NUREG-1144
Rev. 1

Nuclear Plant Aging Research (NPAR) Program Plan

Components, Systems and Structures

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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Rev. 1
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Nuclear Plant Aging Research (NPAR) Program Plan

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ABSTRACT

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 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.

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

- Identify and characterize aging effects that, if unchecked, could cause degradation of components, systems, and civil structures and thereby impair plant safety.
- Identify methods of inspection, surveillance, and monitoring, and evaluate residual life of components, systems, and civil structures that will ensure timely detection of significant aging effects before loss of safety function.
- Evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

The NPAR Program is based on a phased approach to research. The objectives of the Phase I studies are: to identify and characterize aging and wear effects; to identify failure modes and causes attributable to aging; and to identify measurable performance parameters, including functional indicators. The functional indicators have a potential use in assessing operational readiness of a component, structure, or system in establishing degradation trends, and in detecting incipient failures.

The objectives of the Phase II studies are: perform indepth engineering studies and aging assessments based on in situ measurements; perform postservice examinations and tests of naturally aged/degraded components; and identify improved methods for inspection, surveillance, and monitoring, or for evaluating residual life, and make recommendations for utilizing research results in the regulatory process.

The objective of the projected Phase III or the extended portion of research is to provide for the resolution of issues that may be raised during the "results utilization efforts."

FOREWORD

The U.S. Nuclear Regulatory Commission's (NRC's) hardware-oriented engineering research program for plant aging and degradation monitoring of components and systems was first discussed in the initial version of the program plan, NUREG-1144, issued in July 1985. It was stated in the plan that NUREG-1144 would be a living document and would be revised periodically. The revisions would reflect the experience gained in implementing the plan and incorporate comments received from within the NRC, industrial codes and standards committees, and domestic and foreign organizations and institutions.

The Office of Nuclear Regulatory Research (RES) staff has received numerous comments from various offices within the NRC as well as from individuals, organizations, and institutions outside NRC, both domestic and foreign, since issuing the original program plan. The NRC provided planning guidance for needed safety research on plant aging and license renewal in its 1986 Policy and Planning Guidance document (NUREG-0885, Issue 5). The Executive Director for Operations provided specific program guidance to the staff for FY 1986 to 1988 planning and program development. The NRC staff provided their comments on the current research program and needs for additional research and prioritization by "user-need" letters to RES and through the Technical Integration Review Group for Aging and Life Extension review of the Nuclear Plant Aging Research (NPAR) Program.

As a part of the overall phased approach to aging research, significant progress has been made in completing the Phase I engineering research for selected components and systems during the past 24 months. These components and systems include: motor-operated valves, check 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 relative impacts of aging on risk. The Phase I segment of research for evaluating systems-level aging effects, from operating experience and risk evaluation of aging phenomenon, has been completed. In consideration of plant life extension/license renewal, progress has been made in identifying major technical safety issues and defining major light water reactor components and structures according to their risk significance. A preliminary study also has been completed identifying degradation sites and life-limiting processes for each major component. Finally, more has been learned from operating experience and from expert opinions.

Reflecting all the aforementioned inputs, this document presents a revised research plan, which addresses identifying and resolving technical safety issues relevant to plant aging and license renewal. This plan focuses on plant safety systems, electrical and mechanical components, civil structures, and the utilization of technical data in the regulatory process.

This program plan for components, systems, and civil structures, in conjunction with its sister plan for primary system pressure boundary components, form the overall framework for NPAR within the Division of Engineering, Office of Nuclear Regulatory Research of the NRC.

Comments on this document are welcome and will be considered in developing subsequent editions of this plan. Comments need not be restricted to the research activities described herein; comments identifying omissions and/or recommending additional research are also welcome.

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ACRONYMS AND INITIALISMS

ACRS	Advisory Committee on Reactor Safeguards
AECB	Atomic Energy Control Board
AEOD	Office of Analysis and Evaluation of Operational Data (NRC)
AFWP	Auxiliary Feedwater Pump
AIF	Atomic Industrial Forum
ALEXCC	Aging and Life Extension Coordinating Committee
ASME	American Society for Mechanical Engineers
ASPS	Accident Sequence Precursor Study
BNL	Brookhaven National Laboratory
BV	Block valve
BWR	Boiling Water Reactor
CB	Circuit Breaker
CCW	Component Cooling Water
DE	Division of Engineering
DOE	Department of Energy
DRAA	Division of Reactor Accident Analysis (NRC)
DRPS	Division of Reactor and Plant Systems (NRC)
ECCAD	Electrical Circuit Characterizations and Diagnostics
ECCS	Emergency Core Cooling System
EDO	Executive Director for Operations
EMEB/DE	Electrical and Mechanical Engineering Branch of the Division of Engineering
EPRI	Electric Power Research Institute
FRG	Federal Republic of Germany
HPCI	High Pressure Coolant Injection System (PWRs)

IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronic Engineers
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IPRDS	In-Plant Reliability Data System
IS&MM	Inspection, Surveillance and (condition) Monitoring Methods
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCC	Motor Control Center
MEB/DE	Materials Engineering Branch of the Division of Engineering
MIC	Microbiologically Influenced Corrosion
NBS	National Bureau of Standards
NDE	Nondestructive Examination
NOAC	Nuclear Operations Analysis Center (ORNL)
NPAR	Nuclear Plant Aging Research
NPRDS	Nuclear Plant Reliability Data System
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation (NRC)
NSAC	Nuclear Safety Analysis Center operated by the nuclear industry-supported Electric Power Research Institute (EPRI)
NUMARC	Nuclear Utility Management and Resources Committee
NUPLEX	Nuclear Utility Plant Life Extension
OL	Operating License
ORNL	Oak Ridge National Laboratory
OSRR	Operational Safety Reliability Research

PNL	Battelle Pacific Northwest Laboratories
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QOA	Quantification of Aging
RCIC	Reactor core isolation cooling
RES	Office of Nuclear Regulatory Research
RHR	Residual Heat Removal
RPS	Reactor Protection System
SCSS	Sequence Coding and Search System
SEA	Systems Engineering Associates
SNL	Sandia National Laboratories
SRP	Standard Review Plan
SSEB/DE	Structural and Seismic Engineering Branch of the Division of Engineering
SWS	Service Water System
TIRGALEX	Technical Integration Review Group for Aging and Life Extension

1. INTRODUCTION

Since the early 1980s, it has become clear that the current generation of commercial nuclear power plants has gone beyond the development stage and is reaching a stage of relative maturity. The prototype reactors of the late 1950s and early 1960s in the United States have led to the development of two types of commercial light water reactors (LWRs): the pressurized water reactor (PWR) and the boiling water reactor (BWR). The United States now has approximately 100 reactors in commercial operation and a few of these reactors have been operating for over 20 years. As the population of LWRs has matured and advanced in age, the need for a research program that would provide a systematic assessment of the effects of plant aging on safety was recognized. The Director of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission (NRC), in his comments on the Long-Range Research Plan, identified a need for a research program to investigate the safety aspects of aging processes in commercial nuclear power plants. Initiating an aging research program was also recommended by the Advisory Committee on Reactor Safeguards (ACRS) in their 1983 report to Congress.

The NRC provided guidance for needed safety research on plant aging and license renewal in its Policy and Planning Guidance document (NUREG-0885). Also, the Executive Director for Operations (EDO) has provided specific program guidance to the staff for FY 1986 to 1988 planning and program development.

The NRC Office of Nuclear Regulatory Research (RES) has developed and implemented a hardware-oriented engineering research program for plant aging and degradation monitoring of components and systems. This program is called the Nuclear Plant Aging Research (NPAR) Program, first described in the July 1985 issue of NUREG-1144 (Ref. 1), and discussed at length at the July 1985 International Conference on Nuclear Plant Aging, Availability Factor, and Reliability Analysis (Ref. 2). This report describes the NPAR Program for components, systems, and civil structures, which is being conducted by the Electrical and Mechanical Engineering Branch of the Division of Engineering (EMEB/DE). A similar program on aging that focuses on vessels, piping, steam generators, and nondestructive examination techniques is being conducted by the Materials Engineering Branch of the Division of Engineering (MEB/DE). The program plan developed by MEB/DE is a sister plan to the NPAR plan. The two plans form the overall framework for aging research within the Division of Engineering, Office of Nuclear Regulatory Research of the NRC.

Significant progress has been made since issuing the original program plan. The Phase I engineering research has been completed for selected components and systems. These components and systems include: motor-operated valves, check valves, auxiliary feedwater pumps, emergency diesel generators, electric motors, batteries, chargers and inverters, and circuit breakers and relays in safety-related systems and reactor protection systems. Also, onsite assessments of electrical circuits have been performed and aged components and materials are being retrieved from the Shippingport Atomic Power Station. Progress also has been made in

developing models and approaches to evaluate relative impacts of aging on risk. The main objective of this document is to revise the original research plan by incorporating what has been learned from the NPAR Program activities and the comments received from the various industry and government institutions and organizations, domestic and foreign.

This revised NPAR Program Plan describes the research effort currently being implemented to resolve the technical safety issues relevant to plant aging and operating license renewal and describes the utilization of the technical data in the regulatory process.

1.1 Background and Need

Aging affects all reactor structures, systems, and components to various degrees. For the NPAR Program, aging refers to the cumulative degradation of a system, component, or structure that occurs with time, and, if unchecked, can lead to an impairment of continuing safe operation of a nuclear power plant as it advances in age. Necessary measures must be taken to ensure that age-related degradation does not reduce the operational readiness of a plant's safety systems, components, and structures and does not result in common-mode failures of redundant, safety-related equipment, thus reducing defense in depth. It is also necessary to ensure that aging does not lead to failure of equipment in a manner that causes an accident or severe transient.

To establish a perspective for describing the NPAR Program, it is of interest to examine the current status of commercial operating nuclear power plants. As of June 1987, there were 102 licensed commercial power plants in operation in the U.S. The age distribution of these plants is listed below.

Operating Lifetime [Years Since Operating License (OL)]	Number of Plants
More than 20	3
Between 15 and 20	17
Between 10 and 15	40
Between 5 and 10	13
less than 5	29

The two oldest operating plants, Yankee Rowe (OL Date--July 9, 1960) and Big Rock Point (OL Date--August 30, 1962), have been in operation for 26 and 24 years, respectively. However, they are demonstration plants with design power of <200 MWe. The next oldest plant is San Onofre 1 (OL Date--March 27, 1967), which has a net capacity of 430 MWe. In

addition to the plants currently in operation, there are approximately 18 more plants under construction. Most of these plants are expected to be in operation within the next decade.

As the population of U.S. LWRs has matured, problems have already occurred that are the result of time-dependent degradation mechanisms such as stress corrosion, thermal aging, radiation embrittlement, fatigue, and erosion. These problems include failures in pumps, valves, and relays, embrittlement of cable insulation, and cracking of the heat-treated anchor heads for posttensioning systems in containment. Although progress is being made to mitigate the age-related degradation that has already been identified, significant questions still remain because of the variety of components in a commercial power reactor, the complexity of the aging process, and the limited experience with prolonged operation of these power plants.

The NPAR Program has been developed to provide a systematic research effort into how aging affects the safety of the plants currently in operation. This program provides a comprehensive effort to: learn from operating experience and expert opinion; identify failures due to age degradation; foresee or predict safety problems resulting from age-related degradation; and develop recommendations for surveillance and maintenance procedures that will alleviate aging concerns.

The aging program also provides key information to enable the NRC to resolve technical safety issues and define its policy and regulatory position on plant life extension and license renewal. License renewal in this document refers to renewing an OL. Reactors are licensed for up to 40 years of operation under the current regulations. Current regulations also permit license renewal. The Technical Integration Review Group for Aging and Life Extension (TIRGALEX) developed a working definition for life extension. Life extension is defined to include license renewal beyond the original license term of 40 years and a program for systematic hardware renewal of plant systems, equipment, components, and structures.

Utilities currently are planning to apply for license renewals and have defined a tentative schedule for several key steps in the process. Two representative LWRs have been the subject of an EPRI/DOE utility-sponsored pilot study on plant life extension (Ref. 3). The two plants that are the subject of this project are Monticello, a 545-MWe BWR (OL Date--September 8, 1970) and Surry 1, a 788-MWe PWR (OL Date--May 25, 1972). At a technical level, this project is to provide an initial evaluation of the effects of aging on commercial nuclear plants and establish the scope of the effort needed to extend the operating lifetime of these plants beyond their initial 40 years of licensed operation. The first submittal to the NRC is expected in 1993. A large number of additional submittals for license renewal can be expected shortly thereafter. To keep pace with these industry plans and prepare for the large number of submittals, the NRC will need to devote substantial efforts over the next several years to define the requirements for license renewal. The first license for a large plant (>400 MWe) will not expire until about the year 2007 (assuming the license term is defined from

OL issue date). However, the utilities need to decide between requesting a plant license renewal or planning new generating capacity approximately 10 to 15 years before the end of the licensed period, to allow for the long lead times required for planning and construction. A firm NRC policy will be required for license renewal by early 1990. Based on this policy, appropriate regulations, guides, and review procedures can then be written and issued by 1992, to allow preparation and submittal of the first license renewal application by 1993. Reviewing these applications at this early stage will show the viability of the life extension option in sufficient time (by 1995) for a utility to elect an alternative option, if necessary.

Thus, the NRC needs to clearly define its policy and regulatory positions in the near future to ensure the safe operation of aged plants during the current license period and for extended life. Clearly defined policies and criteria are needed to ensure that requests for license renewal address the primary regulatory concerns and issues.

1.2 Framework for Identifying and Resolving Technical Safety Issues

The TIRGALEX was established in 1986 by the EDO to facilitate the planning and integration of NRC plant aging and license renewal/life extension activities. The initial objectives of TIRGALEX have been to clearly define the technical safety and regulatory policy issues associated with plant aging and life extension and develop a plan for resolving the issues in a timely, well-integrated and effective manner.

Figure 1.1 shows the framework recommended by TIRGALEX, and adopted in the NPAR Program, for planning and integrating agency activities related to plant aging and license renewal/life extension. As can be seen on the left side of the figure, technical information on aging and license renewal is already being developed by a variety of sources. This information is compiled and will be updated periodically by RES to ensure that all NRC offices involved in aging and license renewal have current information on ongoing related efforts.

Using the TIRGALEX Integration Plan, the technical data currently being developed in related projects and the regulatory user needs, identified by the Office of Nuclear Reactor Regulation (NRR), are the key inputs used to establish the priority of the research program elements. The RES then has the responsibility for carrying out the necessary research programs.

Hardware-oriented engineering research needed to resolve the issues related to aging is being conducted in DE where two programs are being conducted. The NPAR Program for components, systems, and structures is being performed by the EMEB/DE. The aging research program on the vessels, piping, steam generator, and nondestructive examination techniques is being performed by the MEB/DE.

As the principal technical elements in these research programs are completed, the data and information is made available for use in the

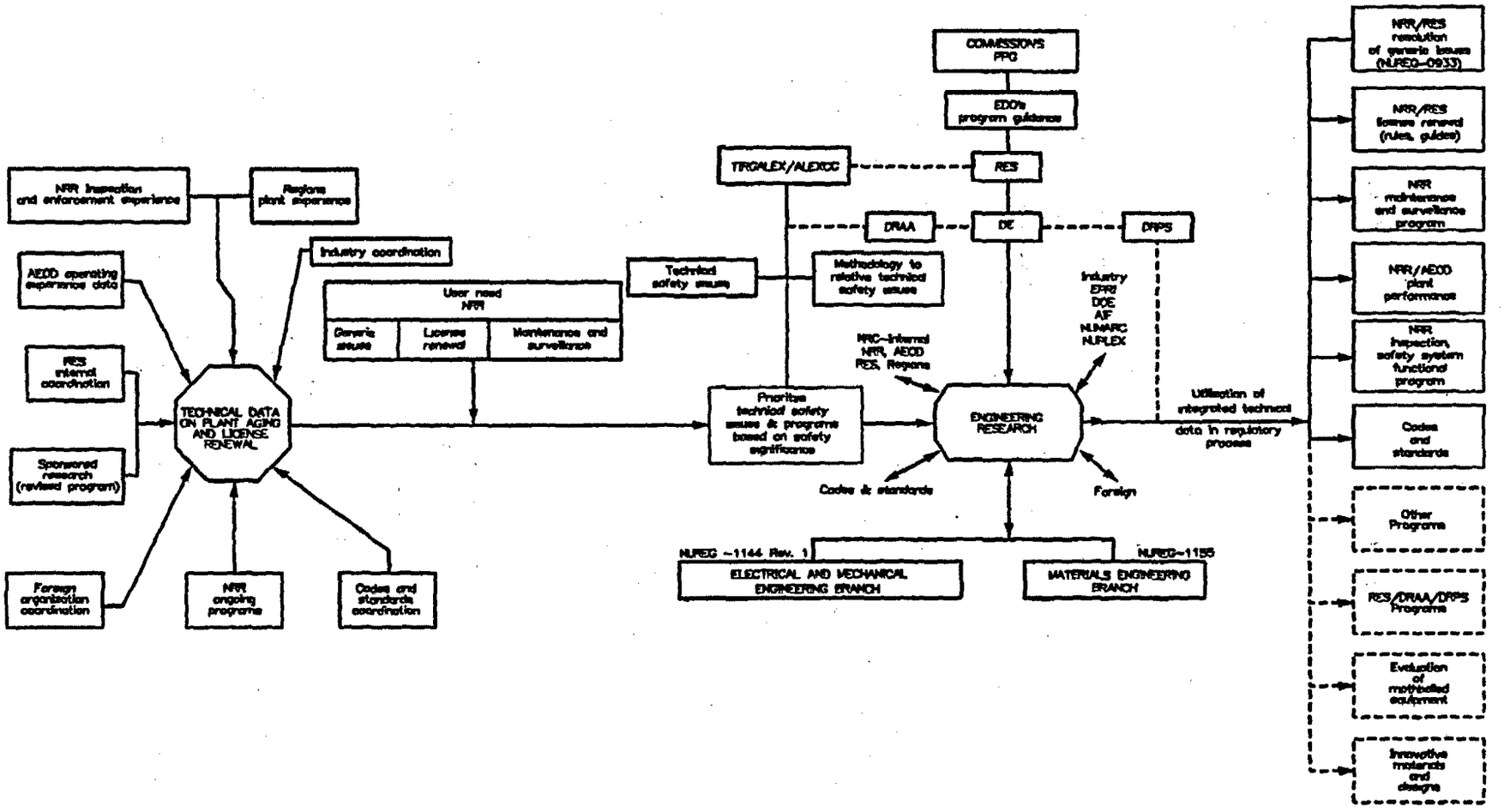


Figure 1.1. NPAR coordination and technical integration.

regulatory process. RES will also make use of research findings as they impact the RES responsibility for developing regulatory criteria, guides and standards, and review procedures.

1.3 Organization of NPAR Plan

Section 2 contains a discussion of the technical safety issues related to aging and safe operation of plants of all ages. The nature of aging processes are discussed first, followed by a discussion of the potential impact of aging on safety and the technical objectives of the research considered in the NPAR Program.

Section 3 contains an overview of the utilization of the NPAR Program results in the regulatory process. The discussion of utilization is divided into seven categories: License Renewal, Generic Issues, Maintenance and Surveillance, Plant Performance, Inspection of Safety Systems and Components, Codes and Standards, and Other Programs. Also, a brief synopsis of the utilization of research results in support of the regulatory process is included.

Section 4 contains an outline of the systematic approach used in the NPAR Program for assessing the effects of aging on plant safety systems, components, and structures. The criteria used to identify systems, components, and structures important to safety are discussed, as is the phased approach developed to study the effects of age-related degradation.

Section 5 contains the description of major program elements and the scope of work for the subjects related to the systems, components, and structures included in the NPAR Program.

Section 6 contains a description of the program coordination and technical integration performed within the NRC, with other government agencies, and with external institutions and organizations, domestic and foreign.

Section 7 contains a discussion of the schedules developed for the various NPAR activities. The schedules include research activities in consideration of license-extension efforts to be completed by the early 1990s, and the continuation of age-related confirmatory research.

Appendix A contains a description of the NPAR strategy and a phased approach used in conducting research. Included also are the methods used for the initial selection of systems and components for aging studies and the development of application guidelines and recommendations.

Appendix B contains a description of the major program elements being addressed by the NPAR Program.

Appendix C contains a description of the research activities being performed as part of the NPAR Program. The scope and current status of each of the major projects are discussed.

Appendix D contains an overview of other ongoing programs related to aging and life extension. The coordination required between the NPAR effort and other ongoing activities is discussed, with emphasis on the need to optimize the use of available resources.

2. TECHNICAL SAFETY ISSUES

A broad set of technical safety issues has been developed to provide focus and direction for the NPAR Program. These issues are based on operating experience, expert judgment, and risk significance. These technical issues include the questions that need to be answered, problems that need to be solved, and measures that must be taken to ensure that safety levels are maintained as the present generation of reactors age. The technical safety issues will be developed further and prioritized by first examining the nature of the aging process, and then examining the potential role aging plays in plant safety and the agency's mission to address plant aging and life extension/license renewal.

The specific technical objectives of the research program have been developed to address this broad set of technical safety issues. The program technical objectives and the technical issues then provide the framework required for developing and guiding the individual research projects in the program.

2.1 Nature of Aging Processes

Commercial nuclear power plants are large engineered complexes comprised of many different systems, components, and structures that cover a broad spectrum of materials and designs. The plants operate in a variety of different environments and must meet different functional requirements. The various components, systems, and structures are inspected and maintained by a variety of methods and general approaches. Consequently, a number of factors can cause degradation of the functional capability of a component, system, or structure. For example:

- Material degradation mechanisms are active during storage and operation. Typical causes of degradation include: neutron embrittlement, fatigue, erosion, corrosion, oxidation, thermal embrittlement, and chemical reactions.

- Stressors can be introduced by improper storage, operating environment, or external environment. Irradiation, primary and secondary coolant chemistry, and vibratory loads are the typical examples of stressors introduced by the operating environment. Freezing and thawing, brackish water, and humidity are typical examples of stressors introduced by external environment. Synergistic influence of electrical and mechanical stressors in combination with other internal and external environment also contribute to degradation processes.

- Service wear: accumulation of fatigue damage due to plant operational cycling, service wear of rotating equipment, and wear of the drive rod assembly in a control rod drive mechanism are typical examples.

- Excessive testing: frequent testing of emergency diesel generators is a typical example.

- Improper installation, application, or maintenance: investigation by NRC (Ref. 4) has indicated that 30% of the nuclear plant abnormal occurrences can be attributed to faulty and improper maintenance.

These factors and others, with time, can act either singly or synergistically to degrade a component, system, or structure.

"Aging" is defined in this report as the cumulative degradation that occurs with the passage of time in a component, system, or structure. This degradation takes place because of one or more of the factors listed above. This degradation can, if unchecked, lead to a loss of function and an impairment of safety. Aging is a complex process that begins as soon as a component or structure is produced and continues throughout its service life. Aging plays a significant role in the operation of a nuclear plant and must be factored into the determination of safe operating lifetime limits. It also is important in the evaluation for license renewal. No nuclear plant, including those still under construction or being mothballed, should be considered immune from its effects.

2.2 Potential Impact of Aging on Safety

The main concern addressed by this research program is that plant safety could be compromised if degradation of key components, systems, or structures is not detected before a loss of functional capability, and timely corrective action is not taken. In this way, aging can result in an undetected reduction in the defense-in-depth concept. The defense-in-depth concept requires the public be protected from the accidental release of fission products by a series of multiple barriers and engineered safety systems.

Age degradation of the reactor components, equipment, and structures can reduce the overall level of safety. Experience at operating power plants provides examples where age degradation of vital components could lead to a loss of the margins provided by the defense-in-depth concept. These examples include items such as failure of emergency diesel generators, degradation of valves, and stress corrosion cracking of heat-treated anchor heads in prestressed concrete containments.

Age degradation of the major components must be evaluated when considering plant life extension and license renewal. The major components are the large, expensive, permanent parts of the reactor system, not routinely replaced or refurbished. Age degradation must be assessed, and an evaluation of the residual life of the major components is required if plant safety is to be ensured during extended life operation.

Age degradation can also cause a loss of operational readiness in engineered safety systems, which are required to mitigate the consequences of a failure of a vital component, such as an assumed break in the primary system boundary. Examples of safety systems are the emergency core cooling system, reactor protection system, and containment spray system.

A survey (Ref. 5) of licensee event reports (LERs) conducted by Oak Ridge National Laboratory (ORNL), as part of the planning for this aging research plan, shows that numerous instances of aging-induced failures of equipment have been reported. The reported events indicate that essentially all types of safety-related systems have been affected by a variety of degradation processes. Also, ORNL described the background of selected age-related LERs in more detail to provide a better perspective regarding the safety significance of age degradation (Ref. 6). Based on these studies, aging effects can contribute to both: (a) the probability of initiation of transients and accidents, and (b) the probability of failure of the mitigating equipment during operation.

Aging can also lead to a higher probability of common mode failures in nuclear power plants. This is an area of potentially the greatest concern. Aging can lead to wide-scale degradation of a physical barrier or to simultaneous degradation of redundant components. If such degradation occurs in part of the reactor coolant pressure boundary, as in the steam generator tubes, then an excess stress, resulting from an event such as a pressure transient or a seismic event, could result in multiple, simultaneous tube failures. This has the potential of releasing radioactivity outside the containment.

A second type of common mode failure is simultaneous failure of redundant components. Age-related degradation can occur in redundant components of safety systems, causing the components to simultaneously fail during a transient or accident. This could lead to loss of functional performance of the safety system. Thus, aging can lead to common mode failures that can result in accident initiation or in loss of safety function and the capability for accident mitigation.

