LETTER REPORT

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SUMMARIES OF RESEARCH REPORTS SUBMITTED IN CONNECTION WITH THE NUCLEAR AGING RESEARCH (NPAR) PROGRAM

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ABSTRACT

As a result of implementing the Nuclear Plant Aging Research (NPAR) program plan, the results of Phase I efforts for selected electrical and mechanical components have been published since 1984. To reduce the difficulty of maintaining cognizance of this wealth of information, summaries of each of 14 NUREG/CR reports have been developed and collated into this report. The purpose is to make the result of these studies more available for rapid survey, direct attention to specific reports of interest and for the utilization of research results in regulatory process.

The summaries of these reports are grouped into three categories. The first category involves the early scoping and background studies and includes: Survey of Aged Power Plant Facilities, Operating Experience Reviews of LERs to Identify Aging Trends, workshops to obtain experts opinions and Aging/Risk Considerations. The second category contains two reports. One on developing a Methodology for Aging Analysis and a second one on evaluation and the use of a signature analysis technique (MOVATS). The third category contains (Phase I) results of aging research on 9 components including: Electric Motors, Battery Chargers/Inverters, Electrical Cable, Pressure Transmitters, Diesel Generators, Motor Operated Valves, Check Valves, Auxiliary Feedwater Pumps and Snubbers. Each report summary has four sections: Background, Summary, Results/Findings, and Utilization of Research Results in the Regulatory Process.

This report is considered a living document. That is research results and additional summaries of reports may be added. In addition, selected future reports also may be summarized and incorporated periodically.

I. INTRODUCTION

The results of the Phase I efforts of the Nuclear Plant Aging Research (NPAR) program have been published since 1984. This report of summaries of these published reports was developed to provide an efficient method to communicate program results. This report is considered a living document. Additional summaries of selected future reports are planned to be incorporated into this report.

The overall United States Nuclear Regulatory Commission (USNRC) objective of aging research, as given in the USNRC Long Range Plan and the NPAR program plan (NUREG-1144), 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 and systems during service; and the potential impacts of degradation upon public safety. Specifically, the aim of this research is to develop methodologies to identify such potential impacts on safety, including the prevention or correction procedures, well in advance of their actual occurrence.

To meet this objective the Nuclear Plant Aging Research Program Plan (NUREG-1144) has been implemented under the sponsorship of the USNRC, Office of Nuclear Regulatory Research. The Nuclear Plant Aging Research Program Strategy is illustrated in Figure 1, including the Phase I and Phase II segments of the program. The program goals are to: (a) identify electrical and mechanical component and systems level aging effects likely to impair plant safety, (b) identify methods of inspection and surveillance of components and systems that will be effective in detecting significant aging effects prior to loss of safety function so that proper maintenance and

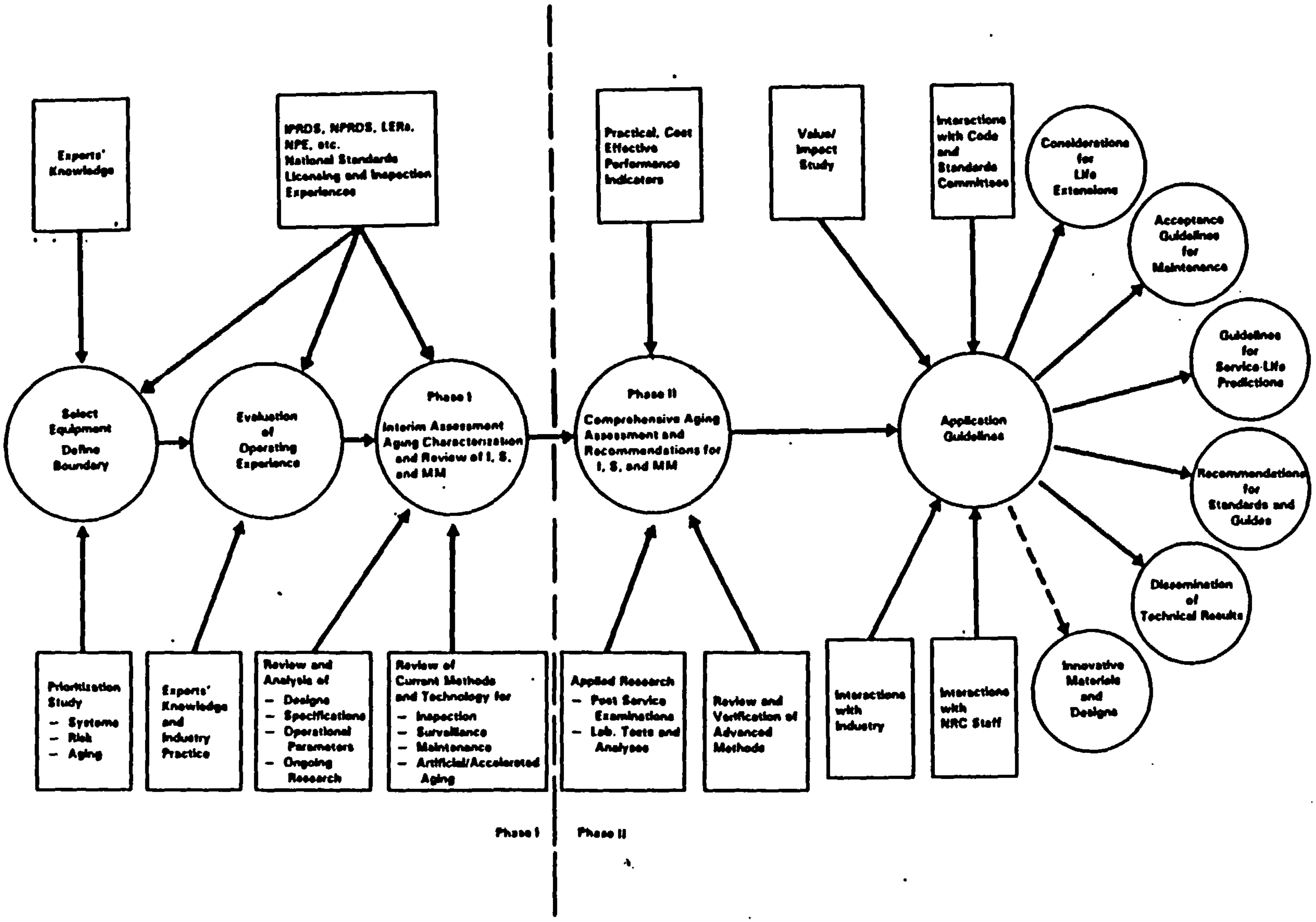


FIGURE 1 Nuclear Plant Aging Research Program Strategy

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. The initial effort in the Nuclear Plant Aging Research program was directed at: reviewing operating experience and experts opinion; establishing a data base containing the known and necessary information for aging assessments of nuclear power plant components and structures; and identifying and prioritizing aging issues and future research needs. The first category of summaries report on the results of this activity. They include the Survey of Aged Plant Facilities, Operating Experience Review of LERs to Identify Aging Trends, workshops to obtain experts' opinion and Aging/Risk considerations.

The second category of summaries, report on analytical and monitoring techniques that have been developed or are being tested as a part of the Nuclear Plant Aging Research program. One summary covers a status report on the development, by SEA, Inc. of a methodology for the evaluation of complex aging effects. The approach consists of the combined application of an interactive modeling technique, the N-Square diagram, and system engineering to develop a methodology for evaluation of complex aging systems interactions. The other summary covers the report that describes the results of a limited field test program to evaluate the Motor-Operated Valve Analysis and Test System (MOVATS) developed and commercialized by MOVATS Inc.

The third category includes summaries of the studies of nine key nuclear plant components. They include: electric motors, battery chargers/inverters, electrical cables, pressure transmitters, diesel generators, motor-operated valves, auxiliary feedwater pumps and snubbers. These reports characterize all the time dependent and environmental effects acting on these components. Also, methods of age degradation detection and monitoring are identified and evaluated for overall contribution to plant safety through mitigation of aging effects.

For uniformity of presentation, each of the report summaries contain four sections: background, a summary, results/findings and utilization of research results in the regulatory process.

The status of specific activities in the Nuclear Plant Aging Research program, as of March 1986, is presented in Table I.

TABLE 2
NUCLEAR PLANT AGING RESEARCH (NPAR)
Technical Progress Reports of Interest to NRR, IE, AEOD, REGIONS

#	COMPONENT/ SYSTEM/TOPIC	CONTRACTOR	REPORT NUREG	REPORT DRAFT	PROJECT STATUS	REMARKS
1	Motor Operated Valves	ORNL	/CR-4234		Phase I study is complete.	Some progress has been made in Phase II activity. It will have recommendations for S&M.
2	Use of Signature Analysis Technique-MOVATS	ORNL	/CR-4380		Project complete. No additional work is contemplated.	Results included in IE Bulletin 85-03.
3	Check Valves	ORNL	/CR-4302		Phase I study is complete.	
4	Aux. Feed Water Pumps	ORNL	/CR-4597	DRAFT	Phase I study is complete.	
5	Electric Motors	BNL	/CR-4156		Phase I study is complete.	Phase II, the effort will be confined to motors inside containment.
6.	Battery-chargers/ Inverters	BNL	/CR-4564	DRAFT	Phase I study complete.	The next phase will include testing of aged unit from Shippingport.
7.	Snubbers	PNL	/CR-4279		Phase I study is complete.	

TABLE 2 (Continued)

#	COMPONENT/ SYSTEM/TOPIC	CONTRACTOR	REPORT NUREG	REPORT DRAFT	PROJECT STATUS	REMARKS
8	Diesel Generators	PNL	/CR-4590	DRAFT	Part I of Phase I report complete.	Have identified # of sub-systems and components susceptible to aging. Has some recommendation for S&M.
9	Room Coolers	PNL	/CR-	DRAFT	Project complete. No additional work has been comtemplated.	Motors, fan belts, cooling coil have propensity for aging.
10	Transmitters	FRC ORNL	 /CR-4257-2	DRAFT	INEL will do detailed engineering studies, i.e., Phase I and Phase II.	These two projects were to review condition monitoring techniques only. Detailed aging assessments and recommendations for S&M will be addressed in Phase I - Phase II studies.
11	Cables	FRC	/CR-4257-1			
12	Aging/Risk Consideration	INEL PNL	/CR-4144		This specific task is complete.	INEL has new activity to identify where in plant systems aging is a safety concern.
13	Survey to Identify Aging Trends from LERs	ORNL	/CR-3543		Project complete.	Identified components and systems which have propensity for aging.
14	Results of Aging Workshop	SNL	/CR-3818		Project complete.	Identified susceptible components from experts' opinion.

TABLE 2 (Concluded)

#	COMPONENT/ SYSTEM/TOPIC	CONTRACTOR	REPORT NUREG	REPORT DRAFT	PROJECT STATUS	REMARKS
15	Kick-off Aging Workshop	SNL	/CR-0036		Project complete.	Comprehensive meeting with industry.
16	Survey of Aged Facilities	INEL	/CR-3819		Project complete.	Corrosion, erosion, vibration and contamination have resulted in aging of fluid mechanical systems.

ORNL = Oak Ridge National Laboratory
S&M = Surveillance and Maintenance
BNL = Brookhaven National Laboratory
PNL = Pacific Northwest Laboratories
FRC = Franklin Research Center
INEL = Idaho National Engineering Laboratory

2.1 Aged Plant Survey NUREG/CR-3819 June 1985 INEL/FIN No. A6389

Survey of Aged Power Plant Facilities

1.0 Background

As part of the overall Nuclear Plant Aging Research Program, the Idaho National Engineering Laboratory (INEL) was asked to perform a survey of aged light water reactor (LWR) power plant facilities. They were 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 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.

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.

Licensee Event Report (LER) data either have previously been used 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. 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) 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 problems or, (b) of high enough significance that action should be taken to assure the problem cannot arise.

2.0 Summary of Approach

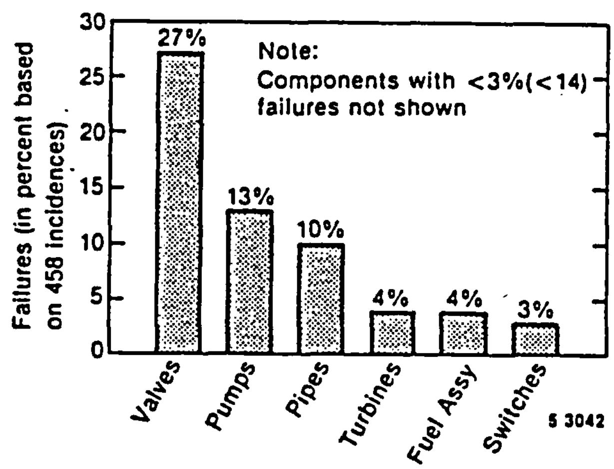
The results of this report recommend methods to help formulate a comprehensive research program that will systematically identify aging and service wear effects which are likely to affect plant safety. The survey centered on safety related plant systems with regard to component failures from operating 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 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 disput 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.

As 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.

In a paper by E. J. Brown of the USNRC, Office of Analysis and Evaluation of Operational Data (AEOD), he 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. 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 below) we limited this portion of the search to these items.

REPORTED FAILURES BY COMPONENT FOR 4 BWRs AND 4 PWRs

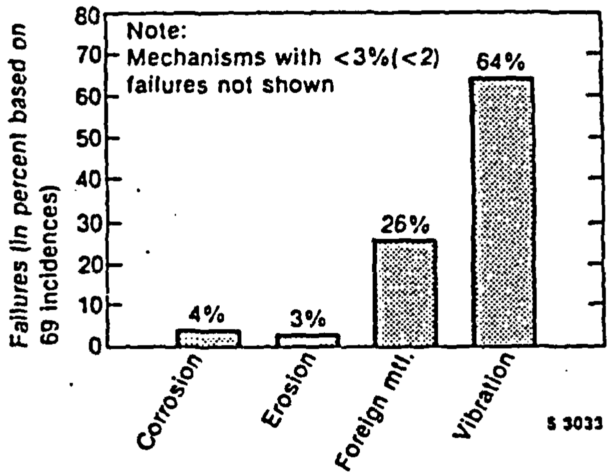


3.0 Results/Findings

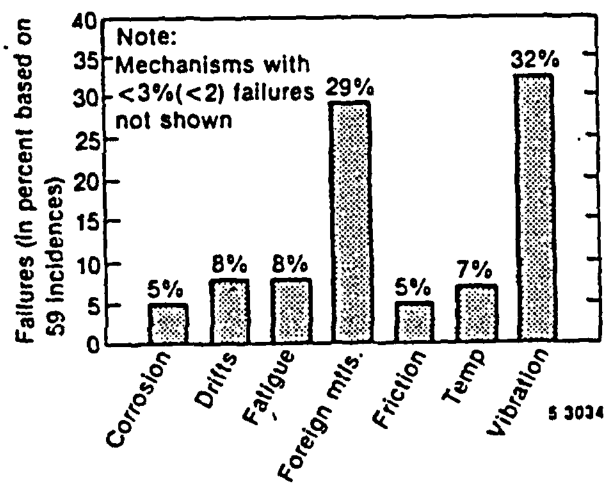
A result of the early analysis was, as expected, in agreement with the Oak Ridge National Laboratory (ORNL) findings on nuclear plant aging trends. But, in addition, a very specific recurring pattern of failure mechanisms (causes) emerged. Four failure mechanisms were responsible for 70% of the reported plant equipment problems in fluid-mechanical systems. These mechanisms were corrosion, erosion, vibration, and foreign materials (either chemical or solid contamination).

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; (2) the Safety Injection System (SIS) for PWRs; (3) the Cooling Water System (CWS) for PWRs.

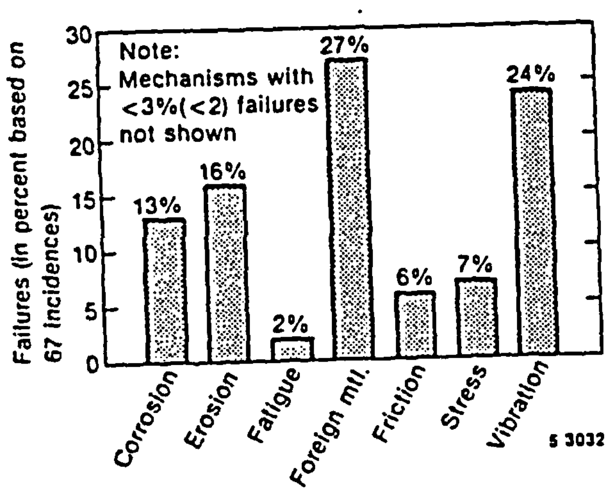
Figures



Reported failures by mechanisms for RHR systems in 13 BWRs.



Reported failures by mechanisms for SI systems in 17 PWRs.



Reported failures by mechanism for CW systems from 17 PWRs.

Additional findings include:

1. 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.

2. There appears to be 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].

3. 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.

4. 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:

1. 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.

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 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.

4. 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.

5. 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.

4.0 Utilization of Research Results in the Regulatory Process

The results of this and subsequently planned studies will be useful to the regulatory process and can be factored into this process through support of implementation of:

a. 10CFR.50.49(e)(5) - Aging

b. 10CFR.50.36(e) - Surveillance Requirements

c. 10CFR.50, Appendix B - Quality Assurance Criteria

2.2 LER Data-Aging Trends NUREG/CR-3543 January 1984 ORNL/FIN No. B0822

Survey of Operating Experience from LERs to Identify Aging Trends - Status Report.

1.0 Background

The objective of this study was to review the currently available sources of light-water reactor operating experience information contained in the Licensee Event Reports (LERs) to identify and evaluate age-related events and trends. The review focused on time-related degradation mechanisms that affect mechanical, structural, and electrical systems and/or components which could result in compromising a safety function. The study was conducted to evaluate the suitability of LERs as a source of data for evaluation of aging trends and is in response to recommendations made at the NRC-sponsored Workshop on Aging, conducted in August 1982.

The scope of this study was the review of LER(s) and, prior to 1976, LER predecessor abstracts for failures resulting from identified age-related degradation mechanisms. An LER is generated by a licensee upon a deviation from the plant Technical Specifications and generally only includes failures that affect safety-related components or systems.

Although abstracts of documents dating back to 1969 were reviewed, the majority of the data was obtained from the more detailed reports available from 1976 to 1982. The abstracts were obtained from the Nuclear Operations Analysis Center (NOAC) [formerly the Nuclear Safety Information Center (NSIC)] file of over 35,000 LERs and LER predecessors. The selected LER abstracts were reviewed, and data were collected on plant, system, component, subpart, failure mechanism, severity of failure, and method of failure detection.

The data were entered into a computer file, which allowed sorts of various parameters to identify predominant age-related failure mechanisms, affected systems, failed component/subpart, or other data sorts of interest.

The review process involved elimination of non-aging effects, consolidation of information from multiple abstracts concerning a single event, and discarding events involving other than commercial nuclear power plants. For each event judged to be an age-related failure, the reviewer prepared an input record for entry into a data file established in the Oak Ridge National Laboratory (ORNL) computer for the aging study project.

2.0 Summary of Data Evaluation

In this study, specific time-related degradation mechanisms are identified as possible causes of a reportable occurrence. Data collected on domestic commercial nuclear power plants covering 1969 to 1982 yielded over 5,800 events attributable to age-related failures. Of these events, 2,795 were attributable to instrument drift, which are addressed separately in the report. The remaining events (3,098) were reviewed, and data were collected for each event, identifying the specific system, component, and subpart; the information included age-related failure mechanism, severity of failure, and method of detection of the failure. About two-thirds of the failures were judged to be degraded, with one-third listed as catastrophic failures. No events were found to be incipient failures, because and LER is prepared only on degraded or catastrophic failure conditions that place plant operation outside the Technical Specifications. The study found that information desired for evaluation of aging effects (equipment, age, service life, and environment) was seldom available from LERs. This reflects the intent of the LER system as a regulatory instrument, rather than an engineering data collection system.

Distribution of Aging Study LERs by Year

Year	Total number of LERs[a]	Percentage examined in aging study (%)
1982	4784	12
1981	4632	14
1980	3835	12
1979	3543	13
1978	3567	12
1977	3414	12
1976	2740	13
1975	2518	9
1974	2007	9
1973	1327	8
1972	763	7
1971	447	9
1970	265	11
1969	238	12
1968	218	11
1967	198	12
1966	170	8

[a]Number of NOAC accessions for LERs or LER predecessors including updates and revised reports, as of March 1983.

3.0 Results/Findings

The Tables below show the distribution of failure mechanisms for the 3,098 events selected as age-related failures.

Failure Mechanisms for Age-Related Events

Failure Mechanism	Number of events
Wear	522
Corrosion	414
Contamination, internal	382
Contamination, external	331
Fatigue	324
Crack*	259
End of Life*	226
Contamination contact	205
Vibration	165
Stress Corrosion	110
Erosion	102
Other Miscellaneous mechanisms	58

*While not actual failure mechanisms, these classifications are discussed also.

The 3,098 events described in the proceeding paragraph were each associated with one of 68 system classifications. The most frequently reported systems are listed in the Table below with those ten systems representing 53.4% of the events.

Age-Related LERs by System

System designation[a]	System	Number of age-related events involving system	Percent
SF	Emergency core cooling system and controls	227	7.3
EE	Emergency generator system and controls	222	7.2
SD	Containment isolation system and controls	215	6.9
PC	Chemical/volume control and liquid poison systems and controls	212	6.8
WA	Station service water systems and controls	144	4.6
CB	Coolant recirculation systems and controls	138	4.5
SB	Containment heat removal systems and controls	134	4.3
HB	Main steam supply system and controls (other than BWR steam supply)	130	4.2
CF	Residual heat removal systems and controls	129	4.2
SA	Reactor containment systems	104	3.4
	SUBTOTAL	1655	53.4
	Balance of systems (56)	1443	46.6
	TOTAL	3098	100.0

[a]From *Instructions for Preparation of Data Entry Sheets for Licensee Event Report* (LER) File, NUREG-0161, July 1977.

Age-Related LERs by Component

Rank	Component	Number	Percent
1	Pipe	446	14.4
2	Valves, Other	277	8.9
3	Monitor	273	8.8
4	Valve, isolation	243	7.8
5	Pump	239	7.7
6	Diesel	151	4.9
7	Valve, Check	101	3.3
8	Steam generator	87	2.8
9	Heat exchanger	83	2.7
10	Snubbers	69	2.2

Age-Related LERs by Part

Rank	Part	Number	Percent
1	Weld	324	10.5
2	Miscellaneous subcomponent	266	8.6
3	Pipe and tubing	233	7.5
4	Valve seat	212	6.8
5	Contacts	192	6.2
6	Packing, seal	163	5.3
7	Wall (pipe)	137	4.4
8	Shaft	113	3.6
9	Housing	102	3.3
10	Bearing	69	2.2

Another study also utilized the LER data. The initial Accident Sequence Precursor (ASP) program examined about 19,400 Licensee Event Reports (LERs) from the period 1969-1979. The initial screening of these LERs resulted in selection of 529 events for detailed review. The detailed review identified 169 events considered to be precursors of potential severe core accident sequences. These events were either initiating events for the sequences or failures that could have affected the course of postulated off-normal events or accidents. These are the events that the ASP program subsequently quantified.

