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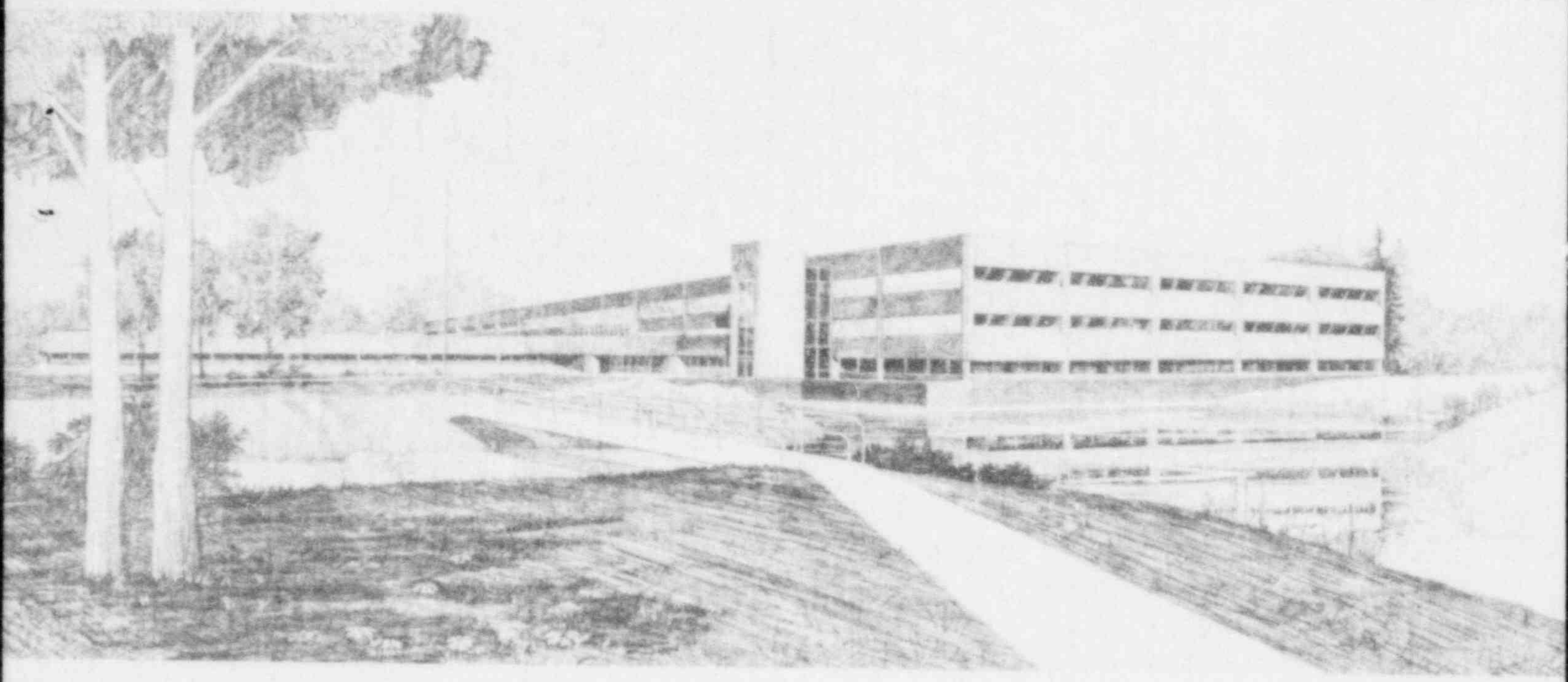

Idaho National Engineering Laboratory
Operated by the U.S. Department of Energy

Survey of Aged Power Plant Facilities

James A. Rose
Robert Steele, Jr.
Kevin G. DeWall
Bruce C. Cornwell

June 1985

Prepared for the
U.S. Nuclear Regulatory Commission
Under DOE Contract No. DE-AC07-76IDO1570

EG&G Idaho

Available from

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U.S. Government Printing Office
Post Office Box 37082
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National Technical Information Service
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NUREG/CR-3819
EGG-2317
Distribution Category: R2

SURVEY OF AGED POWER PLANT FACILITIES

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Washington, D.C. 20555
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FIN No. A6389

ABSTRACT

This report presents the results of the survey of Aged Nuclear Power Plant Facilities that was conducted under the sponsorship of the United States Nuclear Regulatory Commission (USNRC), Office of Nuclear Regulatory Research. The work was conducted in FY-1984 by EG&G Idaho, Inc., in support of the USNRC Long Range Plan. The survey concentrated on boiling water reactors and pressurized water reactor safety related systems, with regard to component failures as determined from operating histories. Only failures that were determined to be age related were included.

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

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

This survey points out, with verifying evidence, that the identification and elimination of the system level causes of component failure is a viable approach to prevention and mitigation of the major reported aging effects.

The results of this survey are to be used by the USNRC to implement a research program that will systematically identify aging and service wear effects, which are likely to affect plant safety.

FIN No. A6389—Survey of Aged Power Plant Facilities

EXECUTIVE SUMMARY

This report describes the results of the survey of Aged Nuclear Power Plant Facilities that was conducted for the United States Nuclear Regulatory Comission, Office of Nuclear Regulatory Research. This research activity is one portion of the overall Nuclear Plant Aging Research (NPAR) program. This research will identify Nuclear Power Plant aging concerns and additional research needs.

For the purpose of this research, Aging and Age related failure are defined as:

Aging: A continuing process of time related degradation of a component, subassembly, or system resulting in a partial or complete loss of function

Age Related Failure: A failure of a component, subassembly, or system that can be attributed to time related degradation due to operating or environmental effects.

The primary objective of this study was to identify aging issues and future research needs by conducting a survey of aged nuclear power plant facilities. The methodology used was to survey eight older commercial power plants by first analyzing plant operating experiences, as put forth in the published literature, and then to corroborate these results by actual field inquiry. This approach did not, nor was it intended to, produce results that could be considered beyond dispute in all regards. The intent was merely to go to the detail necessary to point out, with a basis in fact, some currently unexplored aging issues and to recommend the direction in which future aging research should proceed.

In the published literature [i.e., Licensee Event Report (LERs) and other plant history data files] component loss of function (failures) is well documented but detailed evaluation of plant systems with regard to failure causes is not. The data available are limited for issue identification.

The following results were obtained from the survey.

- Approximately 70% of reported significant Age Related failures for the fluid systems analyzed in commercial plants are the result of four mechanisms; erosion, corrosion, vibration, and foreign material.

- High incidence of component failure in a plant system may or may not indicate a weakness in a component but rather a change in the system, its maintenance, or its mode of operation.

- There appears to be a strong correlation between cause of failure for components and the functional system in which they operate, [e.g., failures due to vibration and foreign materials, in Residual Heat Removal (RHR) Systems, 90%; in Safety Injection Systems (SIS), 61%; and in the Cooling Water Systems (CWS), 51%. The CWS has an additional 30% due to corrosion and erosion].

- Data from interviews with personnel from commercial plants suggests that with certain failure mechanisms (e.g., water hammer, overnormal vibration, and chemistry control) the heatup and cooldown cycles and cold shutdown modes of operation are associated with high failure rates.

- Test reactor facilities do not experience the same magnitude of failure due to foreign materials as commercial plants. This is probably because efforts are made to keep them clean.

Conclusions drawn from the study are:

- Since commercial power plant system environments are directly responsible for most age related component failures, examination of individual components to determine failure mechanisms should be supplemented with aging/systems interaction studies. System design, maintenance, and operational problems are so predominant that it is probable that failures due to the aging of component materials could not be identified with any certainty. Only after the effects of the major failure mechanisms are mitigated could material analysis, coupled with the understandings of stressors and environment, yield definitive results.

- System cleanliness with regard to foreign materials and chemistry control should have strict limits placed on it and should be monitored as part of the normal maintenance procedures.

- Judging from the number of vibration failures evidenced in the survey, flow, and equipment induced vibration is a problem in plant operation. Prevention of vibration and thermal cycle effects could be enhanced by anti-rotation features being added to all fasteners on safety or safety-related components in the plant.

- Any changes contemplated for the system, or component design, operation, or maintenance must take into account possible adverse effects on every component in the system and related systems. System and component interactions are much more prevalent and subtle than most realize.

- Condition monitoring has obvious advantages and should be considered as part of a comprehensive surveillance program. Because many conditions that govern component performance in todays plants are system effects, component condition monitoring alone is not adequate. To be of maximum benefit each component and system should have its degradation rate characterized.

The following recommendations are made for future research based on the findings of this survey:

- The stressors that develop with time and that effect component reliability, in significant number of cases, have been due to functional system operation or deterioration. Reports of failed components often indicate only the stressor and not the root cause of failure. It appears from our preliminary evaluation that many root causes of component failure are predictable, preventable, and/or correctable. Additional research is required to confirm this, perhaps on a single reactor safety system.

- Additional work is needed to characterize systems, prioritized by component failure rate or safety significance, to provide insight into root cause(s) for the major reported failure mechanism(s).

- The present concept of condition monitoring should be expanded to include monitoring system level parameters that can most directly affect component reliability. This would require research to determine which parameters should be monitored for each type of component in each system and to determine the most appropriate method to be used.

- The heatup and cooldown evolutions of plant operations, as well as the cold shutdown period, appear as times of high component stress and subsequent failure for the fluid-mechanical systems that were studied. Further analysis of these plant operating modes would provide a necessary part of the basis for guidelines that would aid in prevention and mitigation of age related equipment failure.

- A concentrated effort should be made on the part of industry owners groups, technical societies, and governmental agencies to understand and take advantage of the tangible (financial) and intangible (safety) benefits that elimination of system level failure causes would bring.

ACKNOWLEDGMENTS

The technical staff for this aging study would like to express their sincere appreciation to the following persons for their assistance in the preparation of this report. The Aged Facility Survey Review Group, especially Tharan L. Cook, for their review and technical input; L. Robert Fitch for his assistance in establishing the aged facility survey data base; Joan M. Mosher, Louise E. Judy, and Julie A. Sellars for their concentration and timely efforts in text composition, word processing, and figure preparation; and Arthur R. Tetley for his technical editing services.

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NOMENCLATURE

This section contains nomenclature used in this report, definitions of commonly used terms, definitions of failure mechanisms, and references that are used for the sources of these definitions.

Definitions of Commonly Used Terms

Aging—A continuing process of time related degradation of a component, subassembly, or system resulting in a partial or complete loss of function.

Age Related Failure—A failure of a component, subassembly, or system that can be attributed to time related degradation due to operating or environmental effects.

Component—The largest entity of hardware for which data are most generally collected and expected to be available (for example: pump with motor; valve with operator; amplifier; pressure transmitter). It is generally an off-the-shelf item procured by the system designer as a basic building block for his system. It would be distinguished from seals, bearings, nuts, bolts, and other piece parts from which the component is manufactured.[1]

Design Life—The time during which satisfactory performance can be expected for a specific set of service conditions upon which design margins are based.

Failure—A type of fault requiring that a component be repaired in order for it to perform its design function. Failures are sometimes classified as primary or secondary failures. However, in classifying failures for this report, no distinction has been made between these two classifications:[1,2]

Primary failure—The so-called "random failure" found in the literature. It results from no external cause.

Secondary Failure—A failure that results when the component is subject to conditions that exceed its design envelope (for example, excessive voltage, pressure, shock, vibration, temperature).

Failure Mechanism—The identified most direct cause or event that prevented the component from performing its intended function.[2] This may only be a symptom of the true or root cause of failure.

Parts—Definable pieces from which a component is manufactured (for example, seals, bearings, nuts, bolts, resistors, relays, etc.).

Stressor—A load or environment that tends to affect the functional capability of a part or component.

System—A collection of components arranged to interact so as to provide a desired function (for example, Containment Spray System, Residual Heat Removal System, High Pressure Coolant Injection System).

Definitions of Failure Mechanisms

Corrosion (CORR)—"The deterioration of a metal by chemical or electro-chemical reaction with its environment."[3]

Drift (DRFT)—"An undesirable change in output over a period of time, which change is unrelated to the input environment, or load."

Electrical Arc (EARC)—The electrical current or discharge that can leap across a gap between two oppositely charged conducting materials. This often results in damage to one or both of the charged surfaces.

