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Survey of Operating Experience from LERs to Identify Aging Trends

Status Report

G. A. Murphy
R. B. Gallaher
M. L. Casada
H. C. Hoy

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SURVEY OF OPERATING EXPERIENCE FROM LERs
TO IDENTIFY AGING TRENDS

INTERIM PROGRESS REPORT

G. A. Murphy M. L. Casada*
R. B. Gallaher H. C. Hoy

ABSTRACT

At the Nuclear Regulatory Commission- (NRC-) sponsored workshop on nuclear power plant aging held in August 1982, the recommendation was made to assess the information available in operating experience reports pertinent to identifying age-related failures. This report describes the preliminary results of such an assessment by the Nuclear Operations Analysis Center (NOAC) staff at Oak Ridge National Laboratory. The NOAC utilized the computerized files of Licensee Event Reports (LERs) and their predecessors to examine age-related degradation of safety-related equipment. This effort was sponsored by the NRC Office of Nuclear Regulatory Research as part of the Nuclear Plant Aging Research Program.

Abstracts of operating experience reports from commercial power plants reported from 1969 to 1982 were surveyed in this initial effort. Utilizing keywords currently available in the NOAC computer files, the NOAC staff initially selected over 7000 events for review. A total of 4461 event abstracts were reviewed in detail, yielding 3098 events considered age-related. Data collected for each event included the system, component, subpart, the age-related failure mechanism, the severity, and the method of detection of the failure.

Wear, corrosion, crud, and fatigue were the identified failure cause mechanisms in over one-third of the 3098 events. About two-thirds of the failure severities were judged as degraded and one-third judged as catastrophic failures. No events were found to be incipient failures because an LER is prepared only upon degraded or catastrophic failure conditions, which place plant operation outside the Technical Specifications.

Pump and valve problems made up almost 30% of the failed components, which reflects industry experience and emphasizes the need for the recently issued American Society of Mechanical Engineers (ASME) surveillance testing codes to detect degradation of these key components. Almost two-thirds of the reported failures were detected by routine surveillance testing, indicating that such practices are effective techniques for monitoring and detection of age degradation of discrete components and systems. A substantial number of the initially

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selected events (2795) resulted from drift — whenever a safety-related device set point or calibration was found outside acceptance criteria contained in the plant's Technical Specifications. These events are addressed separately in the report.

The study found that information desired for evaluation of aging effects (e.g., equipment age, service life, and environment) was not available from LERs. This is consistent with the purpose of the LER system as a regulatory instrument, rather than an engineering data collection system. Only limited information about the root causes of equipment failures is provided by LERs. By design, LERs are tools to report failure effects on systems and safety functions and as such do not go into detail about the specific failure mechanism, contributing causes, and required repair actions. This makes identification of trends concerning such failures impossible. Also, the NRC's new LER rule, scheduled to go into effect in January 1984, will cause a reduction in the amount of aging-related data available. The new rule will no longer require reporting of certain single failures of safety-related equipment. Because the majority of age-related failures are identified as single failures, they will no longer appear as LERs.

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1. INTRODUCTION

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. This study is a part of the overall Nuclear Plant Aging Research (NPAR) Program being conducted by the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission (NRC).

This 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. However, it should be noted that the LERs are a regulatory instrument; therefore, the LER system is not an engineering data collection system and is not suitable for any definitive statistical engineering data collection or analysis. This study utilizes the data only to determine failed components, the age-related failure mechanisms responsible, the severity of the failure, and the failure detection methods that contributed to a reportable occurrence. The NRC *Regulatory Guide 1.16* contains the specific requirements for reporting operating information in LERs.

Although abstracts of documents dating back to 1969 were reviewed, the majority of the data were 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.

Chapter 2 of this report describes the search methodology utilized in this project. Chapter 3 discusses each of the age-related failure mechanisms and the failures caused by them. Overall results of the data collection and specific conclusions are discussed in Chap. 4, while general conclusions and recommendations are presented in Chaps. 5 and 6, respectively.

2. SEARCH METHODOLOGY

The NOAC maintains a keyworded file of LER abstracts, accessible through the Department of Energy (DOE) RECON system (Appendix A). Keywords are assigned to identify the specific plant, reactor type [boiling-water reactor (BWR)/pressurized-water reactor (PWR)/high-temperature gas-cooled reactor (HTGR)], affected systems and components, and proximate cause(s) of an event, if the information is available in the LER.

For this study, project personnel reviewed the NOAC thesaurus of keywords assigned to LERs and selected 16 keywords most likely to yield age-related failure events. This process was necessary to reduce the number of LERs to be examined in the first phase of the study. Table 2.1 gives the number of failure events obtained for each selected keyword. These keywords extracted a major portion of the LER events that involve age-related failures. Further LER-based aging surveys will require additional effort to review events obtained from other keywords for age-related failure information.