Qualification of electrical equipment is required to demonstrate that it will function in accident environments. The prototype equipment used in some of the qualification tests are artificially aged to simulate service degradation. However, there is some doubt that such techniques realistically represent the effects of inservice degradation. For example, it is known that accelerated radiation aging at the high dose rates typically employed by commercial testing laboratories does not produce the degree of embrittlement of cables as may be caused by radiation at the actual dose rate encountered inside containment during operation. Also, with natural aging rather than artificial aging, the polymeric materials used in certain types of solenoid valves have been observed to become more vulnerable to failure under LOCA conditions. Because of the evidence that artificial or accelerated aging techniques may be inadequate, it is difficult to assess the increased degree of vulnerability of safety equipment at this time. This equipment, degraded by age-related service and wear, may be vulnerable to common mode failure during accidents and transients that involve abnormal stresses and demands on the equipment.

2.3 Technical Objectives of the Research

The NRC has the responsibility for ensuring that licensed reactors can continue to be operated safely during their initial licensed lifetime and

during any period of extended life operation. Because of the complexity of age-related degradation and the diversity of the degradation processes, a coordinated research program is necessary to: (a) identify the measures that are available to manage age-related degradation, and (b) identify anticipated problems that may result from plant aging. Using these two general criteria as guidance, a set of nuclear plant aging and life extension technical safety issues has been developed for the NPAR Program, and these are listed below:

- What structures, systems, and components are susceptible to aging effects that could adversely affect public health and safety? Which of these structures, systems, and components are maintained and are replaceable?

- What are the degradation processes of materials, components, and structures that could, if unchecked (improperly maintained and/or not replaced), affect safety during normal design life and during extended life?

- How can operational readiness of aged structures, systems, and components be ensured during 40-year design life and during extended life?

- Are currently available examinations and test methods adequate to identify all relevant aging mechanisms before safety is affected? If not, what efforts are under way to improve them?

- What criteria are required to evaluate residual life of components and structures? What supporting evidence (data, analyses, inspections, etc.) will be needed?

- How should structures, systems, and components be selected for comprehensive aging assessments and residual life evaluations? Which structures, systems, and components should be selected?

- How effective are current programs for mitigating aging (e.g., maintenance, replacement, and repair)?

- What kinds of reliability assurance and maintenance programs will be needed to ensure operational readiness of aged safety systems and components?

- What additional changes will be needed in codes and standards to address aging? What schedule should be followed?

These safety issues form the basis for establishing the technical objectives of the NPAR Program.

The technical objectives of the NPAR Program are:

- Identify and characterize aging effects that, if unchecked, could cause degradation of structures, components, and systems and thereby impair plant safety.

- Identify methods of inspection, surveillance, and monitoring; and evaluate residual life of components, systems, and civil structures, which will ensure timely detection of significant aging effects before loss of safety function.

- Evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging.

The aging research program has been developed to meet these objectives. The program involves: (a) risk-oriented identification and selection of components, systems, or structures for which assessments of the impact of aging on safety performance are to be conducted; (b) review of design base safety margins, qualification testing, operating experience, and methods for surveillance, inspection, monitoring, and maintenance, leading to the development of recommendations for indepth engineering studies; (c) engineering studies, including verification of inspection, surveillance, monitoring, and maintenance methods, evaluation of residual life models, in situ examinations, collection of data from operating equipment, and cost/benefit analyses.

The program developed to meet the above objectives includes a variety of projects. Because of the multidisciplinary scope of the research and the need to make the best use of the available resources, the research effort is focused on key components and structures in the systems of risk significance. The priority of the research effort has been established by taking into account: (a) information gained from the 3-day workshop that was attended by over 300 people from the U.S. and other countries representing a wide spectrum of interests and expertise (Refs. 7 and 8); (b) information gained from the EPRI/DOE workshop on plant life extension held in 1986 in Alexandria, Virginia, in which results were presented from the pilot projects at Surry 1 and Monticello (Ref. 3); (c) insights gained from the risk assessments completed to date (Refs. 9 and 10); (d) advice from a cross section of knowledgeable people; and (e) plant operating experience, including LERs and Institute of Nuclear Power Operation's (INPO's) Nuclear Plant Reliability Data System (Refs. 5 and 11). Other ongoing NRC programs, industry-sponsored research, and programs being conducted in foreign countries are also considered in developing the program plan. In those cases where relevant information is available or is being developed, the NPAR Program has been planned to avoid duplication of effort.

3. UTILIZATION OF RESEARCH RESULTS

The NPAR Program's goals are to obtain a better understanding of the aging and degradation processes in components and structures and provide improved confidence in available methods for detecting and managing aging degradation. This program will provide a basis for timely and sound regulatory decisions regarding continued safe operation of nuclear plants of all ages as well as for the anticipated requests for license renewals. Understanding aging and degradation processes and detecting and managing degradation damage at an early stage, before functional capability is impaired and continued safe operation becomes questionable, will avoid unplanned and costly plant shutdowns. Also, use of the research results will make operating plant maintenance more effective. Wear from excessive testing can be minimized through using more effective surveillance techniques and result in the improved reliability of equipment.

In addition to the general benefits mentioned above, the NPAR Program is structured to respond to the following specific user-oriented needs:

- Develop data for identifying and resolving technical safety issues related to plant aging and license renewal.
- Support NRR/RES in resolving generic safety issues involving aged plant safety systems, support systems, and electrical and mechanical components.
- Evaluate and recommend surveillance and maintenance methods needed to monitor age-related degradation and to support license renewal.
- Develop technical data and provide recommendations useful for developing plant performance indicators (useful to AEOD and NRR for plant inspections and for review of applications for extended license requests).
- Provide information for developing inservice inspection procedures suitable for aged components, systems, and structures.
- Develop recommendations for revising appropriate industry codes and standards.
- Develop technical data useful to RES for the Operational Safety Reliability Research (OSRR) Program and for NRR to evaluate the status of "mothballed" equipment.

The following sections contain a brief description of the NRC staff-defined needs and of how the results of the NPAR Program can be utilized in the regulatory process.

3.1 License Renewal

The NRC needs to clearly define its policy and regulatory positions in the near future so that utility planning for plant life extension can proceed in an orderly manner.

The extension of the period for a nuclear power plant license is provided for in Section 50.51 of Part 50 of the Code of Federal Regulations, which states that a license is issued for a fixed period of time, not to exceed 40 years from the date of issuance. It also states that "Licenses may be renewed by the Commission upon the expiration of the period." Although specific requirements for a license renewal are not defined, it is clear that the "aged" condition of the plant will have to be considered in any utility request for a license renewal. A pressing need for the NRC at the present time is to develop guidance for industry on regulatory policy, rules, and procedures for life extension. By 1992, at the latest, utilities will need to have NRC policy, rules, guidance, and procedures in hand in order to prepare license renewal applications by 1993.

The NPAR activities include: the review of Sections 3 through 10 of the Standard Review Plan (SRP) and associated guidance to identify technical safety issues to be addressed for license renewal; residual lifetime evaluations of major components and structures likely to be considered for life extension; and the review of appropriate technical specifications of methods for early detection and control of aging degradation. The NPAR Program will ensure that this aging/license renewal perspective is factored into its ongoing programs and activities.

The NPAR Program is developing and integrating the vast amount of aging-related data so that the technical safety issues related to license renewal are identified and resolved in an effective and timely manner. Program coordination and technical integration are important elements of the NPAR program. This integration will be accomplished by maintaining, evaluating, and updating the state-of-the-art information obtained from ongoing programs related to aging and license renewal. These programs are sponsored by NRC, industry, and foreign organizations. With this process, the program has prioritized major components and structures that are considered important to evaluate requests for license renewal.

Any additional research projects that may be needed to resolve the technical safety issues in consideration of license renewal will be added to the NPAR Program as they are defined.

3.2 Generic Safety Issues

One of the objectives of the NPAR Program is to support NRR/RES in resolving aging-related generic safety issues identified in NUREG-0933, "A Prioritization of Generic Safety Issues." NUREG-0933 contains a recommended priority list to assist in the timely and efficient resolution of safety issues that have a high potential for reducing risk. The NPAR Program results, which can be used in resolving several of these generic safety issues, are listed in Table 3.1. For example, the NPAR Program can

TABLE 3.1 Generic safety issues, with elements of aging, benefiting from NPAR program results.

Issue Number	Title
23	Reactor Coolant Pump Seal Failures
29	Bolting Degradation or Failures in Nuclear Power Plant
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems
55	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand
70	PORV and Block Valve Reliability
84	CE PORVs
93	Steam Binding of Auxiliary Feedwater Pumps
107	Generic Implications of Main Transformer Failures
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers
115	Enhancement of the Reliability of Westinghouse Solid State Protection System
118	Tendon Anchorage Failure
120	On-Line Testability of Protection Systems
124	Auxiliary Feedwater System Reliability
125.I.6	Valve Torque Limit and Bypass Switch Settings
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems
127	Testing and Maintenance of Manual Valves in Safety-Related Systems
128	Electrical Power Reliability
130	Essential Service Water Pump Failures
132	RHR Pumps Inside Containment
A-17	Systems Interaction

TABLE 3.1 (continued)

Issue Number	Title
A-44	Station Blackout
A-45	Shutdown Decay Heat Removal Requirements
A-47	Safety Implications of Control Systems
B-56	Diesel Reliability
C-9	RHR Heat Exchanger Tube Failures
HF8	Maintenance and Surveillance Program
II.C.4	Reliability Engineering
II.E.6.1	Test Adequacy Study

support the resolution of the Generic Safety Issue B-56, "Diesel Reliability," by evaluating aging and service wear of emergency diesel generators.

The NRR has provided "users-need requests" to RES for resolving some specific issues that are listed in Table 3.1. The NPAR Program is supporting NRR in resolving the Generic Issue II.E.6.1, "Test Adequacy Study," and by assessing methods for monitoring motor-operated valves. The third issue in this category is GI-70, "PORV and Block Valve Reliability."

A residual life assessment task is being performed as part of NPAR to evaluate the age degradation and residual life of major LWR components. The results of this task will indirectly be supportive to NRR in resolving several generic safety issues related to the primary reactor coolant system components. The results of this task will be used in resolving safety issues related to plant life extension and in developing regulatory guidelines and review procedures for use by NRR in reviewing applications for license renewal.

3.3 Maintenance and Surveillance

Maintenance and surveillance programs at nuclear plants are significant contributors to system and plant reliability. The NPAR Program supports the NRR Maintenance and Surveillance Program by evaluating the role of maintenance in managing aging effects. This evaluation consists of:

- Reviewing current practices and procedures, carried out by nuclear utilities, to maintain equipment.
- Reviewing nuclear equipment vendor's recommendations for maintenance of components or subcomponents selected for aging assessments.
- Performing an evaluation, including a comparative analysis, of the relative merits of performing maintenance when a component has been discovered to be malfunctioning (corrective maintenance), and when an observation has been made through surveillance, inspection, or monitoring, that a component may not function when required during a design basis or "trigger" event (preventive maintenance). Emphasis is placed on the relationship between failures (causes or modes) expected to be experienced during operation and those that would potentially occur under the stresses associated with design basis or trigger events.
- Identifying, where possible, those component failure mechanisms likely to be induced through preventive or corrective maintenance. Specifically, look for those failures that might be detectable through short-term, postmaintenance surveillance, inspection, or monitoring.
- Developing recommendations, for acceptable or preferred maintenance practices, based on the preceding activities.

- Evaluating the relative merits of predictive inspection and monitoring methods that can be used to identify imminent failures (predictive maintenance). Predictive maintenance will enable corrective maintenance or replacement to be scheduled based on actual equipment performance. This approach lends itself to use of reliability methods and condition monitoring to mitigate equipment degradation due to aging.

The major emphasis of all the above activities is on the technical aspects of maintenance rather than on institutional, organizational, programmatic, or human factor considerations.

The NPAR Program has been structured to define maintenance and surveillance needs to ensure the operational readiness of aged power plant safety systems and components and provide support to the NRR staff in their review of the requests for license renewal. The NPAR Program also provides the aging-related information for developing maintenance program criteria and standards and maintenance indicators that NRR staff can monitor for specific components and systems.

3.4 Plant Performance Indicators (Involving Aging Considerations)

The operating performance of nuclear power plants, especially in the 10 to 20 years before the end of a plant's operating license, is a significant factor in evaluating requests for plant license renewals. The term "performance indicators" refers to a set of data that may be correlated with individual plant safety performance. Periodic review of the aging trends indicated by the plant performance indicators can aid in evaluating plant performance as they advance in age.

In accordance with the early NRC (IE and now AEOD) study, these indicators may be divided into two categories: direct indicators of current plant performance, i.e., safety system failures; and indirect or programmatic indicators, i.e., an enforcement action index. The NRC staff has selected an optimum set of six indicators on the basis of the deliberations of a task group on performance indicators and discussions with industry representatives. The selected indicators are:

1. Automatic Scrams While Critical,
2. Safety System Actuations,
3. Significant Events,
4. Safety System Failures,
5. Forced Outage Rate, and
6. Equipment Forced Outage per 1000 Critical Hours.

The third indicator, Significant Events, includes degradation of important safety equipment, primary coolant pressure boundary components, and important associated structures. The research results emanating from

the NPAR Program could be used to evaluate the effectiveness of the first four of the above-mentioned plant performance indicators involving elements of aging.

3.5 Inspection

The NPAR Program can potentially support several ongoing NRR programs that guide the regional activities relevant to aging, aging detection, and mitigation of aging degradation. These programs include the Safety System Functional Inspection Program and the Generic Communication Program.

In general, the Safety System Functional Inspection Program assesses whether plant modifications of selected safety systems have degraded the design margin to the point where the system's ability to mitigate design basis events is impaired. This program consists of an indepth review of a small number of safety systems and is usually conducted at older plants.

The objectives of the Generic Communication Program are to:

- Inform licensees of problems, including those due to aging, that have developed in individual plants, and
- Require action when these problems are shown to be significant and generic.

These programs apply to the pressure boundary hardware, drivers, actuators, electrical power, and the instrumentation and controls of engineering safety features.

The NPAR Program will support NRR in establishing inspection procedures that are relevant to aging; NRR includes these procedures in the Inspection Enforcement Manual issued to guide the activities of the regions. For example, some inspection procedures establish guidance for ascertaining that inservice inspection and testing activities are programmed, planned, conducted, recorded, and reported in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

The NPAR Program has the potential to support the ongoing inspection effort conducted by the regional offices in accordance with the NRR inspection program. The objective of this effort is to ensure that systems and components have not been measurably degraded as a result of any cause, including aging. To provide the inspection staff with an up-to-date knowledge of NPAR research results, the results from the program will be summarized at the conclusion of each phase and a briefing will be given for the regional and NRR inspection staff.

3.6 Codes and Standards

Codes and standards help define the inservice inspection requirements to ensure operational integrity of selected power plant electrical and mechanical components. The NPAR Program will develop recommendations to revise relevant ASME and IEEE Codes and Standards to ensure safe operation

with aged components and systems. These recommendations will be developed through active participation in the relevant technical committees.

The Special Working Group on Life Extension--ASME Section XI--is coordinating activities related to codes and standards of interest to the NPAR Program. A special IEEE working group was established to investigate the codes and standards aspects of plant life extension as it may be affected by instrumentation and electrical control equipment. The components and systems currently of interest and being considered in ASME standards are listed in Table 3.2. Some of the relevant IEEE standards are listed in Table 3.3. Periodic briefings and information exchanges with appropriate codes and standards committees are scheduled as part of NPAR.

3.7 NPAR Interfaces with Other Programs

A number of additional NRR/RES programs and activities that have potential to utilize the NPAR Program results are:

- Equipment Qualification
- Reliability Technology
  - Frantic III
  - PETS
  - PRISM
  - NUREG-1150.
- Evaluation of Mothballed Plants
- Innovative Materials and LWR Designs

3.7.1 Equipment Qualification

The NPAR Program results support implementing of Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," of 10 CFR Part 50, which includes the requirement:

> "Equipment qualified by test must be preconditioned by natural or artificial (accelerated) aging to its end-of-installed life condition. Consideration must be given to all significant types of degradation which can have an effect on the functional capability of the equipment. If preconditioning to an end-of-installed life condition is not practicable, the equipment must be replaced or refurbished at the end of this designated life unless ongoing qualification demonstrates that the item has additional life."

TABLE 3.2. Develop recommendations to revise ASME standards for operation and maintenance of mechanical equipment.

Standard Number	Title
ASME OM-1	Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices.
ASME OM-2	Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water Systems.
ASME OM-4	Examination and Performance of Nuclear Power Plant Dynamic Restraints (Snubbers).
ASME OM-5	Inservice Monitoring of Core Support Barrel Axial Preloads in PWRs.
ASME OM-6	Requirements for Performance Testing of Pumps in Light Water Cooled Nuclear Power Plants.
ASME OM-8	Requirements for Preoperational and Periodic Performance Testing of Motor-Operated Valve Assemblies.
ASME OM-10	Requirements for Inservice Testing of Valves in Light Water Cooled Nuclear Power Plants.
ASME OM-13	Requirements for Periodic Testing and Monitoring of Power-Operated Relief Valve Assemblies.
ASME OM-14	Requirements for Vibration Monitoring of Rotating Equipment.
ASME OM-15	Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems.
ASME OM-16	Inservice Performance Testing of Nuclear Power Plant Diesel Drives.
ASME OM-19	Startup and Periodic Testing of Electro-Pneumatic-Operated Valve Assemblies Used in Nuclear Power Plants.

TABLE 3.3. Develop recommendations to revise IEEE standards for electrical equipment for nuclear power plants.

Standard Number	Title
IEEE 308	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.
IEEE 317	Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.
IEEE 323	Qualifying Class 1E Equipment for Nuclear Power Generating Stations.
IEEE 334	Standard for Type Test of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations.
IEEE 336	Installation, Inspection, and Testing Requirements for Class 1E Instrumentation and Equipment at Nuclear Power Generating Stations.
IEEE 344	Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations.
IEEE 382	Standard for Qualification of Safety-Related Valve Actuations.
IEEE 383	Standard Types Tests of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Station.
IEEE 387	Criteria for Diesel Generator Units Applied as Standby Power Supply for Nuclear Power Generating Stations.
IEEE 501	Seismic Testing of Relays for Nuclear Power Generating Stations.
IEEE 535	Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations.
IEEE 572	Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations.
IEEE 549	Qualify Class 1E Motor Control Centers for Nuclear Power Generating Stations.
IEEE 650	Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations.

TABLE 3.3. (continued)

Standard Number	Title
IEEE 944	Application and Testing of Uninterruptible Power Supplies for Nuclear Power Generating Stations.
IEEE/ANSI C37.13-1981	Low Voltage ac Power Circuit Breakers Used in Enclosures.

The evaluation of actual aging processes through the research program provides a basis for assessing the adequacy of industry methods for preconditioning before qualification testing or may lead to recommendations for surveillance or monitoring. This may involve recommendations for revisions of the IEEE standards related to environmental qualification through participation of researchers in the aging research program and in the relevant IEEE standards committees and through the development of industry consensus based on the results of the research. Some of the relevant IEEE standards are listed in Table 3.3.

3.7.2 Reliability Technology

The reliability program has developed a framework and process that can be applied to maintain LWR safety.

The hardware-oriented NPAR Program has the potential to support the major elements of the NRC's Reliability Research and Technology Program. The NPAR Program is evaluating causes of component and structural aging at nuclear power plants, the safety and risk implication of this aging, and methods for detecting and controlling significant aging effects. As part of its efforts, the NPAR Program is collecting failure-rate data on aging and developing quantitative techniques that can be used to quantify the risk and reliability effects of aging, using probabilistic risk assessment (PRA) event tree and system models. These results from the NPAR Program can assist the Reliability Research and Technology Program to: monitor plant/equipment performance; compare plant and equipment performance to acceptable or desired levels to help early detection of degradation; help identify causes of important problems; and help evaluate corrective action and verify effectiveness through performance monitoring.

3.7.2.1 FRANTIC III. The computer code FRANTIC III, used for time-dependent reliability and risk evaluations, was developed by NRC and is particularly useful in technical specification evaluations. Technical data generated in the NPAR Program will be used in developing and evaluating time-dependent models and in determining risk significance of aging effects.

3.7.2.2 PETS - Probabilistic Evaluation of Technical Specification Program. The PETS Program is developing methods for using reliability and risk analyses to improve technical specifications. The development is focused on approaches for modifying allowed outage periods and surveillance test intervals, on a plant-specific or generic base. The aim is for PETS to be available on software for personal computers for NRC and industry use.

For specific components and systems to be studied in the NPAR Program, surveillance and monitoring methods will be recommended to alleviate aging concerns. The recommendations will include identifying component performance parameters and functional indicators and optimize surveillance intervals. Therefore, for these specific components and systems, the PETS Program could use NPAR results.

3.7.2.3 PRISM - Plant Risk Status Information Management System. The PRISM system is a computer software package written for an IBM-XT personal

computer, to provide plant-specific tools for plant inspectors and others whose jobs require little or no PRA background. It is a decision-oriented, user-friendly, menu-driven program that contains data base management and interactive routines to aid NRC inspectors in allocating their efforts toward those areas that have greatest impact on plant safety. Again, for specific components and systems addressed in the NPAR Program, the PRISM project could benefit from NPAR results.

3.7.2.4 NUREG-1150 - Reactor Risk Reference Document. The NPAR Program has potential to support the data base input for the development of NUREG-1150 to account for aging and time-dependent failures. NUREG-1150 provides the results of major risk analyses for six different U.S. LWRs, using the state-of-the-art methods. It is intended that this document provide a data base and insights to be used in a number of regulatory applications including: (a) licensing--evaluating the risk relevance of proposed plant licensing changes, risk effectiveness of existing regulations, and risk priorities of generic technical issues; (b) inspection--developing the methodology and data to set priorities for inspection activities; and (c) research--establishing programs that directly address the analytical and experimental uncertainties identified in NUREG-1150. The draft report for comment of NUREG-1150 was issued in February 1987.

The technical data integration with the aforementioned research projects will be recommended to the ALEXCC for implementation.

3.7.3 Evaluation of Long-Term Outages and Mothballing of Plants

The NPAR Program results could support NRR in developing criteria for evaluating plans involving: long-term outages of operating plants, prolonged delays in plants under construction reaching operational status, mothballing plants during construction, and reactivating laid up or mothballed equipment. The NPAR Program is integrating information related to the degradation processes active in the nuclear power plant structures having safety and risk implications. Some of this information will be useful in evaluating the integrity of mothballed equipment or equipment inactivated during a long-term outage. For example, the NPAR Program is integrating information on microbiologically influenced corrosion (MIC) that is active in mothballed or inactivated equipment filled with liquid. Such degradation was discovered in the stainless steel service water system of H. B. Robinson 2. It occurred after an extended outage to replace the lower assemblies of the steam generators. Use of suitable lay-up procedures could have mitigated the degradation from MIC.

3.7.4 Innovative Materials and LWR Designs

The NPAR Program is identifying and evaluating materials that are susceptible to aging and the critical degradation sites and mechanisms active in the components and structures that are critical to plant safety. These results from the NPAR Program should be considered in the design of the advanced LWRs to ensure higher safety margins. Specifically, components- and systems-specific inspections, surveillance, and condition monitoring methods identified in the NPAR Program could be incorporated as

a built-in diagnostic system in the advanced LWR designs. This design feature would assist in establishing early baseline data and trending of performance parameters and functional indicators.

3.8 Brief Synopsis of NPAR Results in Support of the Regulatory Process

The support to be given to NRC staff defined needs by the various NPAR activities is shown in Table 3.4. This table summarizes in matrix form how each of the ongoing and planned NPAR activities support (and has potential to support) the various user-oriented needs. These activities are discussed in detail in Appendix C and the milestones and schedules for the various activities are given in Section 7. Results in support of the regulatory process have already been obtained in several NPAR-sponsored research activities. A brief description of these results is provided as follows.

- Issued NUREG/CR-4302. Operating experience review and analysis were completed to determine failure modes and causes due to aging of check valves in plant safety systems. This research supports NRR in the resolution of Generic Safety Issue II.E.6, "In Situ Testing of Valves." The ASME Operation and Maintenance (O&M) Committee has been made aware of the results of the study. This NPAR effort related to check valves has been referred to the NSSS Owners Groups' representatives regarding industry actions in response to the check valve and water hammer event at San Onofre, Unit 1.

- Issued NUREG/CR-4597. A study was completed to characterize aging of Auxiliary Feedwater Pumps (AFWP) and evaluate inspection and degradation monitoring methods. Potential failures of the AFWP have been attributed to the presence of large hydraulic forces, particularly at low flow rates, which are substantially different from the best efficiency flow. Methods for detecting failure modes and differentiating between failure causes were defined. The research will support the upgrading of Regulatory Guide 1.147 and Inservice Inspection Code Case Acceptability--ASME Section XI, Division 1.

- Issued Draft NUREG/CR-4590. The evaluation of operational experience and expert opinion indicated that the aging of nuclear service emergency diesel generators is observable; follows recognizable patterns; shows changes in the modes of aging degradation with time; is confined to few, relatively major components; increases as percentage of all failures with time; and is caused by normal operational stressors. The primary causes of diesel generator aging are vibration, adverse environment, and human errors. The results of this research have been conveyed to NRR and they can be used in resolving Generic Safety Issue B-56, "Diesel Reliability," and to upgrade ASME Sections III and XI pertaining to diesel generators.

TABLE 3.4. Potential use of NPAR results, involving aging consideration, for components, systems, and structures.

