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NRC Research Program on Plant Aging: Listing and Summaries of Reports Issued Through September 1993

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

J. P. Vora

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ABSTRACT

The U.S. Nuclear Regulatory Commission is conducting the Nuclear Plant Aging Research (NPAR) Program. This is a comprehensive hardware-oriented engineering research program focused on understanding the aging mechanisms of components and systems in nuclear power plants. The NPAR program also focuses on methods for simulating and monitoring the aging-related degradation of these components and systems. In addition, it provides recommendations for effective maintenance to manage aging and for implementation of the research results in the regulatory process.

This document contains a listing and index of reports generated in the NPAR Program that were issued through September 1993 and summaries of those reports. Each summary describes the elements of the research covered in the report and outlines the significant results. For the convenience of the user, the reports are indexed by personal author, corporate author, and subject.

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PREFACE

The Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC) is conducting a hardware-oriented engineering research program dealing with the aging of nuclear power plant components and systems. This program is described in NUREG-1144, Rev. 2, "Nuclear Plant Aging Research (NPAR) Program Plan—Status and Accomplishments," published in June 1991.

Significant progress has been made in defining aging degradation mechanisms and in developing effective monitoring and surveillance methods for many of the components and systems identified in NUREG-1144, Rev. 2. These components and systems include motor-operated valves, check valves, solenoid-operated valves, electric motors, emergency diesel generators, chargers and inverters, circuit breakers and relays, batteries, auxiliary feedwater pumps, and reactor protection systems. Progress has also been made in developing models and approaches to evaluate the relative impact of aging on risk. The phase I research for evaluating system-level aging effects based on operating experience and risk evaluation of the aging phenomena has been completed.

Significant accomplishments have included identifying major technical safety issues and defining the risk significance of major light water reactor components and structures. The Nuclear Plant Aging Research program continues to provide the technical bases and regulatory guidelines for the license renewal rulemaking, which is considered a top priority for the NRC.

This document contains summaries of NRC-sponsored reports that were generated in the NPAR Program. Each summary describes the objectives of the research, identifies the contractor and the authors involved, and outlines significant research results. If the readers of this document need additional information on a particular report and the findings discussed therein, they are encouraged to contact the authors of that report directly.

This report is updated annually to incorporate summaries of new NPAR reports. Comments are welcome and will be considered in developing subsequent revisions of this document.

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ACRONYMS AND ABBREVIATIONS

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable (radiation level)
ALEAP	Aging and Life Extension Assessment Program
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AUXFP	auxiliary feedwater pump
BNL	Brookhaven National Laboratory
BOP	balance of plant
B&W	Babcock & Wilcox Co.
BWR	boiling water reactor
CCW	component cooling water
CE	Combustion Engineering
CFR	Code of Federal Regulations
CRD	control rod drive
CVN	Charpy V-notch
DBE	design basis event
ECCAD	electrical circuit characterization and diagnostic system
ECCS	emergency core cooling system
EDG	emergency diesel generator
EPA	electrical penetration assembly
EPRI	Electric Power Research Institute
ESFAS	engineered safety feature actuation system
FSAR	Final Safety Analysis Report
GFRS	generic flaw response spectra
GI	generic issue
GSI	generic safety issue
HDR	Heissdampfreaktor, a decommissioned German reactor
HPCI	high-pressure coolant injection
HPIS	high-pressure injection system
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IPRDS	In-Plant Reliability Data Systems
ISA	Instrument Society of America
ISCM	inspection, surveillance, and condition monitoring
ISI	inservice inspection
KWU	Kraftwerk Union Aktiengesellschaft, a German company
LER	Licensee Event Report
LOCA	loss-of-coolant accident
LWR	light-water-reactor
MCC	motor control center
MCSA	motor current signature analysis
MLE	maximum likelihood estimate
MOV	motor-operated valve
MOVATS	motor-operated valve analysis and test system
MSLB	main steam line break
NOAC	Nuclear Operations Analysis Center
NPE	Nuclear Power Experience
NPRDS	Nuclear Plant Reliability Data System
NRC	U.S. Nuclear Regulatory Commission
NPAR	Nuclear Plant Aging Research
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
PORV	power-operated relief valve
PRA	probablistic risk assessment
PWR	pressurized water reactor
RCP	reactor coolant pump
RHR	residual heat removal
RTD	resistance temperature detector
RTS	reactor trip system
RWCU	reactor water cleanup
SCR	silicon controlled rectifier
SOV	solenoid-operated valve
SQUG	Seismic Qualification Utilities Group
SSE	safe shutdown earthquake
TDR	time-domain reflectometry
TIRGALEX	Technical Integration Review Group for Aging and Life Extension
TI	temporary instruction
UT	ultrasonic testing

INTRODUCTION

This document is a listing and index of reports related to the Nuclear Plant Aging Research (NPAR) Program issued through September 1993. The first listing is in alphanumeric order by report number and includes a summary of each report. Three indexes are provided to aid the user in retrieving a specific report: Personal Author Index, Corporate Author Index, and Subject Index. Finally, there is a listing in chronological order by date of publication.

Most of the reports contain a description of the components or systems being examined and identify the principal stressors leading to aging. They frequently contain an analysis and statistical assessment of failure data obtained from Licensee Event Reports and other sources of component failure data for operating nuclear power plants. Current surveillance and monitoring practices are also reviewed and, when identifiable, recommendations are made for improvements.

The information contained in the reports should be of interest to those assessing the aging and reliability of nuclear power plant components, including researchers and designers as well as maintenance and operations personnel.

Most of the documents cited in this report are available from one of the following sources:

1. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082.

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

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MAIN CITATIONS AND SUMMARIES

The reports listed in this compilation are arranged alphanumerically by report number, with unnumbered reports preceding the numbered reports. The bibliographic information is followed by a summary of each report.

UNNUMBERED REPORTS

Letter Report, M. Subudhi, "Review of Aging-Seismic Correlation Studies on Nuclear Plant Equipment," Brookhaven National Laboratory, January 1985.

During the last decade, the issue relating to aging-seismic correlation of nuclear-grade equipment and their components has received special attention by both the NRC and the utility industry with the aim of preventing catastrophic failures of aged nuclear power plant components during a seismic event. This report summarizes the work performed by the Seismic Qualification Utilities Group (SQUG) based on real earthquake data, by NUTECH for Sandia National Laboratories, and by EPRI at Wyle. Based on the above, an outline of the work to be carried out at BNL under the NPAR scope relating to identifying the aged components sensitive to seismic loadings is provided.

EQE, Inc., sponsored by the Seismic Qualification Utilities Group has gathered a comparative data base on the performance of equipment in five fossil-fueled plants consisting of 24 units and a high-voltage DC-to-AC converter station. These plants have experienced four damaging California earthquakes of Richter Magnitudes 5.1 to 6.6. Peak horizontal ground accelerations (PGA) of these earthquakes ranged between 0.2 g and 0.5 g. The actual earthquake-induced effects on equipment were compared with equipment qualification data from three nuclear plants.

The objective of the pilot program was to determine the feasibility of establishing criteria for assessing the seismic adequacy of equipment in nuclear power plants based on evaluation and application of data to be acquired on the characteristics and seismic performance of equipment in nonnuclear power facilities that have been subjected to strong-motion earthquakes. Application of the criteria would provide a valid basis for assessing the need for subsequent qualification efforts in the nuclear industry and for defining the extent of the effort.

Letter Report, L. N. Rib, "Summaries of Research Reports Submitted in Connection with the Nuclear Plant Aging Research (NPAR) Program," Engineering and Economics Research, Inc. (EER), Reston, VA, September, 1986.

The results of Phase I efforts in the NRC NPAR program for selected electrical and mechanical components since 1984 have been published. To help maintain cognizance of this wealth of information, summaries of 14 reports are presented in this publication. Thus the results of these studies are made more readily available for rapid survey, directing attention to specific reports of interest and facilitating the utilization of research results in the regulatory process.

The 14 reports are grouped into three categories: (1) early scoping and background studies, including a survey of aged power plant facilities, operating experience reviews of Licensee Event Reports (LERs) to identify aging trends, workshops to obtain experts' opinions, and aging/risk considerations; (2) reports on developing a methodology for aging analysis and on evaluation and use of a signature analysis technique (MOVATS); and (3) Phase I results of aging research on nine components, including electric motors, battery chargers/inverters, electrical cables, pressure transmitters, diesel generators, motor-operated valves, check valves, auxiliary feedwater pumps, and snubbers. Each summary has four sections: Background, Summary, Results/Findings, and Utilization of Research Results in the Regulatory Process.

This report is considered a "living" document. That is, research results and summaries of additional selected reports may be added periodically.

Technical Integration Review Group for Aging and Life Extension (TIRGALEX), "Plan for Integration of Aging and Life-Extension Activities," U.S. Nuclear Regulatory Commission, May 1987.

The Technical Integration Review Group for Aging and Life Extension (TIRGALEX) was established to facilitate the planning and integration of NRC activities related to reactor aging and life extension. The initial objectives of TIRGALEX were to identify technical safety and regulatory policy issues related to reactor aging and life extension and to develop a plan to integrate NRC and external activities to resolve the issues. This report contains the plan developed by TIRGALEX, which consists of the following main elements:

1. A summary and discussion of the major technical safety and regulatory policy issues associated with reactor aging and life extension.
2. An overview of ongoing programs and activities related to reactor aging and life extension, including both NRC and external programs and activities.
3. Recommendations for future NRC actions to address reactor aging and life extension in a timely, efficient, and well-integrated manner.

K.R. Hoopingarner and F.R. Zaloudek, "Safety Implications of Diesel Generator Aging," Pacific Northwest Laboratory, *Nuclear Safety*, 31:484–489, October-December 1990.

The emergency diesel generators in a nuclear power plant have an important safety function in that they supply electric power for emergency core cooling and related emergency needs in the event of a loss of offsite power. Typically, a plant has two redundant diesel generators of 3,000 to 8,000 kW (5,000 to 10,000 hp).

Diesel generators have been identified as components with significant safety importance, and their operating history has shown performance degradation and loss of reliability as a result of wear and aging. Consequently, emergency diesel generators are included in the NRC NPAR program, the overall objectives of which are to find the causes of aging-related degradation and to recommend methods for managing it. By ameliorating wear- and performance-related degradation, the reliability of the emergency diesel generators can be improved and the risks associated with loss-of-offsite-power events reduced.

The NPAR Diesel Generator Study consisted of two phases. Phase I used plant operating experience, data, expert opinion, and statistical methods to produce a data base related to aging failures, their causes, and corrective actions. Phase II included the development of a more appropriate testing and aging management program that could enhance the availability and reliability of these diesel generators.

The purpose of this article is to discuss the principal causes of wear- and aging-related degradation of emergency diesel generators and the effects on their reliability and availability and to describe methods by which such degradation can be avoided or detected before it becomes a nuclear safety concern. Operational information assembled on component and system failures and their causes was reviewed to identify the important aging and degradation factors for diesel generators. One important factor contributing to wear and degradation has been the fast-starting and loading test procedure called for by Regulatory Guide 1.108 and the NRC Standard Plant Technical Specifications. A new regulatory approach was recommended to develop a more balanced aging management program that includes (1) slow-start testing during which important operating parameters are monitored, (2) analysis of data trends, (3) training, and (4) maintenance. This approach should improve safety by enabling the timely identification of aging-related degradation that could lead to diesel generator failures so that maintenance could be performed in time to prevent actual failure.

NUMBERED REPORTS

BNL Technical Report A-3270-11-26-84, B. Miller, "Scoping Test on Containment Purge and Vent Valve Seal Material," Brookhaven National Laboratory, December 1984.

Degradation of shaft-seal material used in containment purge and vent butterfly valves may initiate valve seal leakage thus breaching containment. A scoping test was performed to gather information on the behavior of the seal material (ethylene propylene) when exposed to severe accident conditions (i.e., steam at 350°F/120 psig and 400°F/232 psig). Three separate test sequences were performed with the test assembly monitored for leakage. The results of these tests revealed no seal leakage; however, shaft-seal degradation was evident.

For two test sequences, the prescribed procedure was revised to include modified temperature profiles and seal-testing sequences.

Removal and inspection of the valve seat following some test sequences revealed minor remolding of the seat material at the disc/body interface with no deformities noted. Approximately one week later, cracks developed in the seat. The cracks were in an area that would be compressed by the retaining ring and in no instance affected the sealing integrity of the valve.

The results of the scoping test revealed no shaft-seal leakage. The seal degradation and cracking was visually evident in the compressed retaining portion of the seat. However, the result should not be construed as representing the entire ethylene propylene family (elastomers prepared from ethylene and propylene monomers). Varying the relationship of these monomers affects the characteristics of the elastomer and its ability to withstand environmental conditions. It should also be noted that all mechanisms by which rubber deteriorates with time are attributable to environmental conditions. The Parker Seal Company states that it is environment, not age, that is significant to seal life, both in storage and in actual service.

BNL Technical Report A-3270-11-85, J. H. Taylor, M. Subudhi, J. Higgins, J. Curreri, M. Reich, F. Cifuentes, and T. Nehring, "Seismic Endurance Tests of Naturally Aged Small Electric Motors," Brookhaven National Laboratory, November 1985.

Two naturally aged 10-HP electric motors were obtained from an older nuclear power plant that is ready for decommissioning. The motors were utilized to drive fan cooler units in an outside environment for 12 years. These motors were first tested for their dynamic characteristics. They both were subjected to seismic excitation with generic floor response spectra (GFRS) that encompass Safe Shutdown Earthquake (SSE) accelerations applicable to most nuclear plants in the United States. The tests showed that the first fundamental frequency is well above the rigid range of an earthquake frequency. Seismic testing was performed with a motor both unloaded and loaded by an attached hydraulic pump that served as a dynamometer. Significant operating parameters such as current, voltage, and temperature were monitored before, during, and after seismic loading, and no noticeable differences were observed. Existing deficiencies in one of the motor bearings and in the stator winding were not affected or magnified by the seismic excitations.

This report describes the test plan, includes details of the procedure, and presents findings of the seismic tests and operating/static tests on both motors.

This testing was part of the NRC NPAR program, and its results are an integral part of the Brookhaven National Laboratory's overall aging assessment of motors, which was published as NUREG/CR-4156.

BNL Technical Report A-3270-12-85, M. M. Silver, R. Vasudevan, and M. Subudhi, "Pilot Assessment: Impact of Aging on the Seismic Performance of Selected Equipment Types," Brookhaven National Laboratory, December 1985.

The NRC has initiated a number of specific research programs in support of the NPAR program, to better understand the impact of equipment aging on plant safety and to recommend realistic operating and maintenance procedures to improve plant availability and enhance safety. This pilot study was performed to investigate the feasibility of using plant experience data to assess the relationship between equipment aging and seismic performance capacity.

After a brief review of available information on plant experience at many California sites for content and quality, data related to performance, maintenance, and failure history were collected for a sample set of equipment types. This pilot study selected the equipment types for investigation from the highest priority group specified in a previous NPAR study. The equipment types studied were electric motors, motor-operated valves, relays, circuit breakers, and motor control centers.

The acquired equipment data consisted of installation date, chronological listing of preventive and corrective maintenance activities, failed state and cause of failure, earthquake data (i.e., free-field acceleration, Richter magnitude, date), and equipment status before and after the earthquake.

The pilot study was successful in demonstrating that experience data can be extracted and utilized to address the relationship between seismic performance capacity and aging of plant equipment. It is strongly recommended that future research be conducted to acquire experience data for other important equipment types and to investigate other California power plants. Such research will provide the maximum amount of actual experience data to address the aging-seismic relationship in a practical manner. Lessons learned from a review of these data can be used as input to develop practical maintenance and operating procedures to enhance safety and improve plant reliability.

BNL Technical Report A-3270-3-86, A. C. Sugarman, M. W. Sheets, and M. Subudhi, "Testing Program for the Monitoring of Degradation in a Continuous Duty 460 Volt Class "B", 10-HP Electric Motor," Brookhaven National Laboratory, March 1986.

This report presents an evaluation of potential maintenance techniques for monitoring age-related degradation in a continuous-duty 460-volt, Class B, 10-HP electric motor. The program follows up the analyses and recommendations outlined in the draft of NUREG/CR-4156, "Operating Experience and Aging-Seismic Assessment of Electric Motors," by M. Subudhi et al. In this study, the following stressors on dielectrics are evaluated: temperature, frequent starts, overload, and high voltage gradient.

In general, the motor tests are conducted by continuously reversing motor direction for five hours, followed by a half hour with the motor running under no load in a single direction and a half hour with the motor turned off and stationary. During the half hour of running under no load, measurements of bearing vibration and movement of stator end turns (measured with accelerometers epoxied to the end turns) were made. Also, a number of insulation tests were conducted. To accelerate the degradation of the test motor (including insulation, bearings, and lubrication), a plug reverse test was performed.

The results of the exploratory testing program revealed which insulation and bearing tests can best be used in utilities' procedures for preventive maintenance, corrective maintenance, and surveillance for safety-related motors.

This testing is meant for motors rated for continuous use. A separate test plan will be required for intermittent-duty motors (e.g., valve actuator motors); such a plan should include typical valve actuator tests such as the open/close cycling test and the insulation tests discussed in Section 4.0 of the presently reported program.

BNL Technical Report A-3270-12-86, R. Fullwood, J. C. Higgins, M. Subudhi, and J. H. Taylor, "Aging and Life Extension Assessment Program (ALEAP) Systems Level Plan," Brookhaven National Laboratory, December 1986.

This system level program plan for ALEAP presents and explains the BNL structured approach to assessing the effects of the aging of nuclear power plant components and systems on safe operation and the extension of plant operation beyond the originally planned plant life. It should be noted that this plan is prepared in a generic fashion and could be used by anyone for a system assessment.

The plan discusses the criteria for prioritizing plant, system, and component selection for analysis to determine the effects of aging. The use of failure modes and effects analysis in conjunction with the results of natural and accelerated aging tests are discussed as means for understanding and predicting the phenomena. The effects of aging on the failure rates of components are being determined principally from plant data with physical and phenomenological models used for interpolation in areas not included in the data base. These results will be integrated with a plant risk model to be used in addressing the question "how old is old enough."

The NRC NPAR program has completed several component-level aging assessments that include the identification of dominant component failure modes based on plant operating experience. The studies provide recommendations for monitoring as well as mitigating age-related component degradations.

Utilizing results from the component-level studies and work performed by other NRC contractors for system-data assessment and system-level risk analysis, this program evaluates the impact of component failures on plant system performance. The study performs in-depth system-level failure-data reviews, evaluates current industry practices for system maintenance, testing, and operation and probabilistic risk assessment (PRA) techniques to understand and to predict the impact of aging on system availability. Recommendations for improving the system performance by means of degradation monitoring and timely preventive and corrective maintenance are addressed. This project integrates its products with the BNL programs for operational safety reliability research and performance indicators.

The first phase of this research effort concentrates on understanding various system designs from plant safety analysis reports, evaluating failure data from plant operating experience data bases, applying PRA analyses, assessing industry-wide surveillance and maintenance practices, and identifying system functional indicators that are used to monitor the rate of system degradation resulting from aging and service wear. The program separates failures on demand from time-dependent failures. It categorizes age-related failures separately from random and design-type failures. It produces results useful for the resolution of pertinent unresolved safety issues and for review and inspection of operating NPPs. The second phase, if authorized and performed, will provide recommendations for improving system performance through enhanced maintenance practices and reliability monitoring, which will be focused on the most risk-sensitive areas of a system. Recommendations are made for improvements in pertinent regulatory guides, industry standards, etc. This program plan delineates the goals and major tasks to be completed in each phase. The current version of the program plan is considered to be a draft and will be revised and updated as the first few system assessments are completed using this methodology. This will produce a final proven methodology that car [illegible] lied to the remaining systems.

BNL T [illegible] ..eport A-327OR-2-39, A. Fresco and M. [illegible]ubudhi, "Aging Effects of Impoi tant Balance of Plant Systems in Nuclear Power Plar ts," Brookhaven National Laboratory, February 1990.

In recent years, balance of plant (BOP) systems have become major causes of plant transients, e.g., the June 9, 1985 loss-of-all-feedwater event at the Davis-Besse Nuclear Power Plant, and have received increased attention from the nuclear industry and the Nuclear Regulatory Commission (NRC). This interim report describes the activities to date in a study of BOP systems by Brookhaven National Laboratory in support of the NRC Nuclear Plant Aging Research (NPAR) program. The initial phase of the study provides preliminary indications of those BOP systems that may warrant a detailed study of aging effects. An approach for accomplishing the overall objective of identifying the effects of aging in these BOP systems on nuclear plant safety is suggested.

This study on BOP systems covers all non-safety-related systems except for those associated with the nuclear steam supply system (NSSS). Some non-safety-related systems in the NSSS are being studied in other parts of the NPAR program.

From the results of the study, it was concluded that the frequency of unplanned reactor trips has often been cited as an indicator of safety performance and that the most frequent contributors to unplanned reactor trips caused by BOP systems are the power conversion systems, i.e., the feedwater, main turbine, main electric generator, main steam (usually the steam bypass to the main condenser), and condensate systems. Other BOP systems contributing to unplanned reactor trips are support systems such as the electric distribution system and, less frequently, the circulating water, service/instrument air, fire protection, and the heating, ventilation, and air conditioning systems. The electric distribution system includes 120-V AC power distribution systems, the switchyard, large plant load users, the DC power system, and control centers. At the component level, the feedwater regulating valves, the turbine-driven feedwater pumps, and the main turbine electro-hydraulic control subsystem are frequent contributors. Failures in the main electric generators are also important as potential causes of reactor trips.

These results are substantially in agreement with the results of an alternative approach in which important BOP systems were categorized based on insights from probabilistic risk assessments.

Preliminary recommendations are:

1. The frequency of unplanned reactor trips should be considered the most important indicator of current or potential near-term safety problems.
2. BOP systems that significantly contribute to unplanned reactor trips should be included in the NPAR program.

The next phase of the program will focus on several of the oldest nuclear plants. Licensee Event Reports involving unplanned reactor trips from the beginning of commercial operation to the present will be reviewed to determine if there is an increasing frequency of unplanned trips caused by the identified BOP systems. This group of plants will include Monticello and Yankee Rowe, the pilot plants for the life extension study.

Next, some plants of intermediate age and then some of the youngest plants will be examined in the same manner. These three age groups will be compared and analyzed. The ultimate goal is to determine whether aging of individual BOP components is a significant factor affecting nuclear plant safety.

BNL Technical Report TR-3270-6-90, W. Gunther, "Maintenance Team Inspection Results: Insights Related to Plant Aging," Brookhaven National Laboratory, June 1990.

NRC is performing maintenance team inspections in accordance with the NRC temporary instruction (TI) 2515/97 entitled "Maintenance Inspection" to determine the effectiveness of the total integrated maintenance process in nuclear power plants. As specified in the TI, "the goal of the inspection effort is to emphasize the use of plant experience, test and surveillance data, recent component failures, [and] Probabilistic Risk Assessment (PRA) insights..." to identify strengths and weaknesses. Two volumes of inspection guidance supplement the TI and direct the inspectors to evaluate the maintenance/aging relation. For example, the inspector is directed to determine the extent to which management is aware of plant aging. The inspector is also required to evaluate the involvement of corporate management in maintenance activities that address plant aging.

More important than the cited guidelines is the expected assessment of maintenance program activities that reflect on the ability of the plant technical staff to manage aging. Predictive and preventive maintenance programs must include condition monitoring, trending, and recordkeeping in order to competently manage the effects of aging on a timely basis. Inadequacies found in these areas are indicative of the inability of the plant staff to properly address and treat aging.

The in-depth performance-based inspections have explored such areas as overall plant performance related to maintenance, management support of maintenance, and the implementation of the maintenance program. Incorporated in the inspection criteria established for these general areas are specific attributes that are relevant to the understanding and managing of aging. The related activities include root cause evaluation of equipment failure, trending of failure data, implementation of equipment qualification programs, control of spare parts, evaluation of test results (including postmaintenance testing), and implementation of condition-monitoring techniques.

The maintenance team inspection reports were reviewed with the following objectives in mind:

1. Assess the evaluations of those portions of the maintenance program determined to be important for understanding and managing aging.
2. Evaluate the weaknesses noted in utility maintenance programs that could affect the ability of the plant to manage aging.
3. Determine the types of preventive maintenance activities and condition-monitoring techniques that address plant aging.

At this time, 47 inspections have been completed, and 24 more are scheduled for the remainder of 1990. NRC has compiled the findings from these inspections in a computerized data base that assisted in identifying plants where the NRC inspection teams had concerns about how well the maintenance program accounted for the effects of aging. Ten reports contained specific findings that the utility maintenance programs do not address aging. These findings are tabulated and summarized in this report.

It should be noted that nine of the above ten reports concluded that the overall maintenance programs were adequate, satisfactory, or good. The guidance and criteria provided to the NRC maintenance inspection team allow a maintenance program to be judged good if it effectively manages current problems even though it may not effectively manage the long-term aging effects on structures, systems, and components.

BNL Technical Report TR-3270-9-90, E. Grove and W. Gunther, "An Operational Assessment of the Babcock & Wilcox and Combustion Engineering Control Rod Drives," Brookhaven National Laboratory, September 1990.

Control rods and the associated drive and control systems, which ensure safe and reliable operation, are essential components of nuclear reactors. This report describes an aging assessment of the Babcock & Wilcox (B&W) and Combustion Engineering (CE) control rod drive (CRD) systems performed as a part of the NRC NPAR program. Emphasis was placed on the specific components of the systems that may be susceptible to aging-related degradation. This study along with a failure modes and effects analysis and a detailed review of utility maintenance practices and procedures will complete Phase I of the aging assessment of these CRD systems.

Of the fifteen plants that use CE CRD systems, thirteen use a magnetic jack control element drive mechanism, and two use a rack and pinion drive. In all eight B&W plants, the control rods are driven by a roller nut/leadscrew drive mechanism. All CE reactors had basically the same CE logic, control, and rod position systems; all B&W reactors had basically the same B&W logic, control, and rod position systems.

Commercially available operating experience data bases were reviewed to identify failed components and the resultant effects on plant operation for the 1980–1989 time period. Age-related failures that resulted in significant plant events, including dropped rods, power reduction, and shutdown for B&W and CE control rod drive systems were identified. Susceptibility of the system to such external influences as maintenance errors and the operating environment was also shown.

BNL Technical Report A-3270 6-21-91, F. Hsu, W. E. Vesely, E. Grove, M. Subudhi, and P. K. Samanta, "Degradation Modeling: Extensions and Applications," Brookhaven National Laboratory, June 1991. Available from the NRC Public Document Room.

Component degradation modeling includes modeling of occurrences of component degradations and analyses of these occurrences to understand the degradation process and its implications, one of them being that degradation rate serves as a precursor of the failure rate. The degradation modeling discussed in the report focuses on the analysis of times of degradation and failure occurrences in order to understand the aging degradation of components. This report reflects the previous BNL work related to the same basic concepts and the mathematical development of a simple degradation model. Using the degradation modeling methodology, failure data on residual heat removal (RHR) pumps and service water (SW) pumps were analyzed to detect indications of aging and to infer the effectiveness of maintenance in preventing age-related degradations from transforming to failures. In this report, further applications and extensions of degradation modeling are discussed.

Additional applications of degradation modeling are carried out for air compressors, which are continuously operating components. It was demonstrated that the analysis of degradation occurrences is useful in understanding the aging process and the role of maintenance in that process. For the air compressors, the failure rate and degradation rate show an early decreasing trend followed by a significant increasing trend that indicates effects of aging. The failure rate, which is significantly lower than the degradation rate in the first 3 years, increases faster in the later years, reaching approximately the same value as the degradation rate at the end of 10 years of operation. This behavior indicates the ineffectiveness of maintenance in preventing degradations from transforming into failures as the air compressors age.

Another important application of degradation modeling is to predict the age-related failure rate from the degradation rate. Experience shows that time-dependent failures generally pass through a degradation state first. An observed increased aging contribution in the degradation rate can signal an expected increase in aging-related failure rates. A simple linear relationship between these two parameters is studied, considering any delayed effect that degradations may have on failures. An example of an application of the data on RHR pumps shows a time lag of 2 years for degradations to affect the occurrence of failures.

For additional applications, extensions of degradation modeling are presented. The extended models the authors are developing will explicitly show the reliability effects of different maintenance and test intervals, different maintenance and test efficiencies, and different repair times. Thus, the extended model will allow the user to evaluate the reliability effects of different maintenance programs.

EGG-SSRE-8972, C.L. Atwood, "Estimating Hazard Functions for Repairable Components," Idaho National Engineering Laboratory, May 1990.

This tutorial report, applying known formulas and tools in a way suitable for risk assessment, presents a parametric framework for performing statistical inference on a hazard function based upon such data of repairable components as might be obtained from field experience rather than from laboratory tests. This framework encompasses many possible forms of the hazard function, three of which are considered in some detail. The theory is neatest and the asymptotic approximations are most successful when the hazard function—for a set of identical components—has the parametric form of a density in the exponential function family. The parameters are estimated based on sequences of failure times when the components are restored to service immediately after each failure. In certain circumstances, the distribution of the failure counts does not depend on the parameter that determines the shape of the hazard function; this suggests natural tools for diagnostic checks involving the individual parameters. The results presented include formulas for maximum likelihood estimates (MLEs), tests and confidence regions, and asymptotic distributions. The confidence regions for the parameters are then translated into a confidence band for the hazard function. For the three examples considered in detail, a table displays all the building blocks needed to program the formulas on a computer; this table includes asymptotic approximations when they are necessary to maintain numerical accuracy. Diagnostic checks on the model assumptions are outlined.

The report gives an example of an analysis of real data. In this example, the methods applied are unable to discriminate among an exponential hazard function, a linear hazard function, and a Weibull hazard function. The MLE for the two parameters appears to have approximately a bivariate normal distribution under the exponential or Weibull hazard model but not under the linear hazard model. If the analysis using approximate normality is carried out in any case, the results appear similar for all three models. If some model is preferred for theoretical or other reasons, this report indicates a way to use it.

EGG-SSRE-9017, C.L. Atwood, "User's Guide to PHAZE, a Computer Program for Parametric Hazard Function Estimation," Idaho National Engineering Laboratory, July 1990.

The program PHAZE (for Parametric HAZard Function Estimation) performs statistical inference on a hazard function (also called a failure rate or intensity function) based on reported failure times of components that are repaired and restored to service. Three parametric models are allowed: the exponential, linear, and Weibull hazard models. The inference includes estimation (maximum likelihood estimators and confidence regions) of the parameters and of the hazard function itself, testing of such hypotheses as increasing failure rate, and checking of the model assumptions under a choice of parametric models.

Since the approach's concern is the failure behavior of components, these failures are assumed to be governed by a Poisson process, with the time typically measured from the component's installation. It is further assumed that, when a component fails, either it is immediately repaired and placed back in service or it is replaced by a new component. Failures of distinct components are assumed to be independent. Thus the data to be analyzed consist of sequences of failure times of similar independent components.

This user's guide sketches only enough of the theory to permit PHAZE to be used; a full presentation of the theory is given in a companion report. A typical PHAZE application is described. This consists of an initial exploratory phase, in which the various model assumptions are checked, and a final estimation phase, in which the maximum likelihood estimator and a confidence interval are found for the hazard function at times of interest. The format of a data file is given with examples. PHAZE is an interactive command-based program; all the PHAZE commands are therefore listed and explained.

The program has been verified and validated, and this work is summarized. Finally, some of the technical details of interest to statisticians and programmers are given. The appendix shows an entire PHAZE session, including both the user's commands and the program's responses. Virtually all of the PHAZE commands are illustrated, and the resulting output is presented.

EGG-SSRE-9777, J. C. Watkins, R. Steele, Jr., and K. G. DeWall, "Isolation Valve Assessment (IVA) Software Version 3.10, User's Manual," Idaho National Engineering Laboratory, June 1991. Available from the NRC Public Document Room.

The Isolation Valve Assessment (IVA) software is a PC-based computer program developed to assess the performance capabilities of a 5-degree flexwedge motor-operated gate valve with a Limitorque operator in the closing direction. The software provides easy-to-use data entry screens to input system design basis conditions and selected valve and motor-operator parameters. Context-sensitive help information is available to clarify the required input and to provide typical input values in the event actual values for the valve-motor-operator unit are not available.

The software enables the calculation and display of the stem thrust, operator torque, and motor torque required to operate the valve using both the standard industry equation and the correlation developed by the Idaho National Engineering Laboratory (INEL). These calculations (of stem thrust, operator torque, and motor torque) pertain to unit capability of delivery at nominal, minimum, and maximum voltage conditions. The results obtained are then displayed to aid the user in determining whether the motor-operated valve (MOV) is capable of functioning at design basis conditions.

A series of four graphs can also be displayed to aid the user in evaluating the functionality of an MOV. The first graph compares the required versus available stem thrust as a function of stem factor for both nominal and degraded voltage conditions. The second graph provides the same information as a function of stem to stem nut coefficient of friction. The third graph displays the conversion of operator torque to stem thrust for a series of stem to stem nut coefficients of friction. The fourth graph relates the stem thrust to torque switch settings for a number of stem to stem nut coefficients of friction.

The software can also be used to assess low flow/low differential pressure diagnostic test results to determine whether the response of a valve is typical of the valves tested by the INEL. If typicality is confirmed, the INEL correlation can then be used to assess the response of the valve at design basis conditions. A single graph is needed to assist in this assessment. This graph, available on the software, relates the evaluation of the low flow/low differential pressure diagnostic test data to the results of a similar evaluation of the valves tested by INEL.

The IVA software was developed by the INEL under contract to the Office of Nuclear Regulatory Research of the NRC. The software has been verified and validated in accordance with the requirements of the American Society of Mechanical Engineers quality assurance requirements for nuclear facility applications. The NRC has made the software available to the public and it may be obtained through INEL.

EGG-SSRE-9926, R. Steele, Jr., J. C. Watkins, and K. G. DeWall, "Evaluation of EPRI Draft Report NP-7065—Review of NRC/INEL Gate Valve Test Program," Idaho National Engineering Laboratory, November 1991. Available from the NRC Public Document Room.

This report documents an INEL evaluation of an EPRI draft report that was critical of the NRC/INEL gate valve test program and was being used by some utilities to discount the applicability of the INEL gate valve test data. The authors' purpose was to resolve the comments of the review team sponsored by EPRI and provide strong technical bases for areas of agreement and disagreement between the INEL and EPRI positions. Areas of disagreement between EPRI and the NRC/INEL include predictable versus nonpredictable valve performance, motor-operator sizing equations, typicality of the test hardware, applicability of the Phase I and the Phase II (work schedules) test conditions, assessment of valve response as indicated in the measured stem force, and the validity of NRC documents issued relating to valve testing. The INEL positions as described in the report are summarized below.

The most important questions raised by the INEL testing was how to determine whether or not a valve will perform predictably. The EPRI report suggested that valves with different disc designs might perform differently from those tested in the NRC/INEL program. The results from INEL, EPRI-Marshall facility, and European testing have all shown that a large number of the valves tested performed unpredictably. Test experience indicates that internal damage occurs as the body-guide to disc-guide clearances increase. The threshold level where current valve designs begin to perform unpredictably needs to be determined and accounted for.