Embrittlement (EMBR)—"Reduction in the normal ductility of a material due to a physical or chemical change."[3]

End of Life (EOL)—A phrase used to imply the failure was not unexpected, given the use, environment and specified design life for the component or system.

Erosion (EROS)—"The destruction of metals or other materials by the abrasive action of moving fluids, usually accelerated by the presence of solid particles or matter in suspension."[3]

Fatigue (FATG)—"The phenomenon leading to fracture under repeated or fluctuating stresses having a maximum value less than the tensile strength of the material."[3]

Foreign Material (FRMA)—Any kind of undesirable solid or chemical material which is suspended in, or deposited by, a fluid. (Note: In the literature sometimes referred to as "contaminates" or "crud".)

Friction (FRTN)—The resistance to the relative motion of two materials in physical contact. Often results in impairing the function of one or both of the objects in contact (also see Wear).

Hardening (HRNG)—"Increasing the hardness (of a material or object), usually involving heating and cooling."[3]

High Temperature (HTEM)—Temperatures of a material or object higher than its normal or specified range.

Other (OTHR)—Used to include all mechanisms not otherwise named.

Oxidation (OXID)—"A reaction in which there is an increase in valence (of an element or ion) resulting from a loss of electrons"[3] caused by the union of the element or ion with oxygen. Often results in a change to the surface of a material that reduces its resistance to chemical or physical damage.

Stress (STRS)—"Force per unit area, often thought of as force acting through a small area within a plane."[3] When forces and moments are applied to a rigid body, stresses result. Failures due to stress often appear as cracks, separation, and changes in shape or size.

Stress Corrosion (STCR)—"Failure by cracking under combined action of corrosion and stress, either external (applied) or internal (residual). Cracking may be either intergranular or transgranular, depending on metal or corrosive medium."[3]

Temperature (TEMP)—Temperature of a material or object different than its normal or specified range.

Unknown (UNKN)—A term used when the failure mechanism is unknown or not otherwise specified.

Vibration (VIBR)—A rapid, rhythmic motion of the particles of a fluid or an elastic solid across a position of equilibrium.

Wear (WEAR)—To impair, consume, or diminish by constant use or by the friction of rubbing, scraping or flowing, etc. A "catch-all" term often used in lieu of other more definitive statements of the method of failure.

Water Hammer (WTHM)—A general term used to describe any of a set of phenomena that are characterized by hydraulic shocks induced or transmitted in the piping system.

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2. C. F. Miller, *Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commerical Nuclear Power Plants, January 1, 1976 to December 31, 1978*, NUREG/CR-1740, May 1981.

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SURVEY OF AGED POWER PLANT FACILITIES

INTRODUCTION

The overall FY-1984 United States Nuclear Regulatory Commission (USNRC) objective for aging research, as given in the USNRC Long Range Plan, is the "identification of significant component/environment aging mechanisms with respect to potential risk to public safety." "This research applies principally to the time-related degradation of electrical and mechanical components during service and the potential impacts of degradation upon public safety. Specifically, this research should develop methodologies to identify such potential impacts on safety, including the prevention or correction procedures, well in advance of their actual occurrence."[1]

To meet this objective the Nuclear Plant Aging Research (NPAR) Program has been initiated under the sponsorship of the USNRC, Office of Nuclear Regulatory Research. The program goals are to: (a) identify electrical and mechanical component aging and service wear effects likely to impair plant safety, (b) identify methods of inspection and surveillance of electrical and mechanical components that will be effective in detecting significant aging effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented, and (c) identify and recommend acceptable maintenance practices that can be undertaken to mitigate the effects of aging and to diminish the rate and extent of degradation caused by aging and service wear. In the near-term, the NPAR program is directed at the reviews of operating experiences and establishing a data base containing the known and necessary information for aging assessments of nuclear power plant components and structures, and at identifying and prioritizing aging issues and future research needs.

To accomplish the near-term objective in the minimum time and to foster diverse views regarding aging research needs, the NPAR Program was separated into eight research activities by the technical monitor. The NPAR research activities are to identify risk-oriented aging effects, assess component aging, and evaluate inspection and surveillance monitoring methods.

In April 1983, the INEL was asked to undertake one of the research activities as part of the overall NPAR Program. The primary objective of this research was to perform a survey of aged light water reactors (LWRs) power plant facilities to determine what, if any, loss of function can be attributed to aging and to evaluate the potential for any identified aging process to be significant for LWRs. The product of this research effort will be a set of recommendations identifying aging concerns and additional research needs.

To initiate the project, a general approach to the conduct of the survey was devised. This consisted of:

1. Identifying nuclear plant facilities that could be considered aged

2. Identifying the best sources of information for each of these facilities within schedule and budgetary constraints

3. Performing a survey of safety related mechanical, electrical, and structural components identified in these facilities to determine what loss of function could be attributed to aging and service wear

4. Identifying components that have a relatively high rate of age related failures

5. Characterizing the environments in which components experienced aging effects.

The "Discussion" section presents an overview of the methods used and a discussion of the data analysis. The "Survey Results" section describes the results of the study. Recommendations for future research and conclusions are contained in the "Conclusions and Recommendations" section. The selection criteria of plants surveyed, information source identification, interviews, and review group are located in Appendices A through D.

DISCUSSION

The four major subtasks to be undertaken in conducting this survey were (a) to identify the facilities to be surveyed, (b) to identify the sources of information to be used, (c) to design and implement an automated data system, and (d) to actually conduct the surveys and to identify the components and environments where aging was a factor in loss of function. Each of these subjects are discussed briefly in the following paragraphs and in more detail in Appendix A.

Aging Defined

To establish guidelines and to bound the aging concept applicable to the Idaho National Engineering Laboratory (INEL) participation in the Nuclear Plant Aging Research (NPAR), and since there appeared to be so many different definitions and usages of the terms *aging* and *age related failures* it was imperative that we define these terms at the outset of the study. For this project we have defined *aging* and *age related failure* as:

Aging. "A continuing process of time related degradation of a component, subassembly, or system resulting in a premature loss of function." If a loss of function, (i.e., failure), occurs after the component or system has performed its intended function for its specified design life, it is not considered an aging problem for the purposes of this survey, it is considered a maintenance problem. What is considered an aging problem for this study is a component or system that fails to perform its intended function for its full design life and the cause for the loss of function is time related.

Age Related Failure. "A failure of a component, subassembly or system which can be attributed to time related degradation due to operating or environmental effects."

Additional definitions used in this report are given in the "Nomenclature."

Survey Methodology

Since it would be impractical to survey all commercial reactors in the United States to determine the direction future aging research should follow, the number of plants surveyed was limited by use of plant selection criteria. The facilities initially chosen included 32 operating commercial nuclear plants at 23 sites and four test reactors at two INEL sites. The rationale for the choice of each of these facilities and a listing of the facilities chosen is given in Appendix A.

Once the facilities to be surveyed were identified it was necessary to choose the most practical method of getting the required information to obtain a reasonable indication of aging issues and the data to be utilized. The method chosen was a survey based on published literature, backed by interviews with plant personnel.

In trying to locate the published information that would best meet the survey objectives we found that others[2,3,4,5,6] have previously used Licensee Event Report (LER) data either to identify aging trends or to estimate gross failure rates for specific plant equipment. Hence, it did not seem to be particularly fruitful to expand that effort as part of this study. Furthermore, LERs, as an instrument of obtaining aging information have severe limitations.[2,7,8] After reviewing several other documents and data bases in detail it was decided to use two primary sources of data. The first of these was the Nuclear Power Experience (NPE)[9] published by the S. M. Stoller Corporation, Boulder, Colorado, and the second was all the age related USNRC Inspection/Enforcement (IE) documents published to date.

The published NPE data is a compilation of about 20,000 separate pieces of information from periodicals, technical papers, technical reports, LERs and correspondence between plant owner and USNRC pertaining only to plant operating problems. By necessity the data is condensed somewhat before it is published, but since the purpose of the NPE is to more fully and more objectively explain problems and their suspected causes, we judged that the information obtainable from this source would meet our criteria exceptionally well.

A review of all the existing IE Bulletins, Notices, and Circulars (~575 reports and supporting data) for age related problems was conducted. This source was used because, if an IE document is written on a subject, it is considered by the USNRC as (a) a potentially wide-spread problem or, (b) of high

enough significance that action should be taken to assure the problem cannot arise. In addition, we eliminated duplication of data from the NPE and from the USNRC IE documentation, in the relatively few instances where it existed.

As a part of this task, it became obvious that a considerable amount of data associated with age related component failures would have to be filed and managed. A review of existing automated data files was made and none were found that appeared adequate for this specialized work. Consequently, a new computerized data file was initiated for this task.[10]

It should be noted that this data base was initiated as a tool to help manage the information which, of necessity, would need to be accumulated. It was not intended as a product of the survey. However, as the survey progressed it became evident that this *tool* could be expanded to fulfill one of the NPAR near-term objectives as given in the "Introduction" section of this report.

In order to minimize subjectiveness on the part of the reviewers every piece of data analyzed had the same set of questions asked of it and was either accepted or rejected as an aging concern based on the same criteria. The criteria used in this process are given in Appendix A.

Initially, the appropriate data from eight of the oldest plants (4 BWRs and 4 PWRs) containing components and system similar to modern designs was entered into the data base (698 reports). Entry of the NPE and IE data was suspended at this point for preliminary analysis. The initial effort was to review the data for age related equipment or structure degradation with reoccurring failure patterns. The emphasis was on isolating specific equipment or structures with the highest failure *rates*. The result of this portion of the analysis was, as expected, in agreement with the Oak Ridge National Laboratory (ORNL) findings on nuclear plant aging trends.[11] But, in addition, what also started to materialize was a very specific reoccurring pattern of failure *mechanisms* (causes). Four failure mechanisms were responsible for 70% of the reported plant equipment problems in fluid-mechanical systems. These mechanisms were corrosion, erosion, vibration, and foreign materials (either chemical or solid contamination).

As a result of this, it was becoming obvious to the authors that to achieve the stated USNRC objectives[1] our efforts should be directed not at identifying degraded components and materials but rather at identifying the predominate *causes* of component failure in plant safety systems. This idea, with supporting information, was presented to a group of INEL specialists (see Appendix D) and was endorsed as a valid approach.

To further validate our initial results the NPE reports describing failures within the Residual Heat Removal (RHR) systems of nine additional BWRs and within the Safety Injection Systems (SIS) and Cooling Water Systems (CWS) of thirteen additional PWRs were encoded and added to the data base. The criteria by which these three systems were chosen are given in Appendix A. The results of the more detailed study of these three systems confirmed our previous findings within reasonable limits. Therefore, we gained sufficient confidence in our initial findings to judge that data from the compliment of plants originally chosen need not be encoded and analyzed at this time. Inherent in that decision is the assumption that little would be found in the remaining plants that would materially alter our recommendations for future aging research. Indeed, one of the recommendations must be to expand our work to verify that assumption.

To corroborate the much greater volume of objective data obtained from the published literature, interviews with plant personnel were then conducted from three additional commercial plants and four test reactor facilities (refer to Appendix A). To minimize subjectiveness on the part of the interviewers, and possible discrepancies between our findings and those of the Plant Aging Workshops, conducted by the Sandia National Laboratory (SNL), we chose to use substantially the same questionnaire as was used by SNL. The primary difference is in the emphasis we placed on identifying the suspected cause of failure rather than on identifying if there is any evidence that aging problems do, in fact, exist.

The complete results of the interviews along with comparisons to the results of the published data survey are given below.

Published Data Analysis

For the first four PWRs surveyed there were 218 reported incidences of age related failures distributed among 61 components. The components that showed the highest percentage of failure were,

as expected,[7] valves at 17%, pumps at 16%, and pipe components at 11%. For the first four BWRs there were 240 reported incidences among 47 components. As for the BWRs, the highest failure rates were in valves at 37%, pumps at 10%, and pipe components at 9% (refer to Figures 1, 2, and 3). These results seem reasonable since there are many more valves than pumps in a power plant and both of these high stress, active components contain moving parts that could be expected to fail more frequently than passive pipe components. It should also be noted that by the criteria used to accept or reject data, event reports were excluded if they described conditions that could not be characterized as complete failures. Faults found during periodic surveillance, most electrical parts for example, are not, therefore, included (see Appendix A).

