NUREG/CR-5700
(ORNL/TM-11806)

Aging Assessment of Reactor Instrumentation and Protection System Components

Aging-Related Operating Experiences

Prepared by
A. C. Gehl, E. W. Hagen

Oak Ridge National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555

2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082

3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grant publications, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5700
ORNL/TM-11806

Aging Assessment of Reactor Instrumentation and Protection System Components

Aging-Related Operating Experiences

Manuscript Completed: July 1992
Date Published: July 1992

Prepared by
A. C. Gehl, E. W. Hagen

Oak Ridge National Laboratory
Managed by Martin Marietta Energy Systems, Inc.

Oak Ridge National Laboratory
Oak Ridge, TN 37831-6285

Prepared for
Division of Engineering
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN B0828
Under Contract No. DE-AC05-84OR21400

ABSTRACT

A study of the aging-related operating experiences throughout a five-year period (1984–1988) of six generic instrumentation modules (indicators, sensors, controllers, transmitters, annunciators, and recorders) was performed as a part of the Nuclear Plant Aging Research Program. The effects of aging from operational and environmental stressors were characterized from results depicted in Licensee Event Reports (LERs). The data are graphically displayed as frequency of events per plant year for operating plant ages from 1 to 28 years to determine aging-related failure trend patterns. Three main conclusions were drawn from this study.

1. Instrumentation and control (I&C) modules make a modest contribution to safety-significant events.

   - 17% of LERs issued during 1984–1988 dealt with malfunctions of the six I&C modules studied.

   - 28% of the LERs dealing with these I&C module malfunctions were aging related (other studies show a range 25–50%).

2. Of the six modules studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures.

3. Infant mortality appears to be the dominant aging-related failure mode for most I&C module categories (with the exception of annunciators and recorders, which appear to fail randomly).

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

LIST OF FIGURES

LIST OF FIGURES (continued)

LIST OF TABLES

LIST OF ACRONYMS

AEOD	Office for Analysis and Evaluation of Operational Data
ANSI	American National Standards Institute
BWR	boiling water reactor
CCF	common cause failure
DBA	design basis accident
EEI	Edison Electric Institute
EFIC	emergency feedwater indication and control
EFS	emergency feedwater system
EMI	electromagnetic interference
EPRI	Electric Power Research Institute
ESF	engineered safety feature
HVAC	heating, ventilating, and air conditioning
I&C	instrumentation and control
INPO	Institute of Nuclear Power Operations
ISA	Instrument Society of America
IPRDS	In-Plant Reliability Data System
LCO	limiting condition for operation
LER	Licensee Event Report
LWR	light water reactor
NPAR	Nuclear Plant Aging Research
NPE	Nuclear Plant Experience
NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission
PM	preventive maintenance
PWR	pressurized water reactor
QA	quality assurance
R&D	research and development
RFI	radio-frequency interference
RPS	reactor protection system
RTD	resistance temperature device
SAR	Safety Analysis Report
SCSS	Sequence Coding and Search System
SFRCS	steam and feedwater rupture control system

ACKNOWLEDGMENTS

The authors gratefully acknowledge the continuing support of the U.S. Nuclear Regulatory Commission Nuclear Plant Aging Research Program manager, J. P. Vora, and the counsel of the project manager, W. S. Farmer, in planning and implementing this study. The assessment was carried out under the general guidance of D. M. Eissenberg, Oak Ridge National Laboratory Nuclear Plant Aging Research Program manager; the authors are grateful for his encouragement and advice.

This report could not have been published without the encouragement and substantial efforts of R. A. Kisner (editorial consultation) and R. C. Kryter and J. A. Thie (technical reviewers).

We also thank Karen Ratliff, who patiently worked through many revisions in preparing this report for publication; Robin O'Hatnick for taking care of the endless details; and reports analyst Jackie Miller, and electronic publisher, Karen Vogel, for their tremendous effort and patience despite the many late changes.

1. INTRODUCTION

The United States has over 100 reactors in commercial operation, and a few of these have been operating for over 20 years. As the population of light water reactors (LWRs) has advanced in age, the need for a research program that would provide a systematic assessment of the effects of plant aging on safety has been recognized. Consequently, the director of the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission (NRC), in his comments on the Long-Range Research Plan, identified a need for a research program to investigate the safety aspects of aging processes affecting components and systems in commercial nuclear power plants.[1] This program is called the Nuclear Plant Aging Research (NPAR) Program and was discussed at length at the July 1985 International Conference on Nuclear Plant Aging, Availability Factor, and Reliability Analysis.[2]

The NPAR Program was developed to provide a systematic study of how aging affects the safety of nuclear plants currently in operation. This program provides a comprehensive effort to (1) learn from operating experience and expert opinion, (2) identify failures due to age degradation, (3) foresee or predict safety problems resulting from aging-related degradation, and (4) develop recommendations for surveillance and maintenance procedures that will alleviate aging concerns.[3]

Many of these issues have been and are being addressed by the nuclear industry through research, improved designs, standards development, and, especially, improved operational and maintenance practices. However, significant questions still remain because of the variety of components in a commercial power reactor, the complexity of the aging process, and the limited experience with prolonged operation of these power plants. Nevertheless, aging and degradation of plant safety systems and components will continue, and currently unrecognized degradation effects are likely to emerge as the U.S. light water reactor population ages. Collection and evaluation of operating experience data are necessary to study the effects of aging and degradation on the safety of operating nuclear power plants during their normal design life and for any extended life option.

This study examined the effects of aging on equipment performance and normal service life and concentrated on six specific instrumentation categories: indicators, sensors, controllers, transmitters, annunciators, and recorders. These categories were selected because of their importance in the operations of safety-related instrumentation and control (I&C) systems and because they have not been reviewed previously by the NPAR program.

1.1 BACKGROUND

As nuclear power plants age, it becomes increasingly important for the nuclear community to understand and be able to manage aging phenomena. Aging affects reactor structures, systems, modules, and components to varying degrees. The issue of *aging* safety-related control systems in a world of increased performance demands is a relatively new one. For the NPAR Program, *aging* refers to the cumulative degradation of a system, component, or structure that occurs with time and, if left unchecked, can lead to an impairment of continued safe operation of a nuclear power plant. Measures must be taken to ensure that aging-related degradation does not reduce the operational readiness of a plant's safety systems and components and does not result in common-mode failures of redundant, safety-related equipment, thus reducing defense in depth. It is also necessary to ensure that aging does not lead to failure of equipment in a manner that would directly cause an accident or severe transient.

As the average age of existing plants increases and life extension actions become imminent for the older ones, the subject of aging effects becomes a concern. The need for identifying and resolving technical safety issues related to aging are discussed thoroughly in the NPAR program plan.[1] Many NPAR reports have been issued and are mostly concerned with large reactor components and equipment that were designed for service beyond the licensed life of the plant (40 years). However, the useful service life for many individual I&C modules is shorter than 40 years; therefore, these smaller and less expensive plant components have to be maintained, refurbished, and replaced relatively frequently. The importance of this study is evident from the ubiquity of these modules in a typical nuclear plant design as well as from the need for keeping these modules operationally ready for safe operation throughout the life of a plant. Thus, it is essential to have a comprehensive study of the aging of I&C modules and of the potential for degradation due to operational as well as environmental stressors.

The purpose of any instrument system is to receive as input one or more physical quantities and to produce as output one or more physical quantities that are in more useful forms than the inputs. The outputs may represent measures of the input quantities, as in a measuring system, or they may represent controls on the operation of some equipment or process. More than 10,000 I&C modules are used in a typical nuclear power plant to perform control, protection, monitoring, and service functions. These modules are located in both hostile and benign environments in primary systems and in the balance-of-plant and service systems. The 109 operating units in the United States have been designed and built over a period of 28 years by many architectural and engineering firms that specified the equipment/modules used. At only a few plant sites do multiple units utilize interchangeable replacement modules. Aside from this small set, the numbers of manufacturers and models present for any one module type render impractical failure modes and effects analyses that would lead to a determination of module aging. Because nuclear power plants have an abundance of I&C systems, the scope of this evaluation has been limited to those I&C modules that are vital to safe plant operation. Examples in the assessment have been limited to six classes of modules that are part of reactor safety-related systems, with some overlap into the reactor control and process control systems as appropriate.

Aging stressors or mechanisms can be cyclic (e.g., caused by repeated demand) or continuously acting (e.g., caused by the operational environment). It is reasonable to assume that the I&C module failure probability resulting from such stressor-induced degradation will monotonically increase with the time of exposure to a stressing agent or mechanism until the module is refurbished, repaired, or replaced.

I&C modules are unlike other components or equipment in a nuclear power plant because their modularity enhances the performance of maintenance and repair and makes possible their change-out even during plant operation. This makes the aging assessment of I&C modules inherently different from and more complicated than that of other mechanical and electrical modules in a power plant for several reasons.

1. The same kinds of modules are exposed to different environments and are used in a variety of ways (e.g., transmitters are used both inside the containment and in the balance of plant, and indicators can be positioned locally or mounted in control room panels).

2. Interaction of modules is inherent in the system design (i.e., modules are interconnected and feed signals to each other and also can be cascaded within a control loop).

3. Simultaneous actions of normal and accelerated aging stressors are quite common during normal plant operation (e.g., elevated temperature, radiation, and process transients).

1.2 OBJECTIVES OF THIS STUDY

In accordance with the NPAR program philosophy, this study has three objectives.

1. Identify aging stressors and the likely effects of these stressors on I&C modules.

2. Identify, by examination of operational history, those safety-related I&C modules most affected by aging.

3. Identify, where possible, methods of inspection, surveillance, and monitoring that will ensure timely detection of significant aging effects prior to loss of safety function.

These goals require that a number of subtasks be accomplished. Also, an analysis of the operational, environmental, and accident-related stressors and a detailed review of the operating experience of these modules at nuclear plants should be included.

1.3 SURVEY OF OTHER AGING STUDIES

Several studies of aging effects have been made on the performance of nuclear plant safety-related equipment.[3] Many of these reports mention I&C equipment, and fewer include referenceable data. This report combines much of the extractable I&C data by providing a single reference for this multifaceted information from other NPAR reports, Electric Power Research Institute (EPRI) documents, and papers from conference proceedings. This information is incorporated in other sections of this report, and the sources are listed in Appendix A. Literature from outside the United States was surveyed briefly but with little success in adding to the base of applicable information for this study.

1.4 BOUNDING THE PROBLEM

Philosophically and in the broadest context, aging begins from the day an instrumentation module is fabricated, even while in storage. However, for this study, aging is related to useful service life and achievement of design performance (i.e., the ability of a module to function as intended, given manufacturers' prescribed preventive maintenance, until replacement becomes necessary). Replacement is the key to aging, whether needed because of wear out (as is most common for mechanical components) or some other reason such as catastrophic failure (as is more common for electronic/electrical components).

Mean time between similar *failures* for a particular type of module, trend data for repairs to the module, and required frequency of testing for the module help establish the degradation with time (aging) for the module. Preventive maintenance (PM) is often applied to extend the period of degradation before failure occurs or replacement is required. Each such PM servicing should not, however, be counted as another aging failure event for that module. If this distinction is not made, a large data file can be obtained from repeated routine servicing for a module, from which erroneous conclusions might easily be drawn. Therefore, depending on the reporting methods, either high or low numbers of reported failures might be termed *aging related*. Both have appeared in the literature.

A basis founded on power plant practices was employed in selecting the boundaries for I&C modules used as examples in the study. This basis is primarily the set of modules for which I&C departments at the plants are responsible and that are utilized in systems considered *important to safety*. Their nature is primarily *electronic*, as opposed to *power* electrical systems, which are not within the scope of this study.

A common technology of electronics is used to define the bounds of interest in this study rather than a physical boundary or a functional system boundary. Hence, almost identical I&C modules may be found in a variety of plant systems. From the standpoint of aging, the nature and environment for the categories are more significant than the function of a module in a system. (The latter becomes significant when the interest is in the consequences of a failure.) Hence, it is logical to have groupings such as indicators, sensors, controllers, transmitters, annunciators, and recorders that are used plantwide.

1.5 ORGANIZATION OF THE REPORT

I&C components and modules not previously studied by the NPAR Program were selected to assess their aging status. A review of historical data was made utilizing national databases as sources for operational failure experiences. The open literature was also surveyed to a limited degree. Some embedded information in previous NPAR reports was extracted and collated. However, qualification data were not always sufficient to completely address normal I&C module aging. Therefore, in the data retrieval scheme, besides the terminology used for *aging*, a prime keyword was *replacement*.

Section 2 outlines the approach taken to sort through a large quantity of operating experience data to retrieve selectively that judged to be aging related. The study discusses the effects of stressors in Sect. 3 and evaluates them as factors that accelerate aging. The categories of I&C modules are illustrated in Sect. 4, and the operating experiences reported are analyzed in Sect. 5. Problems affecting the instrumentation are then characterized by the type of conditions contributing to aging. Some aging-related events are examined in Sect. 6, and generalized findings are presented in Sect. 7. The conclusions, observations, and recommendations from this study are contained in Sect. 8, with detailed supportive material provided in the appendixes.

2. STUDY APPROACH

The approach of the study was to examine experiences within the nuclear industry that have affected the aging of I&C modules. This was accomplished by utilizing

1. nationwide industry databases [Licensee Event Report (LER), Nuclear Plant Reliability Data Systems (NPRDS), and Nuclear Plant Experience (NPE)];
2. NRC Daily Headquarters Reports, NRC Daily Operating Events Reports, and NRC Regional Inspection Reports; and
3. published literature for related investigations of instrumentation aging.

This approach of relying on experiences to date was believed to be appropriate because of the diverse nature of I&C modules and systems. It was preferred over other investigatory approaches such as failure mode analyses or probabilistic risk assessment calculations (both of which would have to be highly module specific and would have to be repeated for numerous instruments). Early in the project, emphasis was placed on trying to isolate specific I&C modules having high failure rates that were not already the concern of other NPAR research efforts. However, the databases available to us could not support the identification of specific modules. Therefore, it was concluded that although a general *data inductive* approach to analysis may be satisfactory for studying failure data for large mechanical components such as pumps and valves, it was not necessarily applicable to the study of electronic components, because electronic modules seldom exhibit signs of degradation either as visible or performance indications and because failures are usually rapid and terminal. Therefore, we employed the *hypothetical deductive* approach to analysis, where a failure mechanism thought to be credible is assumed, and then the available data are analyzed to determine the validity of the assumption. The first method may be thought of as sorting the data into piles, with the size of the pile driving the conclusion. In the second method, after postulating a cause, the data are searched for supporting evidence. This approach is likely to be more productive when the quantity of data is not very large because the trend of aging and/or the frequency of required maintenance for components then becomes more apparent.

The information was used to create a computerized database for analysis. Mechanisms of failure were not determined during this analysis; however, several analyses were performed using the failure-category data. Selected groupings of the data were examined to identify the systems in which the module having the aging-related failure was used. This database was designed as a tool to help manage the information that would need to be accumulated. It was not intended as a product of the study. The database was accessible via programmed searches to query the data for various conclusions regarding failures and postulated events.

2.1 INVESTIGATIVE STRATEGY

The characterization of aging in a system in a nuclear power plant is recognized as being a complex task. This is especially true when the system is composed of several interacting parts such as modules in I&C systems (see Fig. 2.1). For this task, modules were treated generically, with the goal being to determine how each category of modules

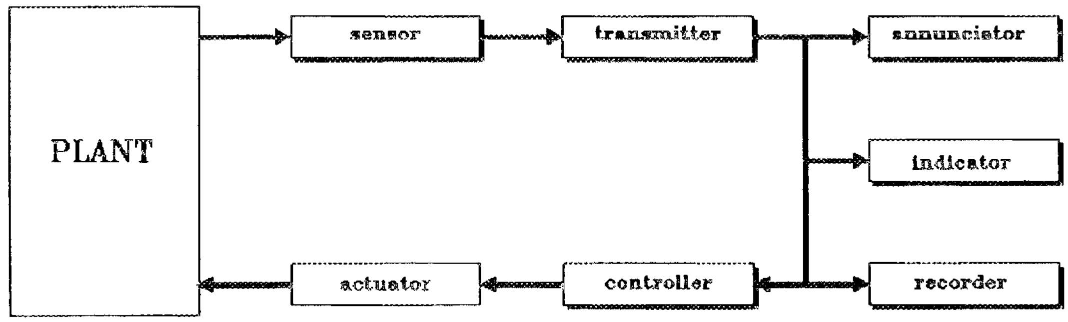


Fig. 2.1. A typical configuration for instrumentation and control safety-related systems in a nuclear power plant.

typically ages. Thus, an overview of the field could be obtained so that zeroing in on specific places in the field would then be more rewarding. Results for the generic analysis were also compared to operating experiences. Effectiveness of surveillance, testing, maintenance, and incipient failure/degradation monitoring was then assessed.

The requirements for generating an LER are spelled out in the Code of Federal Regulations (CFR), 10 CFR, Pt. 50.73 (a) (2) (iv), "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)"; and 10 CFR, Pt. 50.73 (a) (2) (vii) (A), "any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a system designed to shut down the reactor and maintain it in a safe shutdown condition." Therefore, a reported event may be classified easily as safety related or not. Equipment identified in the LER is thus associated with a safety-related event. This greatly facilitates the screening of safety-related experiences from those contributed to reliability databases in the balance-of-plant systems. However, subjective judgment is often required to separate aging-related data from failure data useful for reliability studies. Careful evaluation is required not only for LERs, but for all current national databases.

Information associated with aging-related failures of I&C-category modules was obtained via structured searches of the LER national database. The Sequence Coding and Search System (SCSS) database of LER information, operated by Oak Ridge National Laboratory for the NRC Office for Analysis and Evaluation of Operational Data (AEOD), formed the basis for a compiled aging database for this task. The SCSS database was developed to allow information reported in the LERs and accompanying descriptive text to be encoded such that detailed, comprehensive information regarding component and system failures, personnel errors, and unit effects and their interactions could be retrieved. In addition to its ready availability and accessibility, many features of this database made its use attractive for evaluating instrumentation failures. The intent of the review was to focus on only aging of the six categories of I&C modules. However, because of different levels of specificity used by utilities in preparing their LERs, the review took place at a level of relatively broad functional areas. This was in keeping with the recognition that the characterization of aging is a complex task.

A search of operating experiences was made to cover the 5-year period 1984–1988 for each of the six I&C module categories. This period encompasses 471 reactor years of operation, during which the median plant age was 12 years. This time span was selected because reporting requirements were changed in 1984 and a continuity of reporting was deemed necessary for the analysis. Each abstract was reviewed and analyzed to obtain insights into module aging failures, and then those aging events were placed into an electronic database created especially for this study. Excerpts from the NRC Daily Headquarters Reports, NRC Daily Operating Events Reports, and the NRC Regional Inspection Reports were added. The information is retrievable by structured searches such as for categories, plants, involved systems, methods of detection, and reported causes (see Appendix D). A detailed analysis was then performed on the retained data. The data were compiled first by system affected for each of the six categories. Then, the numbers of module aging failures were matched to the plant age. The data were next regrouped by event year to determine the relative percentages of module failures and to determine the trending. The study then analyzed the trend of module failures by plotting total module failures per plant year vs the year in which failures occurred. Data from the NPRDS and NPE databases were retrieved and the results were compared quantitatively to those obtained from the LER database (see Figs. A.1, A.2, and A.3 in Appendix A). An objective was to determine the advantage of using one database for the analysis of generic problems due to aging stressors. The data were used to rank the effects of the various aging stressors and to observe how these were manifested. Some examples from operating experience reports were used to illustrate the latter. The analysis of the data provided several insights into the effect of aging failures. This strategy provided the means for evaluating the operating experiences and for developing a comprehensive aging assessment for six I&C module categories.

2.2 OTHER INVESTIGATIONS

In a prior investigation, assessments were made of the relative occurrences of aging-related failures vs other failures.[4] Because that study focused on the reactor protection system, there may be general applicability of the results to the I&C modules in this study. A quantity, aging fraction, was defined for a particular piece of equipment as

$$\text{aging fraction} = (\text{failures due to aging})/(\text{total failures}) .$$

The NPRDS was used as a source of data where, with failures divided into five categories (design, aging, testing, human, and other), it was found that different types of I&C equipment had similar aging fractions ranging between 0.2 and 0.4. Another analysis of LERs produced some corroboration of these results,[5] despite judgments about what constitutes aging effects being somewhat different from those in the NPRDS study. Table 2.1 lists the aging fractions for various components described in the 6764 total I&C failures found.

An NPAR study dealing with LWR safety systems noted that the aging fractions for solid state instrumentation components did not vary greatly from system to system.[6] The study noted also that the effect of aging on solid state component failures was minimal relative to other components.

A study that is critical of 20-year-old electronic technology still in use at plants points out that a plant with about 10^3 modules, whose individual failure rates are 10^{-5} to 10^{-6}/h, can be expected to experience a problem every 100 to 1000 hours.[7] Because

Table 2.1. Aging fractions for various instrumentation and control components[a]

Component	Number of faults[b]	Aging fraction
Amplifier/buffer/isolation amplifier	138	0.49
Cable/receptacle/junction-box/terminal	226	0.21
Controller	72	0.21
Comparator/bistable	327	0.58
Converter/conditioner	98	0.64
General Switch	2004	0.75
Hand Switch	35	0.23
Computation Mode	224	0.47
Indicator/meter/annunciator	303	0.40
Computer	45	0.31
Limit Switch	70	0.37
Monitor	979	0.43
Power Supply	227	0.30
Recorder	117	0.53
Relay/solenoid	275	0.37
Sensor	424	0.40
Timer	27	0.41
Transformer	11	0.82
Transmitter	844	0.56
Other/unspecified	318	0.20

[a]*Source*: Based on data from *Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants*, NUREG/CR-1740, U.S. Nuclear Regulatory Commission, July 1984.

[b]Numbers of faults reported in the database for the individual components. The product of these numbers and the aging fractions are the numbers of faults categorized as time related.

individual components within modules are commercial grade, the ability to achieve an order of magnitude of better reliability by judicious use of military grade components is pointed out.

The method by which this enhanced reliability would be achieved is through design improvements in two areas:

1. change-out of selected specific critical components in the equipment modules and
2. redesign of modules to provide enhanced reliability.

In both cases, military-specification components are proposed, especially for rotary switches, operational amplifiers, trimmer resistors, and field-effect transistors. For redesign, other improvements are also suggested (e.g., dual power supplies feeding through diodes instead of individual power supplies).

A study by Idaho National Engineering Laboratory on the effects of aging on the RPS found that a good maintenance program "almost makes aging a nonproblem on redundant systems such as RPS, because the periodic rejuvenation does not allow the system to grow old.[8] The study involved a detailed investigation of generic I&C channels of the Reactor Trip System and Engineered Safety Feature Actuating System by using many of the same sources employed in this study. However, this report concentrates only on the analysis of the six selected I&C module categories rather than on an overall system level.

A recent EPRI study of LERs indicated that ~30% of the reported events during 1980–1982 were attributable to instrument and electrical problems.[9] The report also indicated that close to 70% of the LERs in 1987–1988 were attributable to instrument and electrical problems. Several factors may account for the contrasting results of these two findings of the EPRI study as well as for differences with the findings of this current report: (1) LER reporting requirements were changed substantially in 1984, (2) electrical equipment other than I&C modules was included in the EPRI study, (3) the EPRI study included indirect (secondary) failures of I&C and electrical equipment as well as direct failures, and (4) the two studies used different keywords in searching through databases.

2.3 CONSOLIDATED DATABASE

Although a steady stream of failure data is being generated in the nuclear industry, little can be positively identified with *aging* as defined in this task. Failure data for large mechanical components are more numerous than for electronic/electrical components. This is due in part to degradation in mechanical components being more easily recognized, so when attention is required, another data entry is made. This can (and often does) happen to the same component many times. In contrast, many electronic components give no evidence of gradual degradation, so they receive little periodic maintenance. When they finally give evidence of problems, the failure is often catastrophic, and hence only one data entry is produced. Because of these basic differences, several documents reviewed have stated that electrical components examined were judged to show little or no degradation due to aging, but this may be more a matter of perception than of fact. A parallel (hence confusing) occurrence is that analog modules are gradually being replaced throughout the industry with digital electronics (i.e., the technology employed is continually evolving).

A search of the LER database for the 5-year period 1984–88 showed a total of 13,726 reportable events. Structured searches for the six I&C module categories retrieved 2,276 references, representing 17% of the 13,728 LERs reported during this time. Of the 2,276 LERs dealing with the six I&C module categories, only 628 events were judged to be aging related (i.e., 28% of the LERs involving I&C modules were aging related). The number of aging-related failures is small and amounts to only 4.6% of the 13,726 LERs produced from 1984 through 1988. The six I&C module categories are ranked in Table 2.2 by the frequency of occurrence of their aging-related failures. These low figures appear to show that I&C modules make only a modest contribution to safety-significant events; annunciators and recorders, in particular, show negligible problems.

The search of the NPRDS database retrieved additional data supporting the trends set in Table 2.2 (explained further in Sect. 5). The NPE database was likewise surveyed and was found to contain essentially the same aging information as in the LER database (see Appendix A).

2.4 FOREIGN EXPERIENCE

Some foreign literature, primarily in two collections of papers on aging at international meetings, was examined in this study.[10–12] Essentially all countries in the commercial nuclear power community are addressing aging issues, but relatively few papers address I&C aging issues specifically. However, some useful information was extracted from the referenced meetings, such as that described below.

Belgium uses a rigorous approach to the accelerated aging of electronic equipment.[13] It is based on a formula from Military Standard Specification MIL-HDBK 217C, "Reliability Prediction of Electronic Equipment" (1979), combined with data on individual components that make up complex electronic units.

Japan's practice of formalized inspection of instrumentation during refueling outages has been adopted as a result of regulation.[14] Possible merits of this approach, supplementing surveillance testing during operation, are discussed in further detail in Sect. 7.8.

France has an aging program that starts at the time of plant construction.[15] This approach is in contrast to starting such a program midway through plant life when issues of life extension begin to arise.

Table 2.2. Safety-related findings from Licensee Event Reports (LERs) (1984–1988)

Instrumentation and control module category	Number of LERs retrieved for modules	Number of aging-related LERs	Percentage of aging-related LERs ÷ aging-related total (628)	Percentage of aging-related LERs ÷ total LERs generated (13,726)
Indicator	997	220	35	1.6
Sensor	424	199	32	1.4
Controller	397	105	17	0.8
Transmitter	296	79	12	0.6
Annunciator	101	17	3	0.1
Recorder	61	8	1	0.06
Total	2276	628	100	4.6

3. AGING STRESSORS

An important concern of the nuclear industry is the impact of equipment aging on the capability of plant safety systems to mitigate the consequences of accident and abnormal transients. The effects of aging-induced degradation on the ability of the plant to achieve a safe shutdown includes an evaluation of stressors acting on the equipment as potential causes of aging-related failure. Therefore, consideration must be given to all significant types of degradation that might affect the functional capability of the equipment.

This section examines various stressors that are imposed on six categories of I&C modules during both normal and transient situations. It discusses the operational and environmental stressors that can degrade the mechanical, electrical, and chemical properties of I&C modules; and it includes system- and component-level stresses such as those introduced by testing, human factors, environmental parameters, and their synergistic effects.

Aging degradation occurs when a material is subjected or exposed to a stress condition for a period of time. A stressor is defined as an element that acts upon an I&C module in such a manner as to produce an irreversible physical change. This change may be regarded as the root cause of the aging problem, whereas the stressor is that which induced the physical change. Typical aging mechanisms that cause a material's mechanical strength or physical properties to degrade include fatigue stress cycles (thermal, mechanical, or electrical), wear, corrosion, erosion, embrittlement, diffusion, chemical reaction, cracking or fracture, and surface contamination. Each mechanism can occur in various materials when they are exposed to particular operating and environmental stressors. Abnormal conditions accelerate the aging process, thus weakening the material faster than normal.

The effects of age are particularly important when sensitive modules are operated in environments of elevated radiation, temperature, humidity, vibration, and other conditions that may degrade performance. These processes are reasonably well understood when one type of material is exposed to one kind of stress. However, with the complexities of composite materials (which is the case for most electronic components) and the synergistic effects of several stressors, these processes become difficult to understand. Extensive laboratory testing and material analyses are necessary to characterize these complex phenomena. Also, because aging is by definition a time-dependent process, considerable time is necessary to understand completely the true aging mechanisms operative in a plant environment.

Aging-related failures are difficult to distinguish from random failures solely on the basis of the failure description or reported failure cause provided in most operating experience summaries or abstracts. In some cases, a mechanism identified as having caused a failure could be considered either aging related or random, depending upon the circumstances under which the failure occurred. Environmental effects, normal wear of component parts, erosion, corrosion, cyclic fatigue, etc., affect the component in a continuous fashion with rather slow aging rates. These stressors will commonly act in unison, though one may predominate, making it difficult to recognize the true aging stressor as the cause of the failure. Hence, many aging-related failures are misclassified as being random, and replacement of the failed module is the corrective action taken. Such actions are easier for I&C modules than for other equipment studied in the NPAR program because most I&C modules are accessible and designed to permit easy replacement.

The electronic components in I&C modules often include numerous resistors, capacitors, diodes, and integrated circuits that are used for signal conversion, signal-conditioning, and linearization of the module's output. In some, resistors and/or capacitors are used to maintain the linearity of the output and to set the module *zero* and *span*. These components are adversely affected by long-term exposure to high temperature and humidity, by radiation, and by fluctuations or step changes in the power supply voltage. Any change in the value of electronic components such as the resistors or capacitors can cause calibration shifts and response time changes and also affect the linearity of the output signal. Calibration shifts can occur from deterioration such as loosening or wear of mechanical constituents or aging of electronic parts. Adverse effects from such exposures gradually become apparent over time, and more than one stressor can be responsible for the same ultimate physical degradation.

I&C modules in nuclear power plants are subject to stresses from their environment, their application, the process being monitored, and the services supplied to them such as air and electricity. Under normal plant conditions, the major environmental stressors are radiation, temperature, humidity (moisture), and vibration, all of which contribute to normal wear of components and affect the module in a continuous fashion but with a relatively low aging rate. Module aging can be accelerated by some transient situation such as water hammer, shock, electromagnetic interference, or improper maintenance. Another form of aging-related stressing arises from nonquantifiable forces such as dormancy, human errors, and technological obsolescence. Any one or more of the stressors can severely curtail the service life of I&C modules.

In this aging study, it was found to be more useful and practical to concentrate on classification by stressors rather than by ultimate physical effects. The principal stressors for I&C components considered in this study are listed in Table 3.1 and discussed in Sects. 3.1 through 3.10.

3.1 RADIATION

Ionizing radiation is a dominant stressor for aging studies in nuclear facilities, and its presence makes a nuclear power plant unique in the process and electricity generating industries. Radioactivity is the process by which certain types of unstable atoms or nuclides decay spontaneously until a more stable state is reached. Radiation is the ejected nuclear material (alpha or beta particles or neutrons) or energy (gamma rays) released during the decay process. When alpha, beta, or gamma radiation impinges on materials, the energy is absorbed and may cause structural and chemical changes and damage the material.

Materials such as (1) organic fluids, (2) elastomers, and (3) plastics are especially susceptible to radiation damage. In a nuclear plant, organic liquids can be found in many different forms such as greases, lubricants, coolants or heat-transfer media, and neutron moderating materials. Two of the most striking effects occurring in an organic liquid as the result of exposure to radiation are gas evolution and changes in viscosity. The viscosity usually increases upon continual exposure to radiation until the liquid polymerizes into a solid form.

Table 3.1. Principal stressors

Radiation
Temperature
Moisture/humidity
Vibration and shock
Human interaction
Process pressure surge
Testing
Electromagnetic interference
Cycles of operation
Technological obsolescence

Elastomers are in widespread use in nuclear power plants—for example, as gaskets, flexible connecting tubes, hoses, and elasticlike electrical insulating material and for many allied functions that necessitate the use of a material that is elastic enough to be pressed to fit a given contour or that has a large degree of vibration resistance. Elastomers could be subjected to various degrees of radiation flux, depending upon their site of application in the power plant. Although elastomers are of a specialized nature, they are organic solids and, as such, often exhibit radiation damage at lower dosage levels than do the organic liquids because atoms of a solid have fewer degrees of freedom than those of a liquid.

Plastics also have many applications in nuclear power plants. For example, in that portion of an electrical system involving motors and selsyns, plastics are used as brush holders, grease seals, insulating tapes, spacers, slot insulation, shaft insulation, electrical-wire insulation, and end punchings. As with elastomers, plastics also exhibit radiation damage at lower dosage levels than do organic liquids.

Many components in electrical, electronic, and mechanical systems are susceptible to radiation damage because their modules use components fabricated from the above materials. Certain electronic materials, particularly semiconductors, can change their properties at very low neutron and gamma dosages. Relatively small changes in some electrical properties can render a module functionally useless.

The service life of each module can be estimated upon the premise that a system is no more stable under radiation than its most radiation-susceptible materials. Although radiation damage to organic-type materials is generally related to total dosage and not dosage rate, it is still entirely possible that the ionization produced in an electrical insulation system during irradiation will cause the resistivity of radiation-sensitive insulating materials to be changed in situ as a function of dosage rate. Therefore, it could be possible for a given electrical insulation material or system to receive the same total dosage under different fluxes or dosage rates and yet exhibit somewhat different radiation-induced changes.

3.2 TEMPERATURE

Perhaps the best understood and easiest aging phenomenon to analyze is that due to temperature. In fact, aging degradation of I&C equipment not placed near the reactor vessel is principally temperature dependent. The effects of temperature are often related to an Arrhenius model of aging. This has been applied to power plant equipment.[16] The essence of the model is that the log (service life) is proportional to 1/(absolute temperature). This relationship implies that the log of the ratio of service lives at two temperatures not too far apart will be approximately proportional to the temperature difference. The failure rate of instrumentation can thus be expected to increase rapidly as ambient temperature increases. When the temperature exceeds specified limitations, as set by the manufacturer or the plant Safety Analysis Reports/Technical Specifications, components in the I&C modules are stressed, aging is accelerated, and useful service life defined by the environmental qualification requirements for that module is decreased.

Concerns have been expressed about the performance of safety-related equipment when excessive temperatures are recorded in the containment and equipment rooms/cabinets, especially over prolonged periods. In particular, overheating of electronic components in safety-related I&C modules has raised questions about (1) decreased reliability due to increased failure rates of printed circuit cards and other heat-sensitive components and (2) the potential for common cause failure (CCF) of redundant safety-related instrumentation channels due to extended loss of normal cooling air flow to the cabinets in which the modules are located. A localized elevated temperature can also be generated by poor connections of terminals at the terminal block.

Failures of electronic components in safety-related instrument systems lead to malfunction of control systems, plant transients, inoperability of instrument channels in protection systems, and erroneous indications and alarms in the control room. These concerns prompted NRC to issue an Information Notice regarding the significant problem involving the loss of solid state instrumentation following a failure of control room cooling.[17] These concerns are generic to all operating nuclear plants that use solid state electronic components. Four events involving failures of solid state electronic components in safety-related instrumentation and control systems due to overheating are reviewed and evaluated in reference 18 (see also Sect. 6.1).

Information Notices were also issued alerting licensees to the potential problem resulting from operating a plant at temperatures beyond its analyzed basis[19] (see Sect. 6.2, Example 1) and to potential problems resulting from high-temperature environments in areas containing safety-related equipment.[6] Thermal aging of heat-sensitive components (e.g., cables, splices, terminations, and terminal boards) within the Series SMB Limitorque motor operator limit switch compartments were a concern to NRC inspectors during a 1986 inspection.[20] The compartment space heaters are normally energized during plant operation to reduce the relative humidity. However, the consequences of the increased temperature, which would shorten the qualified life of electrical components within the switch compartment, were not analyzed. Based on the length of time installed (14 years) and the original qualified life (40 years), the qualified lives of some Limitorque Motor operator components were reduced when the effect of the increased compartment temperature was taken into consideration.

Manufacturers design their products for a specified range of operating temperatures. When the extremities of this range are exceeded, stresses greater than normal (accelerated aging) are applied to the product. Effects of off-normal temperatures are much less evident for mechanical devices, which are often more robust than electronic devices.

However, for both, much information has been collected from the survival analyses of equipment and modules removed from Three Mile Island Plant, Unit 2, following the 1979 accident. (See the series of GEND-INF reports prepared by EG&G Idaho for the U.S. Department of Energy.) Excessive temperature can also act in concert with other stressors. For example, detrimental effects of humidity are increased at higher temperatures because of increased diffusion rate.

3.3 MOISTURE/HUMIDITY

I&C modules inside containment are occasionally exposed to moisture from leaking valve seals or broken water or steam lines. These I&C modules are sealed to conform with environmental qualification requirements. However, time and temperature eventually degrade the effectiveness of the seal, and maintenance sometimes damages the seals as well, thus providing a pathway for moisture to enter the module. Moisture can also be elsewhere in the system and enter a module by way of electrical conduits.

Moisture has been found in modules, equipment racks, and instrument cabinets. Some events and NRC concerns are given in two Information Notices.[21,22] For example, Supplement Two to Information Notice 86-106 describes actuation of a carbon dioxide fire suppression system as a result of water entering the control panels through the ends of several open conduits.[23] Fluctuations in ambient temperatures can thus cause condensation with its concomitant adverse effects on electronic and electrical circuitry and equipment. Moisture weakens the dielectric strength of insulators and promotes rot, bacteriological invasion, and reduced tensile strength.[24] It can result in the burning of insulation and the propagation of shorts by increasing surface conductivity and can also cause the failure of relay coils by promoting arcing, which produces contact pitting. Besides stressing insulating materials, spray water and steam can entrap and deposit substances detrimental to electronically conductive surfaces.[21,22] Multiple pin connectors are susceptible to corrosion product buildup around their bases, causing electrical shorts between pins and from pins to ground.

Moisture can adversely affect movable mechanical parts such as linkages, pivots, bearings or electrical pins, contacts, and connectors. In one event, moisture was transmitted throughout an instrument air system when dryers in that system had been shut off because of a malfunction. This was reported at one plant when water from the fire water system inadvertently entered the instrument air system.[25] Once present in I&C instrument air tubing, moisture is difficult to remove and, in the interim, promotes corrosion.

3.4 VIBRATION AND SHOCK

Systems operating with low levels of vibration may eventually experience high-cycle/low-stress fatigue in their components, loosening of fasteners, and shifts in set points or calibration. Vibration generated by nearby machinery during plant operation may be transmitted to sensors and I&C racks through the building structure. Electrical components including relays and breakers exhibit chattering due to vibration that potentially could affect the operation of the component.

Repeated but random mechanical shocks can impose as much cumulative degradation as does cyclic mechanical motion, the principal difference being that failure due to the latter stress can be predicted reasonably well. However, it is the cumulative effect that promotes accelerated aging-related degradation. Therefore, the frequency of occurrences is important when the aging effect of shock is considered. Water hammer in the process piping is a leading contributor to this aging stressor.

3.5 HUMAN INTERACTION

All man-machine interfaces are susceptible to the consequences of human interaction, which can create a situation conducive to the degradation of equipment. This interaction may be via normal practices in maintenance, operation, repair, and testing or inadvertent acts that, as a result, contribute to the wear-out and accelerate aging for the system components. If the human interaction is erroneous, the aging effect can be even more serious. A need exists for studies that (1) include the human as one of the root causes of degradation and failures of modules and (2) determine the sensitivity of new technology systems to operations, testing, and maintenance errors.

3.5.1 Normal Manipulations

Modules are manipulated both while in place during testing and calibration and when moved to another location for maintenance and repair. Interactions can also occur when other equipment and devices are connected or disconnected, when they are turned either on or off, or when the modules are routinely surveyed or serviced. For example, set points (potentiometers) require adjustment, jumpers are installed and removed during some surveillance and testing procedures, and needle valves and block valves are opened and closed during testing and calibrations. Each of these can be unintentionally mishandled, and each such act contributes to wear of some component. In most cases, this wear is predictable.

3.5.2 Erroneous Manipulations

Aging is further hastened by operational misuse such as in-service stresses, installation errors, and incorrectly applied procedures.[26] An incorrect operating procedure can adversely stress the equipment's subcomponents and may accelerate the aging process. For example, frequent starting of certain electrical equipment before allowing them to cool down could age the insulation and cause premature electrical shorts or grounds. Another example is inadvertent overrange of a module, thereby compromising its performance and shortening the expected service life. Maintenance can likewise be a cause of degradation in many components of nuclear power plants when errors in maintenance or the lack of preventive maintenance can cause the module to experience accelerated aging wear. For example, maintenance-induced problems occur when test pressure is applied to the wrong side of a pressure transmitter or when isolation and equalization valves are not manipulated in the correct sequence to prevent exposure of the sensing element to sudden changes in pressure. Even though catastrophic failure may not result, a degradation stressor has nevertheless been applied to the component. Stress may also be applied to instrumentation when the module is cycled above or below its normal operating range. Sometimes when new components are installed, physical as well as functional stressors are imposed on adjacent components or circuitry, and performance tolerances can become marginal, resulting in repeated repairs to the same module. Other abusive conditions created by human interactions can result from general construction and maintenance activities near the installed modules.

3.6 PROCESS PRESSURE SURGE

During normal operation, sensors and process transmitters are continuously exposed to small process pressure fluctuations and to larger pressure surges during reactor trips and other process system transients. Cyclic pressures accelerate normal wear and the loosening of parts in mechanical components. Process excursions and surges force sensors and transmitters to produce an overrange response. High pressure can affect the span and zero offset of a module, which may therefore require more testing and calibration. Such stressors can impose forces outside the design specifications for the module. They can also fatigue the measuring diaphragms through work hardening of the metal, which may result in cracks and stiffness changes.

3.7 TESTING

Components are periodically tested to monitor and maintain their condition during the life of the plant. In current generation plants, testing is performed by human interaction; however, in proposed systems of the future, many testing functions will be automated. Plant technical specifications require that certain safety-related equipment be tested regularly for operational readiness. However, if the tests are performed too frequently, unnecessary stresses could be placed on the component. Also, certain tests such as high-potential testing of electric components or tests requiring fast starts could impart a larger stress than the component is designed to experience normally.

Mechanical tests may include vibration, elevated temperatures, valve stroking, leak tests and measurements, and functional tests. Electrical tests typically measure insulation resistance, dielectric strength tests, and contact resistance and may involve the application of high voltages. These tests affect mostly electronic devices sensitive to high temperature and humidity.

3.8 ELECTROMAGNETIC INTERFERENCE

More than 50 LERs describing electromagnetic interference- (EMI-) related events have been documented from various plants using solid state modules.[27] This information indicates that nuclear power plants have experienced spurious I&C system behavior due to EMI. Forty-two LERs describing EMI-related events were documented between 1969 and 1980. EMI is an aging stressor in that it may cause (1) spurious equipment operation resulting in overcycling of components and systems, (2) damage to components that protect against electrical noise and transients, and (3) progressive degradation to specific components (e.g., insulation). EMI vulnerability is a concern for electronic modules in reactor plant safety systems. These modules include solid state devices from sensitive microelectronics to power electronics, especially in computers and microprocessors, which are finding larger use in various types of reactor systems and equipment critical to the safety and operation of a nuclear power plant. An EMI stressor may originate from within the affected system or external to it.

Electrical *transients* may be either conducted or radiated to the susceptible module. *Radiated* is the term used to characterize nonmetallic paths and *conducted* is the term used to characterize metallic paths. Coupling paths are referred to as conductive if they result from the ohmic contact between two components or wires; radiative if they are caused by the stray capacitance between two components or inductance between two separated but adjacent conductors.

3.8.1 Transient Impulses

Arc welding equipment may generate radiated and power line-induced electromagnetic interference. Plant maintenance routinely requires the use of arc welding equipment for cutting and welding of piping and structures. However, when used in close proximity to I&C modules, arc welding equipment may inject false signals into the control and monitoring circuitry. Arcing from welding equipment or relay and breaker contacts, in conjunction with the inductance of equipment wiring, can cause current and voltage surges in associated circuits. Voltage surges may lead to shorts and arcing in locally weakened insulation. Current surges may radiate EMI to nearby wiring. The action of protective devices can also cause transient stresses on other components, resulting in set-point drift, mechanical fatigue, and surface burning (due to arcing). The back electromotive force associated with de-energizing electromechanical relay coils may be sufficient to generate degradation in some circuits.

3.8.2 Lightning

Lightning can cause induced voltages that may penetrate instrumentation and control system signal lines, data processing system cables, and power supply circuits. Lightning-induced energy pulses are sometimes sufficient to break down (ionize) the insulation between adjacent conductors, to weld contacts together, to bridge the gap to an unnatural ground, and to ignite combustible materials. Other times, lightning acts more as a stressor, merely degrading the service life of electronic devices. However, lightning-induced impulses as an aging factor have yet to be fully studied.

Lightning, although well appreciated for its direct and immediate destruction of equipment, may also act as a stressor when the lightning-induced pulse results in only partial damage, leaving equipment degraded and vulnerable to further stress. Lightning can induce stressful energy impulses into I&C systems, both safety related and nonsafety related. The impulse energy produced by a nearby lightning strike is a definite environmental stressor to I&C modules even when protective devices are installed on the system. Sometimes the protective devices are *sacrificed,* leaving the system in a less secure state. Such impulses have caused spurious signals in electronic systems, analog as well as digital, and spurious equipment actuation or its failure when thresholds were exceeded. This impulse energy has at times circumvented the designed protective features and adversely affected I&C modules and control systems via plant structural members, piping, and even the plant electrical grounding grid network. Sometimes the modules fail catastrophically, which is not an aging-related event, but more often the consequence is a degradation that shortens the natural service life of the modules. These incipient failures, depending on the weakness involved, manifest themselves at some later time, and the failure then becomes a part of the *random* failures from *unknown* causes cited in reports about operating experiences.

The database used for this study contains 80 entries associated with lightning strikes. About 58% of them affected an I&C system, with 28% identifying specific kinds of I&C modules. Thirteen percent affected safety systems, and 21% affected other systems related to safety. Tables 1–3 in Appendix B list events that involved I&C systems and modules affected or failed by lightning strikes.

3.8.3 Radio-Frequency Devices

Radio transmitters and other devices producing radio-frequency interference (RFI) may produce spurious operations (i.e., unwanted actuations of modules) even during normal plant operation and have a deleterious and cumulative effect by adding to wear out of electronic/electrical/solid state devices. An NRC IE Information Notice depicted four such RFI events in which portable radio transmitters caused system malfunctions and spurious actuations in nuclear power plants.[28] To date, solid state devices have been the components most susceptible to RFI. As older plants are retrofitted with solid state equipment, more cases of RFI by portable radio transmitters are likely to result. The use of the increasingly popular cordless telephones presents another possible but weaker source of RFI.

3.8.4 Power Supplies

A problem of increasing importance is developing as more digital equipment and solid state devices replace (upgrade) existing analog and electromechanical equipment. Electrical services to this equipment may not be *modernized* at the time of upgrade, and the characteristics of the older power supplies can cause overheating in some modules;[29] for example, nonsinusoidal waveshapes that may be present impose effects that were not considered in the design of the new modules. Digital logic is particularly sensitive to the power supplied by inverters used successfully with former analog devices.[30]

3.9 CYCLES OF OPERATION

A stressor that may not come to mind immediately when studying aging is that of equipment remaining in a dormant or standby state for long periods, even though it is a common practice to recognize *shelf lives* in storing spare parts. Some parts may degrade from lack of use when in storage or when in inactive service. An example is an inactive transmitter or controller which fails when activated because motions are impeded by sticking or increased contact resistance due to the presence of dust or corrosion. Aging is hastened by operational misuse such as in-service stresses and installation errors.