The INEL analyses of the Phase I and Phase II test results were not complete when the EPRI review was performed. At that time, INEL was trying to analyze the test results using standard industry equations. Only after INEL relinquished its efforts to fit test data into the standard industry equation could the apparent randomness in the results be correlated with fluid subcooling and pressure effects. At that point, the conclusion was drawn that, because the disc factor includes other terms along with sliding friction, the industry's traditional stem force equation used to size motor operators is incomplete, regardless of the disc area term or the disc factor used.

INEL selected 5-degree flexwedge gate valves for testing based on surveys of equipment installed in nuclear service. Not every variation produced by every valve manufacturer was tested; however, among the six valves with seven internal configurations, most of the design features (except the double disc design) of most of the wedge gate valves used to formulate Generic Issue 87, "Failure of HPCI Steam Line Without Isolation" (GI-87, September 1985), were tested.

The typicality of the hardware used in the INEL testing and the actual test conditions provided representative GI-87 operating conditions for evaluating representative GI-87 valves. Although the EPRI report questioned whether the test conditions were applicable to GI-87 valves, the research tests performed sufficiently simulated the worst case conditions and provided a correlation that can be used once the design basis conditions for a valve are known and the valve's predictable behavior established.

For a predictable valve, that portion of the closure stroke that occurs just after flow isolation, when the disc is riding fully on the seats, is very well defined on a stem force versus time trace. The sliding friction or disc factor used to size a valve must be based on a correct calculation of this point. Determination of this point and the corresponding final stem force will help utilities evaluate performance margins or establish that the valve is representative of the valves tested by the INEL for predictable extrapolation purposes.

In conclusion, the NRC documents reviewed in the EPRI report have some omissions in the description of circumstances, but after two years of additional analyses, the conclusions presented in the documents are still valid.

EGG-SSRE-10039, T.H. Hunt and M.E. Nitzel, "An Evaluation of the Effects of Valve Body Erosion on Motor-Operated Valve Operability," Idaho National Engineering Laboratory, May 1993.

Engineers at the Idaho National Engineering Laboratory evaluated the effects of erosion-induced valve wall thinning on motor-operated valve operability. The authors reviewed reports that identified the extent and location of erosion damage in nuclear plant valves and chose a globe valve with severe erosion damage to assess the potential for loss of operability. They developed a finite element model of the selected valve and performed a structural analysis with valve closing forces to analyze the effects of the erosion on structural integrity. The results indicate that sufficient margin to yield stress remained. Therefore, erosion-related wall thinning is not likely to create an operability problem for motor-operated valves.

NISTIR 4485, F. D. Martzloff and A. G. Perrey, "Annotated Bibliography: Diagnostic Methods and Measurement Approaches To Detect Incipient Defects Due to Aging of Cables," National Institute of Standards and Technology, July 1991.

This annotated bibliography has been prepared to document the literature search conducted as an initial task for the project of assessing existing test methods for detection of incipient faults in nuclear power plant cables. The combined listing presented in this report can help future work that may be performed by other organizations involved in similar studies. As a further aid to researchers, the references provided in each of the papers included in this bibliography have been consolidated in a citations list. A hard copy of each paper included in this bibliography has been retained at NIST.

This report is organized in three sections:

- Alphabetical listing of all the reviewed docments;
- Review of authored papers;
- Consolidated citations.

The initial search covered the 1970–1986 period, combining several data bases and incidental referrals to reports with limited circulation. The motivation of this search, at the outset of the project, was to identify any and all test methods reported in the literature that might be applied, refined, or developed into a test method for in situ assessment of the cable condition. An additional set of papers was reviewed, covering the 1986–1990 period. Each review is shown on a separate sheet giving retrieval information, author's abstract, table of contents of a reviewer's summary, identification of the technology involved, brief remarks on the contents and applicability to the subject of in situ test methods, and identification, if any, of the criteria and test methods used to characterize aging or residual life of the insulation.

Since these remarks are made from the point of view of applicability to the limited subject of assessing in situ testing, they should not be construed as an overall, definitive review of the papers. For example, some publications might contain a relevant description of a method applied in assessing the aging of the insulation and thus provide a lead toward a good candidate method for other purposes. Also, papers are still listed that were identified in the search as potentially relevant but found upon review not to be relevant.

This inclusion of irrelevant papers can save the readers the unnecessary effort of acquisition and review of seemingly relevant papers that might eventually prove not applicable to their problem. Conversely, significant papers may have been missed in the search, and their absence from this compilation should not be construed as a negative judgment of their significance.

NISTIR 4487, F. D. Martzloff, E. Simmon, J. P. Steiner, and R. J. Van Brunt, "Detection of Incipient Defects in Cables by Partial Discharge Signal Analysis," National Institute of Standards and Technology, July 1992.

As one of the objectives of a program aimed at assessing existing test methods for in situ detection of incipient defects caused by aging in cables, a laboratory test system was implemented to demonstrate that the partial discharge analysis method can be successfully applied to low-voltage cables. Previous investigations generally involved cables rated 5kV or higher, while the objective of this program focused on the lower voltages associated with the safety systems of nuclear power plants.

The system implemented for this demonstration was based on commercially available signal analysis hardware and software packages, customized for the specific purposes of the project. The test specimens included several cables of the type found in nuclear power plants, with artificial defects introduced at various points of the cable. The final demonstration of the system included a cable with several defects inflicted at locations kept undisclosed to the test operator. Using a combination of the standard signal analysis built into the system and experience-based detailed analysis of selected portions of the data base, the operator was able to identify the existence and location of all these undisclosed defects.

These results indicate that, indeed, partial discharge analysis is capable of detecting incipient defects in low-voltage cables. There are, however, technical and nontechnical limitations that need further exploration before this method can be accepted in the industry.

NISTIR 4787, F. I. Mopsik, "The Use of Time-Domain Dielectric Spectroscopy To Evaluate the Lifetime of Nuclear Power Station Cables," National Institute of Standards and Technology, April 1992.

This report describes the work that has been undertaken at NIST to see if the method of Time-Domain Dielectric Spectroscopy can evaluate the aging in reactor cables with the goal to ultimately estimate lifetimes.

The Time-Domain Dielectric Spectrometer is used in a method for measuring dielectric properties over the frequency range from 104 to 10^{-4} Hz by obtaining the sample response to a step application of voltage. The data are then converted to the frequency domain by a numerical Laplace transform. This method renders high precision and sensitivity and allows acquisition of the entire frequency range, typically at least seven decades, in a time that is less than that required for one cycle of the lowest frequency of interest.

One of the current problems in the operation of aging nuclear power plants is the estimation of the lifetime of the electrical insulation used inside the reactor confinement. This insulation, having been manufactured for installation in reactors, has been subject to lifetime testing prior to certification for use. This testing program has been quite successful, as there have been too few failures to make any changes in the requirements.

As the plants age, however, the question has arisen whether the life remaining in the cable insulation can be estimated so that the cables can be used beyond their original certified life. If this were possible, costly and difficult overhauls of the reactor wiring could be avoided. Given the conservative nature of the certification process, this possibility of extended lifetime certainly exists.

The original tests on the cables were acceptance tests in which the cables were subjected to an environment more severe than found in service, and the cables had to maintain their integrity. These tests have resulted in a situation in which cables are possibly used for only a small part of their potential lifetime. For estimation of a lifetime, such tests provide little information other than a possible minimum allowable exposure. These acceptance tests do not follow changes in the cables as they age. The effects of accelerated aging used in these tests are not studied. Finally, there is no indication at all of the ultimate failure mechanisms of the cables and when and how the failures might occur.

The degradation of electrical insulation under combined thermal and radiation stresses is a well-known phenomenon. Unfortunately, the rate at which this degradation occurs is highly material-dependent, and a small change in additives can make a large difference. In addition, the mechanisms are very complex, with many possible deterioration scenarios. All these factors make theoretical prediction of lifetimes very difficult. They also create the possibility of relatively large variations in the useful lifetime in a given material (composition).

This investigation was to determine whether some measure of cable aging (e.g., deterioration) could be found. Ideally, undesired changes would vary smoothly with aging and would display measurable changes that are above any possible sample-to-sample variations. Finally, there might be a possibility for adding instrumentation (preferably nonintrusive) without major modifications to any reactor.

Lifetime estimation is complicated by the requirement that any electrical cable must remain functional during a loss-of-coolant accident (LOCA). Since such an event can put severe thermal and radiation stresses on the cables inside a reactor, the cables must not be close to failure prior to such an event. It should be stressed that although a cable at the end of its useful life could appear to be quite normal, its mechanical and electrical properties may deteriorate quickly when the insulation is near failure.

The estimation of lifetime for reactor cables currently in use is very difficult because of the combination of the above considerations. Not only is a convenient measure of age difficult to establish, but all measurements must be conducted in a relatively harsh environment. Furthermore, any estimation must attempt to predict the future, including the possibility of an event more severe (with respect to lifetime) than the total environmental aging allowable to a real (accident-free) endpoint.

NUREG–1144, B. M. Morris and J. P. Vora, "Nuclear Plant Aging Research (NPAR) Program Plan," U.S. Nuclear Regulatory Commission, July 1985.

NUREG–1144, J. P. Vora, "Nuclear Plant Aging Research (NPAR) Program Plan," Rev. 1, U.S. Nuclear Regulatory Commission, September 1987.

NUREG–1144, "Nuclear Plant Aging Research (NPAR) Program Plan, Status and Accomplishments," Revision 2, U.S. Nuclear Regulatory Commission, June 1991.

The Nuclear Plant Aging Research (NPAR) program described in this plan is intended to resolve technical safety issues related to the aging degradation of electrical and mechanical components, safety and support systems, and civil engineering structures used in commercial nuclear power plants. The aging period of interest includes the period covered by the original operating license as well as the period of extended plant life that may be requested in utility applications for license renewals.

Emphasis has been placed on identifying and characterizing the mechanisms of material and component degradation during service and utilizing the research results in the regulatory process. The research includes evaluating methods of inspection, surveillance, condition monitoring, and maintenance as means of managing and mitigating aging effects that may affect safe plant operation. Specifically, the goals of the program are to:

1. Identify and characterize aging mechanisms and effects that could cause degradation of components, systems, and civil engineering structures and, if unchecked, impair plant safety.
2. Evaluate residual life of components, systems, and civil structures and identify methods of inspection, surveillance, and monitoring that will ensure timely detection of aging effects before loss of safety functions.
3. Evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

NUREG/CP–0036, (Compilation by) B. E. Bader and L. A. Hanchey, "Proceedings of the Workshop on Nuclear Plant Aging," Sandia National Laboratories, SAND82–2264C, November 1982.

The objective of the workshop, held August 4–5, 1982, in Bethesda, Maryland, was to facilitate an exchange of thoughts between the NRC and industry on time-related degradation and its influence on reactor safety. The specific goals were to define the problem, to discuss the state of knowledge on aging phenomena, and to identify future activities necessary to understand the problem.

The need for a comprehensive program to identify the potential safety problems associated with plant aging was stressed. It was suggested that the effects of time-related degradation on the safety of the complete reactor system should be evaluated in terms of the risk to the public. One should consider multiple causes that have typically been associated with abnormal occurrences. Since individual component failures create problems, time-related degradation will ultimately have to be addressed in terms of maintenance, monitoring, surveillance, etc., of components.

A large number of phenomena that can cause failures were discussed; a detailed list of parts/materials, including lubricants and other additives that must be considered, was given; seemingly minor changes in the chemical constituents of a material or in the manufacturing process can cause significant effects and changes in the system during operation (e.g., water chemistry effects).

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

There were extensive discussions on the limitations of accelerated-aging tests. The use of naturally aged equipment for test purposes was suggested. Sacrificial replacement of equipment was identified as a source for naturally aged plant equipment.

Maintenance and surveillance in plants and their relationship to time-related degradation were extensively discussed.

NUREG/CP-0100, A. F. Beranek, "Proceedings of the International Nuclear Power Plant Aging Symposium," U.S. Nuclear Regulatory Commission, March 1989.

This report presents the proceedings of the International Nuclear Power Plant Aging Symposium that was held at the Hyatt Regency Hotel in Bethesda, Maryland, on August 30-31 and September 1, 1988. The Symposium was presented in cooperation with the American Nuclear Society, the American Society of Civil Engineers, the American Society of Mechanical Engineers, and the Institute of Electrical and Electronics Engineers. There were approximately 550 participants from 16 countries at the Symposium.

A total of 48 papers were presented in 7 technical sessions:

1. Aging Research Programs,
2. Aging of Structures and Mechanical Equipment,
3. Aging of Electrical Equipment,
4. Aging of Systems and Components,
5. Reliability,
6. Role of Maintenance in Aging Management,
7. Aging of Vessels and Steam Generators.

NUREG/CP-0105, Proceedings of the Seventeenth Water Reactor Safety Information Meeting, Vol. 3, U.S. Nuclear Regulatory Commission. Paper by J.A. Christensen, "NPAR Approach to Controlling Aging in Nuclear Power Plants," Pacific Northwest Laboratory, PNL-SA-17487, March 1990.

For about 8 years, the NRC NPAR program has been developing a technical understanding of and guidance for mitigating the effects of the time-dependent processes responsible for the aging-related deterioration of structures, systems, and components that can reduce safety margins in a nuclear power plant.

Control of the effects of aging is at the center of the NPAR efforts; it consists of three key elements: (1) selection of the structures, systems, and components in which aging-related degradation must be controlled, (2) understanding the mechanisms and rates of the degradation, and (3) managing the degradation through effective surveillance and maintenance. These elements are implemented through various ongoing NRC and industry programs and initiatives as well as by conventional regulatory instruments. Also, the three elements are being addressed in a compilation of good practices that will integrate the information developed under NPAR and other studies of aging into a systems-oriented format that tracks directly with the Safety Analysis Reports.

The need to mitigate time-dependent deterioration of NPP components is not a new or recent concept. Degradation with time is, or should be, a prime consideration in any design effort. The material specifications and mechanical designs that characterize the structures, systems, and components of nuclear power plants reflect conscious, detailed concern on the part of the designer for the effects of anticipated service environments and stressors on functionality over time. The major codes and standards upon which nuclear power plant designs and inservice inspections are premised (e.g., ASME Boiler and Pressure Vessel Code, Sections III and XI) are based in large measure on recognized needs to achieve acceptable performance over a reasonable time whether or not age-related degradation is explicitly addressed by the wording in the codes. Even though time-dependent degradation is fundamental to the codes and standards that govern the design, construction, and operation of nuclear power plants, additional concern over aging is warranted for the following reasons:

1. The lack of specificity on time rates of deterioration in the codes requires the exercise of considerable judgment in design and other functions. The availability of explicit, detailed information on aging rates and consequences can result in more consistent judgments.
2. Developing technology will require that the understanding of aging-related degradation be pushed beyond that inherent in current codes.

3. Accounting for aging in design and operational guidelines depends on several conservatisms to provide prudent assurance that failures will not occur. The management of aging, however, requires understanding and control of the time-dependent degradation that actually occurs to implement and evaluate maintenance programs.
4. The design process considers aging-related degradation of single components in estimating design lifetimes, but does not take into account the implications of common-cause failures due to aging in redundant components or the amplified effect of aging in failure sequences involving interactions of multiple components. The evaluation of these effects requires that accurate failure rate vs. time data for each component be applied using probabilistic methods that realistically model systems on a plant-specific basis.

Because of these kinds of considerations, specific concerns, independent of plant design and operational parameters, over how and why structures, systems, and components degrade with age must be resolved. The generic content and structure of programs for addressing degradation due to aging are discussed in this paper.

NUREG/CR-2641, J. P. Drago, R. J. Borkowski, D. H. Pike, and F. F. Goldberg, "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report," Oak Ridge National Laboratory, ORNL/TM-8271, July 1982.

The development of a component reliability data base for use in nuclear power plant probabilistic risk assessments and reliability studies is presented. The data sources are the in-plant maintenance work request records from a sample of nuclear power plants. This data base is called the In-Plant Reliability Data System (IPRDS). Its features are compared with other data sources such as the Licensee Event Report (LER) system, the Nuclear Plant Reliability Data (NPRD) system, and IEEE Standard 500. Generic descriptions of nuclear power plant systems formulated for IPRDS are outlined in the text.

The major objective of the program described is to provide an improved multipurpose data base. Components of each type of NSSS are included in the data base.

In addition to providing information on past failure rates and component down times, the IPRDS may be used for:

1. revising component test intervals and allowable down times;
2. identifying generic problems and recurring failures;
3. identifying the variables (e.g., environment, operating mode, system, maintenance policy, etc.) that control component failure rates;
4. providing an extensive data base against which to compare existing data sources (e.g., LERs and NPRDs) to assess the degree to which these data sources accurately reflect the actual component reliability;
5. correlating current incidents with previous failures, allowing for extrapolation in the near future;
6. identifying trends and patterns in the failure characteristics of particular components or aggregations of components; and
7. identifying failure mechanisms over time for use in defining the aging requirements for component qualification.

NUREG/CR-3154, R. J. Borkowski, W. K. Kahl, T. L. Hebble, J. R. Fragola, and J. W. Johnson, "The In-Plant Reliability Data Base for Nuclear Plant Components: Interim Report—The Valve Component," Oak Ridge National Laboratory, ORNL/TM-8647, December 1983.

This document details the collection and preliminary analyses of data related to valves in the In-Plant Reliability Data System (IPRDS). The data base is developed primarily from historical records of corrective maintenance actions obtained directly from nuclear plant maintenance files. A comprehensive valve population is also included. This report presents data from one PWR and one BWR power plant.

The report demonstrates the degree of distinction and refinement in the reliability statistics that is possible with data from the IPRDS and suggests a general format for disclosure of suitable reliability statistics to satisfy needs within the nuclear data-gathering community. The examples given in the various tables and figures suggest a useful method of comparing valve data and are representative of the degree to which reliability statistics for any particular valve can be ascertained.

One objective of this report is to examine the improvement possible using IPRDS in refining the statistics to ultimately focus on the reliability of particular valve types and valve operators in specific working environments. Another objective is to generate comments from members of the nuclear data community as to the efficacy of the suggested formats for documenting valve information and the various methods used for comparison in this report.

The report gives breakdowns of failure rates by failure modes and by failure causes showing calculated maintenance frequencies and repair times. IPRDS repair time distributions, although unavailable from LERs, are also presented and evaluated.

Preliminary results obtained from the pilot data base in this report indicate WASH-1400 statistics to be nonconservative in reliability estimates for some valve types in certain failure modes.

NUREG/CR-3543, G. A. Murphy, R. B. Gallaher, M. L. Casada, and H. C. Hoy, "Survey of Operating Experiences from LERs to Identify Aging Trends," Oak Ridge National Laboratory, ORNL-NSIC-216, January 1984.

This report describes the preliminary results of an assessment of information pertinent to identifying age-related failures available in operating experience reports. This assessment, by the Nuclear Operations Analysis Center (NOAC) at Oak Ridge National Laboratory, utilized the computerized files of Licensee Event Reports (LERs) and their predecessors to examine age-related degradation of safety-related equipment.

Abstracts of operating experience reports from commercial power plants reported from 1969 to 1982 were surveyed. Over 7000 events were reviewed. Data included the system, component, subpart, the age-related failure mechanism, the severity, and the method of detection of the failure. Wear, corrosion, crud, and fatigue were the identified failure mechanisms in over one-third of the 3098 age-related events. About two-thirds of the failure severities were judged as a degraded state, and one-third were judged as catastrophic failures. Pump and valve problems made up almost 30% of the failed components. Almost two-thirds of the reported failures were detected by routine surveillance testing indicating that such practices are effective techniques for monitoring and detecting age degradation of discrete components and systems. A substantial number of events resulted from setpoint drift.

NUREG/CR-3818, N. H. Clark and D. L. Berry, "Report of Results of Nuclear Power Plant Aging Workshop," Sandia National Laboratories, SAND84-0374, August 1984.

The objective of the workshops was to identify whether there is any evidence of component or structural time-related degradation, i.e., aging problems, in a nuclear power plant and, if so, what problems are of greatest importance. Fifteen representatives from national laboratories, architect/engineers, nuclear steam supply system vendors, research firms, and one university participated. Questionnaires and group discussions screened over 112 components believed to be susceptible to excessive aging; pressure and temperature sensors, valve operators, and snubbers emerged by consensus as the most important. Potential aging problems related to off-normal common-mode effects or problems that were just developing at the time were outside the scope of the workshops because little or no first-hand experience was available for these off-normal or yet-to-be-explored circumstances. Recommendations are made for a systematic approach to rating components in terms of overall safety and for a cooperative effort between industry research groups and regulatory research groups to resolve known aging problems and to identify off-normal or yet-to-develop aging issues. In addition to some well-known aging mechanisms (e.g., neutron embrittlement of pressure vessels) or problems that manifest themselves as equipment failures (e.g., steam generator tube degradation), there is concern that other types of aging problems may be developing. Their effects increase as nuclear power plants get older, and some aging processes could eventually affect power plant availability or safety.

NUREG/CR-3819, J. A. Rose, R. Steele, Jr., K. G. DeWall, and B. C. Cornwell, "Survey of Aged Power Plant Facilities," Idaho National Engineering Laboratory, EGG-2317, June 1985.

The survey concentrated on component failures in LWR safety-related systems as determined from operating histories. Only failures that were determined to be age related were included.

The age-related failure information gathered from the plant histories was analyzed for reoccurring failure patterns. Early program emphasis was on isolating specific equipment with high failure rates that were not already the concern of other research efforts. The resulting (gathered) data could not support the identification of specific equipment. It did, however, imply a direct relationship between the failure and the failure mechanism. Thus the emphasis of the program was redirected toward exploring the relationship of the failure to the failure mechanism.

The results of this preliminary investigation indicated that about 70% of the significant failures reported for the fluid systems analyzed were due to only four failure mechanisms (causes): erosion, corrosion, vibration, and foreign materials. This was subsequently verified by detailed study of several more plant systems and corroborated by field data obtained from personnel interviews. In addition, there appears to be a strong correlation between the cause of component failure and the system in which the component operates.

The survey points out, with evidence, that the identification and elimination of the system-level causes of component failures is a viable approach to preventing and mitigating the major reported aging effects.

NUREG/CR-3956, M. R. Dinsel, M. R. Donaldson, and F. T. Soberano, "In Situ Testing of the Shippingport Atomic Power Station Electrical Circuits," Idaho National Engineering Laboratory, EGG-2443, April 1987.

This report discusses the results of electrical in situ testing of selected circuits and components at the Shippingport Atomic Power Station in Shippingport, Pennsylvania. The goal was to determine the extent of aging or degradation of various circuits from the original plant and the two major core plant upgrades (representing a total of three distinct age groups) as well as to evaluate previously developed surveillance technology. The electrical testing was performed using the Electrical Circuit Characterization and Diagnostic (ECCAD) system developed by EG&G for the U.S. Department of Energy to use at TMI-2. Testing included measurements of voltage, effective series capacitance, effective series inductance, impedance, effective series resistance, dc resistance, insulation resistance, and time-domain reflectometry (TDR) parameters. The circuits evaluated included pressurizer heaters, control rod position indicator cables, miscellaneous primary system resistance temperature detectors (RTDs), nuclear instrumentation cables, and safety injection system motor-operated valves. It should be noted that the operability of these circuits was tested several years after plant operation was concluded at Shippingport. There was no need to retain the circuits in working condition following plant shutdown, so no effort was expended for that purpose. The in situ measurements and analysis of the data confirmed the effectiveness of the ECCAD system for detecting degradation of circuit connections and splices along the high-resistance paths; most of the problems were caused by corrosion. Results indicate a correlation between the chronological age of circuits and circuit degradation.

NUREG/CR-4144, T. Davis, A. Shafaghi, R. Kurth, and F. Leverenz, "Importance Ranking Based on Aging Consideration of Components Included in Probabilistic Risk Assessments," Pacific Northwest Laboratory, PNL-5389, April 1985.

The method outlined in the report ranks power plant components by using a risk-due-to-aging sensitivity measure that describes the change in risk due to changes in component failure rate (without describing closely the aging phenomena and the resulting time-dependent component failure rate).

The output from this study can be combined with that from other studies (data, analytical or experimental) that identify the components most susceptible to aging.

The applications use average component unavailability equations currently employed in probabilistic risk assessment (PRA) to calculate the risk-due-to-aging sensitivity. A more exact calculation is possible by using unavailability equations that include the time-dependent characteristics of component failure rates; however, these time-dependent characteristics are not well known. The risk-due-to-aging sensitivity measure presented here is therefore segregated from these time-dependent effects and addresses only the time-independent portion of aging phenomena. The results identify the components that show the highest potential for risk-due-to-aging phenomena.

Three operating NSSS were analyzed, and it was found that the most risk-significant components are in the auxiliary feedwater system, the reactor protection system, and the service water systems, e.g., pumps, check valves, motor-operated valves, circuit breakers, and actuating circuits.

Future research on the time-dependent portion of aging phenomena for these components is needed to completely describe the impact on risk.

NUREG/CR-4156, M. Subudhi, E. L. Burns, and J. H. Taylor, "Operating Experience and Aging-Seismic Assessment of Electric Motors," Brookhaven National Laboratory, BNL-NUREG-51861, June 1985.

A limited number of electric motor categories with direct safety significance were identified, and failures due to insulation degradation were surveyed.

Age-sensitive components (with respect to materials and design features) were reviewed, potential electrical and mechanical hazards were considered, operational and accident stressors were determined, and monitorable functional indicators were identified. The contribution of pertinent seismic effects was assessed, and failure modes, mechanisms, and causes were reviewed from existing data bases.

NUREG/CR-4234, W. L. Greenstreet, G. A. Murphy, and D. M. Eissenberg, "Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants," Vol. 1, Oak Ridge National Laboratory, ORNL-6170/V1, June 1985.

This report deals with motor-operated valves, focusing on monitoring defects and degradation of nuclear plant safety equipment. The contents include the evaluation and identification of practical and cost-effective methods for detecting, monitoring, and assessing the severity, failure modes, and causes (mainly aging and service wear) of time-dependent degradation in nuclear plants. Also being considered are manufacturer-recommended maintenance and surveillance practices and the selection of measurable parameters (including functional indicators) for use in assessing operational readiness, establishing degradation trends, and detecting incipient failures. The report's results are based on information derived from operating experience records, nuclear industry reports, manufacturer-supplied information, and input from architect-engineer firms and plant operators.

Failure modes are identified for both the valve and the motor-operator assembly. For each failure mode, failure causes are listed by subcomponent or subassembly, and parameters potentially useful for detecting degradation that could lead to failure are identified.

The method emerging from this analysis of the data can provide capabilities for establishing degradation trends prior to failure and developing guidance for effective and safe maintenance.

NUREG/CR-4234, H. D. Haynes, "Aging and Service Wear of Electric Motor-Operated Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants: Aging Assessments and Monitoring Method Evaluations," Vol. 2, Oak Ridge National Laboratory, ORNL-6170/V2, August 1989.

Motor-operated valves (MOVs) are located in almost all plant fluid systems. Their failures have resulted in significant plant maintenance efforts. More important, the operational readiness of nuclear plant safety-related systems has often been affected by MOV degradation and failure. Thus, in recent years, MOVs have received considerable attention by the Nuclear Regulatory Commission and the nuclear power industry and were identified as a component for study by the NRC NPAR program. In support of the NPAR Program, a comprehensive Phase II aging assessment on MOVs was performed by the Oak Ridge National Laboratory (ORNL), and the results of this study are presented in this report.

An evaluation of commercially available MOV monitoring methods was carried out, as well as an assessment of other potentially useful techniques. These assessments led to the identification of an effective, nonintrusive, and remote technique, motor current signature analysis (MCSA). The capabilities of monitoring methods (especially MCSA) for detecting changes in operating conditions and MOV degradation were investigated in controlled laboratory tests at ORNL, in situ MOV tests at a neighboring nuclear power plant, and the gate valve flow interruption blowdown test in Huntsville, Alabama.

The background information and the work leading to the selected monitoring method are summarized below. A primary objective of this study was to identify effective methods for monitoring the condition of motor-operated valves used in safety-related systems of nuclear power plants. In response to a need for improved methods for monitoring MOV condition, several systems that use a variety of sensing devices and signal-processing equipment and provide signatures that yield useful diagnostic information have been developed in the last few years. As part of the Phase II MOV study, one of the motor-operated valve analysis & test systems (MOVATS) was evaluated in depth. This evaluation and a description of four other commercially available systems are included in this report.

In addition, the type and potential value of diagnostic information from many measurable parameters were determined by ORNL tests using MOVs mounted on test stands. The selected parameters are (1) valve stem position, (2) valve stem velocity, (3) valve stem strain, (4) torque- and limit-switch actuation (times of occurrence), (5) internal and external motor temperatures, (6) vibration (several locations), (7) torque-switch angular position, and (8) motor current.

The tests led to the conclusion that the single most informative measurable parameter was also the one that was most easily acquired, i.e., the motor current. MCSA was found to provide detailed information related to the condition of the motor, motor operator, and valve across a wide range of values of parameters and their variations. The recording and the analysis can be done during valve operation to render information that characterizes transient and periodic occurrences.

Several tests were carried out to investigate the capabilities of monitoring methods (especially MCSA) for detecting changes in operating conditions and MOV degradation. Results from selected laboratory tests presented in the report illustrate examples of (1) valve stem taper, (2) stem nut wear, (3) degraded voltage, (4) degraded valve stem lubrication, (5) worm-gear tooth wear, (6) obstruction in valve seat area, (7) motor pinion disengagement, (8) degraded worm and worm-gear lubrication, (9) stem packing adjustments, and (10) torque-switch settings.

In situ signature analysis tests were carried out by ORNL on a total of 20 aged MOVs at a neighboring nuclear power plant. Five of these MOVs were later retested after they were refurbished. Selected results from these tests are presented in this report and show, for example, differences in motor current signatures of similar MOVs that were indicative of control-switch setting variations and differences in component wear. The influences of refurbishing and inactivity on MOV operations were clearly seen in motor current signatures as well.

ORNL participated in the gate valve flow interruption blowdown test program carried out under the direction of the Idaho National Engineering Laboratory at Wyle Laboratories in Huntsville, Alabama. This test was an excellent opportunity for MOV diagnostic studies and, more important, a means for determining the influences of high blowdown flow on the operation of boiling water reactor isolation valves. The reduction in operating "margin" of a MOV due to the presence of additional valve running loads was imposed by high flow. This was observed in motor current and torque-switch angular-position signatures, as illustrated in this study. In addition, the effects of differential pressure, fluid temperature, and line voltage on MOV operation were clearly seen.

The report presents information that should be useful in resolving MOV issues concerning the NRC and the nuclear industry. Important areas not covered by the Phase II work are identified, and recommendations for additional work are included.

NUREG/CR-4257, S. Ahmed, A. Carfagno, and G. J. Toman, "Inspection, Surveillance, and Monitoring of Electrical Equipment Inside Containment of Nuclear Power Plants—With Applications to Electrical Cables," Vol. 1, Oak Ridge National Laboratory, ORNL/SUB/83-28915/1, August 1985.

The purpose of this report is to describe currently available methodology for detecting and determining the amount and rate of age-related deterioration of safety-related equipment. The general concepts of monitoring equipment condition for this purpose are described. The goal is to detect deterioration in the incipient stage, prior to inservice failure and prior to the point at which equipment can no longer be expected to perform its function when exposed to design basis accident conditions.

The application of condition monitoring is discussed specifically for electric cables. The goal is to determine the degree of cable degradation and to predict the remaining useful life. In situ nondestructive testing and destructive laboratory testing are discussed as are their limitations. Interim recommendations are given for the implementation of a condition-monitoring program for cables.

NUREG/CR-4257, G. J. Toman, "Inspection, Surveillance, and Monitoring of Electrical Equipment in Nuclear Power Plants," Vol. 2, "Pressure Transmitters," Oak Ridge National Laboratory, ORNL/SUB/83-28915/3/V2, August 1986.

This report describes the types of pressure transmitters commonly used in nuclear power plants according to their application. The stresses that affect these transmitters include ambient temperature, humidity, radiation, process (fluid) medium pressure, and temperature. The most common effects of the stresses on the transmitters are calibration shifts. The evaluation of failure data contained in Licensee Event Reports indicates that total failure of pressure transmitters occurs relatively infrequently.

Comparison of as-found and as-left calibration data is described as a partial means of evaluating the level of deterioration of a transmitter. Care must be taken to ensure that variations in method or procedure do not produce erroneous data and wrong conclusions. The precision of the comparative measurements must also be high.

The evaluation of calibration data alone will not ensure the capability of operating under design basis accident conditions. If, with time, steam or moisture penetrates the transmitter housing, the transmitter electronics will become inaccurate and may fail. Therefore, the integrity of the housing seal must also be evaluated periodically to be able to predict continued performance capability.

Evaluation of inservice failures is recommended to allow further differentiation between sudden failures (having no precursor) and failures that can be detected in the incipient state. Such evaluations would aid in the further development of monitoring techniques. Because some of the transmitter failures are of the sudden type, periodic operability checks are an important means of detecting failures very soon after their occurrence so that a significant number of failed (inactive or inaccurate) transmitters do not remain undetected.

A combination of operability monitoring and condition monitoring may be used to improve the probability of successfully weathering aging processes and accident conditions.

NUREG/CR-4279, S. H. Bush, P. G. Heasler, and R. E. Dodge, "Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety-Related Piping and Components of Nuclear Power Plants," Vol. 1, Pacific Northwest Laboratory, PNL-5479, February 1986.

This report presents an overview of hydraulic and mechanical snubbers used on nuclear piping systems and components. The functions and functional requirements of snubbers are outlined. The real versus perceived need for snubbers is reviewed based primarily on studies conducted by a Pressure Vessel Research Committee. Tests conducted to qualify snubbers, to accept them on a case-by-case basis, and to establish their fitness for continued operation are reviewed.

This report had two primary purposes: (1) to assess the effects of various aging mechanisms on hydraulic and mechanical snubber operation (e.g., leaking of seals, functional failures) and (2) to determine the efficacy of existing tests in determining the effects of aging and degradation mechanisms. These tests include breakaway force, drag force, velocity/acceleration range for activation in tension or compression, release rates within specified tension/compression limits, and restricted thermal movement. The snubber operating experience was reviewed using licensee event reports and other historical data for a period of more than 10 years. Data were statistically analyzed using arbitrary snubber populations. Value-impact was considered in terms of exposure to a radioactive environment for examination/testing and in terms of the influence of lost snubber function and subsequent testing program expansion on the costs and operation of a nuclear power plant. The implications of the observed trends were assessed; recommendations include modifying or improving the examination and testing procedures to enhance snubber reliability. Optimization of snubber populations by selective removal of unnecessary snubbers was also considered.

NUREG/CR-4302, W. L. Greenstreet, G. A. Murphy, R. B. Gallaher, and D. M. Eissenberg, "Aging and Service Wear of Check Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants," Vol. 1, Oak Ridge National Laboratory, ORNL-6193/V1, December 1985.

The report addresses detecting defects and monitoring the degradation of nuclear plant safety equipment. The program is concerned with identifying and evaluating practical and cost-effective methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of check valves in nuclear plants. These methods will allow degradation trends to be detected prior to failure and allow guidance for effective maintenance to be developed.

The topics considered are failure modes and causes resulting from aging and service wear, manufacturer-recommended maintenance and surveillance practices, and measurable parameters (including functional indicators) for use in assessing operational readiness, establishing degradation trends, and detecting incipient failure. The results presented are based on information derived from operating experience records, nuclear industry reports, manufacturer-supplied information, and plant operators.