The order of the keywords shown in Table 2.1 affects the number of event descriptions identified for each keyword. The numbers given are only the additional abstracts identified for each succeeding keyword as the computer keyword search process eliminates events previously selected under previously used keywords. For example, an event yielded by the keyword *vibration* would not appear again under subsequent keywords. For this study, the keyword *vibration* was reviewed as one of the 616 *vibration* accessions and not repeated as one of the *fatigue* accessions.

Using the process described in the preceding paragraph, the RECON search yielded 7256 abstracts for review. These 7256 abstracts were then reviewed in detail to identify actual age-related failure events.

Of the 7256 abstracts, aging study personnel reviewed 4461 in detail. They identified 3098 unique events that were actually considered to be age-related events involving equipment failure. This represents between 8 and 14% of all accessions for reportable events in each of the years 1966 to 1982. Over 50% of the events examined in detail occurred in the years 1979 to 1982 (Table 2.2). A discussion of the review of drift events is included in Sect. 3.10.

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. For a description of the data collected and the definition of each item, see Appendix A.

Table 2.1. NOAC keywords utilized in aging study LER searches[a]

Keyword	Number of abstracts identified[b]
Aged; effect, age	326
Corrosion	581
Stress corrosion	151
Vibration	616
Wear	533
Crud	1128
Erosion	115
Fatigue	28
Failure, fatigue	113
Oxidation	8
Friction	35
Hardening	7
Crack	411
Flow blockage	409
Total number of abstracts reviewed in detail	4461
Accessions keyworded drift (excluding those keywords listed above)	2795

[a]These keywords do not necessarily correspond to each failure mechanism used in the study (Table 3.1). For example, an event extracted using the search keyword *wear* may be coded as *erosion* if such a distinction could be made from the LER abstract. Additionally, some keywords (e.g., crack and crud) are *symptoms* of an age effect, while others are more accurately an age effect.

[b]Only includes unique accessions; those identified by a keyword higher in the list were excluded in subsequent searches.

Table 2.2. 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. LER AGE-RELATED FAILURE MECHANISM ANALYSIS

As discussed in Chap. 2, the study personnel prepared coded inputs for the 3098 events selected as age-related failures for entry into a data file established in the ORNL computer for the aging study project. Table 3.1 shows the distribution of failure mechanisms for these events. This section discusses each of the failure mechanisms and some of the data collected in this study.

Table 3.1. Failure mechanisms for age-related events

Failure mechanism	Number of events
Wear	522
Corrosion	414
Contamination, internal	382
Contamination, external	331
Fatigue	324
Crack[a]	259
End of life[a]	226
Contamination, contact	205
Vibration	165
Stress corrosion	110
Erosion	102
Other miscellaneous mechanisms	58

[a]While not actual failure mechanisms, these classifications are discussed in this section also.

3.1 Wear

This project assigned wear as a failure mechanism when information was not available to be more specific. For example, if a component wore out because it was subjected to vibration, the event was assigned as a vibration failure. However, 77 plants reported 522 failures caused by *wear*, which were not reassignable to other mechanisms due to lack of information about the event. The plants each reported from 1 to 15 such failures in the time frame examined. These events occurred in almost all plant systems, but a larger fraction of wear events pertained to monitoring systems (airborne radioactivity and radwaste monitoring) compared with other mechanisms. This statistic is due to the nature of those systems, because they operate continuously, regardless of reactor status, thereby incurring more service hours than other systems. Also, a large number (35) of containment system events were reported due to *wear*. This number does not include the 64 containment isolation system events that are primarily containment isolation valve-seat wear. The containment system wear events are reports of excess leakage of containment doors due to mechanical wear of the seals and latching mechanisms.

The components and parts for which wear was most often reported reflect the type of events discussed previously (monitors/sample pumps, isolation valves/valve seats, and penetrations/seals). Also, other types of pumps make up a class of active components for which *wear* was reported (124 events). The parts involved were impellers, bearings, seals, belts, and packing.

The detection of wear failures was primarily by routine testing or inspection (59%) and operational abnormalities (35%). The definitions for each method of detection are provided in Appendix A, Table A.3.

3.2 Corrosion

Of the 3098 events described, 414 (13%) identified *corrosion* as the failure mechanism. This figure does not include the 110 events attributed to stress corrosion, which are discussed in Sect. 3.8. Seventy-three of the 84 plants from which events are included in the study reported at least one corrosion failure. One plant reported 16 corrosion-related failures and 11 plants reported 10 or more.