3.9.1 Normal Usage of Installed Equipment

Not all I&C module failures are due to stressors of an unusual magnitude or character, because aging is the generic effect from normal wear. Wear through normal use is common for both electromechanical and electronic devices. During normal operation, some mechanical devices maintain a balance between two opposing mechanical forces. When one force is perturbed by either a change in the process, environment, or a control signal, a new position of equilibrium is established. Such imbalance occurs frequently, sometimes continuously, and can lead to a *hunting* or oscillation in the mechanism. The I&C modules most often affected are some sensors, transmitters, and control actuators. Other examples include electronic devices that have inherent limitations on the number of correct operations that can be expected, chemical exhaustion of battery cells, and batteries that develop a memory whereby energy storage capacity is lost if the battery is not periodically deep-discharged. Contacts are subjected to cyclic fatigue. Because mechanical contact is made each time these components are in operation or change state, wear out can occur, causing set-point drift or open or loose connections. In addition, the surface of these contacts can be damaged by sparks, pitting, lower seal-in forces, vibrations, and sticking.

3.9.2 Quiescence and Hyperactivity

Aging occurs under all circumstances—it can never be entirely eliminated, even if a module is dormant. In fact, some modules in standby systems display increased aging due to inactivity. Inactivity as a stressor for mechanical parts such as linkages, bearings, and pivots can cause the parts to develop a set or to stick, producing a situation where higher than normal forces become necessary for them to break away from this position and allow movement in response to a demand. In electronics, image burning of cathode ray tubes resulting from an unchanging image on the phosphor screen can degrade the display and thereby adversely affect the desired response from an operator. Hyperactivity is the condition in which modules experience rapid changes in their operating modes or are switched on and off many times at a fast rate. Some susceptible components are lamp filaments and indicators, chart drives (high-speed tracing), computer disk drives, and power supplies. Certain pieces of safety equipment remain on standby and are required to become operational anytime the safety of the plant is challenged. The technical specifications may require periodic start/stop testing of such equipment to ensure its operational readiness. This requirement could involve cold starts of the equipment, which introduces a higher stress than is usually experienced during normal operation.

3.9.3 Storage

Degradation and premature failure can sometimes be traced to the effects of storage environment on I&C equipment. Storage areas often have uncontrolled environments and therefore can subject stored equipment to extreme temperature, humidity, and even corrosive atmospheres. Elevated temperatures can accelerate the effects of some of the other stressors and thus further reduce the anticipated useful service life of I&C modules. Varying temperatures cause metals to expand and contract, seals to weaken, moisture to condense, and metallic surface contamination to develop.

Another factor is shelf life. For example, the deterioration of the dielectric film of electrolytic capacitors is a time-dependent physical effect and occurs regardless of the physical or functional status of the capacitor. Although certain actions can accelerate the reaction, storage does not prevent the reaction from continuing. One recent example occurred on July 9, 1988, at an operating plant when the reactor water cleanup system was reported to have isolated because of an erroneous signal from the steam leak-detection instrumentation.[31] The temperature-monitoring switch was found to be reading above its trip set point. This switch had been installed on May 12, 1987, after being purchased and remaining in warehouse stock since 1978. Bench testing of the temperature switch verified deterioration of capacitors within the internal power supply circuitry.

Sometimes the same storage area contains bottled gas, cleaning agents, building and maintenance supplies, and water treatment chemicals and is ventilated by unconditioned outside air. Exposure resulting from other activities is inadvertent but not all that unusual and, as such, can present an uncontrolled and unanalyzed condition.

3.10 TECHNOLOGICAL OBSOLESCENCE

Obsolescence does not produce degradation of equipment; however, it directly affects maintenance and repair practices and therefore should be discussed in relation to other stressors. With a lack of spare parts for many I&C modules due to their obsolescence, undesirable compromises required during PMs and repairs accelerate the aging process. Technological obsolescence spawns from three sources.

1. Suppliers withdraw support from I&C equipment they sold to plants years earlier.
2. Advances in I&C have produced modules with capabilities that far surpass those that were state of the art 20 years ago.
3. Requirements that equipment must meet tend to change over time.

3.10.1 Supplier Support

A growing problem facing utilities occurs when the original manufacturer of a needed part is no longer in business and a spare part must be found. One such recent case involved failure of a General Electric relay that was no longer manufactured.[32] Although these relays were used in several other safety-related applications, the problem of replacement had not been given prior consideration. This is not an isolated example in the industry. Compounding the problem is the general lack of interchangeability and standardization of parts in nuclear plants because of the continual modifications of plant equipment over the years. This leads to not having a secure supply of parts, and in many cases, a second source does not exist.

Initially, nuclear industry I&C followed the technological lead of fossil-fueled power stations. Other industries such as pulp, paper, chemical, and petrochemical contributed technology as well. Over the past decade, fossil-fueled plants, driven by competitive economic forces and equipment obsolescence, felt the need to modernize by using newer generation I&C components and systems. Thus, they abandoned most of the older technology and methods that nuclear power industry relied on for measurement and control of both nuclear and nonnuclear systems. Even the national and international standards reflect this shift. Unlike fossil generation and other industries, however, the nuclear industry has remained technologically stagnant.

Issues of vendor support, including withdrawal, have been recognized in some detail.[33] With this recognition by NRC and industry, some positive steps have been taken, including enhanced maintenance and replacement programs. In view of the growing concern over the availability of spare parts, NRC-AEOD sponsored a course in November 1988 on the procurement of replacement parts and components. The course presented a practical approach to procuring electrical and mechanical replacement parts including commercial-grade items for safety-related systems in operating nuclear power plants.[34] Plant maintenance organizations are spending increasing time on evaluating substitute materials for components no longer manufactured or available for purchase. For two reasons, many spare-parts companies have gone out of business in the past decade or have changed their business to cater to a different market: (1) the drop-off in new plant construction and (2) stricter NRC and utility documentation and certification requirements, particularly in quality assurance (QA) programs. Many vendors that produced the same product for nuclear and nonnuclear applications found the nuclear option financially inferior and have exited the market. Availability of spare parts also has a direct impact on the timely completion of maintenance.

3.10.2 Advancing Technology

As I&C technology advances, so does the desire for better performance from the I&C systems, modules, and equipment. However, once installed, actual system performance seldom improves and, in fact, slowly degrades with age. Only by upgrading can expectations for improved performance and reliability be realized. A heightened awareness of the superior capabilities of current I&C equipment has emerged. Concurrently, the limitations of older technologies have become clear. Older (analog) equipment is prone to failure and signal drift. Analog systems require more frequent surveillance, testing, and calibrations than do microprocessor-based systems.

The relatively rapid turnover in I&C technology compared to mechanical equipment like valves, motors, and control rod drives is a consideration that would not have been expected to arise in other NPAR studies. The problem is not one of degradation but rather one of absence of the desired improved performance.

3.10.3 Changes in Regulations

Requirements for the commercial nuclear power industry are continually being reviewed and revised to ensure that the latest operational events and technological changes are accommodated by current regulations so that public health and safety are not compromised. The Three Mile Island incident resulted in massive efforts to improve postaccident monitoring systems,[35] to enhance the control room design,[36] and to ensure the qualification of safety-related equipment.[37] Currently under study are guidance requirements for digital modules, microprocessors, and other equipment and for the service requirements and peripheral needs of the advancing technology.

3.10.4 Upgrading

The goal of new technology in I&C equipment (mostly digital) is to provide improved performance, reliability, and maintainability. Immediately observable improvements are reduced instrument set-point drifts and significant reductions in the equipment count required to implement a control system. Consider that a single analog card may be more reliable than a more complex digital microprocessor card; however, the overall digital control system may be more reliable because one microprocessor module replaces several analog modules.[29] The inclusion of microprocessors and their concomitant software in the RPS marks a significant departure from the original analog electronic design. While the transition to digital systems may provide significant performance and safety advantages, it may also introduce issues and concerns that have not been encountered previously and have not undergone a thorough safety review.

An upgrade at one plant was essentially a replacement-in-functional-kind of an obsolete RPS and nuclear steam supply system control system analog module with a modern microprocessor-based module.[29] The licensee stated that the existing Taylor recorder/controllers and original Foxboro equipment were about 20 years old and that recent inspections had shown widespread instances of age degradation. Some equipment used at the plant had been out of production since 1968, and spare parts were increasingly difficult to obtain. The licensee concluded that modernization of the reactor protection and control systems should result in a reduction in time spent for equipment repair and maintenance. This upgrade would enhance equipment reliability, thus providing a basis for potential availability gain.

4. INSTRUMENTATION AND CONTROL MODULE CATEGORIES

The six I&C modules in this study are generically represented by several manufacturers each of which produces several distinct models. Each of the six categories consists of many different models but all of a singular kind. As such, each category is identified by the general descriptions in the following sections. Definitions in italics were taken from the Instrument Society of America (ISA) "Pressure Instrumentation Terminology," ISA-S51.1.[38]

4.1 INDICATORS

Instrument, indicating: a measuring instrument in which only the present value of the measured variable is visibly indicated.[38]

Indicators are modules that associate an input quantity to a measured variable in some directly observable fashion. Indication implies a representation to the eye from which the mind infers either an individually distinct state or, in most instances, a quantity in which it is interested. This magnitude of measurement can be conveyed to the eye either individually by a digit, by a combination of digits, or on a graduated scale on which digits are shown in a logical sequence (Figs. 4.1–4.3). In the latter case, a movable reference such as a pointer is required to indicate the digit of interest on the scale. In general, indicator scale markings may be arbitrarily placed without regard to degrees, inches, or any other measure of positions.

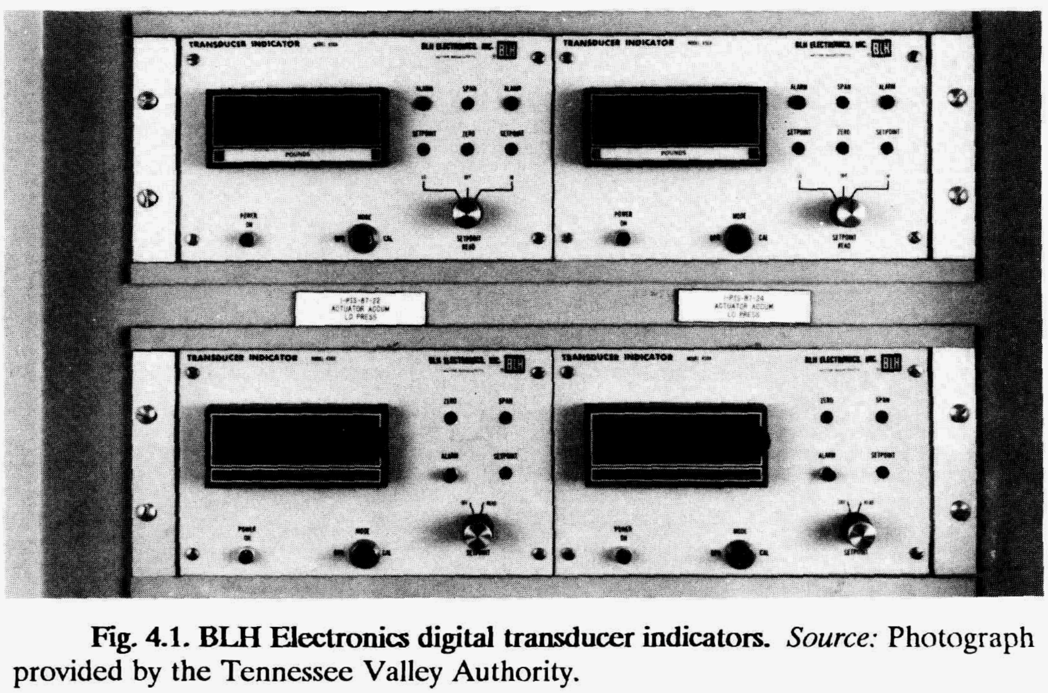


Fig. 4.1. BLH Electronics digital transducer indicators. *Source:* Photograph provided by the Tennessee Valley Authority.

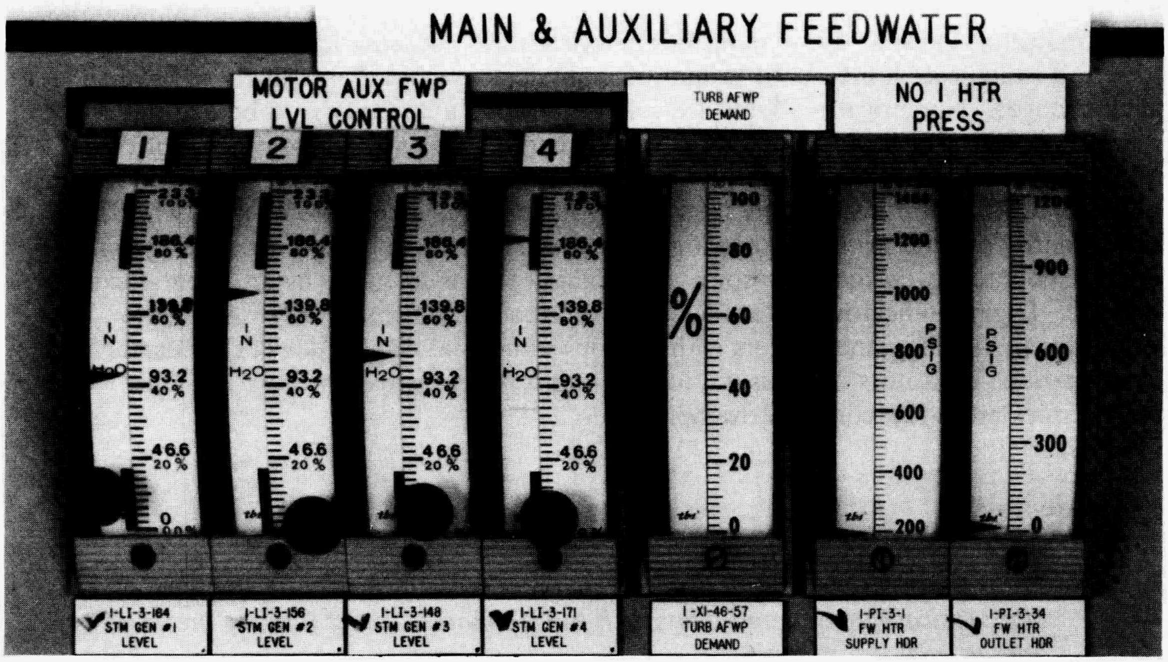


Fig. 4.2. Westinghouse indicators: level, demand, and pressure. *Source:* Photograph provided by the Tennessee Valley Authority.

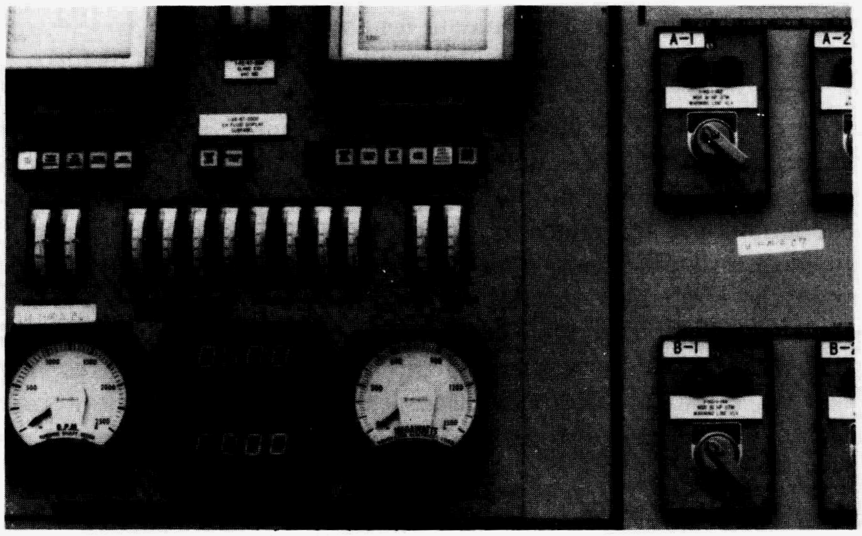


Fig. 4.3. A variety of dial, vertical, and digital indicators.

Analog instruments use one physical variable (e.g., angular position, color, brightness) to indicate another (e.g., temperature, flow rate, neutron flux level), whereas digital indicators present the readout in numerical form. Analog instruments are often preferred to indicate a measured instant of a physical variable that is likely to have sudden quick changes. Numerical readouts are called for when a quantity is to be counted. For example, an automobile speedometer is usually configured as an analog indicator of instantaneous speed, whereas mileage traveled is digitally shown on a counter (odometer). Likewise, a flowmeter customarily uses analog indication for the rate of flow, while the total quantity of fluid having passed a given point is usually indicated numerically on a separate counter, even though a somewhat complicated conversion by an integrator is required. Digital indicators are available for the readout of almost any variable: voltmeters (including panel meters with gaseous numerical display devices); thermocouple thermometers; indicators of pressure, load, strain torque, or pH; oscilloscopes; and stroboscopes are a few common examples.

4.2 SENSORS

1. *Transducer: an element or device which receives information in the form of one physical quantity and converts it to information in the form of the same or another physical quantity.*
2. *Element, sensory: the element directly responsive to the value of the measured variable.*
3. *Element, primary: the system element that quantitatively converts the measured variable energy into a form suitable for measurement.*[38]

A sensor then is the module that measures directly the process variable and produces an output signal or indication proportional to the measured variable. Sensor output signals must be accurate and up to date for the safety-related systems to perform properly. Sensors are exposed to the harshest conditions and process disturbances of all I&C modules. During normal plant operation, sensors are exposed to a variety of conditions that can cause performance degradation over time. Their inaccessibility during plant operation severely restricts the times that many of these modules can be serviced and repaired. Hence, the failure rate per module can be expected to be high in relation to the other five modules in this study.

An example of a simple sensor is a thermocouple for measuring temperature. In the thermocouple, the temperature difference between the hot junction and the reference junction creates a dc voltage directly proportional to the temperature difference. Some other sensors are resistance temperature devices (RTDs), strain gauges, resonant wires, piezoelectric devices, variable reluctance devices, capacitive elements, Bourdon tubes, and linear variable differential transformers. Two different RTDs are shown in Figs. 4.4 and 4.5.

Fig. 4.4. Resistance temperature device sensor.

Fig. 4.5 Weed resistance temperature device.

4.3 CONTROLLERS

Controller: a device which operates automatically to regulate a controlled variable.[38] *Another ISA proposed definition is that a controller is a device that compares the value of a variable quantity or condition to a selected reference and operates in such a way as to correct or limit the deviation.*[38]

Many industrial processes require that certain variables such as flow, temperature, level, and pressure remain at or near some reference value, called a *set point*. The device that serves to maintain a process variable at the set point is called a *controller* (Figs. 4.6 and 4.7). The controller looks at a signal that represents the actual value of the process variable, compares this signal to the set point, and acts on the process so as to minimize any difference between these two signals. This control function may be implemented by using pneumatic, fluidic, electric, magnetic, mechanical, or electronic principles or combinations of these.

Controllers are the command centers that determine and set the action for the I&C systems. Controllers now offer computer and other unique control capabilities that had previously been impractical. In this respect then, input signals, power supplies, and operational environment are more important than for the other I&C modules. Controller modules follow indicator modules in the replacement activity for I&C modules.

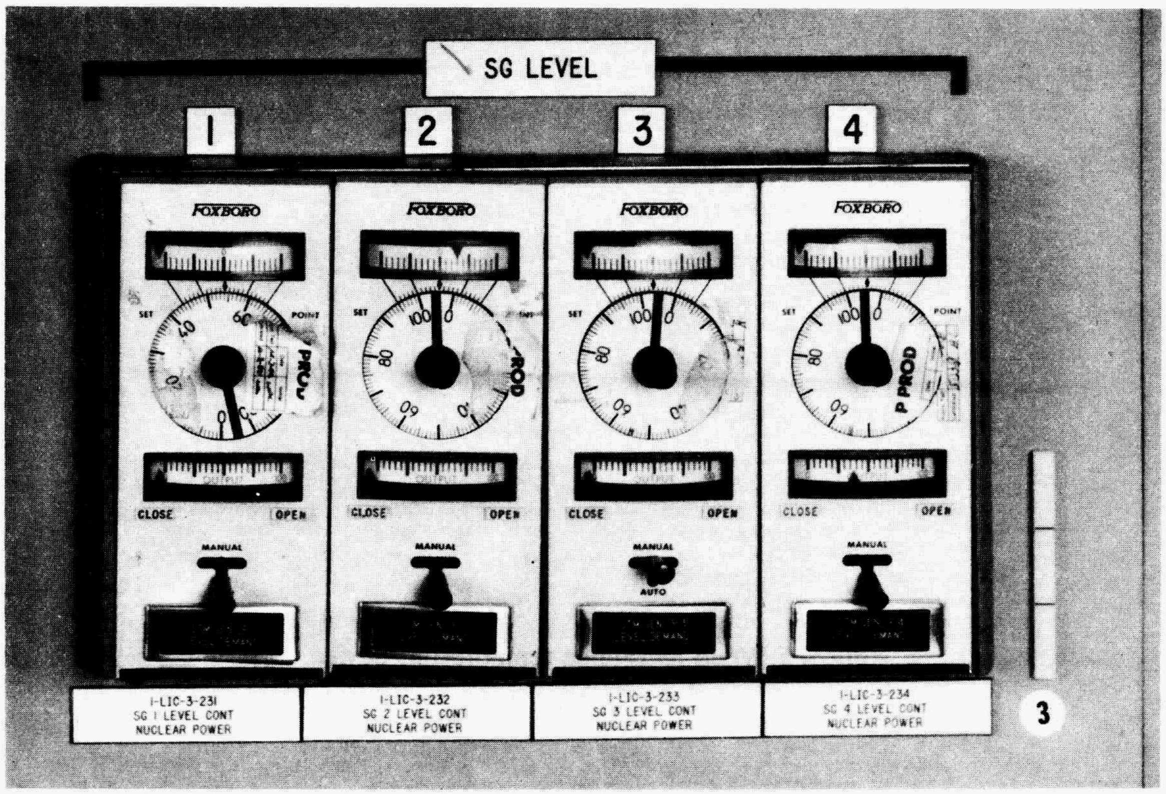


Fig. 4.6. Foxboro set-point level controllers. *Source:* Photograph provided by the Tennessee Valley Authority.

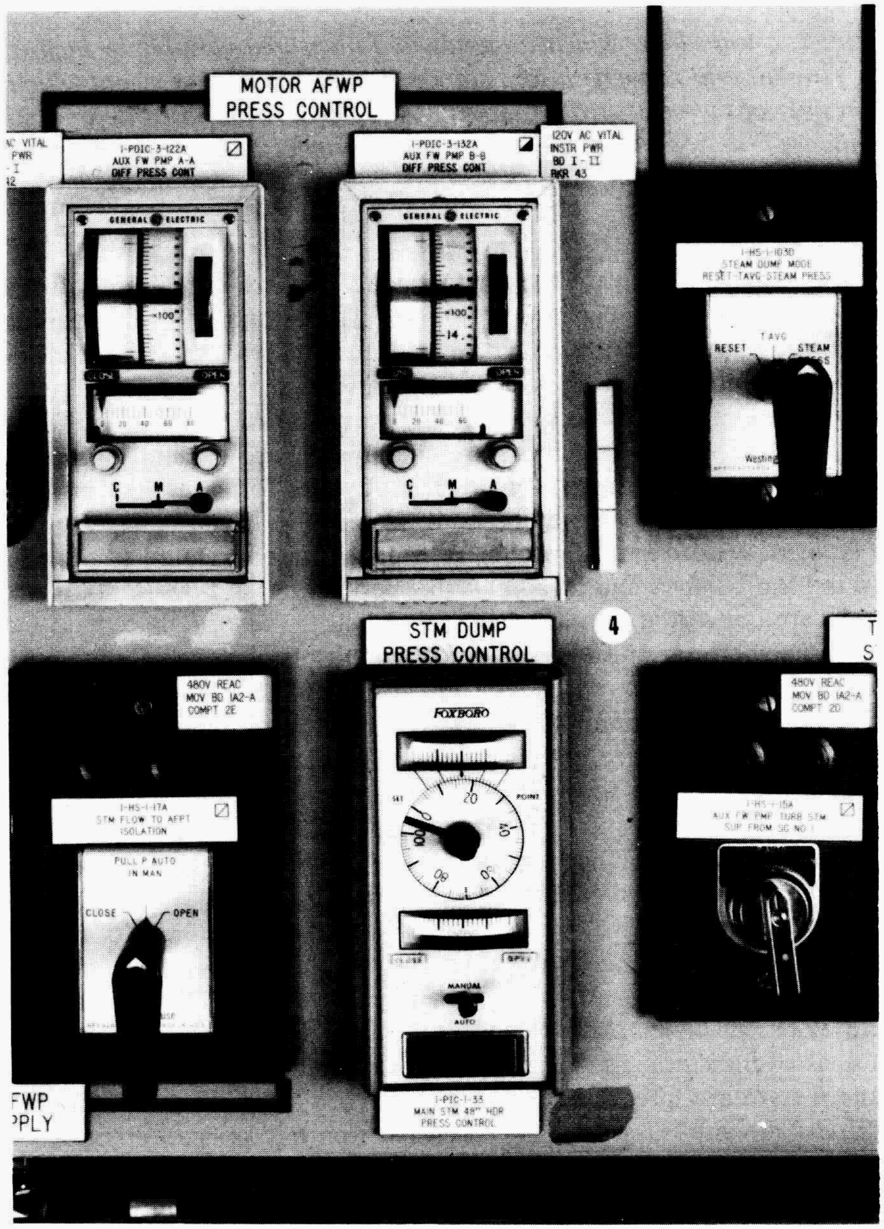


Fig. 4.7. Set-point differential pressure controllers: General Electric (top left and center), Foxboro (bottom center). *Source:* Photograph provided by the Tennessee Valley Authority.

4.4 TRANSMITTERS

Transmitter: a transducer which responds to a measured variable by means of a sensing element and converts it to a standardized transmission signal which is a function only of the measured variable.[38] *Another ISA proposed definition is that a transmitter is a device which receives an analog signal and converts it into a suitable input signal to a controller. The transmitter module can, optionally, include a primary sensor.*[38]

Pressure transmitters provide important signals that are used for control and monitoring of the safety of nuclear power plants (Figs. 4.8 and 4.9). Depending on the plant, 50 to 200 pressure transmitters may be in the safety system, with newer plants having the greater number of transmitters. Aging affects the performance of these transmitters, and temperature is the dominant stressor in most cases.

Nuclear plant pressure transmitters are complex electromechanical systems designed for measurements of pressures from a few inches of water to about 3000 psi. The transmitter usually converts the sensed pressure to a proportional voltage or current signal. Two types of pressure transmitters are found in safety-related systems in nuclear power plants: motion balance and force balance, depending on how the movements of the sensing element are converted into an electrical signal.

In motion-balance transmitters, the displacement of the sensing element is measured with a strain gauge or a capacitive detector and converted into an electrical signal that is proportional to pressure. An example of this type of transmitter is one that consists of an oil-filled cavity with a capacitance plate as the sensing element. As differential pressure changes, the capacitance of the sensing element changes accordingly. This capacitance change is measured, amplified, and linearized by suitable electronics to provide the transmitter output. In force-balance transmitters, a position-detection device is used to establish the displacement of a diaphragm bellows or a bourdon tube resulting from pressure change, and a restorative mechanical force is then generated to null the displacement as it develops. A feedback control system uses the displacement signal to control the force and simultaneously provides an electrical signal that is proportional to pressure. The electrical signal is then transmitted to a remote location where the signal is displayed on a visual indicator, used as an input to a signal conditioning device or a trip unit that changes state at a predetermined level, or used as an input to a controller.

Transmitters are subjected to aging stressors from misapplication, poor design/fabrication, testing and maintenance practices, their environment, the process they are monitoring, and their electrical power supply. Under normal plant conditions, the environmental stressors are temperature, humidity, and radiation. Nuclear application transmitters are sealed for design-basis-accident (DBA) environmental steam conditions; therefore, normal environmental humidity should not pose a problem. However, when that occasional steam line leaks, fire sprinklers actuate, or compartment flooding occurs, nearby equipment is likely to be adversely affected and the remaining service life becomes questionable. Elevated temperatures and radiation may also affect the environmental seals and the electronic subassemblies over time. Instances have been noted where ambient temperatures have been elevated over prolonged periods, causing seals to harden, crack, or take a set, thus allowing moisture to enter even during normal plant service.

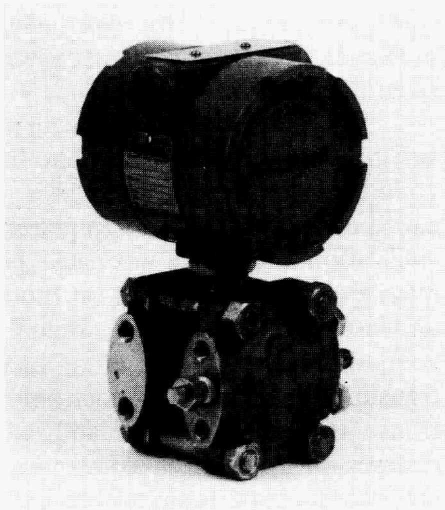


Fig. 4.8. Rosemount pressure transmitter. *Source: Effect of Aging on Response Time of Nuclear Plant Pressure Sensors*, NUREG/CR-5383, U.S. Nuclear Regulatory Commission, June 1989, p. 13, Fig. 5.3.

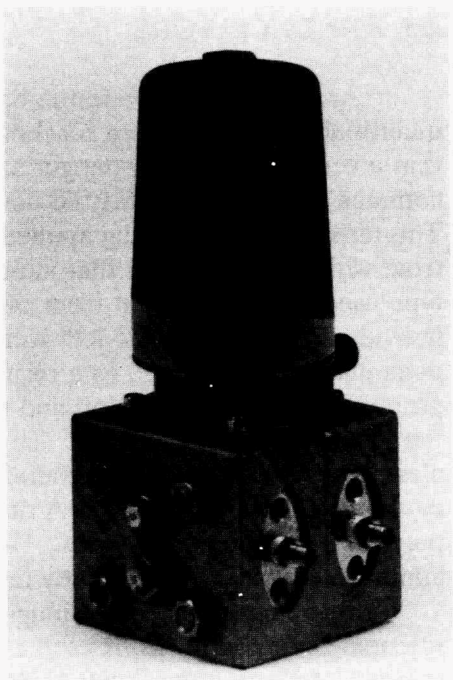


Fig. 4.9. Tobar differential pressure transmitter. *Source: Effect of Aging on Response Time of Nuclear Plant Pressure Sensors*, NUREG/CR-5383, U.S. Nuclear Regulatory Commission, June 1989, p. 13, Fig. 5.4.

Stresses from transient process overpressure conditions such as water hammer may cause significant zero shifts in the sensing element or mechanical linkages within the transmitter mechanism. Normally, these zero shifts can be removed by recalibration. However, recalibration is usually not possible for in-containment devices during periods of reactor operation.

Pressure surges may occur when a transmitter is valved in or out of service if the valving operations are not done in the proper sequence. For many applications, this condition would cause the sensing cell to be forced out of its calibrated span and could cause a calibration shift. In general, this shift would be correctable during the next calibration and would not cause permanent damage to the transmitter. Calibration shift may be caused by stresses on the transmitter from system conditions and personnel errors and could be indicative of gradual aging-related deterioration of the transmitter components such as loosening or wear of mechanical components in force balance transmitters or aging of electronic components such as capacitors in any type of transmitter. Temperature and radiation conditions approaching DBA levels impose severe stresses on transmitter electronics. Calibration shifts may also occur if the transmitter is calibrated at one temperature and operated at a significantly different temperature, temperature being a stressor.

Failures due to these degradation stressors are not always prompt but progress at a rate faster than the manufacturer's specifications would normally indicate. Adverse effects from aging stressors develop over time and, therefore, are often mistakenly reported as random failures. A tracking of maintenance records and a noting of repair and calibration trends are needed to justify replacing the suspected transmitter before catastrophic failure.

4.5 ANNUNCIATORS

Annunciators provide the operator with a visual or an audible indication that a quantifiable limit has been reached (Fig. 4.10). Warning lights indicate to an operator that a certain potentially dangerous condition exists within a process. The literature contains little to document the development of current industrial annunciator systems. The term *drop* was initially applied to individual annunciator points in process applications, from which we may infer that annunciator systems developed from paging systems of the type used in hospitals and from call systems used in business establishments to summon individuals when their services were needed. These systems consisted of solenoid-operated nameplates that dropped as a result of gravity when de-energized. The drops were grouped at a central location and were energized by pressing an electrical push button in the location requiring service. The system also included an audible signal to sound the alert. By the late 1940s, centralized control rooms were introduced from which the plant could be remotely operated. A drop-type annunciator could be used in these general-purpose central control rooms. However, more compact, reliable, and flexible annunciators were subsequently introduced.

In the early 1950s, the plug-in relay annunciator was developed. Instead of solenoid-operated drops, it used electrical annunciator circuits with small telephone-type relays to operate alarm lights and to sound a horn when abnormal conditions occurred. The alarm lights installed in the front of the annunciator cabinets were either the bull's-eye or backlighted nameplate type. The annunciators were compact and reliable, and because of the hermetically sealed relay logic modules, they could be used in certain hazardous areas in addition to the general-purpose control rooms. Miniaturization of

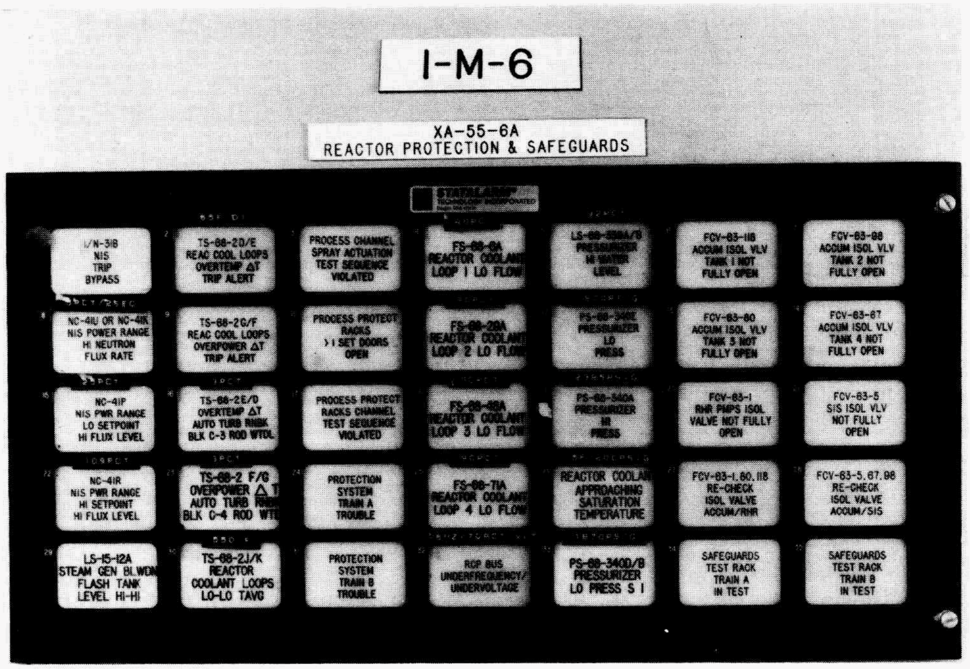


Fig. 4.10. Annunciator panel for reactor protection and safeguards. *Source:* Photograph provided by the Tennessee Valley Authority.

instruments and the use of graphic control panels initiated the development of remote annunciator systems consisting of a remotely mounted relay cabinet connected to alarm lights installed at appropriate points in a graphic or semigraphic diagram.

Solid state annunciator systems with semiconductor logic modules were developed in the late 1950s. These systems permitted additional miniaturization and lowered both the operating power requirements and the heat generated. The semigraphic annunciator was introduced in the late 1960s and fully used the high-density capabilities of solid state logic. It has permitted compact and flexible semigraphic control centers. The trend toward additional miniaturization is the result of the greater availability and reliability of integrated circuit logic components.

4.6 RECORDERS

Instrument, recording: a measuring instrument in which the values of the measured variable are recorded.[38]

A recorder is an instrument module that produces a trace of an input signal and provides a permanent, reproducible copy of that process signal (Figs. 4.11 and 4.12).

The process variable actuates a recording mechanism such as a pen, which moves across a chart. The chart moves constantly with time. These two motions produce an analog record of variable vs time. Any point on the continuous plot obtained in this manner can be identified by two values called *coordinates*. Several coordinate systems are in use, but Cartesian coordinates are most widely encountered.

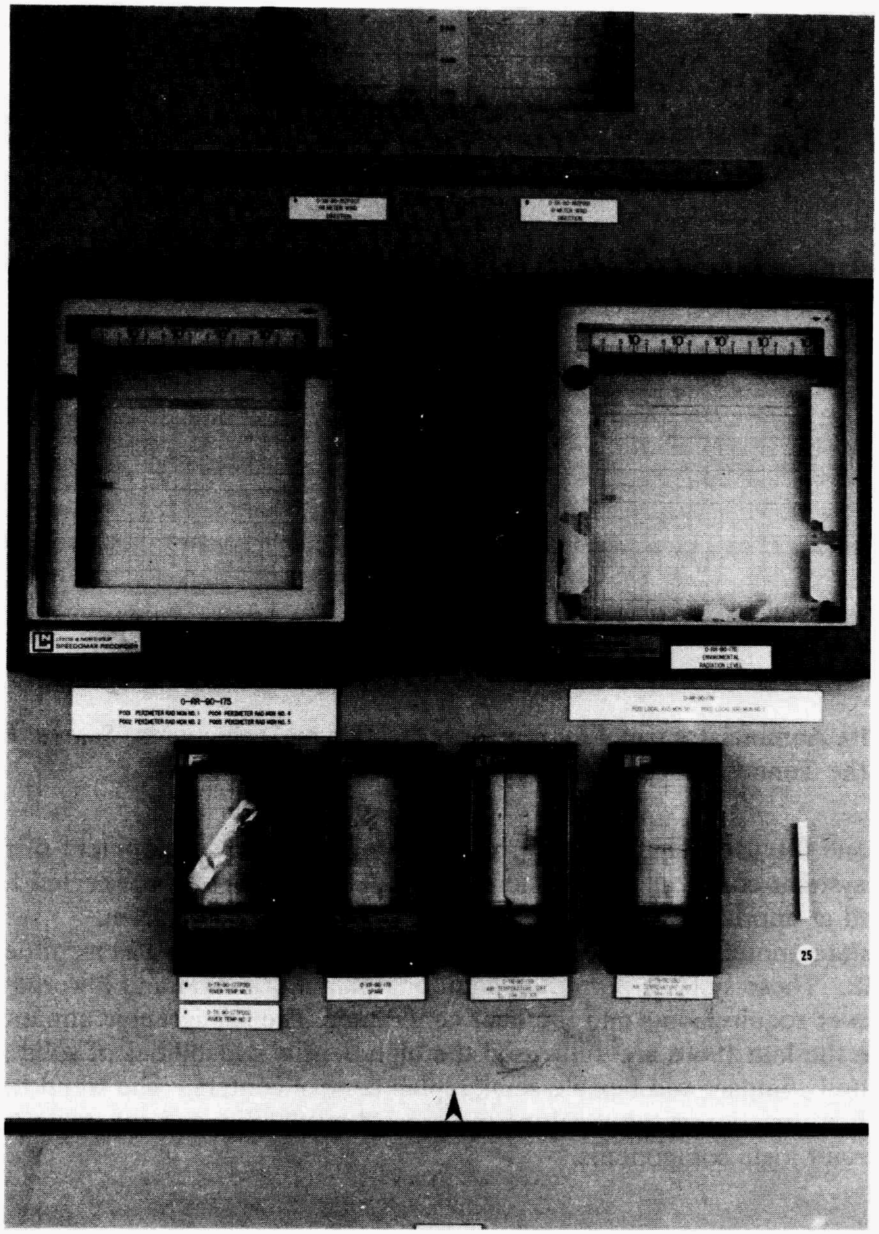


Fig. 4.11. Leeds & Northrup and Fischer & Porter strip recorders. *Source:* Photograph provided by the Tennessee Valley Authority.

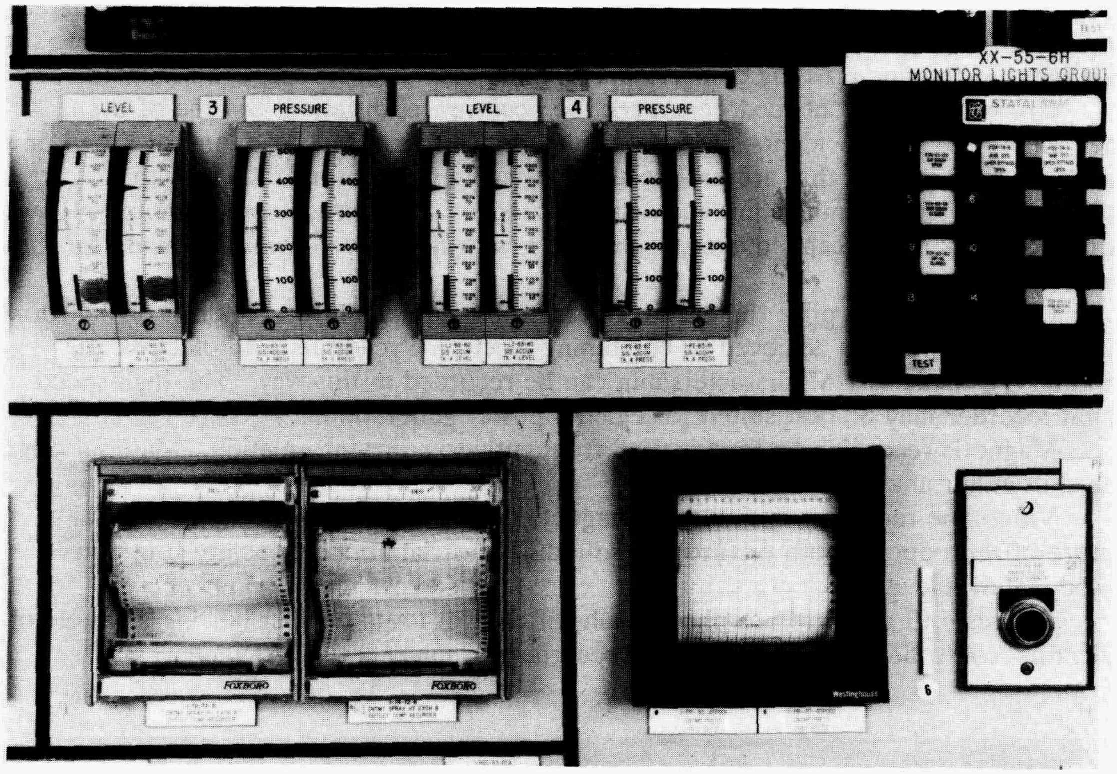


Fig. 4.12. Westinghouse (bottom center) and two Foxboro (bottom left) strip recorders. Also shown on top are vertical indicators for pressure and level. *Source:* Photograph provided by the Tennessee Valley Authority.

The shape of the chart provides a primary means of classification into (1) circular charts and (2) rectangular charts, in sheet or strip form. Strip charts can be torn off and can be stored in rolls or folded in Z folds. Strip-chart lengths vary from 100 to 250 ft (30 to 75 m).

With X-Y recorders, two variables are plotted simultaneously, such as stress vs strain or temperature vs pressure. Either the chart is stationary and the scriber is moved along both the abscissa and the ordinate by the two signals or the chart is moved in one direction while the stylus slides on an arm in the other direction. Combination function plotters, called X-Y-Z recorders, allow the pen to be driven along either axis at a constant speed, thus making recordings of X vs Y, Y vs T, and X vs T possible. Recorders with the three independent servo systems allow the recording of two variables against a third.

The signals entering the function plotter can be analog or digital. Digital signals require transducers to obtain an analog plot. Likewise, digital recorders can be provided with analog-to-digital conversion and conventional digital printout.

When several variables are to be recorded on the same chart, such as several temperatures from thermocouples in various locations, multiple recorders are used. Circular chart recorders can handle up to four variables, whereas strip chart recorders handle as many as 24 to 36 measurements. To identify each variable, symbol or numerical coding or color printing is used, as well as full digital alphanumeric printouts on chart margins.

With the advent of microprocessor technology, multivariable recorders have become available. These allow recording a multitude of variables such as flow and an associated temperature and pressure. Likewise, for comparison of several variables on the same time scale without the line crossing or overlapping, multichannel recorders are available.

Operations or event recorders mark the occurrence, duration, or type of event. They record multiple incidents such as on-time, downtime, speed, load, and overload on the chart. The records that are produced are usually in the form of a bar, with interruptions in a continuous line indicating a change. Microprocessor technology allows scores of points to be scanned every millisecond, with high-speed printouts made for the events that occur.

Digital recorders print the output of electronic equipment on paper in digital form. High-speed recording is achieved by combining electronic readouts with electrostatic printing. As many as 1 M characters/min can be recorded in this way. The high-speed digital recorder may be best suited for electric power generation.

Magnetic recording on tape is used frequently to store information. Material can be stored in either analog or digital form.

Videotape recording is similar to magnetic tape recording, with the additional feature of reproducing both picture and sound. Industrial applications are found in closed-circuit television.

Figures 4.13 and 4.14 are photographs of control room panels showing a variety of indicators, controllers, annunciators, and recorders.

Fig. 4.13. Control room panels showing indicators, annunciators and alarms, controllers, switches, and strip recorders. *Source:* Photograph provided by the Tennessee Valley Authority.

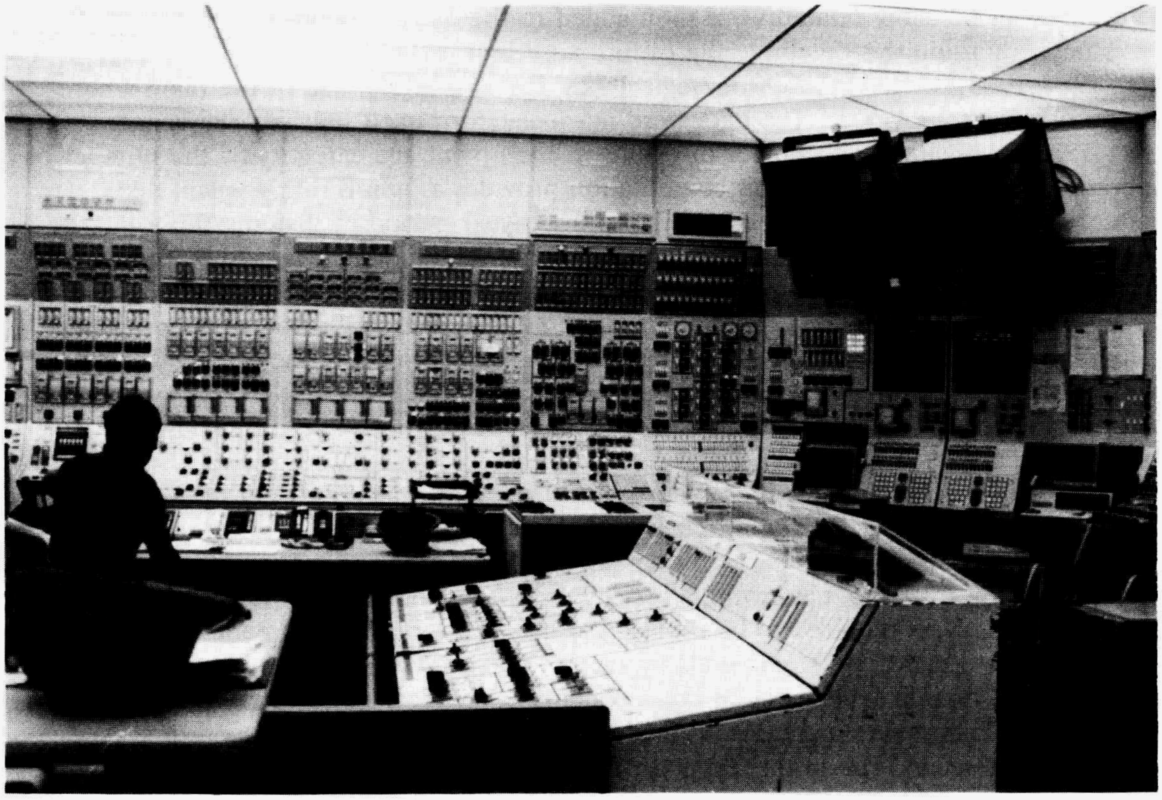


Fig. 4.14. Control room panels and consoles. A variety of indicators, controllers, annunciators, and recorders as well as several cathode-ray-tube display monitors are shown.

