetic balance of mechanisms could be affected with a resulting equilibrium in the net change of the polymer structure, so that the corresponding physical or electrical properties which are ordinarily observed would not change rapidly.

Additional evidence supporting this theory can be found from the results of a detailed study of the effects of irradiation at different temperatures on the crosslinking to scission ratio that was conducted by Bowers and Lovejoy (6). They irradiated a copolymer of tetrafluoroethylene and hexafluoropropylene at different temperatures ranging from 20°C to 380°C. The predominating mechanism occurring was determined by a measurement of melt viscosity. When irradiated at room temperature, net scission results, as shown by the decrease in melt viscosity. At temperatures above the glass I transition temperature (T_g) of approximately 80°C crosslinking predominated. If the material is held precisely at its T_g during irradiation in a nitrogen atmosphere, the effects of both mechanisms counterbalance and there is no net change in the melt viscosity. This example and a number of others, found in other references, indicate that the mechanisms and rates of reactions are very dependent on the characteristics of the material and vary even within classes of polymers. Therefore, each material must be studied individually in order to determine its aging rate in the particular service conditions of concern.

CONCLUSIONS

From the results of this study and the other references discussed it is obvious that predictions of equipment life in radiation ambients cannot be calculated easily from data obtained in a limited test program. Further, the more nearly the test conditions can be designed to simulate service environments, the greater the reliability that can be expected from the resulting observations. And only through extensive testing in the combined environments of anticipated service will it be possible to obtain a reliable estimate of the service life of any material.

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Aging of Electrical Equipment

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During the past two days at this workshop, we have heard about various approaches employed to attempt to predict the life of equipment out to 40 years, in a "nuclear" environment. Life predictions represent an area common to all electrical equipment. I thought it would be of interest to this group to hear how we in the Electrical Systems Division of the Electric Power Research Institute are facing this problem with respect to high voltage cables and power transformers, and how we are attempting to approach potential solutions. I operate out of the Washington, DC, office of EPRI. Prior to joining EPRI, I worked for 15 years in the area of radiation effects on materials, so I feel familiar with life prediction concerns for both areas. After yesterday's session with its detailed emphasis on, and criticisms of the Arrhenius relationship, I concluded that it would be of value to outline specifically where the Arrhenius relationship is valid, why it is not valid in most (but not all) cases relating to equipment aging, and how alternate approaches to life prediction are developed.

One point that this Conference has not noted earlier, however, is how this Industry came to employ and accept the Arrhenius relationship for the aging of insulation. This was first done by Dr. Tom Dakin in the late 1940's[1]. With unique insight, he applied reaction rate methodology to the aging of cellulosic insulation in a hot oil transformer environment over a limited temperature range. The reason that he was successful in employing the Arrhenius relationship for that system, is that the degradation of cellulose follows a single degradation mechanism in this environment. Hence, one is able to predict loss of tensile strength (or elongation) with time under the limited conditions he defined. From that point, it was a relatively short step to seek to project what might happen at constant temperature over long periods of time to achieve equivalent retention of tensile strength, and in the transformer industry, 50% retention of tensile strength has become generally accepted.

In a sense, one might conclude that some luck was involved, as transformers can continue to operate in most cases with cellulosic insulation tensile strength reduced to below 50% of the initial value. However, in recent years the overall situation has changed, as some Utilities (for reasons beyond the scope of this discussion) have had to operate their transformers more often under thermal overload conditions. What has been observed in the laboratory, at the higher resulting temperatures, is that bubbles are evolved. What this means is that the cellulose degradation (which normally occurs at "slow" rates), is now proceeding at these higher temperatures, at rate(s) so rapid that the gases evolved cannot dissolve rapidly enough in the transformer oil; hence, bubbles occur. A Utility engineer responsible for transformer operation gets concerned about possible bubble formation because partial discharges can then occur which, in turn, can lead to premature electrical breakdown of the insulation, and ultimate transformer failure.

The first Figure shows the results of a recently published EPRI report (RP-1289-1) on work performed by the General Electric Company[2]. Note that as the temperature of the insulation increases above 140°C there is a very rapid drop in breakdown strength. The curve on the left of the figure shows the strong influence of moisture on breakdown.