Eleven events, from the common LER data base were selected in both the ASP and Aging studies. These events are summarized in the table on the following page. The reason only 11 events are common to the results of the two studies is due to the strict selection criteria of the ASP effort. Unless the event caused a transient or affected a safety function response to a transient of interest, it was not included in the ASP events. The ASP program also focused on events where multiple failures occurred, particularly common-cause failures. Although aging processes affect all components to some degree, there generally is not a coordinating mechanism to cause failure to occur simultaneously in multiple components. An exception to this is standby components where demand for operation itself provides the coordination.

The LER system is not an engineering data collection device; rather, it is a regulatory event reporting scheme for the purpose of measuring licensee compliance to their respective plant Technical Specifications. Consequently, any analysis of events and failures from LERs must be tempered by the non-statistical nature of the data.

This study utilized the LERs to determine failed components, the age-related failure mechanisms responsible, the severity of the failure, and the

SUMMARY OF ASP/AGING STUDY COMMON EVENTS

Event No.	Accession No.	Event description	Failure mechanism
1	152563	Steam generator tube break	Wear from external debris
2	147400	Both main steam stop valves fail	Corrosion product buildup
3	120293	Plant service water strainers plugged	Contamination buildup
4	105540	Low flow feedwater line severed	Vibration
5	97107	Safety injection valve failed to open	Corrosion product buildup
6	89205	Reactor coolant pump shaft failed	Fatigue
7	60227	Steam isolation valves fail to close	Contamination of pilot valves
8	44751	Three of four safety system level sensors fail	RCS[a] crud buildup
9	128906	HPCI[b] fails due to governor actuator drift	Drift
10	124222	Six main steam relief valves fail to lift	Drift
11	93553	Diesel generator fails due to time set point drift	Drift

[a]Reactor coolant system.

[b]High-pressure coolant injection.

failure detection methods for reportable occurrences due to possible age-related failures. The data are by no means statistically accurate, but serve to indicate possible areas where further study should be performed to better characterize aging of components and systems.

These general conclusions can be drawn from the study.

1. Surveillance testing is an effective technique for detecting degradation.

2. Study data support recent ASME Code emphasis on pump and valve testing.

3. Confirmation of In-Plant Reliability Data System (IPRD) study emphasizes reliability of pumps, valves, and diesel generators.

4. More research is needed to characterize instrument drift causes and piping failures.

2.3 AGING/RISK CONSIDERATIONS NUREG/CR-4144 APRIL 1985 PNL/FIN NO.B2865

Importance Ranking Based on Aging Considerations of Components Included in Probabilistic Risk Assessments

1.0 Background

This study utilizes existing probabilistic risk assessments (PRAs) to gain insights on the relationships between aging of nuclear power plant components and public risk. A method is developed and applied to determine the potential risk significance of aging effects. The study objective is to identify components in nuclear power plants that adversely affect risk if aging processes decrease component reliability or degrade performance characteristics. This objective does not include identifying specific aging processes or describing aging effects (time dependence) on component failure rates.

PRAs are a method to mathematically estimate the likelihood and the consequences of potential accidents at nuclear power plants. In the process of performing a PRA, the potential accident initiators (LOCAs, transients, loss-of-offsite power, etc.) are identified and their likelihood quantified. The safety systems and their support systems that must function to safely shut down the reactors are then identified for each initiator. The safety systems and their support systems are modeled using event tree and fault tree methodology. The safety systems generally considered in a PRA are the reactor protection system, main and auxiliary feedwater systems, high pressure and low pressure injection systems, residual heat removal systems, containment sprays, containment coolers, and accumulators. Support systems include electric power, service water, and engineered safety feature actuation systems. Operator actions are also included in the models.

The event tree and fault tree model solutions determine the combinations of component failures that lead to a core melt for each of the initiators. The combination of an accident initiator and the system failures that result in core melt is referred to as an accident sequence. The combinations of individual component failures that cause "the required systems to fail" is referred to as a cutset.

The probability of each individual component being unavailable is referred to as its unavailability. The probability of the cutset is the product of the unavailability of the individual events. The frequency of an accident sequence can be approximated by the sum of all the cutsets that result in failures of the same set of safety systems. The overall plant risk is simarily approximated by the sum of the accident sequences, or equivalently, the sum of all the accident cutsets.

In addition, a probability of containment failure can be assigned to each accident sequence. In some PRAs, the consequences of accident sequences are evaluated in terms of man-rem, fatalities, or economic impact.

The scope of PRAs vary greatly. Some consider internal events only; others include seismic events, floods and fires, etc. The depth of the analyses of the systems and sequence consequences also varies considerably. The scope of the PRA, as well as the level of detail considered, limit the information that can be extracted from the analysis.

The aging program in general and the risk significance task in particular can benefit from the products of other NRC and industry programs including the Accident Sequence Evaluation Program (ASEP) and the data gathering programs

(LER's, NPRDs, and others). The ASEP program is designed to provide analysis of the dominant accident sequences for most LWR's in the United States. As a part of this program the cutsets for the dominant sequences are identified and risk importance measures calculated for a large number of components. When the results are made available it will be possible to apply the methods outlined in this report to a broad range of plants. This will provide a good basis for assigning priorities to component classes based on the risk estimates at a large number of plants rather than the three (Oconee, Calvert Cliffs and Grand Gulf) analyzed here. The approaches used in ASEP will allow identification of the most risk significant components and systems based on plant design and other operating characteristics. This information will assist in making specific recommendations as to what type of inspection and preventive maintenance programs will be most effective in controlling risk at different plants based on plant design.

The output from this study can be combined with other studies (data, analytical or experimental) that identify the components that are most susceptible to aging mechanisms. The combination of identification of risk significance and aging susceptibility will provide a good basis for effectively focusing resources.

2.0 Summary of Approach

The approach taken in this study uses the results of existing probabilistic risk analyses (PRAs) to gain insights about the relationship between risk and component aging. PRAs performed to date do not explicitly model risk as a function of time, but calculate an average risk level. This report defines a risk importance measure, that measures the sensitivity of risk to changes in a component failure rate. This measure is the partial derivative of the core melt frequency with respect to the failure rate of a specific component. Those components that have the highest sensitivity have the potential for causing the greatest change in risk if their failure rates increase due to aging.

The information presented in a standard PRA does not include time dependent effects. In determining the risk level at a plant, PRA's generally use a time averaged unavailability. Aging issues deal with the time dependent nature of risk. This limits the nature of the information that can be extracted from a PRA without extensively modifying the PRA. This report suggests a method for determining the potential risk significance of aging effects that is based on determining the sensitivity of risk to increases in failure rate. This adaptation of PRA results enables us to identify the components that have the most significant impact on risk if their failure rates increase due to aging effects without describing the time-dependent behavior of the failure rate. The information extracted from PRA's in this manner can be quite useful in guiding research efforts if used in context.

The term "component" can be considered to be individual pieces of hardware, e.g., a valve casing, a valve stem, wiring, etc. The "component" can also be considered as a functional unit such as a motor operated valve that consists of a number of component parts. Components as defined in most PRAs and in this report represent functional units. A motor operated valve for instance is interpreted as consisting of the valve, the motor operator, the circuit breaker, and the electrical cable and control circuitry specifically associated with the valve. A brief description of the component boundaries for each type of component is included in Table 1.

Frequently, components are subject to a number of different failure modes. For instance, motor operated valves could fail to function by several modes including: failure to open, failure to close, and gross leakage. Table 1 also includes the most important failure modes for each component type. These failure modes represent component functional failures and do not indicate the root cause of the failure or the failure mechanism. From an aging perspective, the time dependent processes that lead to a functional failure are of the most concern.

3.0 Results/Findings

The results of risk aging sensitivity measure calculations are presented for plants analyzed as part of the Reactor Safety Study Methodology Application Program (RSSMAP). These studies represent limited-scope PRAs in that they do not include external events and do not specifically include analysis of piping and wiring. The plants included in this analysis are two PWR's, (Oconee and Calvert Cliffs) and one BWR (Grand Gulf). Also included are bounding calculations for three other components: a reactor vessel, steam generator tubes, and snubbers.

Table 2 presents the aging sensitivity rankings for component groups at PWR's. These results are obtained by adding the results of the component groups at the two PWR's. Check valves of the auxiliary feedwater systems and breakers/contactors and trip relays/trip modules of the reactor protection system have the highest potential risk impact as measured by the aging sensitivity measure.

TABLE 1: COMPONENT BOUNDARIES

Component	Boundary	Failure Modes of Concern
Pumps (Electric)	Includes pump, motor, and the control circuitry and electric power components specifically associated with the pump.(1)	• failure to start on demand • failure to run • gross leakage
Pumps (Turbine Driven)	Includes pump, turbine, and control circuitry specifically associated with the pump.	• failure to start on demand • failure to run • gross leakage
Motor Operated Valves	Includes valve, motor operator and the control circuitry, and electric power components specifically associated wtih the valve.(1)	• failure to open on demand • failure to remain open
Control Valves (Air Operated)	Includes the valve, the air actuator, and the control circuitry specifically associated with the valve.	• Failure to go to the "fail safe" position on signal • failure to provide control capability
Check Valve	Includes the check valve only	• failure to open
Relief Valve	Includes the relief valve only	• stuck open
Circuit Breaker/ Contactor (RPS)	The circuit breakers that provide power to the control rod drive mechanisms.	• failure to open
Relay (RPS)	The relays that actuate the trip breakers on signal from trip module.	• failure to open
Trip Module/	Includes the sensors, cables, bistables, and relays that measure plant parameters such as reactor coolant pressure and send a trip signal to trip breakers.	• failure to send trip signal when plant parameters require

TABLE 1: (Concluded)

Component	Boundary	Failure Modes of Concern
Actuation Channel/ Subchannel	Includes the sensors, cables, bistables, and relays that measure plant parameters and send out an Engineered Safety Feature actuation signal.	• failure to send ESAS signal when required
Battery	Includes the battery and the battery charger.	• failure to provide DC power to components requiring DC power (given loss of AC power)
Diesel Generator	Includes the diesel and its support sytems (lube oil cooling, fuel supply, etc.).	• failure to provide AC power to components requiring AC power (given loss of off-site power)
Room Coolers	Includes the fan and cooling coils that provide room cooling to pump rooms.	• failure to cool pump room

(1) The electrical components specifically associated with the pump or motor operated valve would include the connector, cable, and circuit breaker that power the motor, but does not include the electric power distribution system that feeds the circuit breaker.

Table 2. Aging sensitivity of component groups in PWR's.

Rank	Type	System	Aging Sensitivity
1	Check Valves	Auxiliary Feedwater	5.5×10^{-3}
2	Circuit Breaker/Contractor	Reactor Protection	3.2×10^{-3}
3	Trip Relay/Trip Module	Reactor Protection	2.2×10^{-3}
4	Control Valves (air operated)	Auxiliary Feedwater	1.4×10^{-3}
5	Motor Operated Valves	Auxiliary Feedwater	1.4×10^{-3}
6	Pumps	Auxiliary Feedwater	1.4×10^{-3}
7	Motor Operated Valve	High Pressure ECC	4.7×10^{-4}
8	Motor Operated Valve	Service Water	2.9×10^{-4}
9	Pumps	Service Water	2.6×10^{-4}
10	Actuation Channels	Safeguards Actuation	2.1×10^{-4}
11	Check Valve	Low Pressure ECC	1.8×10^{-4}
12	Motor Operated Valve	Low Pressure ECC	1.7×10^{-4}
13	Turbo Generator/Diesel Generator	Emergency Power	1.6×10^{-4}
14	Check Valve	High Pressure ECC	1.0×10^{-4}
15	Batteries	Emergency Power	7.3×10^{-5}
16	Pumps	High Pressure ECC	5.3×10^{-5}
17	Room Coolers	Service Water	3.3×10^{-5}
18	Pumps	Low Pressure ECC	2.0×10^{-5}
19	Relief Valves	Reactor Coolant Pressure Boundary	1.5×10^{-5}
20	Check Valves	Service Water	1.3×10^{-5}

Table 3 presents the combined results for component types of the two PWR plants. Check valves, circuit breakers/contactors, trip modules/actuation channels, motor operated valves, pumps, and air operated control valves have the highest values of the aging sensitivity measure.

Table 4 shows the combined results for component groups at the Grand Gulf BWR. Motor operated valves of the low pressure emergency core cooling system and service water system and actuators of the engineered safety actuation system have the highest potential risk impacts as measured by the aging sensitivity measure.

Estimates are listed in Table 5 for the aging sensitivity measure for three additional component types: the reactor vessel, steam generator tubes, and snubbers using existing PRA's and related studies. The calculations are bounding calculations intended to compare the importance of these components to other components at the plant. Table 5 presents the results of these calculations.

Reactor Vessel

The reactor vessel has the highest potential impact on risk of any component in the plant. PRA's generally make the conservative assumption that a failed reactor vessel results in an uncoolable configuration that leads to core meltdown. The aging impact as measured by the aging sensitivity measure is high compared to the other components in the plant.

Steam Generator Tube

A rupture in a steam generator, as an initiating event, results in a small LOCA and consequently loss of heat removal capability of one steam

Table 3. Aging sensitivity of component types in PWR's.

Rank	Type	Aging Sensitivity
1	Check Valves	5.8×10^{-3}
2	Circuit Breaker/Contactor	3.2×10^{-3}
3	Trip Module, Relay/Actuation Channel	2.4×10^{-3}
4	Motor Operated Valves	2.3×10^{-3}
5	Pumps	1.7×10^{-3}
6	Control Valves (air operated)	1.4×10^{-3}
7	Turbo Generator/Diesel Generator	1.6×10^{-4}
8	Batteries	7.3×10^{-5}
9	Room Coolers	3.3×10^{-5}
10	Relief Valves	1.5×10^{-5}

TABLE 10. Aging Sensitivity of Component Groups at Grand Gulf

Rank	Type	System	Aging Sensitivity
1	Motor Operated Valves	Low Pressure ECC	2.3×10^{-4}
2	Motor Operated Valves	Service Water	1.3×10^{-4}
3	Actuators	Safeguards Actuation	9.9×10^{-5}
4	Pump	Service Water	5.9×10^{-5}
5	Check Valves	Service Water	5.9×10^{-5}
6	Motor Operated Valves	High Pressure ECC	5.4×10^{-5}
7	Check Valves	High Pressure ECC	5.4×10^{-5}
8	Check Valves	Low Pressure ECC	2.8×10^{-5}
9	Batteries	Emergency Power	2.4×10^{-5}
10	Pump	Low Pressure ECC	2.4×10^{-5}
11	Pump/Tubine Pump	High Pressure ECC	1.3×10^{-5}
12	Diesel Generator	Emergency Power	9.5×10^{-6}
13	Relief Valves	Reactor Coolant Pressure Control	2.6×10^{-6}

generator. In this situation, core cooling requirements generally are the operation of the auxiliary feedwater system and at least one high pressure injection pump. Table 6 gives an estimate of the tube aging impact based on the cooling requirement for four plants. Consistent with the aging sensitivity measure definition, these estimates are based on simply adding the conditional failure probabilities of the auxiliary feedwater system and the high pressure injection system. The average value from these four plants is included in Table 5. The potential risk impact of steam generator tubes as measured by the aging sensitivity measure is higher than that of the standby components.

Snubbers

In order to determine the aging impact of snubbers the results of the Seismic Safety Margins Research Program were reviewed. The case of snubber failure is specific in that it has been done for the Zion plant.

The risk associated with snubber failures is characterized by an increased likelihood of a LOCA induced by an earthquake. The earthquake also degrades the safety system that cools the core in the event of a LOCA. In this situation, it is assumed that snubber failure will result in a large or medium LOCA for any earthquake with a magnitude larger than design basis. The dominant core melt sequences for an earthquake induced LOCA, contain the failure of the Safety Injection System (SIS) to cool the core.

The aging sensitivity measure for snubbers as calculated here is moderately high. This calculation is an approximation and subject to high uncertainty. Further, the information used is for only one plant that is not located in a high seismic activity zone. The potential risk significance of snubbers will be very site-dependent in general.

TABLE 5. Aging Sensitivity Measures for Selected Components

Component	Aging Sensitivity
Reactor Vessel	1
Steam Generator Tube	3×10^{-3}
Snubber	1.8×10^{-5}

TABLE 6. Aging Sensitivity Measure Calculations for Steam Generator Tubes

Plant Name	Cooling Requirements	Aging Sensitivity
ANO	1/2 EFWS 1/3 HPIS	6.5×10^{-4} + 4.0×10^{-4} = 1.1×10^{-3}
Oconee	1/2 AFWS 1/3 HPIS	2.4×10^{-4} + 1.4×10^{-3} = 1.6×10^{-3}
Calvert Cliffs	1/2 AFWS 1/3 HPIS	3.0×10^{-3} + 1.7×10^{-3} = 4.7×10^{-3}
Sequoyah	1/3 AFWS 1/3 HPIS	4.3×10^{-5} + 3.5×10^{-3} = 3.5×10^{-3}

Limitations and Assumptions

The most important limitations of this study are the limited number of plants analyzed and limiting the scope of components studied to those analyzed in the PRA's. The analysis is limited to the effects of complete failure (loss of function); the effects of degradation are not specifically addressed. Also common-cause failures attributed to aging are not specifically addressed.

This report considers only some of the components that are potentially important to risk. Components whose primary purpose is to mitigate the consequences of severe accidents such as containment spray nozzles, piping and pumps were not considered. The importance to risk of components that mitigate accident consequences is not easy to determine in light of the large uncertainties associated with the phenomenology and fission product behavior of severe accidents. Structural components such as the containment and containment lining were also not considered. Piping and wiring are not explicitly considered in these analyses and components such as the reactor vessel, steam generator tubes and snubbers are treated only superficially for example purposes.

4.0 Utilization of Research Results in the Regulatory Process

The results of this study can be used to provide a relative sense of the unavailability of selected components. Future efforts on the time dependent portion of aging for these component types are needed to more completely describe the risk impact due to component aging.

3.1 A METHOD FOR AGING ANALYSIS NUREG/CR- Jan. 1986 SEA CONSULTANTS, INC.

A methodology for Evaluation of Complex Aging Effects on Systems' and Components' Performance and Interactions.

1.0 Background

A primary objective of the Nuclear Plant Aging Research (NPAR) Program is to analyze the nature and extent of aging on safe reactor operation. Simulation of the aging process is a method by which understanding can be obtained. Correct simulation is predicated upon accurately modeling the simultaneously occurring inherent and operational aging effects (stressors) in combination with typically complex physical interactions that result from the component and system design configuration. Similar to finite element methodologies, aging simulation must begin with reduction of the problem to the lowest state or element where aging effects can be reliably demonstrated. The lowest element for components important to safety, for our purpose, is the lowest unit with aging significance (LUWAS). A LUWAS may be defined as the lowest constituent part that is susceptable to aging effects that must perform within specification for safe nuclear operation. As with finite element analysis, the initial step is development of the mathematical representation or transfer function of the input-output ration of the LUWAS. The transfer function represents the LUWAS behavior at the beginning of life or period prior to installation and startup. The transfer function is empirical (ratio of input to output) and no knowledge of the LUWAS internal structure is required. At the component level the combination of all LUWAS transfer functions provide a relationship describing the dynamics of the component.

Aging is a time domain phenomenon that alters the state of the LUWAS and can be represented as a set of state variables (temperature, radiation, voltage, humidity, position, etc.). Utilizing time-domain methods, the transfer function is written as a differential equation in terms of the state variables. Empirical relationships, government and industry supported test programs and plant experience are the bases to establish the differential changes in LUWAS response and the resulting component behavior. State variables representing the time domain aging effects provide basis for prediction of the future state and output of the component.

The NPAR strategy recognizes that the evaluation of the affects of aging on safety reactor operation must also include system interactions. Evaluation of spatially and/or functionally coupled system interactions require a methodology that is structured and interactive such that system interdependencies are reliably displayed. Aging effects, analogous to system interactions, are the result of internal and external interactions within a component. Evaluation of aging effects also require a structured and interactive approach. The N-Square diagram is an interactive modeling technique that is both structured and interactive.

An evaluation of complex aging effects on components and systems performance and systems interaction is developed under the auspices of the NRC's Office of Nuclear Regulatory Research, Nuclear Plant Aging Research (NPAR) Program. To assist the Nuclear Plant Aging Research purpose, a comprehensive approach has been developed that consists of the combined application of an interactive modeling technique, the N-Square diagram, and systems engineering to develop a methodology that facilitates understanding and evaluation of complex systems interactions, such as those that result from the effects of aging.

2.0 Summary

This report presents a comprehensive approach for evaluating complex aging effects on components and systems performance and systems interaction. This approach was developed under the auspices of the NRC Nuclear Plant Aging Research (NPAR) Program and consists of the combined application of an interactive modeling technique (the N-Square diagram) and systems engineering. The resulting methodology facilitates analysis of complex systems interaction, with specific illustrative applications to components (differential pressure transducer, compressor and an invertor) and to systems (High Head Safety Injection system and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal system). Emphasis is placed on evaluating the effect of component aging on functional and spatial systems interactions.

The systems interaction methodology consists of development of several levels of N-Square diagrams to obtain a dynamic plant model at the plant function level. Consecutive levels of N-Square models are developed to reduce the system to the component level and then to the Lowest Unit With Aging Significance (LUWAS). From here the effects of postulated and empirically generated age stressors can be assessed, as potentially critical systems and component interactions are identified.

The results of a comprehensive model of the components and systems interactions required to complete a safety or safety-related plant function could be used to identify inspection, surveillance, and monitoring methods or corrective actions needed to assure plant performance.

3.0 Results

3.1 The NSQ © Program

The approach taken in Phase I was to design and develop an N-Square analytical tool using the Turbo Pascal language, version 3.01A. Turbo Pascal is a standard pascal implementation that supports programming access to the PC-DOS/MS-DOS operating system, (the hardware interface to the IBM PC) and the math co-processing chip. Also, the Turbo Pascal compiler leads to code which executes extremely fast (about 3 times faster than the Lotus 1-2-3 example investigated).

The program has been called NSQ, and consists of a single executable file (NSQ.COM), less than 36K bytes in size, and two screen image files. When invoked, NSQ presents a screen permitting up to 4 LUWAS components to be identified, and a 4 x 4 = 16 cell array of transfer functions to be entered. All models were developed and tested within this prototype limitation.

The user can also open up and work within 4 "windows" that overlay the N-Square matrix. These windows are invoked through use of 4 of the 10 function keys. Each window pertains to a group of parameters that relate to the N-Square calculation.