5. OPERATING EXPERIENCES

This section examines the aging-related failure data from the LER national database for the six I&C module types over the period 1984–1988. A structured search of the SCSS database was completed by using the name of the module category as a keyword for each of the six categories. The abstracts of LERs retrieved by these searches were reviewed to determine which events were aging related. Those reports that were determined to be aging related were then added to the database constructed as a tool for this study. By using this database, the events were first sorted by year of occurrence (i.e., 1984–1988), then sorted chronologically by the operational age of the plant where the occurrence took place. The data were normalized for each year of occurrence, 1984–1988, by dividing the number of events in each plant age category by the number of operating plants of that age. This computation provides a failure rate vs plant age. Failure rates were also determined for the entire 5-year period by dividing the total number of events for each given plant age category over the 5-year period by the total number of plant years of operation contained in that 5-year period. Appendix C gives an example of how this was done for the indicators module category. Data summaries and accompanying plots for each of the six I&C module categories are also included in Appendix C.

Aging-related failure data can be represented in various ways, but in this study, three basic patterns were hypothesized to exist (see Fig. 5.1). The first pattern, designated Type I, begins with an initially high failure rate that rapidly decreases before leveling off as the module gets older. This pattern is typical of most modules that suffer *infant mortality* or burn-in problems early in their service lives. The second pattern, Type II, has been postulated for a module having a service life that is shorter than that for the plant and, consequently, is required to be replaced at least once during the plant lifetime. Like the Type I pattern, a Type II curve also shows an initial high failure rate, due to infant mortality, that rapidly decreases. In the second case, however, following the rapid decrease, the failure rate increases again steadily before decreasing a second time (see Fig. 5.1). This second *rise* in the Type II pattern could be due to a module's reaching the end of its service life and consequently being replaced by a new or refurbished unit which then undergoes its own burn-in and infant mortality phase. The third pattern, Type III, is typical of modules with failure rates that are so low that accurate conclusions cannot be drawn from the data. A Type III pattern is *noisy* and appears random because of a very small number of reported failures, which results in low failure rates.

5.1 INDICATORS

Of the six I&C modules studied, indicators appear to have the highest rate of aging-related failures per plant per year (see Fig. 5.2 and Table 2.2). Indicators in the mild environment of the control room receive frequent operator attention during each shift because of the necessity to observe plant operations and record the routine process data. However, many other indicators are installed on panel boards outside of the control room, a harsher environment without the continual attention of operators. This may explain the higher rate of aging-related failures reported for indicators.

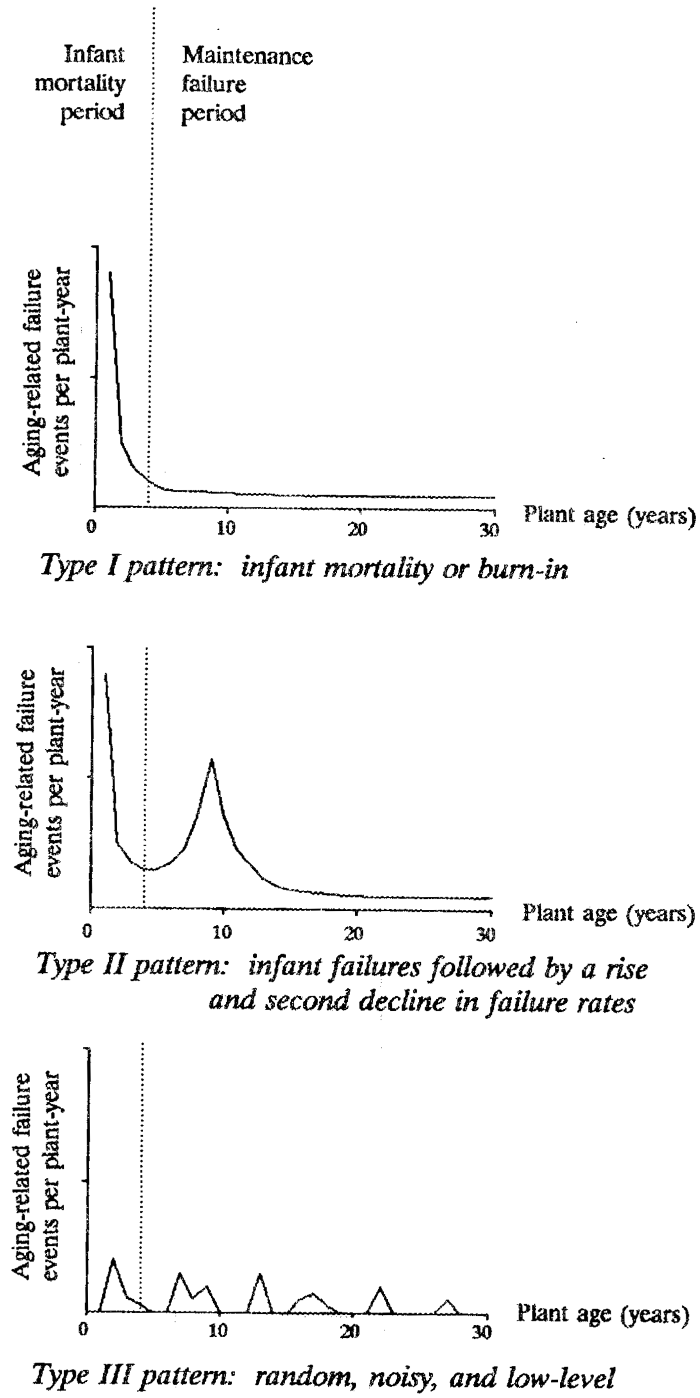


Fig. 5.1. Generic module failure rate patterns.

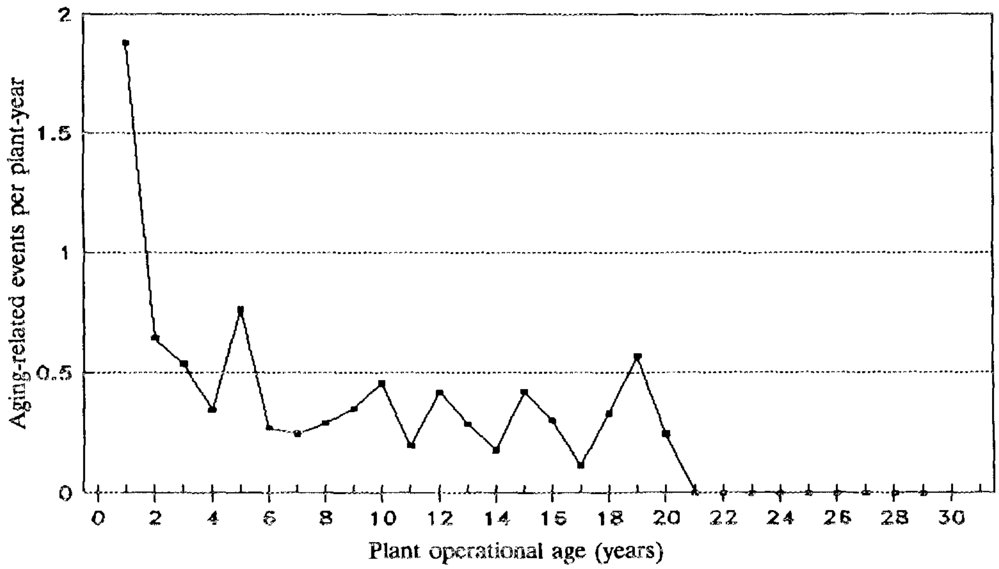


Fig. 5.2. Indicator module aging failures by plant age (1984–1988).

A structured search of the LER database retrieved 997 failure events for indicator modules, of which 220 were judged to be aging related. The indicator module category represented 35% of all aging-related events reviewed and about 1.6% of all the LERs written during this study period. The plot of the normalized failure rate data in Fig. 5.2 suggests a Type I pattern with an evident infant mortality or burn-in period during the first couple of years. An average failure rate calculated by dividing the total number of aging-related events (220) by the total number of plant-years (471) is 0.47 aging-related failures per plant-year, or one indicator module aging-related failure per operating plant every 2.1 years.

The indicator module data summary reveals some insight as to the sharp changes in failure rate around plant ages ten and nineteen (see Appendix C, Table C.1). The peak at plant age 7 years in 1985 is due to two aging-related failures in the single 7-year-old plant operational in 1985 (see Appendix C, Fig. C.5). This same plant also experienced three failures in 1986, five failures in 1987, and three failures in 1988 when the plant was 10 years old. This explains the shifting peak for the successive annual event profiles (see Appendix C, Figs. C.5 and C.6) from plant age 7 years to plant age 10 years and therefore the peak around plant age 10 years in Fig. 5.2. Similarly, two aging-related events in the single 18-year-old plant in 1985 and three events in the same plant the following year at age 19 years account for the peak around age 19 years in Fig. 5.2.

The safety-related systems had a relatively low percentage of the number of aging-related indicator module failures reported in this study when compared to the number of similar failures for other systems (see Table 5.1). The results from this study of operating experiences for indicator modules indicate that no apparent aging-related problem exists that was or could be affecting indicator module performance in safety-related systems.

Table 5.1. Indicator module aging failures by plant system

System	Module failures (%)	System	Module failures (%)
Nuclear	19	Reactor cooling system	3
Heating, ventilating, and air conditioning	18	Stack	3
Radiation monitors	16	Reactor protection system	2
Toxic gas	8	Reactor core isolation cooling system	1
Containment	7	Reactor water cleanup	1
Steam	6	Residual heat removal	1
Feedwater	4	Others	11

5.2 SENSORS

Sensors are exposed over time to a variety of conditions during normal plant operations that can cause performance degradation. Sensors are exposed to the harshest conditions and process disturbances of all I&C modules. Their inaccessibility during plant operations severely restricts the time during which these modules can be serviced and repaired. Hence, the failure rate per module might be expected to be higher than that of the other module categories in this study and more likely to show aging problems because module replacement is more difficult.

The structured search of the LER database retrieved 424 failure events for sensor modules, of which 199 were judged to be aging related. This represented about 32% of all aging-related events reviewed and about 1.4% of all LERs written during this 5-year period. As with the indicator module category, the plot of the normalized failure data for sensor modules also appears to conform to a typical Type I module pattern (see Fig. 5.3). The curves for the sensors module annual data for 1985 through 1988 are shown in Figs. C.9 and C.10 in Appendix C.

Close examination of the database data showed that a large percentage of the aging-related sensor failures were due to toxic gas analyzer sensors and not to ESF system sensors (Table 5.2). An average failure rate calculated by dividing the total number of aging-related events (199) by the total number of plant-years (471) is 0.42 aging-related failures per plant-year, or one sensor module aging-related failure per operating plant every 2.4 years.

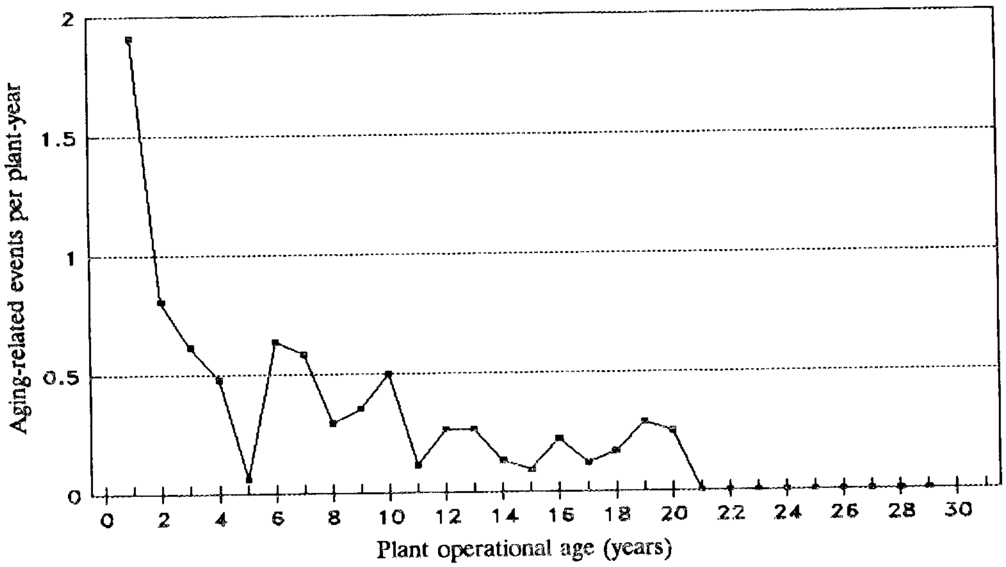


Fig. 5.3. Sensor module aging failures by plant age (1984–1988).

Table 5.2. Sensor module aging failures by plant system

Service	Module failures (%)
Toxic gas analyzers	40
Radiation monitors	21
Nuclear	14
Temperature	9
Pressure and differential pressure	4
Others	12

Another NPAR study analyzed NPRDS data for 315 failures of resistance temperature devices (RTDs), a particular kind of sensor, failures for the period 1974 to 1988.[39] These modules covered a population of 21 vendors. Table 5.3 is adapted from that report and shows that aging-related causes amounted to 40% of the RTD failures, with component faults and personnel error accounting for 53 and 7% respectively. (This same study also looked at a subset of the 498 aging-related pressure instrumentation problems, 287 of them due to sensing lines. As might be expected, blockage dominated and accounted for about 75% of the problems in these.)

Most I&C systems were represented about equally in the LER data reviewed for sensors. The ESF systems are represented in the services depicted for temperature and pressure and are at the lower end of the tally. The results from this study of operating experiences for sensor modules indicate that no apparent aging-related problem exists that was or could be affecting sensor module performance in safety-related systems.

5.3 CONTROLLERS

Controller modules, subjected to more intense testing because of their central function in the I&C system control loop, represented an even smaller fraction of all aging-related failures than did either the indicators or sensors. A structured search of the LER database retrieved 397 failure events for controller modules, of which 105 were judged to be aging related. These represented 17% of all aging-related events reviewed and about 0.8% of all the LERs written during this study period. An average failure rate calculated by dividing the total number of aging-related events (105) by the total number of plant-years (471) is 0.22 aging-related failures per plant-year, or one controller module aging-related failure every 4.6 years.

The plot of the normalized failure data for the controller modules has an apparent Type II pattern with an infant mortality period followed by a second period of higher failure rates around plant age 10 years (see Fig. 5.4). It should be noted, however, that the peak at plant age 9 years of Fig. 5.4 is due largely to the four events in the single 9-year-old plant operational in 1987 (see Table C.3 in Appendix C; the annual data curves for 1985 through 1988 are also shown in Appendix C, Figs. C.13 and C.14). Neglecting these four events at plant age 9 years, the composite curve of Fig. 5.4 would still maintain a Type II pattern. The predominant rise in failure rates occurring around plant age 10 years may be representative of an ~10-year service life of a typical controller module.

Aging-related controller failures were reported in ten different plant systems, with the most often reported system being the auxiliary feedwater system. Table 5.4 gives a breakdown of these ten systems with their respective percentages of the number of module failures. Safety-related systems represented relatively few events. The results from this study of operating experiences for controller modules indicate that no apparent aging-related problem exists that was or could be affecting controller module performance in safety-related systems.

5.4 TRANSMITTERS

A structured search of the LER database retrieved 296 failure events for transmitter modules, of which 79 were judged to be aging related. These represented 12% of all

Table 5.3. Resistance temperature device failure cause description distribution

Cause description	Number of reports	Cause description	Number of reports
Circuit defective[a]	68	Mechanical damage/binding[a]	9
Open circuit[a]	54	Dirty[b]	7
Normal/abnormal wear[b]	41	Material defect[a]	7
Out of calibration[b]	36	Set-point drift[b]	6
Short/grounded[a]	34	Burned/burned out[a]	5
Connection defective/loose parts[a]	31	Out of mechanical adjustment[b]	3
Aging/cyclic fatigue[b]	26	Foreign/incorrect material[c]	2
Abnormal stress[b]	15	Particle contamination[b]	2
Previous repair/installation status[c]	13	Contacts burned/pitted/corroded[b]	1
Incorrect action[c]	12	Foreign/wrong part[c]	1
Corrosion[b]	11	Incorrect procedure[c]	1
Insulation breakdown[b]	11		

Combined categories:
[a]Resistance temperature device or circuit: 208 reports.
[b]Aging related: 159 reports.
[c]Personnel related: 29 reports.
Source: Adapted from *Aging of Nuclear Plant Resistance Temperature Detectors*, NUREG/CR-5560, Nuclear Regulatory Commission, June 1990.

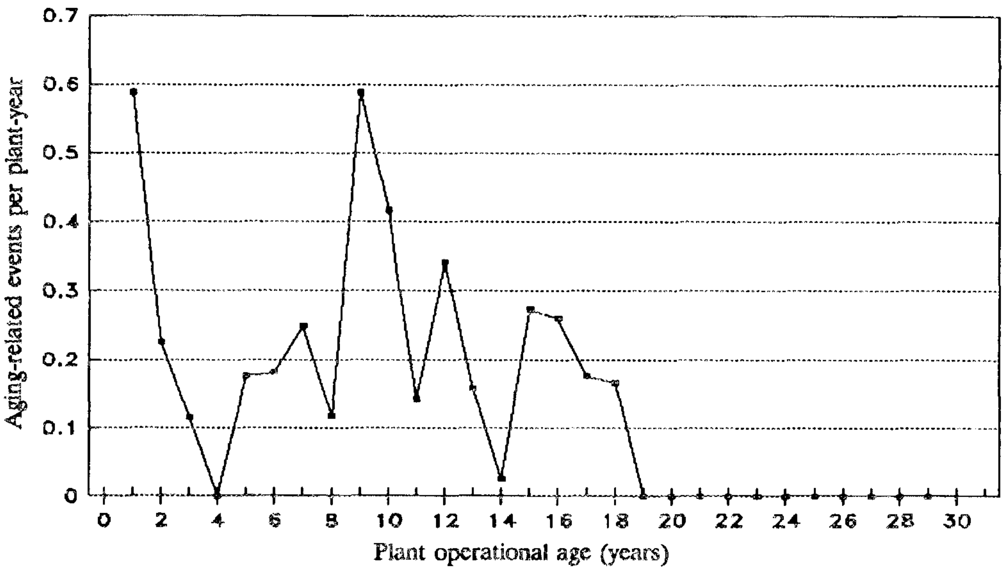


Fig. 5.4. Controller module aging failures by plant age (1984–1988).

Table 5.4. Controller module aging failures by plant system

System	Module failures (%)	System	Module failures (%)
Auxiliary/main feedwater	25	Nuclear	6
Turbine-generator	17	High-pressure coolant injection	5
Steam	11	Residual heat removal	4
Steam generator	10	Control rod drive	2
Reactor core isolation cooling system	7	Others	6
Reactor cooling system	7		

aging-related events reviewed and about 0.6% of all the LERs written during this study period. The plot of the normalized failure data showed an infant mortality during the first few years and a sharp peak in plant year eight. The curve follows a Type II pattern, perhaps indicating a typical service life of 8 years (see Fig. 5.5). Three failures in the single 8-year-old plant operational in 1986 accounted for the sharp peak in the eighth year of plant operation (see Table C.4 and Figs. C.15 through C.18 in Appendix C). An average failure rate calculated by dividing the total number of aging-related events (79) by the total number of plant-years (471) is 0.13 aging-related failures per plant-year, or one transmitter module aging-related failure per operating plant every 5.9 years.

Between 1985 and 1986, the total number of transmitter aging-related events rose 30% above the average for the study (see Table C.4). This could have been due to the well-publicized problems associated with the Rosemount transmitters (see Sect. 6.3). That problem resulted in many transmitters being replaced with newer models, hence an improved service life (see Sect. 3.9.1). In all, eleven different plant systems were represented where aging-related transmitter failures were reported (see Table 5.5). While the ESF systems involved are in the lower half of the list, they cumulatively represent about 22% of the data. However, no serious operating experiences were reported, other than for the Rosemount issue.

A 1986 investigative report described analyzed failures in pressure transmitters and stated that from September 1982 to April 1984 the failure rate, as reported in LERs, was very low.[40] It further stated that the LERs indicated that in this 3-year period, ~330 reportable failures occurred throughout the nuclear power industry in 65 operating plants. Quoting from the report, "These numbers indicate that less than two reportable failures occurred per year per plant and that the failure rate was 0.02 failures/year (2.4×10^{-6} failures/h) for each transmitter." In a more recent study,[18] a survey of 8 years of LERs was conducted, with 498 of 1325 found to be categorized as aging-related pressure instrumentation problems, a significant part of the total (see Fig. 5.6). The above shows how various studies can categorize differently what are judged to be aging-related effects. The database compiled for this study contains information from LERs from 1984 through 1988 and extends the time frame for the results and conclusions expressed in the earlier reports.

The authors of reference 40 also interviewed utility personnel in regard to calibration shift. The utility personnel did not consider the shifts that had been observed to be excessive. However, in some older plants, set-point drift of pressure switches was

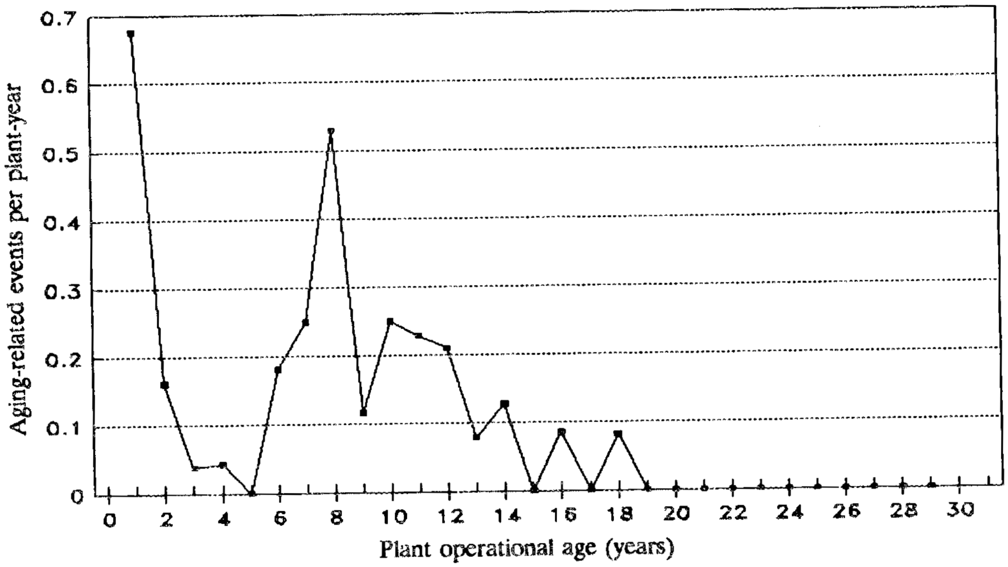


Fig. 5.5. Transmitter module aging failures by plant age (1984–1988).

Table 5.5. Transmitter module aging failures by plant system

System	Module failures (%)	System	Module failures (%)
Reactor cooling	22	Emergency feedwater indication and control	6
Main feedwater	12	Injection	5
Steam	11	Reactor core isolation cooling	5
Steam generator	8	Reactor water cleanup	5
Turbine-generator	8	Others	6
Auixilary power	6		
High pressure coolant injection	6		

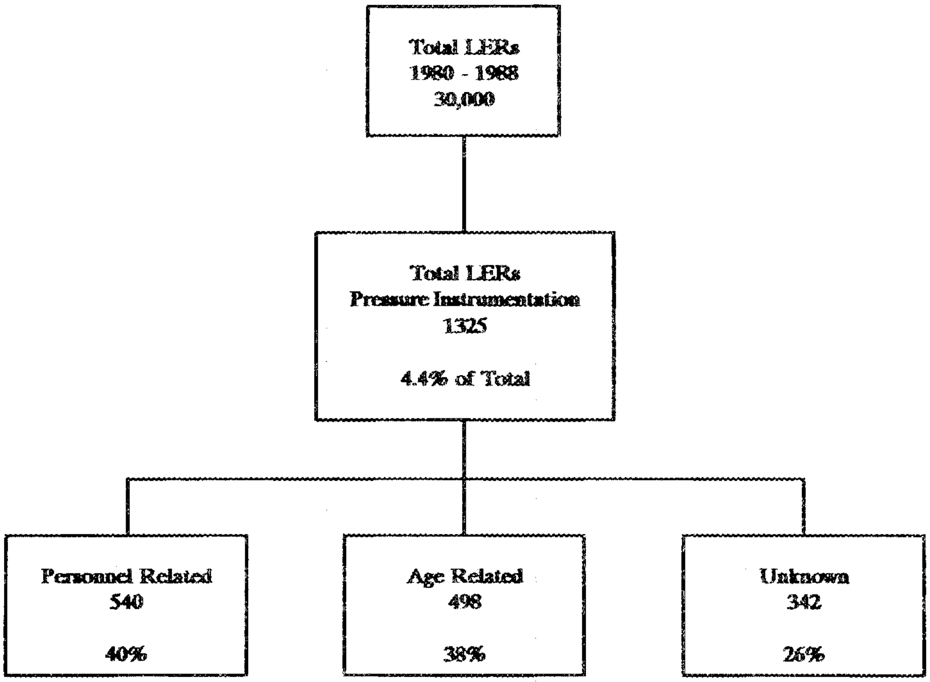


Fig. 5.6. Relative occurrence rates of aging-related problems between September 1982 and April 1984 for various stressors on transmitters. *Source*: Based on Fig. 4.1 in *Effect of Aging on Response Time of Nuclear Plant Pressure Sensors*, NUREG/CR-5383, Nuclear Regulatory Commission, June 1989.

one of the factors leading to their replacement by pressure transmitters that were not prone to the problem. Similar observations concerned with the end of service life for certain transmitters were made during the review of operating experiences where one plant had Fischer-Porter flow transmitters and Barton pressure transmitters[41,42] and another had Gould pressure transmitters.[43,44] (For further discussion on set-point drift see Sect. 7.4.)

Sandia National Laboratory has performed several aging studies on nuclear safety-related equipment. Included was an experimental study with five Barton Model 763 pressure transmitters.[45] These transmitters were tested to determine the failure and degradation modes in separate and simultaneous environmental exposures. The study showed that temperature is the primary environmental stressor affecting the static performance of the Barton transmitters tested. Also performed at Sandia was work on several Barton and Foxboro pressure transmitters which were removed from Beznau Nuclear Power Station in Switzerland and sent to Sandia for testing.[46] These transmitters had aged naturally in the plant for eight to twelve years. The experimental work showed that although some degradation had occurred, the performance of the transmitters remained satisfactory.

Another study performed under the NPAR program involved an evaluation of the stresses that cause degradation in nuclear plant pressure transmitters.[40] The report on this work describes a means of detecting and evaluating the degradation of pressure transmitters and concluded that the major consequences of the stresses on pressure transmitters are calibration shifts. The results from this study of operating experiences for transmitter modules indicate that no other aging-related problem exists that was or could be affecting the performance of transmitter modules in safety-related systems. The study indicated that transmitters are relatively stable devices under normal power operations and illustrated the benefits from a prevailing practice in the industry of replacement before failure.

5.5 ANNUNCIATORS

Malfunction of annunciators is quite apparent (especially the visual units), and at the first indication of a malfunction, service is applied (lamps tested or changed out) by the operator or maintenance is ordered. However, unless such maintenance is the result of a safety-related event or the annunciator itself causes such an event, the only records available are in the plant maintenance log books. When one considers the hundreds of annunciators in a typical plant and the sparsity of aging-related operating experiences reported in LERs, one can easily surmise that early detection by the operator is a principal reason for the scarcity of aging-related events in the national database. The structured search of the LER database retrieved 101 failure events for annunciator modules, of which 17 were judged to be aging related. These represented 3% of all aging-related events reviewed and about 0.1% of all the LERs written during this study period. The normalized failure data are plotted in Fig. 5.7 and show a Type III pattern. The data were sparse and randomly distributed throughout the study period (see Table C.5 and Figs. C.19 through C.22 in Appendix C). The random distribution of the data might best be attributed to the prompt if not constant attention given to these modules by the operators, resulting in much preventive maintenance (PM) service. Proper PM can prolong the service life of annunciators to approximately that of the plant; therefore, aging-related failures are indeed random. However, it is likely that during a control room redesign or as part of a licensing renewal effort, many of these modules would be replaced as part of an updating process. A failure rate calculated by dividing the total number of aging-related events (17) by the total number of plant-years (471) is 0.04 aging-related failures per plant-year, or one annunciator module aging-related failure per operating plant every 25 years.

Table 5.6 was constructed from this study's database and lists the annunciator module aging failures by plant systems. The results from this study of operating experiences for annunciator modules indicated that no apparent aging-related problem exists that was or could be affecting annunciator module performance in safety-related systems.

5.6 RECORDERS

Much the same can be said for recorder modules as was said for annunciator modules. These modules are referred to several times every shift and, consequently, service is prompt— hence, the relatively low rate of aging-related failure data (see Table C.6 and Figs. C.23 through C.26 in Appendix C). The normalized failure data plotted in Figure 5.8 show a Type III pattern and are not sufficient to hypothesize the

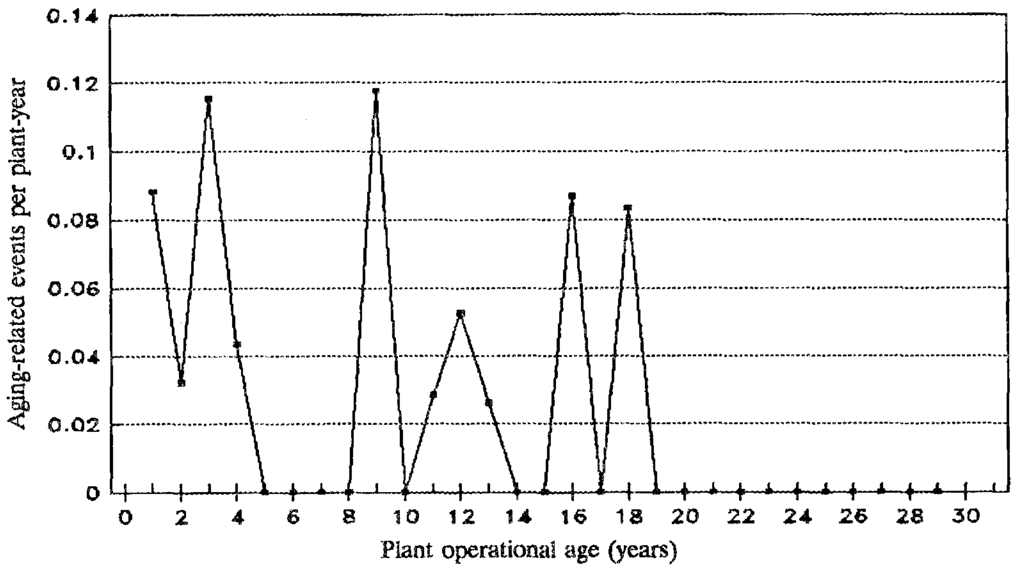


Fig. 5.7. Annunciator module aging failures by plant age (1984–1988).

Table 5.6. Annunciator module aging failures by plant system

System	Module failures (%)
Feedwater	21
Nuclear	17
Reactor cooling system	12
Radiation monitoring	12
Reactor core isolation cooling	5
Reactor water cleanup	5
Others	28

circumstances of failure. The more frequently identified systems involved in the reports for aging-related recorder failures are listed in Table 5.7, where no ESF system was identified.

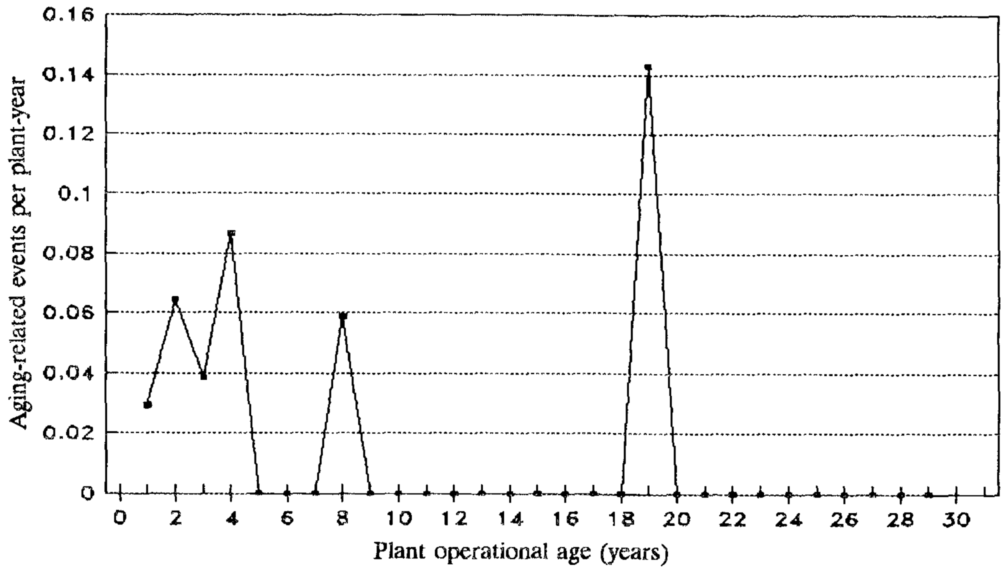


Fig. 5.8. Recorder module aging failures by plant age (1984–1988).

Table 5.7. Recorder module aging failures by plant system

System	Module failures (%)
Feedwater	38
Turbine generator	25
Reactor coolant	13
Radiation monitoring	12
Seismic	12

A major portion of the operating experiences for this study were collected in 1985 for both the LER and NPRDS databases. These experiences could be related to the debugging in new installations of modules and cabinets that resulted from the control room design reviews.[36] Problems associated with the design, installation, and implementation of the Safety Parameter Display System were also noted in the literature. These are not the result of aging-related module failures but are the consequences of actions taken in response to the aging stressor (see Sect. 3.10).

6. EXAMPLES OF AGING

Some significant plant experiences about aging of I&C modules form the basis for this section. The manifestations of these problems, the potential consequences, and the approaches taken to solve the problems provide material of interest to this study.

6.1 ELEVATED TEMPERATURES INSIDE INSTRUMENT CABINETS

Elevated temperatures can unknowingly exist in inadequately cooled instrument cabinets, and a pattern of component failures had to be recognized before the effect of the age-accelerating stressor was identified. The atmosphere in I&C cabinets exhibits elevated (from ambient) temperature due to the heat generated by components inside. Inadequate ventilation exacerbates the temperature buildup. This heat can affect insulation resilience and can cause bearing lubrication breakdown, wear out of mechanical components, and set-point drift. The temperatures experienced by electronic components in inadequately ventilated cabinets can be much higher than both the ambient room temperature and the temperature within the cabinets and can exceed the design limit for some of the components in the cabinets. Determining that elevated room temperatures or instrument cabinet internal temperatures is the root cause of the failure of some electronic components has not been immediate or easy. For example, some licensees experienced several failures over an extended period during which time many corrective actions were attempted before identifying overheating of components as the underlying reason for many of the failures they had experienced. Conformance with regulatory requirements regarding equipment environmental qualification has been assumed because these instrument cabinets are normally located in areas classified as mild environment. Hence, consideration of aging has, in general, not been considered for these cabinets and their components. Usually, even when elevated ambient temperature and/or inadequate cooling was eventually established as the principal root cause of many of the electronic component failures, several years of review were involved before the cause was determined and corrective action programs established.

Example 1

Elevated temperatures inside I&C control room cabinets that went undetected by plant personnel accelerated the degradation of heat-sensitive components. The unexpected failure of the components perturbed an operating plant.[19,47] Measurements taken inside the cabinets revealed temperatures 23–29°C (42–52°F) above the control room ambient temperature.[17] Following this finding, airflow in the control area ventilation system was rebalanced to provide additional cooling to the cabinets. In the five years preceding this event, 35 card failures had been experienced. During the following five months, an additional 13 cards failed, indicating that degradation of components in these cabinets was more widely distributed than had been thought. Elevated temperatures set in motion a degradation process that continues after the problem is corrected, as illustrated by these events. Therefore, other heat-sensitive components that were not replaced can be expected to have a shorter than normal service life.

A similar example occurred at another plant, where the event was attributed to a component breakdown in a circuit board that was exposed to the normal cabinet ambient conditions. The licensee concluded that this circuit board had deteriorated and experienced intermittent failure even though the licensee believed that the cabinet internal temperatures were within their design limits.[48]

Example 2

A pattern of failures must be observed before a problem is identified and after the root cause for accelerated aging is found. In this example, a malfunctioning power supply was sent to the manufacturer for analysis, where it was determined that an inadequate design of the steam and feedwater rupture control system (SFRCS) cabinets had resulted in overheating of the power supplies in these cabinets; and the resultant overheating was a primary contributing factor that has led to the failures of the power supplies.[47,49] The plant had been experiencing power supply and other component failures in the SFRCS cabinets since 1979; yet, only by late 1984 did the licensee determine that the root cause of the failures was overheating of the components. Another example that took multiple occurrences before the aging cause was recognized is an event caused by an inadvertent actuation of a primary containment isolation logic system, where the problem was traced to a specific model relay.[50] Aging was suspected, and a review of plant maintenance records for the past two years revealed a high failure rate in a population of about 200 relays during a two-year period. Aging was believed to have been aggravated by elevated cabinet temperatures; however, neither the failure mechanism nor the cabinet temperatures was given in the report. A program was then put in place to replace all these relays that are in safety-related applications.

6.2 ELEVATED CONTAINMENT BUILDING TEMPERATURES

Example 1

Temperatures in containment buildings have been found to exceed the design specifications for I&C modules. NRC inspectors at one plant noted that the average containment building temperature was 78 to 83°C (140 to 150°F) higher than the design temperature used in the Safety Analysis Report (SAR) and 11°C (20°F) higher than that assumed for equipment qualification during normal service life.[51] It was then determined that such temperatures had existed for the past 13 years, that is, since plant start-up.

Because the plant had been operating at elevated containment temperatures for an extended period, the NRC staff had several concerns—one being that the higher temperatures implied accelerated aging of equipment required for postaccident safe shutdown in accordance with regulation 10 CFR, Pt. 50.49, on equipment qualification. NRC sent an inspection team to investigate the effects of these high temperatures.[52]

Safety-related equipment affected by higher temperature levels than that used to qualify the equipment is listed in Table 6.1, which shows the life predicted at the original expected temperature of 49°C (120°F), life predicted at the higher temperatures

Table 6.1. Calculated component qualified life
(All values are in years.)

I&C module	Predicted life at 49°C (120°F)	Revised life at 65°C (150°F)	Remaining life at 65°C (150°F)
Rosemount pressure transmitter	10	2.3	0.25
Acoustic monitor preamplifier	4	1.04	0.4
High point vents solenoid valves	>40	6.0	2.2
Rosemount hot leg RTD[a]	>40	>40	27
Reactor vessel level detector	30	30	29
Acoustic monitor sensor	>40	>40	33
Conax hot leg RTD[a]	>40	>40	38
Radiation monitor	>40	>40	40

[a]RTD = resistance temperature device.
Source: NRC Inspection Report, Arkansas Nuclear One, Unit 1, 50-313/87-29, U.S. Nuclear Regulatory Commission, Dec. 12, 1987.

actually prevailing, and the life remaining at the higher temperatures.[52] Consequently, the replacement dates for some modules were advanced, and for others, replacement took place at the next refueling outage. The licensees were requested to make a temperature survey of their containment areas. The major findings are given below.[52]

- Some pressurized water reactors (PWRs) experienced high containment temperatures, but the licensees failed to recognize the safety significance and take corrective actions.

- Areas were found in some plants where the local ambient temperature exceeded that specified for the equipment qualification. Hot spots existed even when the area temperature, as measured by a limited number of sensors, was lower than the maximum specified in the qualification report for the equipment.

- Such modules that could be affected are (1) sensors for flow, level, pressure, and temperature; (2) valve operators and limit switches; and (3) pressurizer relief valve positions and relief flow monitors.

On December 23, 1987, NRC issued an Information Notice alerting all licensees to the potential problems resulting from operating a plant beyond its analyzed basis.[53]

Example 2

High-temperature reactor cooling water sometimes inadvertently backflows. This flow raises the temperature of associated piping, and the ambient temperature exceeds the qualification temperature for local I&C modules. Therefore, other plant systems to monitor for accelerated aging-related problems are the emergency feedwater system (EFS) and the emergency feedwater initiation and control (EFIC) system. Several occurrences of elevated temperatures in parts of the EFS were reported at one plant, causing system piping temperatures to exceed design conditions.[54-57] Damage to mechanical and structural components drew attention to the problem. Again, a tracking of I&C maintenance records for flow, pressure, and temperature sensors could determine whether an accelerated aging factor exists and if so, based on experience, could indicate when problems could be expected to surface.

6.3 PRESSURE TRANSMITTER AGING PROBLEM

An example of aging accelerated by temperature and pressure was the case for some differential pressure transmitters manufactured by Rosemount, Inc., when the modules would not maintain their calibration.[58] One problem occurred when those transmitters were exposed to excessive overpressure or reverse pressure. Another example was when the operating environmental temperature for the transmitter was elevated above the design value.[59] This condition would result in reduced performance prior to a detectable failure. This reduced performance manifested itself as an output shift, or as an altered scale factor, and/or as an increase in response time. After attempts to recalibrate failed, the malfunctioning modules were returned to the manufacturer for analysis, where the failure cause was determined to be degradation of the seal (gradual loss of fluid) in the transmitter sensing unit.

6.4 LIGHTNING INFLUENCES VIA GROUNDING

It is commonly assumed that when lightning strikes the electric grid, an area near the power plant, or substation/switchyard, the design of the electrical power distribution system will be adequate to divert and/or suppress the induced impulse voltage. However, haphazard impulses have been known to circumvent protective devices and traverse other pathways to reach suitable ground. In such instances, the consequences have ranged from perturbation of normal plant operations to a system disturbance involving catastrophic failure of some equipment.

A recent example illustrates how the grounding system caused I&C modules important to safety to be stressed by nearby lightning strikes. A reactor trip and damage to plant instrumentation occurred when a lightning strike to the containment building was apparently conducted to ground through the containment penetrations.[60] The induced potentials in the cables passing through these penetrations were high enough to damage many modules in both safety-related systems and the balance-of-plant instrumentation. Several failed catastrophically, and many more were undoubtedly stressed. Similar cases are described in references 61 through 65. These cases definitely show lightning to be a stressor and as such to have an adverse effect on the environmental qualification and useful service life of I&C modules.

7. GENERALIZATION OF FINDINGS

The problems treated in this report can be diagnosed at various levels ranging from the system and its interface with the operator to the microstructure of components on circuit boards. In the middle of this range are the basic components and their clustering into equipment modules. Unfortunately, failures are often analyzed at only these intermediate component levels in LERs. Thus, it is often difficult to determine the root causes for aging-related failure because events are not examined at a sufficiently low level. Also, no information would be given as to whether the problem was aging related unless correlation of the event with a stressor happens to be possible and happens to be written into the failure report.

Analysis of instrument failure databases has been reported to be difficult by previous authors. This is still found to be the case here, especially when the interest is in aging phenomena. Interestingly, the difficulties cited for the LER data collection system in the 1970s still exist today.[66]

- Incomplete information stems from nonuniformities in reporting.
- Incipient failure detections of importance to aging are sometimes not reported.
- Evolutions in reporting requirements influence consistency.

This investigation bears out the findings of previous investigators.

- Major existing databases have shortcomings in tracking I&C module aging.
- A primarily regulatory tool cannot give the type of statistically reliable engineering information desired in aging studies, although qualitative information obtained can be useful.

7.1 TECHNOLOGICAL OBSOLESCENCE

Today's nuclear power plants in the United States are still using I&C technology that was available in the 1960s, despite remarkable improvements in this technology that have increased reliability, performance, and maintainability at a reduction in cost and physical size of equipment. The result has been a critical shortage of replacement parts for I&C systems and potentially more important, a loss of the supporting infrastructure. Many parts necessary for repair of equipment still being used are no longer available or are available from only a single source. This problem of technological obsolescence is addressed in Sect. 3.10.

Maturation and change of equipment requirements over long time periods can also contribute to technological obsolescence. Methods and understanding of control processes are also evolutionary and lead to improvements in technology. System performance and safety requirements change and mature with experience that brings about corresponding evolutionary changes in equipment. As this happens, equipment in place and being used becomes outdated.

With a lack of spare parts, compromises may be made during preventive maintenance that would not otherwise be made. Replacement of components for failure prevention may not even be possible. As a result, those modules affected may fail prematurely or perform poorly.

Few upgrades have been made in response to extreme performance deficiencies or by regulatory requirement mandates. Examples of instrumentation upgrades are to be found in the design change documentation at the plants.[67] These have been as extensive as changing from an analog to a digital reactor protection system to digitizing the feedwater control system at a boiling water reactor (BWR).[29, 68-70] Simple replacements of individual modules have improved performance.

7.2 DATA OBFUSCATION

The obvious and expected direct correlation between I&C module aging-related failures and the age of the plant could not be clearly determined from the available data in this study. The quantitative data was diffused by:

- industry practice of replacing those modules that required increased service or repair (when no failure occurred, no failure report was required),
- unclear history of failed modules in event reports because of difficulty in identifying past module replacement, and
- changes and replacements made to modules because of required environmental qualification.

All of the above were compounded by the modularity of I&C modules that provide easy surveillance, preventive maintenance, and changeout.

Plant-specific data are the most desirable (when obtainable) because of the availability of maintenance histories associated with the failed components. An additional feature of plant-specific data is the ability to identify plant-specific environmental and human contributors to aging-related failures. High incidence of failure of a particular module, for example, may indicate weakness in a specific design or merely a change in the system's maintenance procedure or mode of operation.

7.3 SOME UTILITY APPROACHES TO INSTRUMENTATION AND CONTROL PROBLEMS

Current practices of utilities for finding solutions to problems such as corrective actions for specific events center around internal group consultations and following historical precedents. However, when problems involve broader issues that get management attention, the efforts are more extensive: ad hoc internal task forces, owners' group activities, and outside organization assistance that may even extend to long-term research and development (R&D).[71]

A special sensitivity to problem areas at nuclear plants exists, as exemplified by events. This sensitivity arises from (1) how business concerning safety at these plants is conducted, (2) company desires to minimize costs, and (3) pressures from outside the utility [e.g., NRC and the Institute of Nuclear Power Operations (INPO)] to identify and correct problems. Identifying, reporting, and solving plant problems has always been and continues to be a major effort.