NPAR Research Activities	License Renewal	Generic Safety Issues	Maintenance and Surveillance	Plant Performance Indicators	Inspection	Codes and Standards
Motor-operated valves	*	✓	✓	✓	✓	✓
Check valves	*	✓	✓	✓	✓	✓
Solenoid-operated valves	*		✓		✓	
Auxiliary feedwater pumps	*	✓	✓	✓	✓	✓
Electric motors	*		✓		✓	✓
Chargers/inverters	*		✓		✓	✓
Batteries	*		✓		✓	✓
Power-operated relief valves	*	✓	✓		✓	
Snubbers	*	✓	✓		✓	✓
Circuit breakers	*	✓	✓		✓	✓
Penetrations/connectors/cables	✓		✓		✓	✓
Diesel generators	✓	✓	✓	✓	✓	✓
Transformers	✓	✓	✓		✓	✓
Heat Exchangers	*	✓	✓	✓	✓	
Compressors	*		✓		✓	
Bistables/switches	*		✓		✓	
High pressure ECCS	*		✓	✓	✓	✓
RHR/low pressure ECCS	*		✓	✓	✓	✓
Service water system	*	✓	✓	✓	✓	✓
Component cooling water system	*		✓	✓	✓	✓
Reactor protection system	*	✓	✓	✓	✓	✓
Class 1-E distribution system	*	✓	✓	✓	✓	✓
Auxiliary feedwater system	*	✓	✓	✓	✓	✓
Control rod drive system	✓		✓		✓	✓

TABLE 3.4. (continued)

NPAR Research Activities	License Renewal	Generic Safety Issues	Maintenance and Surveillance	Plant Performance Indicators	Inspection	Codes and Standards
Civil structures	✓					
Risk evaluation of significant aging effects	✓	✓	✓	✓	✓	
Residual life assessment	✓		✓		✓	
Shippingport aging evaluation	✓		✓		✓	

* *under review.*

- Issued NUREG/CR-4564. Operating experience review and analyses were completed to determine failure modes and causes, due to the aging of battery-chargers and inverters that are used in plant safety systems. The identified major contributors to failures are fuses and capacitors, and involve overheating and aging/wearout. The results of this research in combination with the output of Phase II studies will be used to provide recommendations to upgrade IEEE Standard 650, Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Plant Generating Stations. The approval of this standard has been temporarily placed on hold to incorporate aging-related conclusions from NUREG-4564 into the standard.

- Issued NUREG/CR-4380. A field test program was carried out to evaluate a technique of valve signature analysis to detect and differentiate abnormalities, including time-dependent degradation (aging), and incorrect adjustments in motor-operated valves. Measurements were made at four operating plants to verify monitoring techniques and to obtain characteristic "signatures" indicative of degradation and misadjustments of motor-operated valves. This research supports NRR in the resolution of Generic Issue II.E.6, "In Situ Testing of Valves." The research results emanating from the NPAR effort were used in Bulletin No. 85-03: Motor-Operated Valve Common-Mode Failures During Plant Transients Due To Improper Switch Settings.

- Issued NUREG/CR-4279. A study was completed to identify aging of hydraulic and mechanical snubbers used on safety-related piping and components of nuclear power plants. The ASME Section XI Code Committee and ANSI/ASME/OM4 Committee have been made aware of the results of the study, and a value-impact analysis reflecting the reduction in the number of snubbers in existing plants is being incorporated in draft Regulatory Guide SC 708-4, Rev. 1, "Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety."

- Issued Draft NUREG/CR-4692. A study was completed for NRR using NPAR data in the resolution of Generic Issue No. 70, "PORV and Block Valve Reliability." The report contains a review of nuclear power plant operating events involving failures of power-operated relief valves (PORVs) and associated block valves (BVs). Aging-related data include failure mode, failure mechanism, and severity. The report also addresses questions such as: (a) how do operator/maintenance actions contribute to valve failures?; (b) are certain designs more prone to failures than others?; and (c) to what extent would upgrading (valves, operators, and control systems of safety-related systems) have prevented the failure?

- Issued NUREG/CR-4234. Review and analysis of operating experience data were accomplished to determine the failure modes and causes (due to aging) for motor-operated valves. This research supports the NRR efforts to resolve Generic

Issue II.E.6, "In Situ Testing of Valves." The ASME Operation and Maintenance (O&M) Committee has been made aware of the results of this study. Also, this NPAR effort has been referred to the NSSS Owners Groups' representatives involved in responding to Bulletin No. 85-03: Motor-Operated Valve Common-Mode Failures During Plant Transients Due to Improper Switch Settings.

4. RESEARCH APPROACH

4.1 Risk and System Oriented Identification of Aging Effects

The research projects in the NPAR Program use the phased approach to research, as shown in Figure 4.1. An initial selection process is followed to establish priorities for detailed aging assessments of specific systems and components. The selection criteria include nuclear plant experience, "user"-defined needs, expert judgment regarding susceptibility of the system or component to aging degradation, and the potential contribution to risk from failure of systems or components.

4.1.1 Operating Experience and Expert Opinion

One of the sources available for failure data of components and systems is the operating experience currently being obtained from commercial LWRs. This information is obtained from the LERs and the Nuclear Plant Reliability Data System (NPRDS). The information is being analyzed to identify systems and components that are susceptible to aging-related failures (Ref. 12). The current effort has followed the initial scoping studies of plant operating experience (Refs. 5 and 11). Also, once the components and systems are selected for an indepth engineering study, data bases are evaluated further to identify failure modes, failure frequencies, failure causes, and methods used to remedy the causes of the failures.

Expert opinion has also been used for selecting systems and components. Expert opinion was used early in the program to establish the priority of the research effort (Refs. 7 and 8). Since that time, recommendations and advice have been sought from experts throughout the industry. Most recently, expert opinion has been used by the TIRGALEX to assess the priorities that will be used in the NPAR Program.

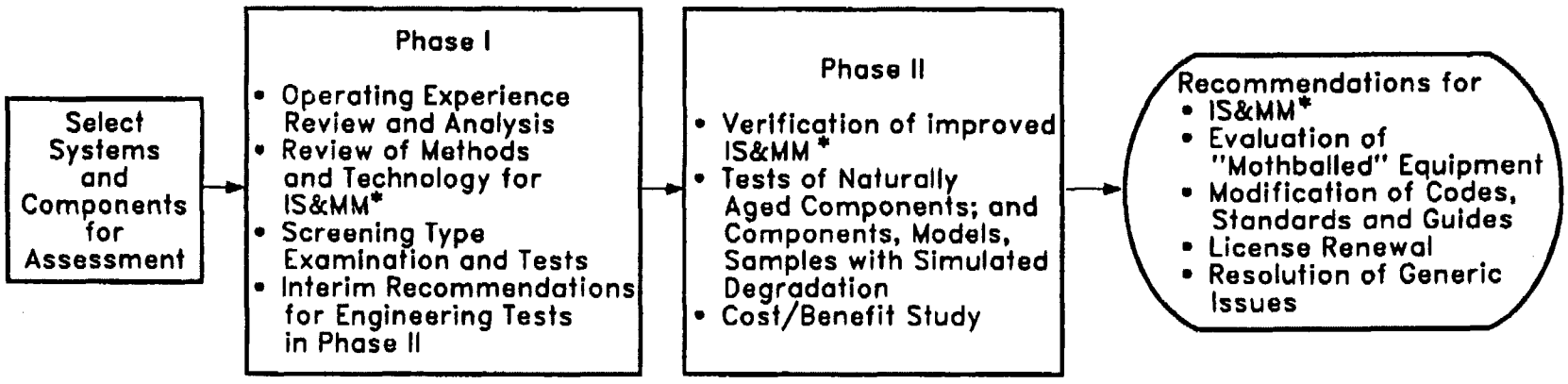
4.1.2 Risk Evaluation

Risk studies have been used to evaluate the potential consequences of component or system failures due to aging degradation. The results of the initial effort are given in References 9 and 10. More recently, work has begun on a time-dependent calculation of risk that can evaluate the effects of age-related failures on system availability and core melt frequency.

4.2 Phased Approach to Aging Assessment and Indepth Engineering Studies

4.2.1 Phase I

The aging assessments of the components or systems selected for evaluation primarily involve two stages. The first stage, Phase I, is based on readily available information from public and private data bases, vendor information, open literature, utility sources, and expert opinions. The products of the Phase I analysis include: an identification of failure



* IS&MM -- Inspection, surveillance and monitoring methods

Figure 4.1 NPAR approach.

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. 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 Phase I, recommendations are developed to identify detailed engineering tests and analyses to be conducted in Phase II. The Phase I evaluation is used to decide if a Phase II assessment is warranted and on occasion may lead to a recommendation of a Phase I assessment of a component or system not yet selected for evaluation.

4.2.2 Phase II

In those cases where Phase II assessments are needed, they generally involve some combination of: (a) tests of naturally aged equipment or equipment with simulated degradation; (b) laboratory or in-plant verification of methods for inspection, monitoring, and surveillance; (c) development of recommendations for inspection or monitoring techniques in lieu of tests that cause excessive wear; (d) verification of methods for evaluating residual service lifetime; (e) identification of effective maintenance practices; (f) in situ examination and data gathering for operating equipment; (g) verification of failure causes using results from in situ and postservice examinations; and (h) cost/benefit analyses.

With the completion of the aging assessment research, a technical basis is available for use in the regulatory process. Examples of the uses include: implementing improved inspection, surveillance, maintenance, and monitoring methods; modifying present codes and standards; developing guidelines and review procedures for plant life extension; and resolving generic safety issues.

A detailed description of the individual steps in the Phase I and Phase II assessments is given in Appendix A.

5. PROGRAM DESCRIPTION

The NPAR Program elements are listed in this section along with a description of their role in plant aging assessment and license renewal. The components, systems, and civil structures that are of current interest were selected on the basis of the reviews of safety-significant items, user needs, operating experience, and expert opinions and may be revised in the future.

5.1 Components, Systems, and Structures Studied in NPAR

The components, systems, and civil structures of current interest have been selected by the risk-oriented identification of aging effects, operating experience, user needs, and expert opinions. The selection processes are discussed in Section 4, and in more detail in Appendix A. The currently selected items are listed in Tables 5.1, 5.2, and 5.3.

Table 5.1 contains a listing of three groups of components that have been identified as having an aging-related impact on plant safety and support systems and their availability and margins. The components in Group 1 were evaluated in the early part of the NPAR Program. Phase I research has been completed for all of the Group 1 components. The Phase II effort has been initiated on motor-operated valves, check valves, auxiliary feedwater pumps, electric motors, batteries, chargers/inverters, snubbers, and solenoid-operated valves. Additional research effort will be defined when, or if, issues are identified during the Phase II efforts.

The components listed in Group 2 are currently included in the NPAR scope of work. Depending upon the availability of funds, Phase I engineering evaluations on these components are scheduled to begin in FY 1988. The Group 3 components have been recommended for aging studies by various sources. Engineering evaluations of the components in Group 3 are not included in the NPAR Program at this time.

Table 5.2 is the list of nuclear plant systems that are of current interest. These systems are considered important for accident prevention or mitigation.

Phase I and Phase II evaluations are now proceeding on the systems in Group 1. The Phase I evaluation of the control rod drive system (Group 2) is planned for FY 1988.

The systems listed in Group 3 have recently been identified as having importance to the evaluation of plant performance. The systems in Group 3 are currently outside the NPAR Program scope, due to limited availability of resources.

The formal study of the residual life of the major LWR plant components and structures listed in Table 5.3 was initiated in FY 1986. These major components and structures are the large, relatively expensive parts of a nuclear power plant that are not routinely or frequently

TABLE 5.1. Components of current interest in the NPAR program.

Group 1

- Motor-Operated Valves
- Check Valves
- Auxiliary Feedwater Pumps
- Electric Motors
- Batteries
- Chargers/Inverters
- Snubbers
- Circuit Breakers and Relays
- Solenoid Valves
- Power-Operated Relief Valves
- Emergency Diesel Generators

Group 2

- Cables (power, control, instrument)
- Electrical Penetrations
- Connectors, Terminal Blocks
- Heat Exchangers
- Compressors
- Transformers
- Bistables/Switches

Group 3

- Fan Chillers
- Purge and Vent Valves
- Safety-Relief Valves
- Service Water and Component Cooling Water Pumps
- Air-Operated Valves
- Main Steam Isolation Valves
- Accumulators
- Surge Arrestors
- Isolation Condensers (BWR)

TABLE 5.2 Systems of current interest in the NPAR program.

Group 1

- High Pressure Emergency Core Cooling System
- Low Pressure Emergency Core Cooling System
- Service Water System
- Component Cooling Water System
- Reactor Protection System
- Residual Heat Removal System/Auxiliary Heat Removal System
- Class 1E Distribution System
- Auxiliary Feedwater System

Group 2

- Control Rod Drive System

Group 3

- Engineered Safety Feature Actuation System
- Recirculation Pump Trip Actuation Instrumentation (BWR)
- Reactor Core Isolation Cooling System
- Standby Liquid Control System (BWR)
- Containment Isolation
- Containment Cooling Systems
- Instrument and Control Air System

TABLE 5.3 Major LWR plant elements of current interest in the NPAR program.

PWR

- Reactor Pressure Vessel*
- Containment and Basemat
- Reactor Coolant Piping and Safe Ends*
- Steam Generator*
- Reactor Coolant Pump Body
- Pressurizer
- Control Rod Drive Mechanism**
- Cables and Connectors**
- Emergency Diesel Generators**
- Accumulator
- Reactor Pressure Vessel Internals
- Reactor Pressure Vessel Support
- Biological Shield
- Pressurizer Line

BWR

- Reactor Pressure Vessel*
- Containment and Basemat
- Recirculation Piping, Safe Ends, Safety System Piping*
- Recirculation Pump Body
- Control Rod Drive Mechanism**
- Cables and Connectors**
- Emergency Diesel Generators**
- Reactor Pressure Vessel Internals
- Reactor Pressure Vessel Support
- Biological Shield

* These LWR primary system components are the subject of indepth engineering studies sponsored by the MEB/DE.

** These LWR components are the subject of indepth engineering studies (as part of NPAR Program) sponsored by the EMEB/DE.

replaced. The major components to be studied were selected on the basis of the safety criterion that the release of fission products that may occur during an accident should be contained within the plant (Ref. 13).

The major components selected for evaluation include the pressure boundary components, containment, and supporting structures. Also included are components related to reactor control systems and reactor safety systems. The reactor internals are included here also, as their failure may prevent control rod insertion or may cause fuel failure. Major components are identified from both PWRs and BWRs and are listed in order of priority in Table 5.3. Detailed evaluations of reactor pressure vessels, reactor coolant piping and safe ends, and steam generators are being performed in the program sponsored by the MEB/DE.

NOTE: It is not the intent of the NPAR Program to do indepth engineering evaluations of: aging and defect characterization, and methods for inspection, surveillance, and monitoring of all significant plant elements. The plant aging research program efforts have to be focused to consider: (a) PWRs from three different NSSS vendors, (b) BWRs, (c) plants with numerous variations in design, applications, and suppliers, and (d) operation and maintenance with differing practices and philosophies. The intent of the NRC-sponsored NPAR effort is to study a few selected electrical and mechanical components and a few representative safety systems and support systems; then, to demonstrate how the NPAR strategy can be applied by the industry to components, systems, or structures of interest. It is the industry's responsibility to characterize and evaluate their own plant systems, components, and structures and ensure their operational safety as the plants advance in age.

5.2 Program Elements

The NPAR Program has been implemented to develop technical data related to plant aging and license renewal. The research being performed in the NPAR Program can be categorized in the following major subjects or technical areas.

5.2.1 Risk Significance of Aging Effects

In this effort, aging-related failures, identified from plant operating data, are evaluated to determine their risk significance to plant safety. This study has provided information needed to select components, systems, and structures for detailed Phase I and Phase II aging evaluations. The program details are described in Appendix C, Section C-3.1.

5.2.2 Aging Assessment of Specific Components and Systems

Aging assessments of components and systems are in progress and consist of the Phase I and Phase II aging evaluations. These evaluations are being made on the components and systems considered vital to plant safety during normal operation, as well as during accident and postaccident

conditions. The technical safety issues that need to be addressed in reviewing applications for license renewal (involving specific components and systems) will be identified and resolved. A listing of the contractors performing aging assessments of components and systems and a description of the various tasks is given in Appendix C.

5.2.3 Aging Assessment of Civil Structures

This study involves an indepth assessment of the aging degradation of the concrete civil structures in nuclear plants. It includes identifying the principal structural safety issues; developing materials properties data for aged civil structures; and evaluating the functional capabilities of aged structures in a postaccident environment. The data generated will be useful in evaluating applications for plant life extension/license renewal. A preliminary scoping study has been completed and an indepth, Phase I aging assessment is planned for FY 1988 (Ref. 14). This task is being addressed by the Structural and Seismic Engineering Branch of the Division of Engineering (SSEB/DE).

5.2.4 Inspection, Surveillance, and Monitoring Methods

The methods used for inspection, surveillance, and monitoring of each of the various components, systems, and structures are reviewed as part of the Phase I and Phase II engineering evaluations. Current industry practices and procedures are assessed and recommendations developed for improved or preferred methods. This review includes identifying performance parameters and functional indicators that can be used for the early identification of age-related degradation for each component, system, or structure.

5.2.5 Role of Maintenance in Managing Aging

Evaluations are made on the role of maintenance in managing aging effects. A review of present practices, in terms of a comparative analysis of corrective versus preventive maintenance, and recommendations for preferred practices are included for each component, system, and structure in this study.

5.2.6 Component Lifetime Evaluation

The current methods for predicting service life of major mechanical components and structures are reviewed and work started on an approach for lifetime evaluation. Consideration is given to the resources required for inspection and monitoring of components and structures and as to whether technically acceptable methods for predicting service life could be substituted.

5.2.7 Special Topics

5.2.7.1 Aging/Seismic Shock Interaction. This is a study to determine the vulnerability of age-degraded components to seismic events. Seismic qualification of electrical equipment already requires consideration of

preaging. However, no requirement currently exists for preaging the mechanical equipment to be qualified. This effort is aimed at assessing how aging degradation will affect the performance of electrical and mechanical equipment during or after a seismic event.

5.2.7.2 Quantification of Aging. This topic consists of the development of a practical approach to the quantification of aging and a residual life assessment of major electrical components. The topic includes an effort to identify the life-limiting processes of each of the major components under study, trending of performance parameters and functional indicators, and determining margins. Methods are then developed for determining major component lifetimes.

5.2.7.3 Decommissioning of the Shippingport Atomic Power Station. This effort involves in situ assessments, acquisition of selected data/records and specimen samples and components from the plant, some of which have completed over 20 years of service. Also, postservice examinations and tests are planned and coordinated with NRC staff and NRC contractors. Program details are described in Appendix C, Section C-1.1.

6. COORDINATION WITH OTHER PROGRAMS, INSTITUTIONS, AND ORGANIZATIONS

Various institutions and industry organizations have performed studies and instituted programs relevant to the aging research. Results of the more important activities are reported in References 1 and 7. Also, there are a number of ongoing programs producing significant results that cannot and should not be duplicated. A major emphasis in the NPAR Program plan is that proper coordination and integration of plant aging research activities are obtained at various levels. This approach will help achieve overall program goals and objectives and ensure the efficient use of available resources.

Ongoing NRC programs related to aging and license renewal are being conducted by Office for Analysis and Evaluation of Operational Data (AEOD), NRR, and RES. The NPAR Program coordination and technical integration with other agency programs are in place or will be implemented through the direction of the coordinating committee, the ALEXCC. In addition to NRC-sponsored research, aging and nuclear plant life extension programs are being sponsored in the U.S. by nuclear industry groups including EPRI, NSSS vendors, utilities, architect engineers, and the DOE. Also, nuclear plant aging and life extension programs are being conducted in a number of foreign countries.

A listing of the more relevant programs, external to the NPAR Program, is given in Table 6.1. Appendix D contains a detailed description of ongoing aging-related programs.

NPAR Program interfaces with ongoing NRC programs have been established and will be maintained. External programs, involving both domestic and foreign organizations, have also been contacted. It is very likely that new aging-related programs will be sponsored in the future by outside organizations. An important and continuing activity in the NPAR Program is identifying new projects and establishing appropriate interfaces.

TABLE 6.1 Selected programs relevant to NPAR aging and life extension programs

RES

- Plant Life Extension Policy Issues--This program has been initiated in FY-87 and will establish a proposed regulatory policy for license renewal.
- Numerous projects addressing generic safety issues and specific structures/systems/components have aging and life extension implications. In most cases, however, aging and life extension are not the dominant considerations in the work.

NRR

- Maintenance and Surveillance Program--Maintenance and surveillance are key mechanisms for controlling aging. Phase I of the program has been completed, and Phase II is under way. The work addresses many other safety and regulatory issues in addition to aging and life extension.
- Safety System Functional Inspection Program--The objective of this program is to ensure that plant modifications have not impaired the effectiveness of selected safety systems. This includes the detection, control, and prevention of selected safety system degradation, whatever the cause; aging-related degradation is covered along with other casual factors not related to aging.
- Safety System Outage Modifications Inspection Program--The objective is to verify that modifications and repairs are conducted properly so that safety is not compromised. This covers modifications and repairs that may be aimed at managing aging or permitting life extension.

AEOD

- NPRDS Analyses--AEOD analyzes data for the Nuclear Plant Reliability Data System (NPRDS), producing statistical and engineering evaluations of component failure modes, time to failure, operating conditions that affect failure, and chemical and physical conditions affecting component-wearout rates.
- LER Database and Analyses--AEOD has developed the Sequence Coding and Search System (SCSS), which can be used in studies of component life and component aging. SCSS is a comprehensive, computerized LER-based system with features that are highly useful in aging-related studies, e.g., cause codes specifically identifying aging degradation.

EPRI/DOE/INDUSTRY

- EPRI/DOE/Industry Cooperative Program--This cooperative program is addressing the full spectrum of policy, safety, regulatory, economic, and technical issues associated with life extension. The first phase is nearing completion.

CODES AND STANDARDS COMMITTEES

- ASME Board on Nuclear Codes and Standards--A special coordinating committee is being established with representation of IEEE, ASME, ACI, etc. Its purpose is to direct the thrust of life extension code activities.
- ASME Section XI--A Special Working Group on Plant Life Extension has met seven times. They have active participation from EPRI, NRC, DOE, NSSS suppliers, and utilities. ASME XI is just beginning to develop changes to the code based on life extension considerations.
- IEEE Working Group 3.4, under the IEEE Nuclear Power Engineering Committee, is developing recommendations for changes to IEEE standards based on aging and life extension considerations.
- IEEE Working Group 3.3, under the IEEE Nuclear Power Engineering Committee, is developing recommended methods for the industry to mitigate equipment aging and to preserve equipment operability.

FOREIGN COUNTRIES

- Canada, France, Italy, West Germany, and Japan have programs that address aging and life extension.
- Numerous programs address specific components and aging mechanisms. For example, Westinghouse and Framatome have programs that address aging of cast stainless steel.

7. SCHEDULES AND RESOURCE REQUIREMENTS

The currently estimated schedules and milestones for completing specific research activities for aging and license renewal are provided in this section. These general schedules, and particularly schedules for evaluating specific components, systems, and structures, depend on funding, assignment of priority, and degree of coordination and participation by other institutions and organizations. The NRC/RES staff and its contractors are actively pursuing participation from domestic and foreign institutions and organizations. The active interest in requirements for license renewal (plant life extensions) should facilitate industry cooperation and active participation in aging research.

Two critical program requirements influencing the schedules and resource requirements of the NPAR Program are:

1. The timely availability of data for naturally aged equipment from operating power plant facilities.

   In situ aging assessments and trending of component and system performance are necessary to aging characterization and detection of defects. Also, postservice examination and testing of naturally aged equipment are essential to relate artificial aging (preaging/accelerated aging) to normal aging and to ensure operational readiness of aged equipment. Evaluation and analysis of naturally aged equipment is intended to generate recommendations for criteria and guidelines for decisions. Cooperative programs among the industry and research organizations should be carried out to facilitate the availability of naturally aged equipment for aging research. Schedules and resource requirements will need adjustments to reflect the extent of these cooperative programs.

2. Identification and resolution of the technical safety issues supporting NRC's definition of policy and regulatory position for license renewal.

The schedule and resource requirements reflect the need for early identification and prioritization of the technical safety issues based on their risk significance. Other major elements affecting these requirements are the availability of topical reports from industry programs, the work scope required to resolve essential safety issues, and the development of proposed regulatory policy review procedures.

The current NPAR schedule has been adjusted to support the regulatory activities now anticipated for nuclear plant life extension. NRC currently plans to have a policy in place by the early 1990s. To support the schedule for the NRC renewal policy, the research activities related to plant life extensions are currently scheduled to be completed by the early 1990s.

The schedules for the effort on major elements of the NPAR are given in Figures 7.1 to 7.4. As an example of a detailed schedule with milestones, the research plan for the assessment of the residual life of major reactor components is shown in Figure 7.4. The scheduling of the Phase I and Phase II assessments for specific components and systems are shown for each of the major activities. Also shown is the schedule for utilizing the research effort. An additional activity, indicated by the dotted lines, is to tentatively plan for tasks that may be needed for resolving issues that may be raised during the results utilization efforts. The vertical dotted line in each figure indicates the expected issue date for this revised plan. The schedules and major elements are based on the current NPAR research priorities. It must be recognized that the activities and schedules can change as information is developed in the new program and as additional inputs are provided and program needs are identified.

It should be emphasized that the number of systems and components and the degree and depth of assessments and analyses that can be carried out effectively will depend upon the availability of funds and the period of time over which the results are required. The timely availability of naturally aged equipment from operating and decommissioned facilities and the opportunity for in situ assessments will determine, in a significant manner, the resource requirements and the completion schedule for the various activities.

NRC funding available to the program for FY 1987 is $5.2 million. The FY 1988 funding level has not been firmly established. In addition, NRC staff participation at an average level of five full-time professionals per year (FTE) will be needed for the duration of the program. When participation by outside organizations is achieved, the resources provided from outside NRC will be identified in future revisions of this plan.