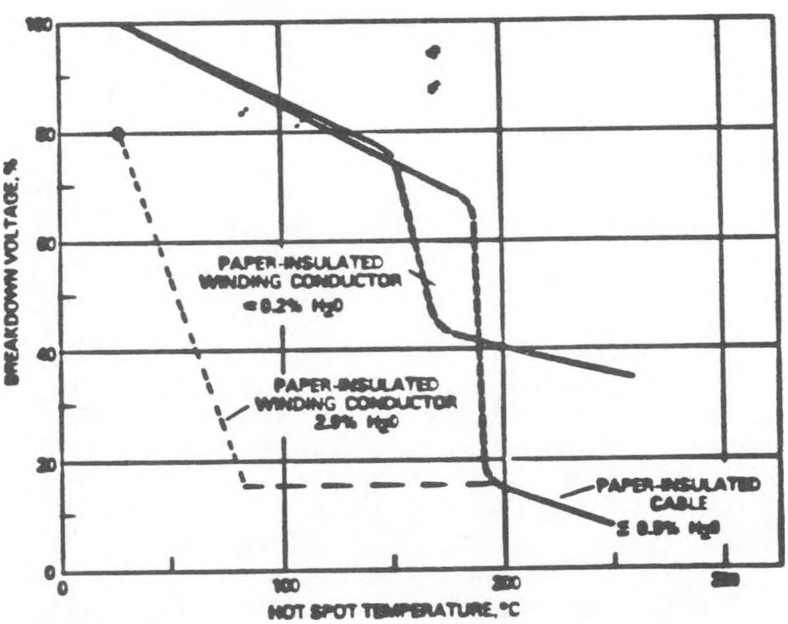

Figure 1 Sixty-Hertz Breakdown Voltage In Percent Of 25°C Strength Of Dry Paper ($\leq$ 0.5% H_2O)

The higher operating temperatures induce failure at shorter aging times than predicted by the Arrhenius relationship. Therefore, what does all this mean with reference to validity of the Arrhenius relationship whose use Tom Dakin pioneered some years ago? The fact is that it is <u>not</u> related. The original projections and the original use of the Arrhenius relationship remain as valid as ever. However, what has happened is that, because of the altered operating environment for the equipment, we have a new degradation mechanism superimposed upon the old one. The original projection remains valid as long as the aging conditions remain as originally defined. When they are changed, which in this case means higher temperatures, the new degradation mechanism assumes greater significance and supercedes the other one. Projections become highly complicated unless this is understood. In a sense, the situation is analogous to radiation of systems in an environment where both radiation and thermal effects (and moisture) are involved. One stress parameter may accelerate the influence of the other one, or it may induce an entirely new mode of failure (or it may do both. We must develop a fuller understanding of the mechanisms of failure involved, before we can have confidence in any projections.

From an electrical systems operation point of view, a question arises as to how serious the above-described problem may be in a real-world environment. The second Figure below shows results from a report by Westinghouse (RP-1289-2) which will be published shortly[3]. In that Figure, we can see that breakdown strength of small coils are compared to larger size model transformers, and while the data at 200°C seems to correlate to some extent, the 180°C data does not correlate at all.