Once the model has been entered, it can be saved on a disk, again by means of a function key. Similarly the model can be retrieved from disk for manipulation, much like a word processor. Use of a function key begins the execution of the model. Once started the program switches into color graphics mode (199 x 359 pixels) and displays the magnitude of up to 3 color-coded LUWAS as a function of time. An N=3 model (simple R-L-C circuit diagram) with 100 time steps runs in about 30 seconds, and an N=4 model (the differential pressure transmitter) with 50 time steps executes in 110 seconds

(1 minute 50 seconds). These benchmarks are for an 8088 CPU without 8087 co-processing support.

A number of improvements to the program are under consideration to extend the software beyond its prototypical testing stage.

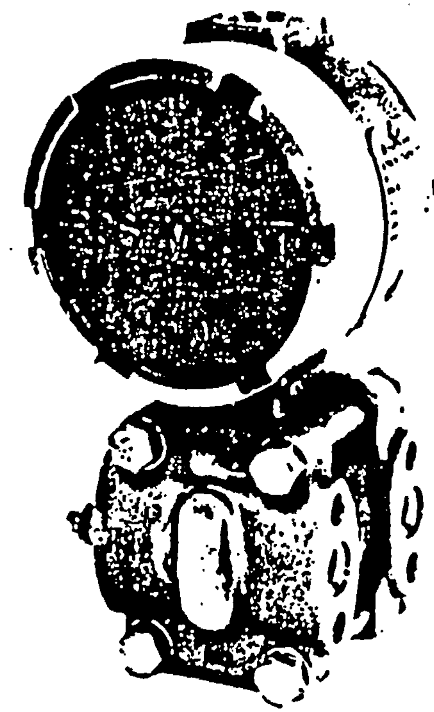
3.2 Illustrative Component Analysis

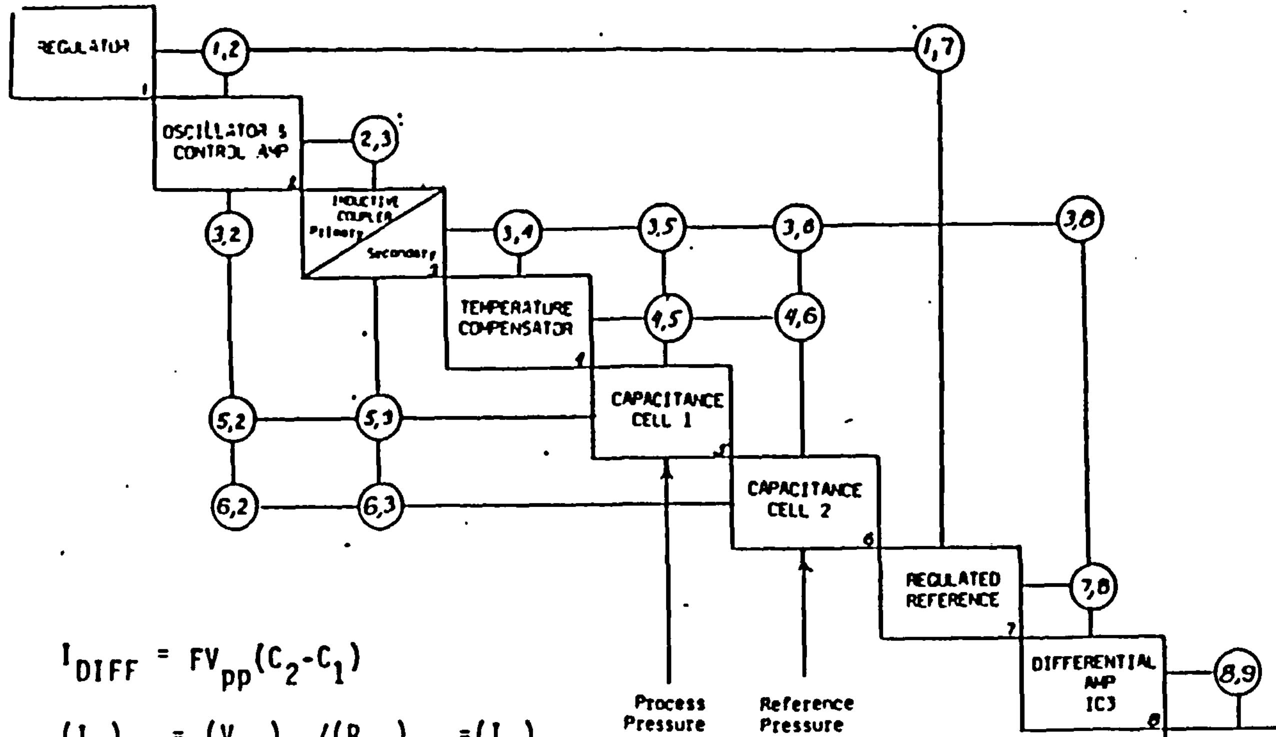
The results of the Phase I program are presented through application of the methodology, presented on three (3) safety related components common to a majority of commercial reactor safety systems: 1) a differential pressure transmitter, a primary component in LOCA and ESFAS instrumentation and control systems; 2) an air compressor used in instrument air systems; and 3) a 125 Vdc to 120 Vac invertor, typically applied as a 120 Vac emergency power source. These examples are provided for demonstration only. The vendor documentation utilized in the preparation of these examples is uncontrolled and no representation of product acceptability or plant impact should be pursued. The N-Square diagram and the summary table of Independent Stress Effects are illustrative of the evaluation of the differential pressure transmitter.

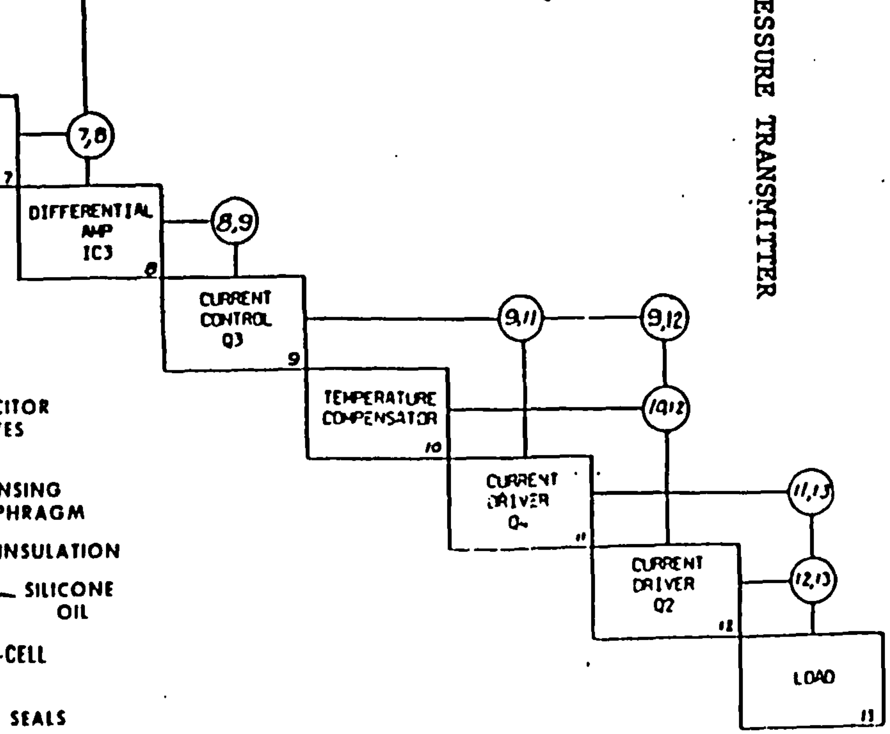
3.3 System N-Square Diagrams

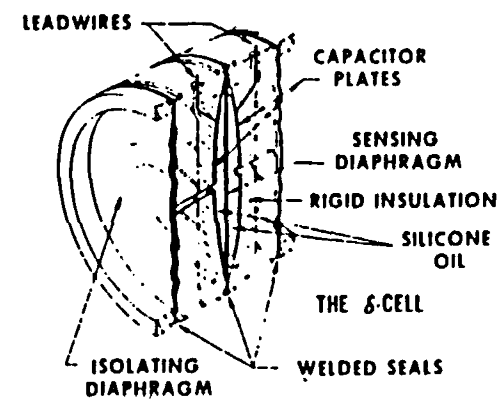
Figure 1 and Table 1 are N-Square diagrams representing the functional and spatial system interactions for reactor core water level maintenance for a LaSalle vintage BWR (LPCI) and a Watts Bar vintage PWR (HPSI). It is easily

DIFFERENTIAL PRESSURE TRANSMITTER









THE δ-CELL

(3,8) $I_{DIFF} = FV_{pp}(C_2 - C_1)$

(9,11) $(I_B)_{Q4} = (V_{CE})_{Q3}/(R_{CE})_{Q3} = (I_C)_{Q3}$

(9,12) $(I_C)_{Q2} = -(V_{CE})_{Q3}/(R_{CE})_{Q3} + I_{R18}$

(8,9) $(I_B)_{Q3} = (I_{DIFF}Z \times A_G)/4.7K$

INDEPENDENT STRESS EFFECTS

DEVICE: Pressure Transmitter

I.Layne-6/85

LUWAS	STRESS	EFFECTS OF LUWAS	EFFECTS ON PERFORMANCE	RELATIONSHIP
1 - ISOLATING DIAPHRAGM	Neutron Radiation	Embrittles stainless diaphragm.	Decreases diaphragm deflection per unit pressure which reduces signal for given pressure.	$I_{sig} = K \times \text{Deflection} = \frac{K'}{f(\emptyset n)}$
2 - PROCESS FLANGE	None			
3 - SILICON OIL	Gamma Radiation	1. Gassing	1a. Increases compressibility which reduces coupling between sensing and isolating diaphragms.	1a. Response time = K x f(sonic velosity)
			1b. Dielectric constant decreases, causing capacitance to decrease which increases signal current.	1b. $I_{sig} = K \times I_{ref} \frac{C_1 - C_2}{C_1 + C_2}$
			1c. Decrease in capacitance causes oscillator frequency to increase and V_{pp} to decrease.	1c. $f\ V_{pp} = \frac{I_{ref}}{C_1 + C_2}$
		2. Viscosity increases.	2. None (sonic velosity is unchanged).	2. None
	Temperature	3. Viscosity decreases.	3. None (sonic velosity is unchanged).	3. None
4 - SENSING DIAPHRAGM	Neutron Radiation	Embrittlement	Decreases deflection per unit pressure.	$I_{sig} = K \times \text{Deflection} = \frac{K'}{f(\emptyset n)}$
5 - CAPACITOR PLATES	None			
6 - RIGID INSULATION	Gamma Radiation	Dielectric constant changes.	None - Capacitance between capacitor plates and housing changes, but circuit is not grounded to housing.	None
7 - δ-CELL HOUSING	None			
8 - ELECTRONIC HOUSING & COVER	None			
9 - LEAD WIRES	Gamma Radiation	Properties of insulation changes.	Capacitance between leads changes.	C = K x f(gamma radiation)

INDEPENDENT STRESS EFFECTS (Continued)

I.Layne-6/85

LUWAS	STRESS	EFFECTS OF LUWAS	EFFECTS ON PERFORMANCE	RELATIONSHIP
10 - O RINGS	Gamma Radiation	Embrittlement and change in dimensions.	Moisture could enter instrument and cause degraded function or failure.	Literature search or tests are required to determine whether O rings in compression will lose their sealing ability at the radiation and temperature levels encountered. Enter as a table.
	Temperature	High temp will have a similar but lesser effect as gamma radiation.		
11 - Oscillator Control				
o IC-1	Gamma Radiation @ 10^6 Rad	Changes properties of semiconductor.	Changes freq/voltage <1% which changes I_{ref}, but I_{diff} does not change since X_{C1} and X_{C2} change identically.	Run "SPICE" and enter results as a table.
	Neutron Radiation	Changes electronic characteristics.	Changes freq/voltage or causes failure.	Run "SPICE" and enter results as a table.
o Resistors	Insignificant			
12 - Oscillator				
o Switching Transistor Q1	Gamma or Neutron Radiation	Increases freq of switching.	Changes I_{ref} but not I_{diff} which changes I_{sig}. Feedback will mitigate.	Run "SPICE" and enter as a table.
o Resistors	Insignificant			
o Transformer	Insignificant			
o Capacitors	Gamma or Neutron Radiation	Changes capacitance within tolerance.	Changes freq which changes I_{ref} which changes I_{sig}. Feedback will mitigate.	Run "SPICE" and enter as a table.
13 - Voltage Regulator				
o Zener	Radiation	Changes regulated voltage.	Changes biasing and setpoints which could cause malfunction or failure.	Run "SPICE" and enter as a table.
o IC2	Gamma Radiation (10^6 Rad)	Changes semiconductor properties.	Changes voltage approx. 1% which could cause malfunction.	Run "SPICE" and enter as a table.

INDEPENDENT STRESS EFFECTS (Concluded)

I.Layne-6/85

LUWAS	STRESS	EFFECTS OF LUWAS	EFFECTS ON PERFORMANCE	RELATIONSHIP
Voltage Regulator (Cont'd)				
o Resistors	None			
o Capacitors	Insignificant			
14 - Zero Span Adj				
o Resistors	None			
o Pot	None			
15 - Current Control				
o IC3,Q2,Q3	Gamma Radiation @ 10^6 Rad	Changes semiconductor properties.	Improper signal current generated which could cause loss of calibration or failure.	Run "SPICE" and enter as a table.
o Resistors	None			
o Capacitors	Insignificant			
16 - Current Limiter				
o Q4	Gamma Radiation	Changes semiconductor properties.	Allows overcurrent condition or improperly limits signal current.	Run "SPICE" and enter as a table.
o Resistors	None			
o Capacitors	Insignificant			
17 - Linear Adj				
o Diodes	Gamma Radiation	Destroy junction.	Failure of linearity circuit.	Run "SPICE" and enter as a table.
o Resistors	None			
o Capacitors	Gamma Radiation	Change capacitance.	The first order non-linearities of capacitance as a function of pressure would not be correctly compensated.	Run "SPICE" and enter as a table.

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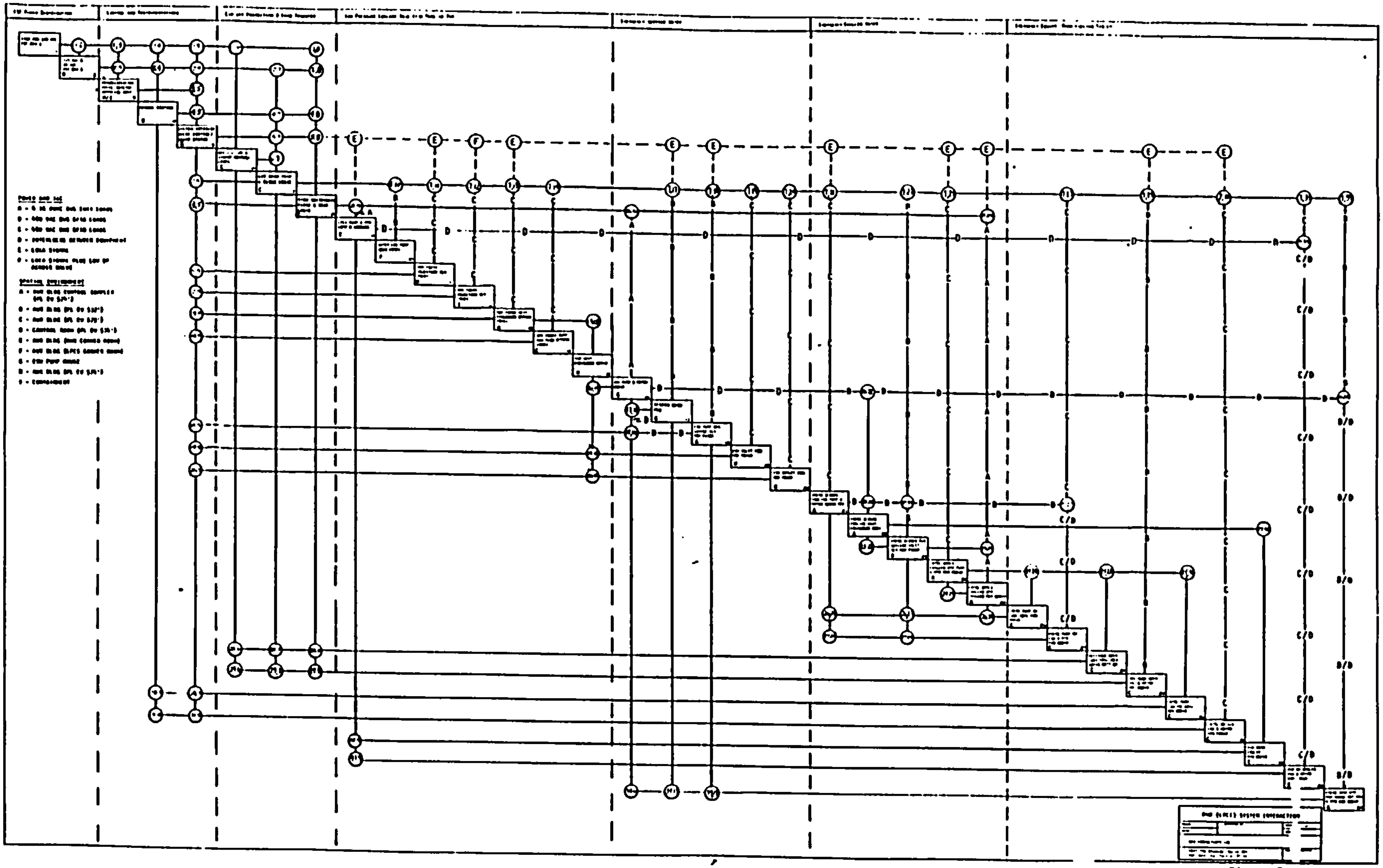


Figure 7

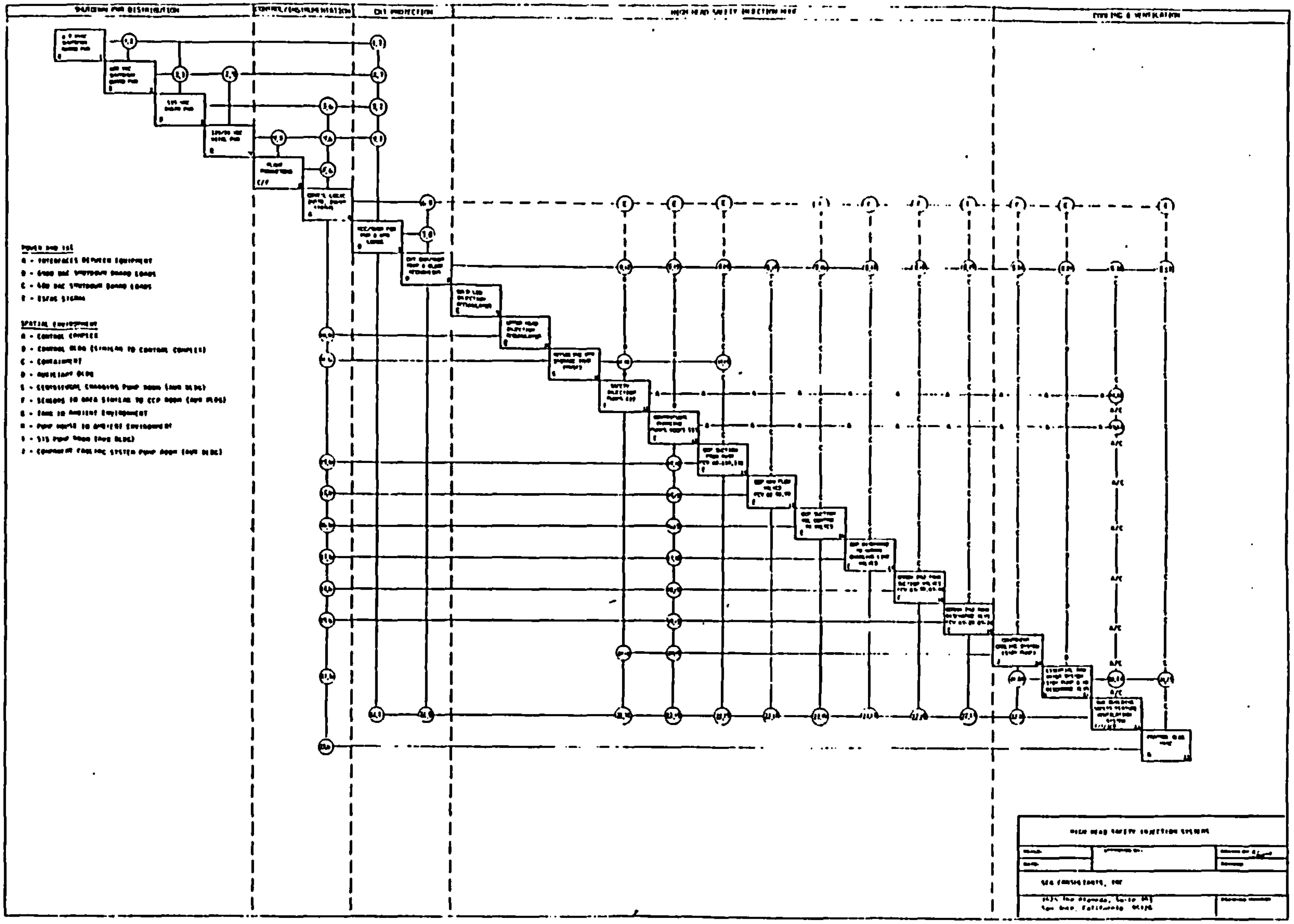


Figure 8

determined by visual observation that system components such as circuit breakers, LOCA initiation instruments and system interlock components have multiple interactions in the performance of reactor core water level maintenance. The detailed analysis is discussed in the report.

4.0 Utilization of Research Results in the Regulatory Process

4.1 Application for Inspection, Surveillance and Monitoring Methods (ISMM)

The preoperational and startup phase of a commercial nuclear power plant provide an initial insight into unplanned plant unique system and component interactions. During this phase of plant construction, components and systems are tested for conformance to test specifications which are developed to demonstrate proper performance. Nonconformances discovered during acceptance testing typically result from adverse interaction(s) that are readily detected.

The system interaction model for effect of aging can identify potentially critical interactions within components or systems that directly affect plant function. The results of a comprehensive model of the system and component interactions required to complete a plant function, (e.g., water level, depressurization, scram, etc.), could identify ISMM actions or corrective actions needed to assure plant performance.

4.2 Applications for PRA Enhancements

The nuclear industry and Nuclear Regulatory Commission rely on probabilistic risk assessment (PRA) for assurance of acceptable risk to public health and safety during plant operation. PRA seeks to quantify the public risk due to plant operation. Risk quantification is typically achieved through use of component failure rates obtained from plant specific data, Mil. Std 217 and IEEE 500. The nature and extent of the aging process on component failure rates has been acknowledged as an important element in risk quantification. Statistical testing to determine the nature and extent of the aging process on the component failure rate and performance is necessary for acceptable PRA quantification and removal of uncertainties associated with environmental qualification. Recognizing the complexity and cost of regulated statistical testing, the N-Square interactive model at the LUWAS level is a cost effective methodology to simulate and quantify the effect of aging on component behavior.