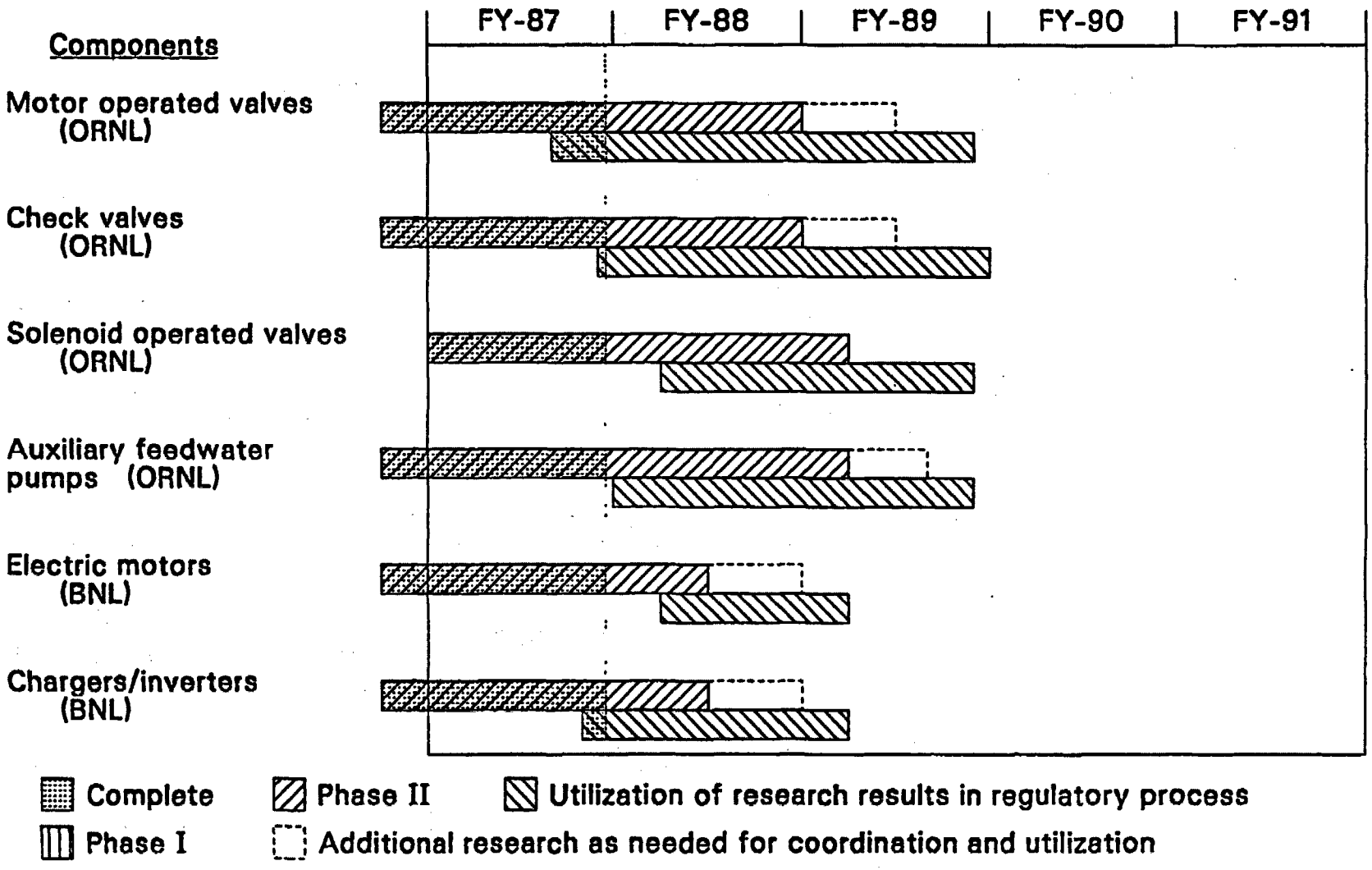
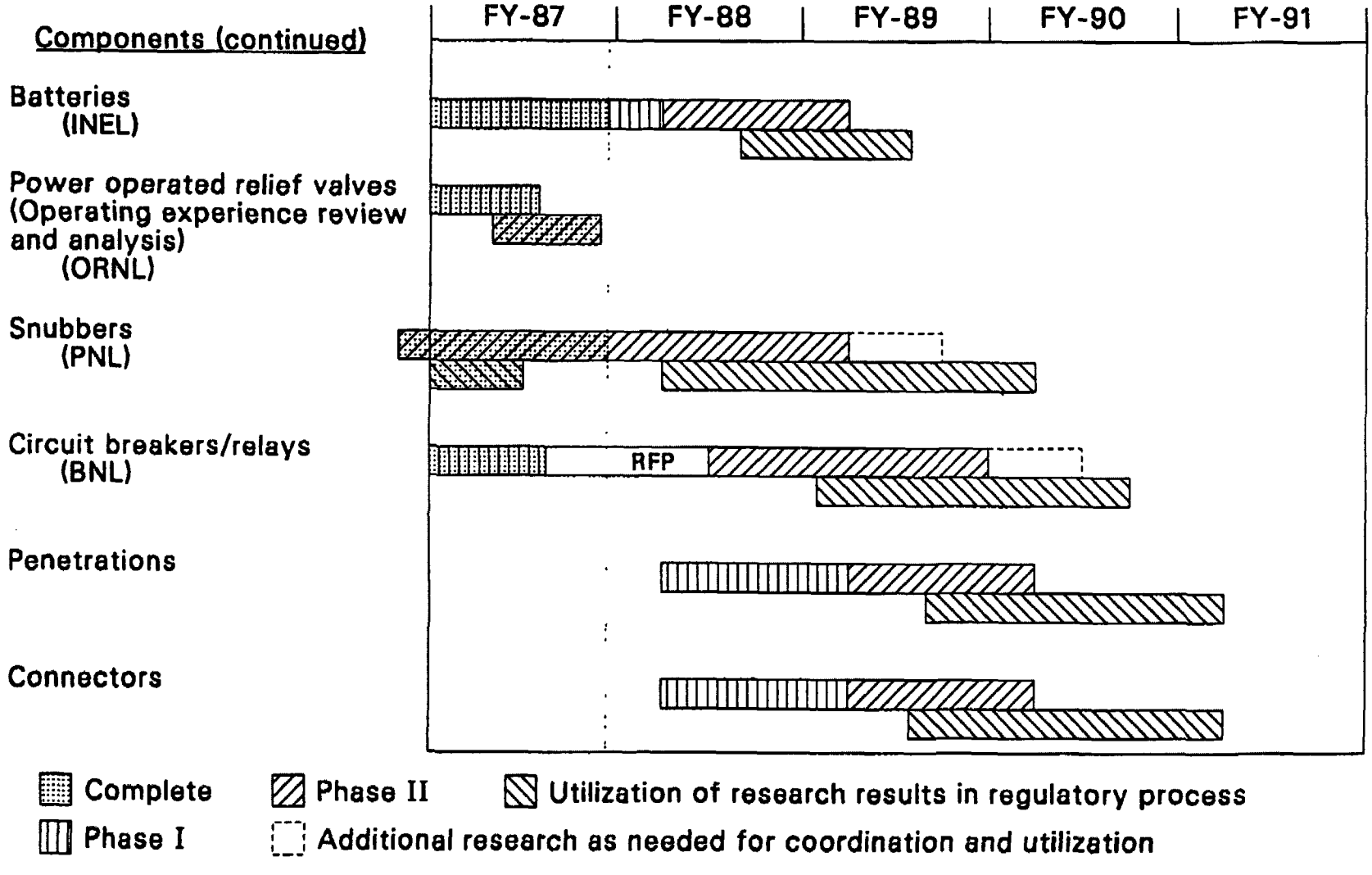


Figure 7.1. NPAR milestones and schedules--components.



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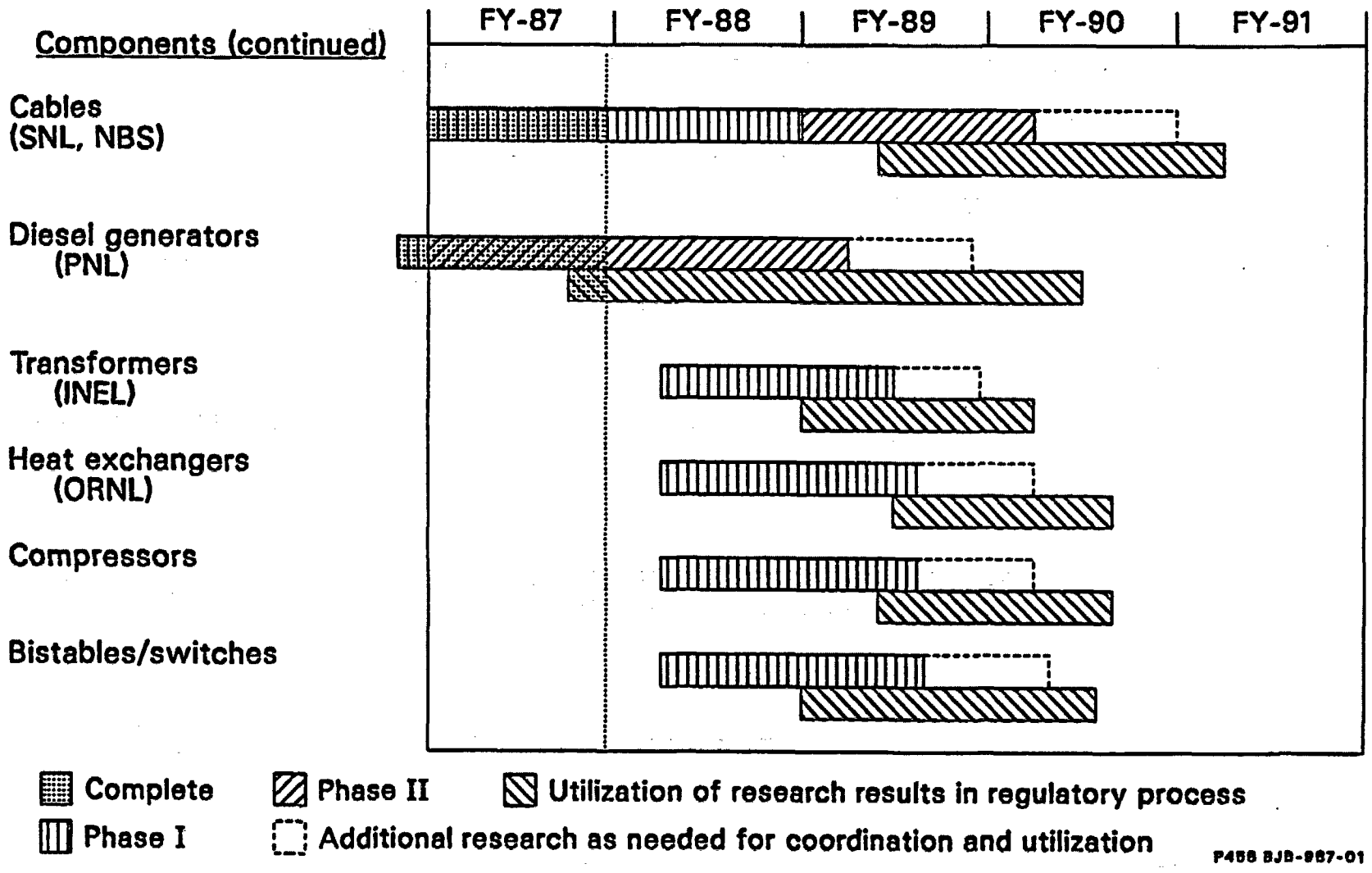

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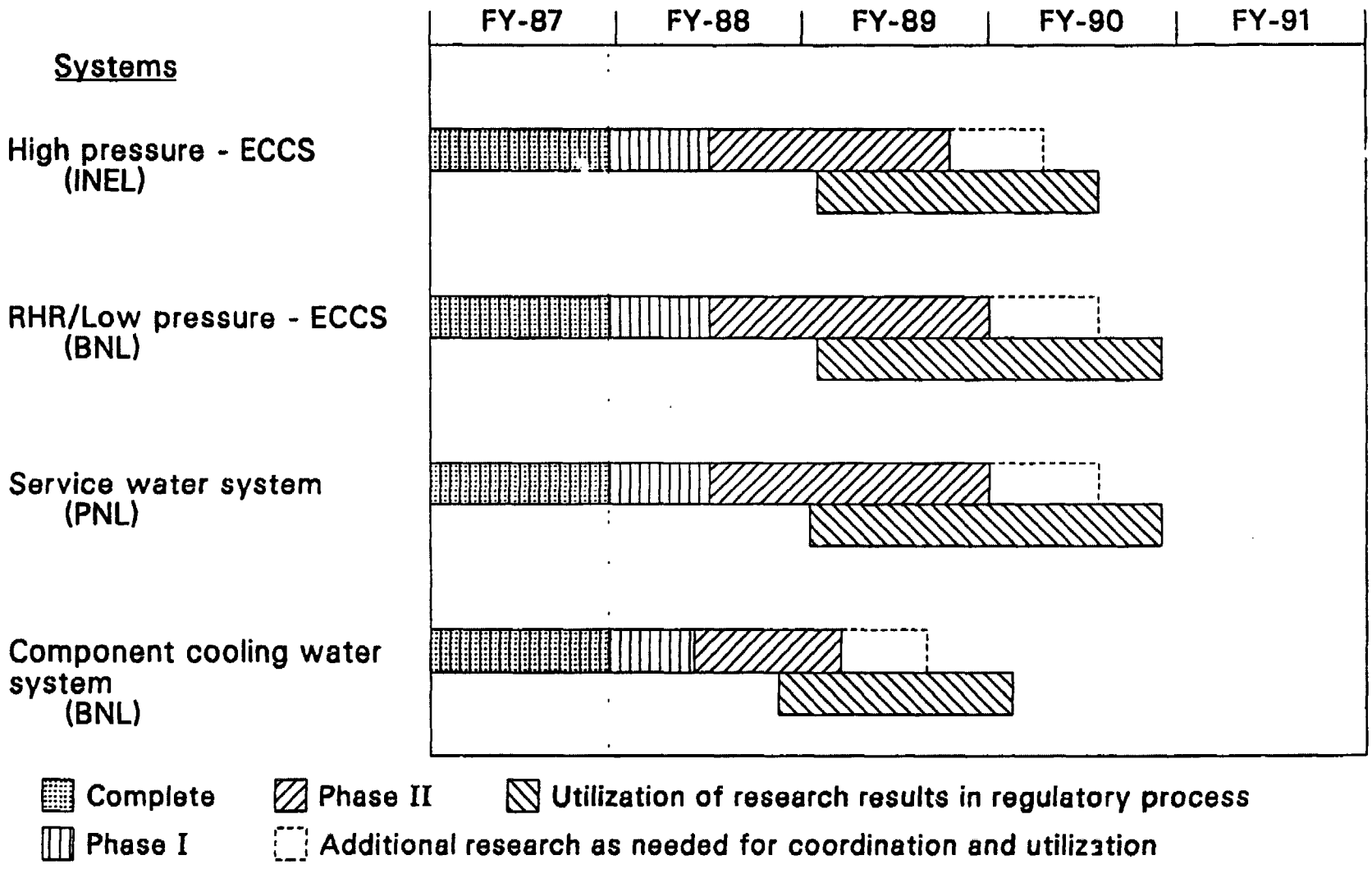
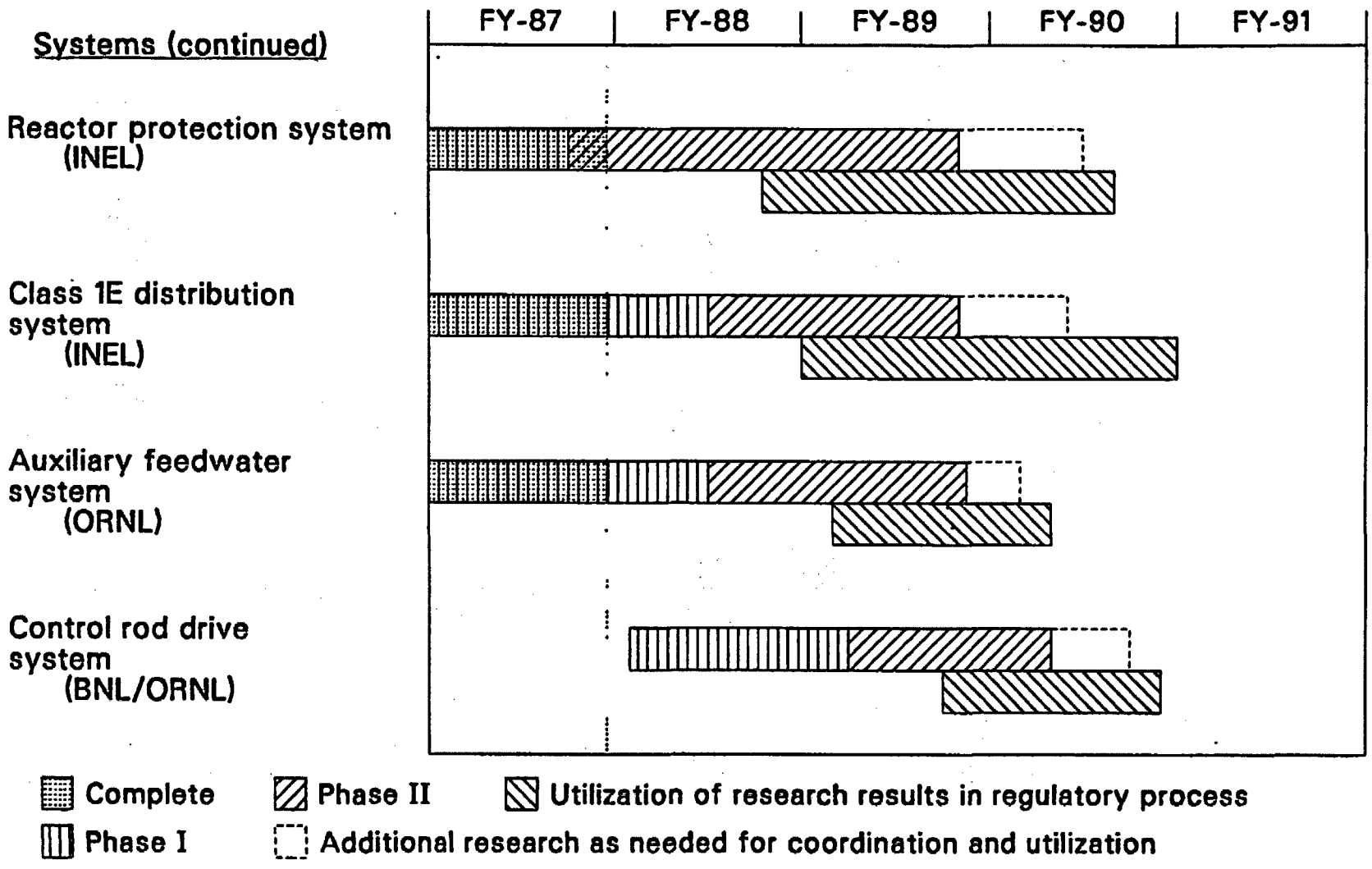


Figure 7.2. NPAR milestones and schedules--systems.



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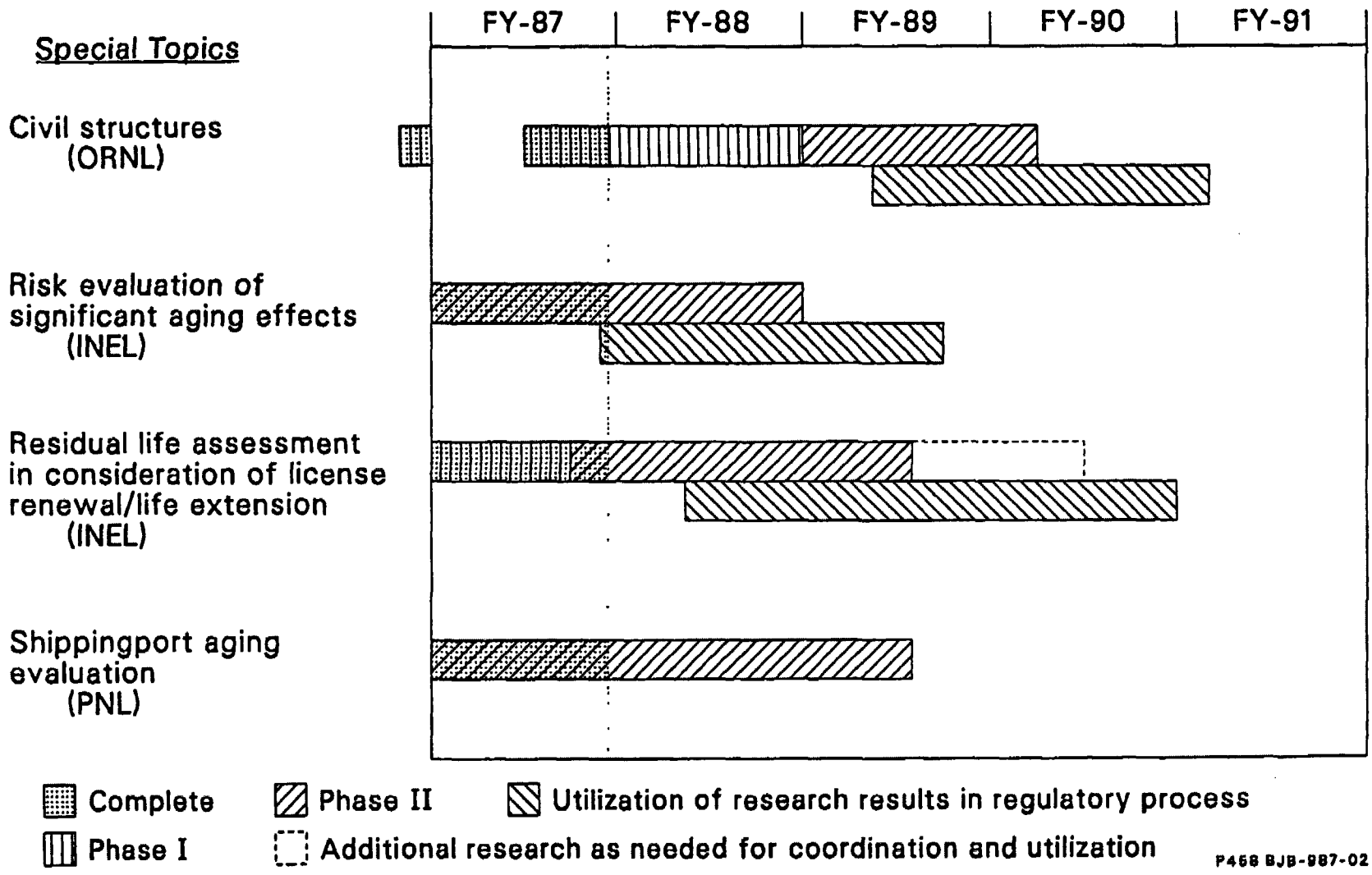

Figure 7.3. NPAR milestones and schedules--special topics.

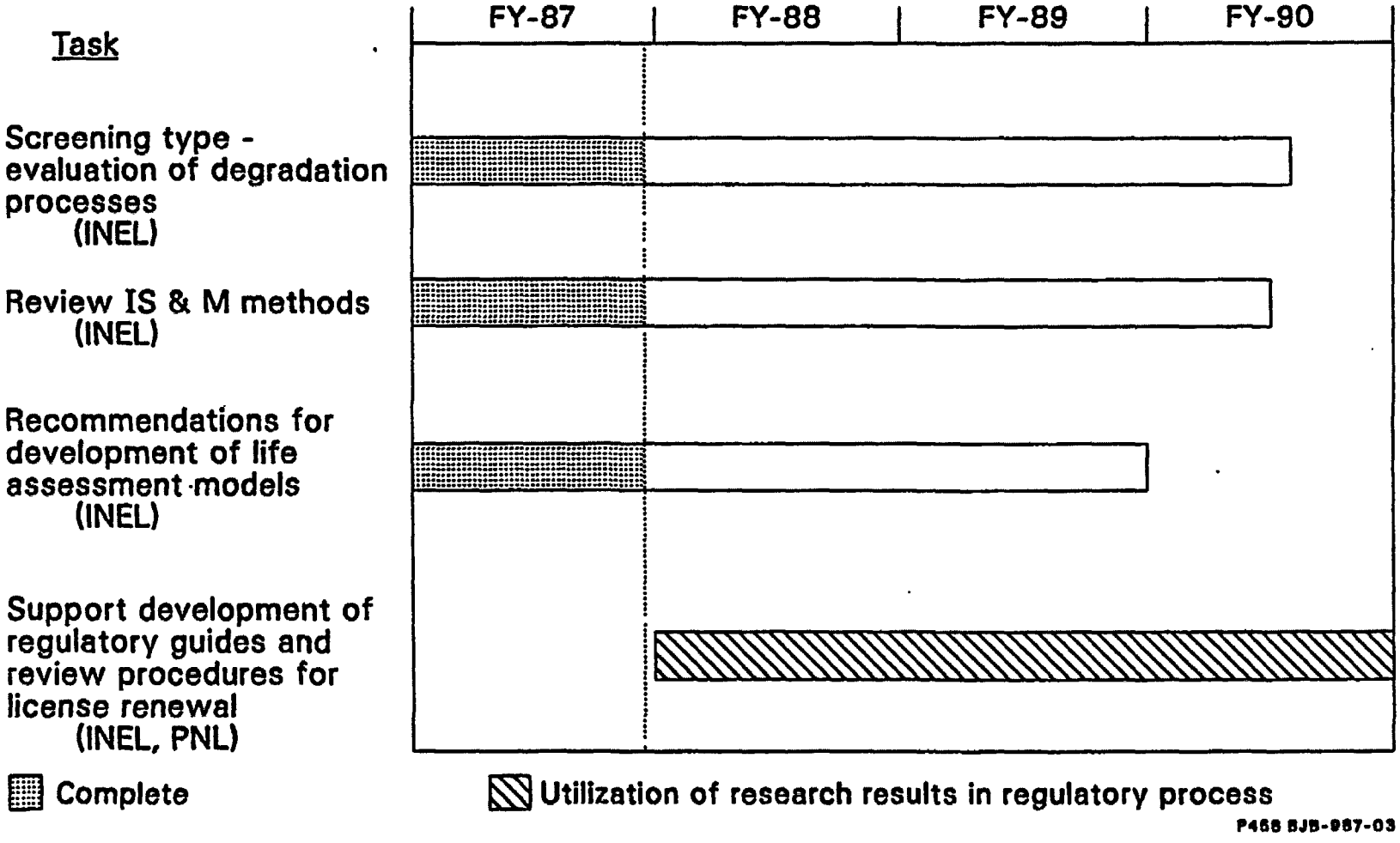

Figure 7.4. NPAR milestones and schedules--residual life assessment of major components.