Corrosion events were reported for almost every system for which LERs are filed. The largest numbers of reports pertained to the service water system (49 events), main steam supply system (43), safety injection (39), liquid radwaste system (28), emergency power system (23 events, almost all relating to diesel generators), and containment isolation system (21).

The components and parts most affected by corrosion were the items that make up the fluid piping systems (pipes, valves, and welds). Also, steam generators and other heat exchangers represented a large number of reports due to tubing corrosion. Valve operators and electrical connections are two other plant items for which corrosion failures were common.

The majority (302) of the 414 corrosion events were reported as being discovered during routine testing or inspection. In addition to scheduled nondestructive testing, this includes routine surveillance, such as plant walk-throughs and periodic area inspections. Other primary means of detection were operational abnormality (48) and special tests or inspections in response to alarms (38). All three of these means of detection can identify corrosion that announced itself via system leakage.

Several corrosion-related problems currently being addressed in operating power plants were reflected in the events examined. These include corrosion induced in cooling-water systems that handle raw water and severe corrosion in chemical waste treatment and boric acid environments.

Event descriptions of steam generator tube failures generally identified the cause as *corrosion* or *stress corrosion*; consequently, some of these failures were reported under each category. Not all steam generator tube degradation events are documented via an LER. Tube-thinning (incipient failure) reports are contained in the periodic surveillance tests or in-service inspection (ISI) records of a particular plant, but are not generally reported; whereas a tube crack or rupture (catastrophic part failure) placing the steam generator (component) in a degraded failure condition *is* occasion for a reportable occurrence.

3.3 Contamination

This project examined three failure mechanisms that were originally all classified as failures due to *contamination* or "crud" (not to be confused with "radioactive contamination"). Although crud is not a very specific term, it is commonly reported in LERs as a failure cause. Three types of contamination failures are defined in this study.

1. External contamination — This represents externally generated material and may cause failures due to external effects or may be ingested by equipment and cause internal problems.
2. Internal contamination — This describes contaminants to a system or component that are internally generated and may include corrosion products, wear debris, and similar effects.
3. Contact corrosion/contamination — Electrical contacts have unique problems with contamination and were examined separately. In addition to being susceptible to external contamination, they commonly generate oxidation layers that can cause failure. Generally, LERs only report "failure due to dirty contacts," so distinctions between the two types of contact contamination were not made.

3.3.1 External contamination

Externally generated contamination resulted in 331 reported events at 73 units. Nine units reported 10 or more events, with Salem 2 reporting 19 events, the most for a single unit. Salem's contamination problems were all reported from July 1981 throughout 1981 and 1982. They were primarily events involving inadequate service water flow to containment fan cooler units (CFCU) due to blockage by silt and marine life.

The external contamination events occurred in 39 systems, with 7 systems accounting for 52% of the events. These systems were

1. containment heat removal (12%),
2. fire protection (9%),
3. service water (8%),
4. containment (7%),
5. containment isolation (7%),
6. emergency power (4%), and
7. process and effluent radiological monitoring (4%).

The service water system was also involved in most of the containment heat removal system events and many of the emergency power events.

Various types of valves were involved in a large number of the external contamination events. The valve parts involved were primarily valve seats, housing, operators, and shafts. Pumps were the second most common component affected by external contamination. The subcomponents involved included strainers, housings, impellers, and associated piping. Monitors were also involved in a large number of these failures due to contamination effects on filters, blowers, and sensing elements. This was expected because monitors cannot be totally protected from environmental effects and still accomplish their monitoring function.

Heat exchanger contamination events were dominated by coil blockage due to silt and marine growth. Many of these events were from Salem 2, but other plants have experienced the same type events for CFCUs and other heat exchangers (e.g., diesel coolers and pump coolers).

Detection of external contamination events was primarily accomplished by two means. Routine testing or surveillance was responsible for discovering 55% of the events, and operational abnormalities accounted for 36%.

3.3.2 Internal contamination

Internally generated contamination resulted in 382 events involving 77 plants. Individual plants reported from 1 to 19 such events, with 10 plants reporting 10 or more.

Internal contamination events were reported for 47 different systems, with the following 10 systems accounting for 250 of the events:

1. containment isolation (73),
2. emergency power (57),
3. safety injection (20),
4. control room habitability (19),
5. main steam isolation (17),
6. containment air purification and cleanup (15),
7. containment (14),
8. condensate and feedwater (14),
9. chemical and volume control (CVCS) and liquid poison (11), and
10. main steam supply (10).

The large number of containment isolation system events is due primarily to contamination of valve seats for containment isolation valves. The emergency power events are a combination of lubricating oil, air start system, and heat exchanger contamination for diesel generators.