3.2 MOV Analysis & Test System (MOVATS) NUREG/CR-4380 Jan. 1986 ORNL/FIN No.B0828

Evaluation of the Motor-Operated Valve Analysis and Test System (MOVATS) to Detect Degradation, Incorrect Adjustments, and Other Abnormalities in Motor-Operated Valves.

1.0 BACKGROUND

Current surveillance requirements for determining the operability of motor-operated valves (MOVs) installed in nuclear plant safety systems are described in plant Technical Specifications. These consist of periodic functional tests in accordance with the in-service inspection provision of Section XI Division 1 of the ASME Boiler and Pressure Vessel Code. Numerous events have been documented in which MOVs have failed to operate as required either during the above surveillance tests or as a result of actuation of safety systems during plant operation. This has raised concerns about the general state of readiness of MOVs to perform their safety function under all anticipated accident conditions.

Oak Ridge National Laboratory (ORNL), as a part of its support of the NRC/RES Nuclear Plant Aging Research (NPAR) program, has been investigating MOV operating experience in nuclear plants. The purpose is to identify the factors which contribute to MOV failure, and to evaluate and recommend suitable monitoring methods. These methods may be used to detect and diagnose degradations and other abnormalities in MOVs prior to loss of operability. One such monitoring method that has been identified is the Motor-Operated Valve Analysis and Test System (MOVATS) developed and commercialized by MOVATS Inc. of Marietta, Georgia.

2.0 Summary

This report describes the results of a limited field test program carried out as part of the ORNL/Nuclear Plant Aging Research program to learn what MOVATS can provide about safety related MOV operational readiness above and beyond the currently used ARME Section XI methods. The limited test program was carried out at the request of NRC/NRR to support their ongoing review of plant in-service testing process and also to assist in resolution of generic issue II.E.6.1, "In-Situ Testing of Valves."

The test program consisted of obtaining signatures, using the MOVATS method, from 36 MOVs located in four commercial nuclear power plants. The signatures are analyzed to determine the types of degradations present and evaluated using the MOVATS method. Based on the results obtained, they determine what it states about MOV operational readiness. The valve signatures and their analyses were provided by MOVATS Inc. under subcontract to ORNL.

The MOVATS method utilizes portable data sensing, acquisition and storage equipment which is connected to an MOV during a plant shutdown. The sensors measure motor current, switch actuations and the axial displacement of the worm in the motor operator, all during actuation of the valve. The data is analyzed in-situ and also later at the MOVATS office using additional data reduction equipment. Detection and diagnosis of abnormalities is made based on comparison with signatures obtained from normal valves and from the experience of the MOVATS technicians.

Valve abnormalities can be classified into two types, each of which can affect operational readiness. The first type consists of time dependent degradation of MOV parts resulting from the effects over time of environmental stressors and service wear. The second type consists of incorrect adjustments and other similar abnormalities which either can lead to acceleration of the time dependent degradation or to failure of the valve to operate under some anticipated operating condition.

3.0 Results/Findings

ABNORMALITIES DETECTABLE BY MOVATS CLASSIFIED BY TYPE

Time-dependent degradations	Incorrect adjustments & other abnormalities
Bent stem	Excessive inertia
Gear wear	Inadequate stem lubrication
Motor pinion binding	Improper seating
Stem wear	Valve backseating
Grease hardening	Incorrect torque-switch calibration*
Motor degradation	Unbalanced torque switch*
	Excessive spring-pack gap*
	Excessive packing tightness*
	Improperly set bypass switch*
	Loose stem-nut locknut*

*Abnormalities that can cause valve failure under some anticipated operating conditions.

SUMMARY OF SIGNIFICANT MOV ABNORMALITIES IDENTIFIED BY MOVATS

Abnormality	Percent*
Improperly set bypass switch	75
Incorrect torque-switch calibration	50
Unbalanced torque switch	33
Excessive spring-pack gap	17
Excessive packing tightness	8
Excessive inertia	8
Loose stem-nut locknut	8
Valve backseating	8
Steam wear	8
Grease hardening	8
Gear wear	6
Motor degradation	3
Miscellaneous abnormalities	35

*The percentages shown in this table are based on a limited sampling of valves available for use in this test program. As a result, the values should be considered as indicators of frequency of occurrence of the listed abnormalities rather than absolute measures applicable generally to all MOVs.

Based on the field tests carried out as part of this program and on an analysis of the capabilities and limitations of the MOVATS methods, the following conclusions are drawn:

1. The MOVATS method can provide valuable diagnostic information regarding the operational readiness of MOVs well beyond that obtainable from ASME Section XI type surveillance tests.

2. The field tests carried out as part of this task demonstrated that incorrectly adjusted bypass switches and torque switches are common. Both of these abnormalities, but particularly the incorrectly adjusted bypass switch, can directly affect MOV operational readiness under some anticipated operating conditions.

3. The MOVATS method can provide useful diagnostic information regarding time dependent degradation (aging) of MOV parts. However, in order to utilize that information in determining operational readiness, the method must be applied periodically and trending data obtained from which to extrapolate operability into the future.

4. The MOVATS method has limitations with regard both to its ability to detect all abnormalities and its convenience. Neither limitation affects its utility in providing diagnostic information regarding MOV operational readiness well beyond that obtained using the ASME Section XI tests.

4.0 Utilization of Research Results in the Regulatory Process

This report, NUREG/CR-4380, in conjunction with NUREG/CR-4234 are contributors towards the resolution of generic issue II.E.6.1. "In-Situ Testing of Valves."

4.1 ELECTRIC MOTORS NUREG/CR-4156 JUNE 1985 BNL/FIN NO. A3270

Operating Experience and Aging-Seismic Assessment of Electric Motors

1.0 Background

Nuclear power generating stations utilize electric motors as vital system components both for the performance of normal operations as well as for the accomplishment of safety-related functions. It is therefore, imperative that motors remain capable of driving required loads.

Motor design and materials of construction are reviewed to identify age-sensitive components and to characterize the dielectric, rotational, and mechanical hazards on motor performance and operational readiness. Operational and accidental stressors are determined, and their effect on promoting aging degradation is assessed. The functional indicators which can be monitored to assess motor component deterioration due to aging or other accidental stressors are identified. Failure modes, mechanisms, and causes have been reviewed from operating experiences and existing data banks.
The study has also included consideration for the seismic correlation of age-degraded motor components.

Only a few categories of electric motors are of direct safety significance in nuclear power plants: (1) three phase induction motors; (2) direct current (dc) motors; and (3) three-phase synchronous motors. The squirrel cage induction motor is the "work-horse" of the nuclear industry, comprising nearly 90% of the total population. Synchronous and dc motors constitute an additional 9% with the balance comprised of specialty applications. The percentage of motor failures of the total population in each category ranges from 2.4% for synchronous motors to 6.3% for dc motors. The dc motor failure rate is higher than normal and attributable to commutator related problems. One of the data bases indicated that motors with ratings of 1-99.9 horsepower (hp) represent nearly 47% of the total motor population. Fractional hp (< 1.0 hp) motors and motors with ratings of 100-999 hp represent another 41% of the total population. Large motors (> 1000 hp) essentially make up the balance. It can be deduced from the data analysis that failure rate typically increases with horsepower rating, even though large motors are often equipped with sophisticated surveillance, monitoring and protection systems.

Three-phase induction motors are versatile and reliable, and speed can be selected to suit the load. Dc motors are reliable and have accurate speed control as well as efficient performance over the entire speed range, but commonly require more maintenance. Where constant speed is an absolute necessity, the synchronous motor is available.

With regard to motor applications, valves and pumps constitute nearly 95% of the total motor population and are predominantly driven by squirrel cage induction motors.

The insulating system of a typical electric motor consists of various materials in association with conductors and supporting structural parts. Insulating systems are NEMA designated as A, B, F and H, in ascending order of maximum operating temperature for a given life. Class B insulation systems are consistently in the highest failure category while Class F and H exhibit significantly lower failures. This is partly because a large population of motors in a typical nuclear plant have Class B insulation.

2.0 Summary of Applications

REACTOR FLUID SYSTEM MOTOR FAILURES (LER Data: 1974-1983)

*Residual Heat Removal (RHR)	41
Service Water (SW)	38
*High Pressure Coolant Injection (HPCI)1	35
Gas Radiation Waste Management	20
Containment Heat Removal (CHR)	19
Emergency Generating System (EGS)	18
Chemical Volume and Control System (CVCS)2	16
Heating Vent. Air Conditioning (HVAC)	12
Containment Isolation System (CIS)	11
Containment Gas Control (CGC)	11
*Reactor Core Isolation Coolant (RCIC)1	10
*Core Spray (CS)1	9
Reactor Coolant Recirculation1	9
Fire Protection	8
Radioactive Monitoring	8
Control Room Habitability	8
Others	<6

* Emergency Core Cooling System (ECCS)

1 BWR Application only

2 PWR Application only

(unmarked) Both BWR and PWR application

3.0 Results/Findings

During motorized system operation, key parameters can be utilized to generally assess motor integrity. Various performance or functional indicators can serve to characterize the behavior of any electric motor. When normal values for these parameters are observed to adversely change, the incipient stage of degradation, potentially leading to ultimate failure, is occurring. Therefore, characteristic parameter performance can be linked to failure modes, mechanisms, and causes that are representative for all types of motors.

A comprehensive survey table is available in NUREG/CR-4156. The table length (pages 4-23 through 4-30 precludes its presentation in the summary. However, it is recommended as a reference for more detail on failure modes, failure causes, failure mechanisms, aging, aging-seismic correlation, categorization of failure and relative probability of occurrence for electric motor components.

The following three figures present a summary of failure modes, mechanisms and performance indicators.

STATOR/ROTOR WINDINGS (INSULATING SYSTEM)

Performance Hazards	Failure Effects
Environment • Temperature • Humidity • Radiation • Chemical • Contaminants	• Degradation of Mechanical* and Electrical Strength with Time. • Shorted windings*
Vibration • Loose Laminations, Wedges* Blockings, Balance Weights*	• Turn-To-Turn Electrical Short • Winding-To-Ground Short
Excess Current • Frequent Starts • Overload	• Overheating, Burnout Windings • Excess Losses
High Voltage Gradient	• Corona or Ionization of Insulation • Electrical Short and Breakdown
Improper Ventilation • Design • Foreign Particles	• Overheating or Windings • Turn-To-Turn Short
Water/Steam	• Shorted Electrical Circuits • Corrosion of Laminations • Poor Electrical Connections*

Functional Indicators:

- Temperature (windings)
- Vibration signature
- Current signature
- Voltage gradient, corona
- Insulation Resistance
- Power factor/loss factor
- polarization index

Dominant Failure Mechanism (Motor Subsystem) ⟷ Performance Hazards ⟷ Failure Effects ⟷ Functional Indicators

* Aging Susceptibility per Table 4-15

Functional Indicators for Dielectric Integrity

BEARINGS
&
LUBRICATION

(ROTOR ASSEMBLY)

Performance Hazards	Failure Effects
Excessive or Under Lubrication • Damage Oil Sling Ring • Seal or Gasket Leakage	• Overheating • Cracking, Scoring, Brinelling, or Splitting of bearings
Water/Steam	• Corrosion
Vibration • Misalignment	• Noise* • Cracking
Bearing Insulation Damage	• Bearing Current Leading to Overheating
Improper Cooling • Seal or Gasket Leakage	• Overheating
Mechanical Problems • Oil/Foreign Particle Deposits • Rotor-Stator Interference • Excessive Axial Movement • Too Much Clearance Between Bearing and Shaft • Frozen Bushing	• Jamming/Freezing of Balls* • Stator Lamination Damage • Tripping of Accessories • Shifting of Rotor C.G.

Functional Indicators:

- Lubrication
  - Lubricity
  - Viscosity
- Vibration signature
- Temperature (Bearing, oil)
- Eddy current losses
- Alignment
- Clearances
- Speed (torque)

Dominant Failure Mechanism (Motor Subsystem) | Performance Hazards | Failure Effects | Functional Indicators

* Aging Susceptibility per Table 4-15

Functional Indicators for Rotational Integrity

MISCELLANEOUS FAILURES

(ALL COMPONENTS)

Performance Hazards	Failure Effects
Vibration • Bolts • Welding/Brazing • Dowel pins	• Loose or Damaged Leads* • Weld Failure • Loss of Balance Weight • Bending or Distortion of Shaft • Shear of dowel pins
Environmental Contamination/Corrosion of Leads	• Improper Electrical Contacts
Damaged Seals & Gaskets	• Oil or Water Leaks*
Wrong Adjustments and Settings	• Overloading, Overheating
Faulty Heaters	• Too Much Moisture
Misapplication/Misoperation • Valve Seating • Phase Error • Frequent Starting	• Overloading, Overheating
Commutator Brushes (wear)	• Poor Connections*

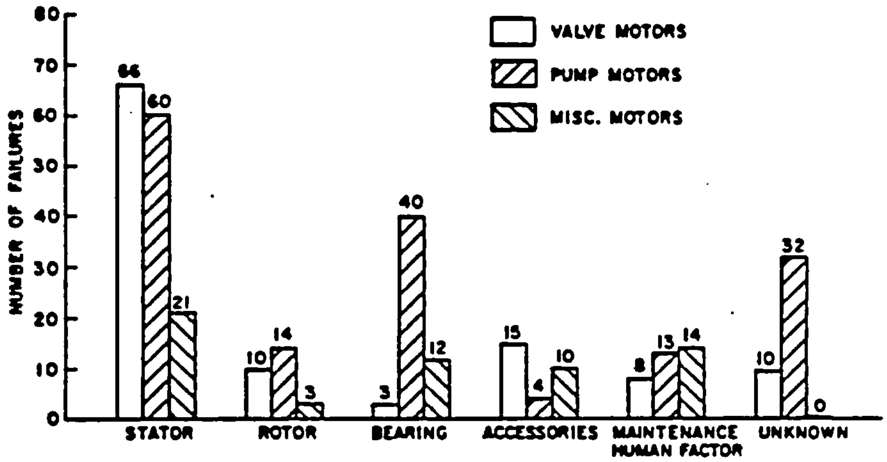
Functional Indicators:

- Vibration signature
- Environmental conditions
- Corrosion
- Cracks
- Leaks (seals & gaskets)
- Wear

Dominant Failure Mechanism (Motor Subsystem) → Performance Hazards → Failure Effects → Functional Indicators

* Aging Susceptibility per Table 4-15

Functional Indicators for Mechanical Integrity



Motor Component Failure Distribution (LER Data: 1974-1983)

Analysis of the LER data provided the following:

- o Motor failures that occur inside of the defined motor boundary are significantly greater than those that occur outside the boundary.

- o For BWR systems, pump and valve motors are equally prone to failure during normal plant operation, whereas for PWR systems, pump failures are more likely to occur.

An IPRDS data review revealed the following:

- o Pump motor failures are often control related.

- o The stresses caused by continuous operation in a motor have less deleterious effect on aging and service wear than intermittent or infrequent operation.

- o Vibration and moisture in-leakage are the prime causes of motor failure.

- o Most reported pump and valve motor failures are catastrophic (Note: This indicated that incipient failures are not being identified.)

A comparison of NRC license SALP (Systematic Assessment of Licensee Performance) ratings with LER motor failures indicates that licensees receiving below average maintenance ratings also experienced a comparatively higher number of motor failures. This condition demonstrates that improved preventative maintenance could prolong motor life and reduce the overall number of failures.

A review of additional data sources utilized served to reinforce foregoing conclusions:

- Motor failures typically occur, and are detected, while the machine is in the operational mode.
- Most degraded motor conditions are either in advanced stages or have resulted in catastrophic failures.
- Stator grounding and bearing related problems are the primary causes of motor failures.

4.0 Utilization of Research Results in the Regulatory Process

Focus on updating national codes and consensus standards including:

a. IEEE-323, Qualifying Class 1E Equipment for NPGS.

b. IEEE-344, Seismic Qualification of Class 1E Equipment for NPGS.

c. IEEE-334, Type Test of Continuous Duty Class 1E Motors for NPGS.

d. Reg. Guide 1.89, Environmental Qualification of Certain Electric Equipment Important for Nuclear Power Plants.

e. ASME-OM8, Performance Testing of Electric MOV Assemblies used in Nuclear Power Plants.

4.2 BATTERY CHARGERS AND INVERTERS NUREG/CR-4564 Dec. 1985 BNL/FIN NO. A3270

Operating Experience and Aging-Seismic Assessment of Battery Chargers and Inverters

1.0 Background

This report provides an aging assessment of battery chargers and inverters, and was conducted under the auspices of the NRC Nuclear Aging Research (NPAR) Program.

In characterizing the aging and service wear effects of battery chargers and inverters, this study considers the predominant designs utilized by the nuclear industry including the size and arrangement of this equipment in operating stations. This equipment is typically located in mild environments and is not subject to containment level environmental parameters. Because of their importance for safe shutdown of the plant, this equipment is required to be environmentally and seismically qualified. A discussion of the effects of component performance under operational and environmental conditions is provided. Several failure data bases are reviewed including Licensee Event Reports (LER), In-Plant Reliability Data Systems (IPRDS), Nuclear Plant Reliability Data Systems (NPRDS), and Nuclear Power Experience (NPE). They are reviewed to determine the failure modes, causes and mechanisms experienced in recent years by the nuclear industry and to prioritize the most significant modes of failures. The study also includes a discussion of manufacturer recommendations for maintaining reliable equipment. A review of industry and government standards relating to testing and maintaining of this equipment is included as well. Standards from the Institute of Electrical and Electronics Engineers (IEEE) and the National Electrical Manufacturers Association (NEMA) specifically address battery chargers and inverters.

Nuclear power plants utilize battery chargers and inverters to supply power to safety-related equipment, instrumentation, and controls. A battery charger converts alternating current (ac) to direct current (dc) to provide power to dc-driven equipment and components as well as to keep the standby batteries in a fully charged condition. Some plants are designed with a standby charger in addition to the required number of units (typically 2-4 per plant). On the other hand, inverters are used to supply ac-power to safety related equipment and equipment important to plant operation after converting the dc-power source to an ac output. A typical plant design requires at least two such units to distribute power to various control equipment vital to power and safe shutdown operations. Plant systems such as the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), Reactor Core Isolation Cooling (RCIC) System, and the ac/dc distribution system utilize these devices to provide the power to satisfy their safety functions.

Both battery chargers and inverters are considered together in this study because of their similarities in design, construction, and materials. The subcomponents, particularly the electronic elements, such as diodes, relays, capacitors, integrated circuits, etc. are the same in both equipment. They also serve related safety functions in the plant and experience the same environment as well as similar operational stresses.

The combination of required and desired options provided by the manufacturer results in each nuclear inverter and battery charger being somewhat unique, although certainly the major circuitry involved with each are similar for the specific models and manufacturers.

Four types of inverters and three battery charger types are presently used in nuclear applications. This equipment is discussed in detail with the advantages and disadvantages of each type noted. The size of the equipment varies with the electrical bus configuration used at the plant. Exceptions to this occur when individual utilities add non-safety but operationally important loads to the inverters or chargers which dictate that a larger unit be employed. The data received indicated a range of station battery charger sizes from ratings of one hundred amps to six hundred amps, while station inverter sizes varied from ratings of five kilowatts to two hundred kilowatts. For inverters, data were also collected for specific application inverters such as those used with the HPCI, RCIC, and Auxiliary Feedwater systems. These smaller inverters are generally rated less than one kilowatt. Some lower voltage battery chargers (24 and 48 volts) were also included in the data if a significant effect was clearly indicated, such as a loss of nuclear instrumentation. Manufactured by the same suppliers of the larger station chargers which usually operate at 125 volts, these smaller units were found to operate on the same principles and contain the identical components as the larger units.

Most of the battery chargers and inverts utilized in nuclear power plants were manufactured by six primary vendors. Although there exists several different designs, the physical appearance and construction of the units including the mounting of subcomponents inside the cabinet are similar. The size of inverters are designated by their voltage-ampere rating. The weight of an inverter varies from as low as 30 lbs. for a 0.5 KVA rated unit to 4,000 lbs. for a 200 KVA unit. Similarly, battery chargers have weights ranging from 60 lb. to 5,000 lbs. The physical size of the equipment also varies significantly.

2.0 Summary

To accomplish the identification of aging and service wear effects and appropriate inspection/surveillance/monitoring techniques, it was necessary to examine potential failure modes, mechanisms and causes. This was achieved by reviewing Battery Charger and Inverter design and materials of construction, by establishing the stressors that are both operational and accident related, and by reviewing existing failure related data. Aging-seismic correlation was addressed during this phase of the program. An interim review of current standards, manufacturer's recommendations, and condition monitoring techniques was performed in order to aid in the determination of future work.

Despite the use of redundant equipment and buses, failures, especially in the case of inverters, have resulted in reactor trips, inadvertent safety system injections, and emergency core cooling system unavailability. The consequences of battery charger failure is not as immediate, since a fully charged battery can supply dc power when a battery charger is unavailable. Depletion of the battery due to charger unavailability can, however, result in the loss of safety systems under post accident conditions. Inverter failure, on the other hand, directly interrupts the 120 volt ac power supply to vital controls, logic and annunciators.

The three types of battery charger designs are the Silicon Controlled Rectifier (SCR) solid state type, the controlled ferroresonant, and the magnetic amplifier (mag amp). While all three types are used at nuclear

facilities, the SCR or thyristor solid state charger is the most widely used, making up nearly 75% of the population, and, in fact, is the only charger type that is qualified to IEEE-323 and IEEE-650.

Four basic inverter designs are currently in use: the ferroresonant transformer, the pulse-width modulated, the quasi-square wave, and the step wave. The former two types are most often used, with the latter two types making up less than 20% of the inverter population.

Since battery charger and inverter are closely coupled in most nuclear plant application they are subjected to very similar system level stresses. That is, the charger output normally is connected to the inverter input, and the ac supply to the battery charger also provides alternate power to the vital bus either directly or through a rectifier in the inverter module. This is illustrated in Figure 1 which also depicts the alignment of offsite and onsite emergency ac sources.

3.0 Results/Findings

The charger and inverter subcomponents that are most susceptible to aging are capacitors, transformers and inductors, silicon controlled rectifiers (including diodes), and fuses. High voltage, current or temperature will affect all of these components, while transformers and inductors are also susceptible to excessive moisture. Fuses may fail due to thermal fatigue.