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APPENDIX A
NPAR PROGRAM STRATEGY

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FIGURE

APPENDIX A
NPAR PROGRAM STRATEGY

The NRC aging research program is carried out in discrete stages as shown in Figure A-1. The phased approach shown here is applied to all components and systems in the NPAR Program. The use of this structured approach is needed in the NPAR Program because of the wide variety of types of systems and equipment that are analyzed, the involvement of a large number of research teams (national laboratories and private sector contractors), and the multidisciplinary nature of the research projects.

The NPAR Program basically employs a two-phase approach to perform detailed aging assessment of components, systems, and structures. In Phase I, an interim aging assessment is performed by reviewing operating data, expert opinion, and industry practices. Where warranted, this is followed by an indepth Phase II comprehensive aging assessment. The technical information generated in Phase I and Phase II is then used in developing criteria and appropriate application guidelines and technical recommendations. An implementation phase (Phase III or the extended Phase II) has been identified for resolving issues that may be raised during the results utilization efforts.

A-1 SELECTION OF COMPONENTS, SYSTEMS, AND STRUCTURES FOR AGING EVALUATION

The first step in the approach used in NPAR is selecting components, systems, and structures for indepth engineering studies. The selection criteria include: the potential contribution to risk from failures of components, systems, and structures; experience obtained from operating plants; expert judgment over the tendency to aging degradation; and user needs. The user needs include resolving generic issues, plant performance indicators, and plant maintenance and surveillance.

The selection process includes establishing a boundary to define what is to be included in the components, system, or structure under consideration; and location of important interfaces.

A-1.1 Risk- and System-Oriented Identification of Aging Effects

An initial evaluation is made of the effect of aging on plant safety systems, support systems, and equipment and its impact on plant safety. The historical operating experience of light water reactors (LWRs), expert opinion, and system risk evaluations are all used in the initial evaluation.

A-1.1.1 Operating Experience and Expert Opinion

Failure data, derived from the operational experience of LWRs, provides information valuable for evaluating the impact of aging-related failures on operating plants. The data are used to categorize systems or components within specific systems that are susceptible to aging-related failures. General trends of aging failures are identified by

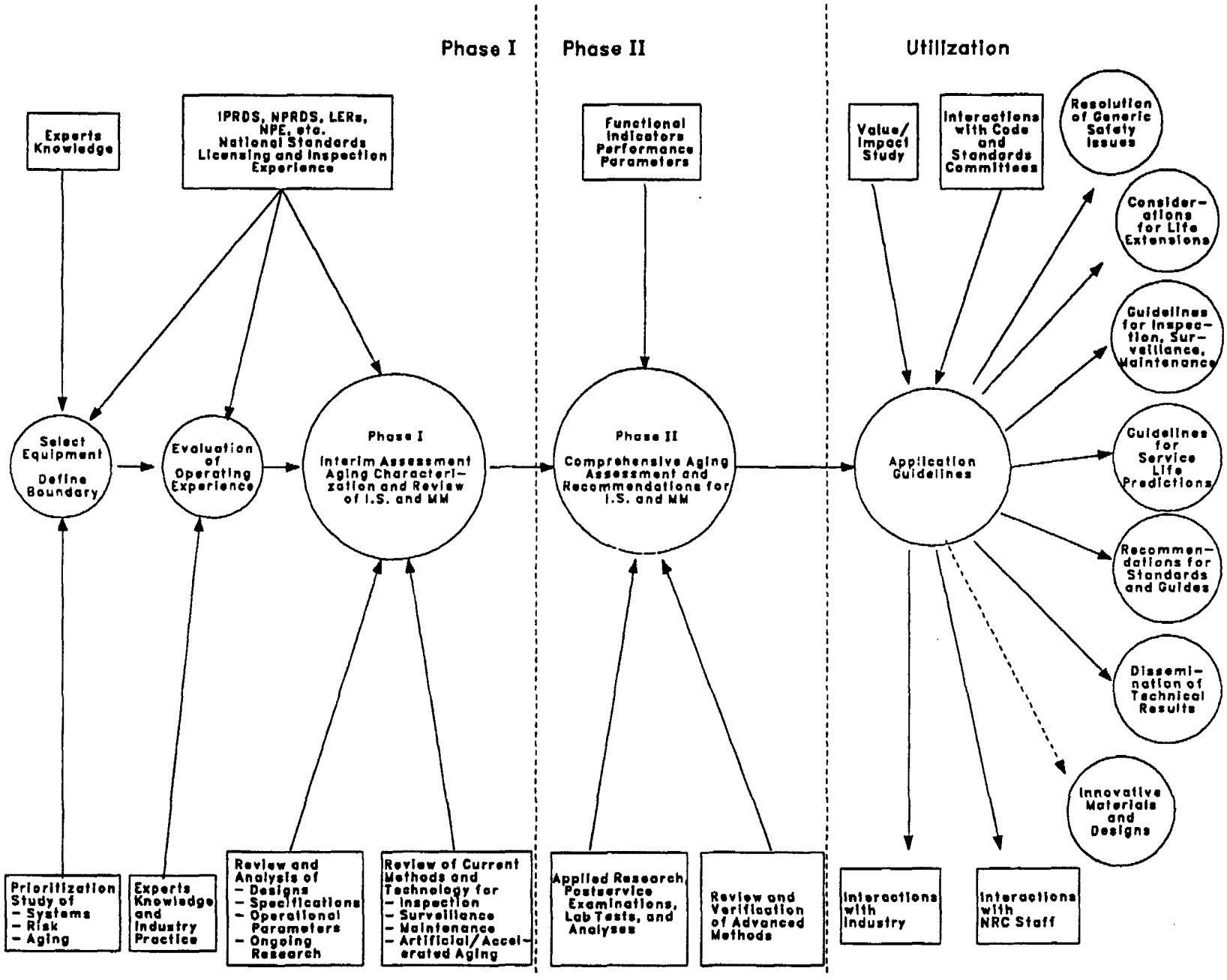

Figure A-1. NPAR program strategy.

categorization of failures, and actual aging mechanisms are identified by using root cause analysis techniques. In general, the failure category information is used to identify systems, and the respective system components, that are susceptible to age-related failures and warrant indepth engineering studies.

Plant operating data have been used since the start of the NPAR Program to identify aging-related degradation. Initial scoping studies (Refs. A-1 through A-3) were made to identify the effect aging has on components, systems, and structures and how aging degrades the required functional performance.

The information, currently being gathered for system level aging degradation, is derived from the Licensee Event Reports (LERs) and the Nuclear Plant Reliability Data System (NPRDS). This information is collected using failure-category and cause-codes that allows identifying the failure as belonging to one of several broad failure categories. Analysis of this failure category information is contained in NUREG/CR-4747 and NUREG/CR-4769 (Refs. A-4 and A-5). The data show that system dependencies of aging exist and can be readily identified. The systems covered in the NUREG reports are a subset of those delineated in NUREG-1144. Efforts are under way to obtain aging failure information on the remainder of the systems. The LER and NPRDS data will be supplemented with the data from the In-Plant Reliability Data System (IPRDS).

In addition to the systems covered in NUREG/CR-4747, information on the root cause of component failure was obtained for several service water and Class 1E electrical power distribution systems. These systems were chosen as a result of their safety significance determinations in past probabilistic risk assessment (PRA) studies. Failure information was also derived from the NPRDS. The aging mechanisms can be identified to the level of resolution provided in the failure records. This information is used in assessing the risk implications of aging mechanisms.

Surveys of expert opinion have also been conducted to identify age-related problems in nuclear plant hardware that impact safety. Workshops were held early in the program (Refs. A-6 and A-7) to identify the issues concerning aging and to review the state of knowledge on aging degradation. Aging problems in components, systems, and structures were identified and a consensus was reached concerning what components were the most important, in terms of aging-related degradation.

A-1.1.2 Risk Evaluation

The objective of this task is to identify components, systems, and structures that will significantly impact nuclear plant risk if aging-related degradation decreases reliability and availability or results in degraded performance. The initial approach taken was to use the results of existing PRAs to investigate the relationship between risk and aging-related degradation (Refs. A-8 and A-9). A sensitivity study was performed to determine what effect increases in failure rates of components

and systems had on overall plant risk. Based on the aging-sensitivity study, a large number of the risk-significant components were identified in the auxiliary feedwater system and the reactor protection system. Pumps, check valves, motor-operated valves, circuit breakers, and actuating circuits were the component types found to have the most potential risk impact.

Time-dependent calculations that propagate the effects of aging throughout plant design life and beyond are necessary to provide a reasonable estimate of aging effects. The reason for this analysis is that the risk ranking of systems and components can change with time when one considers aging effects. Techniques for performing time-dependent risk or core-melt probability calculations have been developed (Ref. A-5). Current efforts focus on the propagation of aging impacts, at the system/component level, using the aging failure information from root cause analysis. These results are not directly relatable to plant effects until the sequence calculations from PRA studies have been combined with the aging impact propagation techniques. Based on the results of the current evaluations and modeling development, it is expected that future efforts will focus on the larger aspects of plant safety or risk.

Risk evaluation of aging mechanisms are used to gage the importance of aging on plant safety. In general, the calculations are carried only to core melt to provide a relative estimate of the change in risk due to aging. Where situations warrant, the effects of aging are carried through to the actual calculation of risk.

A-2 PHASED APPROACH TO AGING ASSESSMENT AND INDEPTH ENGINEERING STUDIES

The NPAR Program essentially uses a two-phase approach for detailed aging assessments to make the best use of the available resources. The two-phase effort ensures that the work being done in the program is focused on the most significant research elements, modes of age-related degradation, and utilization of resources.

A-2.1 Phase I

The Phase I analysis of a selected component, system, or structure includes a review of three elements: (a) the hardware design, operating environment, and performance requirements, (b) a survey of operating experience, and (c) the current methods used for inspection, surveillance, monitoring, and maintenance and for qualifying end-of-life performance.

A-2.1.1 Review of Design Information and Applications

1. Design and Specifications. The first of the three elements of the Phase I assessment begins with a review of the design data and specifications for the hardware being studied. This includes such items as nonproprietary design documents, final safety analysis reports, operating and maintenance manuals, and product

literature. Additional sources of information are also pursued. These include vendor surveys, utility contacts, published reports, and expert opinion.

2. Materials. An important aspect of part of the assessment is identifying all the significant materials that comprise the hardware under review. A list of all significant materials is generated for the hardware. The materials and parts judged most susceptible to aging are identified.

3. Operating and Environmental Stressors. The age-related degradation of the components, systems, and structures is a time-dependent phenomenon and, among other things, depends on operating environment and operating history. The environmental effects that are considered include stressors, such as temperature, radiation, chemicals, contaminants, atmospheric conditions, humidity, and, in the case of primary system components, primary coolant chemistry. Also to be considered are the effects of secondary side coolant chemistry on PWR steam generators and the effects of service water and component cooling water chemistry on safe shutdown components. The environmental conditions considered include the conditions prevalent during operation and also the environmental conditions that prevail during other periods, such as during testing, shutdowns, storage periods, accident conditions, and postaccident situations.

   The operating history assessment includes the thermal, mechanical, and electrical stressors that components, systems, and structures experience during their operating lifetimes. For understanding the influence and effects of stressors and environment on aging degradation processes, one has to consider normal operating conditions, anticipated transients, off-normal conditions, and accident and postaccident conditions. Typical examples of electrical stressors include: slow-switching transients, fast transients of the lightning variety, and low frequency 50-60Hz signals, which can occur singly or in various combinations. Examples of mechanical stressors are static loading stresses, dynamic loading stresses, and seismic and vibrational stresses.

4. Performance Requirements and Functional Indicators. The performance requirements of the hardware are reviewed to assess if aging degrades the ability of the component, system, or structure to perform its required safety function during normal, abnormal, and accident conditions. Here, it may be possible to identify functional indicators. These consist of indicators that are practical to monitor and that provide cost-effective means to identify and manage age-related degradation. Finally, ongoing research is reviewed, and applicable results are included in the assessment of hardware under study.

The aforementioned review and analysis of materials, designs and specifications, stressors and environment, and operational parameters is attempted on all components and systems selected for the comprehensive aging assessment. It is recognized that some of the data search and analysis may not be technically, as well as practically, feasible. Nevertheless, an effort is made to acquire as much of this knowledge on a given component and system as possible. If safety issues warrant such detailed assessments and adequate resources are available, then the logic for reviews and analysis described above are followed. (All NPAR Program contractors follow this same strategy.)

A-2.1.2 Survey of Operating Experience and Failure Evaluation

A second element of the Phase I assessment is a critical survey of the operating experience obtained to date on the components, systems, and structures being evaluated. This review is intended to provide information on the failure rates and reliability that can be expected and the aging-related failure modes and causes that have been experienced.

Information on the failure modes and causes that have been experienced in nuclear power plants are obtained from a variety of sources. These include data from ongoing programs, such as the IPRDS sponsored by NRC and the NPRDS managed by the Institute of Nuclear Power Operation. Other sources include Licensee Event Reports (LERs), Nuclear Plant Experience (NPE), Plant Maintenance Records, and Inservice Inspection Reports.

In this element of the Phase I assessment, an evaluation is made of the hardware failures that have occurred to identify the following:

- Failure Mechanisms. These are established through the process of identifying dominant stressors; studying materials and designs of components and parts; and reviewing service environments and applications; then, evaluating the nature of, and the factors contributing to, age-related degradation and failures.

- Failure Modes. The indicators of failures (e.g., voltage collapse or disturbance in current signature) are assessed, and critical age-related failure modes are identified.

- Failure Causes. The conditions of design, manufacture, and service environments and applications that may lead to failures are determined.

A-2.1.3 Screening Examination and Testing

In addition to the evaluations of operational records, it is necessary to perform some screening-type examinations and tests on selected components to supplement or confirm deduced failure mechanisms. These are limited to screening-type examination and testing in Phase I. This work also assists in identifying key performance parameters that are monitored to determine the ongoing effects of aging.

Test samples may include equipment and components removed from service at operating LWRs and from mothballed or decommissioned reactors. Depending on circumstances, the examinations or tests are conducted in situ; onsite, after equipment is removed; or at various laboratories with appropriate test and examination capabilities.

The Shippingport PWR, now in decommissioning, is a source of test specimens for the NPAR Program. The candidate components for examination and testing have been identified through site visits by NRC and contractor experts representing a range of disciplines and interests. Detailed information for each specific component is developed and used to further assess its relevance to commercial LWR systems.

A-2.1.4 Review and Evaluation of Inspection, Surveillance, Monitoring, and Maintenance

The third element in the Phase I assessment is the review of the inspection, surveillance, monitoring, and maintenance (IS&MM) practices. This element also involves a review of artificial or accelerated aging techniques used to qualify hardware for end-of-life performance.

Existing methods for inspection, surveillance, and monitoring are evaluated to determine those methods likely to be effective in detecting aging degradation in an incipient stage before loss of safety function. Also, it is important that the methods not be unreasonably expensive to implement and not result in unacceptable levels of occupational exposure. Surveillance and monitoring methods being evaluated include periodic inspections, both visual and instrument-aided, and on-line instrumented techniques. The evaluation seeks to identify performance parameters and functional indicators that are capable of representing the functional capability of equipment and useful for managing aging degradation. It should be possible to monitor the selected parameters and indicators at operating plants for reasonable costs.

A review of artificial or accelerated aging techniques is made and compared to data available from naturally aged hardware to demonstrate the applicability of current practices.

A-2.1.5 Interim Assessment and Recommendations

The result of a Phase I evaluation of a component, system, or structure is an interim aging assessment, a defect characterization, and an evaluation of the safety significance of the probable failure modes. An interim evaluation of current IS&MM technology is given and lists of potential performance parameters and functional indictors are developed. Interim recommendations are made for Phase II studies based on the results and reviews of research activities completed in Phase I. The results of the Phase I study are issued in a technical progress report or a milestone report.

Continuation to Phase II activities, for a given component or system, is halted if (a) an adequate data base and experience exist within the

industry; (b) industry-sponsored programs adequately address the research needs; and (c) resources can be used better for other research activities.

A-2.2 Phase II

The Phase II assessment is a long-term effort. It includes validating advanced inspection, surveillance, and monitoring methods through laboratory and field testing of samples and validating accelerated aging techniques. It also includes developing models to simulate degradation, in situ aging assessment, and testing of naturally aged equipment from operating nuclear power plants.

A-2.2.1 Review and Verification of Improved IS&MM and In Situ Assessments

The Phase II research on IS&MM involves reviewing advanced methods and technology for each category of components under study. In this effort, advanced techniques and technologies, either in use or under development, are investigated. When available, the sources of technology both within and outside the nuclear industry are used. The sources outside the nuclear industry include fossil plants, the petro-chemical industry, the aerospace industry, various branches of the Department of Defense, and other government agencies. Also, the practical feasibility of applying these technologies to nuclear plant components are explored.

Laboratory and field application and verification tests of IS&MM candidate technology are carried out. The objective of the tests is to demonstrate that methods are appropriate to follow the dynamics of the performance parameters and functional indicators of interest; methods have adequate selectivity (will not give false indications) and sensitivity (will detect in the incipient stage); and suitable acceptance/rejection criteria are available so that maintenance needs can be correctly identified.

The laboratory tests involve simulating defects of varying degrees of intensity in prototype hardware to determine sensitivity and detection criteria. Various defect and environment combinations are used to determine selectivity. These laboratory tests are carried out to verify that the methods are applicable for in situ use at power plants. The field tests are recommended at cooperating utilities in order to confirm the laboratory results, provide information about the frequency and method of data collection and analysis, and estimate cost effectiveness and practicality of application.

A-2.2.2 Testing of Naturally Aged Components

A second element in the Phase II assessment is examining and testing naturally aged components obtained from operating power plants. This research element is perhaps the most cost intensive and difficult element of the NPAR Program. Yet, it is essential to quantify aging and determine that adequate safety margins exist to ensure the operational readiness of naturally aged components and systems during design basis accident situations.

Considerable effort is needed to acquire naturally aged equipment for examinations and tests. Equipment that has experienced significant operating and environmental stressors are being sought from various sources. The sources under consideration include commercial operating plants, decommissioned facilities, and research reactors. A major thrust of this research element is to evaluate performance of aged equipment before and after it is subjected to the stressors and environmental conditions expected under accident conditions. The evaluation is based on following the dynamics of performance parameters and functional indicators, which were identified in Phase I activity.

In situ monitoring of operating equipment at LWRs is recommended to gain an understanding of the interaction between aging and service wear defect characterization and inspection, surveillance, and maintenance.

When available, aging assessments are performed on equipment that has failed during operation, as well as on equipment that has survived extensive periods of operation. This is done to gain an understanding of those aging effects that would only be excited during a trigger event accompanied by abnormal stresses.

A-2.2.3 Residual Life Evaluations

Residual life evaluations will also be performed using data generated for major components or test specimens from major components, if available. Work is already under way in other programs to evaluate the degradation of the steam generators removed from Surry (Ref. A-10). The structural integrity of the reactor pressure vessel has also been the focus of a large and continuing effort (Ref. A-11). Programs for aging assessments of artificially aged and naturally aged cables are in place by NRC and the industry. Results from these research efforts and similar efforts will be integrated into the residual life evaluation of the major LWR components for use in considerations of plant life extension/license renewal.

A-2.2.4 Service Life Prediction Methods

A third element in the Phase II assessment is developing service life prediction models. This includes a compilation of currently used methods and an evaluation of their applicability. The effort will include testing and examination of aged components, from a participating power plant, and comparing these results to the service life predictions based on the actual service history. This effort also includes developing and qualifying residual life models for the major components of nuclear power plants.

A-3 APPLICATION GUIDELINES

The NPAR strategy flow chart shows the research performed from the Phase I and Phase II assessments will lead to developing application guidelines for codes and standards and recommending improved IS&MM

practices. It will also provide a systematic collection of historical baseline data and trending information for evaluating component and system aging effects. The specific areas of application, shown in Figure A-1, provide highlights of the end uses of the NPAR Program. An important part of the application guideline phase of the NPAR Program is the technical integration of the results obtained by NPAR and other major programs. In developing application guidelines, use will be made of the results available from both NRC and external programs. In developing guidelines, NPAR will work with all the NRC offices involved in aging and life extension, with codes and standards committees, and with industry groups. This work will be done to evaluate the effects new guidelines will have on plant equipment and systems.

A-3.1 Value-Impact Study and Coordination with Users

In the development of application guidelines, a value-impact study will be performed and an interchange with the various end users of the information is planned. For example, before generating application guidelines for improved IS&MM technology, a concerted effort will be made to interact with the NRC staff, code and standards committees, and industry groups, as indicated in Figure A-1.

In this example, value-impact studies will be made of the candidate surveillance and monitoring methods for degraded components considered to have potential for eventual implementation at operating plants. These will be evaluated to determine the occupational exposure likely to occur in conjunction with such methods. The study will assist in identifying those methods that are cost effective and practical for application in a commercial plant environment.

A-3.2 Support Resolution of Generic Safety Issues

The NRC report, NUREG-0933 (Ref. A-12), contains a recommended priority list to assist in the timely and efficient resolution of safety issues that have a high potential for reducing risk. The NPAR Program will generate guidelines, develop criteria, and support resolution of the generic safety issues. The generic safety issues that would directly benefit from the NPAR Program results are listed in Table 3.1 (see Section 3.2 of this report).

A-3.3 Considerations for License Renewal/Life Extension

An important objective of the NPAR Program is identifying and resolving the technical safety issues involved in requests for license renewal of nuclear power plants. The end product of NPAR will be guidance or recommendations to NRC users on subjects such as revisions to IS&MM methods, residual lifetimes of major components, key technical information required in applications for license renewal, and ensuring continued safe operation of plants with license renewals.

A-3.4 Guidelines for Inspection, Surveillance, and Maintenance

Another end product of the NPAR Program is evaluating the role of maintenance in mitigating aging effects and developing guidelines for revised or preferred maintenance practices. This effort consists of the following activities: reviewing current practices and procedures, reviewing vendors' recommendations, evaluating merits of performing preventive or corrective maintenance, identifying failures caused by maintenance procedures, and developing recommendations for a preferred maintenance approach.

A-3.5 Guidelines for Service Life Predictions

Guidelines will be developed for service life predictions of aged components and systems. Improved service life prediction methods for electrical components and residual life models for major components are to be developed and qualified in Phase II. Using these methods, guidelines will be developed taking into special account the effects of plant operating history. These guidelines will be of particular use in the areas of license renewal and maintenance and surveillance.

A-3.6 Recommendations for Standards and Guides

The NPAR Program will develop recommendations for revising relevant industry codes and standards for continued aged plant operation. The NPAR Program will also provide a technical basis for preparing NRC regulatory guides and review procedures concerned with the continued operation of aged plants and also for license renewal considerations.

A-3.7 Dissemination of Technical Results

The research information developed in NPAR is being disseminated by preparing technical papers and reports and sponsoring workshops and symposia and information exchange programs. The program also includes establishing an information data bank that will be available to NRC user organizations and to other user groups (such as utilities, manufacturers, and laboratories).

A-3.8 Innovative Materials and Design

The last application shown for the NPAR Program strategy (in Figure A-1) is innovative materials and designs. Here, recommendations would be provided (and it is up to the industry to implement) to evaluate design changes to existing components, systems, and structures, which would make them less susceptible to aging-induced degradation. It would also find end uses in the other NRC programs (such as equipment qualification and advanced LWR designs).

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*Available in the NRC Public Document Room, 1717 H Street NW., Washington, D.C.

APPENDIX B
MAJOR NPAR PROGRAM ELEMENTS

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APPENDIX B
MAJOR NPAR PROGRAM ELEMENTS

Appendix B contains a description of some major subjects or technical areas being addressed in the NPAR Program.

B-1 AGING--SYSTEMS INTERACTION STUDY

Aging research was initiated in this area to determine how aging is affecting the levels of plant safety at operating plants. This problem is being addressed by establishing the relative contribution to risk from age-related system failures. The initial approach is a data-collection effort to identify aging failures over a broad category and identifying root causes or specific mechanisms of failures for specific systems and components.

While the data collection provides information about aging failures of systems and components, to be useful the data need to be supplemented. Aging is important when it changes the overall levels of plant safety. This type of determination is accomplished through using PRA techniques coupled with time-dependent propagation of aging effects.

The result of the research effort is to establish the importance of various aging-related failures or mechanisms to core-melt frequency or plant risk. This provides a method to identify where (in plant safety systems, support systems, and components) aging is significant to risk; and to prioritize systems and components for indepth aging studies. A second application of this work is in inspection and maintenance. Systems and components susceptible to age-related failures can be identified so that corrective action can be taken.

B-2 AGING ASSESSMENT OF COMPONENTS AND SYSTEMS

Detailed engineering studies of aging effects in selected systems and components are being conducted. The objectives of these studies are to: identify age-related failure mechanisms in systems and components; and investigate the ability of aged components and systems considered vital to plant safety to perform their required safety functions during or after transients and accidents. These efforts include evaluating operating experience, screening-type aging assessments, and onsite and laboratory testing of naturally aged equipment, and, in some cases, laboratory-aged equipment.

The engineering study of aging effects on components and systems will result in identifying:

- The age-related failure mechanisms for the major systems and components.
- Stressors due to environment, maintenance, and operation.

- Aging sites, failure modes, and effects on plant safety or operation.

- Methods to detect and manage aging degradation.

B-3 AGING ASSESSMENT OF CIVIL STRUCTURES

The objectives of this research are to: identify structural safety issues that the NRC will need to address when plant license extension applications are reviewed; conduct confirmatory research to validate long-term behavior of structural materials when subjected to internal and external stressors; and determine the short- and long-term impacts of a post-LOCA (or TMI-2 like) containment environment on structural materials.

An easy-to-use data base containing aged structural materials' properties will be generated from available data, including data obtained using samples from decommissioned facilities. Inservice inspection programs will be used to provide deterioration trends. A methodology will be developed to quantitatively assess structural reliability, as affected by aging. From these, NRC license reviewers will be provided with review issues and acceptance criteria for use in structural reviews of plants for license extension requests.