Corrective actions are always taken after an event. In reading documentation of these corrective actions, one almost always finds relatively simple, ad hoc solutions (e.g., more training or procedural change). Prior occurrences of similar problems at the plant in question or elsewhere and unsuccessful prior implementations of such solutions may not be fully recognized. As a result of this tradition, numerous small improvements are continually being made at plants, thus reducing probabilities of future incidents—especially those identical to incidents that have already occurred. The industry practices replacement of troublesome components before failure. However, this practice often goes unreported in the databases and masks the areas requiring general attention. Generic solutions that would simultaneouly address varieties of problems are not commonly invoked.[52]

7.4 SET-POINT DRIFT

Instrumentation and control systems in nuclear plants are typically provided with adjustable set points where specific actions are initiated. Each of these adjustable set points is assigned a preset value. Set-point drift is the unplanned change in these preset values. When the change is of sufficient magnitude to cause the set point to fall outside specified limits, the event may be classified as an abnormal occurrence that must be reported to NRC. A 1977 report credited set-point drift problems as being "influenced by the initial selection of the instruments, their range, application, calibration, operation, and maintenance procedures."[66] Aging was not considered as a factor in either that report or a 1974 report.[31] The author of the 1974 study of set-point drifts within safety-related instrumentation reported the following observations.

1. Approximately 10% of all abnormal occurrences reported by nuclear power plant licensees involved unplanned changes in the set points of protective instrumentation.

2. Most reported occurrences took place in BWRs.

3. Pressure instrumentation accounted for most of the set-point drifts (69.8%); 11% involved liquid level devices, and nearly 5% involved time-delay devices. Also, 12.6% of the reported occurrences involved temperature instruments; <0.5% involved off-gas radiation monitoring equipment, and 1.4% occurred in nuclear instrumentation.

4. All set-point drifts were discovered during routine surveillance testing [different from I&C failures that are not discovered by testing (see Sect. 7.5)].

5. The most prevalent reason for set-point drift was the use of set points that did not allow sufficient margin for normal instrument error.

Further observations were reported in the 1977 study.[66]

1. Most (51.3%) of all set-point drift events occurred in pressure sensors or pressure-related instrumentation.

2. Pressure sensors accounted for the highest percentage of PWR and BWR set-point problems (38.7% and 57.2% respectively).

Both studies[31,66] agree that most set-point drift problems occur in pressure devices. Level indicators are the second largest contributors to set-point drift problems.

The term *drift* is often used erroneously as a synonym for the term *aging-related*. Set-point drift is sometimes erroneously considered an indication of an aging-related problem. Although modules with aging-related problems may be more likely to have excessive drift problems, set-point drift does not reliably indicate that an age-related problem exists. Set-point drift is affected by the module's design, application, calibration, and maintenance and operation procedures as well as aging-related problems. A search of the LER database was completed to analyze the problem of set-point drift. The search retrieved 370 events, 2.7% of all LERs written during this five-year period, where drift was the cause. A review and analysis of each event determined that of these 370 events, only 74 actually failed because of an aging-related problem. The data are plotted by plant age in Fig. 7.1.

The data show that set-point drift is not an effective indicator of I&C module aging. Reinforcing this conclusion, an article published by *Analog Dialogue*[72] states "long-term instability (assuming no long-term deterioration of some damaged component within the device) is a *drunkard's walk* function; what a device did during its last 1,000 hours is no guide to its behavior during the next thousand." It further states ". . . as a device gets older, the stresses of manufacture tend to diminish and the device becomes more stable (except for incipient failure sources)."

7.5 TESTING

Extensive testing of I&C equipment is required by plant technical specifications to ensure that I&C failures are discovered as soon as possible. The purpose of this testing is to minimize the time over which the plant operates with a reduced level of redundancy in the plant protection system. The potential effects of I&C failure is that either

1. an instrument failure produces a trip signal that, in coincidence with spurious signals on other instrument channels, results in a plant trip causing an unnecessary loss of plant availability; or

2. an instrument failure fails to produce a trip signal, thus reducing the redundancy of the protection system and, therefore, the safety of the system (the degree of decrease in redundancy is different for PWR and BWR plants).

Testing procedures cannot predict the occurrence of a forthcoming component failure or the effects of aging. However, if the test indicates marginal performance of a particular I&C system, then corrective action can be taken to prevent a failure from occurring during operation.

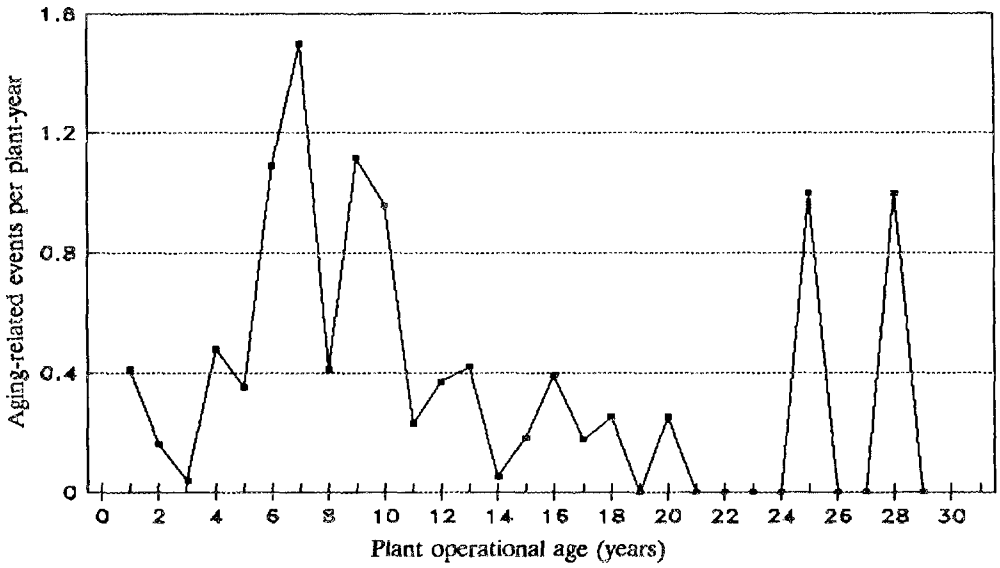


Fig. 7.1. Instrumentation and control module drift failure by plant age (1984–1988).

Standard technical specifications for each of the major instrumentation systems require that operability be demonstrated by the performance of channel checks, channel calibrations, and channel functional tests for specific plant operating modes. Modules in safety-related systems are required to be environmentally qualified and are subjected to more frequent surveillance and calibration tests than are similar models in the balance of plant. In general, the instrumentation for the reactor protection system, the engineered safety features actuation systems, and plant radiation monitoring are to have each channel checked at least once every 12 hours, each channel calibrated every 18 months, and each channel functionally tested every 31 days. For the remote shutdown and accident monitoring systems instrumentation, each channel is checked every 31 days and calibrated every 18 months. Because these are minimum requirements, plant-specific technical specifications could be expected to have more frequent tests and calibrations.

The I&C modules also receive frequent casual attention by the plant operators in the performance of their duties. According to the findings of this report, more module failures are discovered by operator detection than by routine surveillance testing. For example, in the indicators module category, many more module failures were discovered by operator observations (204) than by scheduled testing (28) (see Table D.2 in Appendix D). Only six failures were discovered during related maintenance. For transmitter modules, which are not as accessible to operator observation, this finding that scheduled surveillance testing is not the principal means of discovery for module failures is again true.

The ineffectiveness of existing surveillances in giving some indication of failures (as opposed to minor drifts) has been established by earlier analyses of LERs.[66] In that study of 977 failures due to aging as well as other effects, the breakdown of the methods of discovery was

	Operating	+	Shutdown	=	Total
Discoveries during testing	15%	+	13%	=	28%
Other discoveries	55%	+	17%	=	72%

In another investigation[33] based on 2795 LERs involving only drift, the results are

Drifts detected by scheduled tests	80%
Drifts detected by operators as abnormalities	20%

Another investigation, using 1401 LER events applicable to I&C, reports similar information.[73] A conclusion is that surveillance testing may detect significant degradations due to aging but not at incipient stages. Specific findings were

All faults including drifts discovered by testing	51%
Portion of a class made up of faults other than drifts discovered by testing (deduced here from data presented)	49%
Incipient failures reported	0%

It is concluded that scheduled surveillance tests are not effectively detecting potential aging-related breakdowns. As an explanation for these findings, one might surmise that I&C failures (excluding drift), tending to be more sudden than gradual, are more likely to be discovered by techniques outside existing simple surveillance test methods performed at discrete times.

7.6 MAINTENANCE

Closely associated with surveillance and testing is maintenance. How maintenance in its broad sense can help manage aging problems in all plant areas has already received a great deal of attention.[74] All of the following specific points, taken from reference 74, apply to I&C.

1. Identify the proper quantities to measure to give the best overall picture of a device's condition.

2. Have a data collection program for these quantities to detect early warnings of incipient failures.

3. Understand the trends of performance indicators insofar as their predictability of failures are concerned.

4. Develop databases of information and use these to deduce guidelines and criteria for useful life.

5. Make refurbishment or replacement judgments based upon deciding what functional capabilities are acceptable under normal conditions and under accident-mitigating conditions.

It is likely that the second item, though very worthwhile, could be difficult and might require R&D. Fundamental understandings of so-called *random* or spontaneous failures are needed. Currently, these are not generally predictable except in an approximate probabilistic sense from item 4. Properly developed and spaced preventive and predictive maintenance can extend the operating life of that system or component beyond the original design life. A basic premise is emphasizing predictive and preventive maintenance over curative maintenance. Too much maintenance can actually cause premature or unnecessary wear of some components or systems; therefore, the development of optimal preventive maintenance intervals is important.

7.7 INCIPIENT FAILURE MONITORING

Aging phenomena can proceed without an immediate obvious effect. Therefore, it is natural to think of trending as a monitoring technique where the test data are compared for changes from the original acceptance data. Such changes would indicate degradation or impending failure in an incipient stage before catastrophic failure. Many plants have implemented a practice like this in their preventive maintenance programs in recent years. The increased implementation of trending techniques is due to two factors: (1) encouragement along these lines from the NRC and INPO and (2) the ease of doing this with the growing popularity of computerized databases. This encouragement for plants to implement trending is quite general regarding types of equipment. A specific example of trending beneficially used is the discovery of sustained drifts in Rosemount transmitters[67] (discussed further in Sect. 6.3).

Historically, it is typical to use calibration data card files as a mechanism for trending, visually scanning manual entries for long-term effects. This approach is already technologically obsolete, judging by approaches already in use in other industries. For example, petrochemical plants have calibration results in databases on computers, with computerized data acquisition during tests facilitating data entry.[75]

Techniques for monitoring instrument condition are quite varied. The simplest and most used is trending the results of calibrations. More complex, but potentially much more powerful, is on-line multiple signal analyses.[76,77] Advantages of these methods are that (1) monitoring can be continuous rather than at discrete times; and (2) depending on the instrument and the analysis technique, varieties of static and dynamic properties can be obtained. Also, especially in the newer digital circuitry, built-in self-testing can be essentially ongoing.

Electronic components and circuits tend to fail catastrophically rather than by gradual degradation. Because of this factor, systems that monitor voltages or waveforms internal to the device may not be effective in predicting imminent circuit failure. However, indirect monitoring of environmental stressors acting on the device in question may be a more suitable means of predicting circuit failure. Unfortunately, threshold values above which greatly accelerated aging is likely to occur are probably not currently quantified and are likely to be device specific. The intensity and duration of aging stressors as well as their possible synergistic interaction in the circuit's environment must also be considered. Therefore, for important circuits, it may be desirable to determine threshold values for aging stressors and implement a means to perform real-time measurement of stressors acting on the circuit of interest.

An informal version of an on-line monitoring program is actually the ongoing observation by operators as they compare instruments and controllers with expected performance and listen for alarms. These are not as sensitive or encompassing as computer algorithms directed at accomplishing the same objective. However, human intelligence capabilities would not be found in computer algorithms.

7.8 INSPECTION

Inspection methods are applicable to monitoring aging. These are analogous to practices with mechanical equipment, especially rotating machinery. The established tradition of frequent surveillance tests conducted by I&C personnel right at the instruments and their cabinets affords numerous opportunities for informal inspections. With ample training and experience on what to look for, both regarding stressors and their effects, this can continue to be effective. The importance attached to observation of equipment by personnel can be seen especially in Japan.[14,78] Plants have formal, rigorously scheduled programs of inspections. Instrumentation receives its annual outage inspection as would piping and vessels. These inspections are based on using diverse types of personnel: shift operators, instrumentation specialists, management, and manufacturer staff. Moreover, some inspections require regulatory personnel involvement. The basis for this attention to inspections performed at levels beyond that found in the United States is required by Japanese law.

Historically, inspections of I&C equipment have been understood as being visual and not involving technological aids. However, techniques are in limited use for enhancing inspections such as (1) time-domain reflectometry along with impedance measurements for contact resistance changes and (2) infrared thermography for detection of circuit board components operating at elevated temperatures. A successful program using the first technique has been demonstrated on aging circuitry at Shippingport.[79] It was concluded that such measurements should be integrated into PM programs.

8. CONCLUSIONS, OBSERVATIONS, AND RECOMMENDATIONS

A study of the operating experience for six selected I&C modules (indicators, sensors, controllers, transmitters, annunciators, and recorders) throughout a five-year period, 1984 to 1988, was performed. Various operational and environmental stressors were identified, and the effects of aging due to each of these stressors were examined. The study relied on investigation of national databases and examination of the published literature from related aging studies. The bulk of the data was derived from LERs, which by nature report safety-significant events. Limitations of the study of which the reader should be aware are that (1) plant maintenance records were not examined, so detail is lacking on most events; (2) only failures discovered as a result of safety-significant occurrences are reported in the databases used; and (3) installation date of individual I&C modules is not known, so module age had to be assumed equal to plant age.

8.1 CONCLUSIONS

Three main conclusions were drawn from this study.

1. I&C modules make a modest contribution to safety-significant events (see Sect. 2.3).

   - 17% of LERs issued during 1984–1988 dealt with malfunctions of the six I&C modules studied.

   - 28% of the LERs dealing with these I&C module malfunctions were aging related (other studies show a range 25–50%).

2. Of the six I&C module categories studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures (see Table 2.2).

3. Infant mortality appears to be the dominant failure mode for most I&C module categories (with the exception of annunciators and recorders, which appear to fail randomly) (see Chap. 5).

4. Although aging is a contributor to instrument drift, it is not the sole source. Drift, in the databases used for this study, was not a reliable indicator of age degradation (see Sect. 7.4).

8.2 OBSERVATIONS

Three main observations were made in this study.

1. I&C modules in nuclear plants are replaced as often for reasons of technological obsolescence as for aging. This replacement practice obviously obfuscates aging

studies that rely on industry-wide databases (such as LERs), where the reason for module replacement is not usually given (see Sects. 3.10 and 7.1).

2. The issue of replacement in response to technological obsolescence [almost-one-for-one substitution of newer hardware as advances in technology and/or dwindling supplies of spare parts dictate] may have far greater impact on regulatory matters in the I&C area in the years ahead than the issue of aging (see Sects. 3.10 and 7.1).

3. Monitoring of environmental stressors acting on the module in question may be a suitable means of predicting approaching circuit failure. It may be possible to determine threshold values for aging stressors and perform real-time measurement of stressors acting on the circuit of interest. Systems that monitor voltages or waveforms internal to electronic devices may also be useful in predicting imminent circuit failure, but because of the tendency of electronic components to fail catastrophically rather than by gradual degradation, further development may be necessary to achieve this goal (see Sect. 7.7).

8.3 RECOMMENDATIONS

Three main recommendations were derived from this study.

1. Consideration should be given to methods that would be helpful in reducing the incidence of infant mortality, particularly for the indicators and sensors categories, which dominate aging-related I&C module malfunctions and failures. For example, an attempt should be made to find and identify marginal components prior to installation. Military standards provide good indication of those practices likely to be beneficial.

2. Consideration should be given to testing selected I&C modules for synergistic effects of aging stressors. The purpose would not be qualification of specific equipment but rather identification and quantification of generic stress and failure relationships. Tests would not be intended to demonstrate operating envelopes of any specific brand of equipment.

3. Consideration should be given to creating an industry-wide database dedicated to aging-related information. Earlier studies pointed out that existing databases, while reporting stressors, do not adequately indicate the root-cause failure mechanisms; this current study also encountered difficulty in drawing conclusions as a result of this deficiency. A good source of information for aging studies would be the maintenance records of the individual plants. An industry-wide, readily accessible database devoted specifically to aging-related events and information would provide a most helpful and efficient service for those interested in plant and equipment aging.

9. REFERENCES

1. *Nuclear Plant Aging Research (NPAR) Program Plan*, NUREG-1144, U.S. Nuclear Regulatory Commission, July 1985; Rev. 1, September 1987.

2. G. A. Arlotto, "Understanding Aging: A Key to Ensuring Safety," pp. 7–10 in *Proceedings of the International Conference on Nuclear Power Plant Aging, Availability Factor and Reliability Analysis*, ed. V. S. Goel, San Diego, July 7–11, 1985, American Society of Metals, 1985; *Nucl. Saf.* 28 (1), 46–51 (1987).

3. *NRC Research Program on Plant Aging: Listing and Summaries of Reports Issued Through May 1990*, NUREG-1377, Rev. 1, U.S. Nuclear Regulatory Commission, July 1990.

4. P. T. Jacobs, *An Interim Assessment of Reactor Protection Aging*, NUREG/CP-0082, U.S. Nuclear Regulatory Commission, 1986.

5. *Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants*, NUREG/CR-1740, U.S. Nuclear Regulatory Commission, July 1984.

6. *An Aging Failure Survey of Light Water Reactor Safety Systems and Components*, NUREG/CR-4747, Vol. 1, U.S. Nuclear Regulatory Commission, July 1987.

7. C. W. Mayo et al., *Improved Reliability for Analog Instrument and Control Systems*, Vols. 1 and 2, EPRI NP-4483, Electric Power Research Institute, 1986.

8. L. C. Meyer, *Nuclear Plant-Aging Research on Reactor Protection Systems*, NUREG/CR-4740, U.S. Nuclear Regulatory Commission, EGG-2467, Idaho National Engineering Laboratory, January 1988.

9. L. C. Oakes, W. B. Reuland, and C. D. Wilkinson, *Instrumentation and Control Strategies for Power Plants*, "Volume 2: Problem Definition and Recommendations," NSAC-153, Electric Power Research Institute, December 1990, p. 2-2.

10. A. Beranek, comp., *Proceedings of the International Nuclear Power Plant Aging Symposium*, Bethesda, Md., Aug. 30–Sept. 1, 1988, NUREG/CP-0100, U.S. Nuclear Regulatory Commission, March 1989.

11. *Proceedings of an International Symposium on Safety Aspects of the Ageing and Maintenance of Nuclear Power Plants*, Vienna, June 29–July 3, 1987, International Atomic Energy Agency, January 1988.

12. V. S. Goel, ed., *Proceedings of the International Conference on Nuclear Power Plant Aging, Availability Factor and Reliability Analysis*, San Diego, July 7–11, 1985, American Society of Metals, 1985.

13. M. J. Declerq, "A Methodology for Accelerated Aging of Electronic Systems," pp. 87–90 in *Proceedings of the International Conference on Nuclear Power Plant Aging, Availability Factor and Reliability Analysis*, San Diego, July 7–11, 1985, American Society of Metals, 1985.

14. T. Hattori, "Maintenance Management of Nuclear Power Plant in Japan: Present Situation of Preventive Maintenance," pp. 291–96 in *Proceedings of the International Nuclear Power Plant Aging Symposium*, Bethesda, Md., Aug. 30–Sept. 1, 1988, NUREG/CP-0100, U.S. Nuclear Regulatory Commission, March 1989.

15. M. Coute, G. Deletre, and J. Y. Henri, "Safety Aspects of Nuclear Power Plant Component Aging," pp. 177 in *Proceedings of the International Nuclear Power Plant Aging Symposium*, Bethesda, Md., Aug. 30–Sept. 1, 1988, NUREG/CP-0100, U.S. Nuclear Regulatory Commission, March 1989.

16. S. P. Carfagno and R. J. Gibson, *A Review of Equipment Aging Theory and Technology*, EPRI NP-1558, Electric Power Research Institute, September 1980.

17. *Potential Loss of Solid-State Instrumentation Following Failure of Control Room Cooling*, IE Information Notice 85-89, U.S. Nuclear Regulatory Commission, Nov. 19, 1985.

18. *Effect of Aging on Response Time of Nuclear Plant Pressure Sensors*, NUREG/CR-5383, U.S. Nuclear Regulatory Commission, June 1989.

19. *Control Area Ventilation Trains A and B Inoperable, McGuire Unit 1*, Licensee Event Report 369/84-018, U.S. Nuclear Regulatory Commission, March 22, 1985.

20. NRC Inspection Report, Noncompliances and Violations Noted: Burnt Wiring Observed on Safety-Related Limitorque Motor Operators Due to Close Proximity to or Contact W-Limit Switch Compartment Space Heaters, 99900100-86-01, pp. 3–7, U.S. Nuclear Regulatory Commission, Nov. 7, 1986.

21. *Systems Interaction Events Resulting in Reactor System Safety Relief Valve Opening Following a Fire-Protection Deluge System Malfunction*, IE Information Notice 85-85, U.S. Nuclear Regulatory Commission, Oct. 31, 1985.

22. *Actuation of Fire Suppression System Causing Inoperability of Safety Related Equipment*, IE Information Notice 83-41, U.S. Nuclear Regulatory Commission, June 22, 1983.

23. *Supplement 2: Feedwater Line Break*, IE Information Notice 86-106, U.S. Nuclear Regulatory Commission, March 18, 1987.

24. L. E. C. Hughes and F. W. Holland, *Electronic Engineer's Reference Book*, 3d ed., Haywood Books, London, 1967, pp. 609–21.

25. NRC Inspection Reports, Fort Calhoun Station, Unit 1, 50-285/87-27 and 50-285/87-30, Instrument Air Water Intrusion Event (42701), U.S. Nuclear Regulatory Commission, October 1987.

26. D. L. Shurman, "Maintenance Effectiveness Technical Area Summary," pp. 353–56 in *Proceedings of the 1988 IEEE Fourth Conference on Human Factors and Power Plant,* June 5–9, 1988, IEEE Publication 88CH2576-7, Institute of Electrical and Electronics Engineers, June 1988.

27. NRC Memo, R. W. Houston to T. M. Novzk, Hope Creek Generating Station–Documentation and Evaluation of the Results of Review Meeting Regarding Electromagnetic Interference Disabling Baily 862 Solid State Logic Modules, U.S. Nuclear Regulatory Commission, March 22, 1985.

28. *Lightning Strikes at Nuclear Power Generating Stations*, IE Information Notice 85-86, U.S. Nuclear Regulatory Commission, Nov. 5, 1985.

29. Memorandum from S. Newberry to J. F. Stolz, Haddam Neck–Reactor Protection System Upgrade (TAC NO. 66948) Phase One, Docket 50-213, Feb. 26, 1990.

30. *Unanticipated Equipment Actuations Following Restoration of Power to Rosemount Transmitter Trip Units*, Information Notice 90-22, U.S. Nuclear Regulatory Commission, March 23, 1990.

31. R. A. Hartfield, *Setpoint Drift in Nuclear Power Plant Safety-Related Instrumentation,* 00E-ES-003, Office of Operations Evaluation, U.S. Atomic Energy Commission, August 1974.

32. NRC Inspection Report, Cooper Nuclear Station, 50-298/88-28, U.S. Nuclear Regulatory Commission, Oct. 21, 1988.

33. Relaxation of Staff Position in Generic Letter 83-29, Item 2.2, Part 2, "Vendor Interface for Safety-Related Components," Generic Letter 90-03 to all power reactor licensees, U.S. Nuclear Regulatory Commission, March 20, 1990.

34. Items of Interest for Week Ending November 9, 1989, Office of Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission.

35. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. NRC Regulatory Guide 1.97, U.S. Nuclear Regulatory Commission, 1980.

36. *Guidelines for Control Room Design Reviews*, NUREG-0700, U.S. Nuclear Regulatory Commission, September 1981.

37. "Qualification of Class 1E Equipment for Nuclear Power Plants," U.S. NRC Regulatory Guide 1.89, U.S. Nuclear Regulatory Commission, 1974.

38. "Pressure Instrumentation Terminology," Standard ISA-S51.1, Instrument Society of America, 1976.

39. *Aging of Nuclear Plant Resistance Temperature Detectors*, NUREG/CR-5560, U.S. Nuclear Regulatory Commission, June 1990.

40. *Inspection, Surveillance and Monitoring of Electrical Equipment in Nuclear Power Plants*, Vol. 2, *Pressure Transmitters*, NUREG/CR-4257, U.S. Nuclear Regulatory Commission, August 1986.

41. *Inadvertent Reactor Trip Due to Steam Flow Transmitter 1FT-512, Zion, Unit 1*, Licensee Event Report 295/85-44, U.S. Nuclear Regulatory Commission, Jan. 6, 1986.

42. NRC Inspection Report, Zion Unit 1, 50-245/88-03, U.S. Nuclear Regulatory Commission, Aug. 12, 1988.

43. NRC Inspection Reports, Rancho Seco Nuclear Generating Station, 50-312/88-33, Sept. 28, 1988, and 50-312/88-42, Feb. 13, 1989, U.S. Nuclear Regulatory Commission.

44. NRC Inspection Reports, Rancho Seco Nuclear Generating Station, 50-312/88-23, Sept. 28, 1988, and 50-312/88-42, Feb. 13, 1989, U.S. Nuclear Regulatory Commission.

45. D. T. Furgal, C. M. Craft, and E. A. Salzar, *Assessment of Class 1E Pressure Transmitter Response When Subjected to Harsh Environment Screening Tests*, NUREG/CR-3863, U.S. Nuclear Regulatory Commission, SAND 84-1264, Sandia National Laboratories, March 1985.

46. J. W. Grossman and T. W. Gilmore, *Evaluation of Ambient Aged Electronic Transmitters from Beznau Nuclear Power Station*, NUREG/CR-4854, U.S. Nuclear Regulatory Commission, SAND 86-2961, Sandia National Laboratories, May 1988 draft, (available in NRC Public Document Room, 2120 L Street, NW, Washington, DC 20555).

47. *Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems*, AEOD/C604, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 1986.

48. *Reactor Building Pressure Indicator Failure, Virgil C. Summer Station*, Licensee Event Report 395/82-16, U.S. Nuclear Regulatory Commission, Jan. 13, 1983.

49. *Steam and Feedwater Rupture Control System Channel 3, 48 Volt DC/DC Power Supply Lost Causing a Full Trip, Davis-Besse*, Licensee Event Report 346/82-51, U.S. Nuclear Regulatory Commission, Feb. 10, 1986.

50. NRC Safety Inspection Report, Pilgrim Nuclear Power Station Unit 1, 50-293/88-19, U.S. Nuclear Regulatory Commission, June 29, 1989.

51. NRC Inspection Report, Arkansas Nuclear One, Unit 1, 50-313/87-27, U.S. Nuclear Regulatory Commission, Sept. 21, 1987.

52. NRC Inspection Report, Arkansas Nuclear One, Unit 1, 50-313/87-29, U.S. Nuclear Regulatory Commission, Dec. 12, 1987.

53. *Plant Operation Beyond Analyzed Conditions*, IE Information Notice 87-65, U.S. Nuclear Regulatory Commission, Dec. 23, 1987.

54. NRC Inspection Report, Crystal River Unit 3, 50-302/88-35, U.S. Nuclear Regulatory Commission, March 27, 1989.

55. *Small Leaks in Emergency Feedwater System Valves Lead to Elevated System Temperatures and Exceeding Piping Design Basis, Crystal River, Unit 3*, Licensee Event Report 302/88-14, U.S. Nuclear Regulatory Commission, Jan. 23, 1989.

56. NRC Systematic Assessment of Licensee Performance, Crystal River, Unit 3, 50-302/88-35, U.S. Nuclear Regulatory Commission, September 1987–December 1988.

57. Response to NRC Inspection Report 50-302/88-18 by Florida Power Corp., Sept. 16, 1988.

58. *Potential Misapplication of Rosemount, Inc., Models 1151 and 1152 Pressure Transmitters with Either "A" or "B" Output Codes*, IE Bulletin 80-16, U.S. Nuclear Regulatory Commission, June 27, 1980.

59. *Operational Deficiencies in Rosemount Model 510 DU Trip Units and Model 1152 Pressure Transmitters*, IE Circular 80-16, U.S. Nuclear Regulatory Commission, June 27, 1980.

60. *Reactor Trip on High Negative Flux Rate, Byron Unit 1*, Licensee Event Report 454/85-68, U.S. Nuclear Regulatory Commission, July 13, 1985.

61. *Instrument Failures on Unit 1 and Reactor Trips on Unit 2 from Lightning Induced Voltage Transients, Braidwood 1 and 2*, Licensee Event Report 456/88-23, U.S. Nuclear Regulatory Commission, Oct. 17, 1988.

62. *Reactor Scram Induced by Lightning Strikes Affecting Neutron Monitoring System, Grand Gulf Unit 1*, Licensee Event Report 416/88-12, U.S. Nuclear Regulatory Commission, Aug. 15, 1988.

63. *Failure of Steam Pressure Transmitters Resulting in a Safety Injection, Salem Unit 1*, Licensee Event Report 272/80-31, U.S. Nuclear Regulatory Commission, June 8, 1980.

64. *Reactor Trip Due to Lightning Strike, Zion Unit 2*, Licensee Event Report 304/86-16, U.S. Nuclear Regulatory Commission, June 27, 1986.

65. *Reactor Water Cleanup System Valve Closure Due to Lightning Causing Electric Bus Trip, Quad Cities Unit 2*, Licensee Event Report 265/87-07, U.S. Nuclear Regulatory Commission, May 20, 1987.

66. S. L. Basin et al., *Characteristics of Instrumentation and Control System Failures in Light Water Reactors*, EPRI NP-443, Electric Power Research Institute, August 1977.

67. *Loss of Fill-Oil in Transmitters Manufactured by Rosemount*, Bulletin 90-01, U.S. Nuclear Regulatory Commission, March 9, 1990.

68. W. Stone, "Digital Protection System Upgrades at TVA's Watts Bar Nuclear Plant and Sequoia Nuclear Plant," *EPRI I&C Workshop*, New Orleans, Electric Power Research Institute, March 1990.

69. *Requirements and Design Specification of a BWR Digital Feedwater Control System*, EPRI NP-55-2, Electric Power Research Institute, November 1987.

70. *Testing and Installation of a BWR Digital Feedwater Control System*, EPRI NP-5524, Electric Power Research Institute, December 1987.

71. J. A. Thie, *Surveillance of Instrumentation Channels at Nuclear Power Plants, Vol. 2: An Approach to Classifying Problems and Solutions*, EPRI NP-6067, Electric Power Research Institute, June 1989.

72. "A Reader's Challenge," *Analog Dialogue* 24(3), 25 (1990).

73. *Nuclear Plant-Aging Research on Reactor Protection Systems*, NUREG/CR-4740, U.S. Nuclear Regulatory Commission, January 1988.

74. J. P. Vora and J. J. Burns, "Understanding and Managing Aging and Maintenance," pp. 28–38 in *Proceedings of the International Nuclear Power Plant Aging Symposium*, Bethesda, Md., Aug. 30–Sept. 1, 1988, NUREG/CP-0100, U.S. Nuclear Regulatory Commission, March 1989.

75. P. Ryan and K. Parker, "Texaco Tracks Instrument Accuracy," *Chem. Process.* 52(7), 95–99 (1989).

76. B. R. Upadhyaya, "Sensor Failure Detection and Estimation," *Nucl. Saf.* 26(1), 32–43 (1985).

77. J. A. Thie, "Surveillance of Instruments by Noise Analysis," *Nucl. Saf.* 22(6), 738–50 (1981).

78. *Analysis of Japanese–U.S. Nuclear Power Plant Maintenance*, NUREG/CR-3883, U.S. Nuclear Regulatory Commission, June 1985.

79. *In-Situ Testing of the Shippingport Atomic Power Station Electric Circuits*, NUREG/CR-3956, U.S. Nuclear Regulatory Commission, April 1987.

Appendix A

DATA SOURCES

Appendix A. DATA SOURCES

An aging analysis of instrumentation and control (I&C) modules includes an evaluation of past operating experiences from various national databases. Because utilities are required to report only a small fraction of the total number of plant component failures (i.e., those that cause a violation of the plant technical specifications), the representativeness of the data must be continually questioned to ensure that the conclusions drawn are valid. Furthermore, events that are reported are often incomplete in the sense that much of the information pertinent to reliability or availability assessments is missing [such information is not required by the U.S. Nuclear Regulatory Commission (NRC)]. Key elements of information may, in fact, be unknown at the time of the initial report.

However, the national databases do have several virtues that make them suitable sources for aging information. First, they contain a large amount of data representing a broad cross section of nuclear power plants. Second, the data are public, although sometimes difficult to obtain. Third, some of the data include sufficient information to identify basic failure characteristics such as the individual component that failed and the reason for its failure.

Although much useful information is available from these databases, limitations and weaknesses must be recognized. In general, these databases do not contain a complete record of all failures, partially because of the nature of the databases and the failures required to be reported. The result is that failure frequencies determined directly from the database information will probably be lower than actual. An additional concern with the database information is inconsistency in the interpretation of codes used to report events associated with the failure.

Some data sources are listed below and described in Sects. A.1 through A.8. Data printouts are compared in Sect. A.9.

Licensee Event Reports
Nuclear Plant Reliability Data System
Nuclear Plant Aging Research Reports
NRC Inspection Reports
NRC Headquarters Daily Reports and NRC Operations Daily Reports
Electric Power Research Institute Reports
Nuclear Plant Experiences
In-Plant Reliability Data System

A.1 LICENSEE EVENT REPORTS

The Code of Federal Regulations (10 CFR, Pt. 50.72, for occurrences before 1984 and 10 CFR, Pt. 50.73, of events after January 1, 1984) require nuclear power plants to report significant events to NRC. The pre-1984 Licensee Event Report (LER) database has been useful as a source of reliability data. However, events reported to the LER system after January 1, 1984, are only those that are or those that led to safety-significant events. No LER was required if a module failed and could be replaced within the time constraint of the limiting condition for operation.

LER guidelines do not require the reporting of certain single failures, and because most aging-related failures are simple failures, this reporting requirement reduces the quantity of actual aging-related module failures reported. Information in the LERs is often insufficient to determine the failure mechanism in the affected module. The manufacturer of the module was identified in <1% of the reports, and the model type for the module in <1%.

A.2 NUCLEAR PLANT RELIABILITY DATA SYSTEM

The Nuclear Plant Reliability Data System (NPRDS) was developed by the Equipment Availability Task Force of the Edison Electric Institute (EEI) in the early 1970s under the direction of the American National Standards Institute. NPRDS was maintained by the Southwest Research Institute under contract to EEI through 1981. Since January 1982, NPRDS has been under the direction of the Institute of Nuclear Power Operation, an industry-sponsored organization, to provide information on the operation of systems and components for the major nuclear plants. The systems include engineering and failure data for these components. The NPRDS database contains detailed information describing failures of a broad range of components. The information on the module failures is submitted voluntarily to NPRDS. NPRDS failure reports are to be submitted when a component failure results in the failure of a reportable system to operate properly. The system's operability must be either lost or sufficiently degraded to inhibit proper function. Typically, instrumentation channels are provided in redundancy such that failures of individual modules do not result in the loss of operability of entire systems. Therefore, single failure of instruments—whether from component mechanical defects or from calibration problems—are not reportable to NPRDS.

Some of the limitations concerning this source are summarized below.

- Not all utilities report to NPRDS.
- Incipient failures are not reportable.
- Complete maintenance histories of failed components are not available, and the effects of test and maintenance activities on aging-related failures are masked. Therefore, time-line histories needed for aging evaluations are not available through NPRDS.
- Accurate module service-age calculations are difficult to obtain from these data.
- Approximately 50% of the NPRDS data are placed in the *unknown* or *other devices* failure category.
- Often, the NPRDS *causes description codes* do not reflect the mechanisms causing the failure.
- Many narrative descriptions do not provide sufficient information.

A.3 NUCLEAR PLANT AGING RESEARCH REPORTS

These are reports generated within the Nuclear Plant Aging Research (NPAR) Program and are concerned with specific items of equipment and/or systems, but they contain some useful data for I&C modules. No system in the plant is devoid of controls, protection, alarms, monitoring, etc.; hence, I&C is omnipresent.

A.4 NRC INSPECTION REPORTS

These reports are those of the resident inspector or the result of an unannounced visit to the plant by the regional NRC inspector and cover any item of concern to NRC. Some of the reports may be a follow-up on specific LERs to examine the licensee's corrective or required actions. Some review aspects of the maintenance program, repair and testing of modules and equipment, surveillance, trending analyses, documentation, spare parts availability, and all aspects of management and radiation control. These reports, in 10% of the documents reviewed, supplemented the information from a few specific LERs and provided some new information of a generic nature. From these, an indication can be surmised as to which plants have a good vs a mediocre maintenance program. This, too, would be indicative of the quality of preventive maintenance programs and testing and surveillance techniques and methods.

A.5 NRC HEADQUARTERS AND NRC OPERATIONS CENTER DAILY REPORTS

Sometimes these reports provide an early notification to NRC that an event has happened that will require information from the licensee to be filed (within 90 days). They also keep NRC posted of early significant happenings, whether reportable in another format or not. Some data items of interest have been extracted from these terse statements.

A.6 ELECTRIC POWER RESEARCH INSTITUTE REPORTS

The Electric Power Research Institute continues to produce generic reports each year. While these reports typically describe research oriented to industry problems, some of the justification for a study is based on plant operating experiences, and some useful data were extracted for inclusion in our data bank.

A.7 NUCLEAR PLANT EXPERIENCE

Nuclear Plant Experience (NPE) is a commercial technical service that compiles significant events occurring at operating reactors using LERs, utility operating reports, and a wide variety of current literature to produce an indexed summary of pertinent occurrences. It is routinely updated by the R. M. Stoler Corp. Some specific events at specific plants are reviewed in more detail than in LERs. Items of generic importance and I&C documents are often included.

A.8 IN-PLANT RELIABILITY DATA SYSTEM

This data source has some generic information and data from the maintenance reports from some nuclear power plants. These data, however, were found to be more useful for reliability studies per se than for study about aging.

A.9 COMPARISON OF DATA PRINTOUTS

A survey was made of three national databases (LER, NPRDS, and NPE) for the quantity of annual data each could produce for this study. Relative comparisons are shown in Figs. A.1 through A.3 for the six I&C modules and display quite similar patterns for four of the modules. In each module category, percentages were calculated separately for each of the three databases (LER, NPRDS, and NPE). The percentages indicate the ratio of aging-related events in the given year to the total for the 5-year period studied. Consequently, for this study, it was perceived that the generic patterns could be obtained by using the smaller number of representative events in the LER database.

Patterns for annunciators and recorders provide an interesting speculation. A major portion of the data for annunciator modules was obtained in 1985, while that for recorders was obtained in 1987. Because the open literature does not discuss any major aging or industry problem for these I&C modules, it can be surmised that the causes for these active years could be from either regulatory requirements from the control room design reviews or from technological updating. Section 3.10 discusses accelerating aging stressors.

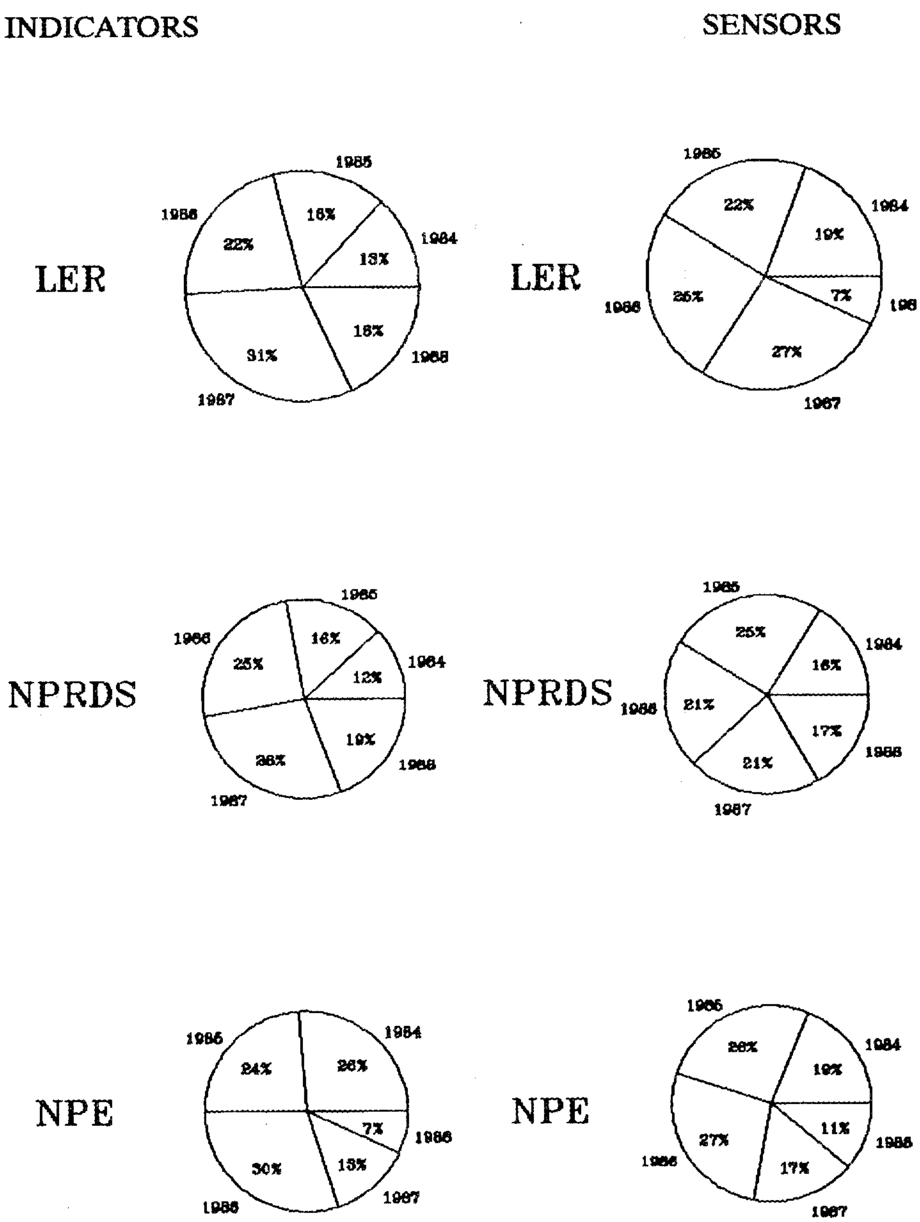


Fig. A.1. Relative percentages of indicator and sensor module age-related data.

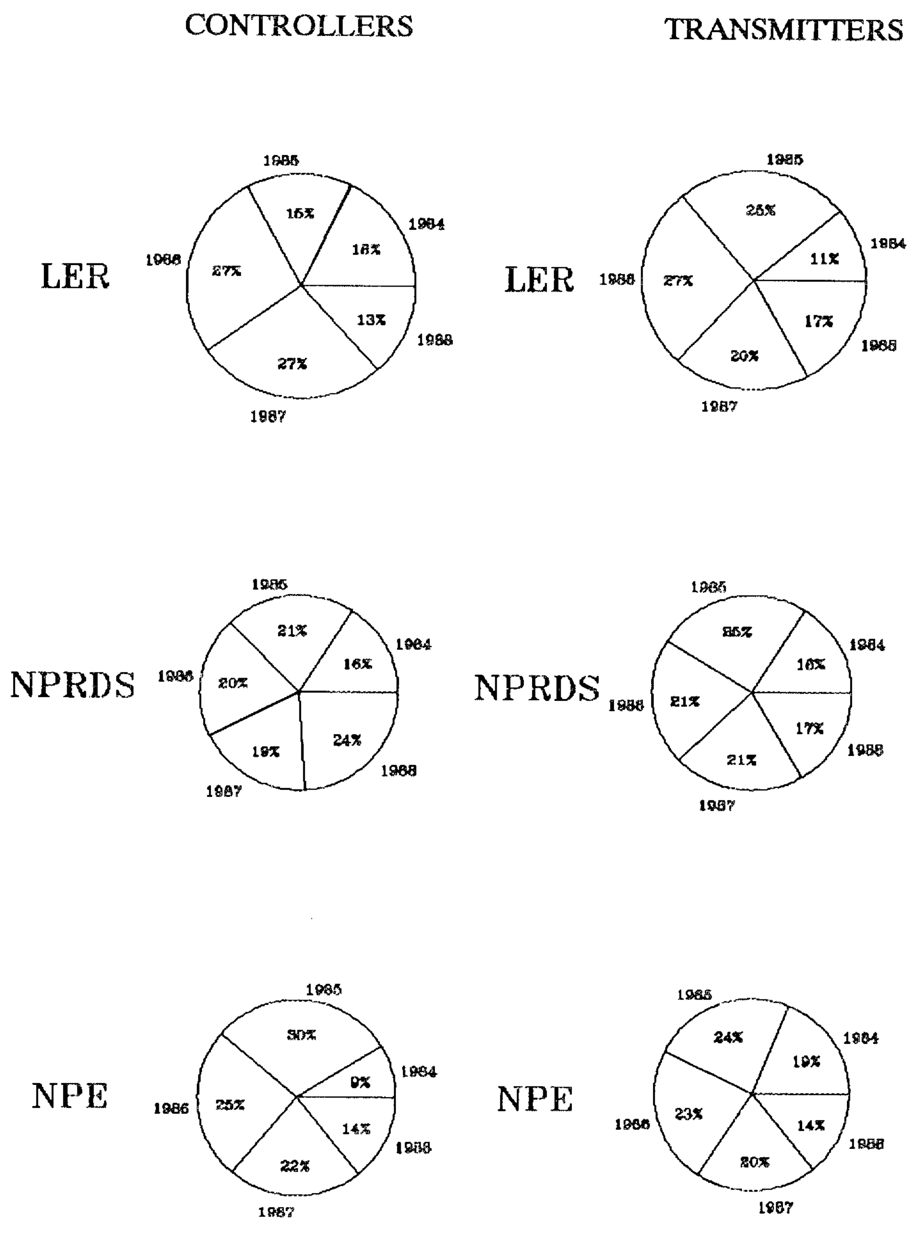


Fig. A.2. Relative percentages of controller and transmitter module age-related data.