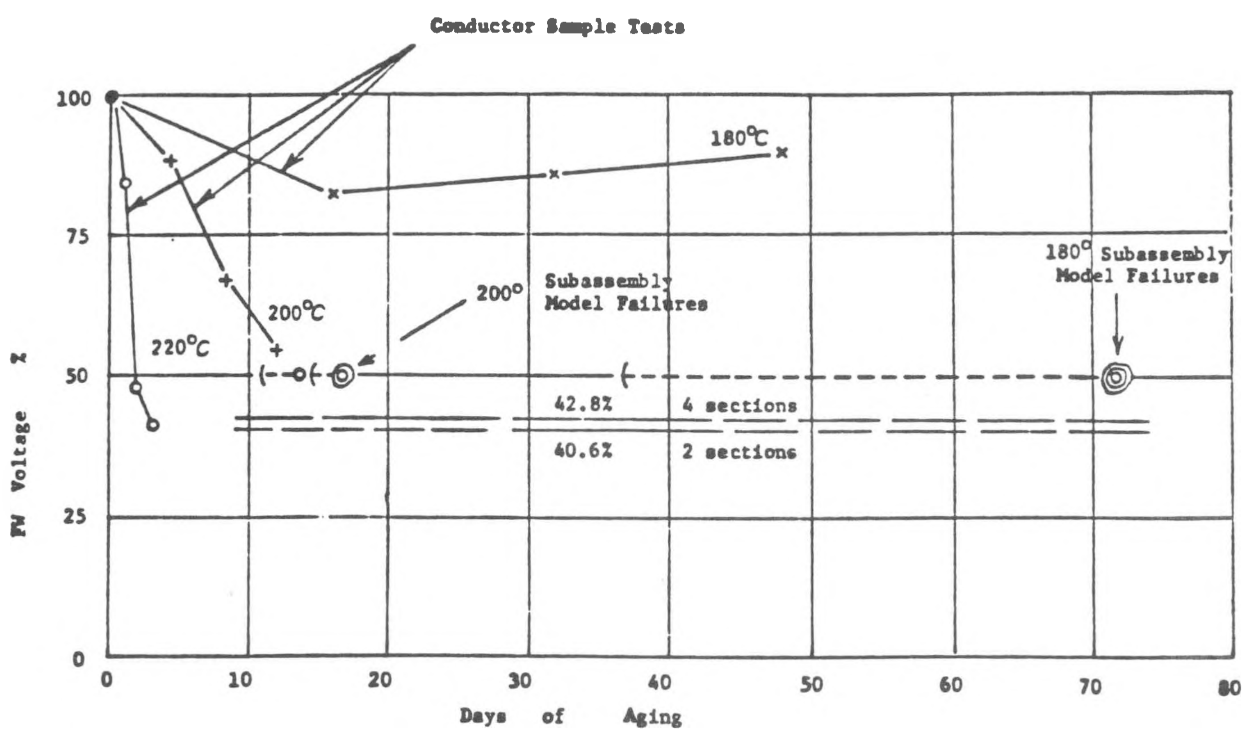

Figure 2 Location of Test Points Based on Area Correction for 4 Sections Exposed and 2 Sections Exposed Initial Test Value for 32 Sections, Plotted with End Point Test Data for Conductor Samples

Westinghouse is evaluating these results employing a modified technique, emphasizing the "area effect" of insulation under stress. This, along with their use of the Arrhenius relationship, will be evaluated in their final report.

How are we to interpret these overall results? We are not quite sure at this time, and we are continuing to fund work in this area to try to understand the significance of "bubbling" in real world operating transformers. The point that I am trying to make here is that to help us make life predictions, one method involves comparison of models with larger systems. This can be performed with success to some extent, but we must be aware of the limitations.

I would like now to discuss some work we are doing in the cable area[4]. In this case, the Arrhenius relationship could never be considered because cables rarely operate above a temperature of 35°C conductor temperature. If the insulation (this case, polyethylene) is exposed to temperatures higher than that, it is extremely rare. We know that the Arrhenius relationship cannot be properly applied; therefore, how are we to develop a proper approach for trying to predict cable life after 40 years?

The answer is that we really don't have a good method yet. Empirically accelerating conventional factors such as electrical stress or frequency (i.e., analogous to dose-rate) does not give reliable projections. We are seeking to develop a reliable method; one that would show some possible promise, involves comparing field-aged cables with accelerated-aged cables prepared in the laboratory. The third Figure below (RP-1357-01, to be published in late 1982) shows cables recovered from a number of utilities (Florida Power Corporation, Duke Power, Memphis Gas Water and Light, Pacific Gas and Electric, Florida Power and Light and Navaho Tribal Utilities) after 7-10 years of aging in the field.