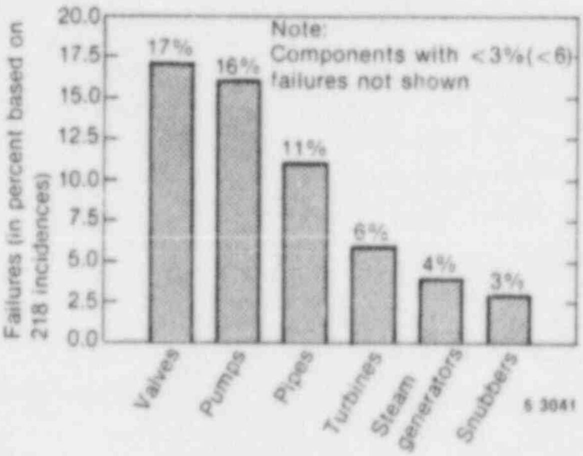

Figure 1. Reported failures by component for 4 PWRs only.

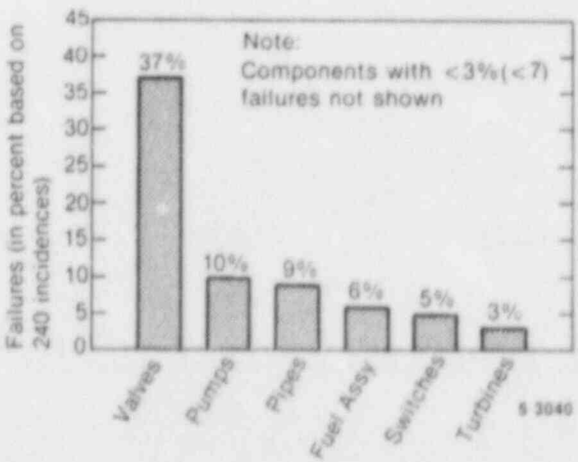

Figure 2. Reported failures by component for 4 BWRs only.

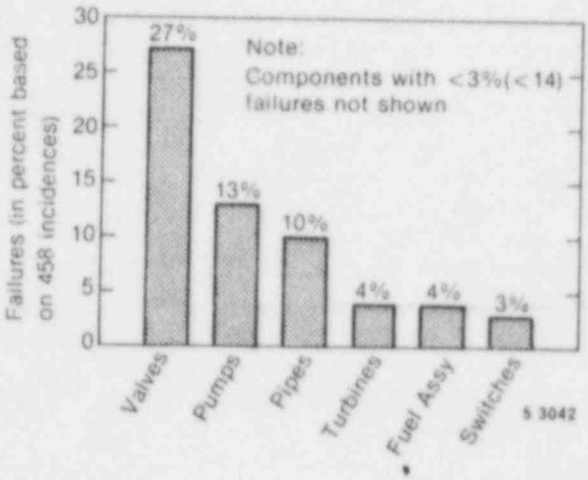

Figure 3. Reported failures by component for 4 BWRs and 4 PWRs.

The search for a discernable pattern of component failures led eventually to analyzing the number of reported failures for each type of component within each functional system. (See Appendix B for listing of systems surveyed.) The result of this effort was as expected; the systems with the highest number of valves had the highest number of valve failures. As with component types, data sorts by part, material, and vendor indicated no clear-cut failure patterns with respect to any individual component or material.

At this point, a paper by E. J. Brown of the USNRC, Office of Analysis and Evaluation of Operational Data (AEOD) was brought to our attention.[13] In it Mr. Brown noted that one of the findings of a previous AEOD study (AEOD/C203) was that after any failure "plant staff efforts are directed toward return (of the plant) to operational status rather than finding the root cause" of the failure. The implication is that the real causes of failure are not typically being determined and corrected. In this paper, Mr. Brown encouraged the industry and the regulatory agencies to use "evidence from operating plants to identify aging mechanisms" as a realistic approach to accommodating the aging problem. This directed our research toward identifying the *causes* of component failure. However, since valves, pumps, and pipes were the components displaying the highest failure rates (see Figure 3) we limited this portion of the search to these items. The results are shown graphically in Figures 4, 5, and 6. In these figures *stress corrosion*, *wear*, *unknown*, and *other* mechanisms were intentionally omitted.

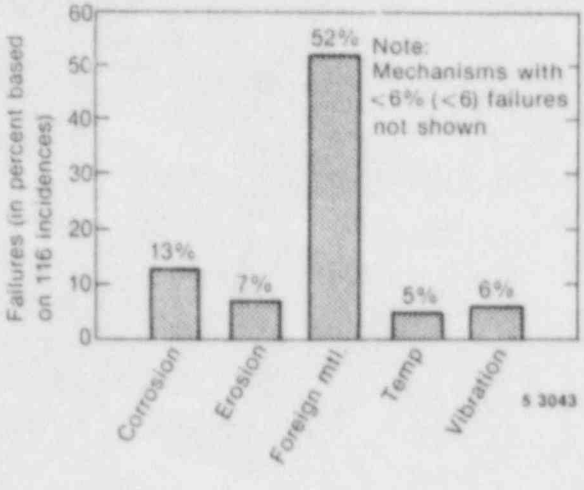

Figure 4. Reported failures by mechanisms for valves in initial 4 BWRs and 4 PWRs.

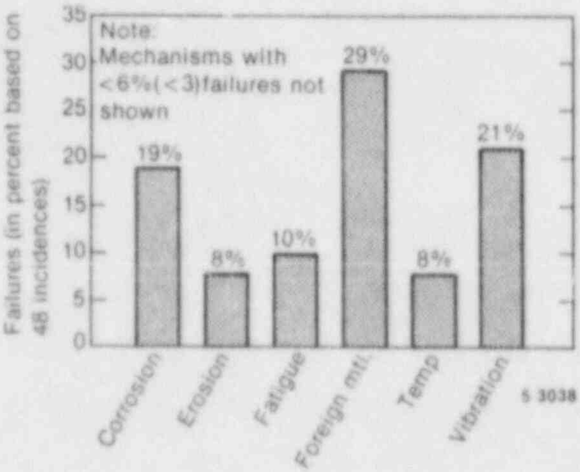

Figure 5. Reported failures by mechanism for pumps in initial 4 BWRs and 4 PWRs.

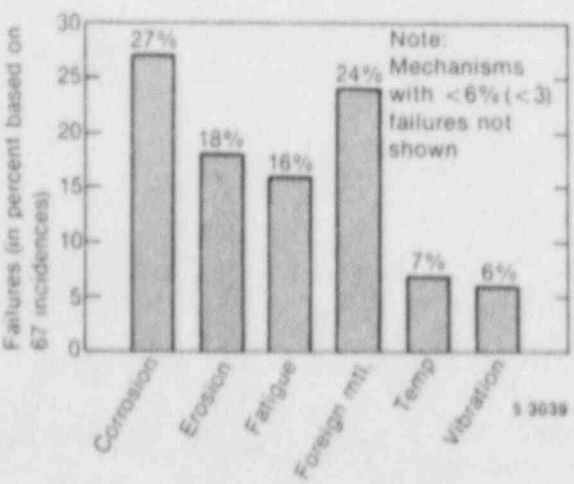

Figure 6. Reported failures by mechanism for pipes in initial 4 BWRs and 4 PWRs.

It is obvious from Figure 4 that valves fail most often (>52% of the time) from damage by foreign materials, either internal or external to the valves (refer to Appendix C for list of failure mechanisms considered). This result is generally supported by Murphy's work at ORNL.[11] Figure 5 shows that pumps are reported to fail nearly 70% of the time due to corrosion, vibration, or foreign materials. Considering the standard design of pumps and their typical usage in plant systems this is not a surprising result. For pipes (Figure 6), not surprisingly, the primary failure mechanisms are corrosion, erosion, fatigue, and foreign materials. (Note: Foreign material is often less appropriately termed in the literature as *crud* or nonradioactive *contamination*.)

Note: Results dealing with failure mechanisms are derived for this report from the data base reduced by the number of failures attributable to (a) stress corrosion, (b) wear, (c) other causes, and (d) unknown causes. The rationale for excluding these data are as follows:

1. Stress Corrosion—Correctly diagnosing stress corrosion (STCR) as the cause of a component having failed, requires specific knowledge and/or training. We have assumed a high degree of error in that diagnosis and have rejected it as a major contributor to failure in components that are not already the subject of other research efforts.

2. Wear—Since *wear* is a generic term that can be used in lieu of another, more definitive statement of the cause of failure, we have assumed that many of the components that are said to have failed for this reason actually failed due to one or more of the other listed causes. Therefore, *wear*, itself, is not considered here as one of the major contributors and the data was adjusted accordingly.

3. Unknown—This category was used in the data collection phase of the survey if the cause of failure was not specified either specifically or in the accompanying text. Since, in most cases this category was large, including it in the data base would tend to reduce the reliability of the statistical analyses performed on the data. It therefore was rejected.

4. Other—Though small, this *catch all* category, used when a reported failure mechanism was not one of the predominant 19 used, was also excluded. (See Appendix B for list of mechanisms considered.)

The results represented by Figures 4, 5, and 6, taken independently, as applied to the specific components in question, (valves, pumps, and pipes) yielded no surprises. However, taken together a pattern started to emerge. For these three components foreign materials contributed to about 40% of the reported failures and corrosion to about 18% of the failures.

This realization led to a data base sort on the number of reported failures versus failure mechanisms across all components. The results are shown in Figures 7, 8, and 9. In these figures *stress corrosion*, *wear*, *unknown*, and *other* mechanisms were intentionally omitted. These graphs plainly show that a very large percentage of reported failures are caused by a relatively few mechanisms. In fact, these data show that for BWRs and PWRs taken together 70% of the reported failures are due to only four of the sixteen causes finally considered as appropriate (see Appendix C). These are erosion, corrosion, vibration, and foreign materials.

One of the results expected out of an examination of reported failures versus failure mechanisms was a high incidence of transmitter drift problems. This expectation was based on the results of NUREG/CR-3543[7] and NUREG/CR-1740[3], which give drift as accounting for between 39 and 71% of reported *reduced capability faults* reported in LERs. Since drift related problems are rarely complete failures as required by the data acceptance criteria set up for this study (refer to Appendix A), they were not included in our data base and, therefore, drift did not appear as a major failure mechanism.

Acknowledging that 70% of the complete failures reported were due to four failure mechanisms may point out that aging and service wear management has to be pursued by more than one method. A qualified component operating outside of its design envelope is not at fault for less than normal performance. The root cause of many component failures is outside the design envelope. The symptom of the age related degradation is a failed component, but, the root cause is the operating

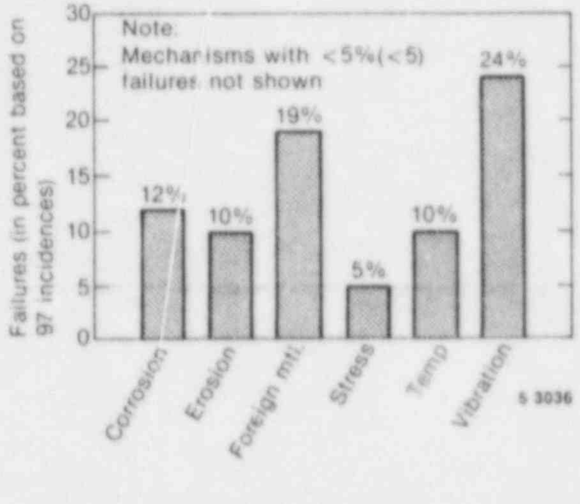

Figure 7. Reported failures by mechanisms for all components in 4 PWRs only.

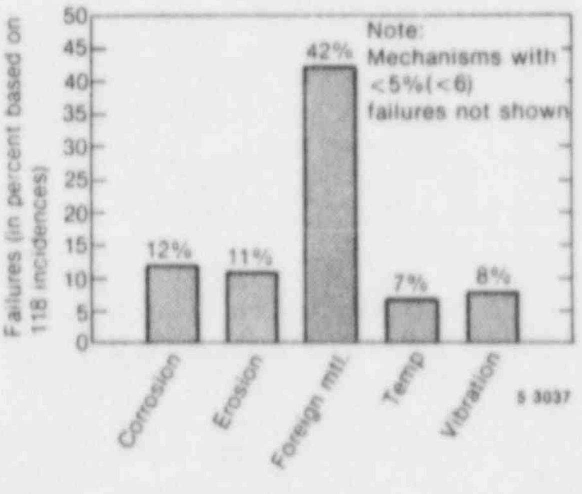

Figure 8. Reported failures by mechanism for all components in 4 BWRs only.