Failure modes for check valves are identified and are examined by identifying methods for detecting failures and differentiating between their causes. For each failure mode, failure causes are listed by component or subassembly, and parameters potentially useful for detecting degradation that could lead to failure are tabulated.

The report also identifies parameters potentially useful for enhancing the detection of degradation and incipient failure; these parameters include dimensions, bolt torque, noise, appearance, roughness, and cracking.

NUREG/CR-4302, M.D. Haynes, "Aging and Service Wear of Check Valves Used in Engineered Safety-Feature Systems of Nuclear Power Plants: Aging Assessments and Monitoring Method Evaluations," Vol. 2, Oak Ridge National Laboratory, ORNL-6193/V2, April 1991.

The failures of check valves have resulted in significant maintenance efforts and, on occasion, in water hammer, overpressurization of low-pressure systems, and damage to flow system components. These failures have largely been attributed to severe degradation of internal parts (e.g., hinge pins, hinge arms, disks, and disk nut pins) resulting from instability (flutter) of check valve disks under normal plant operating conditions. Present surveillance requirements for nuclear power plant check valves have been inadequate for timely detection and trending of such degradation because neither the flutter nor the resulting wear can be detected prior to valve failure. This report describes evaluations carried out in support of the NRC NPAR program of the following developmental or commercially available methods for diagnostic monitoring of check valves:

1. Acoustic emission monitoring,
2. Ultrasonic inspections,
3. Magnetic flux signature analysis,
4. Radiography,
5. Pressure noise signature analysis.

These evaluations were focused on the capability of each method to provide diagnostic information useful in determining the effects of aging and service wear (degradation) and detecting failures and undesirable operating modes. Commercial suppliers of three check valve monitoring systems recently participated in a comprehensive series of tests designed to evaluate the capability of each monitoring technology to detect the position, motion, and wear of check valve internals and valve seat leakage. This report describes these tests, which were directed by the Nuclear Industry Check Valve Group and carried out at the Utah Water Research Laboratory.

Each monitoring method is described and compared with the others, and areas in need of further development are identified. Examples of test data acquired under controlled laboratory conditions and some field test data acquired at operating nuclear plants are presented.

Of the methods examined, acoustic emission monitoring, ultrasonic inspection, and magnetic flux signature analysis provided the greatest level of diagnostic information. These three methods were shown to be useful in determining check valve condition (e.g., disk position, disk motion, and seat leakage), although none of the methods was, by itself, successful in monitoring all three condition indicators. However, the combination of acoustic emission with either ultrasonic or magnetic flux monitoring yields a monitoring system with sufficient sensitivity to detect all major check valve operating conditions. All three methods are still under development and are expected to improve as a result of further testing, analysis, and evaluation.

NUREG/CR-4380, J. L. Crowley and D. M. Eissenberg, "Evaluation of the Motor-Operated Valve Analysis and Test System (MOVATS) to Detect Degradation, Incorrect Adjustments, and Other Abnormalities in Motor-Operated Valves," Oak Ridge National Laboratory, ORNL-6226, January 1986.

An important aspect of the NPAR program strategy is to demonstrate the utility of condition monitoring, signature analysis, and other surveillance methods for detecting, differentiating, and trending various types of abnormalities in the components so that corrective measures can be implemented prior to loss of safety function. A field test program was carried out to evaluate valve signature analysis as a surveillance method to achieve these results as well as to detect incorrect adjustments in motor-operated valves. The technique specified in the title (MOVATS) is the subject of this report. In situ signature traces were obtained in 36 motor-operated valves at four nuclear plant sites. Described are the test equipment package, the method of obtaining the signatures, and determinations made as a result of analyzing them. Based on the results of the signature-analysis technique and those obtained from the field-test program, the capabilities and limitations of MOVATS are discussed.

NUREG/CR-4457, J. L. Edson and J. E. Hardin, "Aging of Class 1E Batteries in Safety Systems of Nuclear Power Plants," Idaho National Engineering Laboratory, EGG-2488, July 1987.

This report presents the results of a study of aging effects on safety-related batteries in nuclear power plants. The purpose is to evaluate the aging effects caused by battery operation in a nuclear facility and to evaluate maintenance, testing, and monitoring practices with respect to the effectiveness of these practices in detecting and mitigating the effects of aging. The study follows the NRC NPAR approach and investigates the materials used in battery construction. It also identifies stressors and aging mechanisms, presents operating and testing experience related to aging effects, analyzes battery-failure event reports in various data bases, and evaluates recommended maintenance practices. Data bases that were analyzed included the NRC's Licensee Event Report system, the Institute for Nuclear Power Operations' Nuclear Plant Reliability Data System, the Oak Ridge National Laboratory's In-Plant Reliability Data System, and the S. M. Stoller Corporation's Nuclear Power Experience data base.

NUREG/CR-4564, W. E. Gunther, M. Subudhi, and J. H. Taylor, "Operating Experience and Aging-Seismic Assessment of Battery Chargers and Inverters," Brookhaven National Laboratory, BNL-NUREG-51971, June 1986.

Battery chargers and inverters are vital components of the nuclear power plant electrical safety system. The objectives of this program are to (1) identify concerns related to the aging and service wear of equipment operating in nuclear power plants, (2) assess their possible impact on plant safety, (3) identify effective inspection, surveillance, and monitoring methods, and (4) recommend suitable maintenance practices to mitigate aging-related concerns and diminish the rate of degradation due to aging and service wear.

The designs of battery chargers (3 types) and inverters (4 types) and the materials for their construction are reviewed to identify age-sensitive components. Operational and accidental stressors are determined, and their effect on promoting aging degradation are assessed. Variations in plant electrical designs, as well as system and component impacts were studied. Failure modes, mechanisms, and causes were reviewed from operating experience and existing data banks. The study also considered seismic effects on age-degraded components of battery chargers and inverters.

The performance indicators that can be monitored to assess component deterioration due to aging or other relevant stressors are identified. Conforming with the NPAR strategy as outlined in the program plan, the study also includes a review of current standards and guides, maintenance programs, and research activities pertaining to safety-related battery chargers and inverters for nuclear power plants.

NUREG/CR-4590, K. R. Hoopingarner, J. W. Vause, D. A. Dingee, and J. F. Nesbitt, "Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience," Vols. 1 and 2, Pacific Northwest Laboratory, PNL-5832, August 1987.

Pacific Northwest Laboratory evaluated operational and expert experience pertaining to the aging degradation of diesel generators in nuclear plant service. The research identified and characterized the contribution of aging to emergency diesel generator failures.

Volume 1 reviews diesel-generator experience to identify the systems and components most subject to aging degradation and isolates the major causes of failure that may affect future operational readiness. Evaluations show that, as plants age, the percentage of aging-related failures increases and failure modes change. A compilation is presented of recommended corrective actions for the aging-related failures identified, and the trend of these failures is discussed. This study also includes a review of current relevant industry programs, research, and standards. Volume 1 presents the results of the Phase I research that identifies the components and systems most susceptible to aging degradation and the major causes of such degradation.

Volume 2 reports the results of a workshop held on May 28 and 29, 1986, with industry representatives to discuss the technical issues associated with aging of nuclear service emergency diesel generators. The technical issues discussed most extensively were man/machine interfaces, component interfaces, thermal gradients of startup and cooldown, and the need for an accurate industry data base for trend analysis of the diesel generator system.

NUREG/CR-4597, M. L. Adams and E. Makay, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants," Vol. 1, "Operating Experience and Failure Identification," Oak Ridge National Laboratory, ORNL-6282/V1, July 1986.

In this report, typical auxiliary feedwater pump features are described in terms of configuration details, materials of construction, operating requirements, and modes of operation. Failure modes and causes due to aging and service wear are identified and explained, and measurable parameters (including functional indicators) for potential use in assessing operational readiness, establishing degradation trends, and detecting incipient failures are outlined.

A series of measures to correct present deficiencies in surveillance, monitoring, and inservice testing practices is discussed. The main body of the report is supplemented by a number of relevant appendices; in particular, a major appendix is included on engineering and design information useful to assess operational readiness.

NUREG/CR-4597, D. M. Kitch, J. S. Schlonski, P. J. Sowatskey, and W. V. Cesarski, "Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants," Vol. 2, "Aging Assessments and Monitoring Method Evaluations," Oak Ridge National Laboratory, ORNL-6282/V2, June 1988.

The subjects specified in the title are described and discussed in four major sections:

1. Failure causes,
2. Description of inspection, surveillance, and condition monitoring (ISCM) methods,
3. Evaluation of ISCM methods, and
4. Role of maintenance in alleviating aging and service wear.

Failure causes attributable to aging and service wear are given and ranked in terms of importance. Cause identifications are made on the basis of experience, postservice examinations, and in situ assessments.

Measurable parameters related to failure causes are identified. ISCM methods are specified, and evaluations are made based on Westinghouse experience. On the same basis, recommendations are given on inspection, surveillance, and condition monitoring. The ISCM methods are intended to yield required capabilities for establishing operational readiness as well as for detecting and tracking degradation and its trends.

The role of maintenance in alleviating and mitigating aging and service wear effects is discussed, and the relationship of maintenance to ISCM methods is identified. Predictive, preventive, and corrective maintenance practices are discussed and evaluated.

Appendices contain a detailed discussion on ISCM methods, failure data base information, auxiliary feedwater pump (AUXFP) installation lists (location survey), a discussion of low-flow testing, auxiliary feedwater system descriptions (with flow-diagrams and schemes), AUXFP minimum-flow-rate criteria, and guidelines proposed by Westinghouse for full-flow testing. Note: The draft of this Vol. 2 (with the same title) was issued by Westinghouse Electric Corporation, Generation Technology Systems Division, in April 1986, coauthored by D. M. Kitch, M. Vuckovich, W. V. Cesarski, and P. J. Sowatskey.

NUREG/CR-4652, D. J. Naus, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants," Oak Ridge National Laboratory, ORNL/TM-10059, September 1986.

The objectives of this study are to (1) expand upon the work that was initiated in the first two Electric Power Research Institute studies relative to longevity and life extension considerations of safety-related concrete components in light-water reactor (LWR) facilities and (2) provide background that will logically lead to subsequent development of a methodology for assessing and predicting the effects of aging on the performance of concrete-based materials and components.

Applications of safety-related concrete components to LWR technology are identified, and pertinent structures (containment buildings, containment base mats, biological shield walls, main building, and auxiliary buildings) and the materials of which they are constructed (concrete, mild steel reinforcement, prestressing systems, embedments, and anchorages) are described. Historical performance of concrete components was established through information presented on concrete longevity and component behavior in both LWR and high-temperature gas-cooled reactor applications. Also, a review of problems involving concrete components in both general civil engineering and nuclear power applications is given. The majority of the problems identified in conjunction with nuclear power applications were minor; they include concrete cracking, concrete voids, or low concrete strengths at an early age. Five incidents involving LWR concrete containments that are considered significant are described in detail.

Potential environmental stressors and aging factors to which LWR safety-related components could be subjected are identified and discussed in terms of durability factors related to the materials used to fabricate the components. The current technology for detecting concrete aging phenomena is also presented in terms of methods applicable to the particular material system that could experience deteriorating effects. Remedial measures for the repair or replacement of degraded concrete components and their effectiveness are discussed. Finally, considerations relative to developing a damage methodology for assessing the durability factor, deterioration rates, and prediction of structural reliability are outlined.

Conclusions and recommendations of the report note the need for (1) obtaining aging data from decommissioned plants, (2) using inservice inspection programs to provide aging trends, (3) developing a methodology to quantitatively and uniformly (i.e., using the same procedures) assess structural reliability as affected by aging or degradation of structural materials, and (4) performing research in support of all these needs. It should be stressed that there is no widely accepted standardized methodology for quantifying the condition and capacity of an individual concrete structure.

NUREG/CR-4692, G. A. Murphy and J. W. Cletcher II, "Operating Experience Review of Failures of Power Operated Relief Valves and Block Valves in Nuclear Power Plants," Oak Ridge National Laboratory, ORNL/NOAC-233, October 1987.

This report contains a review of nuclear power plant operating events involving failures of power-operated relief valves (PORVs) and associated block valves (BVs). Of the 230 events identified, 101 involved PORV mechanical failure, 91 were attributable to PORV control failure, 6 involved design or fabrication of the PORVs, and 32 involved BV failures. The report contains a compilation of the PORV and BV failure events, including failure cause and severity. The events are identified as to plant and valve manufacturer. An assessment of the need to upgrade PORVs and BVs to safety-grade status concludes that such action would improve PORV and BV reliability. The greatest improvement in reliability would result from using newer, more reliable PORV designs and improving testing, diagnostics, and maintenance applied to PORVs and BVs, particularly to the BV motor operators. A summary of interviews conducted with four PORV manufacturers is also included in the report.

NUREG/CR-4715, G. J. Toman, V. P. Bacanskas, T. A. Shook, and C. C. Lodlow, "An Aging Assessment of Relays and Circuit Breakers and System Interactions," Brookhaven National Laboratory, Franklin Research Center, Philadelphia, PA, BNL-NUREG-52017, June 1987.

As part of the NRC NPAR program, Franklin Research Center analyzed the effects of aging on safety-related circuit breakers and relays under contract to Brookhaven National Laboratory. Circuit breakers and relays in a PWR safety injection system were evaluated with respect to the aging caused by system operation. The effect of circuit breaker and relay deterioration on the ability of the system to perform its safety functions was also evaluated. The study included protective, control, and logic relays, as well as molded-case and metal-clad switchgear circuit breakers. Analysis of nuclear power plant failure data confirmed that normally energized relays commonly used in safety systems suffer from more rapid deterioration than do deenergized relays. The failures were attributable to coil deterioration, changes in dimensions of critical organic components, and changes in characteristics of timing diaphragms from thermal deterioration. Some of the failure modes will prevent fail-safe operation. The electrical control and mechanical portions of metal-clad switchgear were found to be more failure prone than the main contacts and arc extinguishing systems. Analysis of failure data for circuit breakers and relays indicated a general trend of increasing failure rates in the period of 6 to 11 years following the start of commercial operation of the plants. The aging interaction study evaluated the interaction of aged relays and circuit breakers in a safety injection system with regard to five events requiring the system to start operation. Failure of redundant trains from common-mode failure of a particular type of circuit breaker or relay is not expected. However, the number of different types of potential failures supports the need for a strong maintenance and surveillance program to prevent multiple age-related failures from affecting redundant safety trains.

NUREG/CR-4731, V. N. Shah and P. E. MacDonald, "Residual Life Assessment of Major Light Water Reactor Components," Vol. 1, Idaho National Engineering Laboratory, EGG-2469, June 1987.

NUREG/CR-4731, V. N. Shah and P. E. MacDonald, "Residual Life Assessment of Major Light Water Reactor Components—Overview," Vol. 2, Idaho National Engineering Laboratory, EGG-2469, November 1989.

This report presents an assessment of the aging (time-dependent degradation) of selected major light water reactor components and structures. The stressors, possible degradation sites and mechanisms, potential failure modes and currently used nondestructive examinations, inservice inspection (ISI), and life assessment methods are discussed for major light water reactor components. Volume 1 covers PWR and BWR pressure vessels, PWR containment structures, PWR reactor coolant piping, PWR steam generators, BWR recirculation piping, and reactor pressure vessel supports. Volume 2 covers PWR reactor coolant pumps, PWR pressurizers, PWR pressurizer surge and spray lines, PWR reactor coolant system charging and safety injection nozzles, PWR feedwater lines, PWR control rod drive mechanisms and reactor internals, BWR containments, BWR feedwater and main steam lines, BWR control rod drive mechanisms and reactor internals, PWR and BWR electrical cables and connections, and PWR and BWR emergency diesel generators. Unresolved technical issues related to understanding and managing the aging of these major components, including requirements for advanced ISI and life assessment methods, are also discussed.

NUREG/CR-4740, L. C. Meyer, "Nuclear Plant-Aging Research on Reactor Protection Systems," Idaho National Engineering Laboratory, EGG-2467, January 1988.

This report presents the results of a review of operating experience for the reactor trip system (RTS) and the engineered safety feature actuating system (ESFAS) reported in Licensee Event Reports (LERs), the Nuclear Power Experience data base, the Nuclear Plant Reliability Data System, and plant maintenance records. The purpose of the review was to evaluate the potential significance of aging, including cycling, trips, and testing, as a contributor to degradation of the RTS and ESFAS. Tables show the percentage of events for RTS and ESFAS classified by cause, components, and subcomponents for each of the nuclear steam supply system vendors. A representative Babcock and Wilcox plant was selected for detailed study. The NRC NPAR guidelines were followed in performing the detailed study that identified materials susceptible to aging, stressors, environmental factors, and failure modes for the RTS and ESFAS and the relevant generic instrumentation and control systems. Functional indicators of degradation are listed, testing requirements evaluated, and regulatory issues discussed.

NUREG/CR-4747, B. M. Meale and D. G. Satterwhite, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," Vol. 1, Idaho National Engineering Laboratory, EGG-2473, July 1987.

NUREG/CR-4747, B. M. Meale and D. G. Satterwhite, "An Aging Failure Survey of Light Water Reactor Safety Systems and Components," Vol. 2, Idaho National Engineering Laboratory, EGG-2473, July 1988.

This report describes the methods, analyses, results, and conclusions of two different aging studies. The first study was a survey of light water reactor component failures associated with 15 selected safety and support systems. Analysts used computerized sorting techniques to classify component failures into generic failure categories. The second study was a careful examination of component failure records to identify and categorize the reported causes of component failures. The systems evaluated in the failure-cause analysis were the auxiliary feedwater, Class 1E electric power distribution, high-pressure injection, and service water. Tables and figures indicate the systems and the components within the systems that are most affected by aging. Engineering insights drawn from the data are provided. Volume 2 presents all of the Volume 1 data from FY-86 combined with the data gathered in FY-87.

NUREG/CR-4769, W. E. Vesely, "Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions," Idaho National Engineering Laboratory, EGG-2746, April 1987.

A model for failure rates of light water reactor safety system components due to aging mechanisms has been developed from basic phenomenological considerations. In the treatment, the occurrences of deterioration are modeled as following a Poisson probability process. The severity of damage is allowed to have any distribution; however, the damage is assumed to accumulate independently. Finally, the failure rate is modeled as being proportional to the accumulated damage. Using this treatment, the linear aging-failure-rate model is obtained. The applicability of the linear aging model to various mechanisms is discussed. The model is also extended to cover nonlinear and dependent aging phenomena. The implementation of the linear aging model is demonstrated by applying it to the aging data collected in the NRC NPAR program.

NUREG/CR-4819, V. P. Bacanskas, G. C. Roberts, and G. J. Toman, "Aging and Service Wear of Solenoid-Operated Valves Used in Safety Systems of Nuclear Power Plants," Vol. 1, "Operating Experience and Failure Identification," Oak Ridge National Laboratory, ORNL/SUB/83-28915/4/V1, March 1987.

An assessment of the types and uses of solenoid-operated valves (SOVs) in nuclear power plant safety-related service is provided in the report. Through a description of the operation of each SOV combined with knowledge of nuclear power plant applications and operational occurrences, the significant stressors responsible for degradation of SOV performance are identified. A review of actual operating experience (including failure data) leads to the identification of potential nondestructive in situ testing which, if properly developed, could provide the methodology for monitoring the degradation of SOVs. Recommendations are outlined for continuing the study into the test methodology development phase.

NUREG/CR-4819, R. C. Kryter, "Aging and Service Wear of Solenoid-Operated Valves Used in Safety Systems of Nuclear Power Plants," Vol. 2, "Evaluation of Monitoring Methods," Oak Ridge National Laboratory, ORNL/TM-12038, July 1992.

Solenoid-operated valves (SOV) were studied at Oak Ridge National Laboratory as part of the USNRC Nuclear Plant Aging Research (NPAR) Program. The primary objective of the study was to identify, evaluate, and recommend methods for inspection, surveillance, monitoring, and maintenance of SOVs that can help ensure their operational readiness, that is, their ability to perform required safety functions under all anticipated operating conditions. The failure of one of these small and relatively inexpensive devices could have serious consequences under certain circumstances.

An earlier (Phase I) NPAR program study described SOV failure modes and causes and identified measurable parameters thought to be linked to the progression of ever-present degradation mechanisms that may ultimately result in functional failure of the valve. Using this earlier work as a guide, the present (Phase II) study focused on devising and then demonstrating the effectiveness of techniques and equipment with which to measure performance parameters that show promise for detecting the presence and trending the progress of such degradations before they reach a critical stage.

Intrusive techniques requiring the addition of magnetic or acoustic sensors or the application of special test signals were investigated briefly, but major emphasis was placed on the examination of nonintrusive, condition-indicating techniques that can be applied with minimal cost and impact on plant operation. These include monitoring coil mean temperature remotely by means of coil dc resistance or ac impedance, verifying unrestricted SOV plunger movement by measuring current and voltage at their critical bistable (pull-in and drop-out) values, and detecting the presence of shorted turns or insulation breakdown within the solenoid coil using interrupted-current test methods. The first of these techniques, though perhaps the simplest conceptually, will likely benefit the nuclear industry most because SOVs have a history of failure in service as a result of unwitting operation at excessive temperatures.

Experimental results are presented that demonstrate the technical feasibility and practicality of the monitoring techniques assessed in the study, and recommendations for further work are provided.

NUREG/CR-4928, H. M. Hashemian, K. M. Petersen, T. W. Kerlin, R. L. Anderson and K. E. Holbert, "Degradation of Nuclear Plant Temperature Sensors," Analysis and Measurement Services Corporation, Knoxville, TN, June 1987.

A program was established and initial tests were performed to evaluate long-term performance of resistance temperature detectors (RTDs) of the type used in U.S. nuclear power plants. This report addresses the effect of aging on RTD calibration accuracy and response time. The Phase I effort (lasting about 6 months) included exposure of 13 safety-grade RTD elements to simulated LWR temperature regimes. Full calibrations were performed initially and monthly, sensors were monitored and cross-checked continuously during exposure, and response time tests were performed before and after exposure. Short-term calibration drifts of as much as 1.8°F (1°C) were observed. Another result was that small response times were essentially unaffected by the testing performed.

This program has demonstrated that there is a sound reason for concern about the accuracy of temperature measurements in nuclear power plants. These limited tests should be expanded in a Phase II program to involve more sensors and longer exposures to simulated LWR conditions in order to obtain statistically significant data. Such data are needed to establish the length of meaningful testing or replacement intervals for safety-grade RTDs. An important corollary benefit from this expanded program would be a better determination of achievable accuracies in RTD calibration.

NUREG/CR-4939, M. Subudhi, W. E. Gunther, J. H. Taylor, R. Lofaro, K. M. Skreiner, A. C. Sugarman, and M. W. Sheets, "Improving Motor Reliability in Nuclear Power Plants": Volume 1, "Performance Evaluation and Maintenance Practices"; Volume 2, "Functional Indicator Tests on a Small Electric Motor Subjected to Accelerated Aging"; Volume 3, "Failure Analysis and Diagnostic Tests on a Naturally Aged Electric Motor," Brookhaven National Laboratory, BNL-NUREG-52031, November 1987.

Volume 1: Performance Evaluation and Maintenance Practices

This report presents recommendations for developing a cost-effective program for performance evaluation and maintenance of electric motors in nuclear power plants. These recommendations are based on current industry practices, available techniques for monitoring degradation in motor components, manufacturers' recommendations, operating experience, and results from two laboratory tests on aged motors. The test results (on a small and a large motor) provide the basis for recommending the various functional indicators for maintenance programs.

The overall preventive program is separated into two broad areas of activity aimed at mitigating the potential effects of equipment aging: performance evaluation and equipment maintenance. The latter involves actually maintaining the condition of the equipment, while the former involves monitoring degradation due to aging. The monitoring methods are further categorized as periodic testing, surveillance testing, continuous monitoring, and inspections.

This study focuses on relevant methods and procedures with the goal of maintaining the motors in a nuclear facility operationally ready. This includes an evaluation of various functional indicators to determine their suitability for trending assessments when monitoring the condition of motor components. The intrusiveness of test methods and the present state of the art for using the test equipment in a plant environment are discussed.

Implementation of the information provided in this report will improve motor reliability in nuclear power plants. The study indicates the kinds of tests to conduct, how and when to conduct them, and to which motors the tests should be applied.

Volume 2: Functional Indicator Tests on a Small Electric Motor Subjected to Accelerated Aging

A 10-horsepower electric motor was artificially aged by plug reverse cycling for test purposes. The motor was manufactured in 1967 and was in service at a commercial nuclear power plant for twelve years. Various tests were performed on the motor throughout the aging process. The motor failed after 3.79 million reversals (3 seconds per reversal) over seven months of testing. Each test parameter was trended to assess its suitability in monitoring aging and service wear degradation in motors. Results and conclusions are discussed relative to the applicability of the tests performed to motor maintenance programs of nuclear power plants.

Volume 3: Failure Analysis and Diagnostic Tests on a Naturally Aged Large Electric Motor

Stator coils of a naturally failed 400-hp motor from the Brookhaven National Laboratory test reactor facility were tested for their dielectric integrities. The motor was used to drive the primary reactor coolant pump for the last 20 years. Maintenance activities on this motor during its entire service life were minimal, with the exception of meggering it periodically. The stator consisted of ninety individual coils, which were separated for testing. Seven different dielectric tests were performed on the coils. Each set of data from the tested coils indicated a spectrum of variation depending on their aging conditions and characteristics. By comparing the test data to baseline data, the test methods were assessed for application to motor mainte-nance programs in nuclear power plants. Also included in this study are results of an investigation to determine the cause of this motor's failure. The aged condition of a second identical primary pump motor, which is of the same age and is presently in operation, is discussed. Recommendations relating to the applicability of each of the dielectric test methods to motor maintenance programs are formulated.

NUREG/CR-4967, L. C. Meyer, "Nuclear Plant Aging Research on High Pressure Injection Systems," Idaho National Engineering Laboratory, EGG-2514, August 1989.

This report presents the results of a review of light water reactor high-pressure injection system (HPIS) operating experience reported in the Nuclear Power Experience Data Base, Licensee Event Reports (LERs), the Nuclear Plant Reliability Data System, and plant records.

Operating experience of nuclear power plants was evaluated to determine the significance of aging-related service wear on equipment and its possible impact on safety. The HPIS and those portions of related systems needed for operation of the HPIS were selected for detailed study in order to evaluate the potential significance of aging as a contributor to the degradation of that system. Tables show the percentage of significant events for HPIS classified by cause, component, and subcomponent for PWRs and BWRs. A representative Babcock and Wilcox plant was selected for detailed study.

The NPAR guidelines provided the framework through which the effect of aging on HPIS was studied, and these guidelines were followed throughout the report, which presents an identification of failure modes, a preliminary identification of failure causes due to aging and service wear degradation, and a review of current inspection, surveillance, and monitoring methods, including manufacturer-recommended surveillance and maintenance practices. The detailed study identifies materials susceptible to aging, various stressors, and environmental factors. Performance parameters or functional indicators potentially useful in detecting degradation are also identified, and preliminary recommendations are made regarding inspection, surveillance, and monitoring methods.

In addition to the above engineering evaluation, the components that contributed to system unavailability were determined, and the contribution of aging to HPIS unavailability was evaluated. The unavailability assessment utilized an existing probabilistic risk assessment, the linear aging model, and generic failure data.

NUREG/CR-4977, R. Steele, Jr. and J. G. Arendts, "SHAG Test Series: Seismic Research on an Aged United States Gate Valve and on a Piping System in the Decommissioned Heissdampfreaktor (HDR): Summary," Vol. 1, Idaho National Engineering Laboratory, EGG-2505, August 1989.

NUREG/CR-4977, R. Steele, Jr. and J. G. Arendts, "SHAG Test Series: Seismic Research on an Aged United States Gate Valve and on a Piping System in the Decommissioned Heissdampfreaktor (HDR): Appendices," Vol. 2, Idaho National Engineering Laboratory, EGG-2505, August 1989.

This report describes the investigation, results, and conclusions of the INEL effort to determine the cause of the reduced performance of a naturally aged Crane gate valve with a Limitorque motor operator. The motor-operated valve served 25 years in the Shippingport Atomic Power Station as a feedwater isolation valve before being refurbished and installed in a piping system in the Heissdampfreaktor (HDR), where valve operability in typical pressure and temperature environments and during simulated earthquakes was studied. During the test program it was discovered that, under some hydraulic loadings, the motor operator failed to reach torque levels high enough to open the closing torque switch. Failure of the torque switch to open caused the motor to stall. In normal plant service, stalling an operator motor can cause motor burnout and render the valve inoperable for subsequent safety functions.

An extensive investigation was conducted to try to isolate the cause of the poor performance of the motor operator. This investigation included follow-on in situ tests at HDR, dynamometer testing of the motor operator at the Limitorque laboratory, testing of the torque spring at INEL, dynamometer testing of the motor alone at the Peerless Motor laboratory, and a mathematical analysis of the HDR power circuit. The investigation identified three causes of the motor-operator's poor performance: torque spring aging, heating of the motor windings, and resistance in the dc power cabling at HDR. The investigation also demonstrated that normal plant testing of valves is not adequate to ensure proper performance under flow and pressure loads in combination.

During the follow-on tests at HDR, we found that, when the valve was subjected to flow loads and pressure loads in combination, the valve either torqued out in the partially open position, stalled in the partially open position, or stalled in the fully closed position, depending on the load and the torque switch setting. The valve torqued out in the fully closed position only when pressure and flow loads were very low.

Undersized power supply cabling resulting in high resistance has surfaced as a problem in at least two dc motor operators in the field. Though the other factors contributed to the anomalous performance of the valve at HDR, undersized cabling was the main cause. The NRC has recently issued an information notice regarding the issue.

None of the three problems discovered during the HDR tests and follow-on investigation would be detected during the normal in-plant testing where the valves are subjected to no load or to pressure loads alone. The problems are detectable only at higher loadings, that is, flow loads in combination with pressure loads, where the load slows the motor down to the extent that momentum cannot carry the unit through complete closure and torqueout.

NUREG/CR-4985, M. Subudhi, J. H. Taylor, J. Clinton, C. J. Czajkowski, and J. Weeks, "Indian Point 2 Reactor Coolant Pump Seal Evaluations," Brookhaven National Laboratory, BNL-NUREG-52095, August 1987.

This report summarizes the findings on Westinghouse reactor coolant pump (RCP) seal performance at Indian Point 2. This study considered a significant number of RCP seal failures. Consolidated Edison initiated a research effort to determine the causes of these failures and to develop appropriate ameliorative action to enhance seal reliability. The BNL work is an outgrowth of the first-phase effort performed by Failure Analysis Associates. The objectives of the BNL program are to determine the root causes of seal failure and to provide recommendations for improving seal reliability. This program made notable advances in understanding the root causes of RCP seal failure. For the first time, actual failed seals were examined in detail in BNL's hot cell, and laboratory tests were conducted to determine failure causes. This report summarizes findings and presents conclusions and recommendations based on review of plant operating and maintenance data, consultation with Westinghouse and utilities, review of prior RCP seal studies (including previous BNL work), and visual and in-depth examinations of the first batch of service-exposed seals received from the plant.

NUREG/CR-4992, G. C. Roberts, V. P. Bacanskas, and G. J. Toman, "Aging and Service Wear of Multistage Switches Used in Safety Systems of Nuclear Power Plants," Vol. 1, Oak Ridge National Laboratory, ORNL/SUB/83-28915/5/V1, September 1987.

An assessment of the types and uses of multistage switches in nuclear power plant safety-related service is provided. Through a description of the operation of each type of switch combined with knowledge of nuclear power plant applications and operational occurrences, the significant stressors responsible for multistage switch deterioration are identified. A review of operating experience (failure data) leads to identification of potential and recommended monitoring techniques for early detection of incipient failures. Although the operating experience does not justify extensive deterioration monitoring of multistage switches, nondestructive testing methods that could be used to evaluate the condition of switches are identified. The report presents a detailed description of the components, materials of construction, and operation of each of the multistage switches included in the assessment. Also, it provides an analysis of failure data from the LER system. An analysis of the various failure modes of multistage rotary switches and their related causes is also given. The existing recommended and required maintenance and surveillance practices are listed. Several techniques with a potential for assessing the condition of switch components and possibly predicting age-related failures are identified. It is recommended that inservice failures be analyzed to determine whether the failures are due to random defects or are the result of generic deficiencies that would require corrective action.

NUREG/CR-5008, R. D. Meininger and T. J. Weir, "Development of a Testing and Analysis Methodology to Determine the Functional Condition of Solenoid Operated Valves," Pentek, Inc., Coraopolis, PA, September 1987.

The objective of this research was to develop a simple, reliable, condition-monitoring system that will provide surveillance information without requiring disconnection or disassembly of solenoid-operated valves (SOVs) installed in operating nuclear power plants. The information provided must be sufficiently reliable to allow plant operators to conclude that valve performance has or has not degraded to the point where corrective maintenance becomes necessary.

The required information is assumed to be obtainable through analysis of in-rush current to the coil of the SOV. Various SOVs were tested in an experimental air system set up in the laboratory. In-rush current data acquired on degraded and new SOVs were analyzed to determine behavior signature models.

Laboratory conditions provided the opportunity to simulate perturbations caused by the valve function, which would differ from actuation to actuation. A visual examination of this time-varying waveform revealed distinct and repeatable variations for different valve anomalies.

This technique could identify gross changes and render characteristic signatures that could be used for various comparisons and to trend valve degradation mechanisms and their consequences over time.

Utilization of the laboratory technique in an operating nuclear power plant would be somewhat impractical since the installed valves are not equipped with synchronous switching capability. Analytical research was therefore conducted to develop a technique to analyze similar electrical data obtained under asynchronous conditions typical of an operating plant. For such field application, the technique developed would use a clip-on current probe, thus enabling all measurements to be made from outside the reactor building without disturbing any electrical connections. The in-rush current to the solenoid-operated valve is analyzed in real time using a personal computer and fast Fourier transform techniques.

NUREG/CR-5051, W. E. Gunther, R. Lewis, and M. Subudhi, "Detecting and Mitigating Battery Charger and Inverter Aging," Brookhaven National Laboratory, BNL-NUREG-52108, August 1988.

This report is the second on the two-step approach for assessing the safety and operational aspects of battery charger and inverter aging in nuclear power plants. Analyses include an assessment of the recent operating experiences with battery chargers and inverters and a discussion of improvements in reliability that may be achieved through modification of the equipment's configuration and an increased inspection frequency. The results are evaluated from a survey of the current maintenance and test practices used in nuclear power plants, along with the manufacturer's recommendations for maintaining equipment operability. Advanced designs for uninterruptible power systems, subcomponent improvements, and current monitoring and protective equipment are described and related to their potential applicability in nuclear power plants.

A naturally aged inverter and battery charger were tested at BNL to evaluate the naturally aged condition, the effectiveness of condition monitoring techniques, and the practicality of selected maintenance and monitoring procedures. A portion of this research effort is covered in RIL No. 159, "Nuclear Plant Aging Research: Safety-Related Inverters," November 9, 1988.

A maintenance program for battery chargers and inverters is recommended. As described in this report, such a program incorporates inspection, monitoring, testing, and repair activities that should be performed to detect and mitigate aging effects and thereby ensure the operational readiness of this important equipment throughout the plant's operating life.

NUREG/CR-5052, J. C. Higgins, R. Lofaro, M. Subudhi, R. Fullwood, and J. H. Taylor, "Operating Experience and Aging Assessment of Component Cooling Water Systems in Pressurized Water Reactors," Brookhaven National Laboratory, BNL-NUREG-52117, July 1988.