As one would expect, plant configurations help determine battery charger and inverter reliability. Those plants that have a standby charger or a second full capacity charger are the most reliable. For inverters, those

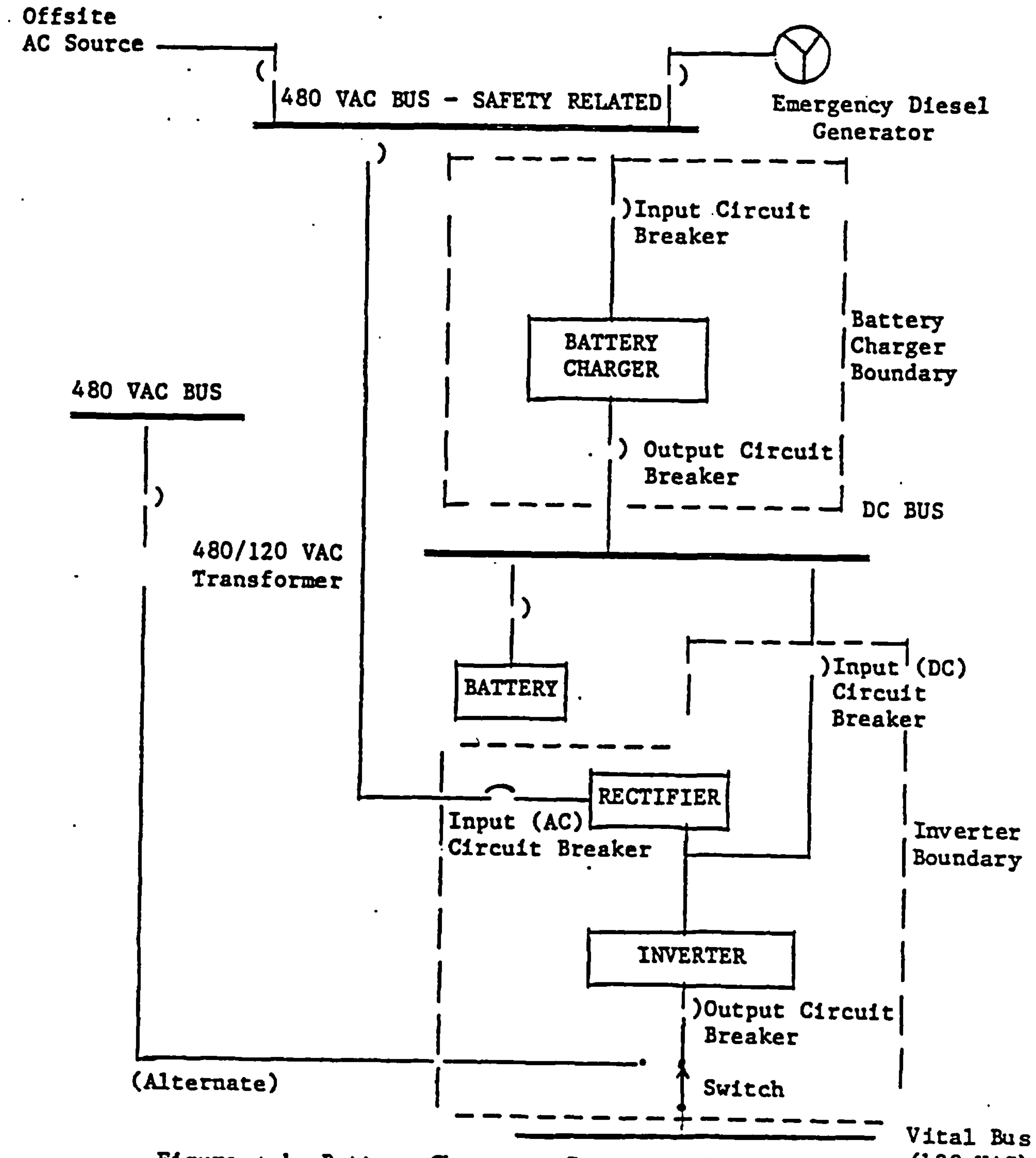
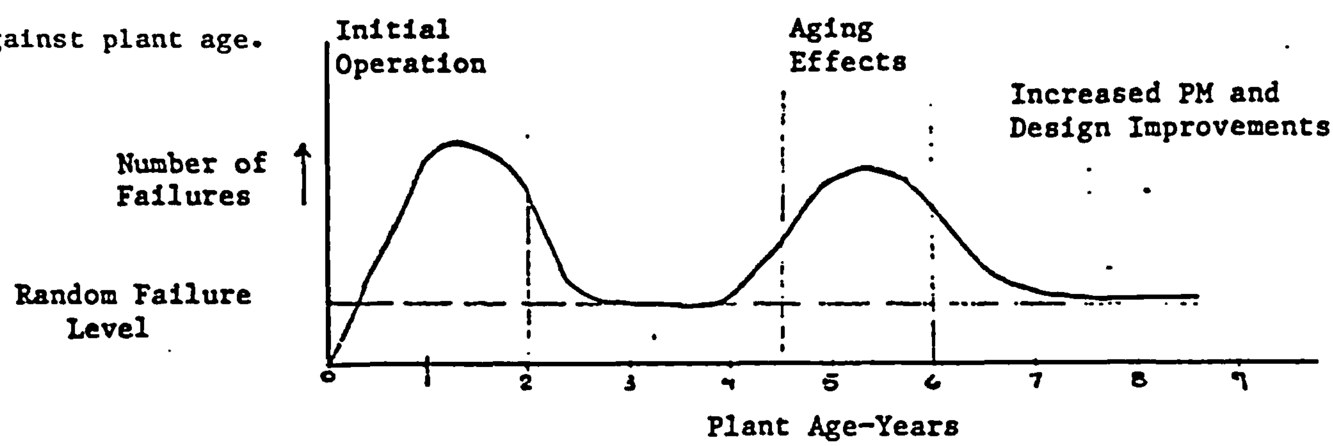


Figure -1 Battery Charger - Inverter Alignment plants with transfer switches, which allow a separate bypass ac feed appear to be the most reliable. Those plants with rectifiers providing an alternate dc feed to the inverters, have not been nearly as reliable.

The weak links which might be susceptible to seismic excitation are: cabinet mountings to floor or wall, subcomponent mountings, wire and cable connections, relays and circuit breakers, transformers, oil filled capacitors, and fuse holders.

Battery charger and inverter failures exhibit the typical "bathtub" curve when plotted against component age. That is, a high number of failures occur in the first year of operation with a pronounced wear-out effect in the fifth and sixth years of operation.

Increasing preventive maintenance scope and intervals, replacing troublesome equipment, and improving system designs are some of the actions taken by utilities who have experienced inverter and battery charger failures which have affected plant safety and availability. Improvements in materials and procedures also help to reduce the failure rate and could explain the shape of the curve obtained when plotting inverter and charger failures against plant age.



A summary of the results of the data analysis, with regard to failure modes and mechanisms and probability of occurrence are listed in Table 1.

The conclusions drawn thus far from this study include: (1) improvements in battery charger and inverter performance can be obtained through implementation of a comprehensive preventive maintenance program supported by appropriate personnel training; (2) due to the extensive systems interactions related to charger and inverter failures, it is recommended that plant procedures be in place to respond to these potential failures. Additionally, periodic capacity testing should be conducted to insure that the capability of this equipment to supply the required loads has not diminished due to the aging of key components.

Future work will consist of testing naturally aged battery chargers and inverters under normal and accident conditions to validate the performance indicators. Additionally, final recommendations will be established for inspection, surveillance, monitoring and maintenance programs.

4.0 Utilization of Research Results in the Regulatory Process

Present NRC interest in vital ac power, of which the inverter is an integral part, is documented in Issue No. 48, LCO for Class 1E Vital Instrument Buses in Operating Rectors. Research on this issue has resulted in recommendations such as the implementation of a 72 hour annual time limitation for energizing a vital ac bus from an interruptible power source, i.e., an alternate bus. Other potential recommendations include the standardization of technical specifications of vital ac buses, and the definition of minimum plant configuration requirements.

TABLE 1. Battery Charger/Inverter Failure Modes/Mechanisms/Causes

COMPONENT	FAILURE MODES	FAILURE CAUSES	FAILURE MECHANISMS	AGING	AGING SEISMIC	PROBABILITY OF OCCURRENCE
CIRCUIT BREAKER (Consists of contacts coil, mechanical linkages, case)	Fails to Operate	Build up of dirt, solidification of lubrication, bearing wear.	Increase in friction, binding.	Yes	Yes	Medium
	Fails Open	Metal fatigue, embrittlement & cracking of insulation.	Trip coil force becomes less than spring force.	Yes	Yes	Medium
		Oxidation & pitting of contact surfaces.	Loss of continuity across contacts.	Yes	No	Low
FUSE	Fails Open	Metal fatigue.	Equipment load cycling.	Yes	Yes	Medium
		Melting of link.	Heat generated by surrounding components.	No	No	Low
RELAY	Contacts Open	Oxidation & pitting of contact surfaces.	Loss of continuity across contacts.	Yes	Yes	Medium
	Open Circuit of Coil	Electromechanical action causing corrosion of fine wires.	Loss of continuity through coil wires.	Yes	No	Low
ELECTROLYTIC CAPACITORS	Loss of Capacitance	Overheating by internal stresses.	Loss of electrolyte.	Yes	No	High
	Open Circuit	Vibration.	Failure of leads.	Yes	Yes	Low
OIL FILLED CAPACITORS	Loss of Capacitance	Overheating forms gasses.	Dielectric breakdown.	Yes	No	High
	Open Circuit	Vibration.	Failure of leads.	Yes	Yes	Low

TABLE 1. Continued

COMPONENT	FAILURE MODES	FAILURE CAUSES	FAILURE MECHANISMS	AGING	AGING SEISMIC	PROBABILITY OF OCCURRENCE
MAGNETICS (Transformer Inductor)	Short Circuit-(turn to turn or to ground)	Temperature cycling/over heating. Low temperature.	Cracking of insulation. Cracking of moisture seals.	Yes	No	Medium
	Short circuit-(turn to turn or to ground)	High voltage stress.	Insulating material deterioration	No	No	Medium
	Change in inductance	Vibration/over temperature.	Change in shunting. Fracture of connecting wires.	Yes	No	Low
SILICON CONTROLLED RECTIFIER	Short or Open Circuit	Overheating.	Overvoltage, overcurrent due to transients.	No	No	Medium
RESISTOR	Open Circuit	Vibration.	Lead fails.	Yes	Yes	Low
	Change in Value	Internal or ambient temperature changes.	Decrease in resistance value as temperature increases.	No	No	Low
PRINTED CIRCUIT BOARDS	Change in Output	Temperature cycling.	Cracking of circuit lines.	Yes	Yes	Medium
		Corrosion.	Open circuit at terminals or within pcb.	Yes	No	Low
		Vibration.	Loose or open connection.	No	Yes	High
SURGE SUPPRESSOR	Short Circuit	Semiconductor barrier breakdown due to overheating.	Overvoltage, overcurrent.	No	No	Low

TABLE 1. Concluded

COMPONENT	FAILURE MODES	FAILURE CAUSES	FAILURE MECHANISMS	AGING	AGING SEISMIC	PROBABILITY OF OCCURRENCE
MISCELLANEOUS						
- Connectors	Open or Short Circuit	Installation stresses	Fatigue of wire at terminals.	Yes	Yes	Medium
- Meters	No Response (Stuck)	Buildup of dirt on movement.	Increase in bearing friction	Yes	No	Medium
		Overheating.	Coil insulation degrades causing shorting.	Yes	No	Low
- Switch	Fails Open or Closed	Contact pitting/corrosion.				
- Potentiometer		Thermal degradation.				

The Office for Analysis and Evaluation of Operational Data (AEOD) has continued to follow inverter performance, and has found unsatisfactory improvement in failure rates and failure consequences despite substantial regulatory and industry recommendations/requirements in this area. Transients, including reactor trips and safety injections, continue to occur because of inverter failure or vital bus degradation. In addition, because of the potential for a significant plant impact when an inverter fails, a second NRC group studying issue A-17, "System Interaction," has also initiated research into inverter performance on the systems level. This includes a review of significant instrumentation which is provided by the vital bus and the effect of its loss on operator performance during a severe transient event.

Similarly, in the area of dc power of which the battery charger is an integral part, the NRC has studied, and continues to research, the performance of safety related dc power supplies and the impact that a loss of dc has on plant safety.

In July 1977, the NRC issued a report (NUREG-0305) which addressed the reliability of dc power supplies at operating nuclear power stations, and discussed the likelihood and the consequences of a postulated failure of all dc during normal operation of a plant. The NRC staff concluded that because of the importance of ac and dc power systems, efforts to review the reliability of these systems should continue to be expended.

In April 1981, a second study of dc power was performed by the NRC (NUREG-0666) as part of the work in Issue A-30, "Adequacy of Safety Related DC Power Supplies." The issue here is primarily one of dc independence to minimize the potential for common mode failure. This study was a probabilistic safety assessment to determine the relative contribution of dc power related accident sequences to the total core damage probability. A significant finding of this PRA study was that a potentially large contribution could be reduced by requiring dc power divisional independence, and improved test, maintenance, and surveillance of dc components, including batteries and chargers.

4.3 <u>ELECTRICAL CABLE-ISMM</u> <u>NUREG/CR-4257-1</u> <u>Aug. 1985</u> <u>FRANKLING RESEARCH CENTER</u>
Inspection, Surveillance and Monitoring of Nuclear Power Plants - with Applications to Electrical Cables

1.0 <u>Background</u>

This report discusses the types of cables that are used in nuclear power plants, the causes and mechanisms of their failure, and the means of detecting age-related deteriorations that may lead to failure. The general concepts of equipment condition monitoring as applicable to the detection of age-related deterioration of safety-related equipment are evaluated. The goal of cable condition monitoring is to determine the degree of cable degradation and to predict the remaining useful life. In-situ nondestructive testing and destructive laboratory testing are discussed. Interim recommendations are given for the implementation of a cable condition monitoring program.

Initially, electrical cable appear to be simple devices that should be easy to monitor for age-related deterioration. The two primary components are insulation and conductors. However, further evaluation indicates that monitoring and evaluation of cable aging is a relatively complex subject. Many types of insulations are presently in use, most of which are organic nature. The types of deterioration vary for the different organic materials and may be significantly different for the same material when it is subjected to different stresses, combination of stresses, and different stress levels. Besides being composed of different materials, different types of cables are used for different functions. Instrumentation cable is significantly different from power cable both in construction and in the stresses to which it is exposed.

The testing of cables in a nuclear plant is limited by many concerns. The number of cable circuits is of a proportion that precludes testing of all circuits. Disruption of circuits for testing can lead to reduction of safety if restoration is not properly controlled. In some cases, tests that are of interest may be precluded by practical limitations of test equipment and suitability of the test to in-situ applications. Monitoring of aging parameters of cables is not simple, however, it is not impossible nor impractical.

Four basic types of electrical cables are used in nuclear power generating stations:

- Low voltage power cables (600 V rating)
- Medium voltage power cables (usually voltages of 4 to 13 kV)
- Control cables (low voltage, below 600 V, at a few amperes. Used to control devices such as valve operators.)
- Instrumentation cables (typically thermocouple wire, resistance temperature detector [RTD], cables, twisted pairs, and coaxial cables for transmission of data or instrument power. Also used for current-controlled devices [e.g., 20 to 50 mA], signal transmission, etc.).

Table 1 provides further breakdown of cable types and applications. It should be noted that 4-kV safety-related cable is not used inside primary containment and that 13-kV power systems are generally not safety-related. Some typical constructions of cable are illustrated in the report.

For a new BWR, the amount of cable within primary and secondary containment is approximately as follows:

Low voltage power	300,000 feet
Control	240,000 feet
Instrumentation (general)	1,300,000 feet
Instrumentation (neutron monitoring)	30,000 feet

Cable mounting configurations vary with the type and vintage of the plant. Cable may be located in trays or conduits inside of containment. In some newer BWR plants, most of the cable is located in conduits inside of primary containment. However, small sections of this cable may be exposed between the device housing and the end of the conduit section. Outside of containment, most cable is located in cable trays.

The most prominent modes of age-related failures of cable are electrical insulation degradation and termination failure. Cable insulation parameters that can be evaluated are insulation resistance, polarization index, high potential withstand capability, tensile strength, elongation, dielectric strength, partial discharge level, and dissipation factor. Test methods and implications of results are discussed as well as the limitations of performing in-situ testing.

An acceptable correlation of cable inspection, surveillance, and monitoring results with the continued ability to withstand DBA conditions is not presently available. Artificially aged cable has been used in accident simulation tests. However, data regarding the level of deterioration in the parameters of the aged cables prior to DBA simulation are not available to allow determination of acceptable levels of deterioration in cables that are in service.

Current utility practices pertaining to surveillance and testing of cables are discussed. Tests to measure insulation resistance are being conducted at some plants to identify trends for cable deterioration. Periodic high-voltage testing of cables connected to large motors is performed by some utilities, but is considered by other utilities to be a source of substantial degradation to the cables. The purpose of existing utility cable testing programs is to determine whether significant deterioration has occurred and whether replacement or further investigation is required. These tests are not condition monitoring tests in that remaining life is not estimated nor is the continuing ability to withstand DBA conditions evaluated.

Information obtained during this study was analyzed to develop interim recommendations for condition monitoring, surveillance, and maintenance of cables. It was found that sufficient data do not exist to formulate quantitative criteria, however, significant condition monitoring parameters were identified, and various observations were made in this report that can help formulate a basis for developing quantitative criteria. Discussion pertaining to interim recommendations is provided; which concludes that further investigation is necessary to develop criteria for choosing between continued operation of cables and taking corrective measures.

2.0 Summary of Testing Status

At present, an adequate correlation between the measurable aging parameters and the future functional capability during normal and DBA conditions of many types of equipment does not exist. Therefore, replacement or repair decisions are heavily based upon prior experience and engineering judgment with regard to test results rather than firm predetermined criteria.

The parameters that are used to monitor the aging of equipment are based on the aging mechanisms of the materials, parts, and interconnections involved. Assuming that the critical parameters indicative of the degradation of the equipment with aging can be identified and the correlation between a parameter and the functional capability is known, one must identify methods that can be used to monitor the parameters of installed equipment. In many cases, suitable techniques are not available that can be applied for continuous or periodic monitoring of the critical parameters. For example, in the case of power cables, the ability to perform in-situ partial discharge testing of long cables is limited. The available portable power supplies for the test are limited to approximately 30 kVA, which may be insufficient for long lengths of cable.

Condition monitoring of equipment is defined as continuous or periodic observation and evaluation of critical parameters to assess the equipment's ability to continue to perform its specified safety functions during a period following the moment of observation. Observation may take the form of measurement or inspection. The specified functions of the equipment include those required in the event of applicable accidents as well as normal service.

Condition monitoring is expected to reveal not only the functional state at the moment of observation, but also the ability of the equipment to remain capable of performing as specified for a period following the observation. Therefore, to be considered as condition monitoring techniques, diagnostic techniques must provide a basis for assessing the expected functional capability of the equipment should a DBA occur subsequent to the observation. Ideally, monitoring a single functional parameter would suffice to indicate the functional capability and exact criteria would exist to decide, based on the observed value of the parameter, what action to take at the time of observation, i.e., to continue operating without change, to perform maintenance or calibration, to repair the equipment, or to replace the equipment.

Conditions monitoring requires a data-collection and analysis system to determine the extent of degradation and the nearness of incipient failure.

The monitored parameter must be evaluated to decide what corrective actions, if any, are to taken. Analysis of data collected over a long time can help to determine the causes of degradation and the systematic preventive actions (such as scheduled inspection, preventive maintenance, or replacement) that will help avoid or reduce equipment degradation.

A brief summary of some of the important available tests pertaining to condition monitoring of cable insulation is provided. Table 1 lists cable type tests, their general application and specific application after cable installation.

Cost-effective techniques for adequate monitoring and detection of degradation and incipient failures are still in the developmental stage for most electrical equipment. In many instances, research may be required before practical techniques can be recommended. However, the existing state of technology has not been fully exploited.

Effective use of condition monitoring in nuclear power plants can improve plant safety and availability. Condition monitoring technology has the potential to allow relaxation of some of the stringent requirements for preventive maintenance and replacement of safety equipment by providing quantitative assessment and prediction of equipment degradation. Evaluation of condition monitoring results will allow establishment of maintenance and replacement intervals based on actual rates of equipment deterioration, thereby allowing improved allocation of resources. Less frequent maintenance will be performed on equipment found to deteriorate more slowly than expected, while more frequent maintenance will be performed on those that degrade more rapidly. The result will be on overall improvement in plant safety.

TABLE 1. Cable Tests and Applications

Test Description	General Application	Specific Cable Application After Installation
INSULATION TESTS		
Low Voltage Insulation Resistance	Troubleshooting, locating existing short circuits	Locating failures (isolating to a particular circuit) (NDE)
Insulation Resistance (500 V)	Determination of general acceptability of insulation. Usually performed at 500 Vdc	Requires shield or ground path to detect deterioration of insulation. If return path not provided, dry air between cable and ground will provide good insulation and may invalidate results. Can be used to determine general insulation level between conductors of paired cable. Does not predict failure. May be trended. May require isolation of certain interconnected equipment from test circuits to prevent damage (NDE)
Polarization Index	Service test to determine general state of insulation system. Frequently used on larger motors. May be used on cables. Uses 500 V dc test equipment	Compares charging and leakage current values at 1- and 10-minute intervals. Requires return path such as shield for measurement between conductor and ground. (NDE)
Elongation	Manufacturing test, provides indication of general acceptability of insulation	Can be used to determine if insulation is losing flexibility and is becoming embrittled (DE)
Tensile Strength	Manufacturing test, provides indication of general acceptability of insulation	Less telling than elongation test since cable conductor provides tensile strength needed after insulation. May indicate excessive softening of insulation (DE)
High Potential Withstand Test	Manufacturing test to prove insulation capability	Usually limited to higher voltage power circuits (4 kV and above). May be performed with connected load in place (e.g., transformer, motor). Limited to applications where isolated from other circuits and cables to prevent extensive damage from flashover or induced voltages. Most often performed at 80% or less of manufacturing test level. May be destructive to insulation and cause failure if not properly performed (test is usually a DC test). If properly performed, NDE

TABLE 1. Cable Tests and Applications (Cont.)