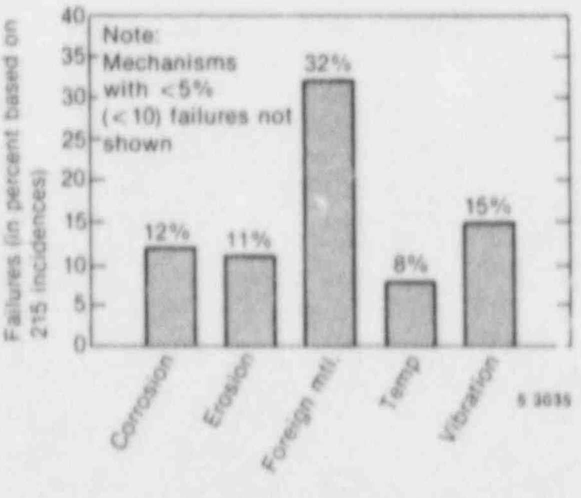

Figure 9. Reported failures by mechanisms for all components in 4 PWRs and 4 BWRs.

environment. In the case of the four failure mechanisms found in this work, the system, in which the components were installed, was operating outside the design envelope. The corrosion erosion and foreign material could be considered age related degradation of the system operating environment. The vibration could be from degradation of a pump, flow induced or original design, all of which over time can produce fatigue or other vibration related failures. Todays technology may not be able to qualify a component to operate in a degraded system. Therefore, management may again require a synergistic approach.

To this end a data sort was made to determine which plant systems showed the most significant failure rates. From this information three plant systems were chosen for a more detailed investigation. The systems chosen were:

1. The Residual Heat Removal (RHR) Systems for BWRs (Figure 10)

2. The Safety Injection System (SIS) for PWRs (Figure 11)

3. The Cooling Water System (CWS) for PWRs. (Figure 12)

In these figures *stress corrosion*, *wear*, *unknown*, and *other* mechanisms were intentionally omitted.

There were other plant systems with overall higher rates of reported failures. However, after application of the choice criteria we had established (refer to Appendix A) the above systems were selected. Data was then compiled for these systems for nine additional BWRs and thirteen additional PWRs.

After the additional system specific data (290 reports) were added to the data base a sort on these three systems was conducted. The results are shown in Figures 13, 14, and 15. In all three cases, failures attributable to foreign materials were between 25 and 30%. However, for RHR Systems 64% of the reported failures were due to vibration. Since RHR Systems use basically the same types of components as do the SI and CW Systems, this result strongly implies that something about the way RHR Systems are designed or operated adversely affects failure rates. A cursory look at a typical system design determined that during system operation the pumps function at a fixed speed and flow is controlled by throttling valves. This sets up vibrations in the system that ultimately damage equipment. This scenario was later confirmed by personal interview.

For CW Systems, it could be assumed that a significant portion of the foreign materials in those systems are suspended, abrasive solids since erosion is shown to be a significant contributor (16%) to failures. A cursory look at the data accumulated for these systems does indeed indicate a high percentage of failures due to sand, silt, shellfish, etc. Personal interviews conducted later agreed with these findings.

The foregoing does not constitute a proof that RHR Systems are poorly designed or that CW Systems are contaminated. What it does seem to show is that failures of components in nuclear plant systems are not necessarily caused by weakness in the components themselves but by mechanisms that originated elsewhere in the system. This suggests a possible system, or operation or maintenance weakness, not a component weakness.

An effort was also made to determine if there was any correlation between a specific plant and the dominant reported failure mechanisms. There did not seem to be any discernible pattern that would be useful to aging research. (Refer to Table 1 for the dominant failure mechanisms for each plant surveyed.) We also did a data base sort on plant age versus reported failure mechanism. Generally, the data showed only that plants that have been on line longer have reported more failures for most failure mechanisms. This seemed reasonable.

At this point we judged that the information contained in our data base had been sufficient to accomplish our objective; i.e., to indicate some areas of needed future research and that more analysis was not necessary at this time. What follows is a synopsis of personal interviews held over the course of this study. The results of these interviews tend to support our findings that are based on the published data. The interviews also suggested other ideas that should be considered further as a part of future aging research.

Synopsis of Personnel Interviews

As indicated earlier, the intended primary value of these interviews was to corroborate the much larger quantity of data obtained from the published

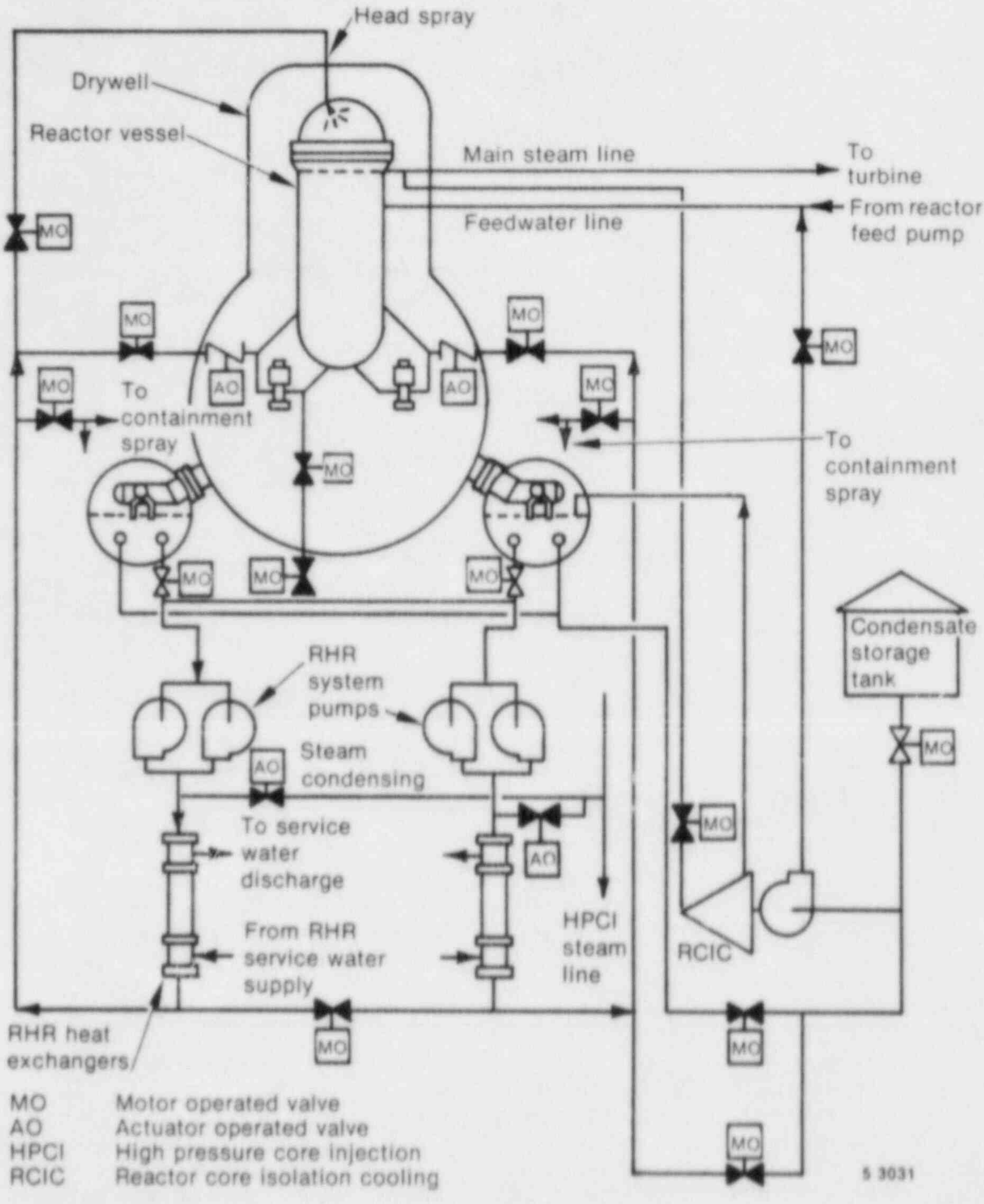

Figure 10. Residual heat removal system typical of BWRs.

literature. To this end, findings from interviews with personnel with operating and maintenance experience in commercial plants are discussed in "Commercial Plant Interview Results" section. Following that ("Test Reactor Interview Results" section), the findings from test reactor interviews are discussed. As with any survey, individual points, taken alone, may not be important but collectively, a pattern may begin to appear that is significant. This did occur in this survey. In the section, "Patterns of Occurrences Based on Interviews," two questions are raised based on recurring patterns of

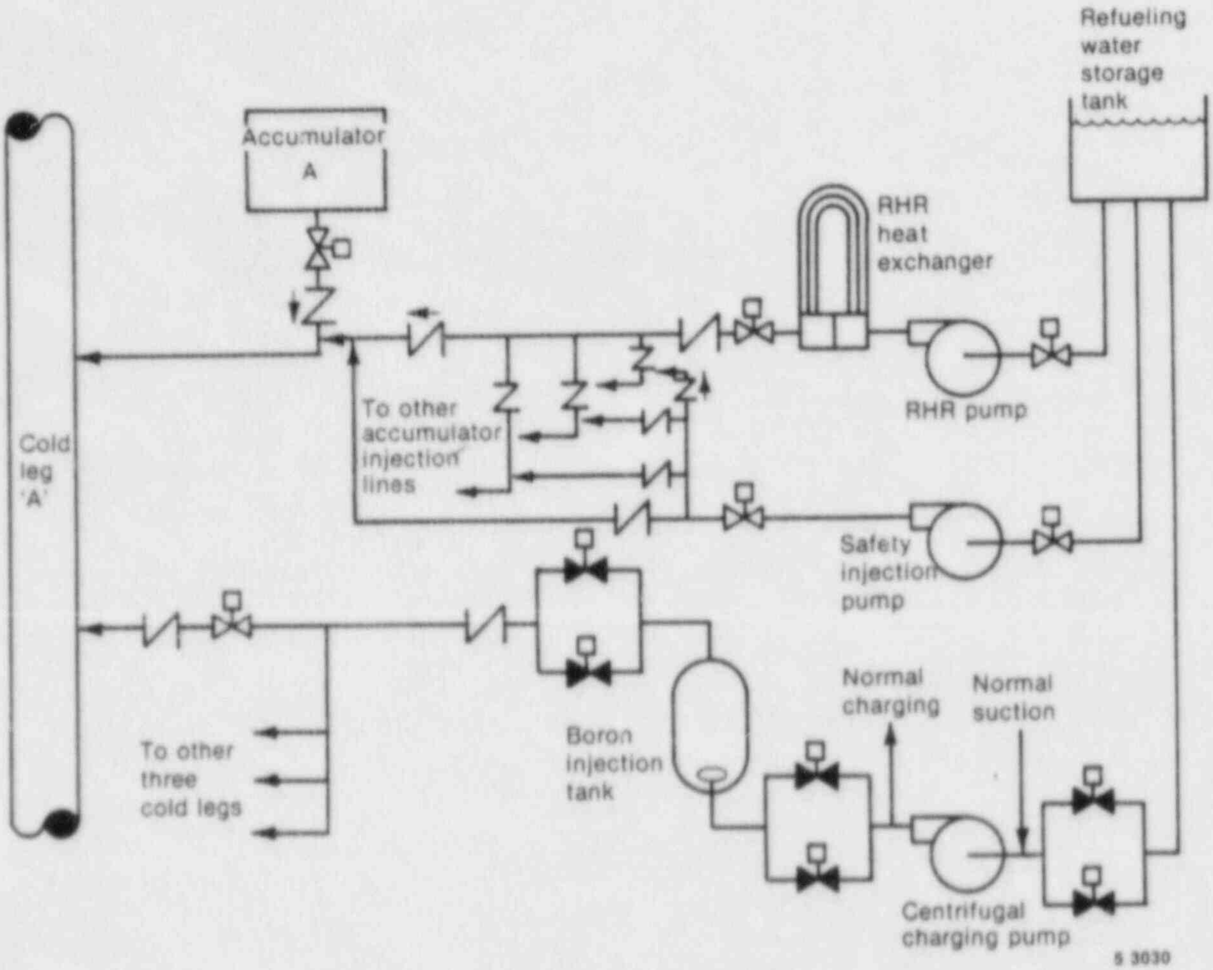

Figure 11. Emergency core cooling system simplified composite of a PWR.