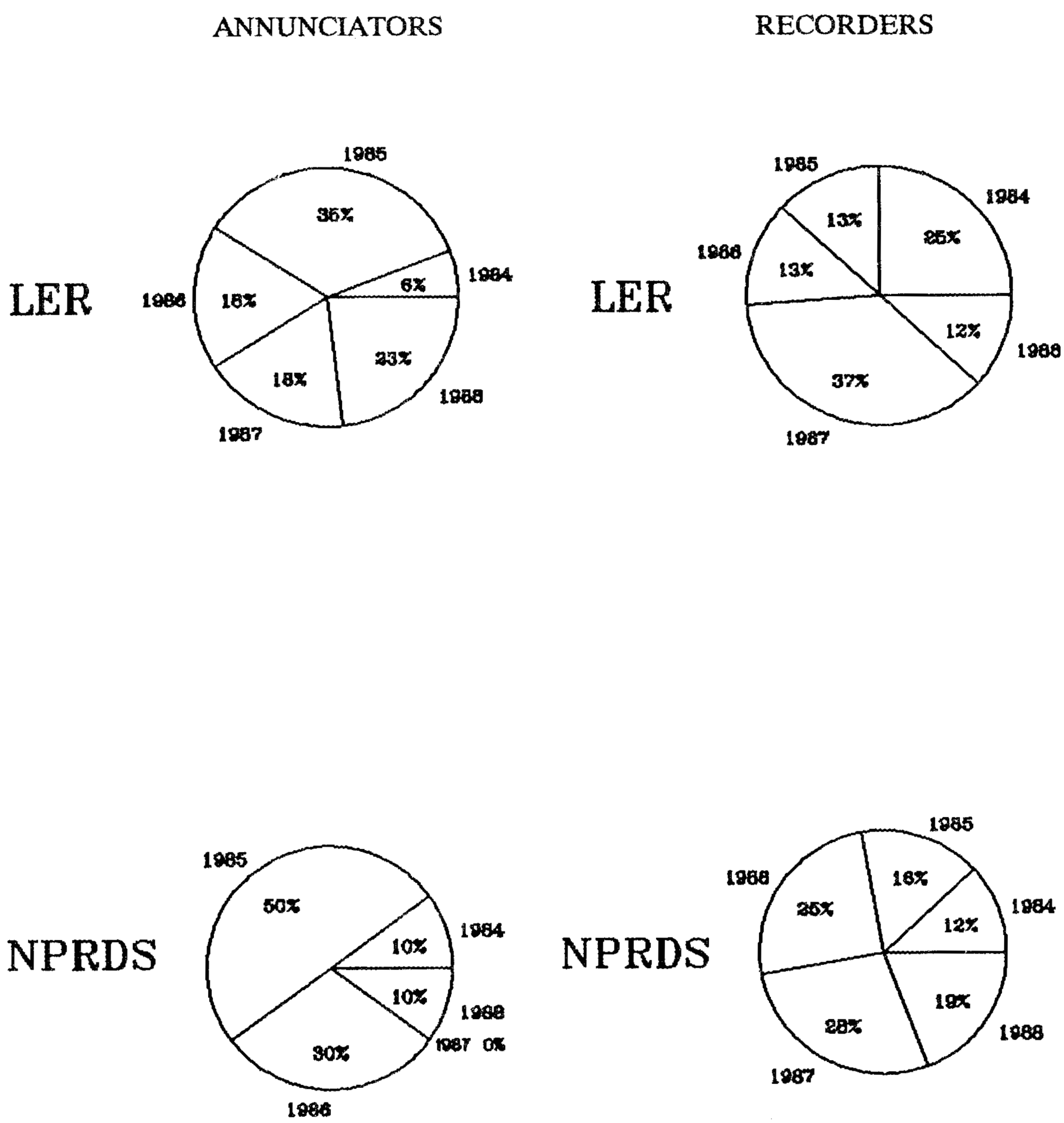


Fig. A.3. Relative percentages of annunciator and recorder module aging-related data.

Appendix B

LIGHTNING EVENTS AS ELECTROMAGNETIC-INTERFERENCE STRESSORS

Table B.1. Lightning events that challenged safety-related systems

	Plant	Document date	LER[a] (other)	Systems affected[b]	Instrumentation and control category affected
1	Braidwood 1	7/18/89	89-06	CRPS	Source range detector
2	Braidwood 2	7/18/89	89-06	CRPS	
3	Braidwood 2	9/07/89	89-04	CRPS	
4	Braidwood 2	9/08/89	(9-8-89)[c]	CRPS	
5	Brunswick 1	9/10/84	84-25		Main steam line radiation high monitor, channel
6	Brunswick 2	9/15/84	84-25	APRM	
7	Byron 1	7/13/85	85-68	CRPS	
8	Byron 1	7/29/87	87-17	CRPS	
9	Byron 1	7/31/87	87-17	CRPS	
10	Catawba 1	7/27/89	(OP C)[d]	RWST	Level channels
11	Farley 2	3/27/84	84-04	CRPS	
12	Farley 2	7/15/85	85-10	CRPS	
13	Grand Gulf 1	8/15/88	88-12	APRM	
14	Palisades	8/5/89	(OP C)[d]	AFW	
15	Quad Cities 1	7/29/87	87-14	CRTGA	Chlorine analyzer
16	Quad Cities 1	3/10/90	(3/10/90)[c]		Turbine generator load mismatch
17	Vogtle 1	7/31/88	88-25	CRPS	
18	Zion 1	8/17/79	(IE)[e]	CRPS	
19	Zion 2	8/17/79	(IE)[e]	CRPS	
20	Zion 2	4/03/80	(IE)[e]		
21	Zion 2	7/16/80	(IE)[e]		

[a]LER = Licensee Event Report.
[b]AFW = auxiliary feedwater.
APRM = average power range monitor.
CRPS = control rod power supply.
CRTGA = control room toxic gas analyzer.
RWST = refueling water storage tank.
[c]NRC Headquarters Daily Report.
[d]OP C = NRC Operations Daily Report.
[e]IE = NRC Information Notice, *Source: Lightning Strikes at Nuclear Power Generating Station*, Information Notice 85-86, U.S. Nuclear Regulatory Commission, Nov. 5, 1985.

Table B.2. Lightning events that stressed nonsafety-related modules

	Plant	Document date	LER[a] (other)	Systems affected[b]	Instrumentation and control modules affected
1	Arkansas 1	4/08/86	86-04	EHC	
2	Arkansas 1	5/17/87	87-02	EHC	
3	Arkansas 2	8/05/85	85-16	RCS	Core protection calculator
4	Braidwood 1	3/15/87	87-18		Containment fuel incident monitor
5	Braidwood 2	11/15/88	88-27		Containment fuel incident radiation monitors
6	Byron 1	9/21/86	86-26		Area radiation monitor and process radiation monitor
7	Byron 1	8/26/88	88-06		Fuel handling building incident area radiation monitor
8	Davis Besse	8/21/87	(12/21/88)[c]		Turbine vibration sensor
9	Duane Arnold	6/17/84	84-20		Control building air intake radiation monitor, channel
10	Monticello	6/05/84	84-12		Reactor building spent fuel pool radiation
11	Monticello	7/18/89	(Op C)[d]		Phone communications
12	Nine Mile Pt. 2	8/15/89	(11/14/89)[c]		Stack radiation monitor
13	Quad Cities 2	7/20/87	87-07	RWCU	Meteorological tower modules
14	River Bend	8/05/87	87-16	AM, SBGT	
15	River Bend	4/14/90	(4/16/90)[e]		Process radiation monitors
16	Robinson 2	8/24/85	85-18	APDMS	Process computer
17	Sequoyah 1	2/21/89	(OP C)[d]		Meteorological instrumentation
18	Sequoyah 2	8/15/88	88-34	CCW	Meteorological monitoring
19	Sequoyah 2	2/21/89	(OP C)[d]		Meteorological instrumentation
20	Shoreham	7/27/86	86-30	RBSVS, CRAC	

Table B.2. (continued)

	Plant	Document date	LER[a]	Systems affected[b]	Instrumentation and control modules affected
21	Shoreham	8/11/86	86-30	RBSVS, CRAC	
22	Summer	7/29/87	87-18	IF&S	
23	Summer	8/30/88	88-10	IF&S	Computer
24	Wolf Creek	8/06/85	85-55		Outside air makeup radiation monitor
25	Wolf Creek	10/09/85	85-71	HVAC	Radiation monitor
26	Yankee Rowe	6/01/86	86-04		Heater drain tank level control channel

[a]LER = Licensee Event Report
[b]AM = annulus mixing.
APDMS = axial power distribution monitoring system.
CCW = component cooling water.
CRAC = control room air conditioning.
EHC = electrohydraulic control.
HVAC = heating, ventilating, and air conditioning.
IF&S = integrated fire and security.
RBSVS = reactor building special ventilation system.
RCS = reactor control system.
RWCU = reactor water cleanup.
SBGT = standby gas treatment.
[c]Nuclear Power Experience Data Base, Stoller Power, Inc.
[d]Op C = NRC Operations Daily Report.
[e]NRC Headquarters Daily Report.

Table B.3. Lightning events that were degrading to safety systems

	Plant	Document date	LER[a] (other)	Systems affected[b]	Instrumentation and control modules affected
1	Braidwood 1	10/17/88	88-23	RPS	
2	Braidwood 2	10/17/88	88-23	RPS, RVLIS	Computer
3	Grand Gulf 1	11/07/89	(Op C)[c]	RPS	Rosemount trip units
4	Oyster Creek	6/11/86	86-17	RPS, VACP	
5	Oyster Creek	7/29/86	86-17	RPS, VACP	
6	Oyster Creek	7/30/86	86-17	RPS, VACP	
7	Salem 1	6/08/80	(IE)[d]	RPS	
8	Turkey Pt. 1	7/21/85	85-19	RPS	Pressurizer pressure protection comparators
9	Turkey Pt. 1	8/13/86	86-32	RPS	Pressurizer pressure protection comparator
10	Washington NP2	5/12/88	88-13		
11	Zion 2	12/02/82	(IE)[d]	RPS	
12	Zion 2	6/27/86	86-16	RPS	Hot leg temperature resistance temperature devices

[a]LER = Lisencee Event Report
[b]RPS = reactor protection system.
RVLIS = reactor vessel level indicating system.
VACP = vital ac power.
[c]Op C = NRC Operations Daily Reports.
[d]IE = NRC Information Notice, *Source: Lightning Strikes at Nuclear Power Generating Station*, Information Notice 85-86, U.S. Nuclear Regulatory Commission, Nov. 5, 1985.

Descriptions of several of the more significant events were presented in *Lightning Strikes at Nuclear Power Generating Stations*, Information Notice 85-86, U.S. Nuclear Regulatory Commission, Nov. 5, 1985. Events involving lightning strikes of switchyards and the consequential anticipated impact on the electrical power distribution systems, as opposed to instrumentation and control (I&C) systems, were not covered by this notice.

The above examples definitely show lightning to be a stressor and as such to have an adverse effect on the environmental qualification and useful service life of I&C modules.

Appendix C

MODULE AGING PROFILES

Appendix C. MODULE AGING PROFILES

An example is given here of how the Licensee Event Report (LER) data were used to examine aging-related failures for six instrumentation and control (I&C) modules. The indicator module category for 1984 was selected for this example.

A structured search of the Sequence Coding and Search System (SCSS) database using the keyword INDICATOR retrieved 997 abstracts of LERs written between 1984 and 1988. The number of events for consideration was reduced to 220 by reviewing each abstract to determine which were aging-related. These were then added to a database constructed as a tool for this study. A sample printout from this database for the indicator module category is included as Appendix D. Sorting of the events by year of occurrence produced the distribution shown in Fig. C.1.

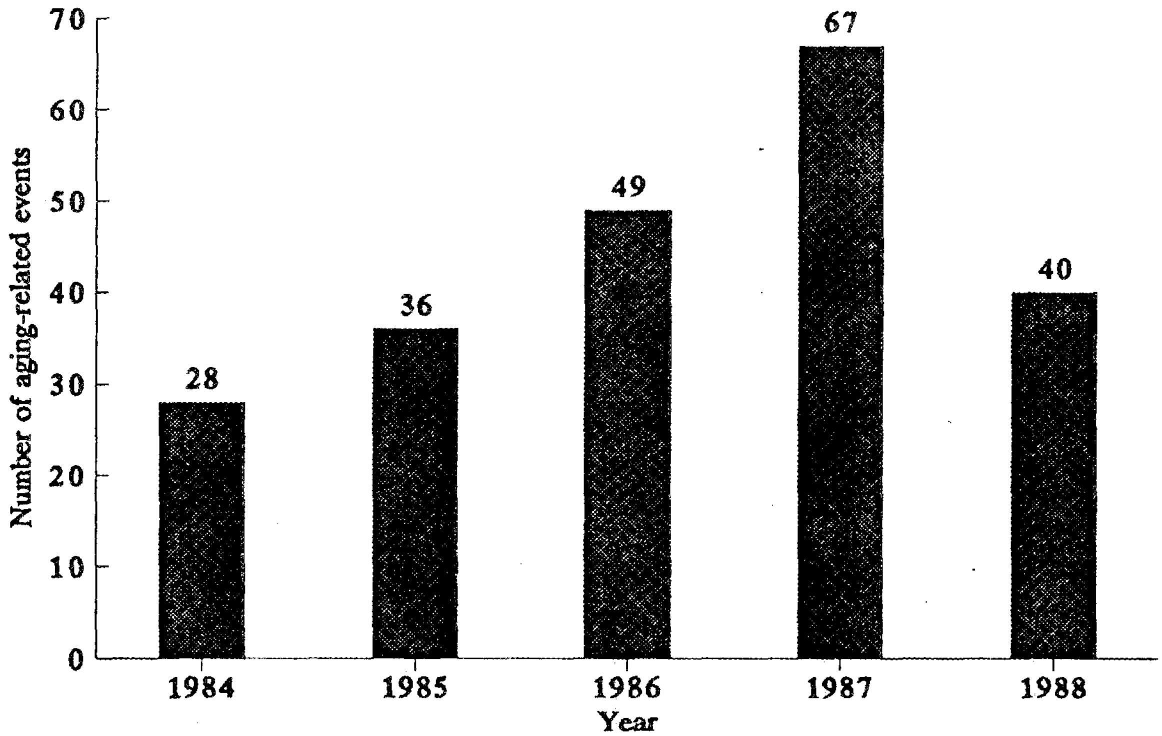


Fig. C.1. Indicator module five-year distribution of events (1984–1988).

The occurrence data for each of the years were next arranged chronologically by the operational age of the plant where the occurrence took place. The date of initial commercial operation was chosen as "age zero" for the I&C module being studied, and the results were tabulated as shown below.

Date year: 1984

Plant age	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20
Number of LERs	17	0	2	1	2	1	0	1	0	0	1	2	1	0	0	0	0	0	0	0

These data were then normalized by dividing the number of events in each plant age category by the number of operating plants having that age in 1984 (see Table C.1). The resultant calculated indicator module aging-related failure rates are shown in Fig. C.2 for 1984.

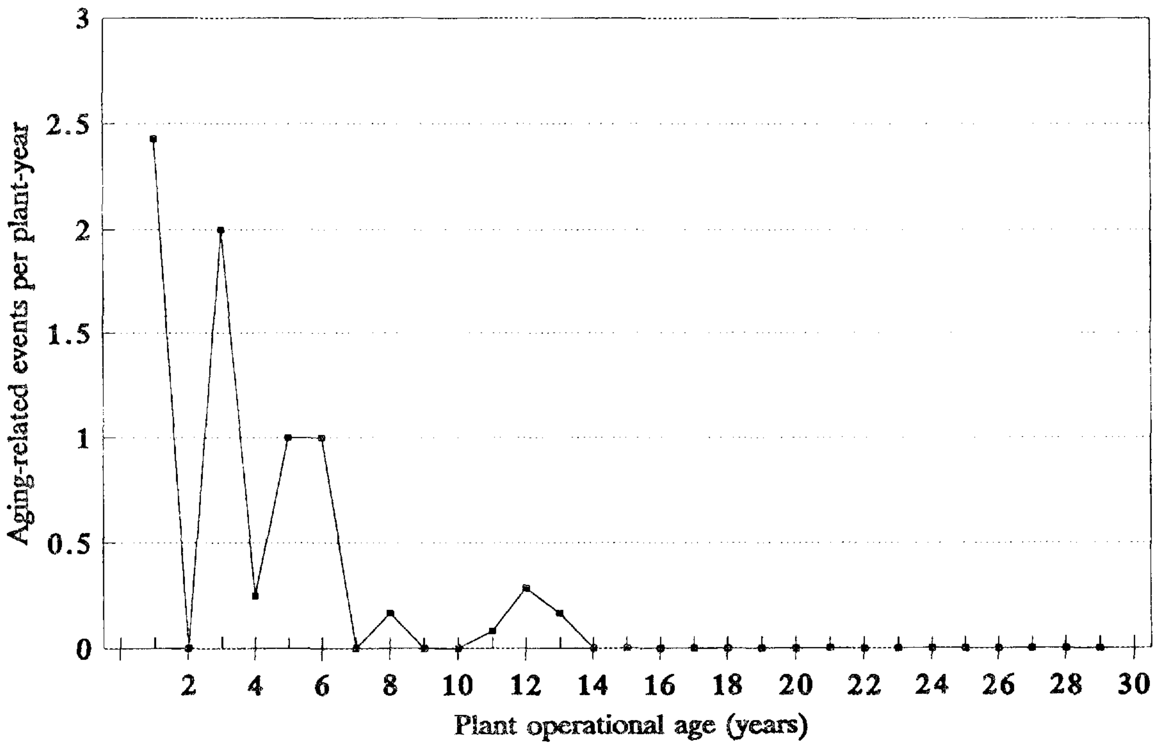


Fig. C.2. Indicator module aging failures by plant age (1984).

Similar failure rates and plots were generated for each year, 1984 through 1988. Failure rates for the entire period were calculated as follows. The number of aging-related events were totaled for each plant age category for the five-year period. For example, the indicator module category indicated a total of 64 aging-related events in one-year old plants and 20 such events for two-year old plants for this five-year period (see Table C.1). This total number of aging-related events was then divided by the total number of plants of that age (plant-years) over the five-year period. Table C.1 shows

34 plant-years for one-year old plants and 31 plant-years for two-year old plants over the period 1984–1988. Therefore, the failure rate is 64/34 aging-related events per plant-year for plant age one and 20/31 aging-related events per plant-year for plant age two over the five-year period for the indicator module category. This failure rate is calculated for each plant age 1 through 29 and then plotted in Fig. C.4. Figure C.3 shows the correlation between the total number of aging-related events and the total number of plant-years over the five-year period.

Similar data for each of the six I&C modules were generated and are documented in Tables C.2 through C.6 and Figs. C.3 through C.26.

Table C.1. Indicator module data summary for 1984–1988
(number of LERs[a] per number of plants vs plant commercial age)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
1	17/7	12/8	12/7	17/6	6/6	64	34	1.882
2	/3	10/7	/8	8/7	2/6	20	31	0.645
3	2/1	/3	3/7	2/8	7/7	14	26	0.538
4	1/4	/1	1/3	5/7	1/8	8	23	0.348
5	2/2	2/4	/1	/3	9/7	13	17	0.765
6	1/1	1/2	1/4	/1	/3	3	11	0.273
7	/4	2/1	/2	1/4	/1	3	12	0.25
8	1/6	/4	3/1	1/2	/4	5	17	0.294
9	/4	/6	/4	5/1	1/2	6	17	0.353
10	/9	/4	7/6	1/4	3/1	11	24	0.458
11	1/12	2/9	1/4	3/6	/4	7	35	0.200
12	2/7	2/12	5/9	7/4	/6	16	38	0.421
13	1/6	3/7	4/12	2/9	1/4	11	38	0.298
14	/5	/6	3/7	3/12	1/9	7	39	0.179
15	/3	/5	4/6	8/7	2/12	14	33	0.609
16	/2	/3	2/5	2/6	3/7	7	23	0.304
17	/1	/2	/3	1/5	1/6	2	17	0.118
18	/1	2/1	/2	/3	2/5	4	12	0.333
19	/0	/1	3/1	/2	1/3	4	7	0.571
20	/0	/0	/1	1/1	/2	1	4	0.25
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.1. (continued)

Plant commerical age (years)	Number of LERs/number of plants 1984	1985	1986	1987	1988	Total LERs	Total plant-years	Events per plant-year
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number of LERs	28/	36/	49/	67/	40/	220		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Report.

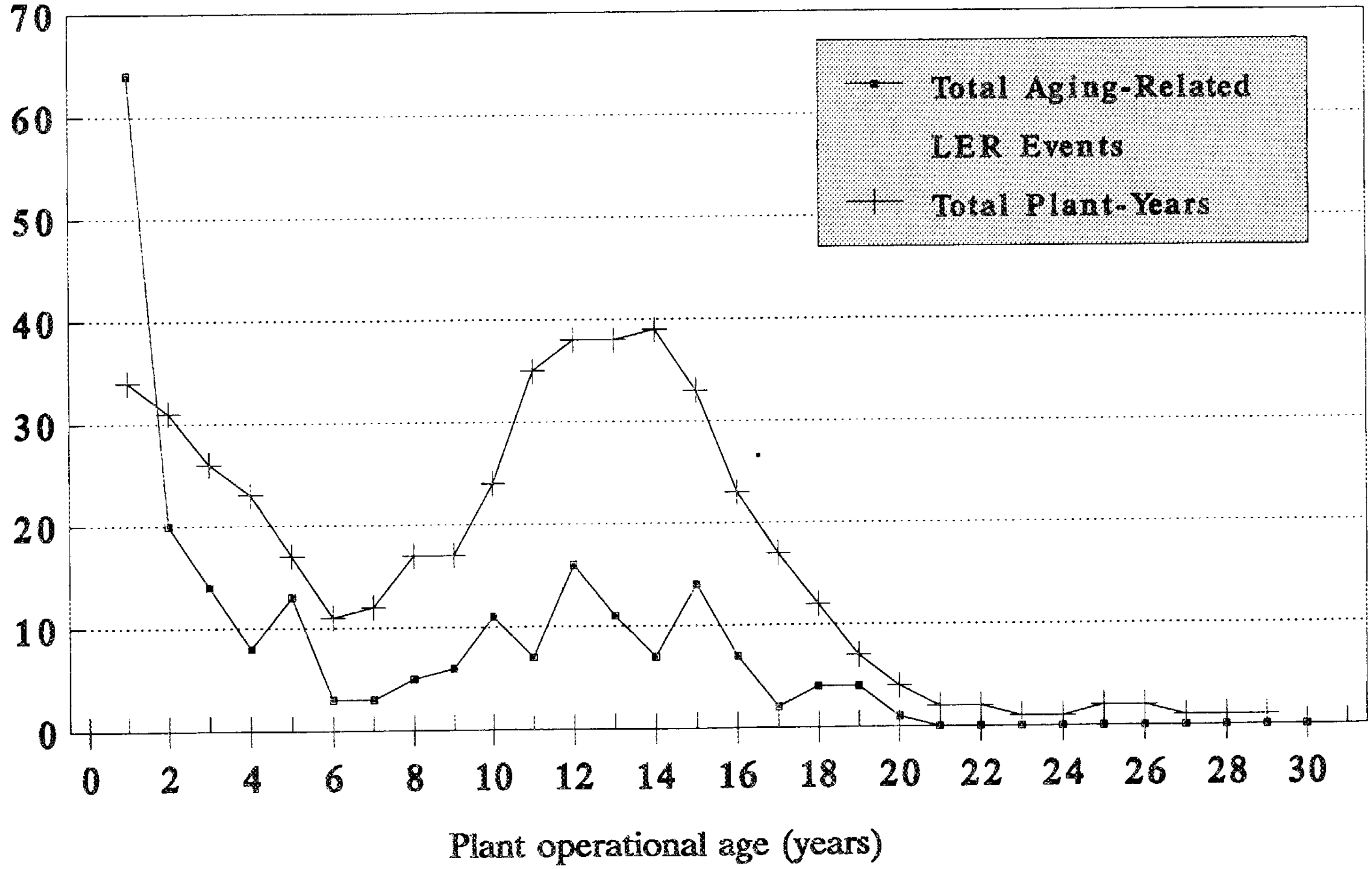


Fig. C.3. Indicator module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

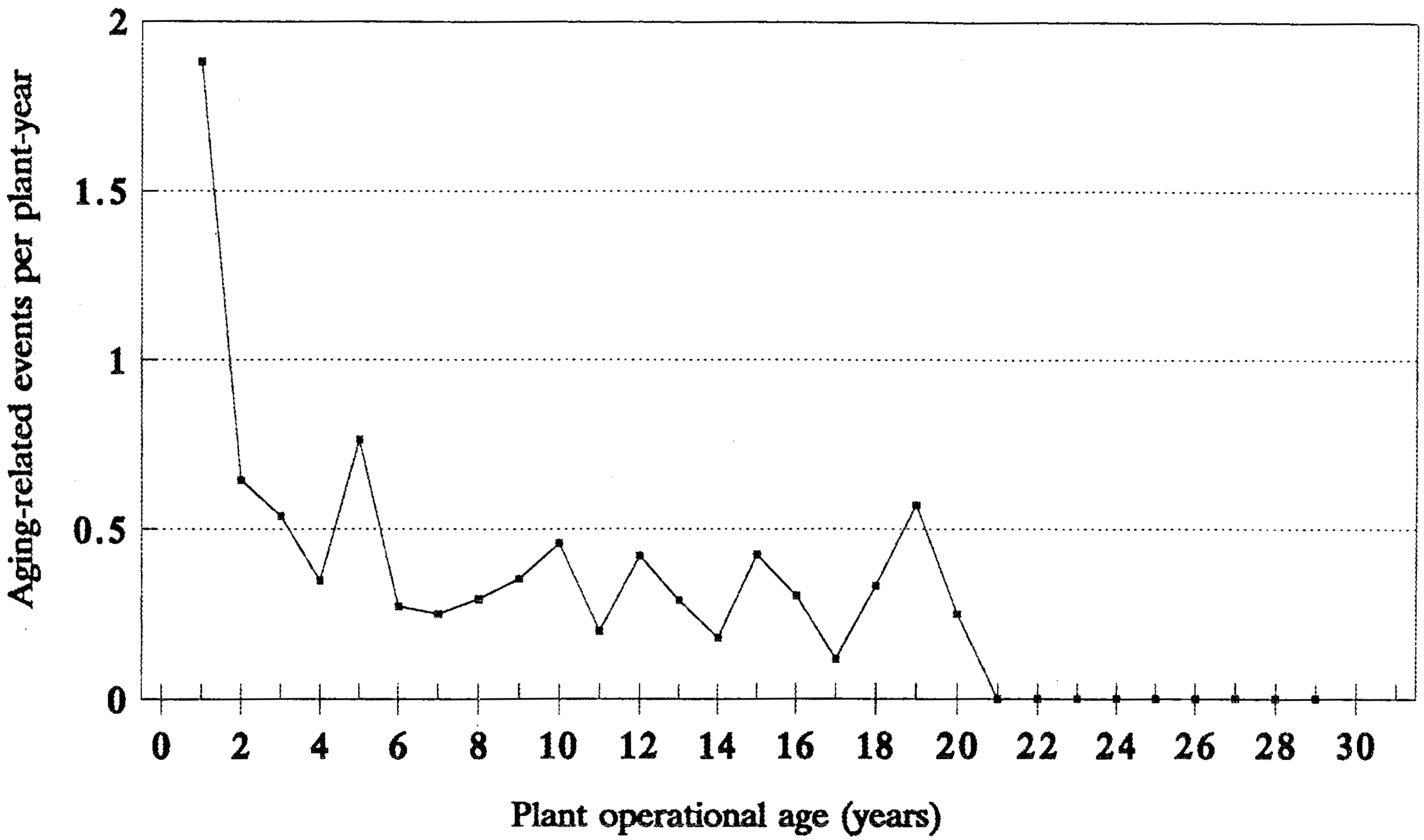
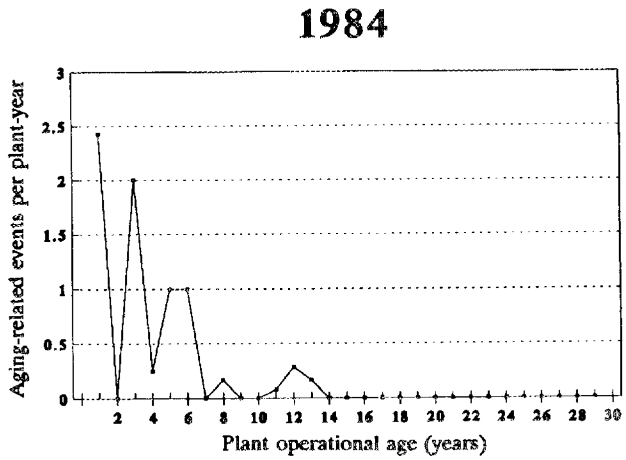
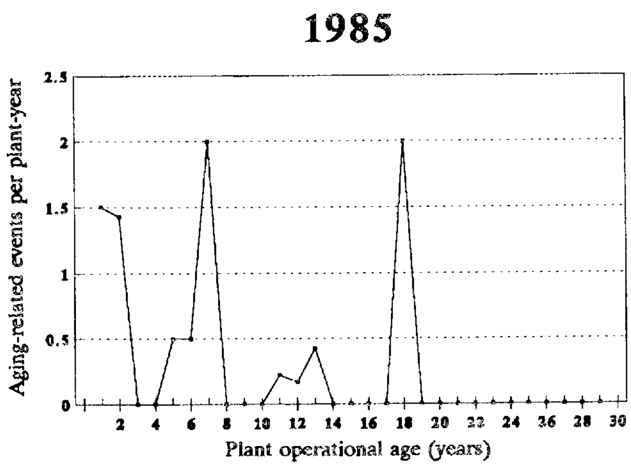


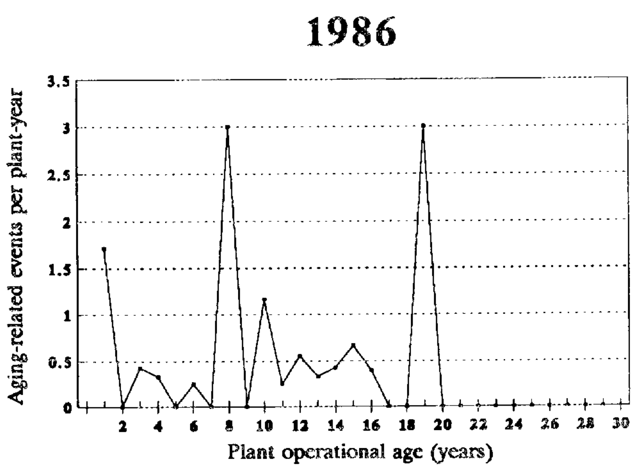
Fig. C.4. Indicator module aging failures by plant age.



(*a*)



(*b*)



(*c*)

Fig. C.5. Indicator module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.

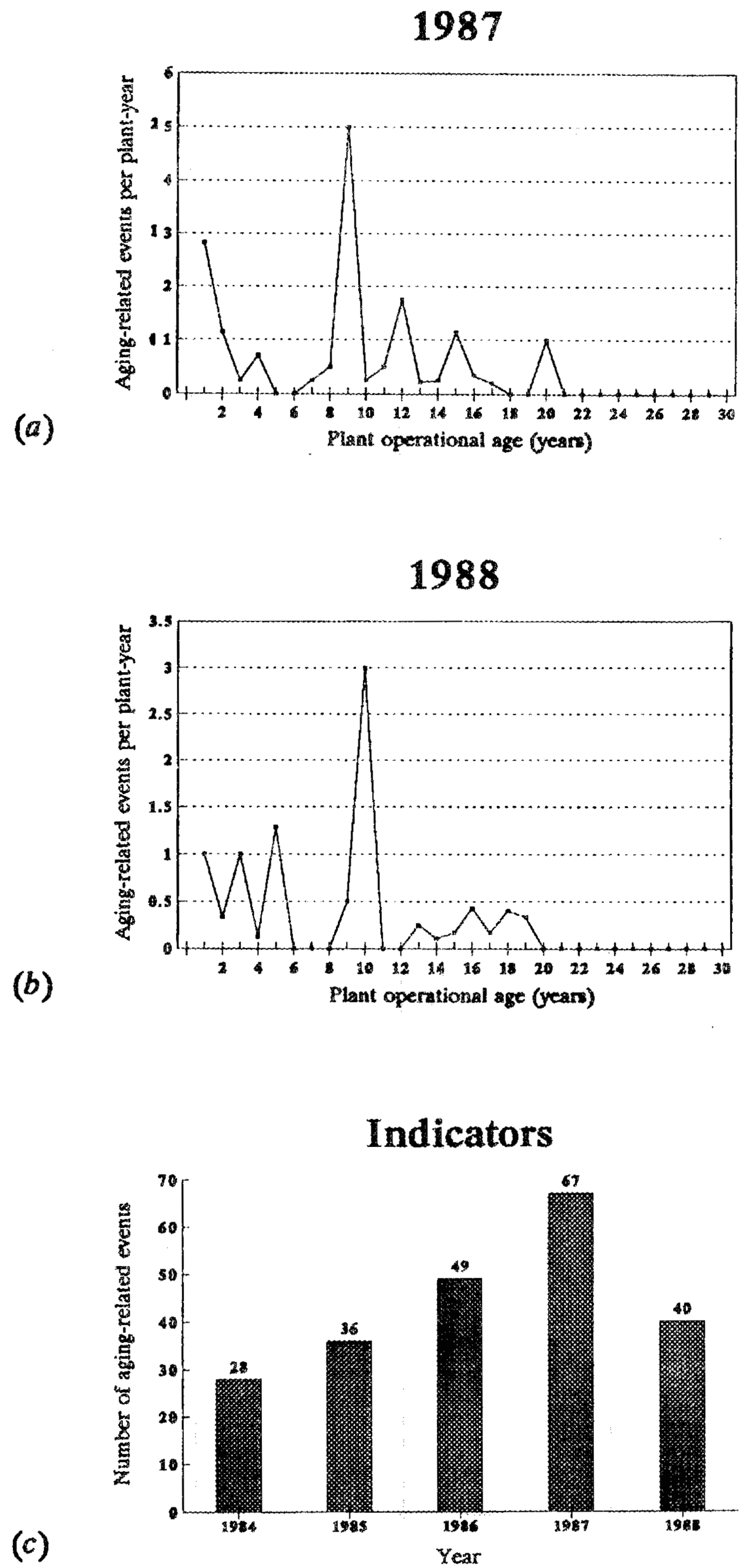


Fig. C.6. Indicator module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Table C.2. Sensor module data summary for 1984–1988
(number of LERs[a] per number of plants vs plant commercial age)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
1	20/	17/8	18/7	8/6	2/6	65	34	1.91
2	/3	8/7	3/8	14/7	/6	25	31	0.81
3	1/1	/3	7/7	2/8	6/7	16	26	0.62
4	1/4	/1	/3	10/7	/8	11	23	0.48
5	1/2	/4	/1	/3	/7	1	17	0.06
6	3/1	2/2	2/4	/1	/3	7	11	0.64
7	1/4	5/1	/2	1/4	/1	7	12	0.58
8	2/6	/4	2/1	1/2	/4	5	17	0.29
9	/4	3/6	1/4	2/1	/2	6	17	0.35
10	3/9	1/4	5/6	1/4	2/1	12	24	0.50
11	1/12	/9	2/4	1/6	/4	4	35	0.11
12	1/7	2/12	2/9	5/4	/6	10	38	0.26
13	/6	5/7	2/12	2/9	1/4	10	38	0.26
14	3/5	/6	1/7	1/12	/9	5	39	0.13
15	/3	1/5	1/6	/7	1/12	3	33	0.09
16	/2	1/3	2/5	1/6	1/7	5	23	0.22
17	/1	/2	/3	1/5	1/6	2	17	0.12
18	/1	/1	/2	2/3	/5	2	12	0.17
19	/0	/1	2/1	/2	/3	2	7	0.29
20	/0	/0	/1	1/1	/2	1	4	0.25
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.2. (continued)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number of LERs	37/	45/	50/	53/	14/	199		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Report.

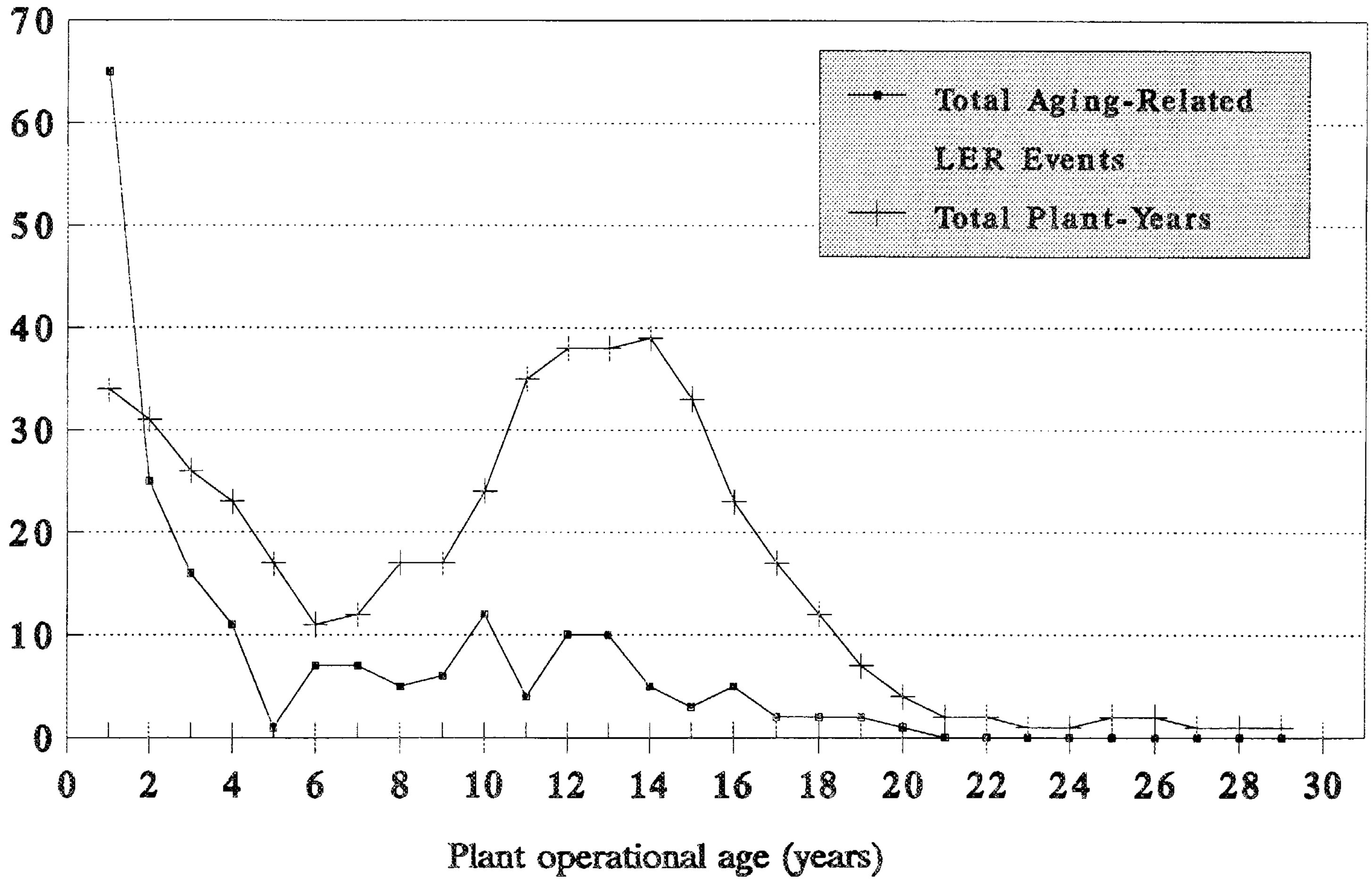


Fig. C.7. Sensor module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

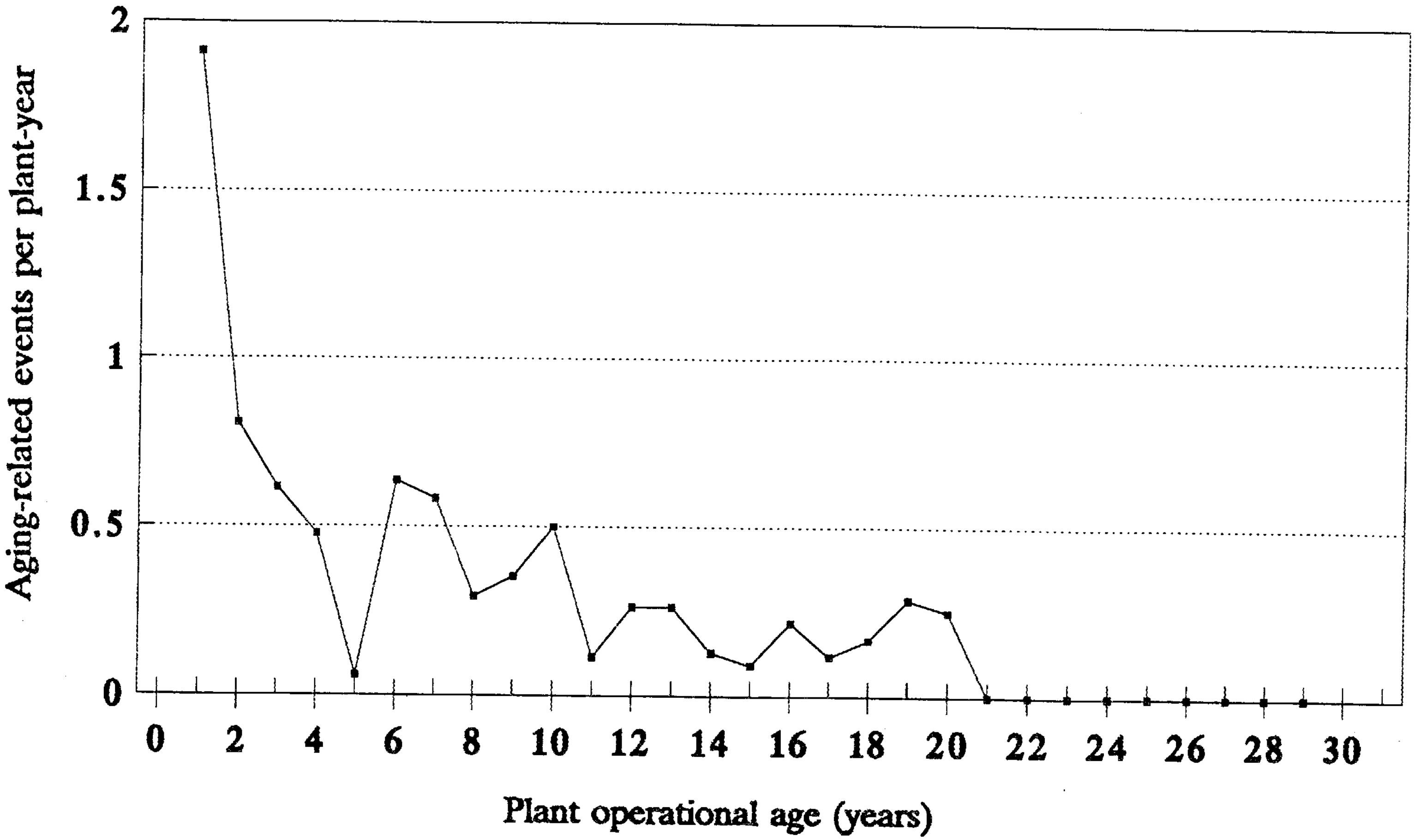


Fig. C.8. Sensor module aging failures by plant age.

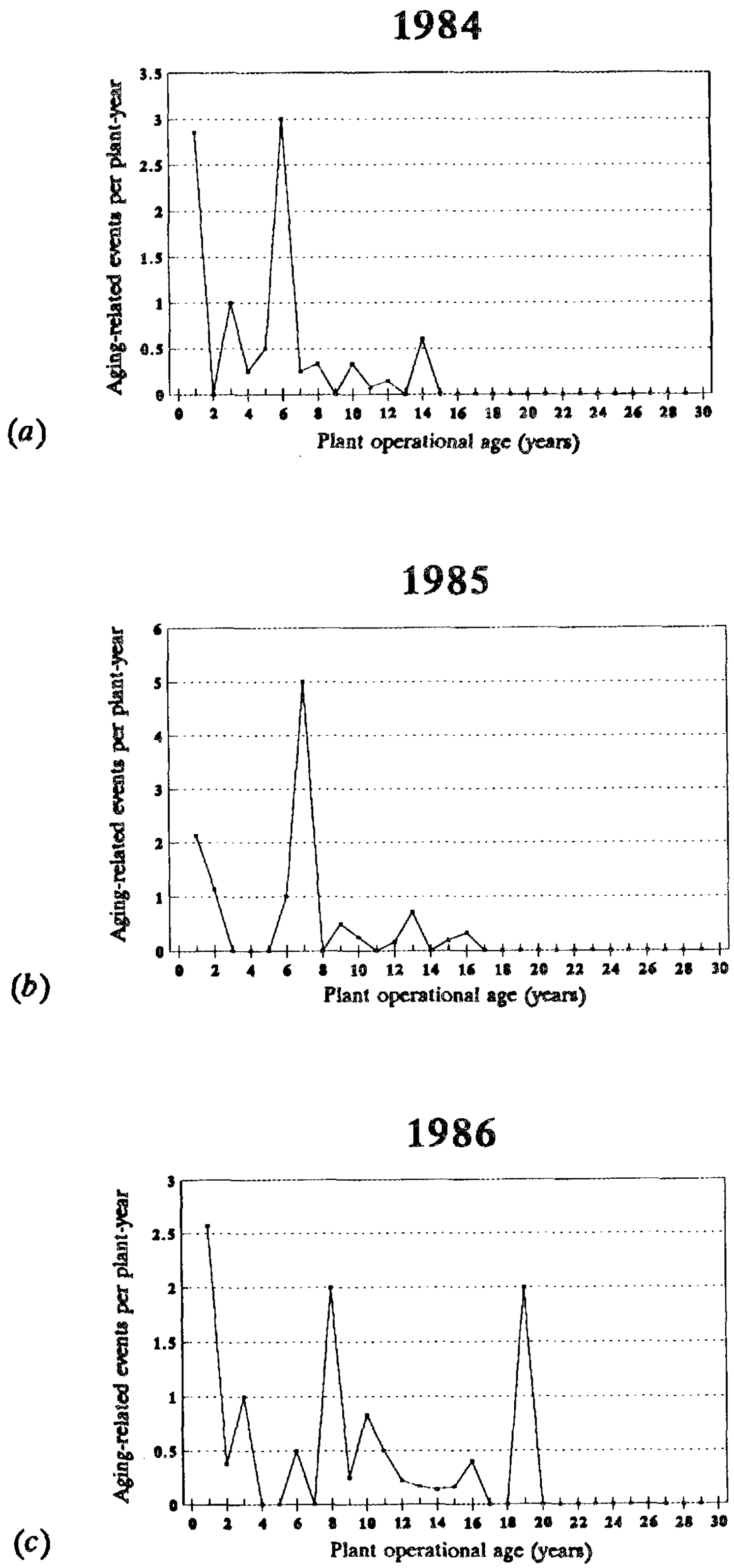


Fig. C.9. Indicator module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.