Test Description	General Application	Specific Cable Application After Installation
INSULATION TESTS		
Dissipation Factor Power Factor	Tests to determine electrical losses within insulation. More important for higher voltage equipment. This is an ac test	The test measures and compares resistive leakage current to capacitive leakage current. The measurements are more important for high voltage systems since losses are proportional to the square of the voltage. Increasing dissipation factors and power factors at higher voltages indicate insulation deterioration due to partial discharging
Partial Discharge Test	Measures corona discharge in voids of insulation. Corona discharge causes high speed ions to attack surrounging insulation and cause further deterioration. A high voltage ac test.	Test is generally not performed on cable in-situ. Would be reserved for suspected problem equipment generally in 13 kVac and above range. Requires shield or return path
Time Domain Reflectometry (Cable Radar)	By using a pulsed signal, the location of a fault (failure) in a cable may be determined by interpretation of the reflected signal. Useful for finding the location of a fault in a long cable	May be used to detect conductor deterioration at remote termination or penetration. May indicate significant deterioration of portions of cable insulation. May indicate contamination of remote terminals. May be used to detect deterioration of series connections or splices
CONDUCTOR/TERMINATION TEST		
Continuity (Loop) Resistance	Determination of circuit resistance variations indicating conductor or termination problems. For use on power circuits	Useful for determining changes in circuit resistance. May be used on motor circuits with windings in circuit. Significant changes in resistance would indicate the need to evaluate terminations, windings, and cable conductors for cause'

NDE Nondestructive Evaluation
DE Destructive Evaluation

The types and extent of cable testing that can be performed in-situ at a nuclear power plant using presently available techniques are limited by practical concerns. However, they do not preclude surveillance and monitoring of the aging of electrical cables through use of presently available techniques. The existence of the limitations requires thoughtful planning to assure that test results are meaningful and that inadvertent damage to the equipment does not occur. Care in selection of tests to be performed and the sample of cables to be monitored will overcome these limitations and help assure the usefulness of results.

3.0 Results/Findings

The results of the screening of representative failure modes and causes are presented in Table 2.

Further in-depth investigations pertaining to data bases such as NPRDS, TMI failure analysis reports, and maintenance records at power plants are necessary to establish correlations of cable insulation aging to various stresses.

Following are the recommendations for further study for developing quantitative criteria for surveillance, maintenance, and replacement of cables.

- Visit selected utilities to collect available data pertaining to age-related degradation of cables concerning failures and recognized flaws.
- Obtain a description of presently established cable testing programs from selected utilities.
- Identify the relevant research activities on cables sponsored by utilities (such as the recently initiated EPRI-sponsored research on equipment aging at the University of Connecticut), or the government, both in the U.S. and abroad, and extract the necessary information.

- o Select a group of experts on cables, consisting of cable users, manufacturers, and research laboratories, to provide input to formulate consensus quantitative criteria for condition monitoring, surveillance, maintenance, and replacement of cables, and to develop further practical tests to relate cable condition to remaining life.

4.0 Utilization of Research Results in the Regulatory Process

The principal application in the regulatory process is in support of the implementation of 10CFR50, Appendix B-Quality Assurance Criteria.

TABLE 2. Insulation Failure Causes and Mechanisms for Electrical Cables in Nuclear Power Plants*

Cable Type	Failure Causes	Failure Mechanisms
Power 600 V, 4 kV, and 13 kV	Loss of flexibility (cracks when flexed)	Chemical changes of polymeric structure resulting from exposure to heat, nuclear radiation, humidity, chemicals, etc.
	Loss of imperviousness (moisture readily reaches conductor through pores and cracks)	Same as above, plus possible physical abuse of cable (chafing, cutting, small radius bending)
	Loss of adequate dielectric properties, inability to withstand voltage stress	Excessive exposure to high electric stress, high temperature, or large radiation dose. Failure frequently coupled with presence of moisture or water. Loss of dielectric properties generally occurs after deterioration of mechanical properties. Loss of dielectric properties would be more rapid at points of insulation stress conditions such as mechanical abuse locations or areas of repeated small radius bending. For 13 kV and above cable treeing, water treeing, and electrochemical treeing may cause relatively rapid breakdown of dielectric capabilities.

* Failure of cable sheathing can also occur, but the sheathing is not relied upon for operability of cables. The major failure modes for sheathing are loss of flexibility and imperviousness; the causes are similar to those for cable insulation.

TABLE 2. Cont.

Cable Type	Failure Causes	Failure Mechanisms
Control	Loss of flexibility (cracks when flexed)	Same as for power cable failure causes
	Loss of imperviousness (moisture readily reaches conductor through pores and cracks)	Same as for power cable failure causes
Instrumentation	Loss of flexibility (cracks when flexed)	Same as for power cable failure causes
	Loss of imperviousness	Same as for power cable failure causes
	Adverse change in dielectric properties [e.g., insulation resistance and/or capacitance between conductor(s) and shield(s)] which may cause attenuation of signal	Exposure to heat, nuclear radiation, humidity, chemicals, etc., creating chemical and/or physical changes to the insulation. Significant changes in dielectric properties can be expected during accident conditions. These effects may be additive with slower changes due to aging.

4.4 PRESSURE TRANSMITTERS-ISMM NUREG/CR-4257-2 ___ FRANKLIN RESEARCH CENTER

Inspection, Surveillance, and Monitoring of Electrical Equipment Inside Containment of Nuclear Power Plants - Pressure Transmitters

1.0 Background

Most of the pressure transmitters in use in nuclear plants are actually differential pressure transmitters, in which a pressure is applied to both sides of the sensing element and the difference between them is measured.

Pressure transmitters are commonly used to measure gage pressure, absolute pressure, differential pressure, liquid level, and flow rate. To measure gage pressure, one side of the detector is exposed to atmospheric pressure and the other side to process pressure. To measure absolute pressure, one side of the detector is evacuated and sealed, and the other side is exposed to the process pressure.

To measure flow rate, a differential pressure transmitter may be connected across a flow-restricting orifice in a process line and the pressure drop produced across the orifice measured. Within limits, the pressure drop is proportional flow rate.

Differing techniques are required to measure liquid level with a pressure transmitter. For nonpressurized vessels, the high-pressure port of the pressure transmitter may be connected to a tap in the vessel located at or below the minimum level to be detected and the low-pressure port of the transmitter vented to atmosphere. The liquid head of the vessel then produces a pressure indication proportional to the level of the liquid in the vessel. For pressurized vessels, the high-pressure port of the transmitter may be connected to a tap in the vessel located at or below the minimum level to be detected as above. However, the low-pressure port is connected to the top of the vessel by a sensing line filled with the liquid contained in the vessel. The filled line acts as a pressure reference to allow the level of the pressurized liquid to be determined from the differential pressure that is indicated.

Pressure transmitters are used in nearly all safety systems having liquid or gaseous process media. They are used to initiate and monitor protective actions requiring reactor trips, containment isolation, and starting and stopping safety actions based on system conditions; to monitor and control reactor pressure, coolant level, and main steam line flow rate. Pressure transmitters prevent operation of low-pressure safety systems, such as residual heat removal in a boiling-water reactor, until reactor pressure is reduced to a safe operating level. Also, they are used to detect high pressure in the containment building and to initiate containment isolation and pressure control. Depending on the vintage of the nuclear power plant, 60 to 200 pressure transmitters are used per plant.

2.0 Summary

This report describes the types of pressure transmitters commonly used in nuclear power plants. The applications of pressure transmitters for detection of gage, absolute, and differential pressures; levels; and flow rates are described.

Pressure transmitters are subject to stresses from their environment, the process they are monitoring, and their electrical power supply. Under normal plant conditions, the environmental stressors are temperature, humidity, and radiation. Nuclear application transmitters are sealed for DBA environment steam conditions; therefore, normal environment humidity is generally not a problem. However, elevated temperature and radiation may affect the environmental seals and the electronic assemblies over a period of time. The seals may harden, crack, or take a set, allowing steam in-leakage under DBA environments or damp conditions during normal service.

The process to which the transmitter is connected may also affect the temperature of the transmitter; however, many transmitter applications have relatively long (50 to 200 ft) sensing lines that limit heat transfer from the process. Stresses from process transient overpressure conditions, such as water hammer, may cause significant zero shifts in transmitter settings. These stresses can cause permanent deformations of sensing elements or of mechanical linkages within the transmitter mechanism. Generally, these zero shifts can be removed by recalibration. However, recalibration is not possible for in-containment devices during periods of reactor operation.

Pressure surges may occur when the transmitter is valved in and out of service if the valving operation is not done in the proper sequence. For many applications, this condition would cause the sensing cell to be forced out of the calibrated span and could cause a calibration shift. In general, this shift would be correctable during the next calibration and would not cause permanent damage. However, the shift would be present for the entire period between calibrations. Calibration shifts may also occur if the transmitter is calibrated at one temperature but operates at a significantly higher temperature.

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

The evaluation of calibration data alone will not ensure the capability of operating under design basis accident (DBA) conditions. If steam or moisture penetrates the transmitter housing, the transmitter electronics will be affected, become inaccurate, and most probably fail. Therefore, the integrity of the housing seal must also be evaluated periodically to be able to predict continued capability to perform under DBA environments. Three Mile Island-2 report GEND-INF-029 provides a good example of water incursion damage to a transmitter. In this case, the water in-leakage is believed to have been by way of the electrical conduit or fittings. The damage was extensive; components were badly corroded, a transformer was badly burned, and the structural integrity of one transistor was completely lost.

The boundary of the transmitter evaluated here is comprised of the process medium connections (ports), not including the sensing lines or test valve manifold; the support backed for the transmitter; and the electrical signal terminals. The boundary also includes the housing seal system, including the wiring penetration seal.

Tables 1, 2, and 3 are a compilation of License Event Report (LERs) for pressure transmitters by application (pressure, level, and flow rate sensing). The utility personnel contacted and the ORNL survey indicated that between 60 and 200 transmitters are used in safety-related applications in each plant. The LERs indicate that in a 3-year period 330 reportable failures occurred throughout the nuclear power industry. During this same period, ~65 plants were in operation. These numbers indicate that less than two reportable failures occurred per year per plant. The approximate failure rate indicated is 0.02 failures/year (2.4×10^{-6} failures/h) for each transmitter.

These values indicate that transmitters are relatively stable devices under normal power plant conditions. However, the above information does not include consideration of DBA environment conditions that may affect failure rates significantly, nor does it include transmitter failure information that was nonreportable under the LER system.

Some of the originally installed transmitters in older plants are being replaced because they do not meet present environmental qualification requirements. The replacement transmitters have been qualified to IEEE Standard 323-1974.

TABLE 1 PRESSURE TRANSMITTER FAILURE DATA FROM LERs FROM SEPTEMBER 1982 TO APRIL 1984 (40 pertinent LERs)

Failure mechanism	Number of events	Percent of total
Calibration drift	32	55
Transmitter failure (exact cause not given)	6	15
Sensing cell failure	5	12.5
Amplifier failure	2	5
Frozen sensing line	2	5
Cause unknown	2	5
Gas in sensing line	1	2.5

TABLE 2 LEVEL TRANSMITTER FAILURE DATA FROM LERs FROM JANUARY 1981 TO APRIL 1984 (128 pertinent LERs)

Failure mechanism	Number of events	Percent of total
Calibration drift	68	53
Transmitter failure (exact cause not given)	23	18
Electronics failure (most frequently amplifier)	17	13
Sensing cell failure	6	5
Frozen sensing line or transmitter	3	2
Sensing line plugging	2	1.5
Gas in sensing line	2	1.5
Force motor failure	2	1.5
Mechanical linkage problems	2	1.5
Corroded internals	1	<1
Unknown cause	1	<1
Loose connection	1	<1

TABLE 3 FLOW TRANSMITTER FAILURE DATA FROM LERs FROM JANUARY 1981 TO DECEMBER 1983 (109 pertinent LERs)

Failure mechanism	Number of events	Percent of total
Calibration drift	47	43
Transmitter failure (exact cause not given)	23	21
Electronics failure (most frequently amplifier)	12	11
Sensing line plugged	8	7
Cause unknown	5	4.6
Sensing element failure	3	2.7
Mechanical linkage problems	3	2.7
Force motor and magnetic transducer failure	4	3.6
Wet transmitter internals	2	1.8
Internal power supply failure	2	1.8

3.0 Results

Table 4 provides a listing of failure modes, failure causes, and potential failure mechanisms and their effects for force balance, capacitance cell, and strain gage type transmitters. While these lists indicate the possibility of many failure mechanisms, the actual failure rate for many of these mechanisms is very low.

The most common effects of the stressors on the transmitters are calibration shifts. The evaluation of failure data contained in Licensee Event Reports indicates that total failure of pressure transmitters occurs relatively infrequently. Discussions with plant technical personnel confirmed that transmitters are not considered a significant source of trouble.

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

A combination of operability monitoring and condition monitoring may be used to improve the probability of successful weathering of DBA conditions.

4.0 Utilization of Research in the Regulatory Process

The results of this and subsequently planned studies will be useful in the regulatory process and can be factored into this process through support of the implementation of:

1. Technical Specifications

2. Surveillance Requirements - 10CFR50.36(e)

3. Updating appropriate regulatory guides, including these endorsing national standards IEEE-317, 323, 383, and 535.

TABLE 1. Transmitter Failure Modes, Causes, Mechanisms, and Effects[a]

Failure mode	Subcomponent	Failure cause	Failure mechanism	Effect on device
		Force balance type transmitter		
Failure to operate	Force bar and linkage	Wear of pivot points	High vibration or improper installation	Decreased accuracy or complete failure
	Feedback coil (force motor)	Coil burnout	Overheating of coil insulation due to coil insulation failure	Loss of output
	Amplifier	Shorting or opening of electronic components (e.g., capacitors, transistors)	Thermal or voltage overload due to power source surges or deterioration of components	Depending on type of failure, may fail high or low, lose accuracy, or fail with steady output
	Housing seals	Inability of seal to provide moisture and pressure barrier	Compressive set of organic seals or embrittlement and cracking due to thermal and radiation stresses	Failure of electronics due to shorting and corrosion from ingress of environmental contaminants
Failure to operate as required	Diaphragm	Diaphragm through leakage	Perforation of diaphragm from corrosion or flaw	Variable instrument drift as pressures across diaphragm equalize
		Permanent deformation of diaphragm	Overpressurization due to transient	Zero shift
	Diaphragm seal	Inability to maintain pressure barrier	Seal deterioration from compressive or decomposition	Variable instrument drift as pressures across diaphragm equalize
	Force bar and linkage	Bending of components in level system	Overpressurization due to transient	Zero shift

TABLE 1. (Continued)

Failure mode	Subcomponent	Failure cause	Failure mechanism	Effect on device
		Capacitance type transmitters		
Failure to operate	Sensing cell	Leakage of cell fluid through diaphragm	Perforation due to flaw	Loss of accuracy and drift; rupture allows equalization of forces on diaphragm
	Terminal cover seal	Inability of seal to provide moisture and pressure boundary	Compressive set of organic seals or embrittlement and cracking due to thermal and radiation stresses	Loss of electronics due to ingress of environmental contaminants
	Electronics housing seal	Inability of seal to provide moisture and pressure boundary	Compressive set of organic seals or embrittlement and cracking due to thermal and radiation stresses	Loss of accuracy to complete failure possible due to ingress of environmental contaminants
	Electronics	Circuit continuity lost	Circuit board retaining screws improperly tightened or loosened due to vibration	Loss of signal, sporadic operation
		Circuit continuity lost, bridging of circuits	Oxidation of contacts due to contamination during repair or after housing seal failures	Loss of signal, sporadic operation
		Shorting or opening of component (e.g., capacitor, transistor)	Thermal or voltage overload due to power source surges or deterioration of components	Loss of output, may fail high or low
Failure to operate as required	Sensing cell	Leakage of process fluid into cell fill fluid	Perforation due to corrosion or flaw	Loss of accuracy (contamination of cell oil changes capacitance of cell)
		Volume of cavity changes when seal moves	Dislodging of tube seal due to overpressurization	Zero shift

TABLE 1. (Concluded)

Failure mode	Subcomponent	Failure cause	Failure mechanism	Effect on device
Failure to operate as required	Sensing cell	Oil chemical changes	Oil breakdown due to thermal or radiation stress	Zero shift, reduced accuracy, or changes in response time
	Electronics	Change in component parameter (e.g., amplification, resistance, capacitance)	Deterioration of component from thermal, radiation, or voltage stresses	Loss of accuracy, drift, zero shift
		Strain gage type transmitters		
Failure to operate	Strain gage	Loss of continuity in bridge circuit	Primary system transient or flaw	Loss of output
	Housing seal, lead wire seal	Inability to provide moisture and pressure barrier	Compressive set of organic seals or embrittlement or cracking due to thermal and radiation stresses	Failure of electronics due to contamination
	Potentiometer	Corrosion of resistive elements in potentiometer	Wire-wound potentiometer corrodes open due to thermal stress and corrosive lubricant[7] (does not occur if proper lubricant is used during manufacture)	Fails over range
	Electric module	Change in component parameter (e.g., amplification, resistance, capacitance)	Component deterioration or failure (transistor, diode, capacitor, resistor)	Loss of full output
Failure to operate as required	Bourdon tube	Permanent deformation of tube	Overpressure transient	Zero shift
		Bourdon tube leaks to transmitter housing	Perforation due to corrosion or flaw	Drift, contamination of transmitter internals, failure to respond

[a]Failures external to the transmitter, such as power source and sensing line problems, are not included.

4.5 DIESEL-GENERATORS NUREG/CR-4590 APRIL 1986 PNL/FIN No. B2865

Aging of Nuclear Station Diesel Generators: Evaluation of Operating and Expert Experience

1.0 Background

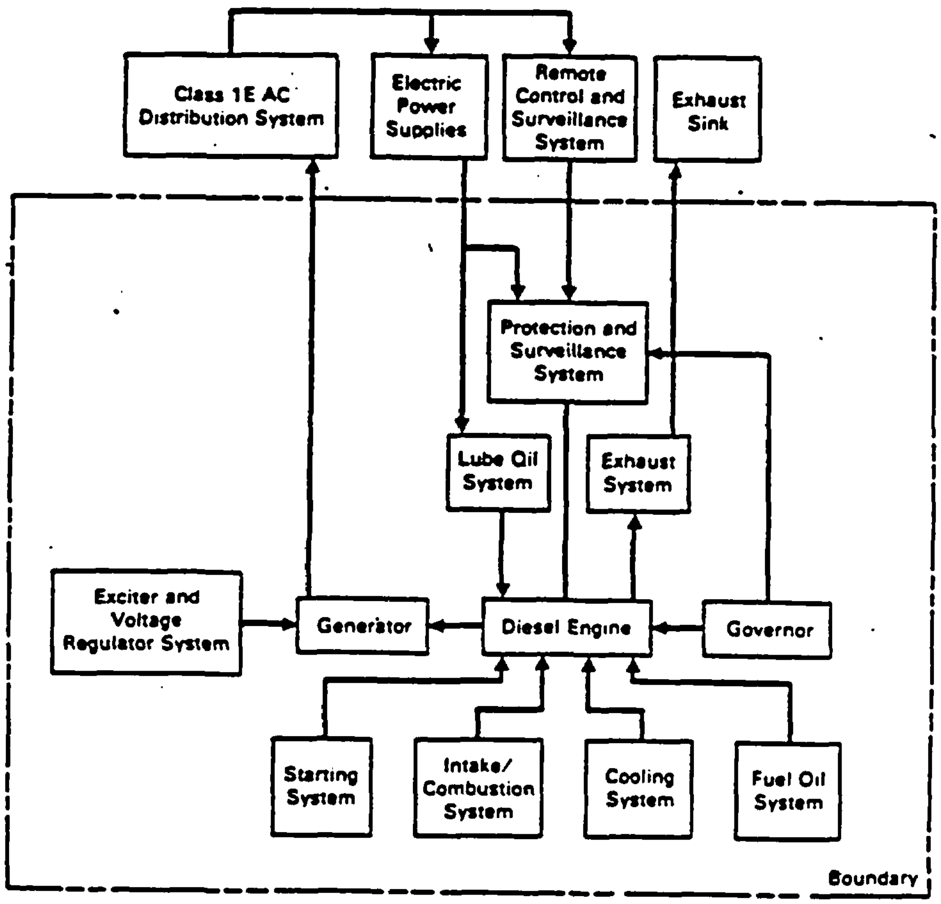
Emergency diesel generators (EDG) at nuclear plants provide class 1 emergency (1E) standby power to support the operation of emergency safety and plant protection system loads. During loss of power to the emergency buses from the main generator or offsite sources, the EDG system must provide backup power to operate critical reactor safety equipment.

More than 200 diesel-generators of various models, by at least nine manufacturers, are in service as EDG at nuclear plants in the United States. The diesel models, and types currently in use at nuclear plants, their major components, and their support systems are briefly identified in this report. The number of engines in nuclear service evaluated in this report, are listed below by manufacturer:

Manufacturer	Number in Service in Nuclear Industry
ALCO	18
Allis Chalmers	3
Caterpillar	5
Cooper Bessemer	11
Fairbanks Morse	49
Electro-Motive Division	84
Nordberg	8
Transamerica Delaval	31
Worthington	4

The diesel-generator is a complex, small power plant within a nuclear facility. Like all mechanical and electrical equipment, diesel-generator systems and components are subject to aging. Each system and component is affected by its operational history and environment. Engine systems deteriorate, or age, both while operating and standing idle. Since diesel generators in nuclear service are not operated for most of their installed lives, the diesel generator and certain related systems should be considered unique from the aging standpoint when compared to many other systems which are used extensively.

2.0 Summary of Systems Associated with the Diesel Generator



Interfacing Subsystems of the Diesel Engine System

3.0 Results/Findings

The evaluation of nuclear plant operational experience and expert opinion concerning the aging of nuclear service emergency diesel-generators indicates: that it is observable; follows recognizable patterns; shows changes in the modes of aging degradation with time; is confined to few, relatively major components; increases as a percentage of all failures with time; is caused by normal operational stressors. As a result of the review of 1,984 failures associated with diesel-generator systems from 1965 through 1984, 1,336 were classified as abnormal failures and 629 as aging related failures.

OVERALL CAUSES OF DIESEL-GENERATOR FAILURE

Failure Cause	Percent of All Failures[a]
Poor manufacturing or construction quality control	22
Adverse conditions: vibration, shock	17
Human error - maintenance	13
Adverse environment: dust humidity, chemicals, etc,	13
Unknown	11
Maladjustment/misalignment	8
Other	16

(a) Includes aging and non-aging failures.