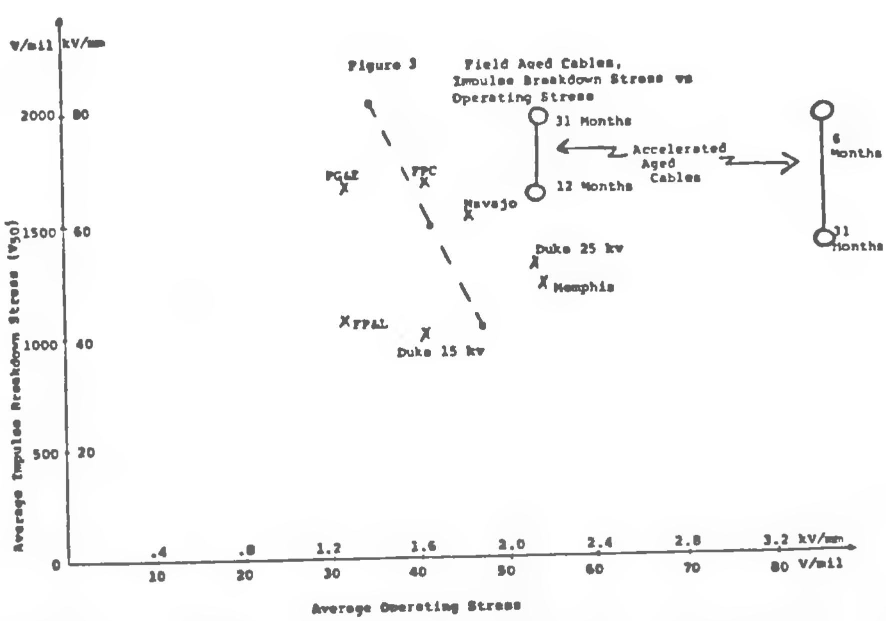

Figure 3 Field Aged Cables, Impulse Breakdown Stress vs Operating Stress

The average operating stress information shown in the Figure was provided by the Utilities, and the breakdown strength on these cables was provided by the Contractor. (Electrical breakdown strength is the generally accepted criteria.) In very general terms, there appears to be a reduction in breakdown strength with operating stress. However, when we compare the effects with accelerated-aged cables at equal average operating stresses from the laboratory, the correlation is not too good. Obviously, this particular set of accelerated conditions does not correlate with the electrical breakdown strength. Another attempt to correlate the field-aged with laboratory-aged cables, involves comparing the electrical breakdown with the size of a particular defect called a "tree." Here there may appear to be a correlation, but nobody is satisfied with using a visual defect that might be inadvertantly overlooked as a correlation factor.

The overall point I am trying to make in presenting this cable work, is that we are not trying to employ a relationship known to be valid elsewhere, albeit with limitations; we are trying to develop an empirical relationship by observing the behavior of field-aged and accelerated-aged cables. Hopefully, at some point in time, we will locate a suitable accelerated aging environment.

I believe that there is an analogy in methodology to be employed in seeking methods to predict life estimates of equipment, whether the environment is nuclear or other. Prediction techniques are fraught with uncertainties. The transformer situation I have described shows how even a valid technique such as the Arrhenius relationship for transformers can be inadvertantly misused when new environmental conditions are imposed. Comparing properties of equipment aged under different conditions (e.g., laboratory and Containment) may be a suitable method for consideration for nuclear equipment.

Finally, I would like to say we have had a greater degree of success by comparing <u>materials</u> properties of field-aged and laboratory-aged cables, using novel advanced analytical techniques, but it would be too involved to review that aspect in the short time allotted to me. Thank you.

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3. Electric Power Research Institute Project RP-1289-02 "Basic Research on Transformer Aging," Westinghouse Electric Corporation, to be published (September, 1982)

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TECHNICAL SESSION IV

Panel Discussion

PANEL DISCUSSION

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The panel discussion opened with the chairman proposing that research programs in the following areas be considered:

(1) the feasibility of using LER information to predict and correct major failures before they occur.
(2) the use of in-service maintenance procedures to monitor for failure trends.
(3) the feasibility of utilizing existing data from DOD, NASA, and other agencies.
(4) the reliability of the national grid over the years.
(5) aging problems unique to "mothballed" plants.
(6) the feasibility of a "lead" plant concept.