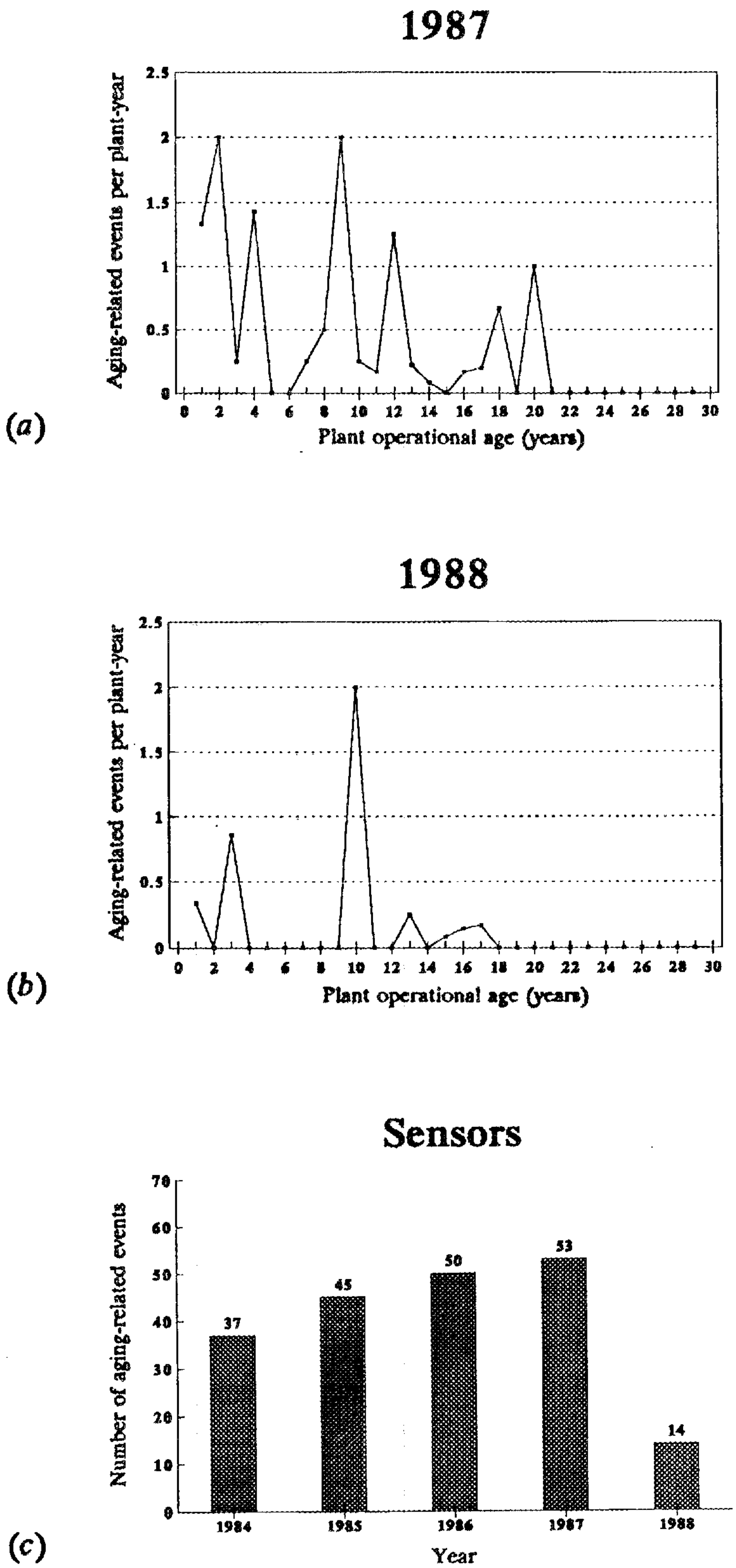


Fig. C.10. Sensor module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Table C.3. Controller module data summary for 1984–1988
(number of LERs[a] per number of plants vs plant commercial age)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-years
	1984	1985	1986	1987	1988			
1	6/7	6/8	3/7	3/6	2/6	20	34	0.59
2	/3	1/7	4/8	2/7	/6	7	31	0.23
3	1/1	/3	1/7	/8	1/7	3	26	0.12
4	/4	/1	/3	/7	/8	0	23	0
5	/2	/4	1/1	/3	2/7	3	17	0.18
6	1/1	/2	/4	1/1	/3	2	11	0.18
7	/4	1/1	1/2	/4	1/1	3	12	0.25
8	/6	/4	1/1	1/2	/4	2	17	0.12
9	2/4	2/6	1/4	4/1	1/2	10	17	0.59
10	6/9	/4	4/6	/4	/1	10	24	0.42
11	1/12	/9	1/4	3/6	/4	5	35	0.14
12	1/7	/12	5/9	5/4	2/6	13	38	0.34
13	/6	3/7	1/12	2/9	/4	6	38	0.16
14	/5	/6	1/7	/12	/9	1	39	0.03
15	1/3	2/5	3/6	3/7	/12	9	33	0.27
16	/2	2/3	1/5	1/6	2/7	6	23	0.26
17	1/1	/2	1/3	1/5	/6	3	17	0.18
18	/1	/1	/2	2/3	/5	2	12	0.17
19	/0	/1	/1	/2	/3	0	7	0
20	/0	/0	/1	/1	/2	0	4	0
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.3. (continued)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number of LERs	20/	17/	29/	28/	11/	105		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Report.

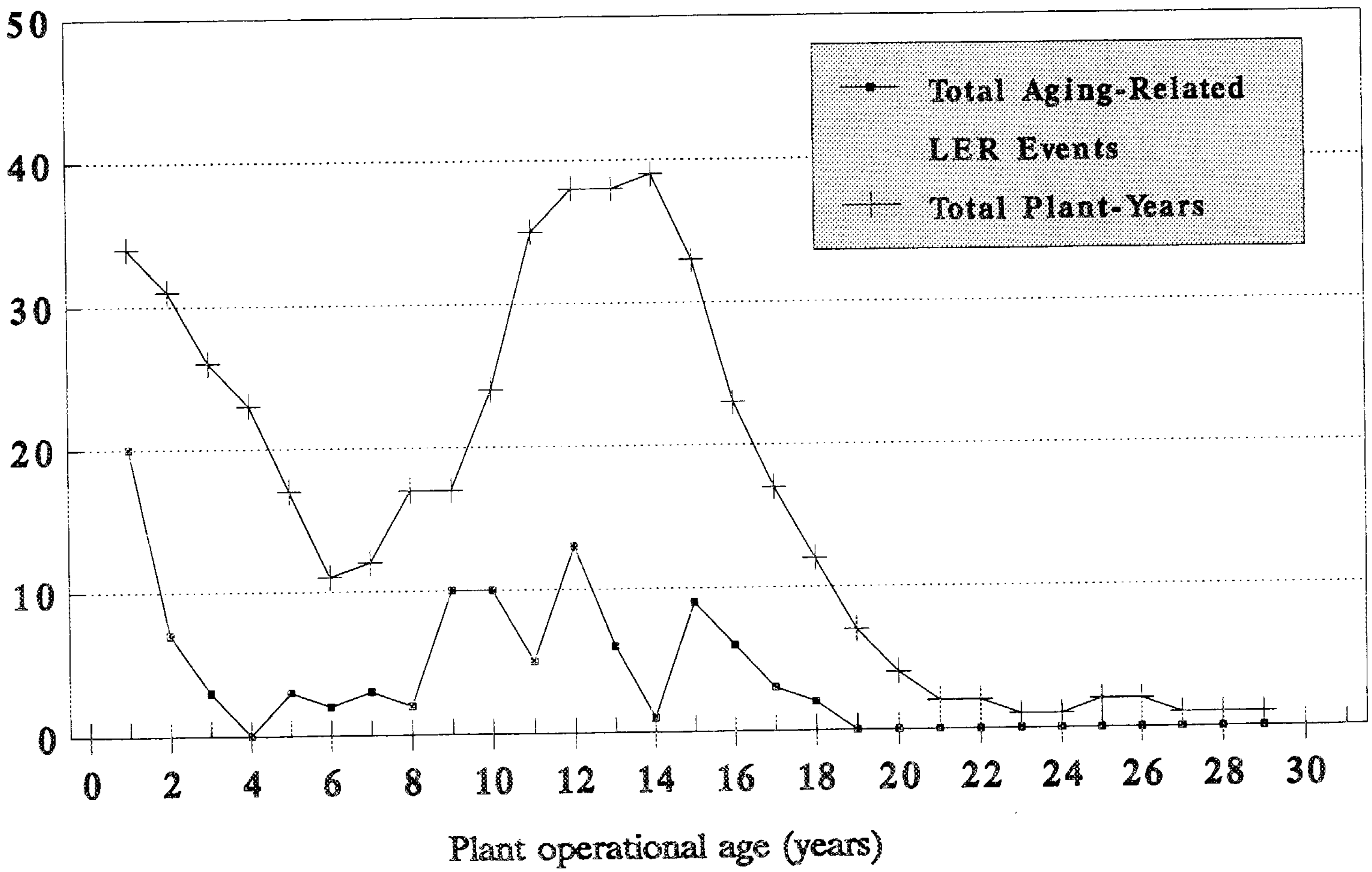


Fig. C.11. Controller module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

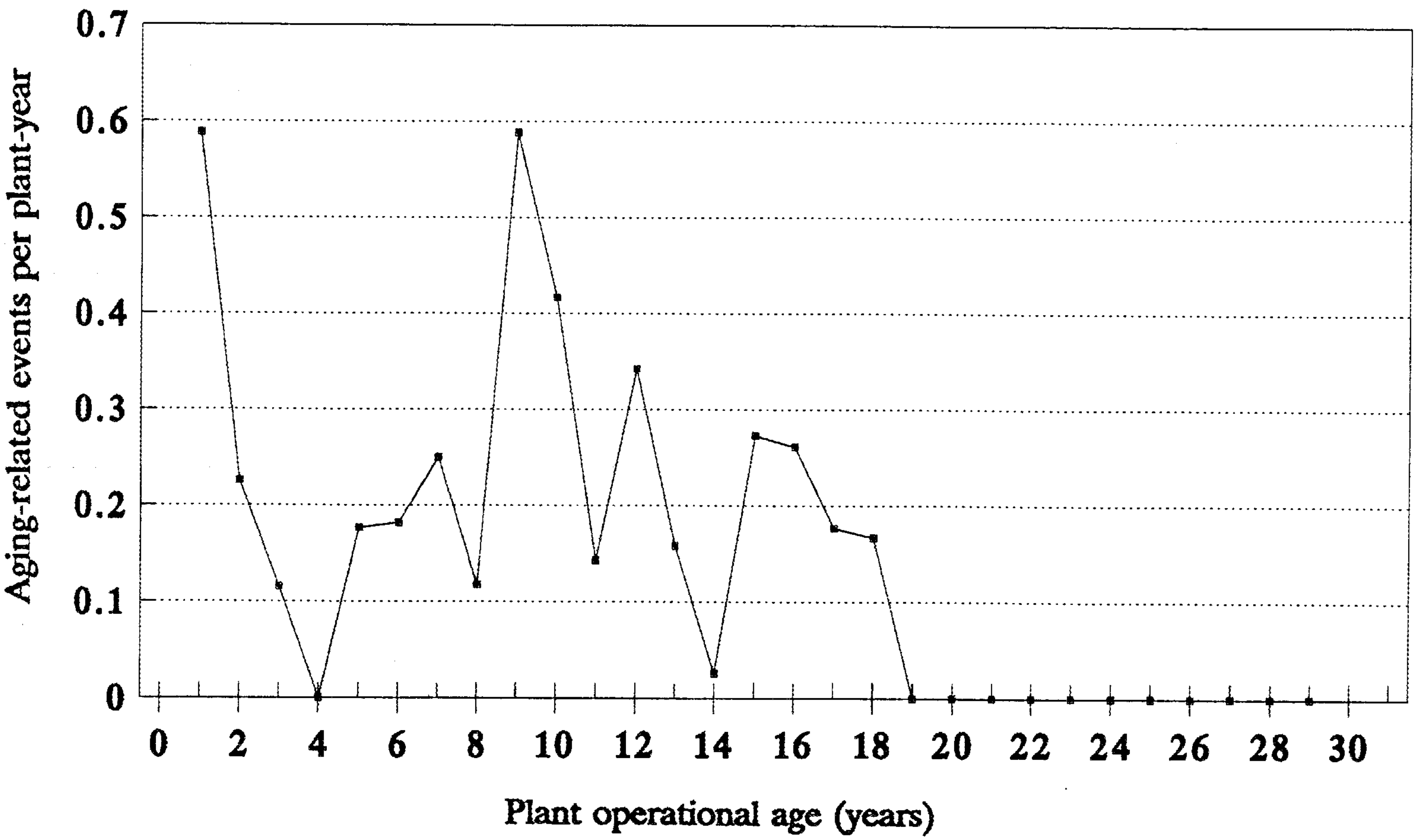
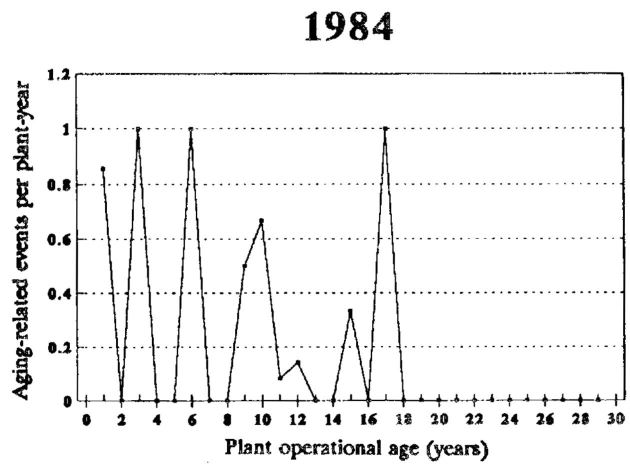
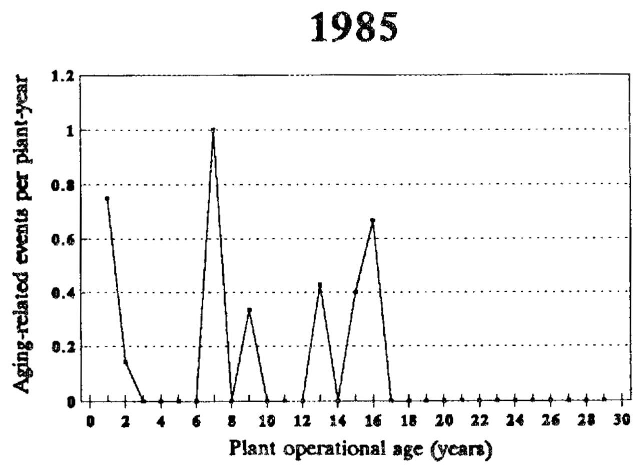


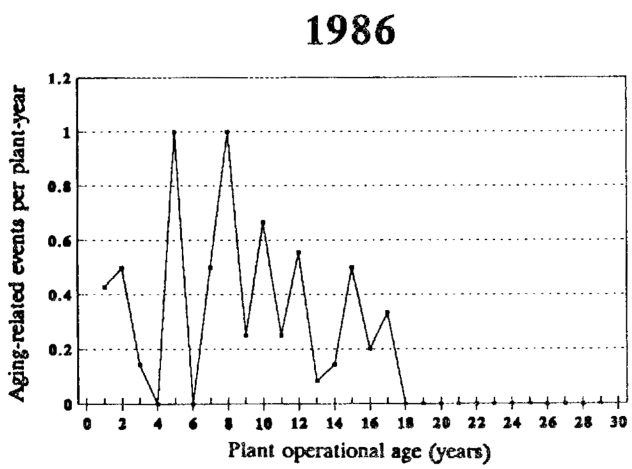
Fig. C.12. Controller module aging failures by plant age.



(*a*)



(*b*)



(*c*)

Fig. C.13. Controller module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.

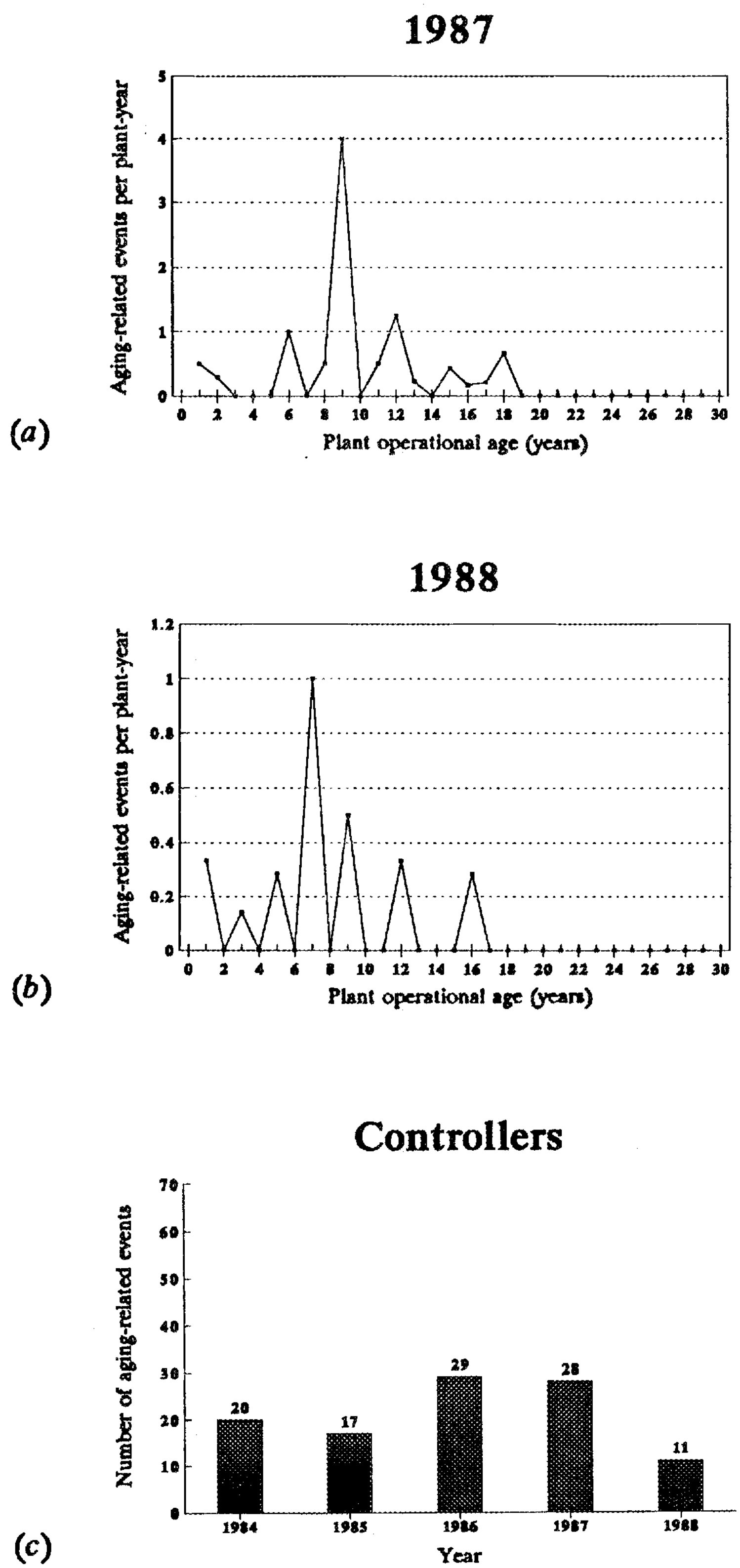


Fig. C.14. Controller module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Table C.4. Transmitter data summary for 1984–1989
(number of LERs[a] per number of plants vs plant commercial age)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
1	2/7	5/8	6/7	4/6	6/6	23	34	0.68
2	/3	/7	1/8	2/7	2/6	5	31	0.16
3	/1	/3	1/7	/8	/7	1	26	0.04
4	/4	1/1	/3	/7	/8	1	23	0.04
5	/2	/4	/1	/3	/7	0	17	0
6	1/1	/2	/4	1/1	/3	2	11	0.18
7	1/4	2/1	/2	/4	/1	3	12	0.25
8	/6	6/4	3/1	/2	/4	9	17	0.53
9	1/4	1/6	/4	/1	/2	2	17	0.12
10	1/9	2/4	1/6	2/4	/1	6	24	0.25
11	1/12	2/9	3/4	/6	2/4	8	35	0.23
12	1/7	1/12	5/9	1/4	/6	8	38	0.21
13	/6	/7	/12	1/9	2/4	3	38	0.08
14	/5	1/6	/7	4/12	/9	5	39	0.13
15	/3	/5	/6	/7	/12	0	33	0
16	/2	/3	1/5	/6	1/7	2	23	0.09
17	/1	/2	/3	/5	/6	0	17	0
18	/1	/1	/2	/3	1/5	1	12	0.08
19	/0	/1	/1	/2	/3	0	7	0
20	/0	/0	/1	/1	/2	0	4	0
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.4. (continued)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-year
	1984	1985	1986	1987	1988			
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number of LERs	8/	21/	21/	15/	14/	79		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Report.

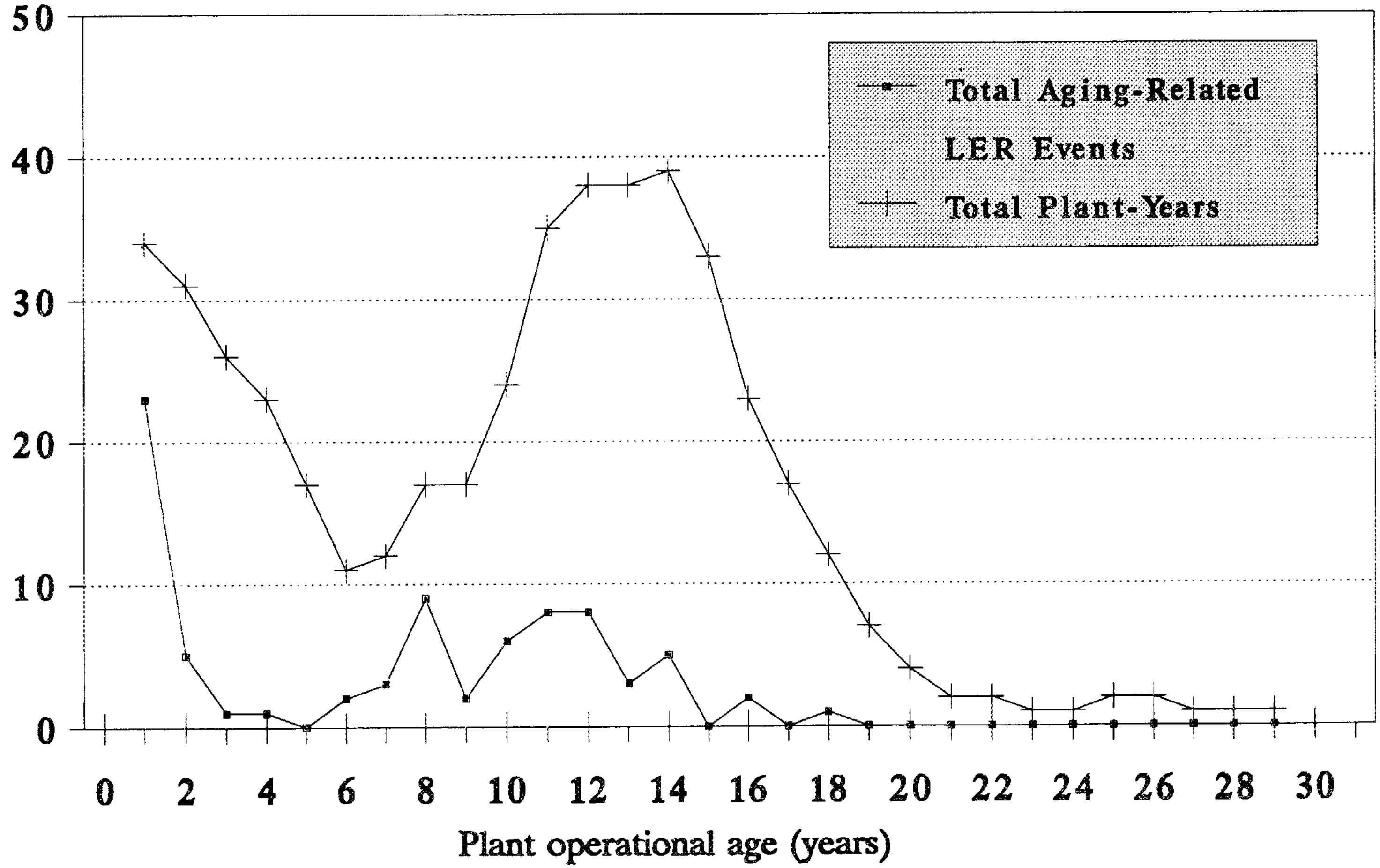


Fig. C.15. Transmitter module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

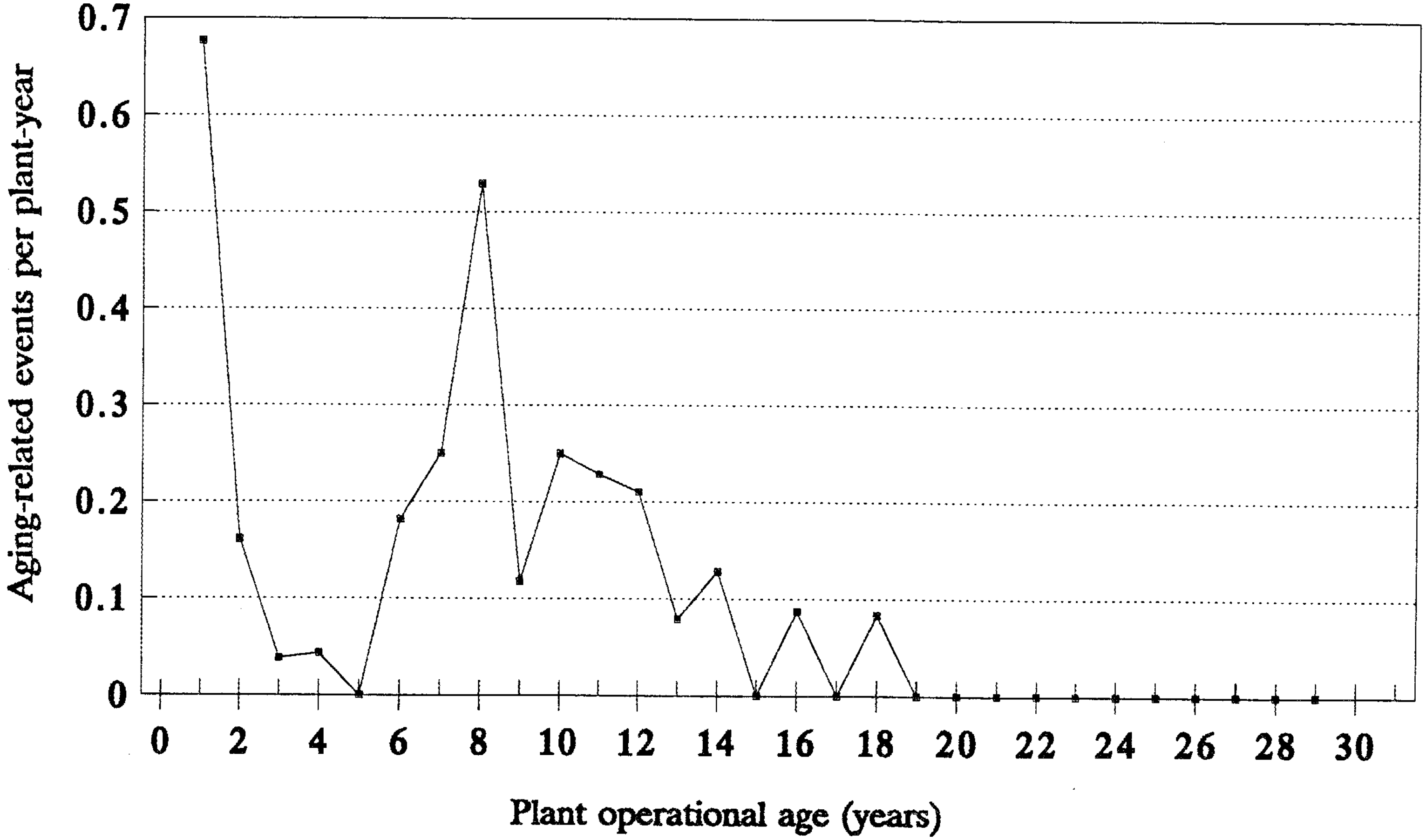
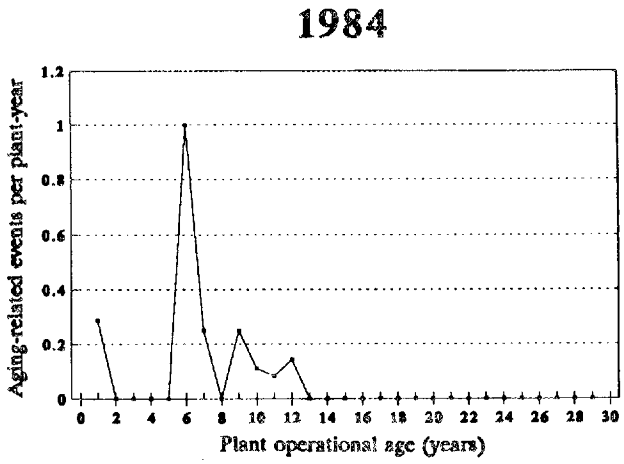
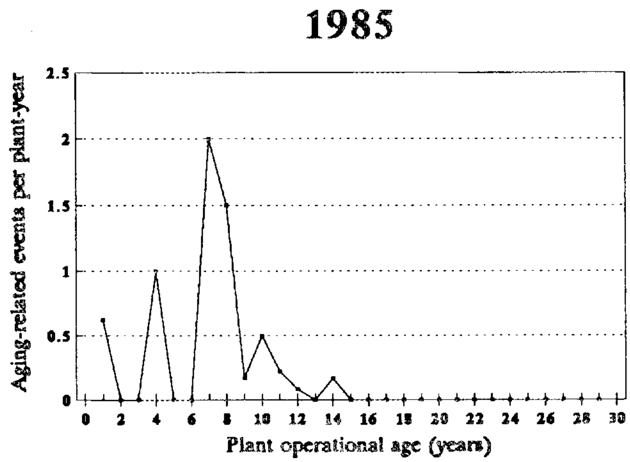


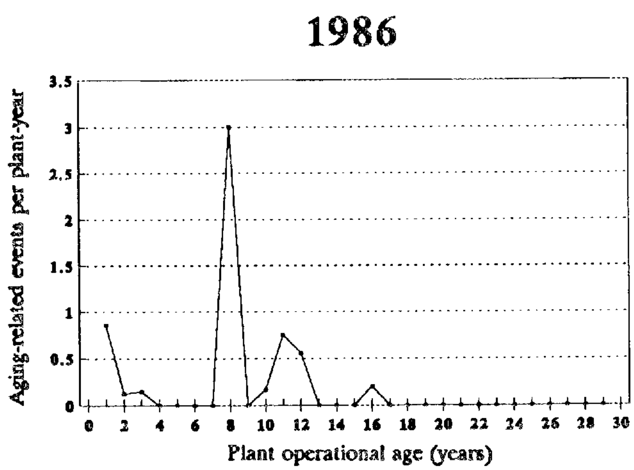
Fig. C.16. Transmitter module aging failures by plant age.



(*a*)



(*b*)



(*c*)

Fig. C.17. Transmitter module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.

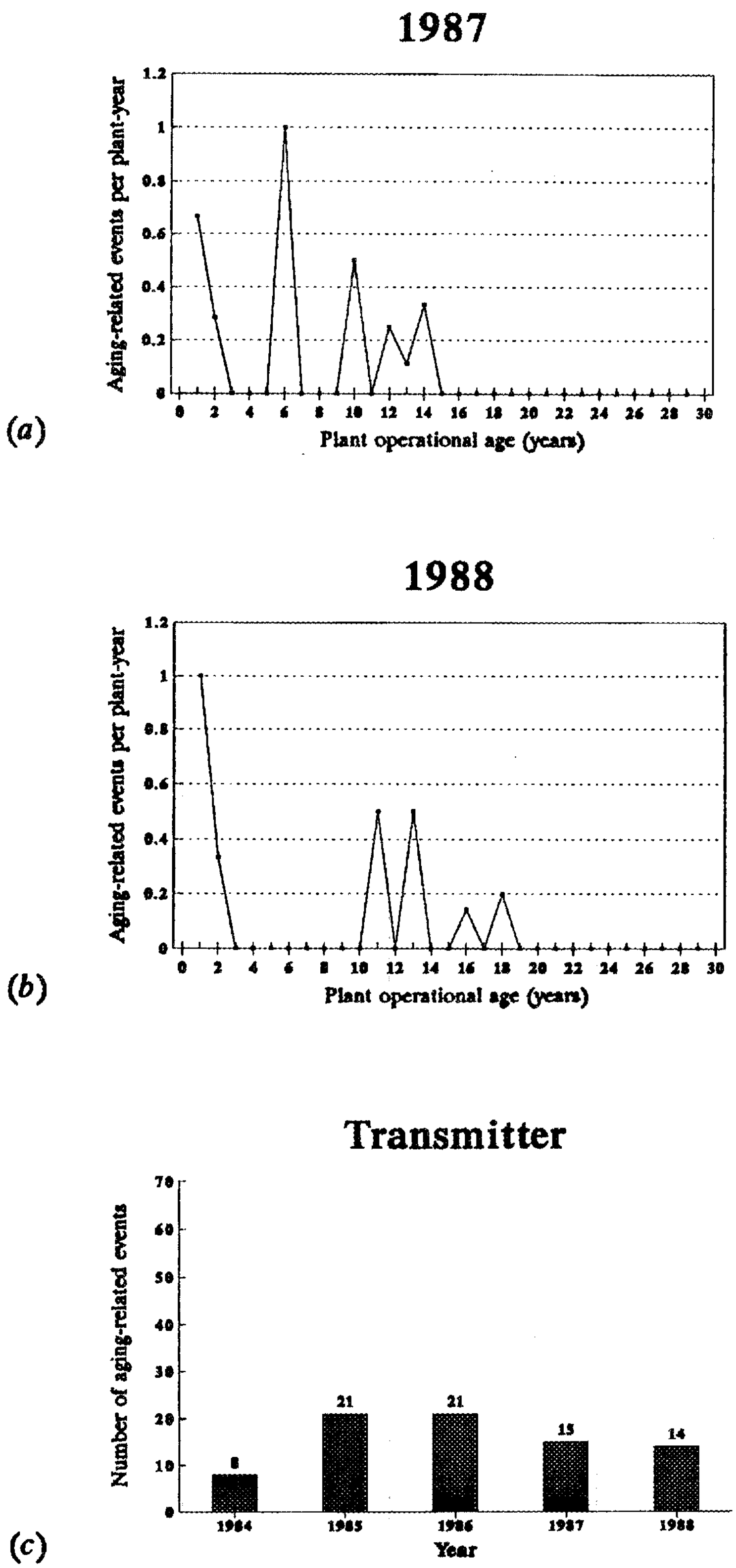


Fig. C.18. Transmitter module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Table C5. Annunciator module summary for 1984–1989
(number of LERs[a] per number of plants vs plant commercial age)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-years
	1984	1985	1986	1987	1988			
1	/7	1/8	/7	1/6	1/6	3	34	0.09
2	/3	/7	1/8	/7	/6	1	34	0.03
3	1/1	/3	1/7	/8	1/7	3	26	0.12
4	/4	1/1	/3	/7	/8	1	23	0.04
5	/2	/4	/1	/3	/7	0	17	0
6	/1	/2	/4	/1	/3	0	11	0
7	/4	1/1	/2	/4	/1	0	12	0
8	/6	/4	/1	/2	/4	0	17	0
9	/4	1/6	/4	1/1	/2	2	17	0.12
10	/9	/4	/6	/4	/1	0	24	0
11	/12	1/9	/4	/6	/4	1	35	0.03
12	/7	/12	1/9	1/4	/6	2	38	0.05
13	/6	1/7	/12	/9	/4	1	38	0.03
14	/5	/6	/7	/12	/9	0	39	0
15	/3	/5	/6	/7	/12	0	33	0
16	/2	/3	/5	/6	2/7	2	23	0.09
17	/1	/2	/3	/5	/6	0	17	0
18	/1	1/1	/2	/3	/5	1	12	0.08
19	/0	/1	/1	/2	/3	0	7	0
20	/0	/0	/1	/1	/2	0	4	0
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.5. (continued)

Plant commercial age (years)	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-years
	1984	1985	1986	1987	1988			
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number LERs	1/	6/	3/	3/	4/	17		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Report.

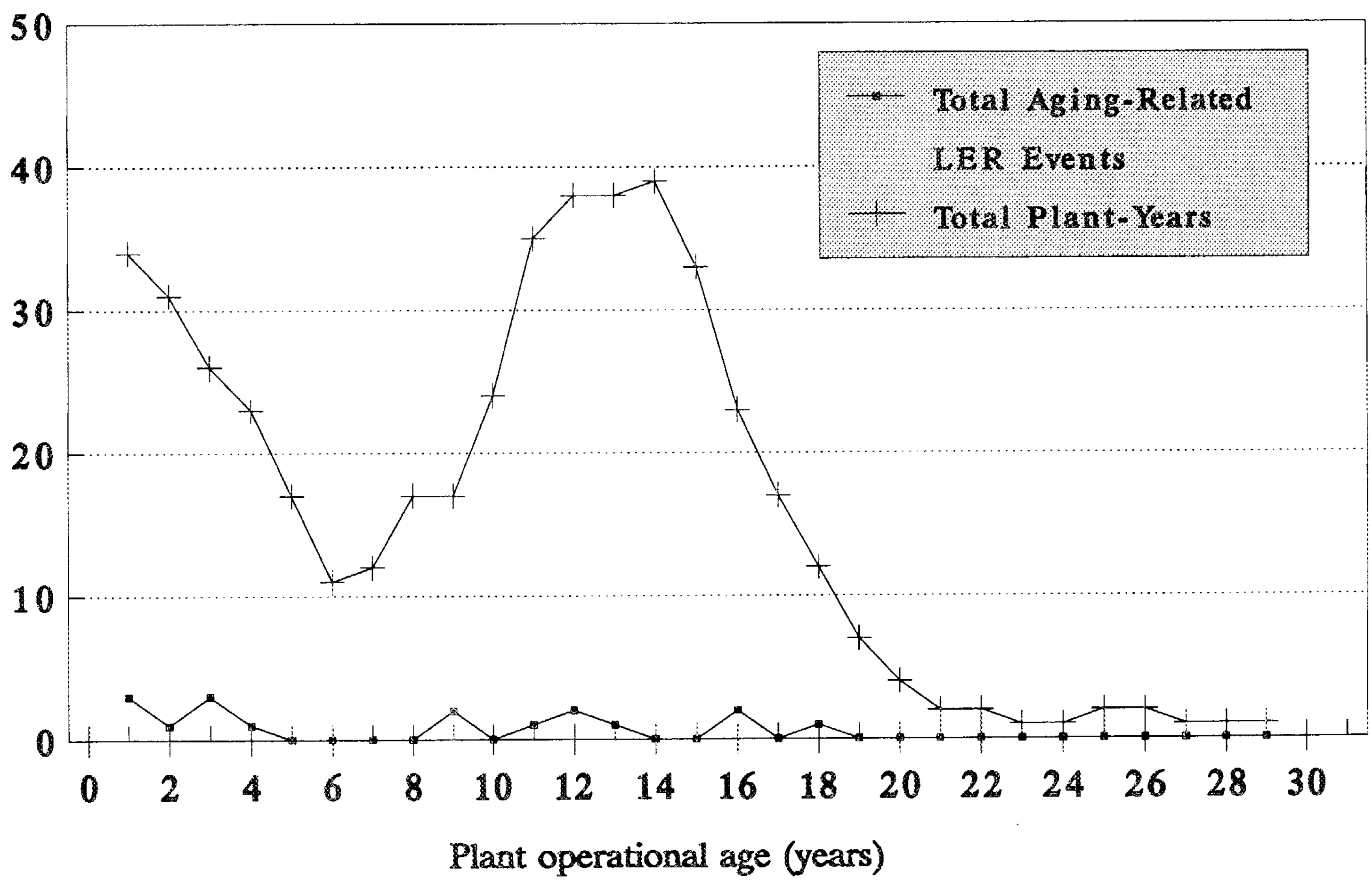


Fig. C.19. Annunciator module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

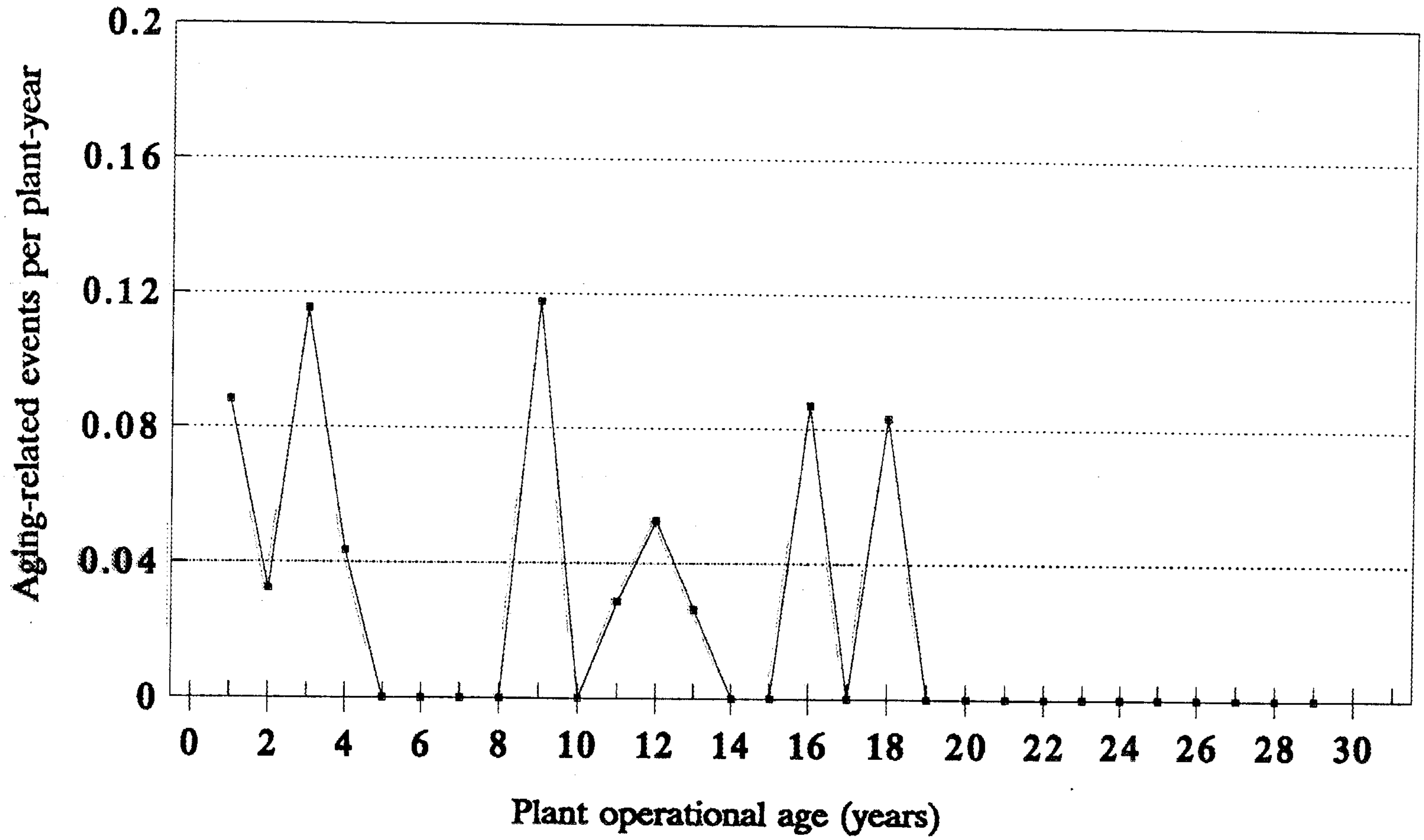


Fig. C.20. Annunciator module aging failures by plant age.

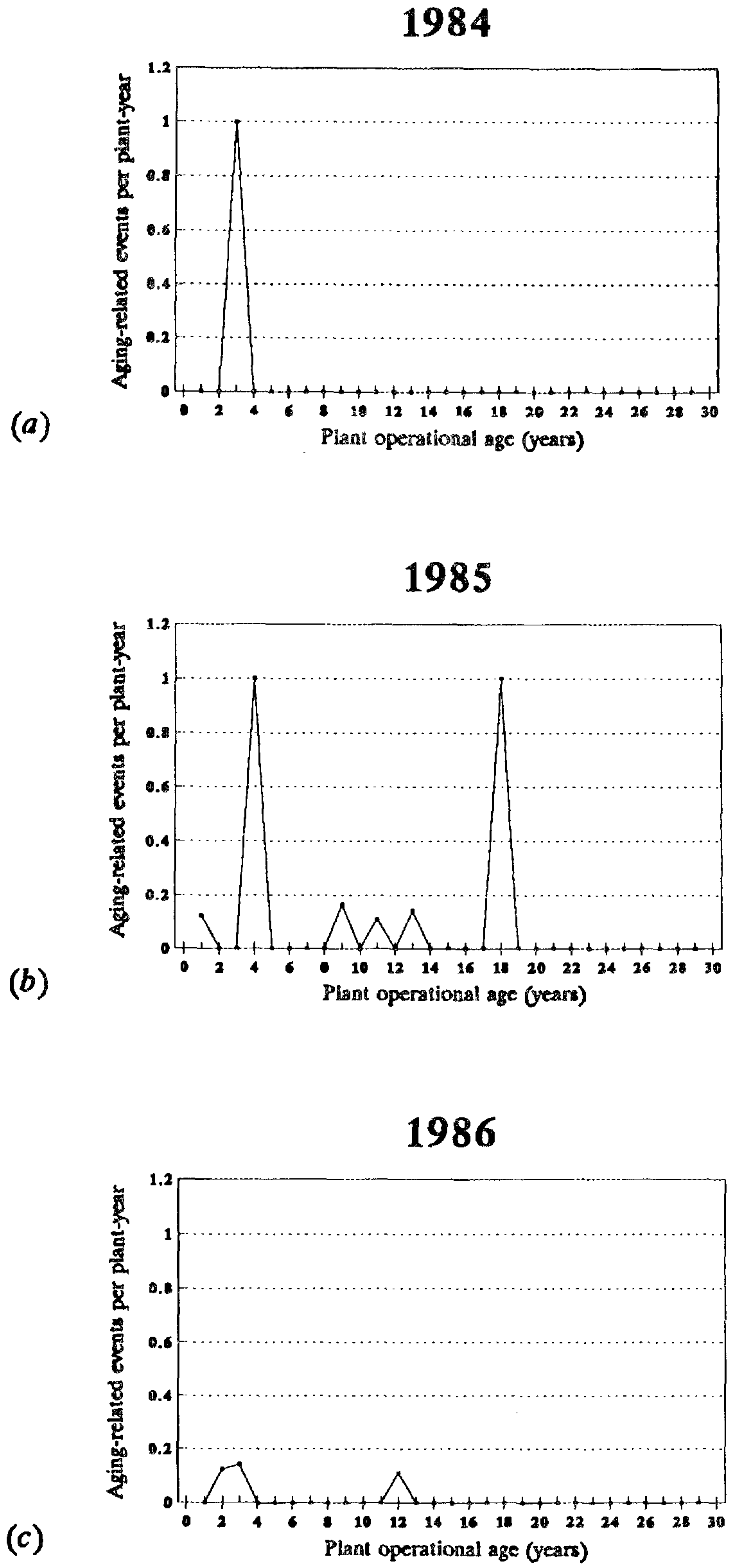


Fig. C.21. Annunciator module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.