An aging assessment of component cooling water (CCW) systems in PWRs was performed as part of the NPAR program. The objectives were to provide a technical basis for the identification and evaluation of degradation caused by age. The information generated will be used to assess the impact of aging on plant safety and to develop effective mitigating actions for the CCW system. The effect of time on this system was characterized by using the "Aging and Life Extension Assessment Program (ALEAP) Systems Level Plan", developed by Brookhaven National Laboratory. Failure data from various national data bases were reviewed and analyzed to identify predominant failure modes, causes, and mechanisms in CCW systems. Time-dependent failure rates for major components were calculated to identify aging trends. Plant-specific data were obtained and evaluated to supplement data base results.

A computer program (PRAAGE) was developed and implemented to model a typical CCW system design and perform probabilistic risk assessment (PRA) calculations. Time-dependent failure rates were input to the program to evaluate the effects of aging on the importance of a component with respect to system unavailability. Time-dependent changes in component importance and system unavailability with age were observed and discussed.

NUREG/CR-5053, W. Shier and M. Subudhi, "Operating Experience and Aging Assessment of Motor Control Centers," Brookhaven National Laboratory, BNL-NUREG-52118, July 1988.

As part of the NRC NPAR program, an assessment was made of the characteristics of aging and service wear of motor control centers (MCCs). MCCs perform an important function in the operation and control of a large number of safety-related motors; thus the operability and reliability of MCCs can affect the overall safety of nuclear plants.

This report follows the NPAR strategy and investigates the operational performance, the design and manufacturing methods, and the current maintenance, surveillance, and monitoring techniques applied to MCCs. A significant result described in this report concerns the identification of important MCC failure modes, causes, and mechanisms from plant operational experience. Frequencies of failures determined for the various subcomponents of MCCs are also described. In addition, recommendations are provided for functional indicators to monitor the performance of MCCs. These functional indicators will be evaluated during Phase 2 of the program.

NUREG/CR-5057, K. R. Hoopingarner and F. R. Zaloudek, "Aging Mitigation and Improved Programs for Nuclear Service Diesel Generators," Pacific Northwest Laboratory, PNL-6397, December 1989.

The study of diesel generator aging for the NRC NPAR program was performed in two phases. In Phase I, plant operating experience and data were used to produce a new data base related to aging, reliability, and operational readiness of nuclear service diesel generators. Phase II is chiefly concerned with measures for mitigating the effects of aging.

This report proposes a detailed management, testing, and maintenance program for emergency diesel generators based on studies and research developed in Phase II of this effort. The proposed program would lead to three expected results: (1) reduction of several of the stressors identified in Phase I that have been shown to accelerate aging of diesel generators, (2) an improved reliability and state of operational readiness, and (3) an increased confidence in the future availability and reliability of diesel generators. The proposed new program would integrate testing, inspection, monitoring, trending, maintenance, and other elements for a better approach to mitigating diesel generator aging. The more important elements of the new proposed program are summarized in the following paragraphs.

The current fast starting and loading requirement for testing diesel generators can produce substantial harm and significant aging effects through the production of large mechanical and thermal stresses, inadequate lubrication during initial acceleration, high rotating and sliding pressures, overspeeding, etc. An improved testing program including slow starting and loading would induce fewer aging effects in the emergency diesel generator by largely eliminating a unique aging stressor. In the course of the monthly testing program, adequate data should be collected for about 30 engine operating parameters discussed in this report that could indicate degrading performance or an impending component failure. For many important components, the implementation of such a program could detect approaching performance failure and allow orderly repair. Monitoring and trending will not be able to detect all components with degraded performance, but the deterioration that will be detected by the recommended tests is significant to aging and reliability concerns. Condition monitoring and trending can provide important indications of possible long-term component or system degradation. This activity should detect many potential component/system failures before the system actually fails. Cost and safety benefits would accrue from avoiding both equipment damage and unscheduled downtime by anticipating these failures and providing timely repair/maintenance. The monthly test program should ensure that the operating parameters listed in the report are within their maximum and minimum limits as applicable. However, it is not necessary to trend every parameter for effective results. When a limiting (maximum or minimum) value is being approached, the utilities should trend the approach to avoid failures and schedule repair before limits are exceeded.

Several recommendations were developed regarding maintenance procedures and training. One important recommendation is that teardown of the diesel engines solely for the purpose of inspection should be avoided unless there is a definite indication that operation is degraded or there is an impending component failure based on performance data trends. The current practical periodic intrusive maintenance and engine overhauls has been found to be less favorable for ensuring safety than engine overhauls based on monitoring and trending results or on a need to correct specific engine defects. Therefore, this report recommends that the periodic overhaul requirements be reevaluated. Further, an understanding of the governor, as well as of the engine/generator, must be developed by providing the maintenance staff with adequate training and motivation. Finally, this report recommends that engine inspections and preventive maintenance be increased to mitigate the aging and wear results of the vibration stressor, focusing on the engine and instrumentation mounted on the engine. Vibration cannot be eliminated, but its effects can be mitigated by keeping fasteners/fittings tight and by frequently recalibrating instrumentation subject to this vibration.

The mission profile for the diesel generator is based on a large-break LOCA with loss of all offsite power. With over 1000 reactor-years of operation in U.S. regulatory history without a large-break LOCA, it may be appropriate to redefine the mission profile for the diesel-generator with consequent benefits. For a loss of offsite power, with or without a small-break LOCA event, the needs for emergency electric power and the diesel mission profile are much less stringent. In this case, the need for power can be delayed and the emergency power needs are reduced, but the need for emergency power may remain for several (3 to 4) days. The prevention of station blackout appears to be the most realistic mission envelope. The technical requirements for the diesel generator are very high reliability with the durability to produce power until the emergency passes and the reactor cooling requirements drop off.

Acceptance of this mission envelope for the diesel-generator system would result in a reduction of the aging degradation of many important engine components through less harmful test requirements. In summary, a more practical mission envelope for the diesel generator system would include an increased start and load time (within 5 minutes), with the power level reduced below the calculated full load (core and containment sprays not needed). From an overall mission standpoint, it appears that safety concerns are better served by testing the engines for reliability rather than for maximum starting accelerations and very rapid loading, which do not seem necessary.

This portion of the NPAR study was initiated to develop for NRC consideration information on potential safety problems related to the aging of diesel generators. General applications of the study results were expected for (1) improvement of diesel reliability, (2) modification of plant technical specifications, (3) improvement in the application of resources by the NRC and the utilities, and (4) development of specific research information needed to change some regulatory requirements. All of these end uses of the research have been accomplished or are under active consideration. Collectively, the safety implications of these changes and research recommendations are important.

NUREG/CR-5141, V. P. Bacanskas, G. J. Toman, and S. P. Carfagno, "Aging and Qualification Research on Solenoid Operated Valves," Franklin Research Center, Norristown, PA, August 1988.

Tests were conducted on three-way direct-acting solenoid-operated valves (SOVs). Some SOVs had been aged naturally through service in nuclear power plants, and others were subjected to accelerated aging. Thermal aging was conducted with both air and nitrogen as the process gas. Operational aging was simulated by putting the specimens through operational cycles at certain intervals during the accelerated thermal aging with the environmental temperature controlled at a level representative of service conditions. The program also included simulation of a design basis event (DBE) that consisted of gamma irradiation and a main-steam-line-break loss-of-coolant accident (MSLB/LOCA) simulation. After each major segment of the test program (aging, irradiation, and MSLB/LOCA simulation), some of the valve specimens were subjected to operational testing and then disassembled for inspection and measurement of physical properties.

Performance of the Automatic Switch Co. (ASCO) SOVs was affected in the early stages of the program by an organic deposit of undetermined origin. Removal of the deposit eliminated the problem.

A naturally aged ASCO SOV with Buna N seals and a new ASCO SOV with EPDM seals were subjected to accelerated aging with nitrogen as the process gas. These valves were the only ones to go through the entire test program without a failure to transfer and without any significant leakage.

Valcor Engineering Co. SOVs suffered from sticking of the shaft seal O-rings, which made it impossible to complete the accelerated thermal aging. Repeated tests and changes in test procedures failed to alter this situation.

It is possible that the stresses of accelerated aging produced effects that are not representative of service aging. Seal deterioration in the Valcor SOVs caused leakage following DBE irradiation. The naturally aged Valcor SOV performed satisfactorily during the first high-temperature portion of the MSLB/LOCA profile but malfunctioned during most of the rest of the test.

Deterioration of the elastomeric parts of the ASCO SOVs did not appear to be sufficient to account for the observed failures to transfer, which evidently were caused by coil deterioration. Elastomeric parts of Valcor SOVs, both from the naturally aged SOV and from the one that had not been aged, experienced substantial deterioration.

NUREG/CR-5159, M. S. Kalsi, C. L. Horst, and J. K. Wang, "Prediction of Check Valve Performance and Degradation in Nuclear Power Plant Systems," Kalsi Engineering, Inc., Sugar Land, TX, KEI No. 1559, May 1988.

Degradation and failure of swing check valves and resulting damage to plant equipment has led to a need to develop a method to predict performance and degradation of these valves in nuclear power plant systems. This Phase I investigation developed methods that can be used to predict the stability of the check valve disk when there are flow disturbances such as elbows, reducers, and generalized turbulence sources within 10 pipe diameters upstream of the valve. Major findings include the flow velocity required to achieve a full-open stable disk position, the magnitude of disk motion developed with these upstream disturbances (with flow velocities below full-open conditions), and disk natural frequency data that can be used to predict wear and fatigue damage. Reducers were found to cause little or no performance degradation. Effects of elbows located within 5 diameters of the check valve must be considered, while severe turbulence sources have a significant effect at distances up to 10 diameters upstream of the valve.

Clearway swing check designs were found to be particularly sensitive to manufacturing tolerances and installation variables making them likely candidates for premature failure. Reducing the disk full-opening angle on these designs results in significant performance improvement.

NUREG/CR-5181, L. C. Meyer and J. L. Edson, "Nuclear Plant Aging Research: The 1E Power System," Idaho National Engineering Laboratory, EGG-2545, May 1990.

This in-depth engineering study of the Class 1E Power System is conducted in accordance with the NRC NPAR program and guidelines. The report provides (1) an identification of failure modes, (2) a preliminary identification of failure causes due to aging and service wear degradation, and (3) a review of current inspection, surveillance, and monitoring methods, including manufacturer-recommended surveillance and maintenance practices. Also, performance parameters potentially useful in detecting degradation are identified in this report, and preliminary recommendations are made regarding inspection, surveillance, and monitoring methods.

A description of a typical Class 1E power system is presented for a pressurized water reactor (PWR) with specific maintenance information from a cooperating utility. The Class 1E power systems provide electric power for the safety systems in the plant, including an emergency power source (usually diesel generators) and three subsystems: the alternating current (ac) power systems, the direct current (dc) power system, and the vital ac power system. Each of the major Class

1E power components is described, and the results of component aging studies are summarized where applicable. The ac power system used in typical nuclear power plants is a dual-train cascading bus system that includes circuit breakers, transformers, relays, load centers, and motor control center switch gear. The dc system includes battery chargers, batteries, inverters, and associated control breakers. The vital 120-Vac loads include the engineered safety feature cabinets and the reactor protection systems.

The review of operating experience included data from the following generic data bases: Licensee Event Reports (LERs), Nuclear Plant Reliability Data System (NPRDS), Nuclear Power Experience (NPE), and plant maintenance data from one cooperating utility. The LER records indicate that the Class 1E power subsystem failures were distributed as follows: emergency power generation, 31.7%; medium-voltage subsystems, 21.2%; low-voltage ac (less than 600 V), 19.8%; and dc system, 9.8%. The most frequent component failures were circuit breakers, 66.3%; inverters, 9.9%; and batteries, 9.5%. The leading causes of circuit breaker faults were mechanical malfunction, 25%; electrical malfunction, 22%; and sticking, 7%. The three leading causes of relay faults were drift, 46%; electrical malfunction, 11%; and sticking, 10%. The NPRDS data review listed the Class 1E power components in order of frequency of failure as follows: diesel engines, inverters, and circuit breakers. The overall fraction of Class 1E electrical component failures related to aging was 32.7%. However, because of system redundancy and fail-safe design, only 2.4% of Class 1E electrical component failures caused total loss of system function.

Approximately 8% of all events in the NPE data base for all systems were associated with the safety electrical system. The NPE listed breakers, motor control centers, and switchgear as having the most frequent failures, at 36.1%. This was followed by inverters and chargers, 15%; diesel generators, 10%; transformers, 3.4%; and batteries, 3%. The plant data also showed that breakers caused the most work requests in the maintenance data base, followed by batteries, the battery charger, and the generator. (Batteries were high on the list because of frequent preventive maintenance tasks such as adding water and testing.)

The review of codes and standards included general design criteria, regulatory guides, and IEEE standards. There were three recommendations for regulatory guides: (1) Regulatory Guide 1.118 should include the issues of testing and inspection for the lightning protection system and power ground system, (2) Regulatory Guide 1.32 should address the issues of cleanliness in switchgear area, and (3) Regulatory Guide 1.9 should be extended to include the problems of diesel generator aging.

Approximately 40 IEEE standards applicable to Class 1E power systems and associated components were reviewed and tabulated. The IEEE reviews each standard approximately every 5 years. The authors recommend that aging be included in this review. Standards provide design and application guidance but generally do not provide specific recommendations for maintenance, testing, inservice inspection, and monitoring of age-related degradation.

Aging research can play a supporting role in solving outstanding safety issues. For example, component degradation due to aging is one factor to consider in the plant coping analysis required by the NRC rule on station blackout.

NUREG/CR-5192, W. E. Gunther, "Testing of a Naturally Aged Nuclear Power Plant Inverter and Battery Charger," Brookhaven National Laboratory, BNL-NUREG-52158, September 1988.

A naturally aged inverter and battery charger obtained from the Shippingport facility were tested as part of the NPAR program. The objectives of this testing were to evaluate the naturally aged equipment state, determine the effectiveness of condition-monitoring recommendations, and obtain insight into the practicality of preventive maintenance and monitoring methods.

Testing indicates that the equipment has retained its ability to respond to load transients. With the exception of silicon controlled rectifiers (SCRs), which were found to be operating with case temperatures (°F) 20% higher than those during the acceptance test, component temperatures and circuit characteristics were similar to original acceptance test measurements. Based on these observations, it is concluded that the inverter and battery charger have not aged substantially.

The two primary monitoring techniques employed were temperature measurements and electrical waveform observation. Internal panel temperature and individual component temperatures were recorded at regular intervals during steady-state and transient operations. Thermocouples imbedded within the transformer and inductor windings and attached to SCR and capacitor surfaces provided a nonobtrusive means of monitoring component operation. Readings taken were compared to original acceptance test data.

Circuit waveforms were observed on an hourly basis during steady-state operation and at the time load transients were applied. The inverter output voltage and the SCR gate current waveforms remained relatively constant regardless of the applied loads.

Finally, this test report recommends that individual fusing of filter capacitors be considered in order to preclude a capacitor failure in the short circuit mode from rendering the inverter inoperable. Also, equipment acceptance testing should be modified to obtain the most limiting design operating conditions for all major subcomponents. Results indicated that aging had not substantially affected equipment operation. On the other hand, the monitoring techniques employed were sensitive to changes in measurable component and equipment parameters. Thus comparing the monitoring results with the original acceptance test data is a viable method of detecting degradation prior to catastrophic failure.

NUREG/CR-5248, I. S. Levy, D. B. Jarrell, and E. P. Collins, "Prioritization of TIRGALEX-Recommended Components for Further Aging Research," Pacific Northwest Laboratory, Science Applications International Corp., PNL-6701, November 1988.

In April 1986, the NRC established the Technical Integration Review Group for Aging and Life Extension activities. In May 1987, TIRGALEX finalized its plan (TIRGALEX 1987), which identified the safety-related structures and components that should be prioritized for subsequent evaluation in the NRC NPAR program. This report documents the results of an expert panel workshop established to perform the prioritization activity. Prioritization was based primarily on criteria derived from a specially developed risk-based methodology that incorporates the effect on plant risk of component aging and the effectiveness of current industry aging management practices in mitigating that aging.

An additional set of criteria was the importance of aging research on structures and components to the resolution of generic safety issues and to identified regulatory needs. The resultant categorization was used to provide additional information to decision makers but was not used to calculate final rankings.

The expert panel workshop was conducted within the following ground rules:

1. Obtain all relevant information on aging of current plants (i.e., during their original license period),
2. Develop an understanding of aging and its effects (i.e., define the contribution of aging to plant risk),
3. Assess the adequacy of current industry practices for managing component aging within acceptable levels of risk,
4. Evaluate the importance of the aging of individual components and component groups on plant risk,
5. Apply the "Risk Significance of Component Aging" methodology (being developed by W. E. Vesely of SAIC under the NPAR program) to the prioritization,
6. Use operational failure data,
7. Use expert judgment through an interdisciplinary panel,
8. Stress the importance of aging research to the resolution of generic safety issues and to user needs identified by the Office of Nuclear Reactor Regulation to aid NRC decision-makers but not to formally prioritize the components.

NUREG/CR-5268, R. Lofaro, M. Subudhi, W. E. Gunther, W. Shier, R. Fullwood, and J. H. Taylor, "Aging Study of Boiling Water Reactor Residual Heat Removal System," Brookhaven National Laboratory, BNL-NUREG-52177, June 1989.

As part of ongoing efforts to understand and manage the effects of aging in nuclear power plants, an aging assessment of a vital system, the residual heat removal (RHR) system in boiling water reactors (BWRs), was performed. This report presents the results and discusses the impact of RHR system aging on plant safety. The work was performed as part of the NRC NPAR program. The RHR study was done according to the methodology developed by BNL as part of the Aging and Life Extension Assessment Program (ALEAP) System Level Plan. The selected approach uses two parallel work paths, one applying deterministic techniques and the other probabilistic techniques, to characterize aging.

The deterministic work performed for the RHR system study involved a review of past operating data from various national data bases. The data covered all operating modes of the RHR. They showed that approximately 70% of the failures reported were due to aging. The dominant cause of failure was found to be normal service, while the dominant failure mechanisms were wear and calibration drift. The predominant failure mode was leakage followed by loss of function and wrong signal. The data also indicated that approximately 65% of the failures were detected by the current test and inspection practices. However, 27% of the failures were not detected until an operational abnormality occurred. This shows that currently employed maintenance and monitoring practices are not completely successful in detecting all aging degradation. In evaluating the effect of failure on RHR performance, it was found that over 50% resulted in degraded system operation, while approximately 20% resulted in a loss of redundancy. Other significant effects of RHR failures include loss of shutdown cooling capability, radiological releases, reactor scrams, and actuation of engineered safety features. Actual plant records for Millstone Unit 1 were obtained and reviewed. The results showed consistency with data base findings.

The probabilistic work entailed the implementation of a personal-computer-based program (PRAAGE-1988) developed to perform time-dependent probabilistic risk assessment (PRA) calculations. The RHR model used was based on the Peach Bottom design. Time-dependent failure rates for major components were developed from the data base findings and were used in the program to calculate system availability and component importances for various ages. The PRA results showed that, when the time-dependent aging factors are accounted for, two significant system effects are seen: (1) system unavailability increases moderately with age and (2) the relative importances of components may change with age. For low-pressure coolant injection operation, miscalibration of instrumentation was the most important contributor to system unavailability. However, during later years, aging can cause motor-operated valves to become equally important. PRA calculations for shutdown cooling operation showed these valves to be the most important contributors to unavailability throughout plant life.

The following conclusions resulted from this assessment:

Aging Effects

1. Aging has a moderate impact on RHR component failure rates (0 to 17% per year increase) and system unavailability (2-fold to 4-fold increase in 50 years). This contribution of aging effects may be attributed to two factors: (1) RHR is a safety system and has relatively stringent testing and monitoring requirements that identify aging degradation before performance is adversely affected and (2) the RHR system is typically maintained in standby, which minimizes exposure to wear-related degradation.
2. Preliminary comparisons of unavailability for standby and continuously operating systems have shown that standby systems are potentially less severely affected by aging. Using this result as a basis, the differences in operation and management of these two types of systems will be further evaluated with the ultimate goal of developing methods that are effective in mitigating aging effects.
3. Examination of plant-specific failure data has confirmed that failure trends for certain components in some plants can differ from industry averages. Although aging was found to have a moderate impact on the RHR system based on average values, the impact on plants for which the data differ from these average values could be significant. This will be addressed in future work.

Data Analysis

1. Results have confirmed that generic failure rates may not accurately represent individual plants for all applications. The uncertainty in risk estimates may be reduced by updating calculations with actual plant data.
2. Mechanical components in the RHR system show a low to moderate increase (8% to 17% per year) in failure rate with age, while electrical components such as switches and sensors show little or no increase (0 to 3% per year).

Design Considerations

1. Plants with a common suction line supplying all loops of the RHR while in the shutdown cooling mode should consider placing increased attention on motor-operated valves (MOVs) in the suction line during later years of plant life since aging can increase the probability of MOV failure and lead to a temporary loss of shutdown cooling capability. Piping and other components in nonredundant supply lines should also be considered.
2. Plants using a common minimum flow line for two RHR pumps should closely monitor pump performance since aging can degrade performance and lead to dead-headed pump operation and possible failure.

The findings presented in this report form a sound technical basis for understanding and managing the effects of aging in RHR systems. The results also provide the framework for future Phase II work. Although the time-dependent aging effects appear to be moderate for the RHR system, additional work is necessary to complete the aging assessment. Since this is predominantly a standby system, exposure to operating stresses is limited, which could contribute to the mitigation of aging effects. However, as plants continue to age and operating time increases, the RHR system could experience rapid increases in failure rates, as was found in previous work on a continuously operating system. This should be addressed in future work. In addition, the relatively stringent tests and inspections performed for the RHR system may contribute to the aging effects.

NUREG/CR-5280, M. Subudhi, W. Shier, and E. MacDougall, "Age-Related Degradation of Westinghouse 480-Volt Circuit Breakers," Vol. 1, "Aging Assessment and Recommendations for Improving Breaker Reliability," Brookhaven National Laboratory, BNL-NUREG-52178, July 1990.

An aging assessment of the Westinghouse DS-series low-voltage air circuit breakers (especially DS-206 and DS-416) was performed as part of the NRC Nuclear Plant Aging Research (NPAR) program. These breakers are used for Class 1E applications in nuclear power plants. DS-416 breakers, in particular, are used for reactor trip applications. The findings from this study form a technical basis for understanding aging effects in DS-series breakers.

This study was initiated following the failure of a center pole lever weld in a reactor trip breaker at the McGuire Nuclear Station and the issuance of NRC Bulletin 88-01 on that subject. The objectives of the study are to characterize age-related degradation in the breaker assembly and to identify maintenance practices to mitigate degradation effects.

The design and operation of DS-206 and DS-416 breakers were reviewed in detail. Failure data from various operational data bases were analyzed (1) to identify all failure modes, causes, and mechanisms, (2) to assess the effectiveness of the requirements formulated in NRC Bulletin 88-01, and (3) to recommend activities that would effectively detect and mitigate age-related problems in breakers. The data bases included Licensee Event Reports (LERs), Nuclear Plant Reliability Data System (NPRDS), In-Plant Reliability Data System (IPRDS), and Nuclear Power Experience (NPE). Additional operating experience data were obtained from one nuclear station and two industrial breaker-service companies to develop aging trends for various subcomponents. The responses of the utilities to NRC Bulletin 88-01 were analyzed to assess the final resolution of failures of welds during reactor trips.

The predominant failure modes in nuclear power plants along with the causes and mechanisms of failure were determined from the operating experience data. Instruction manuals including schematics and manufacturers' maintenance manuals were analyzed to understand the effect of material aging during the service life of the breakers. This analysis was augmented by technical discussions with maintenance and service personnel from the electrical supply industry. Maintenance recommendations by the manufacturer to mitigate age-related degradation, suggestions for improving the monitoring of age-related degradation, and inputs from NRC inspectors involved in assessing breaker problems in the nuclear industry were reviewed.

Volume 2 of this report presents the results from a test program to assess degradation in breaker parts through mechnical cycling that simulated the operating life of nuclear plant breakers.

NUREG/CR-5280, M. Subudhi, E. MacDougall, S. Kochis, W. Wilhelm, and B.S. Lee, "Age-Related Degradation of Westinghouse 480-Volt Circuit Breakers," Vol. 2, "Mechanical Cycling of a DS-416 Breaker. Test Results," Brookhaven National Laboratory, BNL-NUREG-52178, November 1990.

After the McGuire event in 1987 involving failure of the center-pole weld in a reactor trip breaker, the NRC initiated an investigation of the probable causes. During the last decade, NRC has issued a number of information notices and bulletins pertaining to problems encountered in Class 1E breakers. A review of operating experience suggested that burned-out coils, jammed operating mechanisms, and deteriorated contacts were the dominant causes of failures. Although failures of the pole shaft weld were not included as one of the generic problems, the NRC Augmented Inspection Team had suspected that these welds were of substandard quality, which could lead to their premature cracking.

This program involved a commercial grade Westinghouse DS-416 low-voltage air circuit breaker that is typical of breakers used in nuclear power plants for class 1E applications. The test breaker was mechanically cycled for more than 36,000 full cycles with no electrical load, thus accelerating the aging process that could be attributed to breaker cycles to help identify age-related degradations. The test was conducted in accordance with ANSI/IEEE Standard 37.50 (1981) for the life testing of circuit breakers. Three different pole shafts with weld configurations of approximately 60 degrees, 120 degrees, and 180 degrees in the center-pole lever (#3) were used to characterize cracking in the pole lever welds. In addition, three operating mechanism units and several other parts were replaced as they became inoperable.

The mechanical cycling test resulted in the following conclusions on the manufacturing and aging of Westinghouse DS-series breakers:

1. Pole shafts used in this test program were found to have substandard welds. This raises questions as to the effectiveness of the quality assurance program that was followed during welding.
2. Fracture of the trip shaft lever suggested that correct electroplating procedures may not have been followed.
3. The sharper bends at the neck of the hooks on newly purchased reset springs—compared to an older design—led to early spring failures.
4. The hardness of the oscillator surface on newly procured units was 30% less than on older units.
5. Wear, fracture, distortion, and normal fatigue dominated the aging process, with wear being the largest contributor.
6. Excessive wear was evident in the ratchet wheel, holding pawls, oscillator, drive plate, motor crank and handle, cam segments, main roller, and stop roller.
7. Structural components and contact assembly parts showed few effects of aging due to mechanical cycling.
8. A pole shaft with a reduced size weld could fail at as few as 3000 cycles.

The testing yielded many useful results. The burned-out closing coils were found to be the result of binding in the linkages that are connected to this device. Among the seven welds on the pole shaft, #1 and #3 were the ones that cracked first and caused misalignment of the pole levers, which, in turn, led to many problems with the operating mechanism, including burned-out coils, excessive wear in certain parts, and overstressed linkages. Based on these findings, a maintenance program designed to alleviate the age-related degradations caused by mechanically cycling this type of breaker is suggested.

NUREG/CR-5314, C.E. Jaske and V.N. Shah, "Life Assessment Procedures for Major LWR Components; Vol. 3, Cast Stainless Steel Components," Idaho National Engineering Laboratory, EGG-2562, October 1990.

Many critical pressure boundary components in commercial light water reactors (LWRs) are made of cast stainless steels. Life assessment procedures are needed for these components because cast stainless steels are subject to thermal embrittlement during long-term service at LWR temperatures. The components of concern include pump bodies, reactor coolant piping and fittings, surge lines (in a few plants), pressurizer spray heads, check valves, control rod drive mechanism housings, and control rod assembly housings. These are made of grade CF-8, CF-8A, or CF-8M stainless steel in U.S. LWRs; grade CF-3 stainless steel is used in some foreign LWRs. The purpose of this project was to review the available data on thermal embrittlement of cast stainless steels and to develop updated procedures for life assessment by key LWR cast stainless steel components.

Cast stainless steels have a two-phase microstructure consisting of ferrite islands in an austenite matrix. With long-term exposure to LWR temperatures, other phases form in the ferrite phase that cause it to become hard and brittle, while the austenite remains ductile. If the amount of ferrite is small and if it is distributed evenly and finely throughout the austenite, the properties of the casting are not significantly affected by the thermal embrittlement of the ferrite. However, as the amount of ferrite, its coarseness, and its uneven distribution increase, the increased thermal embrittlement of the ferrite adversely affects the properties of the casting.

The properties most affected by thermal embrittlement are Charpy V-notch (CVN) impact energy and fracture toughness (J_{Ic}). Both of these properties decrease as the degree of thermal embrittlement increases. If these values become too low, the structural integrity of a cast stainless steel component could be seriously impaired. Presently, more fatigue-crack-growth data are needed for CF-8 and for all cast stainless steels in the high-cycle regime. Thus, for life assessment of cast stainless steel components, the main concern is loss of fracture toughness and impact energy. Data and engineering models have been developed to help predict the degree of embrittlement as a function of thermal exposure history. For reactor internals, irradiation history may also be a concern.

The minimum CVN impact energy after long-term aging has been found to be proportional to the square of the fraction of ferrite, the mean ferrite spacing, and a chemical-composition parameter. This model should be developed further for application to the assessment of components. A time-temperature parameter can be used to define lower-bound trends to the available impact energy values for cast stainless steels as a function of chemical composition and thermal exposure time. The report proposes a model using that parameter to predict the impact energy decrease for any particular lot of cast stainless steel. This predicted impact energy value or the predicted minimum impact energy value is then used to estimate fracture toughness from correlations between impact energy and fracture toughness at both room temperature and 290°C. This approach should provide a conservative estimate of fracture toughness for use in assessing the structural integrity of cast stainless steel components.

Inservice inspection (ISI) is needed to define type, size, and location of any defects in cast stainless steel components so that their structural integrity can be evaluated. Use of radiography during ISI is less practical than during fabrication. Conventional ultrasonic testing (UT) methods for detecting flaws are not reliable in cast stainless steel components because its coarse grain structures result in a low signal-to-noise ratio. Advanced UT methods being developed have shown an improved capability to detect flaws in cast stainless steel components and have been used in several PWR plants. Because of the difficulties with radiography and UT methods in detecting and sizing flaws, the application of the acoustic emission technique to detecting crack growth in cast stainless steel needs to be evaluated.

The report outlines a procedure developed for estimating the current condition and residual life of key LWR cast stainless steel components. The procedure is implemented in nine major steps. The first three steps involve the collection, examination, and storage of records for fabrication and construction, inservice inspection, and operating history. The fourth step involves a conservative fatigue and fracture mechanics evaluation to determine the worst-case flaw size and the minimum required fracture toughness at the end of the next operating period. In the fifth step, the current condition of the material is assessed using a proposed analytical model, microstructural data, or measured properties (or some combination of the three). In the sixth step, the results of the fourth and fifth steps are combined to evaluate the structural integrity of the component. The seventh step establishes what actions (none, repair, replace, or shut down) are to be taken, and the eighth step establishes the plan for the next ISI.

In the ninth step, the component is reevaluated and the steps are repeated as needed.

NUREG/CR-5334, D. B. Clauss, "Severe Accident Testing of Electrical Penetration Assemblies," Sandia National Laboratories, SAND89-0327, November 1989

Since the Three Mile Island incident, the risk and consequences of severe accidents have been a major focus of reactor safety research. The performance of the containment building has a significant effect on accident consequence; thus considerable effort has been directed toward understanding and predicting the functional failure of containments. The containment pressure boundary typically includes numerous mechanical and electrical penetrations, each of which represents a potential leakage path.

Several studies completed in the early 1980s indicated that electrical penetration assemblies could be an important leak path that merited further study. A report by the Oak Ridge National Laboratory on severe accident sequence analysis for BWR Mark I containments concluded that the temperatures in the drywell were high enough to possibly cause failure of the seals that could result in leakage. NUREG-0772 identified electrical penetration assemblies as having "one of the largest uncertainties associated with predicting the amount of radionuclides released." These studies provided the major impetus for NRC to initiate a research program on these assemblies. Sandia National Laboratories managed a program to conduct a background study and to recommend and perform tests to generate data that could be used to assess the leak potential when the assemblies are subjected to severe accident conditions. These tests are described in this report.

Electrical penetration assemblies are used to provide a leak-tight pass-through in nuclear power plant containment buildings for electrical cables with power, control, and instrumentation applications. The design has evolved to a modular concept that consists of electrical conductors contained within stainless steel tubes (modules) that are sealed.

Three designs, D. G. O'Brien, Conax, and Westinghouse, were tested under simulated severe accident conditions for a PWR, a BWR Mark I drywell, and a BWR Mark III wetwell, respectively, to generate engineering data (leak rate, temperature, insulation resistance, and electrical continuity) for assessing their leak potential. None of the assemblies leaked during the severe accident tests, which can be attributed to the use of redundant seals and to the fact that the outboard containment seals in all three designs were never exposed to temperatures that exceeded the service limits of the seal materials. The exceptional leak integrity of the three designs tested in this program should not be assumed to apply to all other designs in use for at least two reasons:

1. There are a large number of diverse designs in use. In particular, assemblies manufactured prior to 1971 were not subject to national standards and were often manufactured in the field, whereas the three tested in this program were subject to rigorous quality assurance and were designed to meet the standards of IEEE 317-1976 and IEEE 323-1974.
2. The leak potential is highly dependent on the temperatures to which the assembly is subjected. As research continues and more analyses of severe accident sequences are conducted, the "worst-case" loads may change. Therefore, the leakage potential must be reevaluated as the understanding of severe accident loads is improved. Heat transfer effects must be considered to determine the temperature of the outboard containment seals, which end up controlling the potential for leakage.

The results of these tests should not be construed as suggesting that all designs will not leak under severe accident conditions; the performance of all components of the containment pressure boundary must be evaluated on a case-by-case basis with all loads considered. The performance is also affected by thermal and radiation aging. Given good information on the containment loads, a heat transfer analysis to determine the approximate temperature profiles, knowledge of the time-temperature thresholds for the sealant materials used, and the proper exercise of engineering judgment, a reasonable evaluation of the leakage potential of other designs can be made.

The electrical performance of the assemblies was monitored in these tests by measuring the insulation resistance and electrical continuity of the conductors. The resistance degraded rapidly during the severe accident tests, although the rate depended more on the type of cable and loads than on the particular design being tested. Under the specific severe accident conditions that were simulated, the data suggest that all electrical systems supplied in the Westinghouse assembly would have functioned for about 4 days; those supplied in the D. G. O'Brien would have functioned for about 13 hours; and those supplied in the Conax may have functioned for only about 5 hours. Some cables would be expected to function beyond these times. However, it must be noted that conclusions regarding the electrical performance of systems inside the containment building based solely on insulation resistance data must be made with caution. The performance of the electrical systems would depend on the voltage, current, and impedance requirements for a specific conductor application.

NUREG/CR-5378, A.J. Wolford, C.L. Atwood, and W.S. Roesener, "Aging Data Analysis and Risk Assessment—Development and Demonstration Study," Idaho National Engineering Laboratory, EGG-2567, August 1992.

This work develops and demonstrates a probabilistic risk assessment (PRA) approach to assess the effects of aging and degradation of active components on plant risk. The work (a) develops a way to identify and quantify age-dependent failure rates of active components and then incorporate them into PRAs, (b) demonstrates these tools by applying them to a fluid-mechanical system, using the key elements of a NUREG-1150 PRA, and (c) presents them in a step-by-step approach, to be used for evaluating risk significance of aging phenomena in systems of interest.