Several questions for discussion were raised from the panel:

(1) What safety-related components are most susceptible to age-related degradation?
(2) What aging mechanisms are of most concern?
(3) What has been learned from experience?
(4) What can be done to avoid or minimize age-related degradation?
(5) What research programs are needed and who should perform them?
(6) Can aging effects be simulated?
(7) What are the priorities?

Among the points presented by the panel members during their discussions were the following:

- There is a need to look at the whole system to determine how much component degradation the system can tolerate.
- There is a need for benchmarks for identifying when age-related degradation starts.
- A systems analysis can be used to narrow the list of components and parameters of concern, but, ultimately, the problem of age-related degradation will have to be addressed by qualification of components and by developing valid procedures for that qualification.
- A look at the overall system and a matrix of aging mechanisms and damage potential are needed to determine what research programs should be undertaken.

- The problems of software as related to aging must be considered.
- It was suggested that a team be formed with people of different perspectives for the purpose of making predictions relative to age-related degradation and to learn from those predictions.
- The question was raised as to whether much additional testing is needed and whether synergistic effects are of major concern.
- The efforts of the NRC and industry should be coordinated.

The panel discussion and the workshop closed with the chairman emphasizing that the NRC needs industry's cooperation and suggestions.

APPENDIX A

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APPENDIX B

Workshop Program

PROGRAM

Nuclear Power Plant Aging

Workshop
August 4 and 5, 1982

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

PROGRAM

WEDNESDAY, AUGUST 4, 1982

Registration		7:30 am-8:15 am
Introduction	S. K. Aggarwal, Program Manager	8:30 am
Welcoming Address	R. B. Minogue, Director Office of Nuclear Regulatory Research	
Keynote Address	H. R. Denton, Director Office of Nuclear Reactor Regulation	

TECHNICAL SESSION I. 9:00 am-12:00 noon

1. The Effects of Aging on the Performance of Safety-Related Equipment.
   J. F. Gleason, Wyle Laboratories

2. Accelerated Aging Methods of Electrical Equipments at Electricite de France.
   J. Roubault, Electricite de France

3. Equipment Aging Limitations: Key to Research Needs.
   S. P. Carfagno, Franklin Research Center

4. Methodology for Estimating Remaining Life of Components using Multi-factor Accelerated Life Tests.
   R. E. Thomas, G. B. Gaines, M. M. Epstein, Battelle Columbus Laboratories

5. Time Related Degradation, A Key Issue in Nuclear Plant Safety Evaluations.
   L. L. Bonzon, Sandia National Laboratories

6. Condition Monitoring.
   J. W. Wanless, NUS Corporation

TECHNICAL SESSION II. 1:00 pm-5:00 pm

1. Functional Qualification of Mechanical Components and Considerations for Aging Problems.
   E. J. Brown, U. S. Nuclear Regulatory Commission

2. Operation Testing and Maintenance Approach to Aging Problems in Mechanical Components.
   D. Beatty, Burns and Roe

3. The Degradation of Steam Generator Tubing and Components by Operation of Pressurized Water Reactors.
   J. R. Weeks, C. J. Czajkowski, Brookhaven National Laboratory

4. Stress Corrosion Cracking in Boiling Water Reactor (BWR) Piping.
   E. D. Eason, Failure Analysis Associates

5. Overview of Synergistic Aging Effects.
   W. Steigelmann, M. Farber, Synergic Resources Corporation

6. A State Regulator Views the Aging of Nuclear Power Plants.
   J. B. Keating, New York State Public Service Commission

THURSDAY, AUGUST 5, 1982

TECHNICAL SESSION III. 8:30 am-12:30 pm and 1:30 pm-2:45 pm

ORAL PRESENTATIONS

Short presentations will be made by authors who have submitted abstracts which could not be accommodated in Technical Sessions I and II. Also, workshop attendees who may not have technical papers but who wish to contribute their ideas or experiences may make oral presentations upon request to the session chairman.

TECHNICAL SESSION IV. 3:00 pm-5:00 pm

PANEL DISCUSSION

Panel members will discuss and analyze the presentations made during the workshop and make recommendations for future research. Audience participation is encouraged.

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