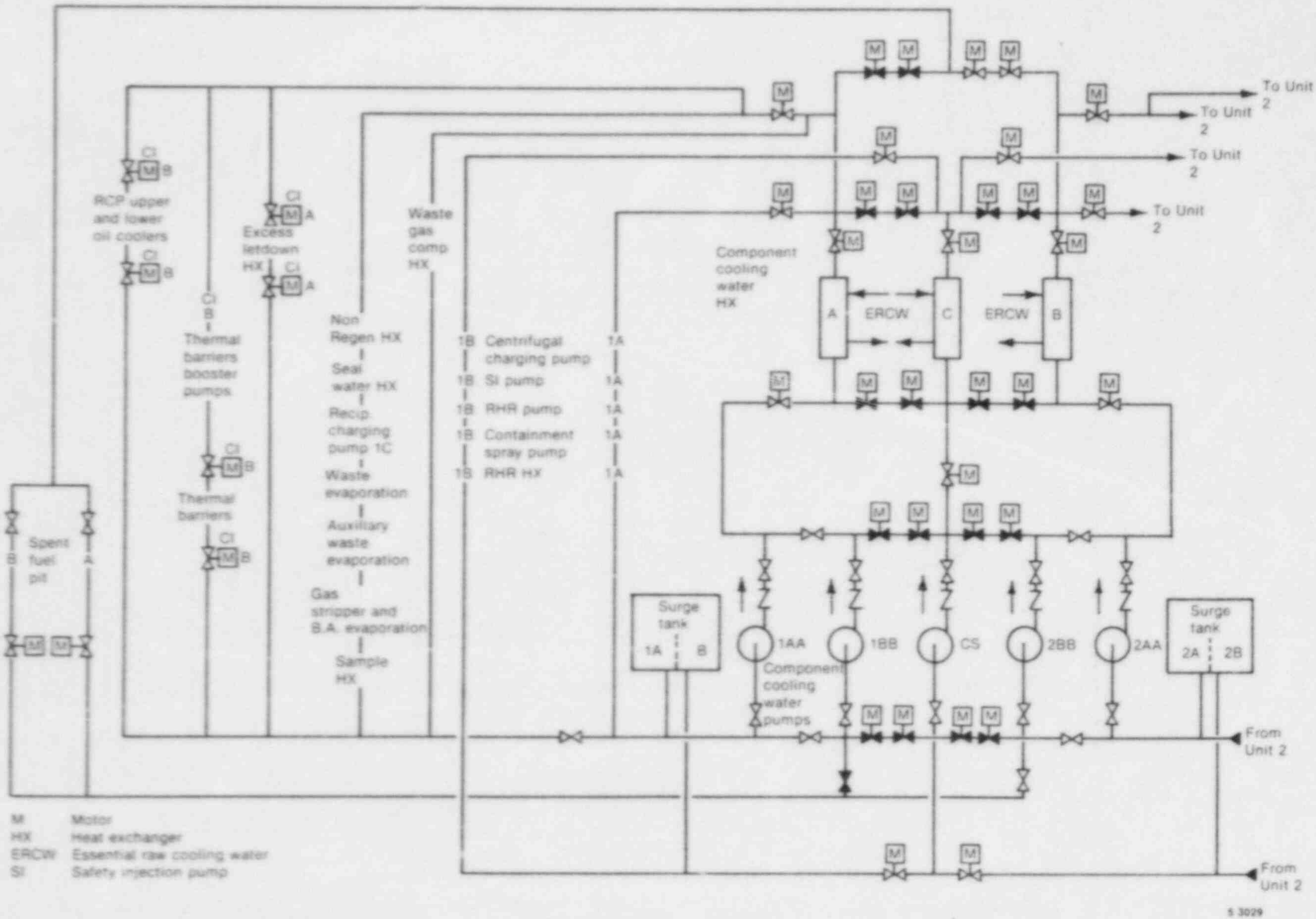

Figure 12. Component cooling water system typical of PWRs.

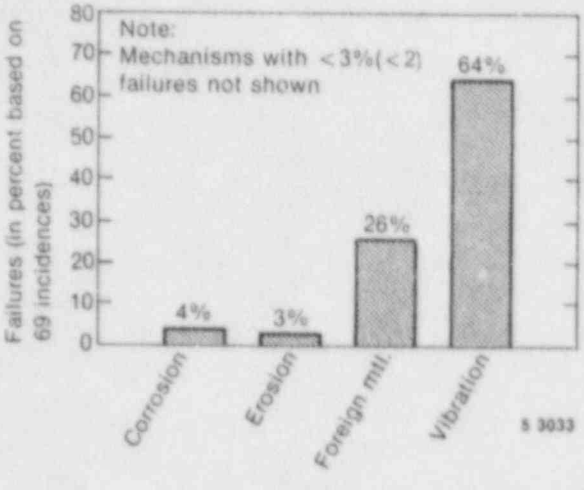

Figure 13. Reported failures by mechanisms for RHR systems in 13 BWRs.

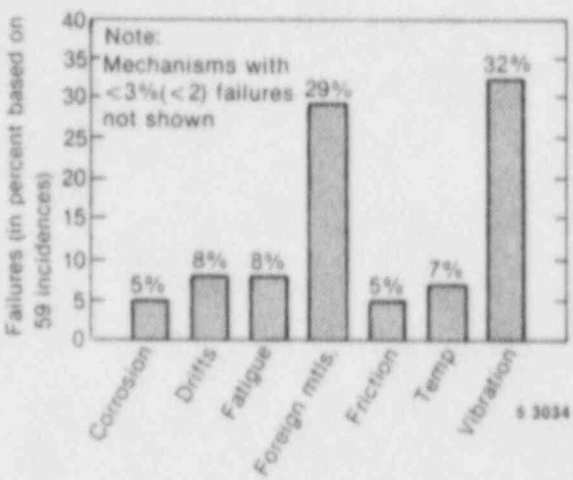

Figure 14. Reported failures by mechanisms for SI systems in 17 PWRs.

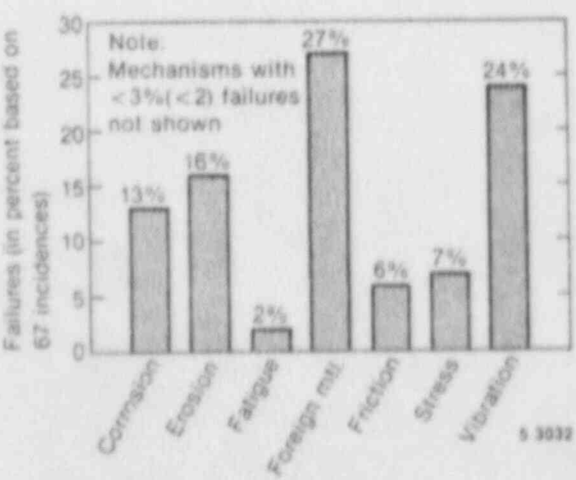

Figure 15. Reported failures by mechanism for CW systems from 17 PWRs.

circumstances that lead to failures in commercial plants but not in the test reactors surveyed. Also, possible answers to these questions are discussed.

Commercial Plant Interview Results. To minimize subjectiveness on the part of the interviewer a questionnaire modeled after that devised by D. L. Berry of SNL for the Plant Aging Workshop[12] was used. The primary difference is in the emphasis we placed on determining the cause(s) for each stated failure. Since the number of problems pointed out in each interview turned out to be small, it was very difficult to draw solid conclusions about the most important failures and their causes. In addition, we suspect that the interviewees may have tended to rate particular failures and their causes as important simply because they, themselves, were aware of the problem, not because of its overall safety significance. However, the data can be used to derive very general conclusions about what the major failure mechanisms may be. Table 1 lists each of the plants surveyed, both through the literature and by personal interview, and the *relative* magnitude of failures attributed to the major failure mechanisms. For the reasons cited above these should not be taken as absolute magnitudes.

The data accumulated by our interviews with commercial plant personnel tended to support findings from the published literature in two areas. First, the components that were most often cited as having failure problems were valves, pumps, and pipes. Second, foreign material and vibration were cited as among the major contributors to failure. However, the people we interviewed did not seem to agree with the results of the Aging Workshops conducted by SNL with regard to decalibration of pressure and temperature sensors or transmitters, nor on the importance of snubbers as an aging issue.[12] Our interviews brought forth very few references to instrumentation or snubbers as major aging concerns. The probable reasons for this are found in the criteria we used to judge age related failures for the purpose of this survey (refer to Appendix A). Another possibility may be in the specific goals of the workshops versus our survey and, therefore, in the choice of people interviewed. Our interviewees were all operators, maintenance people, or site engineers. The SNL workshop attendees, on the other hand, were mostly from national or private research laboratories, universities, Nuclear Steam Supply System (NSSS) vendors, or Architectural and Engineering (A/E) firms.

At the completion of most of our interviews the respondent was asked to relate the circumstances in which the problems described had developed. As

Table 1. Dominant failure mechanisms for each plant surveyed

Plant	Mechanism	Approximate Percentage[a]
Shippingport	Embrittlement	25
	Erosion	25
	Foreign material	25
	Oxidation	25
Humbolt Bay	Fatigue	50
	Foreign material	33
Dresden	Corrosion	30
	Foreign material	32
Yankee Rowe	Corrosion	17
	Vibration	27
Big Rock Point-1	Foreign material	50
San Onofre-1	Corrosion	12
	Foreign material	21
	Vibration	30
Conn. Yankee (Haddam Neck)	Erosion	19
	Foreign material	26
	Temperature	16
	Vibration	19
Monticello	Foreign material	48
Cooper	Foreign material	45
	Temperature	13
	Vibration	16
Davis-Besse	Foreign material	29
	Vibration	43
MTR, ETR, ATR (combined)	Corrosion	15
	Exceeded design life	35
	Erosion	15
	Foreign material	15
LOFT	Corrosion	31
	Exceeded design life	31

a. Since, in most instances, the number of reported occurrences for each mechanism was quite small, these data should be considered only as an indicator of relative magnitude, not as an absolute value.

the interview portion of the survey progressed a recurring pattern started to evolve. Many components were failing in the heatup and cooldown (transient) phases of plant operations. This was a totally unexpected result but is reasonable considering that the two major failure mechanisms reported are foreign materials and vibration. Higher than normal vibration is developed during the heatup phase, perhaps caused by water hammer or pipe thermal expansion, and during the cooldown phase due to throttling of pumps or pipe contraction. Since system vibrations are produced during transients, it is not surprising that failures due to foreign materials would also be present. This would be true since vibration could dislodge foreign materials (i.e., sand, silt, corrosion products, shellfish, etc.), which had previously settled out or that were adhering to pipe walls. The foreign materials thus freed into the system could erode system components (i.e., pipe elbows, valve seats, pump impellers, volutes, etc.), and resettle in valves, pumps, heat exchanger tubes, etc. This obviously could impede system operation.

The seemingly high incidence of component failure during plant transient operation will be discussed further in "Patterns of Occurrences Based on Interviews" section.

Test Reactor Interview Results. The same questionnaire was used for this set of interviews as was used for the commercial plant interviews and for the same reasons. Table 1 lists the predominant failure mechanisms cited for both the INEL Test Reactor Area [TRA—which includes Engineering Test Reactor (ETR), Materials Test Reactor (MTR), and Advanced Test Reactor (ATR)] and for the LOFT Facility.

For TRA the dominant cause of failure cited was that components had simply reached or exceeded the end of their expected design lives. That components would fail frequently for this reason in plants this old (average age—27 years) was not a surprise. But, since these plants experience transients often, what was expected and did not materialize, was that vibration would be cited as one of the major causes of failure. At LOFT we also expected to see a high incidence of component failures due to exceeded design life. This would be true not because it has been so long since fuel load but because much of the equipment was installed and used in nonnuclear tests for many years prior to initial criticality in 1977.