Statistical tests are used for detecting increasing failure rates and for testing data-pooling assumptions and model adequacy. The component failure rates are assumed to change over time, with several forms used to model the age dependence—exponential, Weibull, and linear. Confidence intervals for the age-dependent failure rates are found and used to develop inputs to a PRA model in order to determine the plant core damage frequency. This approach was used with plant-specific data, obtained as maintenance work requests, for the auxiliary feedwater system of an older pressurized water reactor. It can be used for extrapolating present trends into the near future and for supporting risk-based aging-management decisions.

NUREG/CR-5379, D. B. Jarrell, A. B. Johnson, Jr., P. W. Zimmerman, and M. L. Gore, "Nuclear Plant Service Water System Aging Degradation Assessment: Phase 1," Vol. 1, Pacific Northwest Laboratory, PNL-6560, June 1989.

The service water system represents the final heat transfer loop between decay heat generated in the nuclear core and the safe dispersal of that heat energy into the environment. The objective of this assessment is to demonstrate that aging phenomena in the service water system can be identified and quantified so that aging degradation of system components can be detected and mitigated before the system availability is reduced below an acceptable threshold. The following goals of the assessment were directly derived from the NRC NPAR program plan:

1. To identify the principal aging-degradation mechanisms, to assess their impact on operational readiness, and to provide a methodology for mitigating the effects of service water system aging on nuclear plant safety.
2. To examine the current surveillance specifications and evaluate their ability to provide accurate reliability information.
3. To provide a means to evaluate the effectiveness of maintenance on mitigating aging-degradation phenomena.
4. To produce an inspection plan that optimizes the effectiveness of inspections based on system risk reduction.
5. To use the information generated by this assessment to resolve related generic issues and provide guidance for developing regulatory criteria on aging and life extension.

The following approach was used during the initial phase of the assessment:

1. Perform a literature search of government and private sector reports that are related to service water, aging-related degradation, and potential methodologies for analysis.
2. Assemble a data base that contains a listing of the configurations, characteristics, and water sources for the service water systems in all commercial nuclear power plants in the U.S.
3. Obtain and examine the available service water data from large generic data bases, i.e., the Nuclear Plant Reliability Data System, Licensee Event Reports, Nuclear Power Experience, inspection reports, and other relevant plant reference data. Analyze the service water system of a specific power plant for aging-related degradation phenomena from the available data obtained from this data base.
4. Perform a fault-tree analysis of the service water system of a typical plant to examine failure propagation and determine specific input requirements of probabilistic risk analyses.
5. Develop an in-depth questionnaire protocol for examining the information resources at a plant where such resources are not available in the standard data bases. Subsequently, visit a nuclear power plant and solicit the required information.
6. Analyze the information obtained from the in-depth plant interrogation and draw contrasts and conclusions in regard to the data base.
7. Use the plant information to perform an interim assessment of degradation mechanisms and to focus future investigations.

The following is a summary of the conclusions of the assessment to date:

1. Aging-related degradation of open service water systems, i.e., systems that have a direct interface to raw water without chemical control, in nuclear plants is prevalent and constitutes a valid safety concern. Based on actual plant data, the primary degradation mechanism found in the open systems is corrosion compounded by the accumulation of biologic and inorganic material. This conclusion directly contradicts the

results of a failure analysis performed using information obtained from the NPRDS data base, which indicated that the torque switches of motor-operated valves were the prime cause of system failure.

2. Based on multiple plant samplings, the current level of surveillance and postmaintenance testing performed on the system is not sufficient to accurately trend or detect system degradation due to aging phenomena.
3. While postmaintenance surveillance does give some measure of the effectiveness of system modification and repair efforts, sufficient information on monitoring operational condition and postmaintenance testing is not available to characterize more precisely the effectiveness of maintenance.
4. To improve the accuracy of data to a point that would allow a high degree of confidence in the analysis of aging degradation, a root cause logic scheme needs to be developed for use in defining the depth of knowledge and the documentation required to accurately characterize an aging-related component failure.
5. Clear resolution of relevant aging-related safety issues will require the specification of additional documentation of failure data and regulatory requirements to ensure adequate safety margin under aged or extended-life conditions.

NUREG/CR-5379, D.B. Jarrell, L.L. Larson, R.C. Stratton, S.J. Bohn, M.L. Gore, "Nuclear Plant Service Water System Aging Degradation Assessment," Volume 2, Pacific Northwest Laboratory, PNL-7916, October 1992.

The second phase of the aging assessment of nuclear plant service water systems (SWSs) was performed by the Pacific Northwest Laboratory to support the NRC's Nuclear Plant Aging Research (NPAR) program. The SWS was selected for study because of its essential role in the mitigation of and recovery from accident scenarios involving the potential for core melt, and because it is subject to a variety of aging mechanisms. The objectives of the SWS task under the NPAR program are to identify and characterize the principal age-related degradation mechanisms relevant to this system, to assess the impact of aging degradation on operational readiness, and to provide a methodology for the management of aging on the service water aspect of nuclear plant safety.

The primary degradation mechanism in the SWSs, as stated in the Phase I assessment and confirmed by the analysis in Phase II, is corrosion compounded by biologic and inorganic accumulation. It then follows that the most effective means for mitigating degradation in these systems is to pursue appropriate programs to effectively control the water chemistry properties when possible and to use biocidal agents where necessary.

A methodology for producing a complete root-cause analysis was developed as a result of needs identified in the Phase I assessment for a more formal procedure that would lend itself to a generic, standardized approach. It is recommended that this, or a similar methodology, be required as a part of the documentation for corrective maintenance performed on the safety-related portions of SWSs to provide an accurate focus for effective management of aging.

NUREG/CR-5383, H. M. Hashemian, K. M. Petersen, R. E. Fain, and J. J. Gingrich, "Effect of Aging on Response Time of Nuclear Plant Pressure Sensors," Analysis and Measurement Services Corporation, Knoxville, TN, June 1989.

A research program was initiated to study the effects of normal aging on the dynamic performance of safety-related pressure transmitters (i.e., sensors) in nuclear power plants. The project began with an experimental assessment of the conventional and new testing methods for measuring the response time of pressure transmitters. This was followed by developing a laboratory setup and performing initial tests to study the aging characteristics of representative transmitters of the type used in nuclear power plants.

There is need to ensure that the current testing methods, regulatory requirements, and industry standards and practices are adequate to track age-related degradation. The project examined the validity of the available methods for response-time testing of pressure transmitters and reviewed the historical data for evidence of performance degradation problems or trends. Current intervals for response-time testing and calibrating pressure transmitters are based on refueling schedules, apparently for two reasons:

1. There is no method available for on-line calibration of pressure transmitters, and, until recently, response-time testing could not be performed on line.
2. The available data base of degradation rates and trends is not sufficiently reliable to justify testing intervals longer than one refueling cycle.

While testing based on refueling intervals may be adequate, there is concern that the rate of degradation of pressure transmitter performance may increase as the current generation of plants becomes older. Furthermore, on-line testing methods based on new technologies are becoming available to permit more frequent testing of transmitters and to predict incipient failures. These considerations have motivated research such as that covered in this report to ensure that practical test methods and adequate test schedules are used to verify proper and timely performance of safety-system pressure transmitters in nuclear power plants.

The project included a search of the licensee event report (LER) data base for pressure-sensing system problems and reviews of Regulatory Guide 1.118 and of the industry standards on performance testing of pressure transmitters. The following conclusions have been reached:

1. Five reasonably effective methods are available for response-time testing of pressure transmitters in nuclear power plants. These methods are referred to as step test, ramp test, frequency test, noise analysis, and power interrupt test. Two of the five methods (noise analysis and power interrupt test) have the advantage of providing on-line measurement capability at normal operating conditions.
2. The consequences of aging at simulated plant conditions were calibration shifts and response-time degradation, the former being the more pronounced problem.
3. The LER data base contains 1,325 cases of reported problems with pressure-sensing systems over a nine-year period (1980–1988). Potential age-related cases account for 38% of the reported problems in this period. A notable number of LERs reported problems with blockages, freezing, and void (bubble) formation in sensing lines.
4. Regulatory Guide 1.118, IEEE Standard 338, and ISA Standard 67.06 can benefit from minor recommended revisions to account for recent advances in performance testing technologies and from new information that has become available since these documents were initially generated.

The six-month study of the dynamic performance of pressure transmitters covered the following areas:

1. *Assessment of Response-Time Testing Methods.* An experimental assessment of the five methods mentioned above involved laboratory testing of more than twenty pressure transmitters with all five methods. Results showed that the methods are equally effective but vary widely in difficulty of implementation in nuclear power plants. Two of the five methods (noise analysis and power interrupt test) can be performed remotely on installed transmitters while the plant is at normal operating conditions.
2. *Aging Study.* Laboratory research on aging was initiated and preliminary results were obtained. The work involved response-time testing and calibration checks of a number of transmitters after exposure to heat, humidity, vibration, pressure, cycling, and overpressurization conditions. The effect of these conditions was an increase in response time and calibration shifts, the latter being the more pronounced problem.
3. *Review of Related Studies.* All published experimental work on aging of pressure transmitters has concentrated on the effects of aging on static performance of the transmitters as opposed to the dynamic performance reported herein. The related studies concluded that aging affects the performance of pressure transmitters and that temperature is the dominant stressor. Most of the studies on performance of nuclear plant pressure transmitters were sponsored by the NRC. The only other major work was performed by manufacturers for environmental and seismic qualification of transmitters. However, the transmitter qualification data are not sufficient to address normal aging.

The aging research covered in this report was a feasibility study; it used accelerated aging to accommodate the short (6 months) duration of the project. Since accelerated aging does not necessarily simulate normal aging, the aging results in this report must be viewed as preliminary. Furthermore, this study was concerned with the performance of the portion of the pressure-sensing system and electronics located in the harsh environment of the plant; the power supply and other components of the pressure-sensing channel that are located in the control room, cable spreading room, or other mild environments were not studied.

NUREG/CR-5386, D. P. Brown, G. R. Palmer, E. V. Werry, and D. E. Blahnik "Basis for Snubber Aging Research: Nuclear Plant Aging Research Program," Pacific Northwest Laboratory, Lake Engineering Company, Wyle Laboratories, PNL-6911, January 1990.

This report proposes a research plan to address the safety concerns of aging in snubbers used on piping and large equipment in commercial nuclear power plants. The proposed program will provide the structure for the Phase II Snubber Aging Study for the NRC NPAR program, to be performed at nuclear power plants and in test laboratories. This research would be an extension of the work performed by the Pacific Northwest Laboratory (PNL) in the Phase I Snubber Aging Study, the primary objectives of which were to conduct an initial aging assessment of snubbers and to evaluate the concept of reducing the number of snubbers in commercial nuclear power plants. Although snubber reduction programs may reduce their total population by 50 to 80%, this will not mitigate the concern for managing the aging of the remaining snubbers. Indeed, the remaining snubbers may become more important to plant safety than the original population. The proposed Phase II research work is based, in part, on a study of snubbers in U.S. nuclear power plants by the Lake Engineering Company conducted for PNL under the NPAR program. A survey of U.S. utilities conducted for PNL by Wyle Laboratories on the use of snubbers in nuclear plants was also used to identify research needs.

The following are key elements of the proposed snubber research:

1. Review of existing service data,
2. Development of service-life monitoring guidelines,
3. Evaluation of the effects of compression set in hydraulic seals,
4. Evaluation of accelerated methods for predicting seal life,
5. Identification of seals most affected by aging.

The benefits to be derived from the research are principally safety related, including enhanced failure prediction and seismic protection of safety-related piping and equipment, mitigation of snubber aging effects, reduction of staff radiation exposures, and reduction of rad waste. Numerous technical benefits are also expected, including the identification of aging trends, information useful in developing guidelines for monitoring service life, the technical bases for determining service life, the effects of compression set in seals, and improvements in snubber design, materials, and maintenance. Regulatory benefits anticipated include contributions to Standard Review Plans, Regulatory Guides, Plant Technical Specifications, and ASME/ANSI OM-4 Standards based on the broader, more comprehensive data base that would be developed.

The research proposed is designed to address the following questions about the aging of mechanical and hydraulic snubbers:

1. How do snubbers age and degrade?
2. What are the failure characteristics of snubbers?
3. What are the safety implications of snubber aging?
4. What technical information is needed to improve the performance and life expectancy of snubbers?

The results will contribute toward more reliable and predictable snubbers in the nuclear power industry and thus will improve nuclear plant safety. Implementation of the research plan will also provide a data base for use in addressing regulatory and snubber technology issues. The data base will be made available to nuclear utilities, snubber manufacturers, snubber service companies, and the NRC. Planned interfaces will ensure technology transfer to utilities and manufacturers.

NUREG/CR-5404, D. A. Casada, "Auxiliary Feedwater System Aging Study," Vol. 1, Oak Ridge National Laboratory, ORNL-6566/V1, March 1990.

This review of the auxiliary feedwater (AFW) system used at pressurized water reactor (PWR) plants has been conducted under the auspices of the NRC NPAR program. The primary purposes of the review were to (1) determine the potential and historical sources and modes of failure within the AFW system, (2) identify currently applied means of detecting known sources and modes of degradation and failure, and (3) evaluate the general effectiveness of current monitoring practices and identify specific areas where enhancements appear needed.

The report reviews historical failure data available from the Nuclear Plant Reliability Data System, Licensee Event Report Sequence Coding and Search System, and Nuclear Power Experience data bases. The failure histories of AFW system components are considered from the perspectives of how the failures were detected and the significance of the failures. Results of a detailed review of operating and monitoring practices at a plant owned by a cooperating utility are presented. General system configurations and pertinent data are provided for Westinghouse and Babcock and Wilcox units.

The report includes an identification of the general types of AFW system design configurations, an analysis of historical failure data, and a detailed review of a cooperating utility's AFW system design and their current operating and monitoring practices.

Historically, and particularly since the Three Mile Island 2 accident, the AFW system has been recognized as critical to successful mitigation of plant transients and accidents. In recent years, operating incidents involving failures of AFW system components have been among the leading events identified in NUREG/CR-4674, Vols. 1-8, "Precursors to Potential Severe Core Damage Accidents," in which the leading risk-significant events are identified for several calendar years. In the years 1984 through 1986, seven of the top ten events at PWRs, from a core damage risk standpoint, involved partial or total failure of the AFW system. Operational problems with these systems have been diverse in nature. The report lists six events resulting in NRC Bulletins and Information Notices as examples of the diverse types of failures involving the AFW systems. Numerous other operating experiences have resulted in feedback to the industry through both the NRC and the Institute of Nuclear Power Operations (INPO).

In reviewing the role that aging plays in failures such as those of AFW systems, three important points must be considered. First, a combination of factors, including design, maintenance, operation, aging, and other considerations may be involved. These factors are not necessarily independent of one another.

Second, systems age only as the individual components age. Other studies performed under the NPAR program address important components within the AFW system and discuss the aging stressors for these individual components.

Third, a study performed by INEL reviewed historical failure data from the Nuclear Plant Reliability Data

System (NPRDS) and made judgments as to whether or not individual failure episodes were related to aging.

Because of the above three points, the ORNL approach to the AFW system study has been to focus attention on how and to what extent the various AFW system components fail, how the failures have been and can be detected, and what is the value of existing testing requirements and practices, rather than attempting to focus on the extent to which aging (versus design or operating practices, for example) is responsible for failure or degradation.

An analysis of historical failure data involving AFW systems was completed by a detailed review of an existing AFW system and the associated monitoring practices of a cooperating utility. The single largest source of AFW system degradation, based upon the analysis of historical failure data, is the turbine drive for AFW pumps. It should be noted that the turbine proper has been a relatively reliable and rugged piece of equipment. However, the turbine auxiliaries, including the governor control and the trip and throttle valve, have contributed substantially to the overall turbine problems.

The sum of the failures of motor operators and air operators for valves resulted in approximately the same number of AFW system degradations as did failures of the turbine drives alone. Pump failures and check valve failures were also significant contributors to system degradation.

For each type of component and for the various sources of component failures, the methods of failure detection were designated and tabulated. The most notable feature of this aspect of the study was that failures related to instrumentation and control dominated the group of failures that were detected during demand conditions (as opposed to failures detected as the result of periodic monitoring or routine observations made by operators or other personnel). Many of the potential failure sources that were not detectable by the current monitoring practices were related to the instrumentation and control portion of the system.

It was also observed that a number of conditions related to design basis demands are not being periodically verified. Examples of these include pump capacities not being verified at design flow/pressure conditions, turbines not being verified to be capable of delivering required torque at low steam pressures, various control sequences not being checked, and automatic pump suction transfers not being tested.

Another observation made was that some components or certain parts or aspects of components appear to be tested in excess of what failure history indicates to be appropriate. On the other hand, other aspects of certain parts of the AFW systems are either never tested or receive less than thorough testing. It appears that improved testing requirements are needed in order to reduce excessive testing while at the same time ensuring that thorough performance verification is conducted periodically.

NUREG/CR-5404, J.D. Kueck, "Auxiliary Feedwater System Aging Phase I Follow-on Study," Volume 2, Oak Ridge National Laboratory, ORNL-6566/V2, July 1993.

The Phase I study found a number of significant Auxiliary Feedwater System functions that were not tested and verified operable by periodic surveillance testing. In addition, the Phase I study identified components actually degraded by the periodic surveillance tests. Thus, it was decided that this follow-on study would not deal with aging assessments or in situ examination but would instead focus on the testing omissions and equipment degradation found in Phase I.

In this follow-on study, the deficiencies in current monitoring and operating practice are categorized and evaluated. Areas of component degradation caused by current practices are discussed. Recommendations are made for improved diagnostic methods and test procedures that will verify operability without degrading equipment.

NUREG/CR-5406, K.G. DeWall and R. Steele, Jr., "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 1, "Analysis and Conclusions," Idaho National Engineering Laboratory, EGG-2569, October 1989.

Recent testing sponsored by the Nuclear Regulatory Commission (NRC) showed that, for at least some gate valves installed in nuclear applications, the equations used by industry to size the valve operators do not conservatively calculate the thrust needed to close the valves under design basis loadings. The tests also showed that the results of in situ valve testing at lower loadings cannot be extrapolated to design basis loadings. This volume describes the testing conducted by the Idaho National Engineering Laboratory (INEL) to provide technical data for the NRC effort regarding Generic Issue 87 (GI-87) "Failure of HPCI Steam Line Without Isolation." The test program also provides information applicable to Generic Issue II.E.6.1, "In Situ Testing of Valves," and a related document, IE Bulletin 85-03, "Motor Operated Valve Common Mode Failures During Plant Transient Due to Improper Switch Settings."

Of the three boiling water reactor (BWR) process lines covered by GI-87, an unisolated break in the reactor water cleanup (RWCU) supply line was selected for the first phase of testing because such a break would have the greatest safety impact. All three GI-87 process lines have common features: all communicate with the primary system, pass through containment, and have normally open isolation valves.

To meet the new valve operating criteria required by IE Bulletin 85-03 and Generic Letter No. 89-10, industry developed new diagnostic test equipment and methods for in situ motor-operated valve (MOV) testing. IE Bulletin 85-03 succeeded in significantly improving the operability of the selected safety-related valves because it caused many of the utilities to reanalyze the design basis load for the applicable MOVs and to reset the control switches accordingly.

However, very little design basis testing of valves has been conducted outside the plant to verify the analytic assumptions used to determine valve switch settings. Analytic assumptions are necessary because, in many cases, the utility cannot test valves at design basis loadings in situ. The GI-87 testing provides some of the first measured valve responses with which industry's valve operator sizing equations can be compared.

In this initial test program, two representative RWCU isolation valves were subjected to the hydraulic qualification tests described in ANSI B16.41, the qualification standard for nuclear valves, and then to flow interruption tests at full RWCU pipe-break flow. In all, fourteen flow interruption tests were performed, ten on Valve A and four on Valve B. In the Valve A tests, the parametric study included varying both the degree of inlet water subcooling and the pressure. The four Valve B tests were all performed at a normal BWR 10°F subcooling, and only the inlet pressure was varied.

Test results show that, for both valve designs tested, the force required to open and close the valves at temperatures above 100°F was significantly higher than the force predicted by the valve manufacturers. Only during the valve-opening tests at room temperature without flow did the typical industry valve thrust equation predict the valve response. Industry has assumed that the valve-opening thrust requirements would be the highest when the disk lifted off the seat. This was determined not to be true for the valves tested. The highest opening loads (maximum thrust) with flow occurred at different openings for both valves, but in both cases, they were well off their respective seats. Valve-closing thrusts at full line-break flows were higher (up to one third) than anticipated.

The test results provide evidence for two concerns with MOVs in nuclear power plants. First, proper sizing of motor operators is complicated by the fact that the equation used for calculating the stem force needed to close or open a gate valve does not have terms for the effects of temperature, degree of fluid subcooling, internal valve clearances, and the differences in the opening and closing forces that are not accounted for by the stem rejection term. Second, effective in situ testing is very difficult because (1) the tests cannot be conducted at design basis conditions and (2) even with the valve loadings properly quantified during the in situ tests, the results cannot be extrapolated to design basis conditions because the final thrust varies depending on the extent to which disk friction rather than disk seating affects the torque switch.

The disk factor of 0.3 typically used in industry to calculate disk friction force is not conservative for either of the valves tested. A disk factor of 0.5 marginally predicts the forces for one valve during both opening and closing. The response of the other valve is enveloped by the 0.5 disk factor during opening but not during closing. Today's tools for analyzing valve response to fluid loadings are not sophisticated enough to detect small design differences that make large response differences. Temperature also affects the thrust requirements of these gate valves.

All the facts listed justify continued qualification testing of prototypical valves at design basis loadings and stress the need for industry to add new terms to the equation or to increase the disk factor to a very conservative number to account for the missing terms in the equation. Also, test results show that the stem factor is not a constant but changes with stem load, thus making it very difficult to extrapolate normal in situ valve testing to design basis conditions.

When tests or improved sizing equations have determined the thrust needed to operate a valve at its design basis loading, utilities can use one of several modern diagnostic systems to conservatively set the motor operator control switches. However, this method may exceed the allowable thrust on some valve designs. This job will be made easier and the result will be more conservative if both the torque and the thrust can be measured when the switches are set. If further research proves that there is a proportional relationship between stem load and stem factor, the degree of conservatism can be reduced.

NUREG/CR-5406, K.G. DeWall and R. Steele, Jr., "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 2, "Data Report," Idaho National Engineering Laboratory, EGG-2569, October 1989.

This volume presents the 700 pages of actual measured data from the gate valve test program described in Volume 1. They are provided for those readers who wish to look at the data and form their own opinion on the performance of the test valves. For those readers who wish to do their own analysis, the electronic data are available from the Idaho DOE Office of Technology Transfer, (208) 526-8318.

Figure 1 of Volume 2 shows the test loop in schematic form and identifies the instrument location and numbers. Figure 2 converts the differential pressures into flow rates (gallons per minute). Table 1 outlines the test sequence performed on each valve and correlates the data as they are presented here. In the remaining figures, the header on each plot defines the valve (A or B), the test series number, and the test step number. Table 2 lists the test parameters measured during blowdown tests. Table 3 displays the test step matrix for qualification and blowdown tests, and Table 4 lists the test steps and system pressure and temperature for each of the tests performed.

NUREG/CR-5406, K.G. DeWall and R. Steele, Jr., "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 3, "Review of Issues Associated with BWR Containment Isolation Valve Closure," Idaho National Engineering Laboratory, EGG-2569, October 1989.

This volume discusses research performed to develop technical insights for the NRC effort regarding Generic Issue 87, "Failure of HPCI Steam Line Without Isolation." Volumes 1 and 2 describe the relevant valve test program. The research began with a survey to characterize the population of normally open containment isolation valves in those process lines that connect to the primary system and penetrate containment. The qualification methodology used by the various manufacturers listed in the survey is reviewed and deficiencies in that methodology are identified.

Four boiling water reactor (BWR) systems, the emergency cooling system, the high-pressure coolant injection system, the reactor core isolation cooling system, and the reactor water cleanup system, were included in the valve assembly characterization. The "typical" containment isolation valve is a 3 to 10 in., 600 to 900 lb gate valve. The most common design is a cast steel, flexible wedge, pressure-seal valve with a Limitorque operator (AC inside and DC outside of containment). The Anchor/Darling Valve Company manufactures approximately 40% of the valves in the four BWR systems.

The mitigation of a high-energy pipe break is within the design basis for the above valve assemblies, with typical system design conditions of 1250 psi and 575°F. No flow testing has been performed under these conditions to verify the presumptions used by manufacturers in the qualification analysis calculations. Operator torque switch settings are determined using calculations supplied by the valve vendor; torque settings inadequate to close the valve could result if the original calculations are not conservative.

Most of the valve and operator manufacturers use the same equation with minor variations in coefficients to size operators. In this equation, the required thrust to close the valve is equal to the sum of the disk drag load due to differential pressure, the stem end pressure load, and the packing drag load. The service conditions used in the thrust equation are supplied by each individual plant. Four areas have been identified as having the most influence on stem thrust requirements:

1. Repeated cycling can have a significant effect on valve thrust requirements.
2. The typical value of 0.3 for the disk friction coefficient used by the industry is not conservative for all cases.
3. The influence of mass flow/momentum on valve thrust requirements may be significant.
4. Increased temperature causes a significant increase in valve closure loads.

The limited number of tests performed to assess the capability of the gate valve to interrupt the flow of high-pressure steam has resulted in a relatively frequent inability to isolate portions of piping systems. The data now available suggest that industry may be using nonconservative friction factors and possibly underestimating valve stem thrust requirements. Additional work is needed to determine whether present qualification practices are adequate. Recommendations for expanding the qualification of valve assemblies for high-energy pipe break conditions are presented.

NUREG/CR-5419, M. Villaran, R. Fullwood, and M. Subudhi, "Aging Assessment of Instrument Air Systems in Nuclear Power Plants," Brookhaven National Laboratory, BNL-NUREG-52212, January 1990.

As part of ongoing efforts to understand and manage the effects of aging in nuclear power plants, an aging assessment was performed for the instrument air system, a system that recently has been the subject of much scrutiny. Despite its nonsafety classification, instrument air has been a factor in a number of potentially serious events. This report presents the results of the assessment and discusses the impact of aging of the instrument air system on system availability and plant safety. This work was performed as part of the NRC NPAR program. The objective of this study was to identify all the aging modes and their causes that should be mitigated to achieve reliable operation of all safety-related air equipment. Also included is an interim review of typical maintenance activities for air systems in the nuclear power industry.

To perform the complex task of analyzing an entire system, the Aging and Life Extension Assessment Program (ALEAP) System Level Plan was developed by Brookhaven National Laboratory (BNL) and applied successfully in previous studies. The work used two parallel work paths, one using deterministic techniques to assess the impact of aging on compressed air system performance, and the other using probabilistic methods. The results from both paths were used to characterize aging in the instrument air system. The findings from this study, some of which have applications beyond the instrument and service air systems, formed a technical basis for understanding the effects of aging in compressed air systems. The major conclusions from this work are:

1. This study identified aging trends in component failure rates, the relative importance of components, and system unavailability. All these trends could have a deteriorating impact on system availability and consequently on plant safety in later years.
2. Compressors, air system valves, and air dryers, made up the majority of failures. The failures in passive components such as piping, aftercoolers/moisture separators, and receivers increased with time, but these still constituted only a small percentage of overall failures.
3. The effectiveness and quantity of preventive maintenance devoted to a component significantly reduced the number of failures experienced. However, existing maintenance programs within the industry lack uniformity, and quality assurance is not rigorous because the system is classified as "nonsafety."
4. Individual plant maintenance records for instrument and service air systems were found to be the most comprehensive source of data for performing aging analyses.
5. As a continuously operating system with minimal control room instrumentation because of its nonsafety classification, most problems in the air system are detected by local monitoring and indication, walkdown-type inspection, and preventive maintenance inspection or surveillance.
6. Review of compressed air system designs and studies using a PRA-based system model revealed that the redundancy of key components (compressors, dryers, instrument air/service air cross-connect valve) was an important factor in system availability. The overall design configuration affected the pervasiveness of air system problems.
7. Total-loss-of-air events are uncommon. The majority of events resulted in degraded operation (low pressure, air quality out of limits). Normal wear of the system and contamination of the air dominate the problems of system failure. Procedures and testing for the response of personnel and equipment to these conditions should be developed.
8. Human error was a significant cause of failures in critical components such as compressors and dryers, as well as at the system and intersystem level. Training should be augmented in two key areas: (1) operation and maintenance of critical air system components and (2) understanding the importance of instrument air to other plant systems, particularly safety systems.
9. The outside systems that were most often affected by instrument air problems are containment isolation, main feedwater/main steam, auxiliary feedwater, and the BWR scram system. The most commonly affected components were air-operated and solenoid-operated valves.
10. The probabilistic work entailed the development of a computer program (PRAAGE-IA) using a PRA-based instrument air system model to perform time-dependent PRA calculations. Time-dependent failure rates were developed from the data base and other inputs to the program to calculate system unavailability and component importances for various ages. The results showed that, when the time-dependent effects of aging for the worst case are accounted for, there are two significant system effects: (1) system unavailability increases moderately with age and (2) the relative importance of components changes with age. During early operation, leakage in both instrument air/service air piping and support system piping was the most important contributor to system unavailability; during the later years, aging can cause compressors and air dryers/filters to become increasingly important.

The findings presented in this report form a sound technical basis for understanding and managing the effects of aging in instrument air systems. Future work will include improvements to current maintenance, monitoring, training, surveillance, and off-normal response procedures to mitigate degradation due to aging.

NUREG/CR-5448, J.L. Edson, "Aging Evaluation of Class 1E Batteries: Seismic Testing," Idaho National Engineering Laboratory, EGG-2576, August 1990.

Batteries are the only installed source of electric power to provide for monitoring plant conditions and control of some systems of the nuclear reactor in the event of a station blackout (all offsite power is lost and the diesel generators do not start). Approximately 60 individual 2-V cells are connected together to form a typical 125-V dc battery bank that has enough voltage and electrical capacity to provide the needed electric power for the period of time determined for each nuclear plant in accordance with NRC regulations.

Within the NRC Nuclear Plant Aging Research (NPAR) program, a Phase I study of battery aging was performed and reported in NUREG/CR-4457, "Aging of Class 1E Batteries in Safety Systems of Nuclear Power Plants." The study concluded that significant aging effects for old batteries are growth of positive plates, loosening of active material in plates that have grown, loss of active material caused by gassing and corrosion, and embrittlement of the lead grids and straps. The results of these effects are decreased electrical capacity and decreased seismic ruggedness that, during a seismic event, can lead to decreased electrical performance or complete failure.

Since batteries are susceptible to aging degradation that could cause old batteries to be vulnerable to severe seismic events, a test program was conducted to determine if it is possible for the seismic ruggedness of aged batteries in nuclear plants to be inadequate, even though the measured electrical capacity is satisfactory. In addition, selected alternative surveillance methods were evaluated during the testing program to determine if any of them are likely to be more sensitive to battery degradation than the surveillance and testing methods specified in IEEE Std 450-1987, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Storage Batteries for Generating Stations and Substations," and Regulatory Guide 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants."

The batteries in the test program had lead-calcium plates and were manufactured by C&D Batteries. Discussions with C&D personnel indicate that they are typical of batteries presently being installed in nuclear facilities. Each cell had a rated 8-hour electrical capacity of 1350 ampere-hours; was 7-5/8 in. long, 14-1/8 in. wide, and 22-1/16 in. high; and weighed about 240 pounds. They were obtained from a nuclear facility where they were naturally aged to 13.5 years. Records provided by the nuclear facility indicate that the batteries were maintained and tested in accordance with practices that are consistent with those in IEEE Std 450.

The batteries were installed on a shake table using a new battery rack purchased from the battery vendor and were tested to seismic spectra that are typical of those required for the safe shutdown earthquake (SSE) in U.S. nuclear facilities. Information received from selected nuclear plants and the Electric Power Research Institute (EPRI) was used to specify the required response spectrum (RRS) for the seismic tests. The tests were conducted using four different seismic levels representing the best estimate for the RRS encompassing 50%, 85%, 95%, and 100% of the U.S. nuclear plants.

During the seismic tests, the batteries were discharged at 2% of the 3-hour rate while current and battery voltages were monitored to detect the existence of catastrophic failure. During the preseismic, seismic, and postseismic tests, alternative surveillance and monitoring methods were employed to determine whether other methods may be more sensitive to aging-related degradation of batteries than the standard volt-ampere tests that determine their electrical capacity. The alternative monitoring methods employed were (1) measurement of internal resistance, (2) measurement of capacitance, and (3) measurement of battery polarization (comparison of battery voltages measured while increasing discharge current with those obtained with decreasing discharge current). These measurements were suggested as a result of investigations performed by the Westinghouse R&D Center for Sandia National Laboratories in 1986.

Results of the seismic tests indicate that the capacity of lead-calcium batteries of this design did not decrease as a result of shaking at seismic levels that include the most severe SSE levels specified. In fact, the average electrical capacity (ampere-hours) of batteries tested at the 100% seismic level increased from a preseismic capacity of 96% to a postseismic capacity of 98%. The batteries did not show any degradation except for some minor external damage. The battery rack suffered some bending of structural components, but it performed its intended functions. Post-test disassembly of selected batteries showed that some corrosion of the weld joint between the positive plates and the bus/terminal assembly had occurred as a result of the natural aging process. However, this degradation did not interfere with the seismic performance. Metallurgical examinations showed that a large grain structure existed at the weld area. The larger grain structure of the weld makes it susceptible to corrosion and would explain the observed corrosion.

The results indicate that measurements of capacitance and internal resistance can be obtained with repeatability and may provide an indication of battery condition if the measurements were taken over the lifetime of the battery. Polarization and discharge current interruption are two techniques that are capable of measuring internal resistance, while discharge current interruption is also capable of measuring battery capacitance. These measurements would be most useful if they could be made while the batteries were new and then repeated at regular intervals to obtain a pattern of the change in battery characteristics with time.

The results of seismic tests on naturally aged batteries nearly 14 years old showed that, when batteries are maintained and operated in accordance with IEEE Std 450 and Regulatory Guide 1.129, the following may be expected of adequately designed and manufactured lead-calcium batteries:

1. Little, if any, electrical capacity will be lost as a result of seismic shaking at levels that are typical of the most severe SSE levels specified for U.S. nuclear plants.

2. Some internal damage to the plate separators may be expected at the most severe seismic levels. However, this minor loss of seismic ruggedness is not expected to prevent the batteries from providing at least 80% of rated capacity during and immediately following the most severe seismic event.

3. Naturally aged batteries may show evidence of corrosion at the joint between the positive

plates and the positive plate strap (bus). In a well-made joint, this corrosion should not cause the seismic ruggedness to be inadequate for the most severe SSE events expected in the U.S. Operation of batteries at elevated temperature or excessive charging could increase the corrosion, which could then progress rapidly enough to result in inadequate seismic ruggedness.

NUREG/CR-5461, M.J. Jacobus. "Aging of Cables, Connections, and Electrical Penetration Assemblies Used in Nuclear Power Plants," Sandia National Laboratories, SAND89-2369, July 1990.