SYSTEMS & COMPONENTS CONTRIBUTING MOST TO ALL DIESEL-GENERATOR FAILURES

Systems and Components	Percent of All Failures	
Instruments and Controls System	25	
Governor		10
Sensors		3
Relays		2
Startup components		2
Fuel System	11	
Piping on engine		3
Injector pumps		2
Starting System	10	
Controls		3
Starting air valve		2
Starting motors		2
Air compressor		1
Switchgear System	10	
Breakers		3
Relays		5
Instruments and controls		1
Cooling System	9	
Pumps		2
Heat exchangers		2
Piping		2
Lubrication System	7	
Heat exchangers		2
Pumps		2
Lube Oil		1
Other Systems	28	
	100	

CAUSES OF DIESEL-GENERATOR FAILURE DUE TO AGING

Failure Cause	Percent of Aging Failures
Adverse conditions: vibration, shock	27
Poor manufacturing or construction quality control	18
Adverse environment: dust, humidity, chemicals, etc.	17
Unknown	14
Human error - maintenance	9
Maladjustment/misalignment	6
Poor design: wrong application or component	5
Other	4

SUMMARY OF SIGNIFICANT FAILURE RATES(1) DUE TO AGING BY MANUFACTURER, SYSTEM, & FAILURE CAUSE

	ALCO	AC	CAT	CB	EMD	FM	NORD	TDI	WORT
Failure Cause									
a. Poor design: wrong application or component	—	—	—	—	—	—	—	12	—
b. Poor manufacturing or construction/quality control	22	14	17	23	19	12	—	34	21
c. Adverse conditions: vibration, shock	22	—	17	37	23	31	38	24	38
d. Adverse environment: dust, chemicals, etc.	17	43	17	7	20	19	17	9	13
e. Maladjustment/misalignment, etc.	—	—	—	—	—	—	15	—	—
f. Human error - maintenance	14	14	17	9	8	9	—	—	—
g. Unknown/other	15	29	33	10	17	12	11	—	21
Systems									
Instruments and controls	36	43	50	36	27	21	34	24	29
Cooling system	15	14	33	—	12	7	17	11	—
Fuel system	14	—	—	19	7	19	19	18	21
Lubrication system	8	—	—	7	8	7	—	—	—
Generator	—	14	—	—	7	—	—	—	—
Switchgear	—	14	—	—	7	—	—	—	—
Engine structure	—	—	—	9	—	14	—	8	—
Starting system	—	—	—	7	17	—	—	8	21
Intake and exhaust system	—	—	—	—	—	8	—	11	—
Drive train	—	—	—	—	—	—	—	8	—

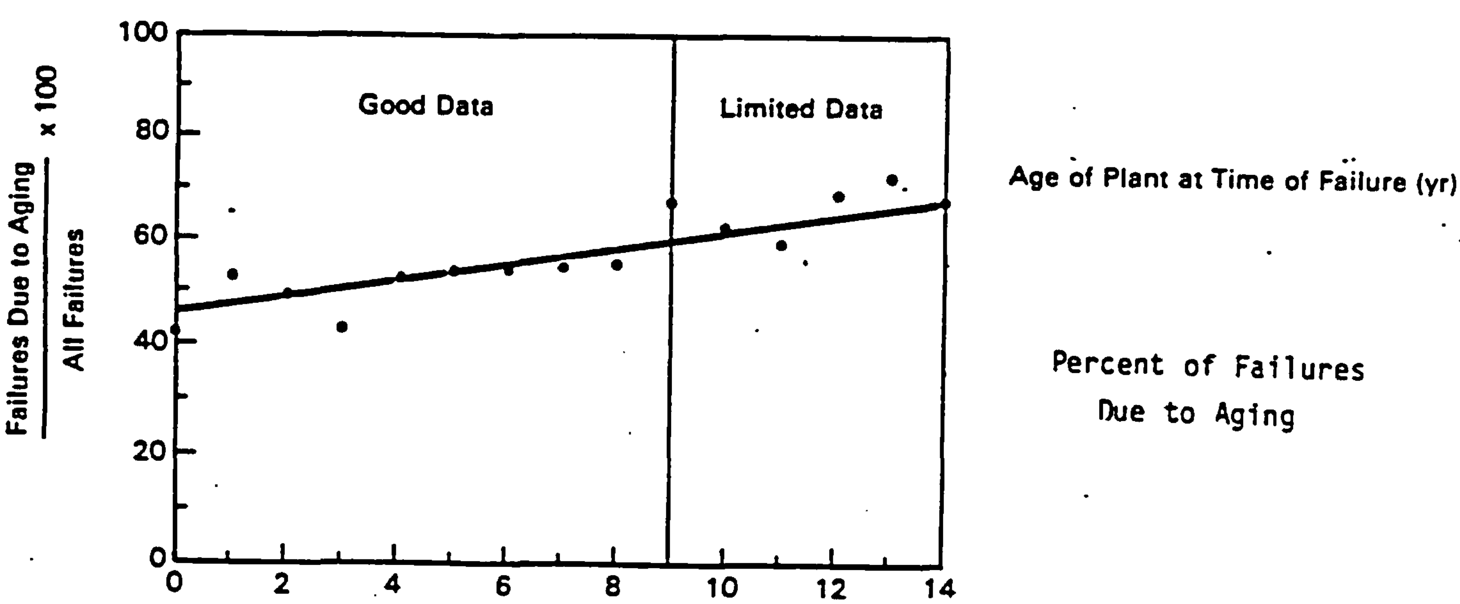
1) Normalized to aging failures per 100 aging failures x 100.

An extensive Aging Failure Matrix is given in Table 4.1 of the original report and extends from page 4.4 through 4.11. This table lists the system and component, the component failure potential (high, medium, or low), the type of failure (chemical or mechanical) plus some comments when appropriate.

In order to develop a potential list of measurable parameters they purpose to conduct an active and thorough trend analysis program to:

- o Identify potentially problematic components early in component life.
- o Employ the latest methods for on-line system monitoring and diagnostics, including full-spectrum, fast Fourier transform (FFT) vibration monitoring.
- o Digitize trend analysis readings for a computerized comparison of results.
- o Compare performance histories with similar installations, nuclear and non-nuclear.

The figure below shows the percent of failures due to aging degradation for any plant at a particular age. The resultant graph shows the changing ratio of aging related failures to all failures. A plot of this ratio for each year from construction and startup (0 years) through 14 years of operation reveals that aging degradation is an increasingly large percentage of total failures for nuclear service diesel generators. This figure also shows the ranges and quality of data used for the plot.



Percent of Failures Due to Aging

4.0 Utilization of Research Results in the Regulatory Process

The results of this study and subsequent studies planned will be useful to the regulatory process and can be factored into this process through:

- A NUREG report on diesel-generator aging
- Recommendations for diesel-generator maintenance
- Changes to RG 1.32, 1.108, 1.118
- Changes to applicability of ASME Section III vibration design standards to diesel-generators
- Changes to ASME Section XI applicability to diesel-generators
- A review of the safety related status of the diesel-generators governor systems
- A revision of the Class 1E electrical equipment qualifications standards as associated with diesel-generators
- Changes to DEMA standard practices
- Changes to related ASTM standards
- Other application to ongoing NRC efforts related to emergency diesel-generators:
  - Analysis of trends and patterns (AEOD LEAD): Evaluate the applicability of methods of analyzing trends and patterns from the NPRDS, (a) survival analysis, (b) multiway tables and log linear modeling, (c) modeling of known reliability models to failure data, (d) trend analysis, (e) detection of shifts in failure rates.
  - Evaluation of Transamerica Delaval, Inc. (TDI) diesel engines (NRR LEAD): Review the technical efforts of the TDI owners' group centered on generic problems, and confirm the design and quality of about 200 other key engine components.
  - Unresolved safety issue A-44. Station Blackout -- (NRR LEAD): The proposed resolution also includes a recommendation that the reliability of each EDG should be maintained at or above specified acceptable reliability levels. The objective of the EDG reliability program is to develop the details in support of improving and maintaining EDG reliability consistent with the resolution of USI A-44.

- Diesel Generator Reliability. (RES LEAD): Evaluate means of assuring diesel generator reliability. The study is analyzing diesel failures and maintenance procedures and uses information from Trojan Nuclear Plant.

- Other Related Tasks -- (IE LEAD): Evaluate the results of NRC/Industry meeting held in April 1985. Review industry responses to: an NRR letter to utilities pertaining to possible changes in R.G.1.108; evaluate ACRS concerns; and analyze Japanese experiences.

4.6 MOTOR-OPERATED VALVES (MOVS) NUREG/CR-4234 VOL.1 JUNE 1985 ORNL/FIN NO. B0828

Aging and Service Wear of Electric Motor-Operated Valves used in Engineered Safety Feature Systems of Nuclear Power Plants

1.0 Background

This is the summary of the Phase I report on electric motor-operated valves (MOVs) to be produced under the Nuclear Plant Aging Research Program (NUREG-1144). The program includes the evaluation and identification of methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of MOVs in nuclear plants. These methods are to provide capabilities for establishing degradation trends prior to failure and developing guidance for effective maintenance.

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

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

2.0 Summary of MOV Application in Nuclear Power Plants

System	Number of MOVs	Valve size (in.)[a]	Valve type
BWR (typical)			
Low-pressure core spray	8–12	2–28	G,[b] Gl[c]
High-pressure coolant injection	8–14	4–24	G, Gl
Low-pressure coolant injection (include residual heat removal and containment spray)	28–50	4–24	G, Gl
Reactor recirculation system	8–10	2–28	G
Reactor core isolation cooling	8–10	3–6	G, Gl
Containment isolation	4–14	3–24	BF,[d] G
BOP systems	50–150	2–60	G, Gl, BF
PWR (typical)			
Auxiliary feedwater	6–10	4–8	Gl
Chemical and volume control	8–10	3–8	G
Containment spray	8–12	6–14	G, Gl, BF
Deisel generator cooling	8–10	4–12	Gl
High-pressure coolant injection	10–24	2 1/2–4	Gl
Heating, ventilation, and air conditioning	6–10	6–48	BF
Low-pressure coolant injection/residual heat removal	12–30	8–10	G
Safety injection	15–54	3–14	
Refueling water storage tank			G
Containment spray			G
Containment sump			G
BOP systems	50–150	2–60	G, Gl, BF

[a]Nominal pipe size.
[b]G — gate valve.
[c]Gl — globe valve.
[d]BF — butterfly valve.

3.0 Results/Findings

The NPRDS, operated by INPO, contains component and failure data that can be obtained on special request. In response to a special request, INPO provided a data search that yielded 722 failure events for this study. The events were then manually sorted as to failed component, which was usually found in the narrative portion of the event description. Additional manual sorting was performed for manufacturer, failure mode, cause, type, and method of detection. Tables 3.1-3.6 summarize the results of this effort.

3.1 Failure Events Related to Failed Components

Failed component	Number	Percent
Limit switch/torque switch	275	38.1
Power supply/controls/wiring	133	18.4
Motor	125	17.3
Mechanical internals	90	12.5
Housing, bolts	27	3.7
Valve	17	2.4
Lubrication	13	1.8
Not determined	42	5.8
Total	722	100

3.2 Failure Events Related to Manufacturers

Manufacturer	Number	Percent
Limitorque	667	92
Rotork	18	3
Other	37	5
Total	722	100

3.3 Failure Mode Distribution

Failure modes	Number	Percent
Will not open	144	20
Will not close	122	17
Will not start/move	93	13
Out of adjustment	68	9
Out of limits	55	8
Will not operate per demand	45	6
Fracture/break	21	3
Spurious operation	18	2
False response	17	2
Crack	16	2
Miscellaneous (23 categories)	76	11
Other	47	7
Total	722	100

3.4 Failure Types

Event (type) failure	Number[a]
Electrical	368
Mechanical	359
Electronic	34
Other instrument	18
Corrosion	10
Other chemical	3
Nuisance	2
Human error	2

[a]Some events had more than one failure type indicated.

3.5 Failure Caused Distribution

Cause	Number	Percent
Burned/burned out	68	9
Set point drift	65	9
Abnormal wear	58	8
Short/grounded	45	6
Connection defective	43	6
Material defect	41	6
Open circuit	40	6
Instrument/switch miscalibration	39	5
Circuit defective	36	5
Miscellaneous (16 categories)	143	20
Other	124	17
None given	20	3
Total	722	100

3.6 Method of detection

Detection	Number
Operational abnormality	257
Surveillance testing	247
Routine surveillance	88
Incidental observation	57
Visual/audio alarm	57
Preventative maintenance	25
Special inspection	23
In-service inspection	20
Other	7

[a]Some events had more than on detection code listed.

3.7 Summary of MOV aging information available from LER-based valve failure studies

Identified failure mode or causes	Frequencies of failures	Mechanical, maintenance, and surveillance modifications	Affected systems[α] BWR	Affected systems[α] PWR	Remarks
nherent[b] aintenance error esign error	No	No	FW and MS conditions Containment isolation ECCS	Containment isolation Reactor coolant ECCS	Data based on RECON keyword search for valve failure LERs
ailed to open or close as required nternal leakage	LER rate = 7×10^{-3}/day for MOVs	No	Containment isolation Main steam LPCI/RHR	Containment isolation Nonsafety systems Main steam	Gross constant failure rates estimated for major valve types in selected safety systems — results should be used with caution
orque switch failure otor damage rong lube	No	Resize motor operator Improve maintenance procedures	No	No	Recommended standards for: MOV actuator specifications Sizing procedures Maintenance and handling procedures R&D of MOVs
versized operators orque switch failure or misadjustment	No	No	Main steam ECCS	Reactor coolant Main steam	Recommended standards for: MOV actuator specifications Sizing procedures Maintenance and handling procedures R&D of MOVs
versized operators mproper thermal overload ndersized operators	No	Larger valve stems Replace overload heaters Resize motor operators Training of maintenance personnel	NSS systems NSS relief valves BOP systems	NSS systems BOP systems Feedwater	Oversized actuators and misadjusted torque/limit switches are predominant cause of valve structural
orque switches imit switches otors	Yes, on a per-plant basis 1978–1980	Evaluate thermal overload bypassing Signature tracing Improved torque switch procedures	HPCI RCIC	Auxiliary feedwater	Some technical inaccuracies; disagreement with NP-241 regarding oversized valve operators: study suggests that changes in valve operating conditions lead to torque switch inaccuracies: propose MOV "signature" tracing

[α] FW — Feedwater
MS — Main steam
ECCS — Emergency core cooling system
NSS — Nuclear steam supply
RCIC — Reactor core isolation cooling

[b] Defined as natural failures such as those due to end of life, spurious, or unexplained.

It is concluded that detection and monitoring methods currently in use for valves and motor operators are relevant, and their use should be continued; the main issue is effective augmentation.

4.0 Utilization of Research Results in the Regulatory Process

4.1 General

Phase I Aging characterization and I, S, M&M of Components

a. Regulatory Guides

- 1.147, Inservice Inspection Code Case Acceptability-ASME Section XI, Div. 1

- 1.148, Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Plants

b. Generic Issues

- Task II.E.6.1, In-Situ Testing of Valves

- Item B-58, Passive Mechanical Failures

- Item C-11, Assessment of Failure and Reliability of Pumps and Valves

- Issue 54, Valve Operator-Related Events Occurring During 1978, 1979, and 1980

- Issue 105, Interfacing Systems LOCA at BWRs

- ASME Boilers and Pressure Vessel Code Section XI - Rules for Inservice Inspection of Nuclear Power Plant Components

Phase II

ASME Operation and Maintenance Committee Standards

OM-2	Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water System
OM-8	Requirements for Proportional and Periodic Performance Testing of Motor-Operated Valve Assemblies
OM-10	Requirements for Inservice Testing of Valves in Light Water Cooled Nuclear Power Plants
OM-15	Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling System

4.2 Specific

In nuclear power plants, testing requirements for identified MOVs are contained in the plant Technical Specifications (TS), which specify the overall In-Service Inspection (ISI) requirements for safety-related systems and components. ISI and testing of ASME Class I, 2, and 3 components are required to be in accordance with Sect. XI of the **ASME Boiler and Pressure Vessel Code.** The purpose of TS requirements is to ensure operability of components within specified limits required by readiness to perform the desired safety function and not to specifically monitor degradation of performance.

Section XI testing requirements for MOVs involve only two parameters - stroke time and seat leakage. With regard to stroke time, the code specifies that the stroke time from test to test be compared. When the increase from the previous test is sufficiently great (>25% or >50%, depending on stroke time), the test frequency should be increased to once per month until corrective action is taken. Corrective action is required when the stroke time exceeds its specified maximum value.

Leakage rate measurements also are compared with previous measurements, and when the margin between measured and maximum permissible leakage is reduced by 50%, the test frequency should be doubled. When projections indicate that the leak rate will exceed the maximum permissible rate by >10% prior to the next test, corrective action is required.

ASME Code Sec. XI contains surveillance intervals, frequencies, and test requirements for Code Class 1, 2, and 3 systems.

4.7 <u>CHECK VALVES</u> <u>NUREG/CR-4302 VOL. 1</u> <u>DEC. 1985</u> <u>ORNL/FIN NO. B0828</u>

Aging and Service Wear of Check Valves Used in Engineering Safety-Feature Systems of Nuclear Power Plants. Operating Experience and Failure Identification.

1.0 Background

This is the summary of the Phase 1 report on check valves (CVs) to be published under the Nuclear Plant Aging Research Program (NUREG-1144). The program is concerned with the evaluation and identification of methods for detecting, monitoring, and assessing the severity of time-dependent degradation (aging and service wear) of CVs in nuclear plants. These methods will allow degradation trends to be established prior to failure and allow guidance for effective maintenance to be developed.

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

Failure modes are identified for CVs. For each failure mode, failure causes are listed by subcomponent or subassembly, and parameters potentially useful for detecting degradation, which could lead to failure, are tabulated.

2.0 Summary of CV Applications in Nuclear Power Plants

System	Number of CVs	Valve Size (in.)	System	Number of CVs	Valve Size (in.)
BWR (typical)			PWR (typical)		
Low-pressure core spray	10-18	2-28	Auxiliary feedwater	4-23	4-8
			Containment spray	4-14	6-14
High-pressure coolant injection (HPCI)	10-14	4-24	HPCI	12-28	2-1/2-4
			LPCI/RHR	5-14	8-10
Low-pressure coolant injection (LPCI) [includes residual heat removal (RHR) and containment spray]	10-21	4-24	BOP systems	200-400	2-60
BOP systems	200-400	1/2-24			

3.0 Results/Findings

3.1 Frequency of Failures

Two data bases contain failure frequencies: the NPRDS and the IPRDS prepared by the Institute of Nuclear Power Operations (INPO). In their "1981 Annual Report," for CVs up to 4 in., 3 failures/10^6 calendar hours were reported during the time period of 7/1/74 to 12/31/81 for leak and failure to stop. For 4- to 12-in. valves, 4 failures/10^6 calendar hours were reported for the same period. Data for CVs of 12 inc. and larger were insufficient to calculate failure rates.

FAILURE MODE DISTRIBUTION[1]

Failure Mode	Percent
Seat leakage	70
External leakage	16
Failed to close	8
Failed to open	2
Damaged internals	4

3.2 Methods of Detection

In an LER survey, surveillance testing found 32%, while 10 CFR 50 Appendix J leakage testing found 27%. Another 28% of the failures were detected during normal operation. Only 1% of the failures occurred during an operational demand.

Detection[1]	Percent
In-service and surveillance test	67
Incidental observation	4
Routine observation	14
Operational abnormally	11
Maintenance	2
Special inspection	2

(1) Based on 382 events involving NPRDS component valve, component engineering Code C (Check Valves)

3.3 Maintenance Actions

The IPRDS, which extracts repair information from plant maintenance records, is the only data base that contains detailed information on maintenance actions performed on failed CVs. However, because of insufficient entries for CVs, this data base could not be included.

Maintenance activity is sometimes stated briefly in the LERs. Based on these reports, the valves were repaired 54% of the time and replaced 11%. About 25% of the LERs did not indicate any maintenance activity.

Activity[1]	Percent
Repair/replace	93
Modify/substitute	4
Temporary measure	3

3.4 Modifications Resulting from Failures

The operating experience data bases do not contain detailed descriptions of postfailure modifications. Some IE publications have outlined a few CV modifications, which are summaried below:

1. Improved soft-seated valve seals - Hard seat valves were modified to a combination soft and hard seat configuration. Several types of soft rings were tried before a molded (one-piece) seal provided a satisfactory leaktightness.

2. Obturator attachment - The locking device that secures the obturator to its hanger wore sufficiently to allow the obturator to fall free of the hanger.

IDENTIFIED FAILURE CAUSE[1].

Failure Cause	Percent
Aging/cyclic fatigue	7
Normal/abnormal wear	50
Binding/mechanical damage	6
Lubrication problem	2
Previous repair/installation	2
Corrosion	4
Weld related	2
Dirty	14
Particulate contamination	1
Out of adjustment	3
Foreign/incorrect material	3
Unknown	1
Connection defect/loose part	3
Material defect	2

3.5

Summary of Valve Part Failure Assessments

Part	Materials	Significant Stressors &	Measurable Parameters
Body, cap (bonnet)	Stainless steel	Mechanical: obturator guide wear, galling body wear, rupture Chemical: corrosion, erosion	Dimensions, appear ance, roughness cracking
Fasteners	Stainless steel	Mechanical: loosening, breakage Chemical: corrosion	Torque, appearance
Seat	Stainless steel hardened alloy Resilient material	Mechanical: wear Chemical: erosion, corrosion	Leakage rate, dimen sions, appearance cracking
Obturator	Stainless steel with hardened alloy seating	Mechanical: wear Chemical: erosion, corrosion	Leakage rate, dimen-sions, appearance cracking
Obturator hanger	Stainless steel	Mechanical: wear, fracture chemical; erosion, corrosion	Dimensions, appearance roughness
Hanger pin	Stainless steel	Mechanical: wear, fracture Chemical: erosion, corrosion	Dimensions, appear-ance, roughness
Hanger pin bearing	Hardened alloy	Mechanical: wear fracture Chemical: erosion, corrosion	Dimensions, appear-ance
Seals, gaskets	Asbestos type. Stainless steel Resilient material	Mechanical: distortion, compression Thermal: hardening, embrittlement (non-metals) Chemical: corrosion	External leakage, appearance, noise torque or force applied for obtur-ator movement, packing gland position

4.0 UTILIZATION OF RESEARCH RESULTS IN THE REGULATORY PROCESS

4.1 General

4.1.1 Phase I Utilization

Regulatory Guides

1.147 Inservice Inspection Code Case Acceptability-ASME Section XI, Div. 1

1.148 Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants

Generic Issues

Task II.E.6.1 In Situ Testing of Valves

Item B-58 Passive Mechanical Failures[1]

Item C-11 Assessment of Failure and Reliability of Pumps and Valves[1]

Applicable Standard Review Plan (SRP) Chapters:

3. Design of structures, components, equipment and systems
5. Reactor coolant system and connected systems
6. Engineered safety features
9. Auxiliary systems
10. Steam and power conversion systems

4.1.2 Phase II Utilization

ASME Boiler and Pressure Vessel Code

Section XI - Rules for In-service Inspection of Nuclear Power Plant Components

- ASME Operation and Maintenance Committee Standards(2)

  OM-10 Requirements for Inservice Testing of Valves in Light Water Cooled Nuclear Power Plants

  OM-2 Requirements for Performance Testing of Nuclear Plant Closed Cooling Water Systems

  OM-15 Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems

(1) Closed out as of July 9, 1985, but will be addressed in NPAR program.

(2) Draft Maintenance Program Plan - RES resources (May 24, 1984).