A major part of the corrosion statistic shown for LOFT in Table 1 is due to corrosion of the storage tank and carbon steel piping in the facility water supply system. These components were between 18 and 23 years old and had been exposed to the soil for most of that time. This type of failure is very similar to the corrosion effects given as a major (in terms of severity and cost to mitigate) problem area cited by one of the utilities interviewed.

Pattern of Occurrence Based on Interviews. In general, there were substantially fewer age related problems cited for the test reactors than we would have anticipated, considering their age and transient operation. Comparison of data from the commercial plants and from the test reactors revealed that the commercial plants displayed a much higher incidence of foreign material caused failure than did the test reactors. Why the difference? Discussion with TRA and LOFT personnel gave a plausible answer. In all of these facilities there was and is a concentrated effort to keep the systems clean. Obviously, effort in this direction would also be a benefit in commercial facilities.

Since LOFT has experienced more than 25 intermediate to large break loss-of-coolant experiments (LOCEs), why has LOFT, and to a lesser degree the TRA facilities not experienced the same magnitude of transient-induced failures as seem to exist for the commercial plants? LOFT should have proven to be an exceptional test facility for accelerated aging when considering transient operation, especially since the individual components are basically the same as those used in commercial plant systems. Why this has proven not to be the case was discussed with both LOFT and commercial plant personnel. There appear to be three reasons for this. The first is that the total operating time for LOFT is small. The second is that any fastener (i.e., nuts and bolts) problems that may have existed were found very early in the plant life and were corrected by design change, usually by installation of anti-rotational devices. The third and most important reason is that LOFT was designed and built for transient operation. Commercial plants are not. Each system at LOFT was designed and analyzed to minimize the effects of transient loads. Every system was designed with LOCE loads being considered as the ASME[a] upset condition. In commercial plants these types of loads would not be considered for normal operation. The lesson that can be learned from this is that transient induced failures, as well as other failures reported in this study, need not be accepted as a fact of life. They can be reduced by judicious design or design change.

a. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III.

SURVEY RESULTS

The nature of surveys do not lend themselves to finding a single solution to a single problem. This survey was no exception. There are intermediate results that, when taken alone, are not overly significant. Eventually these results, when taken together may suggest a pattern of events that allow the researcher to draw some conclusions regarding the problem under consideration.

As this project progressed it became obvious that the study of failures due to aging should not be a study of isolated component problems but should be coupled with a study of systems level problems.

Summary of Results

In the following, what we consider to be the major results are annotated with an asterisk (*) in the left margin.

For Commercial Plants. Presented below is a summary of the intermediate results for commercial plants of this survey, listed in roughly chronological order, which led us to the above conclusion.

1. Valves, pumps, and piping components have the highest incidences of failure due to aging effects. This is as expected. (Refer to Figure 3 and Reference 4)

2. Analysis of component related data did not point out specific types of equipment with an overwhelming aging concern that were not already being considered as part of another research program. Equipment failures followed a random pattern with respect to individual components.

*3. Of reported age related failures in commercial plants ~70% are the result of only four mechanisms; erosion, corrosion, vibration, and foreign materials. (Refer to Figure 9)

*4. High incidence of component failure in a plant system may not indicate a weakness in the component itself but rather a change in the system, its maintenance, or its mode of operation. The stresses resulting in component failure may not have been accounted for or may have increased with time.

*5. There appears to be a strong correlation between cause of failure for components and the functional system in which they are operating:

   a. RHR; 90% of failures are due to vibration and foreign materials. (Refer to Figure 10.)

   b. SIS; 61% of failures are due to foreign materials and vibration and another 30% due to corrosion and erosion. (Refer to Figure 11.)

   c. CWS; 51% of failures are due to foreign materials and vibration and another 30% due to corrosion and erosion. (Refer to Figure 12.)

6. There did not seem to evolve any pattern of correlation between specific plant and dominant failure mechanism. There were dominant failure mechanisms for specific functional systems but they were not plant specific.

*7. During the interviews with commercial plant personnel an underlying, recurring pattern evolved with respect to the occurrence of failure and the plant mode of operation. A review of the data from these interviews suggested that with certain failure mechanisms (e.g., water hammer, overnormal vibration, and chemistry control) were associated with high failure rates during the heat up and cooldown evolutions and cold shutdown modes of operation.

For INEL Test Reactors. Presented below is a summary of the intermediate results for test reactor plants of this survey, listed in roughly chronological order, which led us to the above conclusion.

1. The major cause of failures cited was that components had reached or exceeded the end of their expected lives. (That is, foreign material, vibration, or etc. effects were not cited as the cause for replacement.)

2. These facilities have not experienced the same magnitude of failure due to foreign materials as have commercial plants probably because there is a distinct effort made to keep them clean. This effort is necessary since test results can be affected significantly by relatively low leak rates. Cleanliness is enhanced, at least in the case of LOFT, by frequent large break loss-of-coolant experiments (LOCEs) that tend to *decrud* the system.

3. These reactors are continually in a transient mode of operation. However, they do not seem to experience the same magnitude of transient induced failures as do commercial plants. This is probably because test reactors are designed and analyzed for transient operation.

CONCLUSIONS AND RECOMMENDATIONS

The primary objective of this work was to identify aging issues and future research needs by conducting a survey of aged nuclear power plant facilities. The methodology used was to survey eight older commercial power plants by first analyzing plant operating experiences, as put forth in the published literature, and then to corroborate these results by actual field inquiry. This approach did not, nor was it intended to, produce results that could be considered beyond dispute in all regards. The intent was merely to go to the detail necessary to point out, with a basis in fact, some currently unexplored aging issues and to recommend the direction in which future aging research should proceed.

In the published literature (i.e., LERs and other plant history data files) component loss of function (failures) is well documented but detailed evaluation of plant systems with regard to failure causes is not. The data available is, however, adequate for issue identification.

Conclusions Drawn From This Limited Scoping Study

Conclusions that were drawn during the course of the survey as they relate to identification of aging issues are listed below. The "Recommendations for Future Research" section contains a set of recommendations for needed aging research.

1. Since commercial power plant system environments are directly responsible for most component failures in fluid-mechanical systems, examination of individual components to determine failure mechanisms should be supplemented with aging/systems interaction studies. System design, maintenance, and operational problems are so predominant that it is probable that failures due to the aging of component materials could not be identified with any certainty. Only after the effects of the major failure mechanisms are mitigated can material analysis, coupled with an understanding of the stressors and environment, yield results that are a function of the material itself and not a function of the effects of the operating environment. The major contributors to component failure are corrosion, erosion, vibration and foreign materials in the system and the most effected components are valves, pumps, and pipes.

2. System cleanliness with regard to foreign materials and chemistry control should have strict limits placed on it and should be monitored as part of the normal maintenance procedures.

3. Judging from the number of vibration failures evidenced in the survey, flow and equipment induced vibration are a problem in plant operation. Review of procedures from the ASME Pressure Vessel Code (prior to OM3)[16] dealing with structural analysis, indicate flow and equipment induced vibration would have been approximated in the original design and analysis phase. Systems are typically operated at normal, expected conditions and *walked down* to determine the additional supports required. This determination is, therefore, subjective based on the personnel performing the on-line *analysis*. Rigid supports designed for thermal growth, seismic vibration, and dead weight may not provide sufficient dampening for these vibrations. In addition, flow induced vibration levels can change with time due to such things as pump impeller wear. The number of vibration failures evidenced from the study indicates this area may need additional effort.

4. Prevention of vibration and thermal cycle effects could be enhanced by anti-rotation features being added to all fasteners on all safety or safety-related components in the plant. In addition, all new safety related items should be purchased with fastener lock features. Fastener locks should be checked periodically as part of normal maintenance procedures.

5. Any changes contemplated for the system or component design, operation, or maintenance (including those dictated by regulations) must take into account possible adverse effects on every other component in the system and in every related system. These changes must be verified against the original design specifications since fragility parameters such as fatigue life can be adversely effected. (Fatigue life is determined from a time related analysis of

physical, thermal and other transient conditions.) System and component interactions are much more prevalent and much more subtle than most realize.

6. Condition monitoring as discussed by Sugarman, et al.,[14] has some obvious advantages and should be considered as a valuable part of any comprehensive surveillance program. In light of the results of this survey, however, it also has some disadvantages:

   a. Since the conditions that govern component performance in todays plants are primarily system effects and system conditions can change rapidly, component condition monitoring alone would not be adequate. Monitoring of system conditions would also be necessary to assess the condition of the system. Systems monitoring may also prove to be a practical and cost effective approach.

   b. Even given the unlikely occurrence of system conditions staying steady for some period of time, not all system components degrade linearly with time. The appropriate time/degradation function would have to be determined for each susceptible component.

   c. To be of maximum benefit each component in each safety and safety-related system would have to have its degradation rate characterized over its full design life and over all possible normal operating and design basis environments.

Recommendations for Future Research

The results of this survey are to be used by the USNRC to implement a comprehensive research program that will systematically identify aging and service wear effects, which are likely to affect plant safety. The results of this survey will also help to identify what methods of inspection and surveillance would be most effective in detecting significant aging effects prior to the loss of function. To this end, the following recommendations are made:

1. The stressors that develop with time and, singularly or cumulatively, affect component availability, in a significant number of cases, have been due to functional system operation or deterioration. For instance, functional systems operating with low levels of vibration, in time, have resulted in high cycle/low stress fatigue in components, loosening of fasteners, and shifts in setpoints or calibration. Likewise, a small amount of foreign material (sand, silt, marine growth, etc.) built up in a fluid system with time can result in the start of corrosion in low spots and in entrapment areas. Corrosion products, in turn, add foreign material to the system that can be erosive. Erosion adds to the buildup. Thus, the *cumulative* effect can eventually create an environment in which system components cannot function. The evidence (reports of failed components) often specify only the stressors not the root cause of failure. It appears from our preliminary evaluation that most root causes of component failure are preventable and/or correctable. The survey results, to date, are preliminary indicators of expanded research needs and, as such, require further confirmation.

   In addition, the results should be weighted by years or hours of operation or some other appropriate weighting factor to avoid possible misrepresentation of the magnitude of the problems. The near-term benefit of improving equipment reliability by improving operating environments without major redesign can not be overstated.

2. The results of this survey suggest a philosophy that, though not new in general, does differ from the traditional way of thinking about aging issues. This philosophy is to consider a systems approach. Specifically, to improve the reliability of individual components and reduce the magnitude of the system induced causes first. Therefore, additional work should be initiated to characterize

systems, prioritized by component failure rate or safety significance, to provide insight into the true root cause(s) for the major reported failure mechanism(s). For example:

a. Vibration *from* - - - -
b. Contamination *by* - - - -
c. Corrosion *from* - - - -
d. Erosion *due to* - - - -

From this, guidelines for mitigation of the cause and for prevention of a reoccurrence could be prepared. One of the possible conclusions, certainly, may be to recommend redesign of certain components but it is not the only possibility. To come to this conclusion prematurely may be to treat the symptom not the cause.

3. The present concept of condition monitoring should be expanded to include monitoring system level parameters that can most directly affect component reliability. This would require research to determine which parameters should be monitored for each type of component in each system and to determine the most appropriate method to be used.

4. The personnel contributing to this survey (Appendix D) have experience with plant operations and with equipment problems associated with plant operation. This experience aided in making the following observation and recommendation. The heatup and cooldown cycles and cold shutdown period of plant operations were starting to appear as times of high stress and subsequent failure of components. This aspect of equipment failure was not explored sufficiently in this study. Further analysis of these plant operating modes would provide a necessary part of the basis for guidelines that would aid in prevention and mitigation of age related equipment failure.

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APPENDIX A
DESCRIPTION OF SURVEY METHODOLOGY

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Aged Facility Identification

The facilities chosen for this study included 32 operating[a] commercial nuclear plants at 23 sites and four test reactors at two INEL sites. The rationale for the choice of each of these facilities is given below.