This report covers the examination of the effects of aging on cables, connections, and containment electrical penetration assemblies (EPAs) as part of the NRC NPAR program. Cables and connections are used in every electrical circuit in all nuclear power plants. EPAs are included in every circuit that is inside containment. This NRC-sponsored aging assessment of cables, connections, and penetrations is divided into two phases. Phase I, which is the subject of this report, consists of a review of applicable literature and evaluations of usage, operating experience, and current inspection and surveillance methods. Phase II, currently planned only for cables, includes the development of improved methods for inspection, surveillance, and monitoring; application of monitoring methods to naturally aged and in situ cables; and recommendations for utilizing the research in the regulatory process.

This report includes a review of component usage in nuclear power plants, a review of some commonly used components and their materials of construction, a review of the stressors that the components might be exposed to in both normal and accident environments, a compilation and evaluation of industry failure data, a discussion of component failure modes and causes, a description of current industry testing and maintenance practices, and a review of some monitoring techniques that might be useful for monitoring the condition of these components.

The conclusions of the study are:

1. Cables, connections, and EPAs are highly reliable devices under normal plant operating conditions with no evidence of significant increases in failure rate with aging. Consequently, they receive little or no preventive maintenance. Under accident conditions, however, the reliability of these components is practically unknown.
2. Aging effects that have the potential to lead to common-cause failures during accident conditions have the highest significance.
3. Many of the causes of failures for cables, connections, and EPAs at accident conditions would not cause any detectable manifestations during normal operation because of the absence of high temperatures and humidities. The most important failure mode is expected to be shorting (or reduced electrical isolation). Several different causes may result in this failure mode.
4. Plant operational experience is useful to the extent that it may indicate some possible fast-acting degradation mechanisms for cables, connections, and EPAs that could lead to common-cause failures under off-normal environmental conditions. However, current LER data provide a very limited data base for this purpose.
5. A significant number of manufacturers have produced cables, connections, and EPAs, resulting in many different materials, designs, and construction methods. Consequently, generic assessments of aging effects and vulnerabilities have become much more difficult, particularly where failure modes relate to interfacing stresses.

An experimental assessment of cables is currently under way and will be documented in a future report.

NUREG/CR-5479, B. Damiano and R. C. Kryter, "Current Applications of Vibration Monitoring and Neutron Noise Analysis: Detection and Analysis of Structural Degradation of Reactor Vessel Internals from Operational Aging," Oak Ridge National Laboratory, ORNL/TM-11398, February 1990.

The detection of degradation in PWR internals due to operational aging is becoming more and more important to U.S. utilities as the median age of U.S. nuclear power plants increases. Monitoring and detection of aging effects should aid in justifying plant life extension and result in safer and more efficient operation during the present and extended life period. It has been demonstrated that monitoring programs based on neutron noise and vibration measurements utilizing signature analysis can effectively detect, and in some cases diagnose, degradation of reactor vessel internals. Such programs have the potential to reduce plant downtime, make periodic maintenance more effective, and increase plant safety.

Monitoring of reactor internals can be considered a particular application of the general concept of predictive maintenance, the techniques of which are already widely used in industry to monitor rotating machinery. Predictive maintenance will be further implemented as (1) its benefits become better documented, (2) familiarity with the techniques and their applications grows, and (3) better hardware and software become available. A similar statement could apply to the monitoring of reactor internals. Although this monitoring has been spotty in the U.S., the above-mentioned techniques have been widely applied in Europe, particularly in France and the Federal Republic of Germany, where they are currently (in 1989) 5 to 10 years ahead of those in this country. U.S. utilities could benefit from the experience in Europe, where, in many cases, internals monitoring has been integrated into regular plant maintenance programs. Thus U.S. utilities could implement effective monitoring programs with a minimum of experimentation and wasted effort.

The report begins with a description of some prominent mechanisms through which degradation of reactor internals occurs; the cause of most cases of this degradation is flow-induced vibration. Other mechanisms are also reviewed. This is followed by a brief description of vibration monitoring and neutron noise analysis, including a comparison and evaluation of these two methods. Next, current practices are summarized, and examples of applications of these methods in both the U.S. and Europe (mainly West Germany and France) are given. The report concludes with guidelines for setting up what the authors consider to be a reasonable internals-monitoring program for U.S. utilities.

NUREG/CR-5490, E. V. Werry, "Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components," Vol. 1, Pacific Northwest Laboratory, PNL-7190, October 1990.

This report is the first volume of a review of U.S. nuclear plant regulatory instruments to determine the amount and kind of information they contain on managing the aging of safety-related components in U.S. nuclear power plants. The review was conducted for the U.S. Nuclear Regulatory Commission (NRC) by the Pacific Northwest Laboratory (PNL) under the NRC Nuclear Plant Aging Research (NPAR) program. Eight selected regulatory instruments, including regulations, regulatory guides, technical specifications, standards, Code of Federal Regulations and others, were reviewed for safety-related information on five selected components: reactor pressure vessels, steam generators, pressurizers, primary piping, and emergency diesel generators. Volume 2 is tentatively scheduled for FY 1994, and it will cover selected major safety-related components, e.g., pumps, valves and cables.

The focus of the review was on 26 NPAR-defined safety-related aging issues, including examination, inspection, maintenance and repair, excessive/harsh testing, and irradiation embrittlement. The major conclusion of the review is that safety-related regulatory instruments do provide implicit guidance for aging management, but include little explicit guidance. The major recommendation is that the instruments be revised or augmented to explicitly address the management of aging.

NUREG/CR-5491, R. P. Allen and A. B. Johnson, Jr., "Shippingport Station Aging Evaluation," Pacific Northwest Laboratory, PNL-7191, January 1990.

This report describes a research plan to address safety concerns on aging of snubbers used on piping and equipment in commercial nuclear power plants. The work is to be performed under Phase II of the Snubber Aging Study of the NRC NPAR program with the Pacific Northwest Laboratory (PNL) as the prime contractor. Research conducted by PNL under Phase I provided an initial assessment of snubber operation based primarily on a review of licensee event reports. The work proposed is an extension of Phase I activities and covers research at nuclear power plants and in test laboratories. The report includes technical background on the design and use of snubbers in commercial nuclear power applications and a discussion of the primary failure modes of both hydraulic and mechanical snubbers. The anticipated safety, technical, and regulatory benefits of the work, along with concerns of the NRC and the utilities, are also subjects of the report.

The Shippingport Atomic Power Station, presently (1989) in the final stages of decommissioning, has been a major source of naturally aged equipment for the NPAR and other NRC programs. The evaluation of naturally aged components is an element of the NPAR program strategy. Because naturally aged components and materials experience the actual service-related external stressors, corrosion and wear, testing procedures, and maintenance practices, the evaluation of such components is valuable. One is able to verify degradation models, to validate aging projections based on the extrapolation of accelerated test data, and to detect unexpected aging mechanisms (surprises) that could significantly affect the safety performances of components or systems.

Despite their importance for plant studies, naturally aged components of the desired type and vintage are not readily available. The best source of these components is operational equipment from retired plants. The decommissioning of the Shippingport Station, particularly because it was managed by the U.S. Department of Energy, represents a valuable opportunity to conduct in situ assessments at an aged reactor and to obtain a variety of naturally aged and degraded components and samples for detailed aging evaluations by NRC contractors. As the first U.S. large-scale, central-station nuclear plant, the Shippingport Station parallels commercial pressurized water reactors in reactor, steam, auxiliary, support, and safety systems. The 25-year service life (1957 to 1982) covers almost the entire period of currently operating reactors. Also, because of substantial modifications during the mid-1960s and 1970s, it offers unique examples of identical or similar equipment used side by side with the original equipment but representing different vintages and degrees of aging. As part of the Shippingport Station aging evaluation work, more than 200 items, ranging in size from small instruments and material samples to main coolant pumps, have been removed and shipped to designated laboratories. These items include battery chargers, inverters, relays, breakers, switches, power and control cables, electrical penetrations, check valves, solenoid valves, and motor-operated valves. Samples of piping from various plant systems also have been acquired for radiological characterization studies, and samples from the primary system components will be used for material degradation studies.

Data and records relevant to the procurement, operation, and maintenance of these materials and components have been obtained to support the detailed aging evaluations. In situ assessments of Shippingport Station components also have been conducted, including preremoval visual and physical examinations of components, tests of electrical circuits, and special measurements to assist in the selection of specific components for further evaluation. Although detailed evaluations of the naturally aged components and material from the Shippingport Station have not been completed, the results from preliminary studies indicate the value of the aging information that may ultimately be obtained.

NUREG/CR-5507, W. Gunther and J. Taylor, "Results from the Nuclear Plant Aging Research Program: Their Use in Inspection Activities," Brookhaven National Laboratory, BNL-NUREG-52222, September 1990.

The NRC NPAR program has determined the susceptibility of nuclear power plant components and systems to aging and the potential for aging to affect plant safety and availability. The program has also identified methods for detecting and mitigating the effects of aging in components. A review of the NRC Inspection Program and discussions with NRC inspection personnel revealed several areas where the NPAR results would be valuable to the inspectors. This report describes the NPAR information that can enhance inspection activities and provides recommendations for communicating this information to NRC inspectors. These recommendations are based on a detailed assessment of the NRC Inspection Program and on feedback from resident and regional inspectors.

The emphasis of the NRC Inspection Program is on evaluating the performance of licensees by focusing on requirements and standards associated with the administrative, managerial, engineering, and operational aspects of their activities. The Inspection Program recognizes that licensees may satisfy NRC requirements in ways that differ among the licensees, and inspection guidance is therefore expressed in the form of performance objectives and evaluation criteria. For the resident and regional inspectors, procedures covering such subjects as operations, maintenance, and surveillance have been written. Some of these procedures contain guidance on degradation due to aging.

Associated with each NPAR study is the need to determine the role of inspection, maintenance, and monitoring in counteracting the effects of aging and service wear. The role of maintenance in managing aging is an important area where NRC emphasis has been applied. A review by the NRC of maintenance performed at several plants concluded that most utilities do not perform condition monitoring because of inadequate knowledge of degradation mechanisms and measurable condition-indication parameters. The output from NPAR in this area could provide information needed to assist the inspectors to recognize age-related concerns.

The types of information generated by NPAR that were found to be relevant to inspection needs include:

1. *Functional indicators*—NPAR reports identify parameters that can be monitored or measured to detect aging degradation. The inspectors can apply these results to enhance visual inspections (walkdowns) and to evaluate licensee programs for ensuring the operability of equipment and systems.

2. *Failure modes, causes, effects*—Operating experience data evaluated in NPAR studies can alert the inspectors to the prevalent failure mechanisms of systems and equipment. The potential for changes in failure rate with increasing age is useful in evaluating preventive maintenance.

3. *Stresses that cause degradation*—Inspectors can benefit from knowing the environmental and operational stresses that cause or affect degradation due to aging.

To obtain a complete delineation of the NRC inspectors' needs, presentations summarizing the results of the NPAR program were made to the resident inspectors at three regions. Their comments, supplemented by a written questionnaire, indicated that NPAR results can be of use to the inspectors when provided in a format directed to their activities. Examples of NPAR report summaries and inspection guides for aging-related degradation of components and systems are included in the report.

NUREG/CR-5510, W.E. Vesely, R.E. Kurth, and S.M. Scalzo, "Evaluations of Core Melt* Frequency Effects Due to Component Aging and Maintenance," Science Applications International Corporation, SAIC-89/1744, June 1990.

This report presents the results of a project to develop a methodology using probabilistic risk analysis (PRA) and component aging models to quantify risk effects due to component and structural aging. The approach allows any present PRA and any aging model for the components and structures to be used. An important part of the evaluations is that the effects of maintenance and surveillance programs in controlling aging can be quantified. These programs can be explicitly evaluated to determine their effectiveness in controlling aging impacts on system unavailability, core damage frequency, and public risk. Both point evaluations and uncertainty evaluations can be carried out, and detailed contributors to the aging effects can be identified and prioritized. PRA models are separated from the aging models, allowing available PRAs to be efficiently used in evaluating risk effects of aging.

To demonstrate the methodology, two PRAs, one for a PWR and one for a BWR, were used to calculate the increase in core damage frequency caused by aging for given aging data and assumed surveillance and maintenance programs. The increase in core damage frequency due to aging was averaged over time. This average increase in core damage frequency characterized the effectiveness of the maintenance and surveillance program in controlling aging effects. The average increase in core damage frequency can be added to the baseline PRA core damage frequency to obtain the total projected core damage frequency under a given maintenance and surveillance program with the acting aging process.

The aging of active components was modeled using the linear failure rate aging model in which the component failure rate linearly increases with age according to a characteristic aging rate. To demonstrate the methodology, four aging rate data bases were used: TIRGALEX, MOD1, MOD2, and MOD3. These data bases demonstrated the effects of different aging rates on the core damage frequency for a given maintenance and surveillance program.

Results obtained for different surveillance and maintenance programs clearly show the sensitivity of the increase in core damage frequency to the type of maintenance program and the aging rates. The results are significant from a technical standpoint because they explicitly quantify the impacts that aging and maintenance can have. These evaluations are the first quantifications of aging and maintenance impacts using full-scale up-to-date PRAs.

When the increase in core damage frequency is large for a given surveillance and maintenance program, examination of the detailed aging contributors shows that relatively few components contribute. This implies that a "graded" maintenance program or, equivalently, a "prioritized" maintenance program can effectively control the core damage frequency increase due to aging. In such a maintenance program, most components can have a lower level of maintenance if components important to core damage frequency have a higher level of maintenance.

The dominant aging contributors for the PWR were found to be diesel generators, specific check valves and motor-operated valves in the emergency core cooling system, and motor-driven pumps and turbine-driven pumps in the auxiliary feedwater system. For the BWR, the dominant aging contributors were the diesels, the motor-driven pumps in the service water system, and the turbine-driven pumps in the reactor core isolation system. The aging contribution from every component in the PRA is provided and prioritized. These detailed contributors include specific systems, components, and failure modes and provide a comprehensive means of focusing aging analyses and aging control efforts.

In addition to the point calculations, uncertainty evaluations were carried out. For these evaluations, ranges were assigned to each component aging rate, each effective overhaul interval, and each effective surveillance interval. These ranges described uncertainties and variations in the data. Log-uniform distributions, which are flat distributions on a logarithmic scale, were used for the uncertainty propagation. All the variables were treated as being independent of one another for the evaluations.

NUREG/CR-5515, H.H. Neely, N.M. Jeanmougin, and J.J. Corugedo, "Light Water Reactor Pressure Isolation Valve Performance Testing," Energy Technology Engineering Center, July 1990.

The Light Water Reactor Valve Performance Testing Program was initiated by the NRC to evaluate leakage as an indication of valve condition, provide input to Section XI of the ASME Code, and evaluate emission monitoring for condition and degradation and inservice inspection techniques. Six typical check and gate valves were purchased for testing at typical plant conditions (550°F at 2250 psig) for an assumed number of cycles for a 40-year plant lifetime. Tests revealed that there were variances between the test results and the present statement of the Code; however, the testing was not conclusive. The lifecycle tests showed that high tech acoustic emission can be utilized to trend small leaks, that specific motor signature measurement on gate valves can trend and indicate potential failure, and that inservice inspection techniques for check valves were shown to be both feasible and an excellent preventive maintenance indicator. Lifecycle testing performed

*The current NRC terminology uses "core damage" (as in the accompanying summary) instead of "core melt."

here did not cause large valve leakage typical of some plant operation. Other testing is required to fully understand the implication of these results and the required program to fully implement them.

NUREG/CR-5519, J.C. Moyers, "Aging of Control and Service Air Compressors and Dryers Used in Nuclear Power Plants," Oak Ridge National Laboratory, ORNL-6607/V1, July 1990.

This report discusses work performed as part of the NRC NPAR program on practical and cost-effective methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of compressors and dryers used in the control and service air systems of nuclear power plants. The objective is to provide capabilities for establishing degradation trends prior to failure and for developing guidance on effective maintenance programs.

The topics covered are failure modes and causes resulting from aging and service wear, manufacturer-recommended maintenance and surveillance practices, and measurable parameters (including functional indicators) for use in assessing operational readiness and equipment condition (often related to degradation trends) and in detecting incipient failure. The results are based on information derived from operating experience records, manufacturer-supplied information, and inputs from plant operators. For each failure mode, failure causes are listed by subcomponent, and potentially useful parameters for detecting degradation that could lead to failure are identified.

A brief review of typical compressors and dryers in nuclear power plants showed that the nonlubricated reciprocating compressors and the regenerative desiccant dryer are used in more plants than any other types for both service and control air systems, and the assessment was therefore focused on them. A general description of the equipment that includes illustrations, defined equipment boundaries, functional requirements, and materials of construction is provided. Operational stressors are categorized and listed in detail.

Data bases and nuclear industry reports containing nuclear power plant operating experience were examined. These data bases included the Licensee Event Report (LER) file as cataloged in the Sequence Coding and Search System maintained by ORNL's Nuclear Operations Analysis Center, the Nuclear Power Experience compilation maintained and published by the S.M. Stollers Corporation, the In-Plant Reliability Data System containing maintenance records for one plant, and maintenance records obtained from a cooperating utility for a second plant. During the 1978–1988 decade covered by the LER data, which represents approximately 812 reactor-years, 22 compressor-related and 16 dryer-related events that resulted in loss of control air supply were reported. Equipment failure causes were diverse, with no single type of failure dominating. The records available for the two commercial plants indicated a significant preventive and corrective maintenance effort to take care of service wear and provide reliable equipment operation.

Maintenance recommendations included in operating and maintenance manuals provided by equipment manufacturers were reviewed and compared to the preventive maintenance practices at one plant. The user-applied practices generally were in conformance with or exceeded the manufacturers' recommendations. One troublesome aspect is ensuring the operational readiness of auxiliary compressors that are normally idle for long periods but must provide backup service for critical needs if the main control air supply deteriorates. Manufacturer-recommended mothballing procedures do not appear practical for this application; such failure causes as drive belt set, corrosion of internal parts, and small internal water leaks may present a problem when the compressor is needed.

Measurable parameters that have a potential for enhancing the capabilities for detecting incipient failures and examining degradation trends in compressors and dryers were identified. For compressors, they include periodic delivery capacity tests, trending of stage temperatures and pressures, and motor current signature analysis. Measurable parameters for dryers include moisture sensing within the desiccant column near the exit and periodic monitoring of the axial temperature profile within the column. Use of these measurable parameters in the surveillance and monitoring program might reduce the level and duration of time-directed out-of-service inspection and maintenance, thereby increasing availability and improving overall system reliability.

Nuclear plant control and service air compressors and dryers are not usually considered as safety related because the air systems are not needed to bring the plant to a safe shutdown condition. An effective surveillance and monitoring program with preventive and corrective maintenance can provide reliable service from nuclear plant compressors and dryers. Instances of loss of air supply due to compressor or dryer failure are rare because of the redundancy in most systems. For these reasons, it is recommended that no further consideration of this equipment be included in the NPAR program.

NUREG/CR-5546, S. P. Nowlen, "An Investigation of the Effects of Thermal Aging on the Fire Damageability of Electric Cables," Sandia National Laboratories, SAND90-0696, May 1991.

This report describes the results of a series of tests performed to assess the effects of thermal aging on the vulnerability of cables to fire-induced thermal damage. The tests were part of an effort in support of the NRC NPAR program to identify and investigate fire safety issues for which plant aging might lead to an increased level of risk.

From the standpoint of fire safety, cables represent the single most important class of electrical equipment in a nuclear power plant. First, virtually every plant system includes power, control, and instrumentation cables. Second, cable "pinch" points (that is, locations where redundant train separation is reduced by the merging of cable routings) often represent dominant contributors to plant fire risk as determined by probabilistic risk assessment (PRA) analyses. Third, cables represent the major combustible fuel loading for most plant areas.

The tests described here examined the thermal damageability of two commonly used types of low-flame-spread electric cables qualified to IEEE-383:

1. A Neoprene-jacketed, cross-linked-polyethylene-insulated (XPE), three-conductor, 12 AWG, 600V light power or control cable produced by the Rockbestos Corporation and marketed under the trade name Firewall III.
2. An ethylene-propylene-rubber-insulated (EPR), chlorosulfonated-polyethylene-jacketed (CSPE or Hypalon), two-conductor, 16 AWG, plus shield and drain, 600V instrumentation or signal cable produced by BIW Cable Systems Incorporated and marketed under the trade name Bostrad 7E.

For each of the two cable types, both unaged (i.e., new from the cable reel) and thermally aged samples were tested. No radiation aging was employed in these tests.

The exposure conditions simulated during testing were considered typical of those expected during an enclosure fire when the subject cables are not involved in the fire itself. The most significant difference between the test exposures and anticipated actual exposures was that the tests involved exposure at an elevated steady-state temperature whereas, in actual exposures, equipment would experience a transient time/temperature exposure.

In these cable exposure tests, the walls of the chamber and the air were preheated to the desired uniform steady-state exposure temperature. Two energized cable samples were then quickly inserted through a small door to provide a near step change in environment temperature for the cable samples.

The cable samples were energized by a three-phase 208-volt power source. Each of the three conductors of the Rockbestos cables was connected to one phase of the power source. In the case of the BIW cable, the two conductors and the drain conductor were each connected to one phase of the power source. Leakage currents between power phases were monitored continuously. The time to ultimate cable failure, as determined by the failure of a two-ampere fuse in any one of the three phase circuits, was also recorded. Two measures of thermal damageability can be made based on these tests.

One measure of fire damageability is the thermal damage threshold defined, in the context of these tests, as a temperature range. Its upper limit is the lowest temperature at which electrical failure was observed following exposures of up to 80 minutes. The lower limit is the highest temperature for which no electrical failures were noted following exposures of no less than 80 minutes.

For the Rockbestos cable, the failure threshold of the unaged cable was determined to be 325-330°C, whereas the thermal damage threshold for the aged samples was 350-365°C. For the BIW cable, the thermal damage threshold of the unaged cable was estimated at 365-370°C, whereas that of the aged samples was estimated at 345-350°C. Thus the aging process resulted in the opposite effect on the thermal damage threshold for the two cable products. For the Rockbestos cable, aging increased the damage threshold by approximately 25-35°C while, for the BIW cable, it decreased the threshold by approximately 20°C.

A second measure of thermal damageability is the relative time to failure for exposure temperatures above the damage threshold. The aged Rockbestos samples consistently displayed longer times to failure at a given temperature than did the unaged samples, indicating less vulnerability to thermal damage for the aged samples. The time to failure for the aged and unaged BIW samples was not significantly different for exposure temperatures at which failure was observed in both aged and unaged samples.

It was also noted that, in virtually every case, failure of the cables through conductor-to-conductor shorting resulted in the initiation of intense, sustained, open flaming in the cable samples. As the cables shorted, sparks ignited the gases evolved from the cables. In no case was spontaneous ignition of the cables observed prior to electrical failure. These results indicate that the failure of energized cables is a mechanism for fire spread.

The thermal damage threshold changes observed in the tests on two of the most common nuclear qualified cables in current use in the U.S. nuclear industry are not considered of sufficient magnitude to significantly alter risk estimates for scenarios involving cable thermal damage.

It should be noted that these tests have not explored the impact of other fire environment effects such as suppressant application and high humidity on cable survival. The failure thresholds given above pertain to gross electrical failure. In most cases, significant levels of current leakage were noted prior to gross failure, and specific applications must be examined to determine whether such leakage could constitute the failure of a circuit to perform its design function. Also, because mixed results were obtained for the two cable types tested, no direct conclusion regarding the impact of thermal aging on the fire vulnerability of any other cable type can be drawn based solely on the results of these tests.

NUREG/CR-5555, W. Gunther and K. Sullivan, "Aging Assessment of the Westinghouse PWR Control Rod Drive System," Brookhaven National Laboratory, BNL-NUREG-52232, March 1991.

A study of the effects of aging on the Westinghouse control rod drive (CRD) system was performed as part of the NRC NPAR program. Its objective was to provide a technical basis for identifying and evaluating the degradation due to aging.

The Westinghouse CRD system consists of control rods and the mechanical and electrical components that control the rod motion. The study examined the design, construction, maintenance, and operation of the system to assess its potential for degradation as the nuclear plant ages and evaluated the extent to which aging could affect the safety objectives of the system. Studies are also being conducted for the Combustion Engineering, Babcock and Wilcox, and General Electric CRD systems.

The operating experience for CRD systems as documented in the Licensee Event Reports (LERs), Nuclear Plant Reliability Data System (NPRDS), and Nuclear Power Experience (NPE) data bases was reviewed. These sources provided an average of 30 unique failure events per year over the last 10 years, of which approximately 35% were directly attributable to aging-related degradation. The review resulted in the following observations:

1. The majority of the reported failures occurred in the electrical area, i.e., the power and logic cabinets, and the rod-position indication subsystem.
2. Approximately 40% of the reported failures resulted in a rod drop, which usually challenges the reactor protection system and initiates a reactor trip.
3. Several failure modes such as rod position drift and overheating of power cabinets are common to many plants, which could indicate the need for generic resolutions.

The normal operating and environmental stresses experienced by the system components were assessed to determine their effect on the long-term performance of the system. For example, the regular stepping action associated with control rod motion results in wear of the latch and drive rod components and in electrical surges on the control rod drive mechanism coils. The amount of latch wear measured in another study is presented in this report along with results of other research efforts related to aging of the CRD system. Other examples of stressors associated with aging-related degradation are high temperature, flow-induced vibration, and particle debris carried by the coolant.

A failure modes and effects analysis of the Westinghouse control rod drive system was also conducted, and components with a high safety significance were identified along with the likelihood of their failure. This assessment was based on operating experience data and an evaluation of the susceptibility of the components to age-related degradation. Several components that should receive attention as a plant ages were identified: cables, coils, and connectors (in containment); latch assembly; guide tube; and selected electronics within power and logic cabinets, including the rod-position indicating system.

An evaluation of inspection, surveillance, monitoring, and maintenance was accomplished with information from fifteen plants representing ten utilities. Responses from most plants agreed that two of the required technical specification tests (rod-drop timing and rod excercising) are beneficial in verifying the operational readiness of the system. Preventive maintenance activities for electrical components within containment dominate the overall maintenance of this system. Only a few plants are using circuit-monitoring techniques or nondestructive testing to monitor the long-term operational characteristics of the CRD system (a substantial portion of that system is considered not related to safety). The responses from plants further indicate that some plants have modified the system, replaced components, or expanded preventive maintenance. Several of these activities have effectively addressed the aging issue. However, maintenance practices appear to vary from one plant to another, possibly reflecting inadequacies at some plants.

Techniques to detect and mitigate the effects of aging, including advanced approaches by Westinghouse, the Japanese, and the French are described. Previous research related to the Westinghouse CRD system was discussed, e.g., NUREG-0641 on wear of control rod guide tubes, study E613 on localized wear from the NRC Office of Analysis and Evaluation of Operational Data, and EPRI-sponsored reports on plant life extension and control rod lifetime determination. Research on the extension of plant life for a Westinghouse PWR, for example, identifies the latch, drive rod, and coil stack assemblies as limited-life components.

The findings and recommendations of this aging study may be summarized as follows:

1. Aging-related degradation of the Westinghouse CRD system can compromise the intended function of the system. Therefore, means to detect and mitigate this degradation in the safety- and non-safety-related portions of the system should be pursued.
2. The test requirements on the system (e.g., rod-drop timing) are important in determining the operational readiness of the system, although

they cause some incremental wear on the mechanical part of the system.

3. The preventive maintenance, including inspection and testing of the in-containment cables, connectors, and coils should be increased as these components age. The use of such predictive maintenance concepts as nonintrusive on-line monitoring techniques should be considered. The method used to perform the rod-drop timing test should be modified, or the degradation that can result from the present procedure of pulling the fuses should be accounted for.

4. The logic associated with the speed and motion control of the CRD system is complex. Maintenance errors in this area have resulted in unnecesssary reactor trips and additional stress to the CRD system. Repair and replacement procedures for this portion of the system should be evaluated for completeness and accuracy, and personnel training should be emphasized.

NUREG/CR-5558, P. Steele, Jr., K.G. DeWall, and J.C. Watkins, "Generic Issue 87, Flexible Wedge Gate Valve Test Program: Phase II Results and Analysis," Idaho National Engineering Laboratory, EGG-2600, January 1991.

Qualification and flow isolation tests were conducted to analyze the ability of selected boiling water reactor (BWR) process valves to perform their containment isolation functions at high-energy pipe break conditions and other more normal flow conditions. Numerous parameters were measured to assess industry practices for predicting valve and motor operator requirements. The valves tested were representative of those used in BWR reactor water cleanup systems and high-pressure coolant injection (HPCI) steam lines. Among the objectives of this research program are to determine what factors affect the performance of motor-operated gate valves and to determine how well industry's analytic tools predict that performance.

This program supports the NRC's effort on a generic issue, GI-87, "Failure of HPCI Steam Line Without Isolation." GI-87 covers three boiling water reactor process lines: the HPCI turbine steam supply line, the reactor isolation cooling (RCIC) turbine steam supply line, and the reactor water cleanup (RWCU) process line. All three of these process lines communicate with the primary system, pass through containment, and have normally open isolation valves. The concern with the isolation valves is whether they will close in the event of a pipe break outside of the containment. A release of high-energy steam or hot water in the auxiliary building could result in common-cause failure of other components necessary to mitigate the accident.

One of the major parts of the research program included two full-scale qualification and flow interruption test programs on flexible-wedge gate valves, Phases I and II. The Phase II program was performed in 1989 at the Kraftwerk Union (KWU) facilities near Frankfurt, Germany. Among the valves tested, three were 10-in. valves typical of those used in the HPCI applications. One of the 6-in. valves was also tested at RCIC test conditions. In all, seventeen flow interruption tests were performed, seven at design basis conditions.

Two RWCU valves were tested during the earlier Phase I Test Program. As a result of that work, it was expected that the valves would require more stem force to close than indusry normally would have predicted. Therefore, for the Phase II Program, the motor-operator control switches were set at higher-than-normal torque values to ensure valve closure, and the strengths and weaknesses of a given valve design were determined from the recorded data.

The test results clearly showed that, for the GI-87 concerns, all valves that were subjected to design basis flow interruption tests required more torque and subsequently more stem force to close than would be predicted using the standard industry motor-operator sizing equation for disk load calculations with a common coefficient of friction. The highest loads recorded were the result of internal valve damage caused by the high-differential-pressure loads across the valve disk as it attempted to stop the flow.

The high loads encountered during the test series raise the concern that some valves installed in nuclear power plants may not have large enough motor operators to ensure closure in the event of a design basis accident.

The study into the phenomena affecting the stem loads in a motor-operated gate valve continues. However, the results to date indicate that the phenomena taking place inside the gate valve are more complex than previously thought. The actual disk factor is much higher than previously believed, but this factor can be moderated for some valve applications once the self-closing force balance on the valve disk is understood.

Physical inspection indicated that these valves were very near their physical fragility limits at design basis conditions. The excessive bearing pressure between the disk and the body guide materials resulted in yielding, spalling, and gouging of the surfaces. In some of the designs, the guide clearances were large enough to allow the disk to tilt during closure, which resulted in significant damage to the sealing surfaces.

NUREG/CR-5560, H.M. Hashemian, D.D. Beverly, D.W. Mitchell, and K.M. Petersen, "Aging of Nuclear Plant Resistance Temperature Detectors," Analysis and Measurement Services Corporation, June 1990.

A comprehensive research and development project on aging of narrow-range resistance temperature detectors (RTD) used in the primary coolant system of pressurized water reactors was carried out as part of the NRC NPAR program. The goal was to establish the long-term performance limits of these RTDs in order to verify that objective and adequate measures are implemented to ensure safety.

The project was conducted in two phases. Phase I, a six-month feasibility study, was completed in June 1987. The results, published in NUREG/CR-4928, "Degradation of Nuclear Plant Temperature Sensors," demonstrated the need for additional work in Phase II. This report presents the results of Phase II, which was conducted over a 30-month period beginning in October 1987. The work involved laboratory testing of 72 nuclear grade RTD elements representing several from each of four U.S. manufacturers. The limit for the initial accuracy of these RTDs was established, and a procedure for performing precise calibration was developed. Experimental aging of 30 of these RTDs at simulated reactor conditions resulted in five failures and six major calibration shifts. Two failures occurred in thermal aging, one in vibration aging, one in humidity aging, and one in thermal cycling. The remaining 19 RTDs performed well during the aging tests, maintaining a drift band of $\pm 0.2°C$.

The shelf-life drift of RTDs was also quantified. This involved testing 45 RTDs for storage effects: 24 that had been in normal storage at various nuclear power plants for periods of one to five years and 21 that were aged in the project. The test results for these 45 RTDs showed a shelf life drift band of $\pm 0.1°C$. Most of the storage drifts, the failures, and the normal aging drifts were found to occur in the first few months of aging. A potential remedy is to burn in the RTDs before they are calibrated and installed in the plant.

The performance of nuclear plant RTDs is evaluated by response-time testing in addition to calibration. These two procedures are independent and are therefore done separately. The nuclear industry has about ten years of experience with RTD response time resulting from periodic in situ measurements made in about 60 PWRs at least once in every fuel cycle. Representative results of these measurements were reviewed to identify the range of achievable response times and the response time degradation modes.

Several commercial grade RTDs were also aged and tested for comparison with nuclear grade RTDs. The results showed that the average response time and calibration stability of nuclear grade RTDs is about twice as good as that of the commercial grade RTDs.

The project addressed the following additional topics: sources of errors in RTD calibration, factors affecting RTD accuracy and response time, failures of RTDs as reported in the LER and NPRDS data bases, and the International Temperature Scale of 1990 and its impact on temperature measurements in nuclear power plants. The results of research performed in Phase II did not reveal any unanticipated or major systematic aging problem in the performance of the RTDs tested. The nuclear industry's practice for verifying adequate RTD accuracy and response time is to perform on-line cross calibration and loop current step response tests at least once every fuel cycle. In light of the data obtained throughout this study, this approach is reasonable for managing the aging of RTDs that do not have any major design, fabrication, or installation deficiencies. RTDs that consistently maintain a suitable calibration and reponse time as determined by periodic testing can be used in the plant for their qualified life as specified by the manufacturer. The manufacturers' specifications for the qualified life of nuclear grade RTDs typically range from 10 to 40 years depending on the manufacturer and the conditions at which the RTDs are used.

NUREG/CR-5583, M.S. Kalsi, C.L. Horst, J.K. Wang, and V. Sharma, "Prediction of Check Valve Performance and Degradation in Nuclear Power Plant Systems – Wear and Impact Tests. Final Report, September 1988-April 1990," Kalsi Engineering, Inc., KEI No. 1656, August 1990.

Check valve failures at nuclear power plants in recent years have led to serious safety concerns and have caused extensive damage to other plant components that had a significant negative impact on plant availability. Swing check valve internals may experience premature deterioration if the disk is not firmly held open against its stop. At the present time, no guidelines exist for the prediction of degradation trends and the determination of suitable inspection intervals. A research program aimed at developing a reliable model for quantitative predictions of wear and fatigue for swing check valves was established as part of the NRC NPAR program to improve the safety and reliability of their operation. This report covers Phase II of the research on swing check valves. The work in Phase I was published in NUREG/CR-5159.

The goal of Phase II was to develop predictive models that could be used to quantify the degradation of swing check valves with flow disturbances close upstream of the valve at flow velocities that do not result in full disk opening. Two major causes of swing check valve failure are premature degradation due to wear in the hinge pin and fatigue in the disk stud connection to the hinge arm. Accelerated wear tests were performed using aluminum hinge pins and bushings in 3-inch and 6-inch valves to quantify wear experienced in the hinge pin area. A special disk instrumented with strain gages was used in the 6-inch valve to measure the impact forces and their rate of occurrence to quantify the fatigue damage caused by the disk tapping against the stop. The wear and fatigue prediction models developed in this program show good correlation with laboratory test results as well as with a limited number of check valve failures at plants that had been sufficiently documented.