Preliminary work, already completed in other research programs, is providing input to this research. These include: NUREG/CR-4652 (Ref. B-1), and a report by the U.S. Army Corps of Engineers, Vicksburg (Ref. B-2) on completed testing of a concrete sample from the shield wall of the decommissioned reactor at Gundremmigen, FRG. The sample (from the latter report) shows no significant effect on its structural properties after about 11 years of operational irradiation.

The results of this research effort will include:

- More advanced understanding of structural material behavior, including environmental effects of time, temperature, radiation, moisture, and chemical interactions.

- Improved understanding of the presence, or absence, of synergistic effects on cyclic or fatigue loads and their effect on the useful life of structural elements when combined with service or accident loads (e.g., effect on prestressing levels).

- Developing improved nondestructive techniques for examining concrete structures and prestressing systems for defects, deterioration, or damage.

- Defining and examining the unique environmental conditions that exist in a post-LOCA containment, and the effect on structural materials of unforeseen stress or corrosion, at a time when functional reliability is essential but access for inspection and maintenance are not practical.

Regulatory applications of this research are:

- Improved predictions of long-term structural deterioration.
- Improved predictions of available safety margins at future times.
- Limits on hostile environmental exposures.
- Reduction of licensing reliance on inspection and surveillance.
- Informed review and approval of plant license extension applications.
- Incorporation of research results into national design and inspection standards referenced by the Standard Review Plan (SRP).

B-4 INSPECTION, SURVEILLANCE, AND MONITORING METHODS

A principal element in the engineering evaluation of aging-related degradation is assessing methods used in inspection, surveillance, and monitoring of nuclear plant systems and components. A review is made of methods currently in use in Phase I. Also, an evaluation is provided of their effectiveness in detecting aging degradation at an incipient stage before a loss of safety function. A review is made of advanced techniques in Phase II.

The Phase II efforts include reviewing sources, both inside and outside of the industry, and advanced techniques or technology. Following this, candidate methods are chosen for laboratory and field testing. The objective of the testing is to show that the methods have adequate selectivity (will not give false indication) and sensitivity (will detect degradation in the incipient stage). Laboratory testing is carried out to demonstrate how advanced methods work in a controlled testing environment. Successful candidate methods are then subjected to field testing to confirm the laboratory results and provide information on practical field applications of advanced methods.

In both the Phase I and Phase II assessments, performance parameters and functional indicators that can be used to identify age-related degradation at an incipient stage are identified. In some cases, they can be used to assess the severity of a problem and its specific cause.

B-5 ROLE OF MAINTENANCE IN MITIGATING AGING

Both the Phase I and the Phase II efforts will include evaluating the role of maintenance in mitigating aging effects. This effort will consist of:

- Reviewing current practices and procedures carried out by nuclear utilities to maintain equipment; considering each component

selected for aging assessments; and recommending maintenance methods to ensure safety. For completeness, also including additional components considered important by the utilities.

- Reviewing nuclear equipment vendors' recommendations for maintenance of components or subcomponents selected for aging assessments.

- Performing an evaluation, including a comparative analysis, of the relative merits of: (a) performing maintenance when a component has been discovered to be malfunctioning (corrective maintenance); and (b) performing maintenance when an observation has been made through surveillance, inspection, or monitoring that a component may not function when required during a design basis or "trigger" event (preventive maintenance). A "trigger" event is an operational transient or minor accident which can lead to a more serious event when followed by failures in safety and backup systems. Emphasis is placed on the relationship between failures (causes or modes) expected to be experienced during operation and those that would potentially occur under the stresses associated with design basis or trigger events.

- Evaluating the relative merits of predictive inspection and monitoring methods that can be used to identify imminent failures (predictive maintenance). Predictive maintenance will enable corrective maintenance or replacement to be scheduled based on actual equipment performance. This approach lends itself to use of reliability methods and condition monitoring to mitigate equipment degradation due to aging.

- Identifying, where possible, those component failure mechanisms likely to be induced through preventive or corrective maintenance; specifically, looking for those that might be detectable through short-term, postmaintenance surveillance, inspection, or monitoring.

- Developing recommendations for acceptable or preferred maintenance practices based on the foregoing activities.

In all cases, the emphasis is on the technical or hardware aspects of maintenance rather than on institutional, organization, programmatic, or human factors considerations.

B-6 COMPONENT LIFETIME EVALUATION

An evaluation of the age degradation and residual life of major light water reactor (LWR) components is being performed in the NPAR Program. The information generated in this research effort has two principal objectives. One is assisting in developing criteria that ensure age-related degradation of major components does not impair safe plant operation. The second is generating a technical basis for establishing criteria and developing guidelines to be used in licensing review

procedures and plant life extension/license renewal. The effort is complementary to the ongoing industry-sponsored pilot projects on plant life extension. However, the NPAR effort is fundamentally safety oriented.

The approach used in the residual life assessment project is to first identify and prioritize the major components with respect to safe plant operation. This is followed by an initial effort to establish the life-limiting processes for each of the major components. Included in this effort is identifying degradation sites and failure modes during normal operation and accident conditions.

The initial effort also includes assessing current and potential methods for inspection, surveillance, and monitoring. For this phase of the effort, the work is focused on integrating currently available technical information relevant to aging and life extension.

The results from the initial assessment are then used in developing simple mechanistic models for determining the residual life of selected major components. Developing these models is of particular interest for components that are not readily accessible for routine maintenance and inspection.

As these models are developed, residual life evaluation is planned for the major components using actual plant operating data. In this phase of the project, key plant operating data will be identified. These are operating data necessary for a realistic (rather than conservative) estimate of the mechanical and thermal loading of the components, as well as other environmental stressors. This segment of the NPAR Program is coordinated with the ongoing research programs involving vessels, piping, steam generators, and nondestructive examination techniques.

B-7 INVESTIGATION OF AGING/SEISMIC SHOCK INTERACTIONS

An understanding of the vulnerability of age-degraded equipment to seismic disturbances is necessary for the design life of a nuclear power plant. This includes the original plant life of 40 years and any extended life period. Current industry standards (IEEE 323 and 344) require preaging before seismic qualification of electrical equipment. However, the NRC has not determined such a need for mechanical components and is currently evaluating the significance of aging as a factor in the qualification of mechanical equipment and relevant regulatory guides. Therefore, an assessment is needed of the potential importance of aging in degrading seismic performance of equipment.

Both the nuclear industry and the regulatory agency have ongoing programs to assess the aging-seismic effects. These include:

- Laboratory testing of naturally, as well as artificially, aged components.
- Qualifying equipment using existing test data.

- Using experimental data for qualifying components.
- Developing of seismic fragilities for different components.
- Identifying weak links in certain equipment assemblies.
- Developing surveillance and maintenance programs to alleviate the aging effects on seismic performance of equipment.

In order to avoid duplication efforts, some studies involve both industry and NRC. Recently, it was determined that the qualified life of some equipment may have to be extended in the event that a utility submitted an application to the NRC for a license renewal or plant life extension. In that case, the components originally qualified for a 40-year period would have to be reassessed for extended life.

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APPENDIX C
NPAR PROGRAM ACTIVITIES

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APPENDIX C
NPAR PROGRAM ACTIVITIES

This appendix contains a description of the various research studies being performed as part of the NPAR Program. The scope of each of the elements in the program and their current status are discussed. The major NPAR work under way at this time is being done at the following national laboratories:

- Battelle Pacific Northwest Laboratories (PNL)
- Brookhaven National Laboratory (BNL)
- Idaho National Engineering Laboratory (INEL)
- Oak Ridge National Laboratory (ORNL)
- Sandia National Laboratories (SNL).

Support for the NPAR Program is also being provided by the Franklin Research Center (FRC) as a subcontractor to ORNL and BNL, Systems Engineering Associates, and the National Bureau of Standards.

In addition, work is subcontracted to various private engineering firms and private and academic consultants to make use of special expertise.

The following sections describe the major research activities of the NPAR Program.

C-1 NPAR PROGRAM--PNL ACTIVITIES

Research is being performed on six major elements of the NPAR Program by PNL. The six elements are (a) Shippingport Reactor Aging Evaluation, (b) Aging Assessment and Analysis of Snubbers, (c) Diesel Generator Aging and Life Extension Assessment, (d) Service Water System Assessment, (e) Aging Assessment of Room Coolers, and (f) Quantification of Aging. The following describes the scope of work of each of these elements.

C-1.1 Shippingport Reactor Aging Evaluation

The Shippingport Atomic Power Station, now undergoing decommissioning, is a major source of naturally aged equipment for NPAR component and system evaluations. As the first U.S. large-scale, central-station nuclear plant, Shippingport parallels commercial pressurized water reactors (PWRs) in reactor, steam, auxiliary, support, and safety systems. With its 25-year service life (1957 to 1982), it covers almost the entire time span of currently operating reactors. Also, because of substantial modifications during the mid-60s and -70s, it offers unique examples of identical or similar equipment used side by side but representing different vintages and degrees of aging.

The objective of this NPAR task is to perform in situ assessments, acquire selected components and samples of materials, obtain data and records, and conduct postservice examinations and tests of Shippingport equipment and materials in support of NPAR and other NRC programs. A summary description of this effort including accomplishments and status is given below.

PNL coordinates closely with the DOE Shippingport Station Decommissioning Project Manager and designated site personnel to incorporate the activities of programmatic interest to NRC in the overall site decommissioning plans. Work is in progress to obtain all available data and records for the systems, components, and materials selected from Shippingport. This includes designs and specifications; operation and maintenance manuals; operation and postoperation histories; maintenance history/record; inspection, surveillance, and periodic test procedures/data; etc. Progress includes the acquisition of more than 50 technical manuals for plant components, many original equipment and materials specifications, and the maintenance history and record of changes for key components. The information for several selected systems and components has been compiled and distributed to the assigned NPAR contractors to support their evaluation studies.

Arrangements are coordinated and site services are obtained as required to support the in situ assessment of systems, components, and materials before their removal by the decommissioning operations contractor. These in situ assessments include visual and physical examinations, testing electrical circuits and components, response studies of various electromechanical devices, and different types of nondestructive examinations and tests. EG&G/INEL, for example, has conducted a comprehensive in situ evaluation of 46 Shippingport station electrical components and circuits representing more than 1600 individual measurements of insulation resistance, dc loop resistance, circuit capacitance, inductance, and impedance. In addition, the ferrite content of cast austenitic stainless steel primary system main valves and coolant pump volutes has been measured in situ to identify candidate materials for MEB/DE thermal embrittlement studies. These in situ measurements indicated that 9 of the 24 cast primary system components had sufficiently high ferrite levels to make them of interest for acquisition for detailed materials studies.

Arrangements are coordinated and site services are obtained as required to support the acquisition of components selected by NRC and its contractors for offsite evaluation. This includes identifying, removing, packaging, and shipping components obtained in conjunction with the decommissioning operations and also retrieving selected components after shipment to Hanford, Washington, for disposal.

More than 100 Shippingport station components have been selected for NPAR evaluation and testing through site visits by NRC and contractor experts representing a range of disciplines and interests. Table C-1

TABLE C-1. Shippingport station components.

Offsite Evaluation	Number of Items		
	Selected	Removed	Shipped
Pacific Northwest Laboratories (PNL)			
• PV Nozzle Cutouts	5	5	5
• Coolant Purification Piping	2		
• Rad Waste Piping	2		
• B/D Instrument Piping	2		
• Fuel Pool Piping	2		
• Main Steam Piping	1	1	
• Feedwater Piping	1	1	
Argonne National Laboratory (ANL)			
• Main Coolant Pump	1	1	
• Check Valves	4	2	
• Manual Isolation Valves	3		
• Hot Leg Pipe Section	1		
• Cold Leg Pipe Section	1	1	
Brookhaven National Laboratory (BNL)			
• Motor-Generator Set	1	1	1
• Battery Chargers	2	1	1
• Inverters	3	2	2
• Motor Control Center	1	1	1
• Differential Relays	2		
• Protective Relays	4		
• Agastat Relays	5	1	1
• Scram Breakers	2		
• MG-6 Relays	4	2	2
• DB-50 Breakers	2		
• Circuit Breakers	8		
• Current Transformers	2		
• Potential Transformers	2		
• 480/120 Transformers	2		
• Constant Voltage Transformers	2		
• Relay Panel	1		
• Spare Parts for Charger/Inverter	1 box	1 box	1 box
EG&G Idaho, Inc. (INEL)			
• Motor-Operated Valves	2	2	2
• Limit Switches	8	8	
• Battery Cells	6		
• Nuclear Instrumentation Channels	2		
• Electrical Panel	1		

TABLE C-1. (continued)

Offsite Evaluation	Number of Items		
	Selected	Removed	Shipped
EG&G Idaho, Inc. (INEL) (continued)			
• Thermocouple Signal Box	1		
• Thermocouple Junction Box	1		
• Power Lead Junction Box	1		
• Terminal Strip	1		
• Power Cable	1		
• Instrumentation Cable	2		
• Rod Control Junction Boxes	2		
• Selector Switch	3	2	2
• Pressure Switches	7		
• Rosemount Transducers	4		
• RTDs	16		
• D/P Cells	6	1	
• Transmitters	6	1	
• Level Indicator	1		
• Compensating Ion Chamber Detectors	4		
• BF_3 Detectors	4		
Oak Ridge National Laboratory (ORNL)			
• Solenoid-Operated Valves	7	4	
• Motor Operator	1		
• Motor-Operated Valves	5		
• Check Valves	4	1	1

contains a listing, grouped by contractor, of these NPAR items and other components that are being acquired for the Materials Engineering Branch as part of the NPAR site coordination effort.

Acquisition of many of these components should be completed by the end of FY 1988. Progress will be reported in a milestone report.

C-1.2 Aging Assessment and Analysis of Snubbers

An aging assessment is being performed on the hydraulic and mechanical snubbers used in nuclear power plants. This assessment is being done to establish failure mechanisms and causes and provide recommendations for practical cost-effective inspection, surveillance, and maintenance methods. The Phase I assessment has been completed and the results are documented in the Phase I report entitled "Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety-Related Piping and Components of Nuclear Power Plants" (Ref. C-1).

The report presents an overview of hydraulic and mechanical snubbers based on information from the literature and other sources. Snubber operating experience is reviewed using licensee event reports (LERs) and other historical data for a period of more than 10 years. Data are statistically analyzed using arbitrary snubber populations; implications of the observed trends are assessed; and recommendations to modify and improve examination and testing procedures to enhance snubber reliability are determined. Value-impact was also considered in terms of exposure to a radioactive environment for examination/testing and the influence of lost snubber function and subsequent testing program expansion on the costs and operation of a nuclear power plant. Last of all, optimization of snubber populations by selective removal of unnecessary snubbers was also considered.

The current Phase II research is being conducted in accordance with the NPAR Program strategy starting with a comprehensive aging assessment, postservice examination, and laboratory testing of aged snubbers. The effects of accident conditions, e.g., seismic and loss-of-coolant accident (LOCA) effects are to be included. A significant effort is planned during the testing phase to evaluate performance indicators for snubbers. The results of this last effort are intended to assist in identifying practical and effective performance indicators.

Advanced methods for snubber maintenance and inservice evaluation will be assessed. These results, along with the Phase I results, will be used in developing application guidelines for inspection, surveillance, and monitoring methods for snubbers. This last activity will include interfacing with NRC, operating utilities, and appropriate codes and standards committees.

C-1.3 Diesel Generator Aging and Life Extension Assessment

A multidisciplinary evaluation is being concluded on nuclear service diesel generators. This task consists of an aging assessment of diesel generators, systems, and related components. The evaluation of surveillance, inspection, and maintenance methods used on diesel generators and their role in mitigating aging effects is also included.

The Phase I effort on this element has been completed. An evaluation was made of current operating experience and an interim aging assessment was performed in Phase I. A large data base was established by diesel experts and placed into a computer-managed format for development and analysis. The draft report on Interim Aging Assessment was subjected to an industry-wide peer review in a Diesel Generator Aging Seminar. After the peer review, selected results of the workshop were incorporated into the Phase I report as Volume II and a revised final report (Volumes I and II) was issued (Ref. C-2).

The Phase II effort extends the evaluation of aging assessment started in Phase I, including selected studies of specific installations and the testing, inspection, and diagnostic instrumentation in use. It will

include a review of current IS&MM practices and recommendations for improved surveillance and monitoring for early detection of aging effects. Phase II will also include a study of current and potential service life prediction methods.

The Phase II effort will lead to guidelines for improved IS&MM methods leading to mitigation of aging effects. It will also provide a technical basis for establishing the scope and documentation requirements for license renewal submittals. Another benefit from this effort may be improved reliability of emergency diesel generating systems. Some of the current testing and inspection requirements for these systems have been identified as having an adverse effect on reliability and availability. This effort includes an investigation of alternative methods of inspection and alternative testing schedules that will result in better overall reliability.

C-1.4 Service Water System Aging Studies

This element consists of an aging assessment of the safety-related areas of the Service Water System (SWS) of a nuclear power plant.

The function of the SWS is to transfer the heat loads from various sources in the plant to the ultimate heat sink. The three safety-related heat sources served by this system are identified to be:

- Core decay heat,
- Decay heat removal components, and
- Emergency power sources.

Because of the wide variation in the nature of each plant's ultimate heat sink and the application of a multiplicity of system design approaches, the system is defined from a functional standpoint as: All components, their associated instrumentation, controls, electrical power, cooling and seal water, lubrication, and other auxiliary equipment that comprises the final heat transfer loop between the safety-related heat sources and the ultimate heat sink.

A Phase I assessment is currently being performed on this system.

A detailed task plan is being prepared, incorporating an interdisciplinary approach, and site visits are planned to acquire actual SWS operational data. The visits will verify the adequacy of current data base information. The data will be analyzed to produce interim recommendations with respect to:

- System inspection guidelines.
- Component monitoring methodology (surveillance).
- Systematic approach to component maintenance.

- System life extension regulatory requirements.

The results will be documented in a NUREG/CR report on Phase I.

Additional work on this element will follow the NPAR phased approach. Phase I conclusions that require additional verification or investigation will be pursued, along with any further analysis required. The end result of this task is to produce a set of recommended guidelines and to support the specific NRC license renewal regulatory uses. All Phase II data and reporting requirements will be satisfied before the end of FY 1989 to ensure timely support of the NRC life extension effort.

C-1.5 Aging Assessment of Room Coolers

This element consisted of an aging assessment of room coolers. An NPAR Phase I study on room coolers was completed in FY 1986. The results are summarized in a report entitled "Operating Experience and Aging Assessment of ECCS Pump Room Coolers," issued in October 1986 (Ref. C-3).

Room cooler operating experience data obtained during this study suggest that pump room cooler operation by itself has been relatively trouble-free; LER data indicate that room cooler failures are rare. Room cooler failures usually develop outside the room cooler boundary in the motor control center electrical components or in the SWS chiller, valves, and pumps. These components are subjects of aging assessments in other NPAR tasks and, therefore, no duplication of effort was made.

It was recommended that Phase II of the emergency core cooling system (ECCS) pump room cooler aging assessment be delayed until NPAR system assessments now under way have further addressed the significance of room coolers and more extensive operating experience is available.

C-1.6 A Practical Approach for the Quantification of Aging (QOA)

The objective of this task is to develop and demonstrate a practical approach to quantify the effects of aging on the safety margins for safety-related electrical equipment. A safety margin is a measure of a component's level of safety compared to a chosen operating limit and is useful when data are insufficient to assess the remaining life. The primary activities in this task are to develop a general methodology for assessing safety margins, demonstrate the method on a selected reactor safety component, and integrate the results with the NPAR Program plan.

This task belongs to the category of special topics and does not follow the usual NPAR phased approach.

The initial development of the practical approach for QOA was completed in FY 1986. One major result of the early work was identifying a general definition for a safety margin: the difference between a selected operating limit for a chosen performance indicator and the current value of that performance indicator. This definition is applicable to a wide range of operating limits, performance indicators, and equipment types.

The current work on this task involves refining the methodology developed to date. Along with this, several electrical components will be selected and evaluated to demonstrate on a preliminary basis how this approach will be applied, including the development of an engineering model. The latter may include statistical data from in situ assessments, postservice examinations and tests, trending performance parameters (functional indicators), and controlled laboratory testing. Such a model is needed to determine if a foundation for extrapolations of trending data can be established.

Based on these results, a generalized methodology to quantify aging of reactor equipment will be generated for comparing equipment operating criteria to expected performance.

Future work is expected to address refinements and applications of the QOA methodology.

C-2 NPAR PROGRAM--BNL ACTIVITIES

Brookhaven National Laboratory is currently conducting a research effort on the characterization and detection of age-related failures of selected components and systems. The main objectives of this research effort are summarized as follows. First, identify and characterize the aging and service wear effects associated with selected safety system components. Next, determine how aging and service wear affect the capability of selected equipment to operate during or after seismic events. Third, determine what methods may be effective in detecting significant aging and service wear deterioration that may compromise component performance.

The systems and components being evaluated include electric motors, battery chargers and inverters, circuit breakers and relays, residual heat removal systems, and component cooling water systems. BNL is also engaged in the following research activities: integration of the effects of aging into the NRC inspection program; aging-seismic considerations; and failure analysis of reactor coolant pump seals.

C-2.1 Electric Motors

Electric motors of all types and sizes are used to drive pumps, valves, fans, and control devices in a typical nuclear power plant. These components serve important roles for performing normal operations, as well as for accomplishing safeguard functions during and/or after an abnormal or accident event.

During motor operation, various parameters such as temperature, vibration, current, voltage, and applicable equipment output can be used to generally assess motor integrity. When normal values for these parameters are observed to adversely change, an incipient stage of degradation potentially leading to ultimate failure is suspected. Therefore, characteristic parameter performance can identify failure modes, mechanisms, and causes representative for all types of motors.

The Phase I aging assessment of motors in nuclear power plants is complete and the results documented in NUREG/CR-4156 (Ref. C-4). In addition to failure-mode assessment, the report identifies functional indicators suitable for monitoring the motor dielectric, rotational, and mechanical integrities.

A Phase II effort on motors is under way to assess the industry and regulatory standards and guides for motor performance in power plants. As part of this effort, a survey on current industry practices was performed to assess the adequacy in the existing surveillance and maintenance activities. Also included in Phase II are two test programs, one on a 12-year naturally aged 10 hp motor, and the other on a failed 400 hp motor. These were conducted to evaluate the suitability of various test methods that can be used to monitor the motor health. The reports on these activities are under review.

The Phase II efforts, which include the above research activities, are being documented in a NUREG/CR and the report(s) will be published shortly. The report will provide recommendations for developing motor maintenance programs in nuclear facilities in order to improve motor reliability both under normal and accident conditions.

Recommendations for maintenance guidelines are now being prepared. Periodic testing, surveillance techniques, and continuous monitoring methods are reviewed and assessed. Methods are presented for performance evaluation and trend analysis, as well as for a value-impact analysis.

Current industry maintenance practices were assessed by reviewing the motor maintenance requirements at four nuclear power stations. Insulation resistance is always measured, whereas motor-operating current is only recorded in some cases. Most testing is done for the driven equipment, such as motor-operated valve stroke time or pump speed, which gives little indication of motor condition. Trend analysis is virtually nonexistent for evaluating motor condition.

The maintenance guidelines, including a discussion of reliability centered maintenance (RCM), are presented, along with a logic chart, to make motor maintenance decisions using RCM philosophy. This logic is applied to the specific application of containment fan cooler motors. The resultant maintenance recommendations do not require any hardware modifications but do require more testing and trend analysis than what currently exists.

C-2.2 Battery Chargers and Inverters

Nuclear power plants use battery chargers and inverters to supply power to safety-related equipment, instrumentation, and controls. A battery charger converts alternating current (ac) to direct current (dc) to provide power to dc-driven equipment and components as well as to keep the standby batteries fully charged. On the other hand, inverters are used to supply ac power to safety-related equipment and equipment important to plant operation after converting the dc power source to an ac output.

Plant systems such as the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), Reactor Core Isolation Cooling (RCIC) System, and the ac/dc distribution system use these devices to satisfy certain nuclear power station safety requirements.

The Phase I report covering the aging assessment and review of operating experience of battery chargers and inverters was published in June 1986 (Ref. C-5). This report describes the aging and service wear characteristics of chargers and inverters based on a comprehensive review of their operating experience in nuclear power plants. It concludes that subcomponent failures due to aging and wearout contribute to equipment reliability with potential impact on plant safety.

The current effort is focused on the testing of naturally aged inverters and a naturally aged battery charger to determine the practicality and viability of surveillance and monitoring methods cited in the Phase I report. Research into advanced inspection, surveillance, and monitoring methods employed in other industries is also being conducted to determine applicability to nuclear plants. Because inverters are used extensively in computer applications, it is expected that investigation in this area will provide useful information. Inputs to regulatory guides and national codes and standards are expected in this area, especially for the inverters.

C-2.3 Circuit Breakers and Relays

Three types of relays are used in nuclear safety-related systems: protective, control, and timing relays. Protective relays detect abnormal conditions on the plant's power system and initiate opening of circuit breakers to prevent damage to the protected equipment such as motors, buses, and transformers. Control relays are used in the logic and protective action initiation systems and are basically two-position relays with contacts that transfer position when the relay's coil is energized. Timing relays are used to relay or sustain a signal for a specific period in accordance with system operating requirements.

Two types of circuit breakers (CBs) that are commonly used in nuclear safety applications are: molded-case and metal-clad. Molded-case CBs are used in low-voltage applications up to 480 V for lower level distribution systems and low power loads. The metal-clad switchgear CBs are used in applications where large loads and higher fault currents are involved. They are more sophisticated than molded-case CBs and are used on safety systems with voltages ranging from 480 to 6900 V.

The aging interaction relating to CBs and relays used in the safety injection system will be described in NUREG/CR-4715 (to be issued). The most prevalent failure causes for the relays and CBs are reviewed and their effect on system operation evaluated. The study concludes that failure of a safety injection train is possible from CB and relay failure if adequate maintenance and testing is not performed. Failure of redundant trains is not expected from common-mode failure of a particular type of CB or relay.