Commercial Plants. Thirty-one commercial plants at 22 locations were selected for this study by use of the following criteria. These plants were chosen to be representative of a wide range of commercial nuclear reactors with respect to:

1. *Age*: The oldest plants that contain components and systems similar to modern designs

2. *Type*: Both PWR and BWR facilities

3. *Site Location*: As many different site locations as practical since it was judged that external environments and the sources of circulating and service water could have an effect on component failure rates

4. *NSSS*: All four major U.S. Nuclear Steam Supply System vendors.

5. *Architect/Engineer*: As many different A/E's as practical since it is possible that balance-of-plant design could affect reliability of safety related components.

In addition to the above, one additional commercial plant (Davis-Besse) was added to the data base late in the survey due to acquisition of information from that facility.

The 32 plants selected are listed by date of first commercial operation in Table A-1.[A-1]

Test Reactors. Four test reactors at the INEL site were also included to some degree in the survey. These facilities are:

1. Engineering Test Reactor (ETR),

2. Materials Test Reactor (MTR),

3. Advanced Test Reactor (ATR),

4. Loss of-Fluid Test (LOFT) Facility.

The rationale for including these facilities was:

1. They all can be considered as aged either in terms of actual operating time or in terms of numbers of transients experienced

2. They are geographically close, therefore plant records would be readily accessible

3. There are numerous personnel who have been actively involved in operating and maintaining these facilities still available at INEL for personal interviews.

Information Source Identification

There are many possible ways to obtain the information necessary to obtain a reasonable indication of aging issues. Among these are:

1. Surveys of *experts* from within the utility industry

2. Surveys of operating experiences based on the published literature

3. Examination of actual plant operating and maintenance records

4. Interviews with plant operating and maintenance personnel

5. Detailed examination of many aged components taken from many plants and plant systems

6. Some combination of the above.

Chosen as the most practical method having a high probability of getting valid results was a survey based on published literature, backed by interviews

a. All operational at the time of this writing except Shippingport, Dresden-1, Humbolt Bay-3, and TMI-1 and -2.

Table A-1. Aged commercial LWR facilities considered (by date of operation)

Plant Name	Operation Date	Cooling Source	NSSS Vendor	Location
Boiling Water Reactors				
Dresden-1[a]	8-60	Reservoir (Once through)	GE	Morris, IL
Big Rock Point[a]	12-62	Towers (Mechanical)	GE	Charlevoix, MI
Humbolt Bay[a]	8-63	Humbolt Bay (Once through)	GE	Eureka, CA
Oyster Creek	12-69	Barnegat Bay (Once through)	GE	Toms River, NJ
Monticello[a]	6-71	Towers (Mechanical)	GE	Minneapolis, MN
Cooper[b]	7-74	Missouri River (Once through)	GE	Nebraska City, NE
Peach Bottom-2	7-74	Towers (Mechanical)	GE	Lancaster, PA
Peach Bottom-3	12-74	Towers (Mechanical)	GE	Lancaster, PA
Browns Ferry-1 Decatur, AL	-74	Combined cycle[b]	GE	
Browns Ferry-2 Decatur, AL	-75	Combined cycle[b]	GE	
Hatch-1	12-75	Towers (Mechanical)	GE	Baxley, GA
Browns Ferry-3 Decatur, AL	3-77	Combined cycle[b]	GE	
Hatch-2	8-79	Towers (Mechanical)	GE	Baxley, GA
Pressurized Water Reactors				
Shippingport[a]	12-57	Ohio River	W	Pittsburg, PA
Yankee Rowe[a]	7-61	Deerfield River	W	Greenfield, MA

Table A-1. (continued)

Plant Name	Operation Date	Cooling Source	NSSS Vendor	Location
Pressurized Water Reactors (continued)				
San Onofre-1[a,c]	1-68	Pacific Ocean	W	San Clemente, CA
Haddam Neck[a] (Conn. Yankee)	1-68	Connecticut River	W	Meriden, CT
R. E. Ginna	7-70	Lake Ontario	W	Rochester, NY
Point Beach-1	12-70	Lake Michigan	W	Manitowoc, WI
Robinson-2	3-71	Reservoir	W	Hartsville, SC
Palisades	5-71[d]	Towers (Mechanical)	C-E	Kalamazoo, MI
Point Beach-2	10-72	Lake Michigan	W	Manitowoc, WI
Main Yankee	12-72	Back River Tidal flow	C-E	Wiscasset, ME
Oconee-1	7-73	Reservoir	B&W	Greenville, SC
Fort Calhoun-1	9-73	Missouri River	C-E	Omaha, NE
Zion-1	10-73	Lake Michigan	W	Waukegan, IL
Zion-2	9-74	Lake Michigan	W	Waukegan, IL
Three Mile Island-1	9-74[e]	Towers (Nat. & mech.)	B&W	Middletown, PA
Oconee-2	9-74	Reservoir	B&W	Greenville, SC
Oconee-3	12-74	Reservoir	B&W	Greenville, SC
Davis-Besse-1[b]	3-77 (Natural)	Tower	B&W	Toledo, OH
Three Mile Island-2	12-78[d]	Towers (Nat. & mech.)	B&W	Middletown, PA

a. Plants selected for initial analysis.

b. Plants surveyed by personnel interview.

c. Mechanical Towers & Tennessee River.

d. Restricted levels until 3-73.

e. Plant operation suspended since 3-79.

with plant personnel. Inherent in this choice of approach was the need to (a) identify the published data to be reviewed, (b) to define the criteria by which the pertinent aging information would be extracted and (c) to determine what specific questions should be asked of the interviewees.

To be practical and nonrepetitious of other ongoing work the sources of data to be used in this study must be:

1. Condensed. That is, the sources contain in themselves plant information condensed from many other sources.

2. Minimally influenced by reporting requirements that could influence either the numbers of incidents reported or the quality of the reported information.

After reviewing several documents and data bases in detail it was decided to use two primary sources of data. The first of these was the Nuclear Power Experience (NPE)[A-2] published by the S. M. Stoller Corporation, Boulder, Colorado, the second was all the USNRC Inspection/Enforcement (IE) documents published to date.

In order to corroborate the aging data thus acquired from the published literature, interviews of personnel, primarily with operating and maintenance experience, were to be conducted. However, as with the review of the published data, subjectiveness on the part of the reviewer had to be minimized. This could be best accomplished by use of a structured questionnaire. As a separate part of the NPAR Program, Sandia National Laboratory (SNL) has conducted several workshops wherein knowledgable industry representatives were asked to identify and rank current aging issues.[A-3] To minimize potential discrepancies between results we chose to use basically the same questionnaire as was used in the SNL study.

Automated Data System Implementation

An initial review of the source material needed for this survey as identified in "Published Data Analysis" section of the main text revealed a considerable amount of data associated with age related component failures would have to be filed. Management of this data for analysis would require that some type of data base be established. The data base would need tne capability for storage, search, retrieval, and correlation. Since a review of existing computer data files was made and none were found that appeared adequate for this specialized work, a new computerized data file was initiated for this task.[A-4] The file uses the codes and basic data points suggested by the USNRC Aging Research Technical Coordinator. It was programmed to search on, and compare, all data inputs and up to four separate parameters. This was necessary to help in the correlation of seemingly unrelated failures that may have some of the same environments, materials, failure mechanisms, or other characteristics that can be identified as age related (the data base system and failure mechanism codes used are given in Appendices B and C respectively).

It should be noted that this data base was designed as a tool to help manage the information which, of necessity, would need to be accumulated. It was not intended as a product of the survey. However, as the survey progressed it became evident that this *tool* could be expanded to fulfill one of the NPAR near-term objectives as given in "Introduction" of the main text.

Conduct of Aged Power Plant Surveys

The initial step was to extract from the Nuclear Power Experience[A-2] all the data related to the plants chosen for review (see Table A-1) and to place these data in a separate hard copy file cataloged by affected facility.

Information Input and Analysis. Once the pertinent published information was extracted the criteria by which the reports could be judged as age related were defined. In order to minimize subjectiveness on the part of the reviewer every piece of data reviewed had to have the same set of questions asked of it and had to be either accepted for inclusion in the data base or be rejected based on the same criteria. The screening criteria used were:

1. NPE reports were excluded if they did not pertain directly to one of the plants being reviewed. (All pertinent IE material was included, none was rejected based on affected plant.)

2. Reports were excluded if the problems occurred prior to the date of initial criticality for the respective plant. (However, we should be careful not to overlook the aging that takes place during the long periods prior to startup.)

3. Reports were excluded if they contained informational items rather than component failures. An example of an informational item is a report stating that, "a scram test was not performed on or before the date required."

4. Reports were excluded if they described conditions that could not be characterized as complete failure. Conditions requiring readjustment or recalibration were, therefore, excluded.

5. Reports were excluded if they described failures caused by noncompliance with the manufacturer's recommendations for application, operation, or maintenance.

6. Reports were excluded if the failed part or component had already exceeded its design service life.

7. Reports were excluded if they described failures of components and/or instrument systems considered of lesser importance (i.e., not safety related). Chemical monitors such as hydrogen, oxygen analyzer systems, pH monitors, and various other sample systems for environmental monitoring were, therefore, excluded.

8. Reports were excluded if they described failures of components that monitored parameters outside the reactor building or primary containment. Component failures involving seismic and meteorological monitors were, therefore, excluded.

The next step taken was to read in detail each of the ~7800 entries previously cataloged and to apply the selection criteria outlined above. To start the analysis, all applicable NPE reports for eight of the oldest plants containing components and systems similar to modern designs [four PWRs and four BWRs annotated by footnote (a) in Table A-1] plus all IE material, was coded and entered into the data base (698 reports). Entry of the NPE and IE data was suspended at this point for preliminary analysis. The initial analysis effort was to review the data for age related equipment or structure degradation with recurring failure patterns. The emphasis was on isolating specific equipment or structures with the highest failure rates. The result of this portion of the analysis was, as expected, in agreement with the Oak Ridge National Laboratory (ORNL) analysis of nuclear plant aging trends.[A-5] The conclusion was, excluding electrical drift, valves, pumps and piping are the generic families having the highest failure rates (see Figure 3 main text). Further analysis of the data did not point out any other specific type of equipment with an overwhelming aging concern that was not already a continuing research item of another program. The failures for all generic families or equipment followed a random pattern with respect to any individual component. What did start to materialize was a very specific recurring pattern of failure mechanisms. Four failure mechanisms were responsible for 70% of the reported plant equipment problems. These mechanisms were corrosion, erosion, vibration, and foreign materials (either chemical or solid contamination).

To further validate our initial results three plant systems were chosen for more detailed scrutiny. The systems chosen were:

1. Residual Heat Removal (RHR) System—BWRs

2. Safety Injection System (SIS)—PWRs

3. Cooling Water Systems (CWS)—PWRs

The criteria in order of priority by which these systems were chosen are:

1. Must be safety related systems (RHR, SIS, CWS)

2. Must not have shown a high rate of failure due to specific mechanisms that are the subject of other research programs, [i.e., steam generator tube corrosion and cracking, fatigue, cyclic crack growth of piping, etc., (RHR, SIS, CWS)]

3. Must have shown a relatively high rate of failure after data adjustment for 2. above (RHR, SIS, CWS)

4. Must include at least one BWR unique system (RHR)

5. Must include at least one PWR unique system (SIS)

6. Must include at least one emergency standby system (SIS, RHR in some modes)

7. Must include at least one continuously operating system (CWS, RHR in some modes)

8. It is preferable to choose systems that are dissimilar with each other with respect to operation and environments (see Table A-2). To this end, the NPE reports describing failures within the RHR systems of nine additional BWRs and within the SIS and CWS systems of thirteen additional PWRs were encoded and added to the data base (182 additional reports). These plants are listed in Sections I and II of Table A-1.

The results of this more detailed study of the three systems confirmed our previous findings within reasonable limits. Therefore, we gained sufficient confidence in our initial findings to judge that the full compliment of plants for which hard copy data had been compiled need not be encoded and analyzed at this time.

Plant Personnel Interviews. As indicated earlier, the data survey was to consist of a review of operating plant histories based on the published literature backed by interviews with plant personnel. The primary value of the interviews, since they are limited in amount of data obtained and are somewhat subjective, is to corroborate the much greater volume of more objective data from the published literature. Though contacts with utility sources had been made and nurtured throughout the course of this survey, no serious attempt had yet been made to acquire data from those with actual operating and maintenance experience.