The results of this research allow inspection and maintenance activities to be focused on those check valves that are more likely to suffer premature degradation. The methodology for quantitative prediction of wear and fatigue can be used to develop a sound and effective preventive maintenance program. The results also indicate certain modifications in the valve design that may improve check valve performance and reliability.

NUREG/CR-5587, W.E. Vesely, "Approaches for Age-Dependent Probabilistic Safety Assessments with Emphasis on Prioritization and Sensitivity Studies," Science Applications International Corporation, SAIC-92/1137, August 1992.

This report describes approaches for incorporating component aging reliability models into a probabilistic safety assessment (PSA) or probabilistic risk assessment (PRA) of a nuclear power plant. These approaches and procedures are described from a technical standpoint and are not to be interpreted as having any regulatory implications. Component aging failure rate models and test and maintenance aging control models are presented. Different approaches for carrying out the aging evaluations are given. Demonstrations are given involving prioritizing aging contributors, evaluating maintenance effectiveness, carrying out time-dependent evaluations, and carrying out uncertainty and sensitivity analyses of aging effects.

NUREG/CR-5612, P.K. Samanta, W.E. Vesely, F. Hsu, and M. Subudhi, "Degradation Modeling with Application to Aging and Maintenance Effectiveness Evaluations," Brookhaven National Laboratory, BNL-NUREG-52252, March 1991.

An important element of the assessment of risk associated with aging in nuclear power plants is the understanding of the aging phenomena associated with components of safety systems. This report describes a study of aging phenomena at the component level in support of the NRC NPAR program to develop an aging reliability model representing the aging process experienced by components in nuclear power plants under presently existing test and maintenance practices. A new model was developed to process information on component degradation in order to analyze the degradation process and its implications. The focus was on modeling the degradation rate, i.e., the rate at which degradations occur, with the specific objective of developing explicit relationships between degradation characteristics and the component failure rate.

The research program goes beyond an analysis of times of degradation and failure. First, theoretical models that relate the degradation rate of the component to its failure rate are developed. With the relationships derived, information on component degradation can be used to predict the component failure rate and its significance. Specifically, this methodology can use aging trends in the component degradation rate to predict future aging trends in the component failure rate.

The capability of making such a prediction is important because information on component failure rates due to aging is required to quantify the effects of aging on core damage frequency and risk. This information is also needed to quantify the effectiveness of a given maintenance program in controlling the effects of aging on the core damage frequency and risk. However, failure data are often sparse. On the other hand, degradation data are more abundant because degradations occur at a higher rate than do failures. Thus the methodology developed in this report allows component failure rates due to aging to be estimated from component degradation rates. This has the potential of greatly increasing the quantity and accuracy of component failure rates due to aging for use in risk evaluations of aging effects.

It is important that, in addition to the identification of aging trends in degradation and failure data, the methodology allows maintenance indicators to be selected in such a way that component degradations are related to impacts on reliability and risk. When the degradation indicators show significant impacts of degradation on the component failure rate and the resulting risk, maintenance should be performed to correct the degradations. Thus the degradation indicators can provide a practical and effective means of monitoring component condition and signal for the correction of degradations before they have significant impacts on reliability and risk. In addition, the methodology was used to develop initial estimates of the effectiveness of maintenance in preventing degradations from becoming failures.

Specific applications of the theoretical approach resulted in quantitative models of component degradation rates and component failure rates derived from plant-specific data. As part of the data analysis, statistical techniques that identify aging trends in failure and degradation data were developed. The aging trends can be of any kind and can exist in any segment of the data. Specifically, an analysis of residual heat removal (RHR) system pump data shows a "bathtub" curve for the degradation rate where a distinct increasing trend is observed at the later ages. Interestingly, the pump failure rate does not shown any increasing trend for the same period, which demonstrates the need to identify aging trends through analyses of component degradations.

These results are important first steps in showing that degradations can be modeled to identify aging effects. The theoretical methdology that was developed represents an advancement demonstrating that degradation characteristics are explicitly related to failure rates and hence ultimately to risk. The next step would be to use the methodology and statistical techniques to develop and validate practical procedures for predicting failure rates due to aging from degradation data. This ability would provide powerful tools for analyzing aging effects in terms of degradation data and for predicting their implications for reliability and risk.

NUREG/CR-5619, S.P. Nowlen, "The Impact of Thermal Aging on the Flammability of Electric Cables," Sandia National Laboratories, SAND90-2121, March 1991.

This report describes tests on the fire vulnerability of aged electrical components performed for the NRC NPAR program. The objective was to identify and investigate issues of plant aging that might result in an increased fire risk at commercial nuclear power plants. The particular issue investigated in these tests is the impact of thermal aging on the flammability of electrical cables.

The cable insulation represents the dominant source of combustible materials in most nuclear power plant areas. Current USNRC standards require the use of low-flame-spread cables, as certified by the IEEE-383 qualification standard, in all new installations. However, should these cables lose their fire-retardant properties as a result of material aging, an increase in fire risk could result based on the role cable installations have played in past fire risk assessments. To assess this issue, four large-scale cable flammability tests were performed. Two commonly used types of nuclear grade electrical cables were tested in both the new (unaged) and a thermally aged (through accelerated aging) condition:

1. Rockbestos FIREWALL III, 3-conductor, 12 AWG, Neoprene jacketed, cross-linked polyethylene (XPE) insulated light power or control cable, and
2. Boston Insulated Wire (BIW) Bostrad 7E, 2-conductor with shield and drain, 16 AWG, Hypalon jacketed, ethylene-propylene rubber (EPR) insulated instrumentation cable.

Both of these cables are certified nuclear grade cables, including certification as low-flame-spread cables. They are among the most commonly used cable types in U.S. commercial reactors.

Since these cables were certified as low flame spread, they were exposed to a fire that was more severe than the standard exposure test. This exposure was based on work performed by Factory Mutual Research Corporation (FMRC). FMRC found that a different gas burner (fire source) configuration with two cable trays placed face to face with insulating backer boards (as compared to a single open ladder tray used in the standard test) would produce enhanced fire propagation. The tests described here used a similar configuration to induce flame spread in the sample cable because, if the cables did not burn during testing, little would be learned.

During each of the four fire tests, it was observed that essentially all of the available combustible materials (the cable insulation and jacket materials) were consumed by the fires. Flame propagation to the full height of the 16-foot vertical cable trays was observed in all cases. However, upon examination of the test data, it was found that, for both cable types, the aged cable samples displayed a reduced flammability as compared to the unaged cable samples. This was reflected in reductions in both the rate of rise and the peak value of the measured fire heat release rates for the aged cable samples as compared to those for the unaged cable samples.

These results indicate that, at least for the two cable types tested, thermal aging resulted in a decrease of material flammability. Hence, for these two cable types, the issue of material aging and cable flammability is not of concern. The use of material flammability parameters obtained from tests of unaged cable samples will therefore provide conservative assessments of material flammability in a thermally aged condition.

These results are consistent with results of previous cable aging studies. It has been observed that the process of thermal aging tends to drive off certain of the more volatile constituents of the cable insulation materials. This will leave less of these compounds available during a fire to support combustion; hence flammability is reduced somewhat. Although other cable types have not been tested, it is expected that similar results would be obtained. No further investigation of this issue is recommended.

NUREG/CR-5643, D. E. Blahnik, D. A. Casada, J. L. Edson, D. L. Fineman, W. E. Gunther, H. D. Haynes, K. R. Hoopingarner, M. J. Jacobus, D. B. Jarrell, R. C. Kryter, H. L. Magelby, G. A. Murphy, and M. Subudhi, "Insights Gained from Aging Research," Brookhaven National Laboratory, BNL-NUREG-52323, March 1992.

Aging, if it is not properly managed, affects the operational safety of all reactor structures, systems, and components, and it has the potential to increase risks to public health and safety. It is therefore essential to understand the aging processes that occur in a system or component so that they can be effectively managed. The NRC's Nuclear Plant Aging Research (NPAR) Program has identified those components and systems that have a propensity for age-related degradation and has evaluated methods for detecting and mitigating aging effects.

This report was developed to consolidate the research results from the assessments of component and system aging sponsored by the NRC for use by industry and by NRC in understanding and managing the aging of systems, structures, and components in nuclear power plants. The report discusses aging-related problems, operating experience, solutions to aging problems, and reference documents. The input for this report was provided by the NRC contractors who were responsible for the research, including:

1. Brookhaven National Laboratory

   Components: Battery chargers, inverters, motors, and motor control centers

   Systems: Component cooling water, control rod drive (Westinghouse), instrument air, and residual heat removal.

2. Idaho National Engineering Laboratory

   Component: Batteries

   Systems: 1E distribution, reactor protection, high-pressure coolant injection and core spray (BWR), and high-pressure safety injection (PWR).

3. Oak Ridge National Laboratory

   Components: Auxiliary feedwater pumps, check valves, motor-operated valves, power-operated relief valves and block valves, and solenoid-operated valves

   Systems: Auxiliary feedwater.

4. Pacific Northwest Laboratory

   Components: Emergency diesel generators and snubbers

   Systems: Service water.

5. Sandia National Laboratory

   Components: Cables.

The document includes a Summary of Research Results and an Aging Assessment Guide. The Aging Assessment Guide is more concise than the Summary of Research Results, and it focuses on the specific inspection activities to be considered when assessing the operational readiness of the component or system. The Guide contains visual inspection techniques for detecting aging degradation, including external and internal indicators and important operating parameters. In addition, the Guide lists those activities associated with maintenance, operations, design, and testing that research has shown to be beneficial for managing the aging of that component or system.

NUREG/CR-5646, R. Steele, Jr., and M. E. Nitzel, "Piping System Response During High Level Simulated Seismic Tests at the Heissdampfreaktor Facility (SHAM Test Series)," Idaho National Engineering Laboratory, EGG-2655, July 1992.

This report describes the analysis and results from the "Servohydraulische Anregung Mashinetechnik" (SHAM) seismic research program, in which the NRC and the Idaho National Engineering Laboratory (INEL) participated. The program was conducted by Kernforschungszentrum Karlsruhe (KfK, the Nuclear Research Center) at the decommissioned Heissdampfreaktor (HDR, the superheat reactor) located near Frankfurt, Germany. The SHAM experiments consisted of the direct excitation of a piping system called the Versuchskreislauf (VKL) that was modified to include a naturally aged U.S.-made 8-in. motor-operated gate valve. The piping system was excited at seismic levels of 100% of the safe shutdown earthquake (100% SSE), increasing up to 800% of the SSE using two large 40-ton servohydraulic shakers mounted to the HDR containment building and attached to the piping system. Experiments were conducted with the piping supported by six different piping support systems. These included support configurations typical of those commonly used in European power plants, a typical stiff U.S. system, and a very flexible system. This report specifically addresses the tests performed with the U.S. stiff support configuration. The objectives of the INEL portion of the research included determining the safety margins and failure modes of nuclear grade snubbers and other support components, determining the effects of support failures on piping response, and determining the effects of the dynamic loads on gate valve operability.

Results from tests at input levels of 200% SSE, 600% SSE, and 800% SSE were examined. The 100% tests were not compared to design predictions because of support failures at this level of input. The 200% SSE tests, with all dynamic supports operable, showed that the design analysis predicted maximum stresses at the same locations where the maximum strains were recorded during the tests; also, the acceleration histories showed that the piping responses were generally in the same frequency bands as the predicted natural frequencies.

The 800% SSE loadings caused overload failures of several snubbers with measured strains greater than yield. The timing of the failures of three of the snubbers and the force and displacement data indicated that a zipper-effect failure phenomenon occurred. However, even with the large displacements and strains, no physical failure of the piping occurred.

Except for two cases, all snubber failures occurred at loads well above their design loading. In one case, a load of 8.67 times the design rating was sustained prior to failure. One snubber and its replacement from the same manufacturer failed well below their design loads. Both snubbers were returned to the manufacturer for inspection and analysis. No further information had been received on this at the time this report was prepared.

The U.S.-made 8-in. motor-operated gate valve operated smoothly during all tests in the SHAM series. Some limit switch chatter was observed; however, the limit switch contacts did not stay open long enough to cause the motor controller circuit to interrupt the current flow to the motor. The data showed that even under the most severe structural loading experienced during the tests, the valve operated smoothly.

The test results indicate that sufficient safety margins exist when commonly accepted design methods are applied and that piping systems will likely maintain their pressure boundary in the presence of severe loading and with the loss of multiple supports.

NUREG/CR-5655, M.J. Jacobus and G.F. Fuehrer, "Submergence and High Temperature Steam Testing of Class IE Electrical Cables," Sandia National Laboratories, SAND90-2629, May 1991.

Many types of cable are used throughout nuclear power plants in a wide variety of applications. Cable qualification typically includes thermal and radiation aging intended to bring the cable to a defined "end-of-life" condition before exposure to a simulated design-basis accident. In some instances, cables must be qualified for submergence conditions. High-temperature steam testing of cables (beyond the design basis) is not currently required for qualification.

This report describes the results of high-temperature steam testing and submergence testing of 12 different cable products. The cable products tested are typical of cables used inside containments of U.S. light water reactors and include primary insulations of cross-linked polyolefin (XLPO), ethylene propylene rubber (EPR), silicone rubber (SR), polyimide, and chlorosulfonated polyethylene (CSPE).

These cables were part of a larger test program in which four sets of cables were subjected to simultaneous thermal and radiation aging for 0 (unaged), 3, 6, and 9 months. Following the aging, each set of cables was exposed to a simulated loss-of-coolant accident (LOCA).

The submergence test was performed on the cables that had been aged for 6 months and then exposed to the simulated LOCA, and the high-temperature steam test was performed on the cables that had been aged for 3 months and also exposed to the LOCA. Both of these tests were added to the scope of the test program because the aged cables had completed all planned testing and many of the cables had not yet failed. The unaged cables and the cables aged for 9 months were not involved in either the submergence testing or the high-temperature steam testing and are therefore not discussed in this report.

The submergence test used a solution close to that specified by IEEE 383-1974 for chemical spray during LOCA simulations. The solution was maintained at about 95°C during the exposure, which lasted a total of 1000 hours. The high-temperature steam test involved exposure to steam at temperatures as high as 400°C (750°F). Cable insulation resistances were monitored throughout the high-temperature steam test and at discrete times during the submergence test. Dielectric withstand testing was performed before the submergence and high temperature steam tests and at the end of the submergence test. The cables that passed the post-submergence dielectric test were subsequently wrapped around a mandrel with a diameter 40 times that of the cable and exposed to a final dielectric withstand test.

The conclusions from this study are:

1. The results of the high-temperature steam test indicate the approximate thermal failure thresholds for each cable type. EPR cables generally survived slightly higher temperatures (370-400°C) than XLPO cables (299-388°C) during the high-temperature steam exposure. The XLPO-insulated conductors had no insulation left at the end of the high-temperature steam test. Silicone rubber failed in the range of 396° to 400°C, Kerite FR at 372° to 382°C, and polyimide at 399°C.
2. The results of the submergence test indicate that a number of cable types can withstand submergence at elevated temperature, even after exposure to a loss-of-coolant accident simulation. XLPO cables generally performed better than EPR cables in the submergence test and in the post-submergence dielectric testing. By the end of the final dielectric test (after the mandrel bend), only 1 of 11 (9.1%) XLPO-insulated conductors, 17 of 20 (85%) EPR-insulated conductors, and 6 of 8 other cables (silicone, Kerite FR, and Rockbestos coaxial) had failed.
3. A number of cables that performed well during the submergence test failed post-submergence dielectric withstand testing (either before or after the mandrel bend). This indicates that the IEEE 383 dielectric withstand tests and mandrel bends can induce failure of otherwise functional cables. Note that this conclusion does not imply a criticism of the IEEE 383 requirements, which are intended to provide a level of conservatism in the testing.
4. The IEEE 383 dielectric withstand tests are very severe even if a mandrel bend test is not

performed. This is evidenced by the failure of nine conductors and the near failure of three more in the post-submergence dielectric withstand test, only two of which were showing a strong indication of degradation during the submergence test.

NUREG/CR-5693, R. Lofaro, W. Gunther, M. Subudhi, and B. Lee, "Aging Assessment of Component Cooling Water Systems in Pressurized Water Reactors—Phase II," Brookhaven National Laboratory, BNL-NUREG-52283, June 1992.

A two-phase aging analysis of component cooling water (CCW) systems in pressurized water reactors (PWRs) was performed. In Phase I (NUREG/CR-5052, July 1988), the effects of aging were characterized, and the predominant failure modes, aging mechanisms, and components susceptible to aging degradation were identified. Failure rate trends were examined, and their effect on time-dependent system unavailability was investigated.

In this Phase II study, the methods used to manage aging degradation in the CCW system were studied. Information was collected and analyzed on inspection, surveillance, monitoring, and maintenance (ISM&M) techniques. Also investigated were advanced techniques for detecting and mitigating aging degradation of construction materials and their relationship to aging mechanisms, as well as codes and regulatory requirements.

Results of this aging study show that there are specific basic ISM&M practices that all plants perform. These practices are typically required by Code or plant technical specifications, and they are capable of detecting degradation arising from many aging mechanisms. However, they are not comprehensive enough to completely control all aging degradation. To more effectively control aging, ISM&M programs should include a combination of basic and supplemental practices that are selected on the basis of plant-specific conditions. The report presents listings of supplemental practices, correlated with the respective aging mechanisms each of the practices helps to detect or mitigate.

NUREG/CR-5699, R.H. Greene, "Aging and Service Wear of Control Rod Drive Mechanisms for BWR Nuclear Plants," Volume 1, Oak Ridge National Laboratory, ORNL-6666/V1, November 1992.

This Phase I Nuclear Plant Aging Research (NPAR) study examines the aging phenomena associated with BWR control rod drive mechanisms (CRDMs) and assesses the merits of various methods of "managing" this aging. Information for this study was acquired from (1) the results of a special CRDM aging questionnaire distributed to each U.S. BWR utility, (2) a first-of-its-kind workshop held to discuss CRDM aging and maintenance concerns, (3) an analysis of the Nuclear Plant Reliability Data System (NPRDS) cases of failures attributed to the control rod drive system, and (4) information exchange with nuclear industry CRDM maintenance experts.

Nearly 23% of the NPRDS CRD system component failure reports were attributed to the CRDM. The CRDM components most often requiring replacement because of normal wear and aging are the Graphitar seals. The predominant causes of aging for these seals are mechanical wear and thermally induced embrittlement. More than 59% of the NPRDS CRD system failure reports were attributed to components that make up the hydraulic control unit. The predominant hydraulic control unit components experiencing the effects of service wear and aging are valve seals, discs, seats, stems, packing, and diaphragms.

Since CRDM changeout and rebuilding is one of the highest dose, most physically challenging, and most complicated maintenance activities routinely accomplished by BWR utilities, this report also highlights recent innovations in CRDM-handling equipment and rebuilding tools that have resulted in significant dose reductions to the maintenance crews using them.

NUREG/CR-5700, A. C. Gehl and E. W. Hagen, "Aging Assessment of Reactor Instrumentation and Protection Systems Components," Oak Ridge National Laboratory, ORNL/TM-11806, July 1992.

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 NRC's NPAR Program. These six categories were selected because of their importance in all the operations of safety-related instrumentation and control (I&C) systems and because they have not been previously reviewed within the NPAR program.

The issue of aging of safety-related control systems in a world of increased performance demands is a relatively new one. If left unchecked, components of that system can lead to an impairment of continued safe operation of a nuclear power plant.

In this report, the effects of aging from operational and environmental stressors were characterized by the results depicted in Licensee Event Reports (LERs). The data are graphically displayed as frequency of events per plant year (on the vertical axis) for various operating plant ages (on the horizontal axis, ranging from 1 to 28 years). Such graphs help determine aging-related failure trends and patterns of events.

Three main conclusions were drawn from this study.

1. I&C modules make a modest contribution to safety-significant events.
   - 17% of LERs 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 of 25–50%).

2. Of the six modules studied, indicators, sensors, and controllers account for the bulk (83%) of aging-related failures.
3. "Infant mortality" (during the early life of instruments) 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).

NUREG/CR-5706, D.A. Casada, "NRC Bulletin 88-04: Potential Safety-Related Pump Loss—An Assessment of Industry Data," Oak Ridge National Laboratory, ORNL-6671, June 1991.

Nuclear utility plants are required to periodically test safety-related pumps to demonstrate proper functioning of the pump. Historically, a substantial number of these pumps have been routinely tested at the flow rate available through the pump's minimum flow recirculation flow path, which in many cases was sized to avoid overheating only. It has become more widely recognized that operation of a pump under low-flow conditions can result in hydraulically unstable conditions that can damage the pump, even though the rate of flow is adequate for heat removal.

Nuclear Regulatory Commission (NRC) Bulletin 88-04 required utilities to examine (1) the potential for dead-heading of pumps due to parallel pump competition and (2) the adequacy of the minimum flow rate provided for each safety-related pump. Utilities have reviewed the currently recommended minimum flow rates with pump vendors and have examined existing system design provisions, operating controls, and historical maintenance experience.

Under the auspices of the NRC's Nuclear Plant Aging Research Program, Oak Ridge National Laboratory has reviewed utility responses to Bulletin 88-04. An assessment of the industry response and resultant conclusions and recommendations are presented.

NUREG/CR-5720, R. Steele, Jr., J. C. Watkins, K. G. DeWall, and M. J. Russell, "Motor-Operated Valve Research Update, Idaho National Engineering Laboratory, EGG-2643, June 1992.

The U.S. Nuclear Regulatory Commission (NRC) is supporting motor-operated valve (MOV) research at the Idaho National Engineering Laboratory (INEL). The MOV tests provide the basis for assessing the effects of various factors on the valves and for evaluating the current industry standards. This report addresses several research items, including (1) the use of in situ test results to estimate the response of a valve at design basis conditions, (2) the methods used by industry to predict required valve stem forces and torques, (3) guidelines for satisfactory performance of MOV diagnostics systems, and (4) the development of performance standards or guidance documents for acceptable design basis tests.

In order to perform an in-depth study, the authors have reviewed all of INEL past, current, and ongoing valve research, including test data. This review revealed that the use of in situ test results to estimate the response of gate and butterfly valves at design basis conditions is possible, but some caveats are necessary. It was found that the methods used by industry to predict the required stem force for a gate valve and required stem torque for a butterfly valve are flawed. Also, it was observed that satisfactory performance of MOV diagnostic systems is possible, but very few of the currently available systems measure enough parameters to be useful.

Additionally, this report discusses INEL participation in the writing of performance standards and guidance documents for acceptable design basis tests. Such participation includes an extensive information exchange with the American Society of Mechanical Engineers standards writing committees and attendance at working group meetings.

NUREG/CR-5754, K.H. Luk, "Boiling-Water Reactor Internals Aging Degradation Study, Phase 1," Oak Ridge National Laboratory, ORNL/TM-11876, September 1993.

This report documents the results of a study on the effects of aging degradation on 25 selected boiling-water reactor (BWR) internal components. The operating environment inside a BWR pressure vessel produces stressors that could lead to the development of aging-related degradation mechanisms. A data base containing aging-related failure information for the selected internal components is established using data from Licensee Event Reports. Results of the failure information survey identified two major aging-related degradation mechanisms for reactor internals: stress corrosion cracking (SCC) and fatigue. SCC includes intergranular SCC and irradiation-assisted SCC (IASCC).

Strategies for controlling and managing aging degradations are based on understanding the relationship between stressors and the associated aging-related degradation mechanisms. The implementation of a plant hydrogen water chemistry (HWC) program is considered to be a promising method for controlling SCC, which is the more prevalent problem for BWRs. Flow-induced vibration (FIV) is the major cause of fatigue problems in BWR internals. FIV problems are resolved either by eliminating the excitation sources or by detuning the structure from input excitations. Questions remain concerning the effectiveness of HWC in mitigating SCC in internals and in the assessment of high-cycle fatigue in a corrosive environment.

Vibration monitoring, based on neutron noise measurements and trending studies, is an inspection method that can provide early failure detection capability and can improve the effectiveness of current plant inservice inspection programs. However, the large water gap and the lack of existing ex-core neutron flux monitors may hinder the use of neutron noise vibration measurements in BWRs.

NUREG/CR-5762, J. F. Gleason, "Comprehensive Aging Assessment of Circuit Breakers and Relays," Wyle Laboratories, WYLE 60101, March 1992.

This report describes the results of a comprehensive aging assessment of relays and circuit breakers that was completed as part of the NRC Nuclear Plant Aging Research (NPAR) Program. This is a Phase II report and the research has followed the established NPAR strategy described in Revision 2 of NUREG-1144, "Nuclear Plant Aging Research (NPAR) Program Plan" (June 1991). Relays and circuit breakers are important safety-related equipment that perform critical functions in the operation and control of nuclear power plants.

It is a challenge to a good preventive maintenance program to be sensitive to the effects of aging. Early identification of age-related degradation increases the probability that the safety significance of the identified aging process can be minimized.

A comprehensive effort was undertaken to verify improved inspection, surveillance, and monitoring (ISM) methods. The Phase II effort was accomplished in four major elements: an investigation into current and advanced ISM methods, tests of aged relays and circuit breakers, tests of degraded relays and circuit breakers, and in situ tests.

Current and advanced ISM methods were ascertained by soliciting information from nuclear and nonnuclear utilities, relay and circuit breaker manufacturers, and maintenance facilities. Testing of naturally aged devices was performed. Test specimens for each of the five relay types (auxiliary, control, electronic, protective, and timing) and two circuit breaker types (molded case and metal clad) were obtained from nuclear and nonnuclear utilities and manufacturers. A total of 39 specimens were tested, utilizing current and improved ISM methods.

Eleven specimens of relays and circuit breakers were purposely degraded and the ISM methods implemented after each degraded condition. The purpose of these degradation tests was to evaluate the effectiveness of the method to detect or predict the level of degradation. This also provided some quantifiable parameters of the extent of degradation. The degradations chosen for each relay and circuit breaker type were purposely severe, but for the most part, did not cause total loss of operability of the device. Thus, an attempt was made to simulate the worst state of deterioration or degradation prior to failure to operate. The degradations were chosen based on a review and evaluation of the failure modes and mechanisms reported in Phase I, experiences of the research team, and requests from the utilities and manufacturers.

These evaluations of degradation conditions showed that generally accepted current nuclear plant maintenance practices do not always detect the effects of significant aging mechanisms. This provides insight into the reason failures of safety-related relays have occurred in service in spite of a comprehensive maintenance program.

At specific plants (Catawba Nuclear Station and Nine Mile Point Unit 1), the research team witnessed plant maintenance personnel performing routine maintenance on relays and circuit breakers. Copies of procedures were obtained, results of plant maintenance tests reviewed, and engineerinig and maintenance personnel interviewed. Additionally, nonintrusive ISM procedures of infrared pyrometry, infrared scanning, and vibration testing were demonstrated.

The research of this report has particular significance with respect to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," Information Notice 84-20, "Service Life of Relays in Safety-Related Systems," and IE Bulletin 84-02, "Failures of General Electric Type HFA Relays in Use in Class IE Safety Systems." These documents require licensees to have preventive maintenance and surveillance programs for circuit breakers and relays. The research provides information on the effectiveness of the required preventive maintenance methods and demonstrates that improved ISM procedures were more effective than current industry practices for detecting aging and mitigating its effects on specific devices and components.

Finally, specific recommendations were made for the introduction of changes in current nuclear industry practices of inspection, surveillance, and maintenance on relays and circuit breakers. These recommendations were based on significant research results (described in this report), that identified improved inspection, surveillance, and monitoring methods. Implementation of these methods could minimize the impact of aging and result in more cost-effective maintenance of relays and circuit breakers.

NUREG/CR-5772, M.J. Jacobus, "Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables," Volume 1, Sandia National Laboratories, SAND91-1766/1, August 1992.

This report describes the results of aging, condition monitoring, and accident testing of crosslinked polyolefin (XLPO) cables. Three sets of cables were aged for up to 9 months under simultaneous thermal ($\simeq 100°C$) and radiation ($\simeq 0.10$ kGy/hr) conditions. A sequential accident consisting of high dose rate irradiation ($\simeq 6$ kGy/hr) and high-temperature steam followed the aging. The test results indicate that most properly installed XLPO cables should be able to survive an accident after 60 years for total aging doses up to 400 kGy and for moderate ambient temperatures on the order of 50-55°C (potentially higher or lower, depending on material-specific activation energies). Mechanical measurements (primarily elongation, modulus, and density) were more effective than electrical measurements for monitoring age-related degradation.

NUREG/CR-5772, M.J. Jacobus, "Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables," Volume 2, Sandia National Laboratories, SAND91-1766/2, November 1992.

This report describes the results of aging, condition monitoring, and accident testing of ethylene propylene rubber (EPR) cables. Three sets of cables were aged for up to 9 months under simultaneous thermal (≃ 100 °C) and radiation (≃ 0.10 kGy/hr) conditions. A sequential accident consisting of high dose rate irradiation (≃ 6 kGy/hr) and high-temperature steam followed the aging. Also exposed to the accident conditions was a fourth set of cables, which were unaged. The test results indicate that most properly installed EPR cables should be able to survive an accident after 60 years for total aging doses on the order of 150-200 kGy and for moderate ambient temperatures on the order of 45-55°C (potentially higher or lower, depending on material-specific activation energies and total radiation doses). Mechanical measurements (primarily elongation, modulus, and density) were more effective than electrical measurements for monitoring age-related degradation.

NUREG/CR-5772, M.J. Jacobus, "Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables," Volume 3, Sandia National Laboratories, SAND91-1766/3, November 1992.

This report describes the results of aging, condition monitoring, and accident testing of miscellaneous cable types. Three sets of cables were aged for up to 9 months under simultaneous thermal (≃ 100 °C) and radiation (≃ 0.10 kGy/hr) conditions. A sequential accident consisting of high dose rate irradiation (≃ 6 kGy/hr) and high-temperature steam followed the aging. Also exposed to the accident conditions was a fourth set of cables, which were unaged. The test results indicate that, properly installed, most of the various miscellaneous cable products tested should be able to survive an accident after 60 years for total aging doses of at least 150 kGy or higher (depending on the material) and for moderate ambient temperatures on the order of 45-55 °C (potentially higher or lower, depending on material-specific activation energies and total radiation doses). Mechanical measurements (primarily elongation, modulus, and density) were more effective than electrical measurements for monitoring age-related degradation.

NUREG/CR-5779, J. C. Moyers, "Aging of Non-Power-Cycle Heat Exchangers Used in Nuclear Power Plants," Vol. 1, Oak Ridge National Laboratory, ORNL-6687/V1, July 1992.

This report presents the results of the Phase I assessment of the time-related (aging) degradation of non-power-cycle heat exchangers used in safety-related systems or that provide normal operating capability in nuclear power plants. This assessment was sponsored by the Office of Nuclear Regulatory Research of the USNRC as an element of the ongoing Nuclear Plant Aging Research Program. The objectives of this Phase I research effort were to review operating experience and other information, to identify failure modes and causes resulting from aging, and to identify measurable parameters that might provide a better indication of equipment condition.

The report briefly reviews the design and application of the heat exchangers in both PWR and BWR plants. Typical design characteristics and materials of construction are given for the various applications. Operational stressors are categorized and discussed.

Operating events described in data bases for nuclear power plants and in nuclear industry reports were examined. These data bases included (a) the licensee event report file as cataloged in the Sequence Coding and Search System maintained by ORNL's Nuclear Operations Analysis Center, (b) the Nuclear Plant Reliability Data System compiled by the Institute for Nuclear Power Operations, (c) Nuclear Power Experience published by Stoller Power, Inc., and (d) maintenance records for a two-unit PWR plant as furnished by a cooperating utility. A total of 710 reported events were examined. Of these, 279 events involved interfluid leakage, 217 involved external leakage, 156 involved tube-side flow blockage, and 25 involved impaired heat transfer; the remaining 33 reportings related to miscellaneous events.

There are only minimal regulatory or Technical Specification requirements for inservice inspection and testing, and they are limited primarily to those inspections and tests required to maintain the integrity of the pressure-containing boundary. The general philosophy of plant operators regarding flow blockage and leakage is that repairs and maintenance will be done as required. Improvements to this philosophy that could lead to enhanced reliability are not apparent, and no such improvements are suggested in the report.

Inservice testing to determine the heat transfer capability of the heat exchangers has normally been done only when possible degradation was indicated from observation of process parameters; scheduled performance testing was not normally done. However, largely as a result of NRC concerns for the capability of safety-related service-water-cooled heat exchangers to perform as required under accident conditions and the resulting issuance of Generic Letter 89-13 ("Service

Water System Problems Affecting Safety-Related Equipment," July 18, 1989), plant-specific inservice performance testing programs are being developed by plant owners. In addition, the Operation and Maintenance Committee of the American Society of Mechanical Engineers is developing standards to address both vibration monitoring and inservice performance testing of heat exchangers.

NUREG/CR-5783, E. Grove and W. Gunther, "Aging Assessment of the Combustion Engineering and Babcock & Wilcox Control Rod Drives," Brookhaven National Laboratory, BNL-NUREG-52299, January 1993.

The effects of aging upon the Babcock & Wilcox (B&W) and Combustion Engineering (CE) control rod drive (CRD) systems have been evaluated. For this study, the CRD system boundary included the control rod assemblies, guide tubes, control rod drive mechanism, control system components, rod position indication components, and cooling system. Detailed operation experience data for 1980 to 1990 was evaluated to identify the predominant failure modes, causes, and effects. The results of this evaluation, along with an assessment of component material and operating environment, lead to the conclusion that both the B&W and CE CRD systems are susceptible to age degradation. Failures of the CRD system have resulted in significant plant effects, including power reductions, plant shutdowns, scrams, and ESF actuations.

Information on current plant system inspection and maintenance practices were obtained from two B&W plants and four CE plants through an industry survey. The results of this survey indicate that some plants have modified the system, replaced components, and established preventive maintenance programs, some of which effectively address the aging issue while others do not. The potential application of some advanced monitoring inspection techniques are discussed.

NUREG/CR-5807, J. K. Wang and M. S. Kalsi, "Improvements in Motor Operated Gate Valve Design and Prediction Models for Nuclear Power Plant Systems," Kalsi Engineering, Inc., KEI No. 1721, May 1992.

This report documents the results of Phase I research proposed and conducted by Kalsi Engineering, Inc., to improve the operability of motor operated gate valves in nuclear power plants. Phase I research, funded by the Small Business Innovation Research (SBIR) program, resulted in the following major accomplishments:

- Opening and closing thrust equations for the common types of gate valves used in U.S. nuclear power plants have been developed and documented.
- An analytical methodology to predict inertial thrust overshoot in a motor operated gate valve has been developed from the basic principles (balance of forces equation). Comparisons against data supplied by Duke Power Company have confirmed that the methodology is sound, and there is good quantitative agreement between analytical predictions and actual test results.
- The results of a comprehensive review of friction and galling data are documented in this report to provide a rational basis for selecting an appropriate coefficient of friction for a given application.
- The concept of index of contact stress severity has been introduced to determine whether or not a gate valve will behave predictably under fluid-flow forces. Preliminary analyses to calculate localized contact stresses at the disc-to-guide contact and at the disc-to-downstream seat contact under disc-tilting conditions have been developed.
- Significant factors that affect the opening thrust requirements of a gate valve have been identified, and quantitative methods that can be used to diagnose valve opening problems have been documented.
- Improvements in gate valve designs to make them less sensitive to pressure/thermal transients and external pipe loads have been identified, and some quantitative examples are included in the report to show the achievable degree of improvement.