However, a sufficient number of different types of failures were found, and this supports the need for a strong maintenance and test program to prevent multiple age-related failures.

The Phase I assessment of circuit breakers and relays has been concluded, and the report documenting the results is under review. A Phase II evaluation of circuit breakers and relays is planned (an RFP has been issued). Phase II will review inspection, surveillance, and monitoring methods that can and should be applied to specific nuclear power plant circuit breakers and relays. The role of maintenance in achieving circuit breaker and relay reliability in nuclear safety systems will also be evaluated.

C-2.4 Motor Control Centers

A Phase I aging assessment is being performed of motor control centers (MCCs), including an investigation into the aging of each of the subcomponents that constitute an MCC. The boundary that has been defined for the MCC analysis consists of the cabinet enclosing the various devices. Because the subcomponents of MCCs can vary, depending on the particular application, the aging assessment includes a number of devices that were identified during the review of various failure occurrences described in the literature. Each subcomponent identified within the defined MCC boundary is then analyzed according to the general NPAR strategy.

The Phase I research for MCCs includes an assessment of the current methods used by industry for inspection, surveillance, and monitoring of MCC performance. This effort has included a trip to the Square D Company for discussions with manufacturing personnel on the methods currently employed for evaluating the operating performance of MCCs and their various components. In addition, a plant tour and review of the manufacturing and quality control inspection methods were completed. It is expected that visits to other manufacturing facilities will provide further data for evaluating inspection, surveillance, and monitoring methods.

Additional information in this area, and in the evaluation of maintenance practices, will be obtained through a survey of various utilities. The results of this Phase I evaluation will include recommendations on potential performance parameters and functional indicators that could be monitored throughout the life of an MCC and be useful in detecting aging degradation. These recommendations will be reviewed and verified in detail during the Phase II work.

C-2.5 Residual Heat Removal and Component Cooling Water Systems

A fifth element in the BNL scope is the Phase I aging evaluation of the residual heat removal (RHR) system and the component cooling water (CCW) system.

The boiling water reactor (BWR) residual heat removal system is a dual purpose system that is used for: (a) removing residual or decay heat from

the reactor by transferring the heat to service water in RHR heat exchangers, and (b) injecting water at low pressure and high volume into the reactor in the event of a loss-of-coolant accident (LOCA) (Low Pressure Coolant Injection-LPCI mode of RHR).

The pressurized water reactor (PWR) component cooling water system is a cooling water system used at PWRs to cool a variety of reactor plant components such as the reactor coolant pumps, RHR heat exchangers, ECCS pumps, and letdown heat exchangers. It is a pure water system that is, in turn, cooled by service water.

The impact of component failures on plant system performance is being evaluated using results from the component level studies and work performed by all pertinent NRC contractors for systems data assessment and systems level risk analysis. The study is performing indepth systems level failure data reviews, reviews of 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 will be addressed. The project will integrate its products with other BNL programs on reliability and performance indicators. The first is the Operational Safety Reliability Research Project. The overall purpose of this project is to evaluate the effectiveness of reliability program elements applicable to the safety of operating reactors and to identify the attributes of successful reliability programs through case studies and trial use. The results of this research will provide a technical basis for evaluation of reliability program elements that may be proposed by licensees in trade for less prescriptive regulatory requirements. The second is the Performance Indicator Project. This project is investigating risk-based indicators to monitor plant safety. The purpose of risk-based indicators is to monitor plant safety. Safety is determined by monitoring the potential for core melt (core melt frequency) and the public risk. Limits can be set at levels of safety consistent with NRC safety goals. Furthermore, the performance of safety systems, support systems, major components, and initiating events are monitored using proper measures (e.g., unavailability, failure or occurrence frequency).

C-3 NPAR PROGRAM--INEL ACTIVITIES

Several studies have already been completed at INEL in support of the NPAR Program effort. A preliminary aging assessment of batteries, cables, connectors, terminal blocks, and penetrations was completed. In this assessment, the materials susceptible to aging, stressors and environmental envelopes, failure modes and causes, functional indicators, and current IS&MM practices were identified and evaluated. The INEL has also completed an early evaluation of the susceptibility of materials in pressure, temperature, and level-sensing systems to aging degradation.

In situ electrical measurements have been made on the plant safety systems at the Shippingport Atomic Power Station before plant decommissioning. Using the EG&G Idaho-developed electrical circuit

Characterization and diagnostics (ECCAD) system, five circuit types were evaluated: pressurizer heaters, control rod position indicators, various primary system RTDs, motor-operated valves, and nuclear instrumentation. The test results are documented in Reference C-6. The remaining assessment effort on these components is being conducted as part of the evaluation of selected systems.

Finally, an aging assessment and defect characterization was made of selected valves from the Shippingport Atomic Power Station. This task was a joint activity between the NPAR Program and the Equipment Qualification Program. In this task, two valves were examined, refurbished, and operationally tested. One was an 8-inch diameter motor-operated gate valve in service for 25 years. The other was a 2-inch diameter motor-operated globe valve in service for 5 years as a high-pressure injection pump throttle valve. The results of this effort are being documented in a NUREG/CR report.

A review of the Standard Review Plan was initiated to identify any age-related technical issues. The critical review includes Chapters 3 to 10. The evaluation is concentrated on the operating experience gained to date. The remaining work on this task is included in the current effort.

The key results of the INEL effort are being documented in a series of NUREG/CR reports. Several reports have been issued to date (Refs. C-6 through C-11).

Currently, research on three major NPAR tasks is being conducted at the INEL. The three tasks are: (a) Task 1, Evaluation of Aging Contribution to Plant Safety; (b) Task II, In-Depth Engineering Studies of Selected Systems; and (c) Task III, Residual Life Assessments of Mechanical Components and Structures.

C-3.1 Evaluation of Aging Contribution to Plant Safety

The work in this task is being performed to identify where there are risk and safety concerns due to aging degradation of components, systems, and structures.

This task belongs to the special topics category and departs from the usual NPAR phased approach. The approach followed in this task is to first review operating experience and then develop a model that incorporates aging effects on system and reactor risk analysis. Using these models, an evaluation is made to identify in what systems and specific components aging has a significant effect on risk. Once the systems and components are identified, recommendations are made for indepth engineering studies.

In the first part of this task, an evaluation is made to determine the extent aging has affected LWR safety system performance based on the operating experience contained in the Nuclear Plant Reliability Data System (NPRDS) data base. The systems being evaluated are listed in Table C-2. The evaluation efforts completed through FY 1986 are indicated in parentheses.

TABLE C-2. System analysis status.

PWR Safety Systems (For additional systems to be studied in FY 1987, priority will be placed on Babcock and Wilcox (B&W) plants.)

1. Reactor Protection Trip System (completed Westinghouse)
2. Chemical and Volume Control System (completed B&W)
3. Engineered Safety Features Actuation System
4. Residual Heat Removal System
5. Power Conversion System
6. Emergency Core Cooling System

    a. High Pressure Injection and Recirculation (completed Westinghouse and B&W)
    b. Low Pressure Injection and Recirculation

7. Auxiliary Feedwater System (completed Westinghouse and B&W)
8. Pressure Control System (e.g., Power-Operated Relief Valves)
9. Safety-Related Reactivity Control System

PWR Support Safety Systems

1. Class 1E Electrical Power Distribution System (completed Westinghouse and B&W)
2. Service Water System (completed Westinghouse and B&W)
3. Component Cooling Water System (completed B&W)

BWR Safety Systems

1. Reactor Protection System (completed)
2. Standby Liquid Control System (completed)
3. Engineered Safety Feature Actuation System
4. Emergency Core Cooling Systems

    a. Coolant Injection--High and Low Pressure (completed RHR)
    b. Automatic Depressurization
    c. Core Spray--High and Low Pressure

5. Reactor Core Isolation Cooling System
6. Pressure Control System
7. Safety-Related Reactivity Control System

BWR Support Safety Systems

1. Class 1E Power Distribution System (completed)
2. Service Water System (completed)
3. Component Cooling Water System (completed)

For these systems, the component failure contributions to five major categories (the "broad brush" analysis) are to be determined. These categories are: (a) aging and service wear, (b) design and installation, (c) testing and maintenance, (d) human related, and (e) other. This will identify component/failure category combinations that will be used to establish relative impacts on system unavailability. Additionally, events in these failure categories that resulted in initiating a transient or accident, loss of system function, loss of redundancy, or degradation of system availability are to be identified. The information gathered should include, where obtainable, age of the plant involved, system, component type, and inservice age of the component. This information will be cataloged and filed for future evaluation. The results of this effort are documented in NUREG/CR-4747 (Ref. C-8). This report includes categorized failure results by systems and components for 5-year increments of service age.

A second part of this task is developing aging models to provide quantitative determination of the effect aging has on plant safety. In addition to developing aging models, a data base of aging-related failure data is being developed to provide aging root cause information for various systems. In FY 1986, BWR service water systems and some PWR Class 1E power distribution systems were analyzed. The work remaining includes root cause evaluation of the Westinghouse auxiliary feedwater system and high pressure injection system.

With the aging models and the aging-related root cause data, an evaluation will be made to assess the relative effects aging has on reactor systems. This analysis will be made on the plant/systems models used in the Surry PRA performed for NUREG-1150. This evaluation will provide a best estimate of the risk and safety implications of aging using the time-dependent models and best available failure data. Using the Surry models will also provide a direct comparison between current PRA results and the time-dependent aging models.

C-3.2 Indepth Engineering Studies of Selected Systems

The indepth engineering studies of systems are being conducted in accordance with the NPAR phased approach to research described in NUREG-1144. The systems of interest for this task are those that provide the protection function for the major boundaries to the release of radioactivity (RCS, containment). Certain Class 1E power components also will be studied as subsystems to determine their importance to safety and as part of the systems of interest. The research results emanating from this task will provide at the system level:

- Identification of aging degradation effects.
- Identification of IS&MM to trend degradation.
- Identification of methods to mitigate degradation (maintenance and replacement).

- Recommendations for modifications to appropriate codes, standards, and regulatory guides.

Three systems are currently being evaluated in this task: the high pressure emergency core cooling system (HP-ECCS), the in-plant Class 1E power distribution system, and the reactor protection system (RPS).

A Phase I evaluation of HP-ECCS is being performed. The evaluation planned for this system includes an evaluation of operating experience (which will incorporate the results already obtained in part of Task 1), a Phase I aging assessment, and a review and recommendation for IS&MM. The Phase I aging assessment of the HP-ECCS also includes in-plant aging studies at the Oconee-3 nuclear power plant. Evaluations are being made of internal event and transient assessment reports, inservice inspection and testing records, performance testing and station modification records, nuclear maintenance and equipment data bases, and results of trend analysis and predictive maintenance programs. The results of the Phase I HP-ECCS evaluation will be documented in a NUREG/CR report.

The second system being evaluated in Task II is the Class 1E power distribution system. A Phase I evaluation is in progress following the NPAR strategy. The work being performed includes an evaluation of operating performance to determine what aging-related problems have been experienced and a Phase I aging assessment, which will include a review, evaluation, and recommendations for IS&MM. Also included in the evaluation of this system is an in-plant study of the Class 1E power distribution system at Oconee-3. A three-part study will be performed of the: (a) dc power system, (b) vital instrument power subsystem, and (c) high voltage power subsystem. This work will be documented in a NUREG/CR report.

A third system currently being assessed is the RPS. A Phase II evaluation is being performed in this subtask. The work being performed consists of a comprehensive Phase II aging assessment, including postservice examinations of naturally aged equipment. A significant effort is planned here on evaluating potential performance indicators for the RPS.

With the completion of the Phase II evaluation, application guidelines will be prepared for surveillance and maintenance and for condition determination (to ensure operational readiness of aged RPS). Recommendations for modifications in testing and inspection methods will be made.

The Task II effort includes plans for future studies of additional systems. Phase II evaluations are projected for the HP-ECCS and Class 1E power distribution systems. This work will include evaluation of performance indicators for aging and service wear effects, advanced methods for IS&MM, and the development of application guidelines.

C-3.3 Residual Life Assessment of Major Components

An evaluation of the age degradation and residual life of major LWR components is being performed at the INEL for the NPAR Program. The NPAR

effort is safety oriented and is complementary to the ongoing industry-sponsored pilot projects on plant life extension.

The approach used in the residual life assessment task is to first identify and prioritize the major components with respect to safe plant operation. This is followed by a Phase I effort to establish the life-limiting processes for each of the major components. Included here is the identification of degradation sites and failure modes during normal operation and accident conditions.

The Phase I effort also includes an assessment of current and potential inspection, surveillance, and monitoring methods. For this phase of the effort, the work is focused on integrating currently available technical information that is relevant to aging and life extension. This is mainly information that has been generated or is now being generated by other NRC and industry programs.

The results from the initial assessment of life-limiting processes will then be used in developing simple mechanistic models for determining the residual life of selected major components. Developing these models is of particular interest for components that are not readily accessible for routine maintenance and inspection.

As these models are developed, residual life evaluations are planned for the major components using actual plant operating data. In this phase of the project, key plant operating data will be identified. These are the operating data that are necessary for a realistic (rather than an enveloping design or conservative) estimate of the mechanical and thermal loading of the components as well as other environmental stressors.

With the completion of the Phase I evaluation effort, the results will be reviewed and areas where additional research is needed will be identified. Where additional work is required, recommendations will be made for Phase II research.

As of FY 1987, the major components important to plant safety have been identified and prioritized. An initial evaluation has been made of five PWR components: the containment, pressure vessel, primary piping, steam generator, and vessel support; and of three BWR components: the pressure vessel, recirculation piping, and vessel supports. In this evaluation, the degradation sites, degradation mechanisms, stressors, and failure modes have been identified. This evaluation also includes a review of the current methods used for inspection and surveillance of these components. The results of this effort have been documented in NUREG/CR-4731 (Ref. C-11).

The identified scope of work that remains is as follows. The initial evaluation of the remaining components will be completed and a more detailed investigation of selected major components, such as BWR and PWR containments, will be undertaken.

A detailed assessment of current and emerging inspection surveillance and monitoring methods will be performed. This will include methods being developed for use in nuclear power plants and methods being developed in allied industries such as fossil power, chemical plants, and aerospace. Recommendations will be made for additional research or engineering qualification efforts for newly developed techniques.

C-4 NPAR PROGRAM--ORNL ACTIVITIES

The NPAR Program is sponsoring research on four major tasks at ORNL. The first task is evaluating the auxiliary feedwater system. The other three tasks involve aging assessments of various components. The results have been documented in References C-12 through C-15.

C-4.1 Aging Assessment and Analysis of Auxiliary Feedwater System

This task includes the aging assessment and analysis of the auxiliary feedwater system (AFWS). The main objective of this task is to apply the NPAR Program strategy and provide recommendations and guidelines for inspection, surveillance, monitoring, and maintenance of the AFWS. Phase I of the task on the AFWS includes evaluating operating experience, determining aging impact upon its operability, reviewing inspection, surveillance, and condition monitoring methods, and evaluating the role of maintenance practices in counteracting aging. ORNL is planning to perform the Phase I study in conjunction with a cooperating utility. The Phase I study will provide recommendations for comprehensive aging assessment of the AFWS to be followed in Phase II study.

C-4.2 Aging Assessment and Analysis of Motor-Operated Valves, Check Valves, AFW Pumps, and Solenoid-Operated Valves. Evaluation of Condition Monitoring Methods for Electrical Cables, Pressure Transmitters, and Multistage Switches

The second task includes the aging assessment and analysis of motor-operated valves, check valves, AFW pumps, and solenoid-operated valves, and evaluation of Condition Monitoring Methods for electrical cables, pressure transmitters, and multistage switches used in nuclear power plants. Phase I of this task is complete, and ORNL is performing a comprehensive Phase II aging assessment. In a separate task, detailed evaluations of the role of maintenance in counteracting aging effects in selected valves will be done. Here, the relative benefits of various predictive, preventive, and corrective maintenance practices will be evaluated and improper maintenance practices causing valve degradation identified. The evaluation of motor-operated valves was completed in FY 1986 (Ref. C-14).

C-4.3 Testing of Naturally Aged Solenoid Valves

Testing of naturally and artificially aged solenoid valves is being performed using IEEE 382 and 323 as a guide. Thermal aging and cycling will be used for artificially aged valves. These valves will be subjected to gamma radiation to simulate the accident exposure and then put through a

30-day LOCA test. The valves will be functionally and electrically tested and physically examined at the end of each phase of testing to determine the degree of degradation.

C-4.4 Diagnostics and Monitoring of Reactor Internals--Structural Integrity

In-core and ex-core neutron noise monitoring is being evaluated to detect degradation in PWR reactor vessel internals. The results of this task will provide input in revising a standard for the use of neutron noise to monitor core barrel vibrations and preparing a standard to monitor loose parts. This task will attempt to predict the effects of various types of degradation on the noise and vibration signatures.

Oak Ridge National Laboratory has analyzed ex-core neutron detector noise data to determine the feasibility of detecting incipient thermal shield degradation in two domestic PWR reactor pressure vessels. Results of the noise data analysis indicate that thermal shield support degradation probably began early in the life of both plants. The degradation was characterized by the appearance of new resonances in the ex-core neutron detector noise. This study shows that the neutron noise analysis program can be used to monitor degradation of reactor internals.

C-5 NPAR PROGRAM--OTHER ACTIVITIES

The National Bureau of Standards (NBS), Systems Engineering Associates (SEA), and Sandia National Laboratories (SNL) are conducting research studies and providing support for the NPAR Program.

C-5.1 NBS Study

The NBS is conducting an independent review of the techniques that have been used for in situ testing of electrical cables inside the containment.

The techniques being evaluated are the diagnostic methods and measurement approaches used for detecting incipient defects due to the aging of both electrical and mechanical components in plant safety systems.

The NBS investigation focuses on the aging of Class 1E electrical cables in nuclear power plants. Previous work has identified three characteristic failure modes of concern in nuclear power plant applications: dielectric failure, localized changes in characteristic impedance, and localized increase in the resistance of the conductor.

While there may be other failure modes for the electrical cables in question, the failure modes listed underscore the fact that the failure criteria are diverse. It is unlikely that a single simple test can be used to evaluate all relevant properties of electrical cables. It is also true that the physics and chemistry of aging are not well enough understood to permit a unique and unambiguous identification of incipient failures of cables inside containment. This evaluation program at NBS will provide a

technical focus for basic investigations into aging and failure mechanisms, and will provide NRC a capability for an independent evaluation of measurement methods and approaches for detecting defects in aged cable systems inside containment, during normal operating life and during extended life.

C-5.2 SEA Study

The research effort by SEA is aimed at evaluating the consequences aged components have on vital light water reactor (LWR) systems and the resulting effect on plant safety.

The Phase I effort, "Method to Analyze and Understand Aging Effects," has demonstrated the application of the N-square diagram modeling of the systems interactions to identify components and parts within components with aging significance. The method involves proper characterization of the functional and spatial systems interactions and information pertaining to:

- The relationship (and effect) of parts to components performance.
- The relationship (and effect) of systems to plant performance.
- The effect of aging and service stresses at the part and component level.
- Critical specification parameters that must be maintained and are affected by age and/or service stress.
- Presentation of simultaneously occurring interactions for evaluation.

The current research effort is directed toward:

- Applying system interaction model procedures, developed in Phase I, to selected safety systems and support systems.
- Investigating the systems ability to mitigate effects of aging leading to common-mode failures.
- Identifying components and parts that have propensity for aging degradation.
- Generating recommendations for maintenance of the systems to alleviate aging concerns.

The systems interaction model procedures are being applied to the following PWR and BWR fluid-mechanical and electrical safety systems.

- PWR Safety Systems and Support Systems
  - High-Pressure Emergency Core Cooling System

- Class 1E Electrical Power Distribution System
- Service Water System
- Auxiliary Feedwater System

- BWR Safety Systems

  - Low-Pressure Emergency Core Cooling System
  - Reactor Protection System

C-5.3 SNL Study

Sandia National Laboratories is conducting an assessment of Class 1E electric cables for aging and their qualification for life extension. This assessment will focus on: (a) determining how the various monitoring indicators for cable aging change with time; and (b) determining by LOCA test whether aged cables can be shown to be qualified for life extension (beyond 40 years).

Selected Class 1E qualified cables, representative of those currently used in nuclear power plants for safety-related functions, will be used in the research. The degree of cable degradation will be determined, by measurement, for various usage periods (i.e., 20, 40, and 60 years). The cables will be artificially aged using a long time period, temperatures based on the Arrhenius Theory, and a reasonably low dose rate (to ensure that dose rate effects are properly considered). The program tasks include:

- Reviewing lists of Class 1E electric cables (EPRs, polyelefines, etc.) in general use for safety-related functions and recommending to NRC a representative selection of cables for testing.

- Developing an aging test plan. This plan will be based on using standard test spools, around which selected cables can be wrapped and aged, to perform a LOCA qualification test. The LOCA test would be run for cables that have been aged to the equivalent of 20, 40, and 60 years. The test results will be used to evaluate the cables' survival and qualified life as a function of age. Also, the test results will be studied for evidence of potential problems that might exist if cable use were extended beyond the normal 40-year (qualified) life.

- Developing a method of and performing an accelerated aging test of the cables on the test spool frames. The aging period should be about 6 months and use the temperature equivalent and radiation dose equivalent of power plant operation of 20, 40, and 60 years.

- Periodically testing the properties of short lengths of the same aged cable materials on the spool frames. The cable samples will be removed from the aging program at selected intervals to measure cable degradation over the equivalent 60-year aging period. Mechanical properties will be measured, including

tensile strength, elongation, hardness, and density. Electrical property measurements will be made for capacitance, resistance, and voltage discharge. The sampling periods will be at approximately 10-year increments of age. The degradation measurements will be plotted and correlated. Based on the analyses of the data, recommendations will be made for actual in-plant cable monitoring measurements to assess cable aging in nuclear power plants.

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APPENDIX D
ONGOING PROGRAMS RELATED TO NPAR

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ONGOING PROGRAMS RELATED TO NPAR

D-1 INTRODUCTION

Aging of nuclear power plant components has a significant impact on plant safety and economy. Aging may reduce the safety margins for critical components, structures, and safety systems, and thus compromise the defense-in-depth concept. Therefore, the U.S. Nuclear Regulatory Commission (NRC) and regulatory agencies in foreign countries are sponsoring research and development programs to evaluate the impact of aging on the safe operation of nuclear power plants. The nuclear industry, including EPRI, NSSS vendors, utilities, and architect engineers, is interested in aging because of the significant economic advantage that can be derived from extending the operating license (OL) of the aged power plant. Therefore, the nuclear industry, and especially EPRI/DOE, is sponsoring pilot studies to evaluate the potential of plant life extension (PLEX), i.e., OL extension, for a typical BWR and PWR plant.

There are several NRC and industry-sponsored aging-related programs that are currently being carried out. Many of these programs are complementary to each other and, therefore, it is essential that they are coordinated. Coordinating these programs will eliminate duplication of efforts and provide a more complete set of aging-related results in a timely and cost-efficient manner. This coordination and integration effort has a special significance for NRC because it is expected that the criteria related to license extension will be required by the early 1990s.

The Executive Director for Operations (EDO) has established a Technical Integration Review Group for Aging and Life Extension (TIRGALEX) to ensure effective utilization of NRC resources. TIRGALEX has recognized the importance of the coordination effort and conducted a preliminary review of relevant programs and activities already under way.

Interfaces between the NPAR Program and other ongoing NRC programs have been established and will be maintained. Similarly, external programs involving both domestic and foreign organizations have been contacted. This section outlines the aging-related programs sponsored by NRC. It also outlines the similar programs sponsored by industry in the United States, Japan, and Europe. The outline of these programs is structured into following six subsections:

1. NRC
2. EPRI
3. Industry
4. DOE
5. Codes and Standards Committees

6. Foreign countries.

The list of age-related programs external to NRC will be updated periodically. The updating will be made as information exchanges take place and information for new programs is made available to the NPAR Program.

D-2 ONGOING AGING AND LIFE EXTENSION PROGRAMS AND ACTIVITIES AT NRC

In this section, a summary of ongoing NRC aging and life extension related programs and activities is presented. Programs that have ended (and, therefore, are not ongoing) are not included. New program starts for FY 1987 are not covered because information on these programs was fragmentary.

D-2.1 Office for Analysis and Evaluation of Operational Data (AEOD)

The AEOD collects, screens, evaluates, and reports on operational experience and data from the entire nuclear industry. A natural byproduct of this process is identifying components and systems with repetitive, unusual, or possibly generic failure modes, some of which are aging related.

The AEOD also conducts evaluations of components, systems, procedures, and management structures in response to specific operational events. In performing these evaluations, AEOD is concerned with acquiring and analyzing data that lead to identifying the root cause(s) of the events. The AEOD does not specifically focus on aging issues per se, unless the incident involves failures or degradations due to aging mechanisms. When an event indicates aging degradation, the information is forwarded to the appropriate NRC office for action. The AEOD then publishes case studies that contain detailed descriptions of the failure parameters, failure causes, and engineering evaluations.

The AEOD has developed, through the Oak Ridge National Laboratory (ORNL), a comprehensive computerized data base to aid in collecting and evaluating licensee event reports (LERs). This data base, the Sequence Coding and Search System (SCSS), contains established procedures and codes for collecting operational data. The SCSS also provides a means of storing and rapidly retrieving root-failure-cause data that can be used in studies of component life and component aging. It also has resulted in codes specifically identified for aging degradation.

The AEOD has an ongoing project to analyze the Nuclear Plant Reliability Data System (NPRDS) data base. In concert with the Institute of Nuclear Power Operations (INPO), which maintains the data base, AEOD has developed a list of critical components on which to focus attention. The trends and patterns analysis of NPRDS data focuses on these key components. The AEOD's analysis of NPRDS data results in statistical and engineering evaluations of component failure modes, times to failure, operating conditions that affect failure, and chemical and physical conditions affecting component-wearout rates.