Initially, contacts with commercial facilities were limited to establishing communication and discussing aging in general terms. From the SNL workshop

Table A-2. System dissimilarities—RHR, SIS, CWS

Parameter	Residual Heat Removal System (BWR)	Safety Injection System (SIS)	Cooling Water Systems (PWR)
Operation	Emergency standby (LPCI, containment spray) Periodic use (shutdown cooling, steam condensing, suppression pool cooling)	Emergency standby	Continuous flow
Flowing medium	Reactor water (clean, demin.)	Borated water	Chemically treated water
Operating temperature	<350°F	100°F	<125°F
Operating pressure	<450 psi	1700 psi	<60 psi
Flow rates	High (>40 K gpm) Low (<4 K gpm)	<600 gpm	(System dependent)

experiences[A-3] and from initial contacts with utilities, it was judged that maximum benefit from personnel interviews would not be realized until the literature search portion of the project had identified specific issues of concern. Toward the end of the survey period these facilities were contacted and ask for input on identification of suspected causes of failure.

The facilities chosen as the objects of our interviews are given below along with the type of plant and the year of first criticality.

1. Three commercial facilities:
   San Onofre; PWR; Westinghouse, 1968
   Cooper; BWR; General Electric, 1974
   Davis-Besse; PWR; Babcock & Wilcox, 1977

2. Four test reactors:
   MTR; low pressure PWR; 1950
   ETR; low pressure PWR; 1957
   ATR; low pressure PWR; 1965
   LOFT; PWR (based on *W* design); 1977

A possible significant difference between the facilities listed above aside from age and type is the source of circulating water; San Onofre uses salt water; Cooper a flowing river; Davis-Besse a large body of fresn water and the test reactors use deep wells. As Mr. Feldman[A-6] points out there are statistically significant differences in performance between salt and nonsalt water cooled plants. It seems reasonable, therefore, to assume similar differences could also exist between plants with clean versus silt-laden water sources. In fact, these differences were reflected in the results of the interviews.

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APPENDIX B
MAJOR SYSTEM CODES USED IN THIS SURVEY

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MAJOR SYSTEM CODES USED IN THIS SURVEY

The following is a listing of the codes used in the data base created for this survey. They are included for reference purposes.

I. The codes used for Boiling Water Reactors and their systems are:

Nuclear Systems—N

N01 Reactor core
N02 Control rod drive system
    Control rod drive hydraulic system
N03 Reactor control system
N04 Reactor recirculation system
N05 Standby liquid control system
N06 Reactor protection system
N07 Neutron monitoring nuclear instrumentation system
N08 Residual heat removal/low pressure safety injection
N09 Reactor water cleanup system

Engineered Safety Systems—S

S01 Reactor core isolation cooling system
S03 Engineered safety features
    Safety injection system
    High-pressure coolant injection/core spray system
    Low-pressure coolant injection
    Low-pressure core spray system
    Automatic depressurization system

Containment Systems—C

C01 Primary containment and penetrations
C02 Reactor building
C03 Containment heat removal system
C04 Containment isolation system
C05 Containment purge system
C07 Combustible gas control system
C08 Containment ventilation system
C10 Containment spray system

Electrical Systems—E

E01 Main power system
    Protection relaying and controls
E02 Plant ac distribution system
    Essential power system
    Nonessential power system
    High-pressure core spray power system
    Protective relaying and controls

Electrical Systems—E (continued)

E03 Instrumentation and control systems
  dc power system
  Vital dc power subsystem
  Plant dc power subsystem
  Instrument ac power system
  Vital instrument ac power subsystem
  Plant instrument ac power subsystem
E04 Emergency power system
  Diesel-generator fuel oil subsystem
  Diesel-generator cooling water subsystem
  Diesel-generator air subsystem
  Diesel-generator lubrication oil subsystem
E05 Plant lighting system
  Essential lighting
  Nonessential lighting
E06 Plant computer
E07 Switchyard
  dc control power system
  Protective relaying

Power Conversion Systems—P

P01 Main steam system
P02 Turbine-generator system
  Electro-hydraulic control subsystem
  Turbine gland seal subsystem
  Turbine lubrication subsystem
  Stator (hydrogen) cooling subsystem
  Hydrogen seal oil subsystem
P04 Condenser and condensate system
  Condenser evacuation system
  Condensate cleanup/polishing system
  Condensate heater drain subsystem
P05 Feedwater system
  Feedwater heater drains subsystem
P06 Circulating water system

Process Auxiliary Systems—W

W01 Radioactive waste system
  Gaseous radwaste system offgas subsystem
  Liquid radwaste system
  Solid radwaste system
W02 Radiation monitoring system
  Plant area radiation monitors
  Environmental radiation monitors
  Process radiation monitors
W03 Cooling water systems
  Reactor building cooling water system
  Turbine building cooling water system
W04 Service water systems
  Demineralized makeup water system
  Station service water
  Essential service water system

Process Auxiliary Systems—W (continued)

Nonessential service water system
Chilled water system
W05 Refueling system
W06 Spent fuel storage system
Fuel pool cooling and cleanup system
W07 Compressed air system
Service air system
Instrument air system
W09 Plant gas system
Nitrogen system
Hydrogen system

Plant Auxiliary Systems—X

X02 Fire protection system
Water system
Carbon dioxide system

II. The codes used for Pressurized Water Reactors and their systems are:

Nuclear Systems—N

N01 Reactor core
N02 Control rod drive system
N03 Reactor control system
N04 Reactor coolant system
N05 Emergency boration system
N06 Reactor protection system
N07 Nuclear monitoring nuclear instrumentation system
N08 Residual heat removal/low pressure safety injection
N09 Chemical and volume control system (CVCS)

Engineered Safety Systems—S

S02 Engineered safety features actuation system
S03 Safety injection system

Engineered Safety Systems—S (continued)

S03 Safety injection system (continued)
High-pressure safety injection subsystem
Safety injection tank/core flood subsystem
Low-pressure safety injection subsystem

Containment Systems—C

C02 Reactor building/containment and penetrations
C03 Containment cooling system
C05 Containment purge system
C07 Combustible gas control system
C08 Containment ventilation system
C10 Containment spray system

Electrical Systems—E

E01 Main power system
- Protection relaying and controls

E02 Plant ac distribution system
- Essential power system
- Nonessential power system
- High-pressure core spray power system
- Protective relaying and controls

E03 Instrumentation and control systems
- dc power system
  - Vital dc power subsystem
  - Plant dc power subsystem
- Instrument ac power system
  - Vital instrument ac power subsystem
  - Plant instrument ac power subsystem

E04 Emergency power system
- Diesel-generator fuel oil subsystem
- Diesel-generator cooling water subsystem
- Diesel-generator air subsystem
- Diesel-generator lubrication oil subsystem

E05 Plant lighting system
- Essential lighting
- Nonessential lighting

E06 Plant computer

E07 Switchyard
- dc control power system
- Protective relaying

Power Conversion Systems—P

P01 Main steam system

P02 Turbine-generator system
- Electro-hydraulic control subsystem
- Turbine gland seal subsystem
- Turbine lubrication subsystem
- Stator (hydrogen) cooling subsystem
- Hydrogen seal oil subsystem

P04 Condenser and condensate system
- Condenser evacuation system
- Condensate cleanup/polishing system
- Condensate heater drain subsystem

P05 Feedwater system
- Feedwater heater drains subsystem

P06 Circulating water system subsystem

Process Auxiliary Systems—W

W01 Radioactive waste system
- Gaseous radwaste system offgas subsystem
- Liquid radwaste system
- Solid radwaste system

W02 Radiation monitoring system
- Plant area radiation monitors
- Environmental radiation monitors
- Process radiation monitors

W03 Cooling water systems
- Reactor building cooling water system
- Turbine building cooling water system

W04 Service water systems
- Demineralized makeup water system
- Station service water
- Essential service water system
- Nonessential service water system
- Chilled water system

W05 Refueling system

W06 Spent fuel storage system
- Fuel pool cooling and cleanup system

W07 Compressed air system
- Service air system
- Instrument air system

W09 Plant gas system
- Nitrogen system
- Hydrogen system

Plant Auxiliary Systems—X

X02 Fire protection system
- Water system
- Carbon dioxide system

APPENDIX C
FAILURE MECHANISM CODES USED IN THIS SURVEY

APPENDIX C
FAILURE MECHANISM CODES USED IN THIS SURVEY

The codes used for failure mechanisms in this survey and the data base are:

CORR	Corrosion
DRFT	Drift
EARC	Electrical arc
EMBR	Embrittlement
EDFL	End of life
EROS	Erosion
FATG	Fatigue
FRMA	Foreign material
FRTN	Friction
HRNG	Hardening
HTEM	High temperature
OTHR	Other[a]
OXID	Oxidation
STRS	Stress
STCR	Stress corrosion[a]
TEMP	Temperature
UNKN	Unknown[a]
VIBR	Vibration
WEAR	Wear[a]
WTHM	Water hammer

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a. Failure mechanisms not used in data analyses (refer to "Nomenclature" section of main text for rationale).

APPENDIX D
AGED FACILITY SURVEY REVIEW GROUP

APPENDIX D
AGED FACILITY SURVEY REVIEW GROUP

The following is a listing of people contributing to the interviews of plant personnel.

J. A. Hunter—Eighteen years experience in reactor systems design and related experience in dynamic and environmental simulation. Currently, EG&G Idaho Manager for Equipment Qualification Programs, which includes aging studies.

Dr. W. A. Reuter—Twenty-five years nuclear and aerospace materials technology experience. Currently Technical Leader for structural materials for the EG&G Idaho Materials Science Division.

R. A. Livingston—Twenty years experience in nuclear engineering specializing in pump and valve design, operation and maintenance.

K. C. Sumpter—Twenty years nuclear plant experience in the materials engineering area. Currently EG&G Idaho manager for TMI-2 technical support under the sponsorship of the DOE.

D. E. Kudera—Twenty-six years nuclear facility experience (both commercial and test reactors) specializing in nuclear chemistry.

T. L. Cook—Eight years experience as commercial power plant Operator five of which were as a Senior Reactor Operator. Also five years experience in review of commercial power plant inservice test plans for the USNRC.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 — U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any): NUREG/CR-3819 EG&G-2317

2. TITLE AND SUBTITLE: Survey of Aged Power Plant Facilities

3. LEAVE BLANK

4. DATE REPORT COMPLETED: MONTH January, YEAR 1985

5. AUTHOR(S): James A. Rose, Robert Steele, Jr. Kevin G. DeWall, Bruce C. Cornwell

6. DATE REPORT ISSUED: MONTH March, YEAR 1985

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code):
EG&G Idaho Inc.
P.O. Box 1625
Idaho Falls, ID 83415

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER: FIN No. A6389

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code):
Department of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

11a. TYPE OF REPORT: Technical

b. PERIOD COVERED (Inclusive dates): FY 84

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This report presents the results of the survey of Aged Nuclear Power Plant Facilities conducted for the USNRC Office of Nuclear Regulatory Research. The results of this report recommend methods to help formulate a comprehensive research program that will systematically identify aging and service wear effects which are likely to affect plant safety. The survey centered on safety related plant systems with regard to component failures from operating histories.

The age related failure information gathered from the plant histories was analyzed for reoccurring failure patterns. Emphasis was on identification of specific equipment with high failure rates and of failure mechanism relationships. The data would not support specific equipment identification. It did imply a direct relationship between failure and failure mechanism. 70% of the failures reported were due to four failure mechanisms. In addition there appeared to be a strong correlation between cause of failure and the system in which the component operates. This is verified by detailed study of several plant systems and corroborated by personnel interviews.

This survey indicates identification and elimination of system level cause of component failure is a viable approach to prevent and mitigate major reported aging effects.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS:
Aged Nuclear Power Plants
Aging and Service Wear
operating histories
age related component failure
failure mechanisms

b. IDENTIFIERS/OPEN ENDED TERMS

15. AVAILABILITY STATEMENT: Unlimited

16. SECURITY CLASSIFICATION: (This page) Unlimited; (This report) Unlimited

17. NUMBER OF PAGES

18. PRICE

EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415