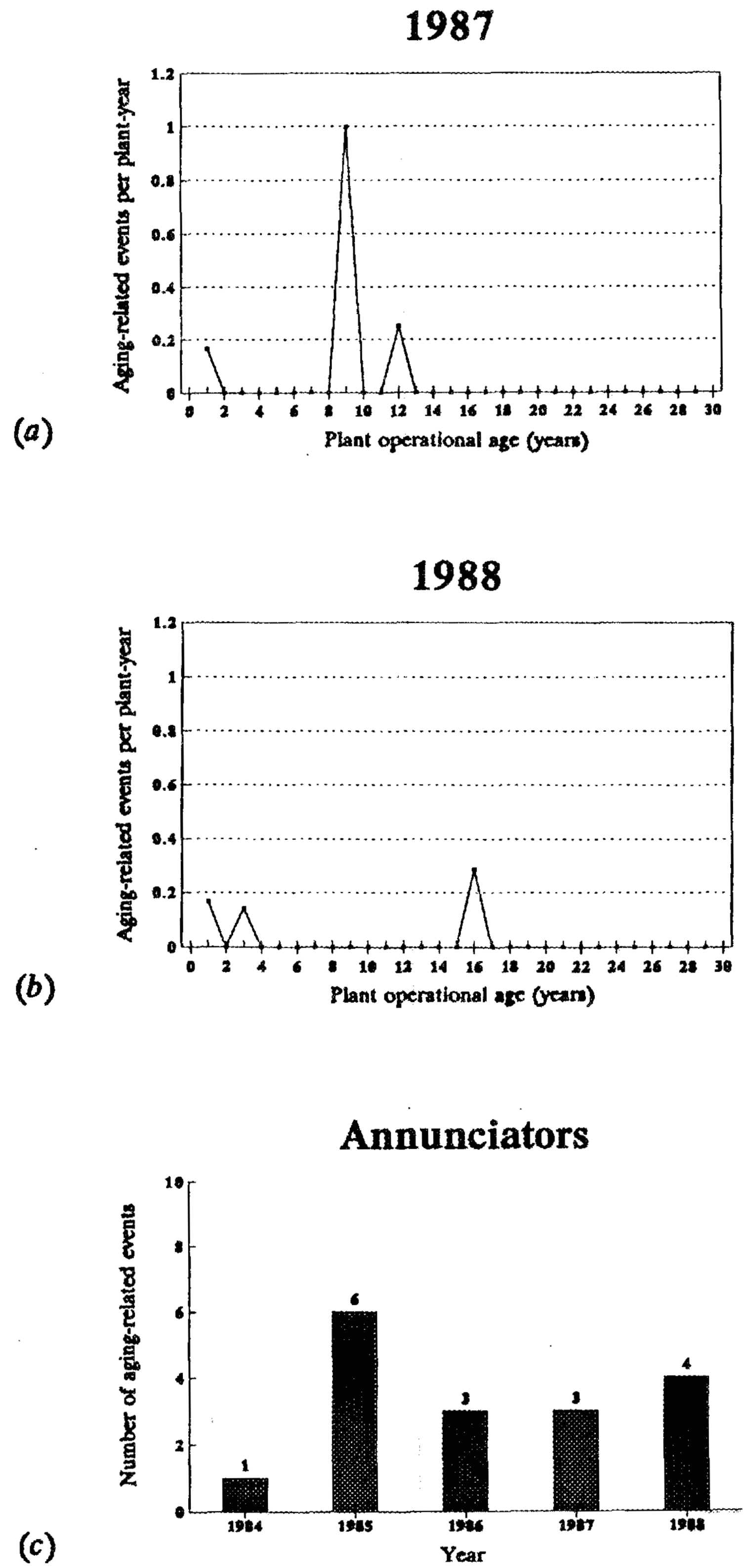


Fig. C.22. Annunciator module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Table C.6. Recorder module data summary for 1984–1989
(number of LERs[a] per number of plants vs plant commercial age)

Plant Commercial Age	Number of LERs/number of plants					Total LERs	Total plant-years	Events per plant-years
	1984	1985	1986	1987	1988			
1	/7	/8	/7	/6	1/6	1	34	0.03
2	/3	/7	/8	2/7	/6	2	31	0.06
3	/1	/3	1/7	/8	/7	1	26	0.04
4	1/4	1/1	/3	/7	/8	2	23	0.09
5	/2	/4	/1	/3	/7	0	17	0
6	/1	/2	/4	/1	/3	0	11	0
7	/4	/1	/2	/4	/1	0	12	0
8	1/6	/4	/1	/2	/4	0	17	0.06
9	/4	/6	/4	/1	/2	1	17	0
10	/9	/4	/6	/4	/1	0	24	0
11	/12	/9	/4	/6	/4	0	35	0
12	/7	/12	/9	/4	/6	0	38	0
13	/6	/7	/12	/9	/4	0	38	0
14	/5	/6	/7	/12	/9	0	39	0
15	/3	/5	/6	/7	/12	0	33	0
16	/2	/3	/5	/6	/7	0	23	0
17	/1	/2	/3	/5	/6	0	17	0
18	/1	/1	/2	/3	/5	0	12	0
19	/0	/1	/1	1/2	/3	1	7	0.14
20	/0	/0	/1	/1	/2	0	4	0
21	/0	/0	/0	/1	/1	0	2	0
22	/1	/0	/0	/0	/1	0	2	0
23	/0	/1	/0	/0	/0	0	1	0
24	/0	/0	/1	/0	/0	0	1	0
25	/1	/0	/0	/1	/0	0	2	0

Table C.6. (continued)

Plant Commercial Age	Number of LERs/number of plants 1984	1985	1986	1987	1988	Total LERs	Total plant-years	Events per plant-years
26		/1	/0	/0	/1	0	2	0
27			/1	/0	/0	0	1	0
28				/1	/0	0	1	0
29					/1	0	1	0
Total number of LERs	2/	1/	1/	3/	1/	8		
Number of licensed plants	/80	/88	/95	/101	/107			

[a]LER = Licensee Event Reports.

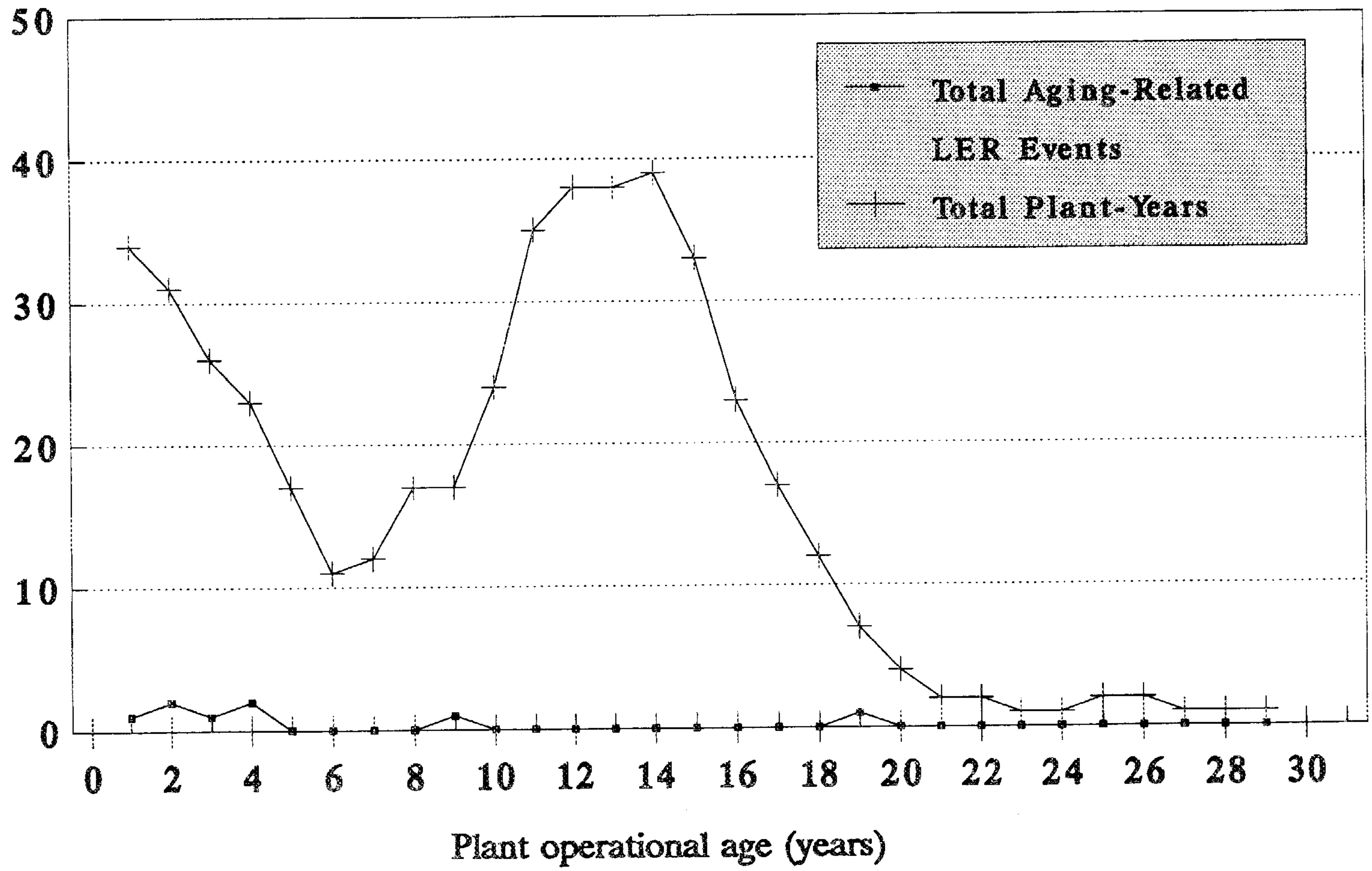


Fig. C.23. Recorder module aging-related Licensee Event Reports (LERs) for 1984 through 1988.

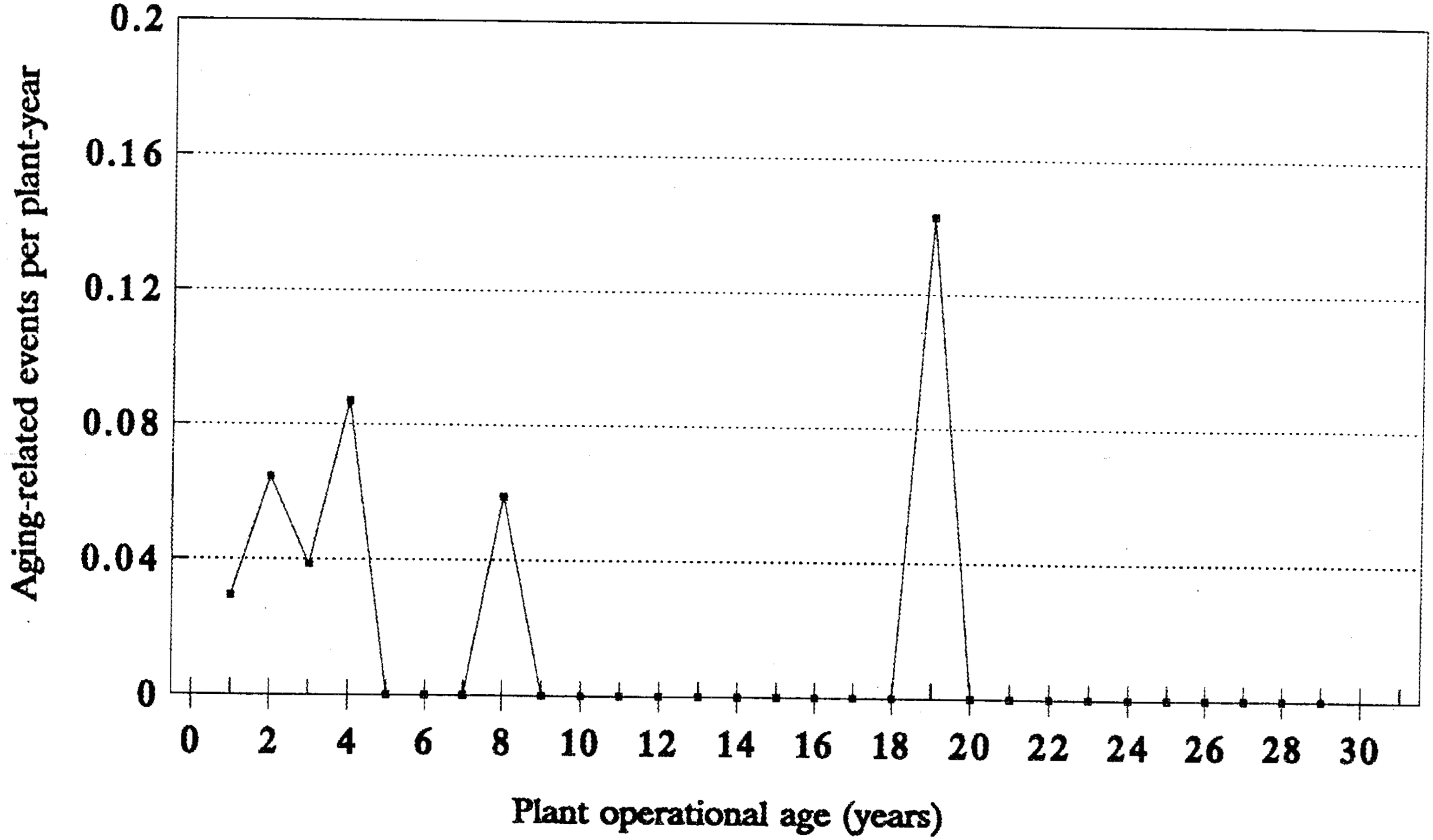
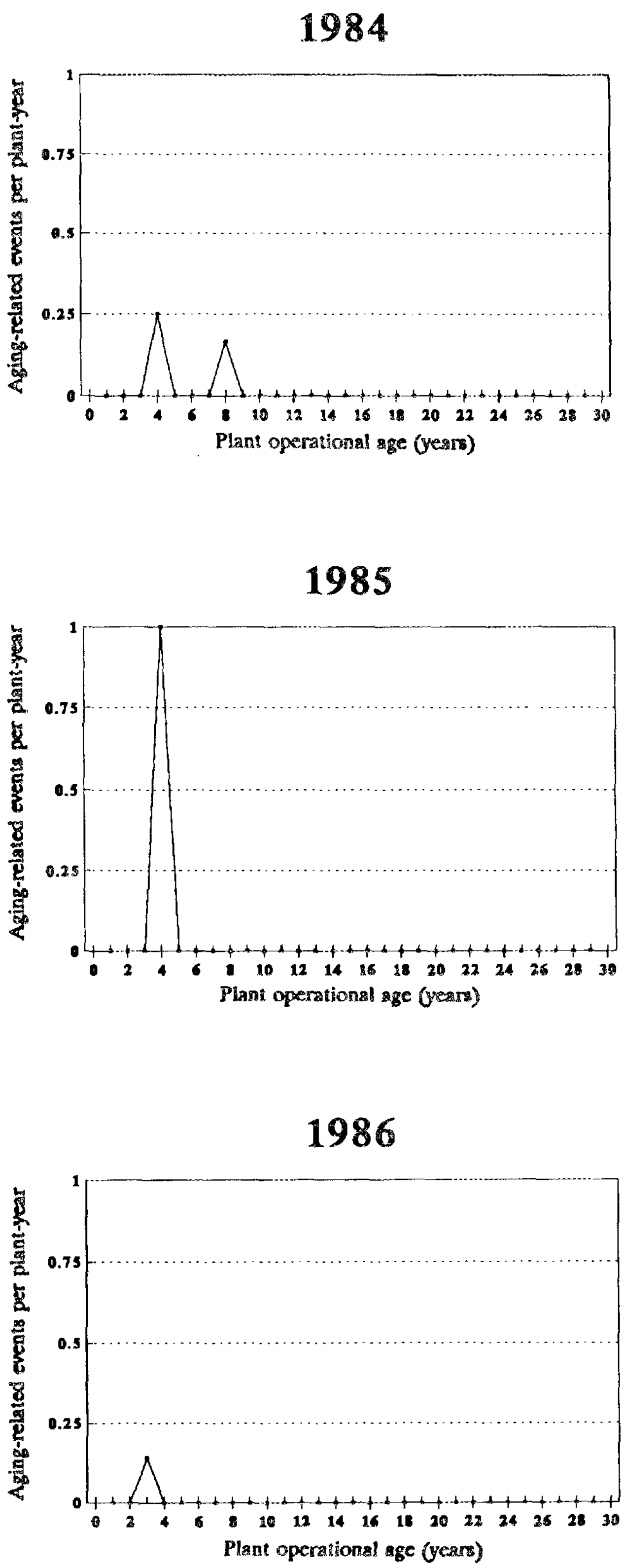
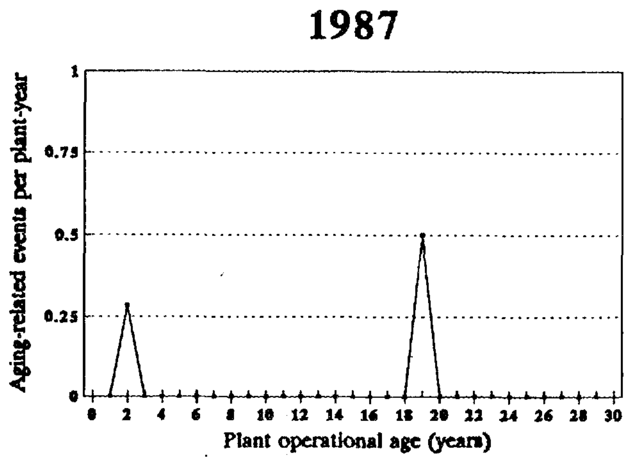


Fig. C.24. Recorder module aging failures by plant age.

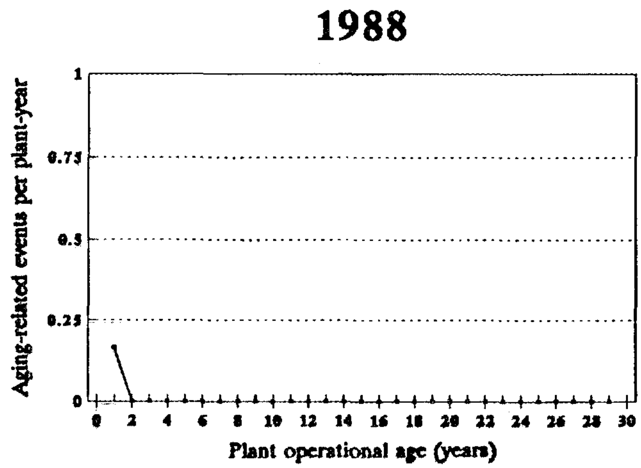


(*a*) (*b*) (*c*)

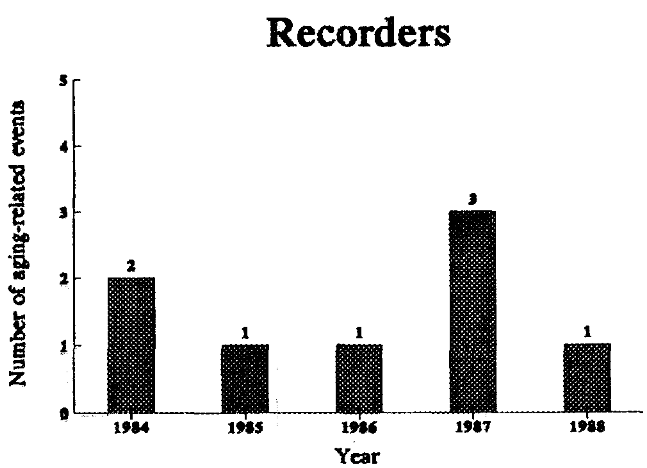
Fig. C.25. Recorder module aging-related annual event profiles for (*a*) 1984, (*b*) 1985, and (*c*) 1986.



(a)



(b)



(c)

Fig. C.26. Recorder module aging-related annual event profiles for (*a*) 1987, (*b*) 1988, and (*c*) five-year distribution of events.

Appendix D

DATABASE TOOL

Appendix D. DATABASE TOOL

A database was created as a tool for this study. The database consists of six sections, one for each of the instrumentation and control modules in this study where each section contains a compilation of aging-related, coded, and programmed information extracted from available sources. It was searchable by column combinations or by keywords in the text or a combination of both. Table D.1 identifies the columns for a full printout, and Table D.2 is such a printout for the sections on indicator modules.

Table D.1 Printout column headings

Column Headings	Description
A	U.S. Nuclear Regulatory Commission plant license docket number
B	Plant type and nuclear system vendor
C	Date of initial commercial operation for the plant
D	Event date
E	Source of event information
F	Coded functional category for the module
G	Type of application for the module
H	Instrumentation and control systems for failed module
I	Reported failure cause
J	Manner of failure detection
K	Reported corrective action
L	References of related events at the same plant

Table D.2 Database printout for instrumentation and control category - indicators

	A	B	C	D	E	F	G	H	I	J	K	L
1	50-133	BWR(GE)		05/21/88	LER 133/88-002	SIGCOND	CKTCOMP	RAD MON, GAS TREATMENT	AGE DEGRADED RESISTOR	OBSER	REMOVE FROM SERVICE BY DECOMMISSIONING PLANT	LERS 113/87-003; 113/88-001
2	50-155	BWR(GE)	03/86	11/23/87	LER 155/87-012	SIGCOND	AMPLFR	NEUTRON INTER RANGE MON	AGING ELECTRONIC TUBE FAILURE	OBSER	REPLACE AMPLIFIER	
3	50-213	PWR(W)	08/67	02/14/85	LER 213/85-004	SIGCOND	SET PT	NEUTRON OVERPOWER TRIP	NOT STATED (POWER SUPPLY DRIFT?)	TEST	REPLACE SET-POINT POTENTIOMETERS	LERS 213/84-027, 029
4	50-213	PWR(W)	08/67	07/03/85	LER 213/85-016	SIGCOND	SET PT	NEUT, DROPPED ROD RUNBACK	NOT STATED (POWER SUPPLY DRIFT?)	TEST	REPLACE SYSTEM WITH STATE-OF-THE ART EQUIPT	LER 213/85-005
5	50-213	PWR(W)	08/67	06/27/86	LER 213/86-030	SIGCOND	SET PT	NEUT, DROPPED ROD RUNBACK	AGING NUCLEAR INSTRUMENTATION	OBSER	REPLACE SYSTEM WITH STATE-OF-THE ART EQUIPT	LERS 213/86-023,022; 213/85-015,011; 213/84-029
6	50-213	PWR(W)	08/67	07/10/86	LER 213/86-032	SIGCOND	SET PT	NEUT, DROPPED ROD RUNBACK	AGING NUCLEAR INSTRUMENTATION	OBSER	REPLACE SYSTEM WITH STATE-OF-THE ART EQUIPT	LERS 213/84-027; 213/85-004,016
7	50-213	PWR(W)	08/67	11/08/86	LER 213/86-043	XDCR	RADIATN	RAD MON, NOBLE GAS STACK	PM TUBE AND HV POWER SUPPLY FAIL	OBSER	REPLACE HV SUPPLY AND PM TUBE	
8	50-213	PWR(W)	08/67	04/21/87	LER 213/87-006	CHANNEL	NEUTRON	MONITOR, POWER RANGE	AGING NUCLEAR INSTRUMENTATION	OBSER	SCHEDULED MODERNIZATION IN 1989	
9	50-219	BWR(GE)	12/69	01/28/88	LER 219/88-002	SIGCOND	AMPLFR	RAD MON, CONTAIN HI-RANGE	DEFECTIVE COMPONENTS IN MONITOR PREAMPS	TEST	REPAIR AND RETURN TO SERVICE	
10	50-237	BWR(GE)	06/70	03/31/86	LER 237/86-008	XDCR	RADIATN	RAD MON, REFUEL FLOOR	GEIGER-MUELLER TUBE FAILURE	OBSER	REPLACE G-M TUBE, RECALIBRATE	
11	50-237	BWR(GE)	06/70	09/15/87	LER 237/87-026	SIGCOND	SET PT	LEVEL, REACTOR LOW WATER	MASTER TRIP UNIT LOGIC CARD DRIFT	TEST	REPLACE LOGIC CARD, REPAIR DEFECTIVE CARD	LER 249/85-004
12	50-244	PWR(W)	07/70	08/04/88	LER 244/88-007	CHANNEL	RAD MON	CONTAINMENT PARTICULATE	BRIDGE RECTIFIER RANDOM FAILURE	OBSER	REPLACE UNIT	LER 244/87-005
13	50-245	BWR(GE)	03/71	08/06/88	LER 245/88-006	CHANNEL	RAD MON	REFUEL FLOOR MONITOR	NOT STATED	OBSER	REPLACE ALL ACTIVE COMPONENTS	LER 245/87-013
14	50-247	PWR(W)	08/74	12/31/87	LER 247/87-020	XDCR	TEMP	TEMP, HOT AND COLD LEG	RTD LEADS NOT VAPOR TIGHT AS REQUIRED	OBSER	REPLACE, PLACE IN CONDUIT AND ENCAPSULATE	
15	50-249	BWR(GE)	11/71	12/22/86	LER 249/86-022	XDCR	RADIATN	RAD MON, FUEL POOL FLOOR	FAILED GM TUBE, RESISTORS, CAPACITOR	OBSER	REPLACED SENSOR AND CONVERTER SECTIONS	LER 237/86-008
16	50-249	BWR(GE)	11/71	09/15/87	LER 249/87-026	SIGCOND	CKTCOMP	LOW WATER LEVEL SCRAM	FAILED ZDIODE, MASTER TRIP UNIT LOGIC CARD	OBSER	REPLACE LOGIC CARD, REPAIR DEFECTIVE CARD	LER 249/85-004
17	50-250	PWR(W)	12/72	11/30/85	LER 250/85-040	SIGCOND	HVPS	INTERMEDIATE RANGE NEUT	FAILED HV POWER SUPPLY, CAPACITOR	OBSER	REPLACE HV POWER SUPPLY, INPUT CAPACITOR	LER 250/84-021
18	50-250	PWR(W)	12/72	03/05/86	LER 250/86-012	XDCR	NEUTRON	SOURCE RANGE NEUTRON	FAILED DETECTOR, FAILED DIODE IN PREAMP	OBSER	REPLACE DETECTOR, DIODE IN PREAMP	LERS 251/84-014, 023; 251/85-025
19	50-250	PWR(W)	12/72	09/21/86	LER 250/86-034	CHANNEL	NEUTRON	SOURCE RANGE NEUTRON	NOT STATED	OBSER	NOT STATED	
20	50-251	PWR(W)	09/73	07/11/86	LER 251/86-012	SIGCOND	CKTCOMP	CONTAINMENT RAD MON, HVAC	FAILED ZDIODE	OBSER	REPLACED ZDIODE	LERS 251/84-004; 251/85-014; 251/86-011
21	50-251	PWR(W)	09/73	11/14/86	LER 251/86-026	SIGCOND	CKTCOMP	CONTAINMENT RAD MON, HVAC	FAILED RESISTOR, METER IN RATEMETER MODULE	OBSER	REPLACED RESISTOR AND METER IN RATEMETER	LERS 251/84-004; 251/85-014; 215/86-011, 012
22	50-251	PWR(W)	09/73	12/05/86	LER 251/86-027	XDCR	RADIATN	CONTAINMENT RAD MON, HVAC	FAILED DETECTOR, RESISTOR; BROKEN SOLDER JNT	OBSER	REPLACED DETECTOR, RESISTOR; REPAIRED JOINT	LERS 251/84-004; 251/85-014; 215/86-011, 012, 026
23	50-251	PWR(W)	09/73	12/12/86	LER 251/86-029	SIGCOND	CKTCOMP	CONTAINMENT RAD MON, HVAC	FAILED RESISTOR, CAPACITOR, SOLDER JOINTS	OBSER	REPLACED CAPACITOR, RESISTOR; REPAIR JOINTS	LERS 251/86-011, 012, 026, 027
24	50-251	PWR(W)	09/73	02/12/87	LER 251/87-005	SIGCOND	CKTCOMP	CONTAINMENT RAD MON, HVAC	FAILED FAILED TRANSISTOR, RESISTOR	OBSER	REPLACED TRANSISTOR, RESISTOR	LERS 250/87-008; 251/84-004; 251/85-014; 251/86-011, 012, 026, 027,029
25	50-251	PWR(W)	09/73	07/29/87	LER 251/87-021	SIGCOND	CKTCOMP	CONTAINMENT RAD MON, HVAC	FAULT PRONE PRM CIRCUITRY	TEST	REPLACED WITH UPGRADED SOLID STATE CIRCUITRY	LERS 250/87-005, 008; 251/86-030
26	50-251	PWR(W)	09/73	09/15/87	LER 251/87-023	XDCR	RADIATN	CONTAINMENT RAD MON, HVAC	DETECTOR, JAMMED PAPER FILTER TAPE	OBSER	REPLACED XDCR, FILTER PAPER; ADJUSTED DRIVE	LERS 250/87-005, 008; 251/86-030; 251/87-021
27	50-251	PWR(W)	09/73	08/12/88	LER 251/88-007	XDCR	NEUTRON	SOURCE RANGE NEUTRON	FAILED DETECTOR	OBSER	REPLACED DETECTOR	LER 250/87-022
28	50-254	BWR(GE)	02/73	05/31/85	LER 254/85-007	SIGCOND	SPECFN	FUEL POOL RAD MON, HVAC	FAILED "SENSOR CONVERTER"	OBSER	REPLACED "SENSOR CONVERTER"	LERS 254/85-005, 012, 014
29	50-254	BWR(GE)	02/73	10/02/86	LER 254/86-029	XDCR	CHEM	AMMONIA ANALYZER, HVAC	FAILED TIMING BELT IN SENSOR (HIGH AMB TEMP)	OBSER	REPLACED BELT, CALIBRATED, TESTED	
30	50-254	BWR(GE)	02/73	06/29/87	LER 254/87-013	XDCR	CHEM	CHLORINE ANALYZER, HVAC	PROBE CORROSION, FAULTY TEST PROCEDURE	TEST	REPLACED PROBE, NEW TEST PROCEDURE	LERS 254/86-029; 254/87-010
31	50-255	PWR(CE)	12/71	09/03/87	LER 255/87-033	XDCR	RADIATN	HIGH-RANGE STACK MONITOR	FAILED ION CHAMBER ANODE INSULATOR	TEST	INSTALLED ALTERNATE; RETURNED TO MFR	
32	50-259	BWR(GE)	08/74	06/08/86	LER 259/86-019	SIGCOND	CKTCOMP	REFUEL ZONE RAD MON, HVAC	FAULTY ELECTRICAL CONNECTOR PLUG	OBSER	REPLACED FEMALE PORTION OF CONNECTOR	LER 259/85-035
33	50-259	BWR(GE)	08/74	07/12/87	LER 259/87-013	XDCR	RADIATN	REFUEL ZONE RAD MON, HVAC	OUTPUT SPIKE TO TRIP POINT	OBSER	REPLACED DETECTOR	LERS 259/86-019; 296/86-010
34	50-261	PWR(W)	03/71	06/24/88	LER 261/88-014	SIGCOND	CKTCOMP	STEAM GEN LEVEL CHANNEL	FAILED CAPACITOR	TEST	REPLACED CAPACITOR	
35	50-263	BWR(GE)	06/71	01/10/84	LER 263/84-004	XDCR	CHEM	CHLORINE ANALYZER, HVAC	TAPE CASSETTE RAN OUT, TAPE BREAK	OBSER	NOT STATED	LER 263/84-002
36	50-263	BWR(GE)	06/71	02/09/86	LER 263/86-005	SIGCOND	POW SUP	REACTOR BLDG VENT RAD MON	FAILED POWER SUPPLY	OBSER	REPAIRED POWER SUPPLY	
37	50-263	BWR(GE)	06/71	06/20/86	LER 263/86-021	XDCR	FLOW	REACTOR BLDG VENT RAD MON	FAILED FLOW SENSOR	OBSER	REPLACED HIGH FLOW SENSOR	LER 263/85-012
38	50-263	BWR(GE)	06/71	02/16/87	LER 263/87-006	SIGCOND	CKTCOMP	REACTOR BLDG VENT RAD MON	FAILED VARISTOR, PROCESSING UNIT, FUSE	OBSER	REPLACED VARISTOR, PROCESSING UNIT, FUSE	
39	50-265	BWR(GE)	03/73	06/28/85	LER 265/85-015	XDCR	RADIATN	REACTOR BLDG VENT RAD MON	FAILED GEIGER-MUELLER TUBE	OBSER	REPLACED GEIGER-MUELLER TUBE	
40	50-272	PWR(W)	06/77	01/07/84	LER 272/84-002	SIGCOND	SPECFN	STEAM GEN LEVEL CONTROL	FAILED SIGNAL SUMMATOR MODULE	OBSER	REPLACED SIGNAL SUMMATOR MODULE	
41	50-272	PWR(W)	06/77	10/02/87	LER 272/87-013	XDCR	NEUTRON	SOURCE RANGE NEUTRON	WATER CONDENSED IN DETECTOR HOUSING	OBSER	REPLACED DETECTOR; NEW HOUSING DESIGN	
42	50-275	PWR(W)	05/85	10/22/87	LER 275/87-017	XDCR	RADIATN	CONTROL ROOM RAD MON, HVAC	SEIZED PAPER DRIVE MOTOR	OBSER	REPLACED PAPER DRIVE MOTOR, CIRCUIT MOD	LERS 275/85-023, 025, 027; 275/86-007, 015; 323/85-001, 003, 005, 008; 323/86-010; 323/87-003,008
43	50-280	PWR(W)	12/72	04/07/84	LER 280/84-008	XDCR	NEUTRON	SOURCE RANGE NEUTRON	NOISY DETECTOR	OBSER	REPLACED DETECTOR	
44	50-280	PWR(W)	12/72	04/07/84	LER 280/84-008	SIGCOND	AMPLIF	SOURCE RANGE NEUTRON	FAILED PREAMPLIFIER	OBSER	REPLACED PREAMPLIFIER	
45	50-280	PWR(W)	12/72	07/27/87	LER 280/87-017	SIGCOND	CKTCOMP	PROCESS VENT RAD MON	FAILED CPU CARD	OBSER	REPLACED CPU CARD	
46	50-280	PWR(W)	12/72	08/07/87	LER 280/87-019	RECEIVER	ANNUNC	XMFR ANNUNCIATOR PANEL	FAILED ANNUNCIATOR PANEL	OBSER	REPLACED TRANSFORMER LOCAL ANNUNCIATOR PANEL	
47	50-280	PWR(W)	12/72	10/18/87	LER 280/87-026	XDCR	CHEM	CHLORINE ANALYZER, HVAC	CHLORINE SENSOR TEMPERATURE COEFFICIENT	OBSER	PLAN TO REPLACE CHLORINE TREATMENT SYSTEM	LERS 280/87-016, 020, 022
48	50-280	PWR(W)	12/72	11/11/87	LER 280/87-032	SIGCOND	SPECFN	TAVG PROTECTION CHANNELS	FAILED SIGNAL SUMMATOR MODULE	OBSER	REPLACED SIGNAL SUMMATOR MODULE	
49	50-280	PWR(W)	12/72	08/02/88	LER 280/88-023	SIGCOND	SPECFN	STEAM GENERATOR FLOW	FAILED MULTIPLIER/DIVIDER POWER SUPPLY	OBSER	REPLACED MULTIPLIER/DIVIDER	LER 281/88-018
50	50-281	PWR(W)	05/73	08/11/88	LER 281/88-018	SIGCOND	SPECFN	STEAM GENERATOR FLOW	FAILED MULTIPLIER/DIVIDER POWER SUPPLY	OBSER	REPLACED MULTIPLIER/DIVIDER	LER 280/88-023
51	50-295	PWR(W)	12/73	05/16/84	LER 295/84-014	SIGCOND	CKTCOMP	VENT STACK RAD MON	FAILED TRANSISTOR	OBSER	VENTING WAS SECURED	
52	50-295	PWR(W)	12/73	12/06/85	LER 295/85-044	XDCR	FLOW	STEAM GENERATOR FLOW	FAILED FISHER-PORTER STEAM FLOW TRANSMITTER	OBSER	REPLACE FLOW XMTR; PLAN UPGRADE	LER 295/85-045
53	50-295	PWR(W)	12/73	12/05/85	LER 295/85-046	SIGCOND	AMPLIF	SOURCE RANGE NEUTRON	FAILED PREAMPLIFIERS (2)	OBSER	REPLACED PREAMPLIFIERS; SCHEDULE UPGRADE	LER 295/85-044
54	50-296	BWR(GE)	03/77	09/19/86	LER 296/86-010	XDCR	RADIATN	REFUEL ZONE RAD MON	FAILED DETECTOR AND CONVERTER	OBSER	REPLACED DETECTOR AND CONVERTER TWICE	
55	50-304	PWR(W)	09/74	06/27/86	LER 304/86-016	XDCR	TEMP	OVER TEMP DELTA T	FIVE (5) HOT LEG RTD'S DAMAGED BY LIGHTNING	OBSER	REPAIR INSTR; UPGRADE LIGHTNING SYSTEM	
56	50-305	PWR(W)	06/74	05/02/88	LER 305/88-006	SIGCOND	SPECFN	OVER TEMP DELTA T	FAULTY CONNECTION IN SET-POINT CALCULATOR	TEST	REPLACED FAULTY WIRES IN FOXBORO BOX	
57	50-312	PWR(BW)	04/75	08/26/86	LER 312/86-019	RECEIVER	INDICAT	AUX BLDG GAS RAD MON	FAULTY FLOW INDICATOR	OBSER	CORRECTIVE ACTION NOT STATED	LER 312/86-003
58	50-316	PWR(W)	07/78	10/22/85	LER 316/85-032	SIGCOND	POW SUP	CONTAINMENT AREA RAD MON	TWO (2) FAILED 12-V POWER SUPPLY MODULES	OBSER	REPLACED FAILED MODULES	LER 316/84-003
59	50-316	PWR(W)	07/78	07/15/86	LER 316/86-023	XDCR	VOLUME	ACCUMULATOR VOLUME	DEFECTIVE INTERNAL INSTRUMENT COMPONENTS	TEST	REPLACED COMPONENTS; ULTRASONIC TESTING	LER 316/86-024
60	50-316	PWR(W)	07/78	07/18/86	LER 316/86-024	SIGCOND	AMPLIF	FEEDWATER FLOW XMTR	FAULTY AMPLIF IN FEEDWATER FLOW TRANSMITTER	OBSER	INITIATE INVESTIGATION BY FOXBORO CO.	
61	50-316	PWR(W)	07/78	07/22/87	LER 316/87-008	RECEIVER	CONTRLR	FW PUMP DELTA P CONTROL	COULD NOT IDENTIFY	OBSER	REPLACED DELTA P CONTROLLER	
62	50-321	BWR(GE)	12/75	08/08/87	LER 321/87-014	XDCR	RADIATN	REFUELING FLOOR RAD MON	G-M TUBE FAILURE DUE TO GAS LEAKAGE	OBSER	REPLACED TUBE, RECALIBRATE CHANNEL	LER 321/87-006
63	50-321	BWR(GE)	12/75	80/27/87	LER 321/87-015	XDCR	RADIATN	REFUELING FLOOR RAD MON	G-M TUBE FAILURE FROM MISHANDLING, STORAGE	OBSER	REPLACED TUBE, RECALIBRATE CHANNEL	LER 321/87-006, 014
64	50-321	BWR(GE)	12/75	12/14/87	LER 321/87-016	XDCR	CHEM	CHLORINE DETECTOR, HVAC	ELECTROLYTE RESERVOIR RAN DRY	OBSER	REPAIR, RECAL, REVISE PROCEDURE	LERS 321/85-033, 037
65	50-323	PWR(W)	03/86	04/05/88	LER 323/88-005	CHANNEL	RADIATN	PLANT VENT RAD MON, HVAC	NOISE PICKUP FROM CYCLING MOTOR NEARBY	OBSER	REPLACED DRIVE MOTOR ASSEMBLY	LERS 275/88-003, 005, 007; 323/88-003, 004
66	50-324	BWR(GE)	11/75	05/04/87	LER 324/87-007	SIGCOND	CKTCOMP	RCIC ISOLATION INSTRUMENT	LONG-TERM THERMAL DEGRADATION OF COMPONENTS	OBSER	REPLACED INSTRUMENT	LERS 324/83-058; 324/84-017; 325/86-008, 014
67	50-325	BWR(GE)	03/77	05/01/85	LER 325/85-021	XDCR	FIRE	CONTROL BUILDING HVAC	ONE FROM DUST, FOUR UNSTATED	OBSER	VERIFIED NO FIRE; RETURNED TO STANDBY	
68	50-325	BWR(GE)	03/77	05/01/85	LER 325/85-021	XDCR	CHEM	CHLORINE DETECTOR, HVAC	ELECTROLYTE RESERVOIR RAN DRY	OBSER	NOT STATED	
69	50-325	BWR(GE)	03/77	01/06/86	LER 325/86-003	XDCR	CHEM	CHLORINE DETECTOR, HVAC	DEFECTIVE CIRCUIT BOARD	OBSER	REPLACE CKT BOARD; STUDY POSSIBLE CHANGES	LER 235/84-033
70	50-325	BWR(GE)	03/77	01/13/86	LER 325/86-005	SIGCOND	CKTCOMP	CONTROL ROOM RAD MON	FAILURE OF RELAY COIL AND CIRCUIT BOARD	OBSER	REPLACE RELAY COIL AND CIRCUIT BOARD	
71	50-325	BWR(GE)	03/77	03/17/86	LER 325/86-008	SIGCOND	CKTCOMP	RWCU HIGH-TEMP ISOLATION	COMPONENT BREAKDOWN, CKT AND MODULE FAILURES	OBSER	REPLACED TEMPERATURE CONVERTER MODULE	

Table D.2 (continued)