In summary, the Phase I research has been successful in completing the preliminary development of improved gate valve operability models. This can serve as an excellent foundation to continue further analytical and experimental development that is necessary to provide reliable and proven gate valve operability models to the nuclear power industry.

NUREG/CR-5848, J.S. Dukelow, "Recordkeeping Needs To Mitigate the Impact of Aging Degradation," Pacific Northwest Laboratory, PNL-7987, October 1992.

This report discusses technical issues associated with the role of nuclear plant records systems in understanding and managing the aging of nuclear plant components, systems, and structures. It considers both the types of technical data useful for verifying continued safe operation and the use of new technology for upgrading records systems. Specific topics reviewed include the need for maintenance and reliability data, operational history data to support the assessment of remaining fatigue life, comprehensiveness and usability of the engineering design basis, improvement of the data input process, and conversion of existing records into machine-readable forms.

The report concludes that successful management of nuclear plant aging will require improvement of existing plant records systems; several generic and specific recommendations are provided. The computer-based technology for meeting this need and implementing these recommendations already exists and can be implemented at a reasonable cost.

NUREG/CR-5870, D. P. Brown, E. V. Werry, and D. E. Blahnik, "Results of LWR Snubber Aging Research," Pacific Northwest Laboratory, May 1992.

Snubbers are safety-related devices used to restrain undesirable dynamic loads at various piping and equipment locations in nuclear power plants. Each snubber must accommodate a plant's normal thermal movements and be capable of restraining the maximum off-normal dynamic loads postulated for its specific location. Snubbers are subject to the effects of aging; the factors that contribute to the degradation of their safety performance need to be better understood.

Snubber operability is mandated by the regulations, which stipulate that systems, structures, and components, including snubbers, be designed to withstand the effects of normal and off-normal dynamic phenomena. In the mid 1980s, the NRC recognized the need to enhance snubber performance through aging studies and improved service-life monitoring techniques. The NRC's Nuclear Plant Aging Research (NPAR) Program Plan of 1987 provided the vehicle and the sponsorship to undertake preliminary investigations into snubber performance and aging. Pacific Northwest Laboratory (PNL) staff and its subcontractors, Lake Engineering and Wyle Laboratories, visited 13 plants, conducted interviews, collected relevant data, and performed the snubber research.

This report describes the Phase II NPAR in-plant aging research conducted to enhance the understanding of snubber aging and its consequences. The in-plant aging research was based on a research plan by Brown et al., which clarified the relationship of snubber aging to snubber degradation and identified additional information on aging that requires further investigation and analysis for both hydraulic and mechanical snubbers.

This report presents the results of snubber aging research, including methodology, evaluation, testing and failure data, as well as service-life monitoring recommendations that emphasize distinguishing between aging- and nonaging-related snubber failures. The graphics, tables, and supporting text clearly illustrate this distinction. The results of reported work support the perspective that snubber failures are closely related to age-related degradation caused by inservice operational environmental influences, e.g., vibration and elevated temperature. Because there is a lack of information on service to mechanical snubbers, special emphasis was placed on gathering such information for these devices.

NUREG/CR-5944, D.A. Casada and M.D. Todd, "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry," Oak Ridge National Laboratory, ORNL-6734, September 1993.

Check valve operating problems in recent years have resulted in significant operating transients, increased cost, and decreased system availability. As a result, additional attention has been given to check valves by utilities (resulting in the formation of the Nuclear Industry Check Valve Group), as well as the NRC and the American Society of Mechanical Engineers Operation and Maintenance Committee. These organizations have the fundamental goal of ensuring reliable operation of check valves.

A key ingredient to an engineering-oriented reliability improvement effort is a thorough understanding of relevant historical experience. A detailed review of historical failure data, available through the Institute of Nuclear Power Operation's Nuclear Plant Reliability Data System, has been conducted. The focus of the review is on check valve failures that have involved significant degradation of the valve internal parts. A variety of parameters are condidered, including size, age, system of service, method of failure discovery, the affected valve parts, attributed causes, and corrective actions.

NUREG/CR-6001, G.D. Buckley, R.D. Orton, A.B. Johnson, Jr., L.L. Larson, "Aging Assessment of BWR Standby Liquid Control Systems," Pacific Northwest Laboratory, PNL-8020, August 1992.

Pacific Northwest Laboratory conducted a Phase I aging assessment of the standby liquid control (SLC) system used in boiling-water reactors. The study was based on detailed reviews of SLC system component and operating experience information obtained from the Nuclear Plant Reliability Database System, the Nuclear Document System, Licensee Event Reports, and other databases. Sources on sodium pentaborate, borates, and boric acid, as well as the effects of environment and corrosion in the SLC system, were also reviewed to characterize chemical properties and corrosion characteristics of borated solutions.

Relatively few SLC component failures were attributed to sodium pentaborate buildup or corrosion. The leading aging degradation concern to date appears to be setpoint drift in relief valves, which has been discovered during routine surveillance and is thought to be caused by mechanical wear. A higher setpoint results in loss of system overpressure protection, and a decrease in setpoint results in a reduction of boron injection rate. Degradation was also observed in pump seals and internal valves, which could prevent the pumps from operating as required by the technical specifications. In general, however, the results of the Phase I study indicate that age-related degradation of SLC systems has not been serious.

NUREG/CR-6029, W.K. Winegardner, "Phase I Aging Assessment of Nuclear Air-Treatment System HEPA Filters and Adsorbers," Volume 1, Pacific Northwest Laboratory, PNL-8594, August 1993.

A Phase I aging assessment of high-efficiency particulate air (HEPA) filters and activated carbon gas adsorption units (adsorbers) was performed by the Pacific Northwest Laboratory as part of the NRC's Nuclear Plant Aging Research Program. Information concerning design features; failure experience; aging mechanisms, effects, and stressors; and surveillance and monitoring methods for these key air-treatment system components was compiled. Over 1100 failures, or 12% of the filter installations, were reported as part of a DOE survey. Investigators from other national laboratories have suggested that aging effects could have contributed to over 80% of these failures. Tensile strength tests on aged filter media specimens indicated a decrease in strength. Filter aging mechanisms range from those associated with particle loading to reactions that alter properties of sealants and gaskets. Low radioiodine decontamination factors associated with the Three Mile Island accident were attributed to the premature aging of the carbon in the adsorbers. Mechanisms that can lead to impaired adsorber performance include oxidation as well as the loss of potentially available active sites as a result of the adsorption of pollutants. Stressors include heat, moisture, radiation, and airborne particles and contaminants.

NUREG/CR-6043, D.E. Blahnik and R.F. Klein, "Phase I Aging Assessment of Essential HVAC Chillers Used in Nuclear Power Plants," Pacific Northwest Laboratory, PNL-8614, September 1993.

The Pacific Northwest Laboratory conducted a Phase I aging assessment of chillers used in the essential safety air-conditioning systems of nuclear power plants. Centrifugal chillers in the 75- to 750-ton refrigeration capacity range are the predominant type used. The chillers used, and air conditioning systems served, vary in design from plant to plant. It is crucial to keep chiller internals very clean and to prevent the leakage of water, air, and other contaminants into the refrigerant containment system. Periodic operation on a weekly or monthly basis is necessary to remove moisture and noncondensable gases that gradually build up inside the chiller. This is especially desirable if a chiller is required to operate only as an emergency standby unit.

The primary stressors and aging mechanisms that affect chillers include vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Aging is accelerated by moisture, noncondensable gases (e.g., air), dirt, and other contamination within the refrigerant containment system; excessive start/stop cycling; and operation below the rated capacity. Aging is also accelerated by corrosion and fouling of the condenser and evaporator tubes. The principal cause of chiller failures is lack of adequate monitoring. Omission of scheduled maintenance and human errors also contribute to failures.

NUREG/CR-6048, K.H. Luk, "Pressurized-Water Reactor Internals Aging Degradation Study, Phase I," Oak Ridge National Laboratory, ORNL/TM-12371, September 1993.

This report is a summary of the results of a Phase I study on the effects of aging degradations in pressurized-water reactor (PWR) internal components. Westinghouse, Combustion Engineering, and Babcock & Wilcox reactors are included in the study.

Stressors associated with the operating environment inside the reactor pressure vessel provide conditions that are favorable to the development of aging-related degradation mechanisms. The dominant stressors are flow-induced oscillatory hydrodynamic forces generated by the reactor primary coolant flow. Results of a survey of the component failure information identified three major aging-related degradation mechanisms: fatigue, stress corrosion cracking, and mechanical wear.

Strategies for controlling and managing aging degradations are formulated based on the understanding of the linkage between stressors and aging degradation mechanisms. Flow-induced vibration problems are resolved by conventional engineering practices: by eliminating excitation sources or by de-tuning the structure from input excitations. Uncertainties remaining in the assessment of aging effects on PWR internals include long-term neutron irradiation effects and the influence of environmental factors on high-cycle fatigue failures.

An effective plant inservice inspection program will ensure the structural integrity of reactor internals. Reactor internals can be replaced if it is deemed necessary. Therefore, an inspection method with early failure detection capability will further enhance the safety as well as the efficiency of plant operations.

ORNL/NRC/LTR-91/25, D.A. Casada, "Throttled Valve Cavitation and Erosion," Oak Ridge National Laboratory, December 1991. Available from the NRC Public Document Room.

In November of 1988, Brunswick plant maintenance personnel discovered significant localized erosion of the valve body of a Unit 1 residual heat removal (RHR) valve, 1-E11-F017B (the maintenance was being performed to repair the valve stem and back seat). The F017B valve is a 20-inch Rockwell angle globe valve that has historically been used to throttle RHR flow. Excessive throttling of the valve had resulted in cavitation-induced erosion damage to areas immediately downstream of the seat. Subsequent investigation indicated that erosion of valve bodies was a generic concern for the other RHR valves used in the same service (F017A on Unit 1 and F017A and F017B on Unit 2). The November 1988 event led to the issuance of NRC Information Letter 89-01. The RHR valves used for suppression pool cooling (16-inch Anchor Darling globe type) were also found to have been damaged by cavitation erosion.

This study was conducted to identify the causes of erosion, valves most susceptible to erosive effects, historical erosion-related experience of valves and adjacent piping, and potential means of correcting the problems.

The principal source of erosion damage in nuclear plant valves and piping is cavitation. Other contributors to erosion damage are corrosion and impingement by high-velocity liquid and by abrasive particles. As fluid passes through the minimal flow area within a valve body, the pressure drops. Depending upon the system conditions and valve design, the pressure may drop below the fluid-vapor pressure. If this occurs, the subsequent pressure recovery downstream of the valve causes the vapor pockets to collapse. The process of the vapor pocket collapse (cavitation) results in substantial energy dissipation. The energy dissipation may damage material in the vicinity of the cavitation site and result in substantial component and system vibration.

Valves used in heavily throttled conditions are frequently the most susceptible to damage, mainly because of the fact that the system conditions that dictate the throttling are more likely to result in cavitation than are those for which throttling is not required or is minimal. Valves used in heat exchanger outlet or bypass service or in other control functions are most likely to experience cavitation.

Test indicated that cavitation existed throughout the range of 4,000 to 16,000 gpm with F017 valves, while F024 valves started to indicate cavitation to 4,500 gpm. Cavitation was most prevalent at higher flow rates, and it was also noted that the location of the cavitation moved throughout the valve's body as flow changed.

Subsequent investigation of seven other valves used in safety-related throttling service, including core spray (CS), high-pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) systems revealed that one other valve, the HPCI system full-flow test isolation valve F008, had experienced notable erosion.

Butterfly valves are particularly susceptible to cavitation because of the fact that the minimum pressure in them is substantially less than that for a globe valve under the same overall pressure drop.

The service water system was found to have experienced the most erosion-related damage to valves and piping. The condensate and feedwater system has also experienced a significant number of erosion-related failures.

The Oak Ridge National Laboratory performed an assessment of the significance of valve body erosion, focusing on the identification of valve types and applications susceptible to erosion.

The material used in valves and piping is significant in determining the rate of erosion. In general, hardfacing materials such as stellite have been used to minimize erosive effects. Non-cobalt-bearing materials such as nickel aluminides have been demonstrated as being capable of providing erosion resistance without the potential radiological effects of cobalt-bearing material use.

Besides erosion, another significant result of cavitation is vibration. Within a valve, it can result in cracked welds or loosened parts of the valve or other adjacent equipment.

The use of appropriate control valve trim is a preferred means of addressing cavitation concerns, since it can help minimize cavitation-caused erosion and resultant vibrations.

PNL–5722, D. E. Blahnik and R. L. Goodman, "Operating Experience and Aging Assessment of ECCS Pump Room Coolers," Pacific Northwest Laboratory, October 1986.

This report provides a preliminary aging assessment of safety-related room coolers for the emergency core cooling system (ECCS) pump rooms in nuclear power plants. The assessment conforms to the NRC NPAR program strategy and is based on limited information obtained through public and private data bases, equipment vendors, utility contacts, literature searches, and expert opinion.

Description of the ECCS pump room cooler systems were based on FSARs and vendor-supplied information. Data from LERs, review of maintenance requests at a reactor plant, and discussions with personnel that do utility repair and maintenance work were used to determine the operating experience of pump room coolers. Failure modes, causes, frequency rates, and methods of detection are summarized from the operating records. Maintenance actions and modifications needed as a result of the operator experience are addressed. Operational stressors are summarized, manufacturer recommendations for maintenance and surveillance are listed, and aging and service-wear monitoring are briefly evaluated.

PNL–6287, K. R. Hoopingarner, B. J. Kirkwood, and P. J. Lonzecky, "Study Group Review of Nuclear Service Diesel Generator Testing and Aging Mitigation," Pacific Northwest Laboratory, March 1988.

As part of the NPAR program, the Pacific Northwest Laboratory is performing a diesel generator aging assessment study. In the on-going NPAR Phase II of the aging study, efforts have been focused on aging mitigation and other success strategies for improving nuclear plant diesel generator operation and maintenance and also increasing its reliability.

A study group of diesel experts, the authors of this report, met on April 29 and 30, 1987, to resolve issues on mitigating diesel generator aging and improving operations, testing, and maintenance. The focus of the study group was to (1) address the diesel generator aging stressors resulting from the present periodic testing practices of the nuclear industry and (2) propose potential mitigating measures. A new recommended testing program was developed and is documented in this report. The report lays out the conclusions and recommendations of the study group. The experts agreed that, if these recommendations are put into practice, many of the engine aging stressors (e.g., those due to fast start) could be reduced or eliminated; another consequence could be a reduction of failures and an improvement in operability and reliability.

PNL-7516, K.R. Hoopingarner, "Emergency Diesel Generator Technical Specifications Study Results," Pacific Northwest Laboratory, March 1991.

This report covers a study in support of the NRC NPAR program on the effects of aging on emergency diesel generators (EDG). The research was performed in two phases. Phase I used plant operating experience, data, expert opinion, and statistical methods to produce a new data base related to aging, reliability, and operational readiness of nuclear service diesel generators. Phase II was chiefly concerned with measures for mitigating the effects of aging.

Insights from a number of sources indicate that there are many opportunities for improving the management of EDG systems. Existing technical specifications, for example, could be modified to yield significant safety benefits by reducing direct effects of aging and increasing system reliability. Thus technical specifications related to the management, testing, and reliability of emergency diesel generators were reviewed. Beneficial specifications were identified as were those that could adversely influence aging and reliability.

Potential improvements in technical specifications and engine and system management aimed at reducing aging effects and increasing reliability would:

1. Significantly reduce the number of total engine starts,
2. Reduce the load application rates for testing purposes by gradually adding load,
3. Reduce the EDG testing loads to 90% of the continuous load rating or to the plant emergency unit load, whichever is less,
4. Increase the maximum EDG start time to 25 to 30 seconds,
5. Make necessary changes to support the reliability emphasis of Regulatory Guide 1.9, Revision 3, and delete statistical emphases,
6. Address fuel oil storage management to permit flexibility and the use of a large fraction of stored fuel before replacement,
7. Eliminate many unnecessary and partially redundant tests and engine starts in the 18-month test period (including those due to false signals),
8. Eliminate, where possible, short engine run times and excessive idle times.

The technical bases for such changes to the specifications are obtained from the NRC NPAR program and from research sponsored by the Electric Power Research Institute (EPRI) and the Nuclear Safety Analysis Center (NSAC) operated by EPRI. Fast starts, fast loading, and the large number of test runs are cited as acting to increase diesel generator stress and wear. The results from this study confirm these stressors and add excessive testing loads as another important stressor.

PNL-7823, A.D. Chockie, K.A. Bjorkelo, T.E. Fleming, W.B. Scott, and W.I. Enderlin, "Maintenance Practices To Manage Aging: A Review of Several Technologies," Pacific Northwest Laboratory, October 1991. Available from the NRC Public Document Room.

The quality of a maintenance program directly affects the ability of a nuclear power plant to detect and mitigate the effects of age-related degradation. The Nuclear Plant Aging Research (NPAR) Program, which is sponsored by the NRC Office of Nuclear Regulatory Research, has contracted this research in order to analyze effective maintenance activities for the management of aging of systems and components.

The maintenance programs used by two commercial industries and two military organizations were selected for this study—the U.S. commercial airline industry, the Japanese nuclear power industry, the U.S. Air Force B-52 bomber, and the U.S. Navy Ballistic submarine.

The maintenance programs of these four entities were examined in this report since they offer valuable lessons for managing aging in the U.S. nuclear power industry. Specifically, they indicate the need for an effective maintenance program to manage the aging degradation of critical systems and components. Such a maintenance program should include three basic elements:

1. A systematic approach to the conduct of maintenance tasks,
2. Methods for monitoring and assessing maintenance activities, and
3. Mechanisms for feedback and corrective actions to improve maintenance effectiveness.

A systematic approach to maintenance includes a comprehensive maintenance policy, clear maintenance program objectives and goals, and the physical conduct of maintenance based on the overall policy, objectives, and goals.

The structure of the maintenance program is important to ensure that aging issues are addressed. This analysis identified four elements inherent in an effective maintenance program that are also important to an aging management program. The elements are selecting critical systems and components, understanding aging through the collection and analysis of equipment performance information, mitigating aging by conducting necessary maintenance, and the use of feedback to improve the aging management program.

Critical systems and components can be selected by several methods. For example, the aviation industry's reliability-centered maintenance program uses a risk-based approach to identify safety-significant equipment.

Understanding of aging processes is accomplished by collecting and analyzing relevant operating characteristics and performance data on the equipment. The maintenance program should be designed to detect, identify, and correct problems caused by aging mechanisms, such as corrosion, wear, and fatigue, before the safety or reliability of the plant is impaired.

Once the aging process of critical equipment is understood, maintenance tasks designed to detect and correct equipment degradation can be selected and scheduled, includeing inspection, surveillance, equipment monitoring, replacement, and overhaul.

Feedback mechanisms ensure continual maintenance program refinement and improvement. These mechanisms may consist of specific maintenance activities that serve as a basis for establishing and scheduling future inspection and maintenance tasks. Feedback is also obtained from groups of personnel who evaluate and improve the maintenance programs. Feedback mechanisms are vital if the maintenance program is to address changing conditions such as the degradation of plant equipment from aging. A summary of the maintenance-related activities to address system and component aging is presented in tabular form.

PNL-SA-18407, A.B. Johnson, Jr., D.B. Jarrell, U.P. Sinha, and V.N. Shah, "Understanding and Managing Corrosion in Nuclear Power Plants," Pacific Northwest Laboratory, August 1990.

The concept of understanding and managing corrosion in nuclear power plants is not new—in various forms, this main theme of the report has been applied throughout the development and maturing of nuclear technology. Too often, however, understanding corrosion has been based on reacting to it rather than on anticipating its occurrence. Regulatory and utility initiatives are creating a climate and framework for more effective application of the concept. This report characterizes the framework and provides some illustrations of how the concept is being applied in support of the NRC NPAR.

Although corrosion has not caused a major accident in a nuclear power plant, it has been a continuing cause of an overall untimely degradation of the plants and, in particular, of nuclear safety-related systems and components, often resulting in reactor shutdowns and extended outages. Examples of major corrosion-induced degradation include intergranular stress corrosion cracking (IGSCC) of piping in boiling water reactor recirculation bypass systems and denting, pitting, intergranular attack (IGA), and IGSCC of steam generator tubes in pressurized water reactors. These types of degradation involve significant phenomena that have been widely recognized and investigated. Major investments have been made to detect and mitigate the effects of these corrosion mechanisms. Numerous other mechanisms and phenomena have been observed in nuclear systems and components; some are obvious, some subtle.

The failure of a valve to open or close on demand, for example, may be a consequence of corrosion. This and similar considerations increase in importance as plants age but take on an additional dimension with consideration of extending the licenses of the plants beyond the current 40-year period. Both the nuclear utilities and the NRC are considering in detail the degradation mechanisms that may have special significance in license renewal.

Mitigation of corrosion impacts in nuclear facilities must involve more than technical considerations. It must involve attitudes of alertness and commitment on the part of regulators, plant management, and the plant work force. It also requires timely and ample allocation of resources.

There is a trend in the nuclear industry to advance the management of aging phenomena, including corrosion. The fact that new corrosion phenomena continue to emerge, however, provides evidence that principles of corrosion control still need to be aggressively applied. Lessons learned in current reactor operation need to be systematically and effectively applied to extended operation and advanced reactor designs.

This report defines the concept of understanding and managing corrosion, references relevant regulatory and industry initiatives, and focuses on an overview of how the concept is being applied, drawing on results from the NPAR program. The overview includes a brief survey of corrosion impacts on major structures, systems, and components, including service water, steam generators, piping, and containment. Mitigation methods are briefly reviewed. The overview is referenced to a major data base that is being developed to assist both utilities and regulators in the important and responsible task of understanding and managing corrosion and other degradation mechanisms in nuclear plants. An effective application of understanding and managing these mechanisms is crucial not only to safe and economic operation of the nuclear plants, but also to public perception of a soundly designed, managed, and operated technology.

PNL-SA-20219, D. P. Brown and D. E. Blahnik, "ASME Subsection ISTD Recommendations Based upon NPAR Snubber Aging Research Results," Pacific Northwest Laboratory, December 1991.

As a result of information obtained through the NRC's Nuclear Plant Aging Research (NPAR) Program and from snubber task research, recommendations were made in three areas for the next revision of the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code, Subsection ISTD (In- Service Testing-D): (1) service-life monitoring, (2) visual examination attributes, and (3) failure grouping and corrective actions.

A service-life monitoring program will be most effective if it distinguishes between service-related and non-service-related failures. It is important that the root cause of snubber failure or degradation (e.g., dynamic transient, vibration, excessive temperature), be identified along with the failure mode (e.g., high drag force, low activation) and the failure mechanism (e.g., deformation of screw or ball shaft, solidification of grease). This information provides the basis to take effective countermeasures.

It is suggested that failure evaluation data sheets provide key information, including failure mode, failure mechanism, failure cause, environment, service time, abnormal conditions, test data, visual observations, and other test-related observations. They include trending and diagnostic tests and post-service as- found tests. Some of them might be included in ISTD. If included, an appropriate statement should be written.

For effective failure cause and root cause determination, it is important that personnel involved in failure evaluation have adequate experience. Failure evaluation data sheets should not be formatted in a manner that might lead the examiner toward a potentially incorrect failure cause.

Suggested service-life monitoring techniques can be added to those in the ISTD, Section 7.0. They take into consideration the capability of the various snubber models to endure the full range of plant environments (from benign to severe). Previously unrecognized severe environments may often be identified by root cause evaluation of failed or degraded snubbers. Information regarding the snubber endurance capability is often obtained from operating experience (i.e., from failure data or by monitoring the degradation).

SAND88-0754 UC-78, K. T. Gillen and R. L. Clough, "Time-Temperature-Dose Rate Superposition: A Methodology for Predicting Cable Degradation Under Ambient Nuclear Power Plant Aging Conditions," Sandia National Laboratories, August 1988.

Time-temperature superposition is an empirical approach that has been used in polymers for more than 30 years to make thermal aging predictions during experimentally inaccessible times. Given the historical success of time-temperature superposition, the authors have expanded this approach for combined radiation-thermal environments, yielding an empirical time-temperature-dose-rate shifting procedure. The procedure derives an isothermal curve for a given amount of material damage versus dose rate at a selected reference temperature. This is done by finding the Arrhenius activation energy that causes higher-temperature dose-rate data to superpose when shifted to the reference temperature. The resulting superposed curve at the reference temperature extends to much lower dose rates that are experimentally inaccessible because of the long time periods that would be required to simulate aging. This procedure therefore allows meaningful predictions to be made for long-term, low-dose-rate radiation aging conditions. Using historical data from Sandia's radiation-aging program on nuclear power plant cable materials, the authors have successfully applied the time-temperature-dose-rate superposition approach to four different materials: hypalon, neoprene, polyethylene, and PVC jacket material. For two of these materials, extrapolated predictions based on the superimposed data were found to be in excellent agreement with 12-year, low-dose-rate nuclear power plant results.

WYLE 60103-X, J. F. Gleason, R. A. DeFour, J. M. Hammond, and P. A. Lubeski, "Test Plan for the Comprehensive Aging Assessment of Circuit Breakers and Relays for Nuclear Plant Aging Research (NPAR) Program, Phase II," Wyle Laboratories, Huntsville, AL, July 1989.

This entry refers to seven individually bound reports, each presenting the test plan for a specific type of circuit breaker or relay:

60103-1	Molded-case circuit breakers
60103-2	Metal-clad circuit breakers
60103-3	Auxiliary relays
60103-4	Control relays
60103-5	Protective relays
60103-6	Timing relays
60103-7	Electronic relays

The purpose of these reports is to provide details of the tests planned for the types of circuit breakers and relays under investigation in Phase II of the Comprehensive Aging Assessment of Circuit Breakers and Relays. This work is being performed by Wyle Laboratories for the NRC NPAR program, which is intended to resolve technical safety issues related to the aging degradation of electrical and mechanical components, safety systems, support systems, and civil structures used in commercial nuclear power plants. The aging period of interest includes the period of normal licensed plant operation as well as the period of extended plant life that may be requested in utility applications for license renewals.

The Phase I report, NUREG/CR-4715, "An Aging Assessment of Relays and Circuit Breakers and System Interactions," showed that relays and circuit breakers are important nuclear plant components that are susceptible to degradation with time. Thus Phase II, a comprehensive aging assessment of relays and circuit breakers, was implemented to provide (1) a review and verification of improved inspection, surveillance, monitoring, and maintenance methods; (2) in situ examinations and data gathering for operating equipment; (3) postservice examinations of naturally aged components or components with simulated degradation; (4) an evaluation of the role of maintenance in mitigating the effects of aging; (5) evaluations of methods for predicting residuals and service life; and (6) recommendations for using research results in the regulatory process. Specific goals of the program are:

1. To identify and characterize aging effects that, if unchecked, could cause degradation of components and subsystems of circuit breakers and relays and thereby impair plant safety.
2. To identify methods of inspection, surveillance, and monitoring and to evaluate the residual life of components and subsystems of circuit breakers and relays that will ensure timely detection of significant aging effects before loss of their safety function.
3. To evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

Methods are available to detect and mitigate aging degradation and thereby to minimize its impact. The reports describe the background of the research strategy, list and elaborate on the objectives of the research, and define the testing to be performed on naturally aged and degraded equipment in order to determine the methods most effective for detecting age degradation. Emphasis has been placed on identifying and characterizing the mechanisms of material and component degradation during service and using research results in the regulatory process.

The testing consists of performing and evaluating various methods of inspection, surveillance, condition monitoring, and maintenance, including simulated degradation, to aid in determining the usefulness of these methods for managing the effects of aging on safe plant operation. The devices chosen for testing are representative of circuit breakers and relays that have been in use in nuclear power plants. New, used, and aged specimens up to 40 years old have been located from a variety of sources, including Wyle stock and nuclear plants.

PERSONAL AUTHOR INDEX

This index lists, in alphabetical order, all participating authors of each report listed in the main citation listing. Each name is followed by the number and the title of the reports prepared by the author. If further information is needed, refer to the main citation by the report number.

NUREG/CR-6001, "Aging Assessment of BWR Standby Liquid Control Systems."

PNL-SA-18407, "Understanding and Managing Corrosion in Nuclear Power Plants."

Johnson, J.W.

NUREG/CR-3154, "The In-Plant Reliability Data Base for Nuclear Plant Components: Interim Report—The Valve Component."

Kahl, W.K.

NUREG/CR-3154, "The In-Plant Reliability Data Base for Nuclear Plant Components: Interim Report—The Valve Component."

Kalsi, M.S.

NUREG/CR-5159, "Prediction of Check Valve Performance and Degradation in Nuclear Power Plant Systems."

NUREG/CR-5583, "Prediction of Check Valve Performance and Degradation in Nuclear Power Plant Systems—Wear and Impact Tests."

NUREG/CR-5807, "Improvements in Motor Operated Gate Valve Design and Prediction Models for Nuclear Power Plant Systems."

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Lewis, R.

NUREG/CR-5051, "Detecting and Mitigating Battery Charger and Inverter Aging."

Lodlow, C.C.

NUREG/CR-4715, "An Aging Assessment of Relays and Circuit Breakers and System Interactions."

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NUREG/CR-5515, "Light Water Reactor Pressure Isolation Valve Performance Testing."

Engineering and Economics Research, Inc. (EER)

Letter Report, L. N. Rib, "Summaries of Research Reports Submitted in Connection with the Nuclear Plant Aging Research (NPAR) Program."

Franklin Research Center

NUREG/CR-4715, "An Aging Assessment of Relays and Circuit Breakers and System Interactions."

NUREG/CR-5141, "Aging and Qualification Research on Solenoid Operated Valves."

Idaho National Engineering Laboratory (INEL)

EGG-SSRE-8972, "Estimating Hazard Functions for Repairable Components."

EGG-SSRE-9017, "User's Guide to PHAZE, a Computer Program for Parametric Hazard Function Estimation."

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NUREG/CR-4731, "Residual Life Assessment of Major Light Water Reactor Components—Overview," Vol. 2.

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NUREG/CR-4967, "Nuclear Plant Aging Research on High Pressure Injection Systems."

NUREG/CR-4977, "SHAG Test Series: Seismic Research on an Aged United States Gate Valve and on a Piping System in the Decommissioned Heissdampfreaktor (HDR): Summary," Vol. 1.

NUREG/CR-4977, "SHAG Test Series: Seismic Research on an Aged United States Gate Valve and on a Piping System in the Decommissioned Heissdampfreaktor (HDR): Appendices," Vol. 2.

NUREG/CR-5181, "Nuclear Plant Aging Research: The 1E Power System."

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NUREG/CR-5314, "Life Assessment Procedures for Major LWR Components," Vol. 3, "Cast Stainless Steel Components."

NUREG/CR-5378, "Aging Data Analysis and Risk Assessment—Development and Demonstration Study."

NUREG/CR-5406, "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 1, "Analysis and Conclusion."

NUREG/CR-5406, "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 2, "Data Report."

NUREG/CR-5406, "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," Vol. 3, "Review of Issues Associated with BWR Containment Isolation Valve Closure."

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Pacific Northwest Laboratory (PNL)

Pentek, Inc.

Sandia National Laboratories (SAND)

Science Applications International Corp.

Wyle Laboratories

SUBJECT INDEX

In this index, each report is listed under one or more of the following subjects:

1. Aging, Including Plans, Surveys, Analyses, Methods, and Models.
2. Diesel Generators and Related Systems.
3. Electric Power Systems, Including Cables, Trays, Connectors, Circuit Breakers, Relays, Switches, Penetrations, and Related Components.
4. Electrical Equipment, Including Transformers, Motors, Batteries, Chargers, and Inverters.
5. Instrumentation, Measurement, and Control Systems.
6. Maintenance.
7. Major Components: Reactor Vessels, Reactor Coolant Pumps, Steam Generators, Pressurizers, and Structures (Including Basemat and Containment).
8. Monitoring.
9. Operating Experience, Field Results, and Related Data.
10. Piping, Including Valves, Seals, Supports, Snubbers, and Related Components.
11. Probabilistic Risk Assessment (PRA).
12. Safety and Protection Systems (Including Injection Systems) and Their Components.
13. Seismic Effects and Aging.
14. Service Water, Auxiliary Feedwater, Instrument Air, and Other Fluid Systems, Including Their Pumps, Heat Exchangers, and Related Components; Balance of Plant Systems and Components.

These subjects are not intended to include every subject covered in all the reports listed. Nor do they represent a "standard" or "official" list of subjects. They were selected to be most helpful to knowledgeable personnel seeking published information on the various aspects of nuclear plant aging.

1. Aging, Including Plans, Surveys, Analyses, Methods, and Models

NUREG/CR-4652, "Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants."

NUREG/CR-4731, "Residual Life Assessment of Major Light Water Reactor Components," Vol. 1.

NUREG/CR-4731, "Residual Life Assessment of Major Light Water Reactor Components—Overview," Vol. 2.

NUREG/CR-4769, "Risk Evaluations of Aging Phenomena: The Linear Aging Reliability Model and Its Extensions."

NUREG/CR-4819, "Aging and Service Wear of Solenoid-Operated Valves Used in Safety Systems of Nuclear Power Plants," Vol. 2, "Evaluation of Monitoring Methods."

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2. Diesel Generators and Related Systems

"Safety Implications of Diesel Generator Aging," *Nuclear Safety*, December 1990.

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3. Electric Power Systems, Including Cables, Trays, Connectors, Circuit Breakers, Relays, Switches, Penetrations, and Related Components

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NISTIR 4787, "The Use of Time-Domain Dielectric Spectroscopy To Evaluate the Lifetime of Nuclear Power Station Cables."

NUREG/CR-3956, "In Situ Testing of the Shippingport Atomic Power Station Electrical Circuits."

NUREG/CR-4257, "Inspection, Surveillance, and Monitoring of Electrical Equipment Inside Containment of Nuclear Power Plants—With Applications to Electrical Cables."

NUREG/CR-4715, "An Aging Assessment of Relays and Circuit Breakers and System Interactions."

NUREG/CR-4731, "Residual Life Assessment of Major Light Water Reactor Components--Overview," Vol. 2.

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NUREG/CR-4564, "Operating Experience and Aging-Seismic Assessment of Battery Chargers and Inverters."

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7. Major Components: Reactor Vessels, Reactor Coolant Pumps, Steam Generators, Pressurizers, and Structures (Including Basemat and Containment)

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The U.S. Nuclear Regulatory Commission is conducting the Nuclear Plant Aging Research (NPAR) Program. This is a comprehensive hardware-oriented engineering research program focused on understanding the aging mechanisms of components and systems in nuclear power plants. The NPAR program also focuses on methods for simulating and monitoring the aging-related degradation of these components and systems. In addition, it provides recommendations for effective maintenance to manage aging and for the implementation of the research results in the regulatory process.

This document contains a listing and index of reports generated in the NPAR program that were issued through September 1933 and summaries of those reports. Each summary describes the elements of the research covered in the report and outlines the significant results. For the convenience of the user, the reports are indexed by personal author, corporate author, and subject.

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