4.2 Specific NRC Efforts Related to Check Valves

1. San Onfore Water Hammer Event (IE Lead): As a result of this event, the staff has been directed to implement and coordinate actions necessary to assure the reliability of safety related-check valves. Including:

   a. The adequacy of check valve design for application in the feedwater and other systems.

   b. The adequacy of Inservice Testing (IST) Programs and procedures to detect degraded and failed valves.

   c. The adequacy of check valve maintenance programs to assure operability.

2. Generic Issue TMI II.E.6.1, In Situ Testing of Valves (NRR Lead): Study and make recommendations to change current in situ testing methods in order to provide assurance that valves (including check valves) will perform their safety-related functions under design basis as well as operating conditions.

3. Generic Issue 105, Interfacing System LOCA in LWRs (NRR Lead): Perform a risk analysis and a cost benefit analysis of leak testing reactor coolant boundary pressure isolation valves, including check valves. The results of this effort will be used to, among other things, support resolution of Generic Issue TMI II.E.6.1.

4. Generic Issue 93, Steam Binding of Auxiliary Feedwater Pumps (NRR Lead): Determine an appropriate means to prevent steam leakage past check valves which could result in steam binding of the auxiliary feedwater pumps. This issue is being treated separately from the more general statement of the problem (Generic Issue TMI II.E.6.1.) because of its direct effect on plant safety.

4.8 AUXILIARY FEEDWATER PUMPS NUREG/CR-4597 MARCH 1986 ORNL/FIN NO. B0828

Aging and Service Wear of Auxiliary Feedwater Pumps for PWR Nuclear Power Plants

1.0 Background

This report addresses time-related degradation of pressurized-water reactor (PWR) power plant auxiliary feedwater pumps (AUXFPs). Since failures of these components can reduce the amount of feedwater available for removing heat when the usual feedwater supply is unavailable, such failures can result in altered safety margins for PWR systems.

Auxiliary feedwater pumps (AUXFPs) are basically small boiler feed pumps as used in small capacity, mostly older fossil fuel electric generating plants. Thus they retain most of the design features, plus the potential operating and reliability problems inherent in the feedwater pumps.

Plants have traditionally used AUXFPs per PWR. In most plants, one of these is steam turbine driven and the other is electric motor driven. The steam turbine driven unit has the advantage of variable speed and the maximum design operating speeds are generally in the 4000 and 5000 rpm range. The motor driven unit is usually designed for two-pole induction motor speed, and powered from either one of two standby diesel driven generators. Newer plants may also have a third AUXFP per reactor which is called the "back-up", and is electric motor driven either from main line power or switchable to the diesel driven generator bus.

The overall approach which now is emerging in the newest plants is to have the "back-up" AUXFP be a non-safety related, non-nuclear class (nonclass) pump (thus less expensive) and to use it for all normal plant start-up, shut-down and non-emergency service.

2.0 Summary of Applications

AUXFPs are used to supply feedwater to the steam generators under plant startup, shutdown, and emergency conditions. These pumps have a best-efficiency-point (BEP) flow at any operating speed and if used in a continuous operating mode, normal delivered flow is between approximately 50% and 120% of the BEP flow. However, the AUXFP application is by definition a transient operational mode, i.e., startup, shutdown, and emergency. Thus, one could say that there are basically two operating regimes: (a) standby and (b) normal. Normal includes any flow from shut off and bypass flow to full run-out flow. Alternately, if the term "normal operation" pertains to that regime in which the pump resides most of the time, then normal operation would be the standby condition and any condition under which the AUXFP is pumping would be considered off-normal.

Nearly all AUXFPs presently in-service are installed with little or none of the specific parameter monitoring devices which are commonly integrated into continuously running main power cycle equipment. Most auxiliary feedwater systems are not presently configured to facilitate regular periodic testing of AUXFPs over the complete range of flow rates that would simulate the various emergency scenarios for which AUXFPs are installed. The present surveillance practice as required by plant technical specifications (TS), which consists primarily of starting each AUXFP once every one to three months for a short duration test at bypass flow, is inadequate to verify the pump's operational readiness. The Current TS surveillance requirements do not require the collection of adequate data for performance trending to establish a correlation between AUXFP operating parameters (e.g., vibration, bearing temperatures, etc.) and wear and aging. Consequently, with present typical installations, aging and service wear monitoring would have to be based primarily upon a detailed and thorough inspection of the pump in the disassembled state. Unless specific operating problems arose with a particular AUXFP, such a detailed inspection would be practical only during a scheduled plant shutdown, such as at refueling time.

3.0 Results/Findings

Three failure modes apply to auxiliary feedwater pumps: (a) failure to operate, (b) failure to operate as required, and (c) external leakage. Failure causes corresponding to these failure modes are identified in general terms and subsequently described in most specific terms in a section on failure cause analysis. The single most important factor relevant to auxiliary feedwater pump potential failures is the presence of large hydraulic forces within such pumps, particularly at low flow rates substantially different than the best-efficiency flow. In most plants the AUXFP bypass line is a single line sized to pass 5-15% of best efficiency flow. Thus, this type of testing could be the main contributor to wear and aging of various AUXFP components. In some newer plants (e.g., Palo Verde) an additional full-flow bypass line is provided to allow testing of the AUXFPs over the full operating flow range, even when the plant is operating in a normal generating mode.

SUMMARY OF AUXFP-TYPE FAILURES REPORTED IN LERS (1973 THROUGH 1983)

	%
Failed component	
Bearings	48
Packing and seal	30
Casing	4
Internal components	4
Impeller	2
Capacity	2
Shaft	2
Other	8
Methods of detection	
Testing	42
Operation	29
Maintenance	6
Not stated	23
Maintenance action	
Replacement	67
Repair	25
Modification	6
No repair required	2
Identified cause	
Lack of lubrication or cooling	23
Maintenance error	17
Wear/end of life	15
Design error	6
Crud	4
Operator error	2
Other	4
Not stated	19
Unknown	10

AUXFP-TYPE FAILURES REPORTED IN NPRDS DATA BASE (1974 THORUGH 1985)

	%
Failed component	
Packing/gasket	50
Bearings	38
Internal components	6
Shaft	6
Methods of detection	
Incidental observation	33
Surveillance testing	25
Routine observation	12
Audiovisual alarm	12
Operational abnormality	12
Special inspection	6
Maintenance action	
Repair/replace	94
Modify	6
Failure cause	
Wear	58
Lubrication	12
Binding	12
Aging	12
Abnormal stress	6

SUMMARY OF IMPORTANT AUXFP PART FAILURE ASSESSMENT

Part	Materials	Significant stressores; and failure causes	Measurable parameters
Shaft and fasteners	400-series S.S.	(a) mechanical; breakage (b) hydraulic; breakage and wear (c) tribological; wear and seizing	(a), (b), and (c) vibration, bearing temperature, appearance, transmitted toque
Impellers	CrNi alloy steels, 17-4Ph	(a) mechanical; breakage (b) hydraulic; breakage and wear (c) tribological; wear-surface wear	(a) and (b) rotor unbalance vibration, appearance, delivered flow (c) rotor vibration, appearance
Thrust runners	400-series S.S.	(a) mechanical, breakage, seizing (b) hydraulic; breakage, seizing (c) tribological; rubbing, lubricant dirt and breakdown	(a), (b) and (c) transmitted torque, rotational speed, rotor axial position
Stationary vanes (diffuser or volute)	400-series S.S.	(a) hydraulic; breakage	(a) delivered flow, appearance, vibration
Wear rings	400-series S.S.	(a) mechanical; seizing (b) hydraulic; seizing (c) tribological; wear-surface wear	(a), (b) and (c) vibration, transmitted torque, delivered flow, appearance, clearance
Thrust balancers	400-series S.S.	(a) mechanical; breakage, seizing (b) hydraulic; breakage, seizing (c) tribological; wear	(a), (b) and (c) rotor axial position, transmitted torque, rotational speed, appearance
Thrust bearings	Rolling contact elements (Specialty steels)	(a) mechanical; breakage (b) hydraulic; breakage (c) tribological; wear	(a), (b) and (c) rotor axial position, transmitted torque, rotational speed, appearance, clearance
Radial bearings	Bearing white metal (typically tin-base babbitt)	(a) mechanical; breakage, seizing (b) hydraulic; breakage, seizing (c) tribological; seizing, wear	(a), (b) and (c) rotor vibraton, bearing temperature, transmitted torque, appearance
Shaft seals	Stuffing-box or mechanical type	(a) mechanical; breakage, wear (b) hydraulic; breakage, wear (c) tribological; wear	(a), (b) and (c) seal leakage rate, rotor vibration, local shaft, temperature
Coupling	Gear type (usually)	(a) mechanical; breakage, wear (b) hydraulic; breakage, wear (c) tribological; breakage wear	(a), (b) and (c) rotor vibration, transmitted torque, appearance

4.0 Utilization of Research Results in the Regulatory Process

The results of this and subsequently planned studies will be useful to the regulatory process and can be factored into this process through the following vehicles:

a. Regulatory Guidelines: 1.147 Inservice Inspection Code Case Acceptability - ASME Section Xl, Division 1

b. Generic Issues: Item C-11 Assessment of Failure and Reliability of Pumps and Valves

c. Technical Specifications: The present technical specification and surveillance testing requirements should be reconsidered to ensure the operational readiness for all AUXFPs after each time the pump surveillance requirements are satisfied.

d. ASME Boiler and Pressure Vessel Code Section Xl - Rules for Inservice Inspection of Nuclear Power Plant Components

e. ASME Operation and Maintenance Committee Standards

   OM-2 Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water Systems

   OM-6 Requirements for Performance Testing of Pumps in Light-Water Cooled Nuclear Power Plants

   OM-14 Requirements for Vibration Monitoring of Rotating Equipment

   OM-15 Requirements for Performance Testing of Nuclear Power Plant Emergency Core Cooling Systems

4.9 Snubbers NUREG/CR-4279 February 1986 PNL/FIN No. B2865

Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety- Related Piping and Components of Nuclear Power Plants.

1.0 Background

This report presents and overview of hydraulic and mechanical snubbers used on nuclear piping systems and components, based on information from the literature and other sources.

The initial effort was to review the applicability of snubbers, evaluate the benefits of reductions in snubber population, and provide input to the preparation of appropriate regulatory guides. Therefore, this report does not cover all of the Phase I elements of aging and wear assessment for snubbers. It is expected that the Phase II effort will be directed at aging and defect characterization and to develop recommendations for inspection, surveillance and maintenance.

The functions and functional requirements of snubbers are discussed. The real versus perceived need for snubbers is reviewed, based primarily on studies conducted by a Pressure Vessel Research Committee. Tests conducted to qualify snubbers, to accept them on a case-by case basis, and to establish their fitness for continued operation are reviewed. A conclusion from this report is that many snubbers installed in nuclear power plants may be unnecessary and could be removed. Work outside the scope of this report has confirmed that the removal of many snubbers can be justified. An approved approach to evaluate snubber removal has been incorporated into ASME III and approved by the U.S. Nuclear Regulatory Commission (NRC) on a case-by-case basis.

As more snubbers were used, several operating problems arose; for example, degradation and leaking of seals on hydraulic snubbers and functional failures of both hydraulic and mechanical snubbers. These problems led to increased qualification and testing requirements. Thus, the original cost of a snubber represents only a small fraction of the overall cost of qualification, installation, maintenance, and testing. In addition, maintenance and testing result in substantial radiation exposure in older plants.

Another problem, that was not recognized initially, was the limitation on in-service inspection (ISI) resulting from the large numbers of snubbers and supports that prevent access to many welds in piping systems. A further problem was the concern that stiff piping systems may be inherently more susceptible to overloading and possible failure than flexible systems.

As a result of these concerns, Technical and Steering Committees on Piping were organized under the Pressure Vessel Research Committee (PVRC) with active industry and NRC participation. In the past two years, these groups have developed a more relaxed interim position on seismic damping, a modified and less conservative position on spectra broadening, and a document on industry practice related to design approaches leading to fewer snubbers. These positions have been accepted by the NRC on a case-by-case basis, and portions have been incorporated into Appendix N of ASME III (the reactor construction code). A task group on seismic design under an NRC Piping Review Committee has recommended that the case-by-case status be converted to generic positions. The NRC Executive Director for Operations has issued a directive to develop such generic positions as cited in NUREG--1061.

2.0 Summary of Applications

Snubbers of the type discussed in this report have two functions: 1) they should move freely at low accelerations and 2) they should lock up at higher accelerations. The valid use of snubbers is in locations of limited clearance and possible high thermal expansion.

Several organizations have reported that a typical 1100-MWe capacity boiling-water reactor (BWR) can have 9,000 to 10,000 supports on seismic Category I piping (as many as 800 spring hangers and 1500 snubbers). An 1100-MWe pressurized-water reactor (PWR) could have 7,000 to 10,000 supports (200 spring hangers and 950 snubbers).

Engineering, fabrication, construction, and hardware costs will be lower with fewer supports; however, when the cost of items such as analysis, computer time, and reconciliation are considered, the cost difference in the two approaches may not be significant. The difference is heavily influenced by two factors: 1) the installed cost per support and 2) the total life (40 years) cost per support. The second factor refers to inspection and maintenance costs associated with snubbers--typically $5,000 to $10,000 per snubber on small-bore piping.

The table below provides a basis for developing a program on older hydraulic and mechanical snubbers. It contains the parameters cited in the Draft Regulatory Guide, ASME OM4, and ASME XI. The table is a qualitative presentation of aging factors such as wear, corrosion, and contamination. A selection of both hydraulic and mechanical snubbers could have a series of tests covering spring rates, release rates, dead band, drag, etc., that would note deviations from normal. The snubbers could then be disassembled to ascertain the causes for the deviations. Any quantification will depend on the snubbers that are selected. ASME OM4 is suggested as a benchmark for such testing.

Common Causes for Degradation in Snubber Operating Characteristics

	Measured Parameter[a]				
Cause	Stiffness (spring rate)	Activation Force	Release Rate	Dead Band	Friction (drag/breakaway)
Wear	-	-	-	M	M
Corrosion	-	-	-	-	H,M
Viscosity	H	-	-	-	-
Temperature	H	H	H	-	-
Entrapped air	H	H	H	H	-
Contamination	-	-	-	-	H,M

(a) M - mechanical
H - hydraulic

The parameters in the above table, such as corrosion and viscosity are strongly influenced by temperature and irradiation. Thus, times to failure may differ markedly from one portion of the plant to another. This is of concern because an inherent assumption in testing a small sample is the homogeneity of the population. Snubbers taken from one region of a reactor (for example, a cooler region) could display a markedly different failure history than ones removed from another region (for example, a BWR dry well). This issue could warrant consideration in selecting the sample to be tested.

3.0 Results/Findings

A review of available data reveals the large number of failure mechanisms, many of which are related to aging/degradation. In the case of hydraulic snubbers, several of the failure mechanisms result in a loss of fluid and a consequent loss of function. This occurrence will be a fail-safe mode in that there should be no lockup and overstressing of the pipe. Based on early arguments on flexible versus stiff piping, the lack of operation of hydraulic snubbers during a seismic event should have limited effects. Whether the same can be said concerning more severe dynamic loads depends on analysis or testing. It can be concluded that the loss of function of some hydraulic snubbers should have a limited effect on the probability of failure of piping during a seismic event.

With mechanical snubbers, the situation is not as clear. One obvious aging/degradation mechanism--corrosion of internals over a period of time--has been known to lead to loss of function and lockup. In this instance, the situation is not fail-safe; and, mechanical snubbers located in regions of high thermal expansion of piping during heat-up and cool-down may severely stress snubbers or piping, or both. If sufficiently high stresses occur, the snubber may be torn from the wall, or the pipe may be damaged. In regions of lower thermal expansion, the locked up snubber may exist for a substantial time. If a dynamic load occurs, the snubbers will behave as rigid members, and they or the pipe may be damaged. Mechanical snubbers have locked up without being detected for several days or weeks.

Several utilities have extensive snubber replacement programs, replacing one form of hydraulic for another or replacing mechanical for hydraulic. Thus, the age factor cannot be quantified using an assumption that the same snubbers were originally in the system. The same can be said for seals and seal materials in hydraulic snubbers where several changes have occurred in original materials as well as replacement materials.

Obvious degradation mechanisms include seal aging in hydraulic snubbers and vibrations in mechanical snubbers. Several load conditions, such as water and steam hammer or valve closure, can render snubbers inoperable. Under these conditions, loss of function can occur whether the snubbers are new or old.

If the Draft Regulatory Guide were converted to an active Regulatory Guide, the aspect of an inspection program should be resolved. While it could be argued that further modifications could result in improvements, it appears that the draft Regulatory guide covers the salient aspects pertinent to an inspection program.

Environmental qualification has been handled on a case-by-case basis, and there is no assurance that a generic program exists. Furthermore, there are several aspects for such a program. Under normal operating conditions, an ocean site will require a different environmental qualification program than an inland site. In addition, a high-temperature dry well of a RWR will be markedly different from reactor regions near ambient temperature. In essence, the areas of environmental qualification are not well defined and specific criteria are needed.

With regard to ISI and testing, there should be a reassessment of existing requirements to establish what modifications may be necessary. The existing visual examination in ASME XI may not be adequate, and the bench testing program probably requires review and modification. Currently, only smaller snubbers are covered by ASME XI; expansion will be required due to the failures of larger snubbers, particularly hydraulic ones. It may be necessary to test in-situ in recognition of the massive size of these larger snubbers. In this case, a simpler test may be necessary to determine the functionality of these snubbers. ASME OM4 represents a positive step toward improving testing; however, it may not be totally adequate.

Example of Failure Modes and Causes for Power Station Snubbers[(a)]

a) Hydraulic Snubber Failure Modes[(a)]

Mode	No. of Failed Snubbers	% of Snubbers Tested	% of Snubbers Failed[(b)]
Low activation in tension	13	8	2
Low activation in compression	3	2	6
Low bleed in tension	11	7	23
Low bleed in compression	10	6	21
High activation in tension	17	11	35
High activation in compression	10	6	21
High bleed in tension	6	4	13
High bleed in compression	3	2	6

b) Hydraulic Snubber Failure Causes

Mode	No. of Failed Snubbers	% of Snubbers Tested	% of Snubbers Failed
No observable defect	6	4	26
Degraded EP seals	3	2	13
Polyurethane piston seals	7	4	30
Poppet upside down	3	2	13
Debris in fluid	1	1	4
Poppet stuck	1	1	4
Activation adjustment screw broken	1	1	4
Piston/cylinder scoring	2	1	9

Example of Failure Modes and Causes for Power Station Snubbers (Cont.)

c) Mechanical Snubber Failure Modes [c]

Mode	No. of Failed Snubbers	% of Snubbers Tested	% of Snubbers Failed
High drag	14	21	66
Locked up	5	8	24
High acceleration	1	2	5
Locked up in compression	1	2	5

(a) A total of 160 hydraulic snubbers were tested; 48 snubbers (or 30%) failed.
(b) Failures total more than 100%, because some snubbers exhibited more than one failure mode or cause.
(c) A total of 66 mechanical snubbers were tested; 21 snubbers (or 32%) failed.

Number and Proportion of Snubber Failures by Manufacturer for Data Set A[1]

	Failure Cause				Failure Mode			
Manufacturer	Unknown	Design	Installation	Operation	Unknown	Frozen	Out of Tolerance	Would Not Lockup
Unknown	22 (1%)	104 (6%)	168 (10%)	1450 (83%)	45 (3%)	150 (9%)	648 (37%)	901 (52%)
Bergen-Paterson	—	425 (63%)	46 (7%)	207 (31%)	1 (1%)	1 (1%)	57 (8%)	619 (91%)
Blaw Knox	—	1 (25%)	3 (75%)	—	—	—	—	4 (100%)
ITT-Grinnell Corp.	—	256 (51%)	33 (7%)	217 (43%)	—	—	215 (42%)	291 (58%)
Pacific Scientific	3 (9%)	9 (27%)	6 (18%)	15 (45%)	3 (9%)	13 (39%)	3 (9%)	14 (42%)
Anchor-Holth	—	—	—	3	—	—	—	3
International Nuclear Safeguards Corp.	—	91	—	—	—	91	—	—
ITT Barton	—	—	—	1	—	—	—	1
McDowell Wellman	—	—	1	—	—	—	—	1
Power Piping Co.	—	—	—	1	—	—	—	1

(1) Developed from detailed computer search of LER abstracts (1973-1983).

Cost of Reducing the Number of Snubbers in Existing Plants

For backfit applications, the cost of redesigning and implementing a more flexible system must be estimated and then subtracted from the predicted cost savings. The cost of backfitting plants to reduce the number of snubbers has been estimated to be from $820,000 to $890,000 per plant. This cost does not include the increased employee exposure that would result from repiping and removing unnecessary snubbers. However, this exposure rate might be expected to be on the same order as that incurred from inspecting and maintaining a snubber (estimated to be 2.5 man-rem/snubber). The $1,000/man-rem rule of thumb was used to quantify this cost. The cost savings would be $2,500/snubber for each snubber eliminated.

In a plant containing 1,000 snubbers, a 30% reduction in the number of snubbers could be expected to result in a total cost reduction of form $526,000 to $3,500,000 (See table below). All of the previous caveats regarding the impact of hidden costs such as increased plant downtime or damage to piping still apply. The exposure/implementation costs were obtained by multiplying the unit cost ($2,500) by the reduction in snubbers (300).

Estimated Cost Savings Resulting from Reducing the Number of Snubbers in an Existing Plant [(a)]

Cost Category	Cost, $	Timing	Total Costs, $ 2% Discount Rate	5% Discount Rate	10% Discount Rate
Capital	0	--	0	0	0
Implementation	855,000	0	855,000	855,000	855,000
Inspection/maintenance	(115,100)	Annual	(2,577,832)	(769,369)	(1,085,038)
Exposure/implementation	750,000	0	750,000	750,000	750,000
Exposure savings	(111,000)	Annual	(2,486,007)	(1,706,342)	(1,046,388)
Total			(3,458,839)	(1,870,711)	(526,426)

(a) The following assumptions were used: 30-year remaining plant life; 2% real discount rate; 300 unit reduction in snubbers (and a representative population).

4.0 Utilization of Research Results in the Regulatory Process

The results of this and subsequent planned studies will be useful in the regulatory process and can be factored into this process through the following vehicles:

a. Regulatory Guides: (1) NRC Standard Review Plan, Section 3.9.3 cites the criteria applied for snubbers. Subsection II.3b,1-7 deals specifically with snubbers. Section 3.9.2 deals with piping preoperational vibration and plant start-up test programs. (2) Draft RG SC 708-4, Rev.1, February 1981 deals with qualification testing, acceptance testing and functional specification.

b. Generic Issues: Generic Letter 84-13, May 3, 1984, Technical Specifications for Snubbers.

c. ASME: (1) PVP-45, Criteria for Nuclear Safety Related Piping and Component Support Snubbers (1980) discussed acceptance testing. (2) Preservice and in-service testing are addressed in ASME XI and ANSI/ASME OM4. (3) ASMEIII, Section NF-3000 covers design rules for supports.