	A	B	C	D	E	F	G	H	I	J	K	L
72	50-325	BWR(GE)	03/77	05/30/86	LER 325/86-013	XDCR	RADIATN	COMMON AIR INTAKE RAD MON	MOISTURE INTRUSION AND CORROSION	OBSER	REPLACED SENSOR/CONVERTER	
73	50-325	BWR(GE)	03/77	07/06/86	LER 325/86-015	XDCR	RADIATN	COMMON AIR INTAKE RAD MON	FAILURE OF SENSOR/CONVERTER	OBSER	REPLACED SENSOR/CONVERTER	
74	50-331	BWR(GE)	02/75	09/04/85	LER 331/85-035	CHANNEL	TACHOM	RCIC TURBINE TACHOMETER	"INACCURATE"	OBSER	ORDERED NEW CONTROL BOX WITH TACH CIRCUIT	
75	50-331	BWR(GE)	02/75	10/05/85	LER 331/85-042	OTHER	TIMER	DRYWELL PUMP/FILL TIMERS	LOGIC CONTACTS REVERSED SINCE 1974	OBSER	CONTACTS REVERSED TO PRỌPER POSITION	
76	50-333	BWR(GE)	07/75	09/05/87	LER 333/87-013	CHANNEL	FLOW	RCIC STEAM LINE FLOW	INVESTIGATION REVEALED NO IDENTIFIABLE CAUSE	OBSER	REPLACED TRANSMITTER AND MASTER TRIP UNIT	LERS 333/85-028; 333/87-012
77	50-334	PWR(W)	10/76	02/10/86	LER 334/86-001	XDCR	NEUTRON	SOURCE RANGE CHANNEL	NOT STATED	OBSER	REPLACED SOURCE RANGE DETECTOR	
78	50-334	PWR(W)	10/76	12/09/87	LER 334/87-019	SIGCOND	CKTCOMP	FUEL BLDG VENT RAD MON	FAILED CAPACITORS IN POWER SUPPLY; WIRING	OBSER	REPLACED CAPACITORS; CORRECTED WIRING	
79	50-338	PWR(W)	06/78	09/17/85	LER 338/85-017	OTHER	ALARM	DROPPED CONTROL RODS	INTERMITTENT FAULTS IN THREE ALARM CKT CARDS	OBSER	REPLACED ALARM CIRCUIT CARDS	
80	50-338	PWR(W)	06/78	11/23/87	LER 338/87-020	XDCR	LEVEL	TURBINE SOLENOID TRIP	FATIGUE FAILURE OF SPRING IN LEVEL SWITCH	OBSER	REPLACED MICROSWITCH; INSPECTED ALL OTHERS	
81	50-338	PWR(W)	06/78	11/24/87	LER 338/87-023	SIGCOND	SPECFN	PROCESS VENT RAD MON	FAILURE OF RAD MON CPU CARD, CONNECTOR	OBSER	REPLACED CPU CARD, TIGHTENED EDGE CONNECTOR	
82	50-338	PWR(W)	06/78	12/04/87	LER 338/87-024	SIGCOND	SPECFN	VENT STACK RAD MON (KAMAN)	PROBLEM WITH NEW PROMS IN MICROPROCESSOR	OBSER	REPLACED PROMS WITH ORIGINAL UNITS, CPU CARD	LER 338/86-003
83	50-338	PWR(W)	06/78	02/10/88	LER 338/88-010	SIGCOND	SPECFN	VENT STACK RAD MON (KAMAN)	MICROPROCESSOR CPU BOARD MALFUNCTION	OBSER	REPLACED CPU CARD, UPGRADE MODIFICATION	LER 338/87-023; 338/88-006
84	50-339	PWR(W)	12/80	09/12/85	LER 339/85-011	SIDCOND	XMTR	VENT STACK RAD MONITOR	MICROPROCESSOR CPU BOARD MALFUNCTION	OBSER	REPLACED CPU BOARD	
85	50-339	PWR(W)	12/80	08/24/87	LER 339/87-006	XDCR	NEUTRON	INTERMEDIATE RANGE MONITOR	NEUTRON DETECTOR FAILURE	OBSER	WILL REPLACE DETECTOR	
86	50-341	BWR(GE)	01/88?	09/03/85	LER 341/85-059	CHANNEL	FLOW	MAIN STEAM LINE	FLOW INDICATION FAILED	OBSER	NOT STATED	
87	50-344	PWR(W)	05/76	11/14/86	LER 344/86-011	CHANNEL	VIBRAT'N	MAIN TURB BEARING (RS100%)	LOOSE CONNECTIONS IN JUNCTION BOX	OBSER	LOOSE CONNECTIONS TIGHTENED	
88	50-344	PWR(W)	05/76	11/14/86	LER 344/86-011	XDCR	NEUTRON	INTERMEDIATE RANGE MONITOR	COMPENSATED ION CHAMBER FAILURE	OBSER	REPLACED COMPENSATED ION CHAMBER	
89	50-344	PWR(W)	05/76	08/30/87	LER 344/87-025	XDCR	RADIATN	LOW LEVEL NOBLE GAS MON	FAULTY DETECTOR GENERATED INTERMITTENT NOISE	OBSER	REPLACED PRM-1C DETECTOR	
90	50-344	PWR(W)	05/76	04/10/88	LER 344/88-007	XDCR	CHEM	CHLORINE DETECTOR, HVAC	NORMAL EQUIPMENT WEAR AND AGING	OBSER	FAULTY DETECTOR REPLACED	
91	50-346	PWR(BW)	07/78	11/20/84	LER 346/84-018	CHANNEL	NEUTRON	NIMBIN MODULE	BAD RATE METER MODULE	OBSER	MODULE REPLACED, PROCEDURE MODIFICATION	LER 346/80-045
92	50-346	PWR(BW)	07/78	09/06/87	LER 346/87-011	SIGCOND	XMTR	FEEDWATER FLOW	NOT STATED	OBSER	NOT STATED	LERS 346/87-001, 006, 010
93	50-346	PWR(BW)	07/78	01/06/88	LER 346/88-001	CHANNEL	SEISMIC	SEISMIC TRIGGER	DOES NOT MEET FREQUENCY RANGE REQUIREMENTS	TEST	NEW TRIGGER WILL BE OBTAINED, TS AMENDMENT	LERS 346/77-013; 346/83-035; 346/87-003
94	50-346	PWR(BW)	07/78	08/09/88	LER 346/88-018	CHANNEL	RADIATN	STATION VENT RAD MONITOR	HIGH-VOLTAGE BOARD FAILURE, PROCEDURAL ERROR	OBSER	REPLACE HIGH-VOLTAGE BOARD, REVISE PROCEDURES	LER 346/87-009
95	50-348	PWR(W)	12/77	04/05/87	LER 348/87-007	XDCR	NEUTRON	SOURCE RANGE DETECTOR	DETECTOR FAILURE	OBSER	REPLACE DETECTOR	
96	50-352	BWR(GE)	02/86	10/23/85	LER 352/85-085	XDCR	CHEM	CHLORINE DETECTOR, HVAC	OPTIC BULB FAILURE IN DETECTOR	OBSER	REPLACE BOTH BULBS IN DETECTOR	LERS 352/84-008,010, 028, 033, 046; 352/85-029, 030, 031, 042, 050, 059, 063, 081, 086
97	50-352	BWR(GE)	02/86	12/07/85	LER 352/85-094	XDCR	CHEM	CHLORINE DETECTOR, HVAC	ANALYZER SERVOMECHANISM FAILURE	OBSER	INSTALLED NEW DETECTOR	LER 352/85-090, 092, 093
98	50-352	BWR(GE)	02/86	02/13/86	LER 352/86-012	CHANNELS	RADIATN	4 RHRSW RAD MON CHANNELS	DESIGN ERROR IN CIRCUITRY	OBSER	MODIFICATION INSTALLING PRỌPER CIRCUITRY	
99	50-352	BWR(GE)	02/86	03/02/87	LER 352/87-008	XDCR	CHEM	CHLORINE DETECTOR, HVAC	CHLORINE DETECTION PROBE (ANACON) FAILURE	OBSER	INSTALLED NEW ANACON PROBE	LER 352/86-039
100	50-352	BWR(GE)	02/86	04/09/88	LER 352/88-013	XDCR	NEUTRON	INTERMEDIATE RANGE MONITOR	INTERMEDIATE RANGE MONITOR DETECTOR FAILURE	OBSER	REPLACE DEFECTIVE IRM DETECTOR	
101	50-354	BWR(GE)	12/86	04/16/86	LER 354/86-004	CHANNEL	NEUTRON	LOCAL POWER RANGE MONITOR	NOT STATED	OBSER	RESTORE FAILED LRPM TO OPERABLE STATUS	
102	50-354	BWR(GE)	12/86	05/04/86	LER 354/86-012	SIGCOND	POWER S	CONTROL ROOM VENT RAD MON	DETECTOR HIGH-VOLTAGE SUPPLY DRIFT	OBSER	RECALIBRATED POWER SUPPLY	
103	50-354	BWR(GE)	12/86	05/08/86	LER 354/86-016	SIGCOND	POWER S	CONTROL ROOM VENT RAD MON	DETECTOR HIGH-VOLTAGE SUPPLY DRIFT	OBSER	RECALIBRATED POWER SUPPLY	LERS 354/86-012, 017
104	50-354	BWR(GE)	12/86	07/07/86	LER 354/86-036	SIGCOND	POWER S	CONTROL ROOM VENT RAD MON	HUMIDITY EFFECTS ON HV POWER SUPPLY	OBSER	REPLACE POWER SUPPLY WITH UPGRADED MODEL	LERS 354/86-012, 016
105	50-354	BWR(GE)	12/86	06/26/87	LER 354/87-027	CHANNEL	TEMP	HIGH-PRESSURE COOLANT INJ	FAILED RILEY TEMPMATIC 86 TEMPERATURE MODULE	OBSER	REPLACE SUBJECT INSTRUMENTS WITH NEWER MODEL	LERS 354/86-051; 354/87-004, 012
106	50-361	PWR(CE)	08/83	05/24/84	LER 361/84-023	CHANNEL	RADIATN	CONTROL ROOM AIRBORNE RAD	GROUND SYS NOISE ENTERING DETECTOR AND PREAMP	OBSER	REPLACE DETECTOR, PROPERLY GROUND PREAMP	LERS 361/84-002, 038, 047, 073; 362/84-030
107	50-361	PWR(CE)	08/83	08/24/84	LER 361/84-049	CHANNEL	RADIATN	CONTAINMENT PURGE ISOLATION	ELECTRICAL NOISE SPIKES	OBSER	INVESTIGATING NOISE SUPPRESSION CIRCUITS	LERS 361/84-002, 010
108	50-361	PWR(CE)	08/83	09/03/84	LER 361/84-052	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	CONSIDERING MANY POSSIBILITIES	OBSER	"CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED"	LERS 361/84-006, 012, 021, 026, 032, 037, 042
109	50-361	PWR(CE)	08/83	09/28/84	LER 361/84-055	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	CONSIDERING MANY POSSIBILITIES	OBSER	"CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED"	LERS 361/84-006, 012, 021, 026, 032, 037, 042, 052
110	50-361	PWR(CE)	08/83	01/10/85	LER 361/85-003	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	CONSIDERING MANY POSSIBILITIES	OBSER	"CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED"	LERS 361/84-006, 012, 021, 026, 032, 037, 042, 052, 055, 065
111	50-361	PWR(CE)	08/83	03/21/85	LER 361/85-010	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	CONSIDERING MANY POSSIBILITIES	OBSER	"CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED"	LERS 361/84-006, 012, 021, 026, 032, 037, 042, 052, 055, 065; 362/85-003, 019
112	50-361	PWR(CE)	08/83	05/07/85	LER 361/85-027	CHANNEL	RADIATN	CONTAINMENT PURGE ISOLATION	ELECTRICAL NOISE SPIKES	OBSER	DETAILED STUDY OF PLANT GROUND SYSTEM	LERS 361/84-002, 004, 049, 061, 071, 076; 361/85-009, 023; 362/84-010
113	50-361	PWR(CE)	08/83	05/04/85	LER 361/85-029	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	CONSIDERING MANY POSSIBILITIES	OBSER	"CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED"	LER 361/85-010
114	50-361	PWR(CE)	08/83	07/03/85	LER 361/85-032	SIGCOND	CKTCOMP	CONTROL ROOM RADIATION MON	FAILED TRANSISTORS	OBSER	REBUILT THE RADIATION MONITOR	LERS 361/85-033, 037
115	50-361	PWR(CE)	08/83	12/08/87	LER 361/87-028	CHANNEL	RADIATN	CONTAINMENT AREA RAD MON	BUILDUP OF DEPOSITS IN CABLE CONNECTOR	OBSER	PROCEDURE FOR ROUTINE CLEANING OF CONNECTOR PINS	
116	50-361	PWR(CE)	08/83	02/07/88	LER 361/88-001	CHANNEL	RADIATN	CONTROL ROOM RADIATION MON	CIRCUIT CARD SUPPORT, EDGE CONNECTOR PROBLEMS	OBSER	PROCEDURE FOR ROUTINE INSPECTION AND CLEANING	LER 361/87-028
117	50-361	PWR(CE)	08/83	02/20/88	LER 361/88-005	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	INTERMITTENT SPIKING OF AMMONIA ANALYZER	OBSER	EVALUATING TECHNOLOGY FOR MORE RELIABLE INSTR	LER 361/85-043
118	50-361	PWR(CE)	08/83	04/28/88	LER 361/88-009	CHANNEL	CHEM	TOXIC GAS ISOLATION, HVAC	FAILED CHLORINE DETECTOR ASSEMBLY	OBSER	CHLORINE DETECTOR ASSEMBLY REPLACED	LERS 361/87-023; 361/88-005
119	50-361	PWR(CE)	08/83	05/12/88	LER 361/88-011	CHANNEL	RADIATN	FUEL HANDLING ISOLATION	FAILED ELECTRONIC COMPONENTS IN POWER SUPPLY	OBSER	MODIFYING RAD MON GAS CHANNEL MODULE	LERS 361/85-034; 361/88-004
120	50-361	PWR(CE)	08/83	06/02/88	LER 361/88-012	SIGCOND	CKTCOMP	PARTICULATE/IODINE RAD MON	FAILED TRANSISTORS IN DETECTOR PREAMPLIFIER	OBSER	REPLACED FAILED TRANSISTORS	LER 361/88-013
121	50-362	PWR(CE)	01/84	03/29/85	LER 362/85-010	CHANNEL	PROTEC'N	STEAM GENERATOR LOW FLOW	FAILURE OF RELAY AND SET-POINT CARD	OBSER	REPLACED BOTH FAILED COMPONENTS	LERS 361/84-043; 362/83-111; 362/84-005, 013, 015, 023, 037, 038, 039; 362/85-001
122	50-362	PWR(CE)	01/84	03/12/87	LER 362/87-002	SIGCOND	SWITCH	CONTAINMENT AREA RAD MON	FAILURE OF KEYLOCK RANGE SWITCH (NORMAL WEAR)	OBSER	NEW SWITCHES ON ORDER FOR REPLACEMENTS	LER 362/85-005
123	50-362	PWR(CE)	01/84	08/30/87	LER 362/87-015	XDCR	RADIATN	CONTAINMENT AIRBORNE MON	PHOTOMULTIPLIER TUBE FAILURE	OBSER	REPLACED PHOTOMULTIPLIER TUBE	LERS 361/84-011, 062; 361/85-036; 362/85-011, 014; 362/86-005; 362/87-006
124	50-362	PWR(CE)	01/84	05/11/88	LER 362/88-004	SIGCOND	CKTCOMP	IODINE RAD MONITOR	INCORRECT VENDOR DRAWINGS, WRONG CAPACITOR	OBSER	CORRECT DRAWINGS, CONFORM EQUIPMENT TO DRAWINGS	
125	50-366	BWR(GE)	09/79	12/13/84	LER 366/84-035	SIGCOND	CKTCOMP	RCIC AMBIENT TEMP INSTR	BAD CIRCUIT CARD	OBSER	NOT STATED	
126	50-366	BWR(GE)	09/79	05/16/85	LER 366/85-024	CHANNEL	SPEED	RCIC ELECT'L OVERSPEED MON	"COMPONENT FAILURE"	TEST	FAILED COMPONENTS WERE REPLACED	
127	50-366	BWR(GE)	09/79	01/26/87	LER 366/87-003	XDCR	TEMP	PRIMARY CONTAINMENT ISOL'N	FAILURE OF TEMPERATURE SWITCH MONITOR	TEST	REPLACE SENSOR; REVISE PLANT PROCEDURES	LERS 321/86-016; 366/85-012
128	50-366	BWR(GE)	09/79	04/17/88	LER 366/88-011	SIGCOND	POWER S	AVG POWER RANGE MONITOR	FAILED REGULATOR CARD, PROCESS COMPUTER BOARD	TEST	REPLACE FAILED EQUIPMENT, EXTEND INVESTIGATION	LER 366/87-003
129	50-368	PWR(CE)	03/80	02/10/84	LER 368/84-005	SIGCOND	PREAMP	EXCORE NEUTRON SUBCHANNELS	FAILED PRE-AMP ASSEMBLY AND FILTER	TEST	REPLACE PREAMP ASSEMBLY AND FILTER	
130	50-368	PWR(CE)	03/80	04/21/86	LER 368/86-005	SIGCOND	CKTCOMP	EXCORE LOGARITHMIC NEUTRON	NOISE FROM FAILED POWER LEAD SURGE CAPACITORS	OBSER	REPLACE SURGE CAPACITORS	LERS 368/84-001, 007; 368/85-020; 368/86-004, 007
131	50-369	PWR(W)	12/81	01/30/84	LER 369/84-002	SIGCOND	CKTCARD	DELTA T/TAVG OF THE PCS	FAILURE OF A LEAD/LAG CARD	MAINT	REPLACE FAULTY CARD	LERS 369/81-125, 172; 369/82-018; 369/83-057, 090, 103, 104, 108; 370/83-060
132	50-369	PWR(W)	12/81	04/26/84	LER 369/84-015	SIGCOND	CKTCARD	CONTROL ROD POSITION IND	CIRCUIT FAILURE ON DETECTOR/ENCODER CARD	TEST	REPLACE FAULTY CARD	
133	50-369	PWR(W)	12/81	01/28/85	LER 369/85-004	SIGCOND	XMTR	MAIN FWP LOW SUCTION PRESS	FAILED PNEU PRESS XMTR, TURBINE CONTROL SYS	OBSER	REPAIR OR REPLACE FAILED COMPONENTS	LER 370/84-025
134	50-370	PWR(W)	03/84	02/03/84	LER 370/84-006	SIGCOND	CKTCARD	CONTROL ROD POSITION IND	FAILED CENTRAL CONTROL CARD AND DISPLAY CARD	OBSER	CIRCUIT CARDS REPLACED	
135	50-370	PWR(W)	03/84	12/11/85	LER 370/85-030	XDCR	NEUTRON	SOURCE RANGE CHANNEL	WATER IN THE DETECTOR CANNISTER	OBSER	REPLACED DETECTOR	
136	50-370	PWR(W)	03/84	07/22/86	LER 370/86-012	RECEIVER	CONTROL	DEH TURBINE CONTROL SYSTEM	COMPONENT FAILURE IN DEH TURBINE CONTROL SYS	OBSER	NOT STATED	
137	50-370	PWR(W)	03/84	03/18/87	LER 370/87-005	SIGCOND	CKTCARD	CONTROL ROD INDICATION SYS	"ELECTRONICS FAILURE OF UNKNOWN NATURE"	MAINT	RESTORED ORIGINAL CIRCUIT CARD IN SYSTEM	
138	50-370	PWR(W)	03/84	07/24/88	LER 370/88-007	CHANNEL	PROTEC'N	CHANNEL III OF RPS	FAILED SIGNAL COMPARATOR CARD	TEST	MFR WILL PERFORM FAILURE MODE ANALYSIS	LERS 369/86-002, 007; 369/88-007
139	50-373	BWR(GE)	01/84	03/27/84	LER 373/84-020	RECEIVER	RELAY	RTR BLDG VENT PROC RAD MON	FAILED RELAY BOARD IN TRIP UNIT	MAINT	REPLACED TRIP UNIT RELAY BOARD	
140	50-373	BWR(GE)	01/84	07/15/86	LER 373/86-031	RECEIVER	RELAY	HI-RADIATION MONITOR	FAILED HV BOARD IN THE INDICATOR RELAY	OBSER	BAD CHANNEL BYPASSED	
141	50-373	BWR(GE)	01/84	11/18/86	LER 373/86-040	CHANNEL	CHEM	AMMONIA DETECTION	DEGRADED WISA SAMPLE PUMP	OBSER	REPLACED WISA PUMP; SCHED BIENNIAL REPLACEMENTS	
142	50-373	BWR(GE)	01/84	12/13/87	LER 373/87-037	XDCR	TEMP	RWCU F/D INLET TEMP SWITCH	FAILED CAPACITOR IN THE TEMPERATURE SWITCH	OBSER	REPAIRED SWITCH	LER 373/87-042

Table D.2 (continued)

	A	B	C	D	E	F	G	H	I	J	K	L
143	50-373	BWR(GE)	01/84	06/09/88	LER 373/88-013	XDCR	CHEM	AMMONIA DETECTION	FAILED FRONT OPTICS LAMP	OBSER	REPLACED FRONT OPTICS LAMP	LERS 373/83-077; 373/84-017; 373/87-035
144	50-382	PWR(CE)	09/85	08/01/85	LER 382/85-036	XDCR	RAD	CR AIR INTAKE RAD MON	HOLES IN PM TUBE LIGHT SHIELDS	OBSER	REPLACED FOIL ON TWO DETECTORS	
145	50-387	BWR(GE)	06/83	09/25/86	LER 387/86-034	CHANNEL	CHEM	DRYWELL O2/H2 ANALYZER	CIRCUIT ERROR ON REPLACEMENT AMP BOARD	OBSER	REPLACED WIRE SHORT WITH PROPER RESISTOR	
146	50-388	BWR(GE)	02/85	04/05/84	LER 388/84-001	SIGCOND	PREAMP	INTERMEDIATE RANGE NEUTRON	FAILED VOLTAGE PREAMPLIFIER	OBSER	REPLACED VOLTAGE PREAMPLIFIER	
147	50-397	BWR(GE)	12/84	01/11/84	LER 397/84-002	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	EXCESSIVE ELECTRICAL NOISE INDUCED INTO SYST	OBSER	RESET AND RETURNED TO NORMAL OPERATION	
148	50-397	BWR(GE)	12/84	03/19/84	LER 397/84-025	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	EXCESSIVE ELECTRICAL NOISE INDUCED INTO SYST	OBSER	RESET AND RETURNED TO NORMAL OPERATION	LER 397/84-002
149	50-397	BWR(GE)	12/84	05/28/84	LER 397/84-053	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	CLOSURE OF AN RCIC VALVE	OBSER	RESET AND RETURNED TO NORMAL OPERATION	LERS 397/84-052, 050
150	50-397	BWR(GE)	12/84	06/17/84	LER 397/84-063	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	STARTING A SERVICE WATER PUMP	OBSER	RESET AND RETURNED TO NORMAL OPERATION	LERS 397/84-053, 067, 066, 069, 068
151	50-397	BWR(GE)	12/84	06/20/84	LER 397/84-067	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	CLOSURE OF AN RCIC VALVE	OBSER	RESET AND RETURNED TO NORMAL OPERATION	LERS 397/84-053, 063, 069, 066
152	50-397	BWR(GE)	12/84	06/28/84	LER 397/84-068	CHANNEL	RADIATN	CR AIR INTAKE RAD MON	CLOSURE OF AN RCIC VALVE	OBSER	RESET AND RETURNED TO NORMAL OPERATION	LERS 397/84-053, 067, 066, 063, 069
153	50-397	BWR(GE)	12/84	07/08/84	LER 397/84-073	CHANNEL	CHEM	CHLORINE ANALYZER	COMPONENT FAILURE IN ELECTRONICS MODULE	OBSER	RESET AND RETURNED TO NORMAL OPERATION	
154	50-397	BWR(GE)	12/84	09/16/87	LER 397/87-028	CHANNEL	TEMP	LEAK DETECTION SYSTEM	FAILED TEMPERATURE MONITORING MODULE	OBSER	REPLACED MODULE; STUDY REPLACEMENT OF SYSTEM	
155	50-400	PWR(W)	05/87	07/30/88	LER 400/88-019	XDCR	NEUTRON	SOURCE RANGE DETECTORS	ONE INTERNAL SHORT, ONE FAULTY CONNECTOR	OBSER	REPLACED BOTH DETECTORS AND ONE PREAMPLIFIER	
156	50-409	BWR(AC)		09/10/86	LER 409/86-026	CHANNEL	NEUTRON	WIDE-RANGE NUCLEAR	NOISE FROM RANGE SWITCH AND OTHER SOURCES	OBSER	REPLACED SWITCH AND INSTRUMENT DRAWER	LERS 409/84-018; 409/85-015; 406/86-015, 027
157	50-409	BWR(AC)		09/10/86	LER 409/86-027	CHANNEL	NEUTRON	SOURCE RANGE CHANNEL	UNDERSIZE PINS ON CHANNEL OUTPUT TUBE	OBSER	REPLACED TUBE, CHECKED OTHER TUBES	LER 409/86-026
158	50-409	BWR(AC)		09/19/86	LER 409/86-030	CHANNEL	NEUTRON	WIDE-RANGE NUCLEAR	SET-POINT AMBIGUITIES BETWEEN SCRAM AND ALARM	OBSER	REPLACEMENT NUCLEAR INSTRUMENTATION ORDERED	LER 409/85-018
159	50-409	BWR(AC)		07/12/87	LER 409/87-005	CHANNEL	RADIATN	CB AND AIR EJECTOR CHANNELS	FAILED TRANSISTOR AFFECTED COMMON POWER SUPP	OBSER	REPLACED CHASSIS	LERS 409/84-017, 020; 409/86-010, 025, 034, 037; 409/87-003
160	50-410	BWR(GE)	03/88	11/09/86	LER 410/86-004	CHANNEL	NEUTRON	LOCAL POWER RANGE MONITOR	FAULTY L&M CIRCUIT CARD, INFANT MORTALITY	OBSER	REPLACED LPRM CIRCUIT CARD	
161	50-412	PWR(W)	11/87	08/25/87	LER 412/87-019	XDCR	SPEED	TURBINE OVERSPEED	FAULTY TURBINE OVERSPEED TRANSDUCER	MAINT	REPLACED OVERSPEED XDCR, UNDERFREQUENCY RELAYS	
162	50-413	PWR(W)	06/85	09/24/85	LER 413/85-057	RECEIVER	CONTROL	CONTROL ROD POSITION	FAILED PDC CONTROLLER 2000	OBSER	NOT STATED	
163	50-414	PWR(W)	08/85	05/08/87	LER 414/87-019	RECEIVER	CONTROL	PROCESS CONTROL SYSTEM	PRINTED CIRCUIT CARD MALFUNCTION	OBSER	REPLACED PRINTED CIRCUIT CARD	LERS 414/86-025, 053
164	50-414	PWR(W)	08/85	03/09/88	LER 414/88-012	RECEIVER	CONTROL	S/G CF CONTROL VALVE	TWO DEFECTIVE CIRCUIT CARDS	OBSER	REPLACED PRINTED CIRCUIT CARDS	LERS 413/86-025; 413/88-015; 414/87-019, 027
165	50-416	BWR(GE)	07/85	11/07/85	LER 416/85-043	CHANNEL	CHEM	HYDROGEN ANALYZERS	HEAT DEGRADED TERMINAL STRIPS BRITTLE, BROKEN	MAINT	LOWERED THERMOSTAT, REPLACED TERMINAL STRIPS	
166	50-416	BWR(GE)	07/85	01/11/88	LER 416/88-005	RECEIVER	CONTROL	RTR PRESS CONTROLLER SET PT	SPECIFIC CAUSE COULD NOT BE IDENTIFIED	OBSER	REPLACED MOTOR DRIVEN POT, SET-POINT PUSHBUTTONS	LER 416/88-002
167	50-423	PWR(W)	04/86	01/18/88	LER 423/88-004	XDCR	CHEM	CONTR BLDG CHLORINE SENSOR	SPECIFIC CAUSE COULD NOT BE IDENTIFIED	OBSER	REPLACED SENSING ELEMENT	LERS 423/86-037, 040; 423/87-028
168	50-424	PWR(W)	06/87	02/13/87	LER 424/87-004	RECEIVER	COMPUTE	CONTAINMENT RAD MONITOR	FAULTY CKT BOARD IN DATA PROCESSING MODULE	OBSER	REPLACED CIRCUIT BOARD	
169	50-424	PWR(W)	06/87	05/01/87	LER 424/87-021	RECEIVER	COMPUTE	CONTROL ROOM RAD MONITOR	FAILURE OF DATA PROCESSING MODULE	OBSER	REPLACED DPM, TO VENDOR FOR TEST AND ANALYSIS	LERS 424/87-005, 019
170	50-424	PWR(W)	06/87	09/21/87	LER 424/87-058	XDCR	RADIATN	CONTROL ROOM RAD MONITOR	FAULTY SENSING TUBE, BAD SOFTWARE DESIGN	OBSER	REPLACED TUBE, PROPOSED SOFTWARE CHANGE	LERS 424/87-005, 021, 073
171	50-424	PWR(W)	06/87	11/17/87	LER 424/87-068	XDCR	RADIATN	CONTROL ROOM RAD MONITOR	FAULTY SENSING TUBE, BAD SOFTWARE DESIGN	OBSER	REPLACED TUBE, PROPOSED SOFTWARE CHANGE	LERS 424/87-058, 065
172	50-424	PWR(W)	06/87	12/21/87	LER 424/87-073	XDCR	RADIATN	CONTROL ROOM RAD MONITOR	FAULTY SENSING TUBE, BAD SOFTWARE DESIGN	OBSER	REPLACED TUBE, PROPOSED SOFTWARE CHANGE	LERS 424/87-058, 065, 067
173	50-440	BWR(GE)	11/87	05/30/86	LER 440/86-019	XDCR	TACH	DIESEL GEN CTRL TACHOMETER	INTERMITTENT SIGNALS DUE TO SYSTEM NOISE	OBSER	EXTENSIVE REVIEW, WILL UPGRADE TACHOMETERS	LERS 440/86-031, 042
174	50-443	PWR(W)		01/08/87	LER 443/87-001	XDCR	RAD	CONTROL ROOM RAD MONITOR	FAULTY GEIGER-MUELLER TUBE	OBSER	REPLACED MONITOR ASSEMBLY	LER 443/86-003
175	50-454	PWR(W)	09/85	12/25/84	LER 454/84-040	RECEIVER	INDICAT	CONTROL ROD POSITION DEMAND	STEP COUNTER STICKING	TEST	CLEAN AND LUBE ALL STEP COUNTERS, REPLACE ONE	
176	50-454	PWR(W)	09/85	02/18/85	LER 454/85-022	XDCR	NEUTRON	POWER RANGE EXCORE NEUTRON	FAILURE OF DETECTOR	OBSER	REPLACED DETECTOR	
177	50-454	PWR(W)	09/85	12/11/85	LER 454/85-099	RECEIVER	COMPUTE	CONTROL ROOM RAD MONITOR	COMPON FAILURE ON MICROPROCESSOR MOTHERBOARD	OBSER	REPLACED MICROPROCESSOR MOTHERBOARD	LERS 454/85-002, 088
178	50-455	PWR(W)	09/85	07/27/88	LER 455/88-010	CHANNEL	RADIATN	SERV WATER OUTLET RAD MON	NOT IDENTIFIED	OBSER	CLEANED INLET STRAINER, REPLACED MICRO CKT BRD	LERS 454/84-014; 454/85-082
179	50-456	PWR(W)		07/16/87	LER 456/87-038	SIGCOND	POWER S	CONTROL ROOM RAD MON	TRANSISTOR FAILURE IN HV POWER SUPPLY	OBSER	REPLACED POWER SUPPLY	
180	50-456	PWR(W)		09/27/87	LER 456/87-055	XDCR	RADIATN	CONTAINMENT VENT RAD MON	DEFECTIVE DETECTOR	OBSER	REPLACED DETECTOR TUBE	
181	50-461	BWR(GE)	11/87	01/06/88	LER 461/88-002	XDCR	RADIATN	CONTAINMENT PURGE RAD MON	RANDOM DETECTOR TUBE FAILURE	OBSER	REPLACED DETECTOR TUBE	
182	50-461	BWR(GE)	11/87	07/12/88	LER 461/88-019	XDCR	LEVEL	CONDENSER PIT LEVEL SWITCH	STRESS CORROSION FAILURE OF SWITCH ARM	OBSER	REPLACED SWITCH	
183	50-482	PWR(W)	09/85	02/20/86	LER 482/86-006	CHANNEL	RADIATN	CONTROL ROOM RAD MON	FAULTY POWER ISOLATION BOARD	OBSER	REPLACED POWER ISOLATION BOARD	
184	50-482	PWR(W)	09/85	12/25/86	LER 482/86-071	XDCR	CHEM	CHLORINE DETECTOR, HVAC	OPTICS CIRCUIT BULB FAILURE	OBSER	REPLACED BULB (MDA SCIENTIFIC, INC., MOD. 7040 FAN)	
185	50-482	PWR(W)	09/85	05/06/87	LER 482/87-019	CHANNEL	RADIATN	CONTROL ROOM RAD MON	BROKEN SHIELD ON CO-AX CONNECTOR	MAINT	REPLACED CABLE CONNECTOR	
186	50-482	PWR(W)	09/85	11/23/87	LER 482/87-053	XDCR	CHEM	CHLORINE DETECTOR, HVAC	PAPER TAPE TRANSPORT FAILURE	OBSER	REPLACED PAPER TAPE	LERS 482/85-011, 014, 033, 052, 056, 061, 062, 081, 085; 482/86-001, 002, 015, 022, 041, 063, 065,071;
187	50-482	PWR(W)	09/85	11/27/87	LER 482/87-064	CHANNEL	RADIATN	CONTROL ROOM RAD MON	MOISTURE INDUCED CORROSION OF CONNECTOR	OBSER	CLEANED CONNECTOR, APPLIED HEAT SHRINK SEAL	LER 482/87-019
188	50-482	PWR(W)	09/85	07/28/88	LER 482/88-012	XDCR	CHEM	CHLORINE DETECTOR, HVAC	PHOTOCELL FAILURE	OBSER	REPLACED PHOTOCELL ASSEMBLY	LERS 482/85-011, 014, 033, 052, 056, 061, 062, 081, 085; 482/86-001, 002, 015, 022, 041, 063, 065, 071;
189	50-482	PWR(W)	09/85	06/02/88	LER 482/88-013	XDCR	CHEM	CHLORINE DETECTOR, HVAC	PAPER TAPE TRANSPORT FAILURE	OBSER	REPLACED PAPER TAPE	LERS 482/85-011, 014, 033, 052, 056, 061, 062, 081, 085; 482/86-001, 002, 015, 022, 041, 063, 065, 071;
190	50-483	PWR(W)	12/84	08/13/84	LER 483/84-028	CHANNEL	PRESSURE	REACTOR COOLANT SYSTEM	DESIGN ERRORS, FAILED INDICATOR LAMP	TEST	REPLACED LAMP; DESIGN CHANGE IMPLEMENTED	
191	50-483	PWR(W)	12/84	12/09/85	LER 483/85-052	RECEIVER	CONTROL	MAIN STEAM AND FEEDWATER	CKT CARD FAILURE IN ISOLATION CONTROL PANEL	OBSER	CKT CARD REPLACED & SENT TO MFR FOR ANALYSIS	LER 483-85-051
192	50-483	PWR(W)	12/84	06/30/88	LER 483/88-008	RECEIVER	COMPUTE	RADWASTE BLDG VENT RAD MON	RAM FAILURE ON MICROPROCESSOR CIRCUIT BOARD	TEST	REPLACED RAM INTEGRATED CIRCUITS	
193	50-498	PWR(W)		09/06/87	LER 498/87-007	CHANNEL	CHEM	TOXIC GAS MONITOR, HVAC	DEFECTIVE FLOW SWITCH	OBSER	REPLACED DEFECTIVE FLOW SWITCH	
194	50-498	PWR(W)		09/26/87	LER 498/87-010	CHANNEL	RADIATN	FUEL HANDLING BLDG RAD MON	UNABLE TO DETERMINE	OBSER	REPLACED I/O AND PREAMP PC BOARDS, INSPECTED	
195	50-498	PWR(W)		10/17/87	LER 498/87-011	CHANNEL	CHEM	TOXIC GAS MONITOR, HVAC	PRINTED CIRCUIT BOARD FAILURE	OBSER	REPLACED PRINTED CIRCUIT BOARD	
196	50-498	PWR(W)		11/28/87	LER 498/87-020	CHANNEL	CHEM	TOXIC GAS MONITOR, HVAC	COMPUTER CHIP FAILURE	OBSER	REPLACED COMPUTER CHIP; SYSTEM EVALUATION	LER 498/87-011
197	50-498	PWR(W)		01/10/88	LER 498/88-004	CHANNEL	CHEM	TOXIC GAS MONITOR, HVAC	LOOSE CONNECTOR OR INTEGRATED CIRCUIT CHIP	OBSER	INSPECT AND CLEAN RAM BOARDS AND CAGES	LER 498/87-022
198	50-528	PWR(CE)	02/86	01/19/85	LER 528/85-003	RECEIVER	COMPUTE	CR VENT PROCESS RAD MON	BAD CPU BOARD IN REMOTE INDICATING CONTROLLER	OBSER	REPLACED CPU BOARD	
199	50-528	PWR(CE)	02/86	12/29/85	LER 528/85-097	RECEIVER	COMPUTE	CONTAINMENT PURGE RAD MON	COMPONENT FAILURE ON CPU CARD	OBSER	REPLACED KAMAN SCIENCES CORP. MOD. 450358-004 CPU	
200	50-529	PWR(CE)	09/86	09/22/86	LER 529/86-046	CHANNEL	SAFETY	SAFETY INJECTION ACTUATION	FAILURE OF MANUAL ACTUATION HANDSWITCH	OBSER	REPLACED FAULTY HANDSWITCH	
201	50-213	PWR(W)	08/67	LONGTIME	IR 50-213/88-11	INDICAT	POSITION	ROD POSITION INDICATION	TEMP-INDUCED ERRORS IN ANALOG SYSTEM OUTPUT	OBSER	ESTABLISHED RPI TASK FORCE TO EVALUATE PROBLEM	NRC/IR 50-213/88-08
202	50-213	PWR(W)	08/67	06/09/88	IR 50-213/88-11	CHANNEL	FLUX DIS	POWER DISTR'N AXIAL OFFSET	WRONG RESISTOR, OVERVOLTAGE POWER	OBSER	REPLACE SYSTEM IN NEXT REFUELING OUTAGE	RIDS 213/JUN88
203	50-213	PWR(W)	08/67	03/19/88	SALP REPT/87-99	CHANNEL	NEUTRON	INTERMEDIATE RANGE CHANNEL	EQUIPMENT AGE, ELECTRICAL NOISE	OBSER	NOT STATED	
204	50-213	PWR(W)	08/67	SUMMER88	IR 50-213/88-23	OTHER	STUDY	EXCESSIVE CONTAINMENT TEMP	HIGH SUMMER TEMPERATURES	TEST	WEIGHTED AVERAGE TEMPERATURE FOR TWO MONTHS	
205	50-213	PWR(W)	08/67	08/13/79	IE/IN 80-12	CHANNEL	PRESSURE	PRESSURIZER PRESSURE CONTR	LIGHT SOURCE FAILURE IN BISTABLE	OBSER	REPLACE BISTABLE WITH SOLID STATE UNIT	NONE
206	50-259	BWR(GE)	08/74	07/20/88	LER 259/88-022	SIGCOND	CKTCOMP	REFUEL ZONE RAD MONITOR	COIL SHORTED DUE TO AGE AND INTERNAL HEATING	OBSER	REPLACED 24-V RELAYS WITH 36-V RELAYS	LISTS SEVEN PREVIOUS SIMILAR EVENTS
207	50-269	PWR(BW)	07/73	04/11/85	LER 269/85-006	RECEIVER	CONTROL	SECONDARY PRESSURE CONTROL	KNOWN UNSTABLE PRESSURE CONTROL SYSTEM	OBSER	RESTARTED REACTOR	LER 269/85-005
208	50-269	PWR(BW)	07/73	04/25/85	LER 269/85-007	RECEIVER	CONTROL	MAIN FEEDWATER FLOW	INTEGRATED CONTROL SYSTEM ACTION	OBSER	RESTARTED REACTOR	NONE
209	50-269	PWR(BW)	07/73	01/02/89	MR/RII/EN14387	SIGCOND	CKTCOMP	MAIN FEEDWATER CONTROL	BAD CKT CARD IN INTEGRATED CONTROL SYSTEM	TEST	REPLACED CIRCUIT CARD	NONE
210	50-269	PWR(BW)	07/73	01/02/89	MR/RII/EN14387	SIGCOND	CKTCOMP	EMERGENCY FEEDWATER CONTROL	BAD CARD IN FDW VALVE CONTROL CIRCUIT	TEST	REPAIRED CIRCUIT	NONE
211	50-271	BWR(GE)	11/72	06/24/88	IR 50-271/88-08	XDCR	VIBRAT'N	TURBINE BEARING, RS(100%)	FAILED SENSOR PROBE	OBSER	REPLACED PROBE, EXAMINED BEARING	NONE
212	50-272	PWR(W)	06/77	09/07/88	IR 50-272/88-17	CHANNEL	FLOW	STEAM GENERATOR STEAM FLOW	DRIFT FROM UNKNOWN CAUSE ON 5 CHANNELS	TEST	UNDER CONTINUING INVESTIGATION	LER 272/88-017, IR 50-272/88-19
213	50-275	PWR(W)	05/85	05/31/89	IR 50-275/88-11	SIGCOND	CKTCOMP	WESTINGHOUSE ARD RELAYS	AGE DETERIORATED SAND-BASED POTTING COMPOUND	OBSER	INVESTIGATING	IE/IN 88-88, IE/IN 89-97, IR 50-275/88-15, NRC DCI-87 EM-N121

Table D.2 (continued)

	A	B	C	D	E	F	G	H	I	J	K	L
214	50-275	PWR(W)	05/85	10/06/89	NRC/DER/EN16786	XDCR	FLOW	DP ACTUATION, SI, RS(100%)	SHARED INSTRUMENT TAPS	OBSER	NOT STATED	NONE
215	50-275	PWR(W)	05/85	06/01/87	IR 50-275/88-15	XDCR	PNEUMAT	ALL PNEUMATIC INSTRUMENT'N	WATER LEAKAGE INTO INSTRUMENT AIR SYSTEM	OBSER	REPLACED ALL RELAYS WITH SIGNS OF DEGRADATION	NONE
216	50-275	PWR(W)	05/85	12/13/87	IR 50-275/88-16	SIGCOND	CKTCOMP	SOURCE RANGE CHANNEL	FAILED CAPACITOR IN HV POWER SUPPLY	OBSER	MONITOR SIMILAR POWER SUPPLIES, PREVENTIVE MAIN	LERS 275/87-024-L0&L1
217	50-275	PWR(W)	05/85	08/30/88	RIDS 275/AUG88	XDCR	SPEED	MAIN FEEDWATER PUMP, RS	SPEED PROBE FAILURE, CKT FAILURE TO DETECT	OBSER	PROBE REPLACED AND SENT TO VENDOR FOR ANALYSIS	LER 275/88-025
218	50-277	BWR(GE)	07/74	05/05/89	NRC/DER/EN15541	RECEIVER	CONTROL	HPCI SYSTEM FLOW CONTROL'R	"ELECTRICAL MALFUNCTION"	TEST	FLOW CONTROLLER REPAIRED	MR/RI/05/08/89
219	50-277	BWR(GE)	07/74	05/19/89	NRC/DER/EN15652	CHANNEL	LEVEL	REACTOR WATER LEVEL CONTROL	FAILURE OF A CONTACT TO CLOSE	OBSER	REPAIR OR REPLACE FAULTY CONTACT	MR/RI/05/22/89
220	50-282	PWR(W)	12/73	07/26/88	IR 50-282/88-12	RECEIVER	CONTROL	FLUX TILT CONTROLLERS	FOXBORO CONTROLLERS INCOMPATIBLE WITH NIS	OBSER	MODIFIED FOXBORO CONTROLLERS FOR COMPATIBILITY	LER 282/88-004-LL, IR 50-282/88-19, RIDS 282/JUL88
221	50-285	PWR(CE)	06/74	06/16/89	NRC/DER/EN15890	RECEIVER	CONTROL	TURBINE-DRIVEN AFW PUMP	MALFUNCT CKT CARD IN STEAM SUPPLY VALVE LOGIC	TEST	REPLACED FAULTY CARD	NONE
222	50-285	PWR(CE)	06/74	05/09/88	IR 50-285/88-19	CHANNEL	FLOW	RTR COOLANT LOW FLOW TRIP	FAILED POWER SUPPLY TO LOW-FLOW TRIP UNIT	TEST	REPLACED POWER SUPPLY	LER 285/88-013
223	50-285	PWR(CE)	06/74	07/06/87	IR 50-285/88-27	SYSTEM	INST AIR	ALL PNEUMATIC INSTRUMENT'N	WATER LEAKAGE INTO INSTRUMENT AIR SYSTEM	OBSER	EXTENSIVE INVESTIGATION, REVIEW AND COMMITMENTS	LER 285/88-033; IR 50-285/87-27, 87-30, 88-15, 88-23; FC-1047-88; NNE OE-2262
224	50-289	PWR(BW)	12/78	12/22/87	IR 50-289/87-24	SIGCOND	POWER S	OTSG LEVEL TRANSMITTERS	FAILED TRANSISTOR/RESISTOR IN DC CHOPPER CKT	OBSER	REPLACED POWER SUPPLY MODULE, CONTINUE STUDY	NONE
225	50-289	PWR(BW)	12/78	05/19/88	IR 50-289/88-05	CHANNEL	D/PRESS	MFW PUMPS, OTSG LEVELS	D/PRESS. INSTRUMENT (SP-11A-DPT-2) FAILED LOW	OBSER	EVALUATION OF THE EVENT	NONE
226	50-289	PWR(BW)	12/78	06/??/87	SIR 50-289/88-07	XDCR	RADIAT'N	TURBINE BLDG SUMP MONITOR	TEMP COEFF OF PM TUBE VOLTAGE DIVIDER	OBSER	MFR REPLACED RESISTORS WITH THERMISTORS	NONE
227	50-289	PWR(BW)	12/78	10/11/88	SIR 50-289/88-24	RECEIVER	CONTROL	INTEGRATED CONTROL SYSTEM	FAULTY ANALOG MEMORY MODULE	OBSER	REPLACED ANALOG MEMORY MODULE	NONE
228	50-293	BWR(GE)	12/72	04/26/88	SIR 50-293/88-19	CHANNEL	LOGIC	REACTOR WATER CLEANUP	AGE-RELATED FAILURES OF GE CR 120-A RELAYS	OBSER	REPLACING ALL SAFETY RELAYS OF THIS TYPE	LER 293/88-014, IR 50-293/88-24, 88-34
229	50-293	BWR(GE)	12/72	02/02/88	SIR 50-293/88-34	CHANNEL	LOGIC	CONTAINMENT ISOLATION	FAILURE OF COIL IN LOGIC RELAY	OBSER	NOT STATED	LER 293/88-005, IR 50-293/88-07
230	50-293	BWR(GE)	12/72	12/31/88	RIDS 293/DEC88	XDCR	NEUTRON	THREE IRM DETECTORS	TO BE DETERMINED	OBSER	REPLACED THREE DETECTORS	NONE
231	50-293	BWR(GE)	12/72	06/09/89	RIDS 293/JUN89	XDCR	TEMP	RCIC, TEMPERATURE SWITCH	INADEQUATE CONTACT PRESSURE BY TERMINAL BLOCK	OBSER	REPLACED TERMINAL BLOCK	LER 293/89-019, NRC/DER/EN15823
232	50-293	BWR(GE)	12/72	08/03/89	MR/RI/EN 16171	XDCR	NEUTRON	LOCAL POWER RANGE MONITOR	UNSTABLE U-COATING, FLAKING IN NA-300 TUBES	OBSER	WORKING WITH GE ON CORRECTIVE MEASURES	SIMILAR PROBLEMS AT, AT LEAST, FIVE OTHER PLANTS
233	50-295	PWR(W)	12/73	07/13/88	IR 50-295/88-13	SIGCOND	SPECFN	STEAM/FEED FLOW MISMATCH	SQUARE-ROOT EXTRACTOR FAILURE	OBSER	NOT STATED	RIDS 295/JUL88, LER 295/88-013, SIR 50-295/88-16
234	50-295	PWR(W)	12/73	07/06/89	MR/RIII/EN16032	RECEIVER	ANNUNC	NSSS ANNUNCIATORS	CONTROL SYSTEM CIRCUIT CARD FAILURE	OBSER	REPLACED FAILED CIRCUIT CARD	NONE
235	50-298	BWR(GE)	07/74	06/04/88	IR 50-298/88-33	RECEIVER	CONTROL	HPCI PUMP GOVERNOR (EGM)	FAILED TRANSISTOR IN EGM SPEED CIRCUIT	TEST	EGM RET TO MFR (WOODWARD GOVERNOR) FOR REPAIR	LER 298/88-022
236	50-298	BWR(GE)	07/74	09/16/88	IR 50-298/88-28	RECEIVER	RELAY	RWC, RBV, RHRS ISOLATIONS	AGE-RELATED FAILURES OF GE CR 120-A RELAYS	OBSER	REPLACE ALL GE CR 120-A RELAYS	NONE
237	50-298	BWR(GE)	07/74	09/23/88	IR 50-298/88-28	SIGCOND	POWER S	REACTOR BLDG RAD MONITOR	FAILED ZENER DIODE IN POWER SUPPLY	OBSER	REPAIRED POWER SUPPLY	NONE
238	50-293	BWR(GE)	12/72	06/06/87	IR 50-293/88-24	CHANNEL	LOGIC	345 SAFETY-RELATED RELAYS	ARMATURE BINDING AND EXCESS PICKUP VOLTAGE	OBSER	MODIFIED/REPLACED GE TYPE HFA RELAYS	GE SAL 188.1, NED-545, RFO NO. 7, NRC/IR 87-46

NUREG/CR-5700
ORNL/TM-11806
Dist. Category RV

INTERNAL DISTRIBUTION

1. J. L. Anderson
2. S. J. Ball
3. R. E. Battle
4. D. A. Casada
5. N. E. Clapp, Jr.
6. R. L. Clark
7. D. F. Cox
8. B. G. Eads
9. E. C. Fox
10. D. N. Fry

11–19. A. C. Gehl

20. R. H. Greene
21. E. W. Hagen
22. H. D. Haynes

23–25. R. A. Kisner

26. K. Korsah

27–31. R. C. Kryter

32. K. H. Luk
33. J. C. Moyers
34. G. A. Murphy
35. C. E. Pugh
36. R. E. Uhrig
37. T. L. Wilson
38. A. Zucker
39. J. B. Ball, Advisor
40. B. Chexal, Advisor
41. T. B. Sheridan, Advisor
42. R. M. Taylor, Advisor

43–44. Central Research Library

45. Y-12 Technical Reference Section

46–47. Laboratory Records Department

48. Laboratory Records ORNL-RC
49. ORNL Patent Section
50. I&C Division Publications Office

EXTERNAL DISTRIBUTION

51. Assistant Manager for Energy Research and Development, U.S. Department of Energy, Oak Ridge Operations Office, P.O. Box 2001, Oak Ridge, TN 37831

52–53. Office of Scientific and Technical Information, U.S. Department of Energy, P.O. Box 62, Oak Ridge, Tennessee 37831

54. D. M. Eissenberg, RD 1, Box 133, Cambridge, NY 12816
55. G. Sliter, Electric Power Research Institute, P.O. Box 10412, Palo Alto, CA 94303
56. J. W. Tills, Institute for Nuclear Power Operations, 1100 Circle 75 Parkway, Atlanta, GA 30339-3064
57. M. Subudhi, Brookhaven National Laboratory, Bldg. 130, Upton, NY 11973
58. A. B. Johnson, Battelle-PNL, MS P8-10, P.O. Box 999, Richland, WA 99352
59. J. P. Vora, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Electrical and Mechanical Engineering Branch, 5650 Nicholson Lane, Rockville, MD 20852
60. E. J. Brown, U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, Reactor Operations Analysis Branch, MS 2104, Maryland National Bank Building, 7735 Old Georgetown Road, Bethesda, MD 20814
61. C. Michelson, Advisory Committee on Reactor Safeguards, 20 Argonne Plaza, Suite 365, Oak Ridge, TN 37830

62. M. Vagins, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Chief, Electrical and Mechanical Engineering Branch, 5650 Nicholson Lane, Rockville, MD 20852
63. W. S. Farmer, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, Electrical and Mechanical Engineering Branch, 5650 Nicholson Lane, Rockville, MD 20852
64. M. J. Jacobus, Sandia National Laboratories, P.O. Box 5800, Division 6447, Albuquerque, NM 87185
65. H. L. Magleby, Idaho National Engineering Laboratory, MS 2406, P.O. Box 1625, Idaho Falls, ID 83415
66. Scott Newberry, U.S. Nuclear Regulatory Commission, NRR/SICB, MS 7E12, 1 White Flint North, 11555 Rockville Pike, Rockville, MD 20852
67. Jerry L. Mauck, U.S. Nuclear Regulatory Commission, NRR/SICB, MS 7E12, 1 White Flint North, 11555 Rockville Pike, Rockville, MD 20852
68. Matthew Chiramal, U.S. Nuclear Regulatory Commission, NRR/SICB, MS 7E12, 1 White Flint North, 11555 Rockville Pike, Rockville, MD 20852
69. Joseph P. Joyce, U.S. Nuclear Regulatory Commission, NRR/SICB, MS 7E12, 1 White Flint North, 11555 Rockville Pike, Rockville, MD 20852
70. A. C. Thadani, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of System Technology, MS8 E2, 1 White Flint North, 11555 Rockville Pike, Rockville, MD 20852

Given distribution as shown in NRC category RV (10–NTIS)

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/CR-5700
ORNL/TM-11806

2. TITLE AND SUBTITLE
Aging Assessment of Reactor Instrumentation and Protection System Components
Aging-Related Operating Experiences

3. DATE REPORT PUBLISHED

MONTH	YEAR
July	1992

4. FIN OR GRANT NUMBER
B0828

5. AUTHOR(S)
A. C. Gehl, E. W. Hagen

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
Oak Ridge National Laboratory
Oak Ridge, TN 37831

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Engineering
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

A study of the aging-related operating experiences throughout a five year period (1984-1988) of six generic instrumentation modules (indicators, sensors, controllers, transmitters, annunciators, and recorders) was performed as a part of the USNRC Nuclear Plant Aging Research Program. The effects of aging from operational and environmental stressors were characterized from results depicted in Licensee Event Reports (LERs). The data are graphically displayed as frequency of events per plant year for operating plant ages from 1 to 28 years to determine aging-related failure trend patterns. Of the six modules studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures. Infant mortality appears to be the dominant failure mode for most I&C module categories. Of the LERs issued during 1984-1988 which dealt with malfunctions of the six instrumentation and control modules studied, 28% were found to be aging-related (other studies show a range of 25-50%).

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
aging, instrumentation, indicator, sensor, controller, recorder, annunciator, transmitter, stressors, drift, obsolescence

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page) unclassified
(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)