One example of a recently completed case study conducted by AEOD involving aging mechanisms concerned motor-operated valve performance in response to the Davis-Besse event. The aging mechanisms highlighted in this case study were (a) inadequate protection devices, which allowed equipment to operate beyond its design limit and suffer accelerated aging, and (b) inadequate design features for human factors, which also accelerated the aging process. The study identified surveillance testing and maintenance as mitigative and corrective measures for the aging mechanism. The study also indicated that developing and implementing of a signature-tracing technique (based upon the results provided by the RES/NPAR program) to monitor motor-operator parameters during actuation would provide better mitigation of this aging mechanism.

D-2.2 Office of Nuclear Reactor Regulation (NRR)

A substantial number of aging-related programs are in progress in NRR. They are found in several of the broad program categories that NRR tracks: Operating Reactors, Casework, and Safety Technology. The major aging-related programs are discussed below.

Operating Reactors. The NRR is responsible for licensing actions and safety assessments of both currently operating reactors and new reactors coming on line. While aging concerns are built into the licensing process, to some extent, the licensing process is not geared toward characterizing the aging process as it occurs or as it might exist at the time of a license renewal request. The capability of equipment to perform satisfactorily for its specified lifetime is NRR's principal area of attention.

Casework. This is one of the larger program categories in NRR and represents projects that are conducted to support individual licensing actions. These projects may be initiated by licensee requests for license amendments or by events that occur at the plants that may require a regulatory response from the NRC. Most aging-related projects are included in this program category.

Safety Technology. The NRR divides this program category into five subgroups: unresolved and generic safety issues; risk assessment; regulatory requirements; code analysis and maintenance; and human factors program issues. Some of these programs have aging-related aspects. For example, unresolved and generic safety issues are sometimes related to aging issues, e.g., pressurized thermal shock. Projects in risk assessment are not specifically related to aging; it is recognized that the ability of probabilistic risk assessments (PRAs) to model the effects of aging is limited at present.

Within the human factors category, the crucial role of maintenance in predicting and correcting aging degradation has been reflected in the Maintenance and Surveillance Program. Recently, Phase I of the Maintenance and Surveillance Program (NUREG-1212) was completed and reviewed by the Executive Director of Operations (EDO). Phase II is currently under way. Phase I was designed to survey current maintenance practices in the U.S.

nuclear industry and to evaluate their effectiveness. Phase II is working toward resolutions of the issues identified and assessed in Phase I.

Several NRR programs guide ongoing regional activities relevant to aging, aging detection, and mitigation of aging consequences. These programs include the Safety System Functional Inspection Program, the Safety System Outage Modifications Inspection Program, and the Generic Communication Program.

The Safety System Functional Inspection Program, in general, assesses whether plant modifications of selected safety systems have degraded the design margin to the point where the system's ability to mitigate design basis events is impaired. This program consists of an indepth review of a small number of safety systems and is usually conducted at older plants. The major objectives of the program are to ensure that:

- Safety systems are capable of performing the safety functions required by their design bases.
- Testing is adequate to demonstrate that the systems would perform all of the required safety functions.
- System maintenance (with emphasis on pumps and valves) is adequate to ensure system operability under postulated accident conditions.
- Operator and maintenance technician training is adequate to ensure proper operations and maintenance of the system.
- Human factors considerations relating to systems and supporting procedures are adequate to ensure proper system operation under normal and accident conditions.

The objectives of the Safety System Outage Modifications Inspection Program are to verify, through sampling inspections, that:

- Licensees have effective controls for conducting modification and repair activities during outages.
- Activities are accomplished in accordance with established procedures and commitments.
- Completed repairs and modifications have been properly designed, installed, inspected, and tested.
- Affected systems are ready for safe startup and operation of the plant following an outage.

The objectives of the Generic Communication Program are to:

- Inform licensees of problems, including those due to aging and wear, that have developed in individual plants.

- Require action when these problems are shown to be significant and generic.

These three programs apply to the pressure boundary hardware, drivers and actuators, electrical power, and the instrumentation and controls of engineered safety features.

The NRR also guides the activities of the regions by issuing the Inspection and Enforcement Manual. Portions of this manual establish inspection procedures that are relevant to aging and life extension. For example, some inspection procedures establish guidance for ascertaining that inservice inspection and testing activities are programmed, planned, conducted, recorded, and reported in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Where applicable, these procedures prescribe inspection of the licensee's recordkeeping for modification, maintenance, and repair activities.

An ongoing routine inspection effort is conducted by the regional offices in accordance with the NRR inspection program. The objective is to ensure that systems and components have not been measurably degraded as a result of any cause, including aging. The components selected for specific inspection efforts are containment, instruments and components (specifically, breakers and relays), and reactor coolant system piping. The programs in the component area are directed at breakers and relays that may cause anticipated transients without scrams and/or loss of safety-related equipment. The containment programs address corrosion, fatigue, and crack propagation in steel, reinforced-concrete, and prestressed-concrete containments. The reactor coolant system piping program addresses stress-corrosion cracking of welds, and erosion and general corrosion of pipe bends and elbows.

D-2.3 Office of Nuclear Regulatory Research (RES)

RES sponsors a number of large and important aging-related programs. The general objectives of the program are: identifying aging mechanisms, evaluating their safety and regulatory impacts, assessing detection methods for particular aging degradation mechanisms, and developing mitigating and corrective actions.

Major program areas with aging-related projects include Mechanical and Electrical Equipment Qualification; Instrument Integrity; Primary System Integrity, which includes the Heavy Section Steel Technology (HSST) program, Degraded Piping, and Steam Generator Integrity projects; Equipment Operation and Integrity, which includes the Nuclear Plant Aging Research (NPAR) Program; and Nondestructive Examination.

RES has initiated an activity, called License Renewal Policy Development, whose objective is to formulate and prepare methods of resolution for PLEX-related policy issues.

D-2.4 Office of Nuclear Material Safety and Safeguards (NMSS)

At the present time, NMSS does not have any specific programs that are considered to be pertinent to reactor aging and life extension. Although NMSS work on waste management does involve aging considerations, these are quite different from those that pertain to operating reactors. The progress and success of the DOE waste repository program could have an institutional impact on extension of plant life. However, this aspect does not appear to be a matter that should be of concern at this time. From the safeguards standpoint, the instrumentation and equipment associated with physical security systems for reactors are subjected to routine tests, surveillance, and maintenance resulting in repair or replacement to ensure performance such that aging is not an important issue. With respect to transportation casks for spent fuel, no data or studies were identified that might be pertinent to plant aging/life extension. Fuel fabrication facilities involve only unirradiated fuel and, therefore, pose no significant risk. The components and systems associated with these facilities are far less complex, and they are routinely maintained, refurbished, and replaced. Also, frequent opportunities exist for review of the status of the facilities by the 5-year license term. The NMSS will continue its participation in TIRGALEX to report on any further NMSS programs that may relate to reactor aging issues and to maintain cognizance of ongoing aging and life extension work that may have benefits to NMSS licensing activities.

D-2.5 Technical Integration Review Group for Aging and Life Extension (TIRGALEX)

A technical integration and review group for aging and life extension (TIRGALEX) was established by the Executive Director for Operations in April 1986. TIRGALEX was established to provide overall planning and integration of NRC activities related to plant aging and life extension. The specific objectives of TIRGALEX are to:

1. Clearly define the technical safety and regulatory policy issues associated with plant aging and life extension.

2. Develop a plan for ensuring resolution of the issues in a timely, well-integrated, and efficient manner.

To meet these objectives, TIRGALEX focused on three main topics:

1. Defining the major technical safety and regulatory policy issues associated with plant aging and life extension.

2. Conducting a review of ongoing programs and activities related to plant aging and life extension.

3. Formulating recommendations based on an appraisal of the issues and the programs and activities already under way.

An important element of the TIRGALEX charter was to ensure that all ongoing NRC technical activities and data related to aging and license renewal are effectively integrated. Coordination and communication with external organizations are an important part of this integration function. To effectively implement this activity, TIRGALEX has recommended that RES with the support of the other offices take the lead role for external communication on technical data relative to aging and license renewal.

The technical data for aging and life extension provide one of the inputs for the prioritization of aging and license renewal technical safety issues. Other key inputs are the regulatory user needs to be supplied by user organizations such as the regional offices and NRR. Once the data and research results have been generated, the utilization of the technical information in the regulatory process will become the responsibility of the user organizations.

D-3 ONGOING AGING AND LIFE EXTENSION PROGRAMS AT EPRI

The Electric Power Research Institute (EPRI) has a number of programs related to aging. Examples include:

1. Nuclear Plant Life Extension and Construction

2. Corrosion Control

3. Component Reliability.

The Nuclear Plant Life Extension and Construction Program approaches aging and life extension from the technological perspective of understanding the aging process for systems and components. It does not address institutional or legal issues. Over the next few years, EPRI plans to give increased attention to component aging and equipment preservation. The Corrosion Control Program addresses environmentally caused cracking and pitting. It specifically emphasizes boiling water reactor (BWR) water chemistry and understanding and mitigating pipe cracking due to corrosion. The Component Reliability Program is related to, but more general than, the Corrosion Control Program. Structural reliability and safety improvement are the main thrusts of the Component Reliability Program. Specific examples of activities include material characterization, flaw detection and assessment, inspection hardware development, and reliability methodology. While the central program focus is materials, the program also addresses piping, reactor pressure vessel, and steam generator tubes.

Two other programs, the Plant Availability Program and the Safety Control and Testing Program, indirectly address aging concerns. Both address or are related to on-line monitoring, plant availability, and plant performance. Diagnostic systems, human engineering, and the man-machine interface form the nucleus of these programs.

The EPRI also co-sponsors a joint program with industry and DOE to address life extension issues. This program is discussed in more detail in Section D-4.

D-4 ONGOING AGING AND LIFE EXTENSION PROGRAMS SPONSORED BY INDUSTRY AND DOE

A joint industry, EPRI, and DOE program was initiated in 1984 to identify issues associated with light water reactor (LWR) life extension. In 1984, DOE and EPRI agreed to co-fund studies and developed a joint R&D plan in 1985. Two pilot studies involving Surry 1 (PWR) and Monticello (BWR) were initiated. An AIF/National Environmental Studies Project regulatory study was also started.

Results, to date, of the studies were presented at a seminar August 25 to 27, 1986, in Alexandria, Virginia. The presentations delineated the roles of program participants, as follows:

- LWR Life Extension Utility Steering Committee. This group's primary objective is to preserve and enhance the option for extending the life of its members' nuclear power plants beyond the initial licensed lifetime of 40 years. To achieve this objective, the Committee must: (a) help ensure the availability of technology and methods to enable individual owners to make informed decisions about life extension, and (b) assist in establishing a stable and predictable regulatory environment for securing renewed licenses from the NRC. The Steering Committee will oversee the work of three subcommittees, ensure that sufficient resources are available to complete the work, and interact with NRC on policy and management issues.

- Electric Power Research Institute (EPRI). The EPRI's primary function is to transfer technology and lessons learned to its members. It supports ongoing life extension work by co-sponsoring the pilot plant studies and providing the results of other EPRI-sponsored projects for use in the ongoing PLEX research. Several years ago, EPRI sponsored initial studies of the feasibility of extending the operating life of nuclear power plants.

- U.S. Department of Energy (DOE). The DOE is concerned with maintaining options for meeting future energy demands and views nuclear power plant life extension as a viable option that could temper the uncertainty in future power load forecasts. The DOE is co-sponsoring the pilot plant studies and performing several research projects that will enhance the life extension option. The other projects include risk assessments, cable aging evaluations, and economic assessments. The DOE has entered a cooperative agreement with EPRI to support R&D that will enhance the feasibility of extended life.

- Northern States Power (NSP). The NSP is conducting the BWR pilot plant study at its Monticello Nuclear Generating Station. This study is co-funded by EPRI and DOE.

- Virginia Power. Virginia Power is conducting the PWR pilot plant study at its Surry Unit 1 plant. This study is also co-funded by EPRI and DOE.

- Atomic Industrial Forum (AIF). The AIF has two life extension programs in progress. The first program, sponsored by the AIF National Environmental Studies Project (AIF/NESP), is a study to determine the regulatory issues that will affect life extension. The results of that study will be published later this year. The AIF Life Extension Subcommittee was created in 1986 to review the second program on licensing issues.

- ASME Section XI Special Subcommittee. This subcommittee has appointed a special working group on nuclear plant life extension (SWG-PLEX) to recommend additions and modifications to ASME Section XI to cover special problems related to extending the life of nuclear power plants. The SWG-PLEX also will coordinate the activities of other codes and standards groups in preparing the ASME Section XI changes and additions to accommodate life extension.

- IEEE PLEX Subcommittee. This group has appointed a working group to investigate the codes and standards aspects of nuclear plant life extension related to electrical and instrumentation and control equipment. The working group has developed an action plan and plans to publish a preliminary report on these issues by early 1988.

Preliminary conclusions of the studies presented in Alexandria, Virginia, in August 1986, are summarized as follows:

- While technical problems were identified, none were felt to be life limiting. With proper understanding of aging and degradation processes, component replacement or effective surveillance and maintenance would resolve the issues.

- There is a need for improved tests, inspections, surveillance procedures, recordkeeping, and maintenance practices to alleviate aging concerns and ensure 40-year plant life extension.

- Specific areas of possible R&D were identified, but no decisions on implementation have yet been made.

D-5 ONGOING AGING AND LIFE EXTENSION PROGRAMS AT DOE

Plant life extension is being pursued by DOE through its nuclear energy organizational element. One of three major efforts within the LWR safety area is the cooperative EPRI/industry/DOE program to extend the productivity of existing and future LWRs. In FY 1987, DOE plans to concentrate on delineating the key LWR structures and components capable of continued service beyond their design life without refurbishment or replacement. Further, DOE intends to evaluate items requiring replacement

in terms of safety significance, licensing, and cost implications. The DOE programs are intended to support relicensing by influencing the Federal regulatory process and evaluating the safety and economic impacts of improved plant performance.

D-6 ONGOING LIFE EXTENSION ACTIVITIES IN CODES AND STANDARDS

In general, only planning efforts have been initiated thus far with regard to life extension; no significant actions have yet been completed by the Codes and Standards Committees.

The following is a summary of status with regard to code activities related to life extension:

Institute of Electrical and Electronics Engineers (IEEE). Working Group 3.4, "Nuclear Plant Life Extension," has held four meetings. Its purpose is to investigate the codes and standards aspects of plant life extension as they pertain to electrical equipment. An action plan has been developed. A preliminary report is to be published in 1988.

Instrument Society of America (ISA). They have no current activities. They take their lead from IEEE. When relevant items arise, IEEE will made the necessary contacts.

American Society of Mechanical Engineers (ASME). Under the auspices of the ASME Section XI subcommittee, the Special Working Group on Plant Life Extension has met six times. They have active participation from EPRI, NRC, DOE, utilities, and NSSS suppliers. To date, the special working group has reviewed a broad spectrum of aging and life extension programs.

The latest activity is a request to all ASME Section XI subgroup chairmen to begin to develop changes to the code factoring in life extension. This probably will require at least 1 year to produce meaningful changes.

ASME Board of Nuclear Codes and Standards. A special coordinating committee is being established with representation from IEEE, ASME, ACI, etc. Its purpose is to direct the thrust of life extension. The first meeting may be sometime this fall.

D-7 ONGOING AGING AND LIFE EXTENSION PROGRAMS IN FOREIGN COUNTRIES

Extensive programs in materials degradation, NDE, fracture mechanics, structures, and other aging-related areas exist worldwide. A few of these have been detailed in NUREG/CR-3040.

The NRC operates large cooperative international research programs, several of which are directly related to aging and life extension. Some of these programs are listed in Table D-1.

TABLE D-1. Examples of some existing international projects.

Engineering Technology Area	Project	Participating Countries/Agencies	Total Funding $ NRC	Total Funding $ Other	Duration and Completion Date
RPV Safety	Validation of Vessel Wall Embrittlement Trends	USNRC/FRG (MPA)/UK (AEA)	215K --	1,000K 500K	1985-1987
	Fracture Mechanics	USNRC/SWI (EIR)	--	--	1984-1988
	RPV Dosimetry and Annealing	USNRC/Belgium (CEN/SCK)	--	--	1982
Piping	IPIRG	USNRC/Belgium FRA/JAP/UK/ Sweden/SWI/ Spain/Canada	1,650K	3,000K	1986-1989 (Proposed)
Equipment Qualification	Beta Irrad. of Polymers	USNRC/France	300K	300K	1985-1988
	Accelerated Aging of Elastomerics	USNRC/JAP	--	--	1986-1989 (Pending)
	Valve Behavior Under Simulated Earthquakes	USNRC/FRG	600K	--	1984-1988
Steam Generator	Steam Generator Project	USNRC/EPRI/FRA (CEA)/ITA (Ansaldo)/JAP (NUPEC)	5,000K	4,000K	1982-1987
NDE	Acoustic Emission Monitoring ZB-1 Test	USNRC/FRG (HDA)	600K	(DM)800K	1981-1985
Containment	Pretest Prediction for Concrete Model Test	USNRC/UK/ (NII)/ FRA (CEA)/ITA (ENEA)	300K	900K	1985-1987

Other programs identified in the analyses of several selected components and structures include:

- Westinghouse and Framatome--performing work separately and jointly on NDE for cast stainless steel.
- JAERI--evaluating the effects of separate and simultaneous application of environmental stressors on cables (NUREG/CP-0071).
- France--developing predictive maintenance programs.

Additional details on foreign activities were provided at an IAEA technical committee meeting held in Vienna, Austria, from September 1 to 5, 1986, on "Safety Aspects of Nuclear Power Plant Aging." Twelve countries attended: Canada, Czechoslovakia, Finland, France, West Germany, Italy, Pakistan, Sweden, Switzerland, the United Kingdom, the United States of America, and Yugoslavia.

Of the 12 participants, West Germany, Canada, France, Italy, and the U.S. have aging programs, although the scope of the programs varies widely. Japan has a program plan for nuclear plant life extension R&D, but it was not discussed at the IAEA meeting. The following is a summary of the aging-related programs of West Germany, Canada, France and Italy as well as Japan's R&D program.

WEST GERMANY

The GRS has been engaged, on behalf of the Federal Minister of the Environment, Nature Conservation, and Regulatory Safety, in a data collection program at power plants for 15 years (since initial operations). This program has demonstrated how long-term data collection and trending of component performance parameters and functional indicators (a strategy/approach similar to that recommended in the NRC's program plan for Nuclear Plant Aging Research--NUREG-1144) provide protection against failures resulting from aging.

Aging degradation has been observed in emergency diesels, PVC power, measuring and control cables, pump motors, valves, and a variety of electric and electronic equipment. The safety and availability problems that may result from aging are counteracted by a system of inspection and planned preventive maintenance measures. The GRS has implemented a long-term systematic collection of data so that the cumulation of defects on certain components can be recognized.

CANADA

Managing aging degradation of nuclear power plant components and maintaining adequate plant safety is part of the overall Canadian approach to reactor safety. Major features of this approach include:

- The design of CANDU nuclear power plants strives to accommodate various aging effects by using appropriate design features such as diversity, physical separation, and testability.

- Operating policies and procedures prescribe practices that are designed to minimize aging effects. Components, particularly pressure-retaining components, important to safety are subject to inservice inspection.

- Deficiencies are systematically detected and recorded; causes (including aging degradation) are determined, and appropriate corrective actions are taken.

- Licensees are required to monitor and periodically evaluate equipment and system performance against the reliability targets set by the Atomic Energy Control Board (AECB), allowing detection of aging trends.

- Research and development, by both the industry and the AECB, provide the basis for predicting the behavior of critical plant components during plant operation, developing better components and developing monitoring methods capable of detecting component degradation before loss of safety function occurs.

- Aggressive monitoring is utilized, employing monitoring instruments, periodic testing, inspection, maintenance, and field patrols. It is recognized that aging degradation can be detected only if appropriate time-variant parameters are monitored.

- Significant Event Reports are prepared for events that have a significant negative effect on reactor safety, worker or public safety, and cost. This system has the following important features:

  - Recording of equipment and operation deficiencies in a specified, systematic manner to allow event review and analysis.

  - Multilevel diverse screening process.

  - Trend lines of equipment and system deficiencies.

  - Communication of lessons learned to other CANDU stations, owners, designers, and equipment manufacturers.

- Reactor Safety Reliability Assessment is an important part of the annual comprehensive and systematic review of nuclear power plant operation and maintenance by both the licensee and the AECB. It gives the actual past-year performance and the predicted future performance in terms of system unavailabilities and serious process failure occurrences. These can be compared to the AECB reliability targets mentioned earlier.

FRANCE

From the outset of the nuclear program, the French Safety Authorities and the EDF (Main Licensee) took into consideration the effects of aging on the installed equipment.

In French designs, qualification is one of the means used to check equipment design. In most cases, it includes testing designed to evaluate the behavior of the equipment with time.

Examples of aging experiences in French reactors include:

- For mechanical equipment--diesel generators are exposed to untimely starting; steam generators are subject to unforeseen corrosion or erosion due to foreign matter.

- For electrical equipment--isolating switches are operated under unscheduled loading conditions and batteries whose autonomy sometimes changes unexpectedly.

In addition, materials such as coatings (paints) or lubricants (oils or greases) have a great effect on the behavior of the equipment with which they are associated.

In conjunction with the safety authorities, Electricite de France has initiated a program of investigations to:

- Develop means of measurement, in theory and practice, of the aging of equipment.

- Determine the influence of nuclear power plant operating orocedures on aging.

In addition, a file of events is kept up to date for each plant and is analyzed to evaluate the behavior of the installation.

Since the beginning of the French nuclear program, means have been set up that ensure the periodic monitoring of operation or of the intrinsic properties of the equipment or systems. Some typical examples are:

- Periodic testing

- Inservice inspection

- Preventive maintenance

Finally, procedures have been set up to evaluate primary circuit performance for each plant (measurement of primary circuit pressure and temperature values above a certain threshold) and allow comparison to design features.

ITALY

Two Italian programs have been identified to understand and manage plant aging. They are (a) Italian Aging Research Program on Electrical and Instrumentation Equipment, and (b) Cycle of Preventive Maintenance.

The Aging Research Program on Electrical and Instrumentation Equipment is a limited program intended to resolve issues related to the aging and service wear of equipment at reactor facilities and their possible impact on plant safety; this program has been started in Italy. The main goal is to provide a basis for assessing the adequacy of industry methods for preconditioning before qualification testing.

The assessment will include examining and testing equipment removed from the Garigliano reactor now awaiting decommissioning. The candidate Garigliano components that have been selected through site visits by ENEA/DISP and ENEL are power and signal cables and thermocouples.

The cable activities will be based on the implementation of qualification test plans on naturally aged (thermal and low dose rates) cables and on cable available at the plant warehouse. The qualification test plan will include definition of electrical characteristics, aging under controlled environment conditions, humidity absorption after aging, experimental determination of activation energies, fire propagation tests, LOCA tests, and radiation exposure damage tests.

The activities relevant to thermocouples will be based on detection of the systematic error affecting the measurements on all the thermocouples presently immersed in shielding water; measurement of time constants on most of the thermocouples, where feasible; metallographic study of the hot junction; qualification of the cold junction in compliance with the procedures presently in use; and dynamic brittleness tests.

The program will be conducted in cooperation with ENEA and ENEL. A dialogue for cooperative research with the ongoing RES/NPAR program in the United States has been initiated.

The Italian Preventive Maintenance Program is implemented through monitoring, inspections, recordkeeping, and repairs and replacements.

JAPAN

Technology development for nuclear power plant life extension is a priority effort in Japan. A 7-year technology development plan was implemented in FY 1985.

The major tasks in plant life extension technology development are related to the diagnosis of nuclear power plant aging deterioration, prediction of remaining plant life, and replacement and improvement of plant equipment. At this stage, the specific items of research and development have not yet been decided. It is expected that the following tasks will be recommended for comprehensive evaluations:

- Development of life diagnosis and prediction methodology (developing life evaluation methods, creating a data base on aging and degradation phenomena, developing monitoring techniques).

- Development of technology for replacing and improving large equipment (developing methodology, verification).

The selection of equipment and structures to be studied in detail was made in 1985. Structures and equipment that could be easily replaced were not subject to further evaluation. The remaining structures/components (108 BWR, 102 PWR) were ranked according to evaluation criteria that included ease of replacement and impact on safety. Those with the 10 highest priority rankings were:

BWR

1. Reactor pressure vessel
2. Reactor containment suppression chamber
3. Containment dry well
4. Reactor pressure vessel support girder anchor bolt
5. Emergency core cooling system
6. CRD housing (outer)
7. Core shroud
8. Reactor recirculation system flow control valve
9. CRD housing (inner)
10. Core support plate

PWR

1. Reactor pressure vessel
2. Reactor cooling system piping
3. Primary coolant pump casing
4. Reactor pressure vessel support structure
5. Steam generator
6. Cables in containment
7. Reactor pressure vessel inside structure

8. Containment

9. Main piping

10. Safety-related pumps

The concrete structures were also subjected to study, and the following parts were identified to be studied in detail: (a) reactor pedestal, (b) biological shield, (c) penetration, (d) basemat, and (e) water intake facility.

The technology development plan anticipates completion of necessary activities to establish the evaluation by the start of 1991.

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The nuclear plant aging research described in this plan is intended to resolve issues related to the aging and service wear of equipment and systems and major components at commercial reactor facilities and their possible impact on plant safety. Emphasis has been placed on identification and characterization of the mechanisms of material and component degradation during service and evaluation of methods of inspection, surveillance, condition monitoring, and maintenance as means of mitigating such effects. Specifically, the goals of the program are as follows: (1) to identify and characterize aging and service wear effects which, if unchecked, could cause degradation of equipment, systems, and major components and thereby impair plant safety, (2) to identify methods of inspection, surveillance, and monitoring, or of evaluating residual life of equipment, systems, and major components, which will ensure timely detection of significant aging effects prior to loss of safety function, and (3) to evaluate the effectiveness of storage, maintenance, repair, and replacement practices in mitigating the rate and extent of degradation caused by aging and service wear.

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