The components and parts involved in internal contamination events reflect the ten systems listed. A total of 166 of the events involved various types of valves with the subcomponent parts including valve seats, valve operators, and shafts. Fifty diesel generator events involved contamination of filters, governors, heat exchangers, piping, valves/valve operators, and air start motors.

Detection of internal contamination events was primarily due to routine testing or surveillance (72%), with an additional 24% discovered due to operational abnormalities. The events were also evenly divided between those considered as catastrophic failure (51%) and degraded failures (49%).

3.3.3 Contact corrosion/contamination

This project examined 205 events reporting contact contamination. The events occurred at 58 units with 4 units reporting 10 or more events. The events occurred in 43 systems with 5 of the following systems having more than 15 reports each:

1. emergency cooling (26),
2. emergency power (21),
3. coolant recirculation (18),
4. reactor protection (17), and
5. engineered safety feature instrumentation (16).

The failed parts involved in these events were contacts and similar electrical devices. The components include various types of switches (60 events), relays (24), circuit breakers (23), and monitors (18).

Seventy percent of the contact contamination events were considered catastrophic in nature. This is a higher percentage than for most failure mechanisms and is a reflection of failures pertaining to electrical equipment. Mechanical items are often reported degraded but less frequently are considered to have totally lost their function. More frequently, electrical equipment is considered to either operate or fail completely.

Sixty-eight percent of these events were discovered due to routine testing or surveillance and another 28% due to operational abnormalities.

3.4 Fatigue

Fatigue was reported as the failure mechanism for 324 events examined in this study. Seventy-five plants reported fatigue failures with from 1 to 25 failures reported per plant. The two units at Calvert Cliffs reported 40 fatigue failures from 1976 through 1981. These failures were primarily events associated with charging pumps and reactor coolant pumps, causing degradation of piping and sensing lines, particularly at piping and socket welds.

Fatigue, as reported by LERs, includes both thermal and mechanical fatigue failure. In many cases, the LER is not detailed enough to determine the underlying cause except by inference from failure location (e.g., charging lines experience vibration conditions, feedwater nozzles experience thermal cycling). The systems for which the 324 fatigue failures were reported include:

1. CVCS (31%),
2. coolant recirculation system (11%),
3. emergency core cooling system (ECCS) (10%), and
4. RHR system (6%).

Because of shared components in some PWR designs, charging pump problems can be reported either as a CVCS event or an ECCS event. Charging pumps and other continuously operating high-pressure pumps provide an environment likely to result in mechanical fatigue failures.

Fifty-two percent of the 324 fatigue events involved piping and pipe fittings. Failure of welds comprise the majority of these events, including recurring problems with socket welds. Other components with numerous failures included pumps, valves, and diesel generators. Detection of fatigue events was accomplished primarily by routine testing or inspection (68%) and operational abnormalities (27%). The severity of the events was considered degraded in 75% of the reports and catastrophic for 25%.

3.5 Crack

The study examined 411 events that were keyworded *crack*. Of these, 152 were eliminated (see Chap. 5) or assigned to other failure mechanisms (stress corrosion, vibration, and fatigue). Although crack is not an actual failure mechanism, the remaining 259 events are summarized briefly in this section. They serve to indicate one of the limitations of the LER information, that is, lack of detail.

Sixty-nine different units reported one or more events for which crack was the most specific information about the failure mechanism. These events occurred in 40 different systems with the largest number of reports concerning safety injection (29 events), CVCS (27), coolant recirculation (22), reactor coolant cleanup (21), and main steam (20) systems. As for the similar mechanisms, the primary components/parts are piping/welds, pumps, valves, and steam generators/tubing. Detection methods are also similar, primarily routine testing or surveillance (55%) and operational abnormalities (41%).

The lack of detail concerning these events is important because of the number of events involved. The 259 crack events compare with only 165 vibration events and 110 stress-corrosion events. This means that reassignment of these events to actual mechanisms (if additional information were available) could significantly change some of the failure mechanism information presented here.

3.6 End of Life

In many cases the LERs describing reportable events do not provide details about the failures that they report. In examining LERs available in NOAC's data files, all events that had been keyworded "effect, age" or "age" were selected. These events were classified into the various mechanisms described in this report. One class of events is where the only information available was that the component failed due to "natural end of life," without any further information. After assigning as many specific mechanisms as possible, 226 events remained classified as *end of life*.

Of the 226 events, the systems involved include

1. fire protection systems (12%),
2. airborne radioactivity monitoring system (7%),
3. emergency power system (6%),
4. reactor protection system (5%),
5. containment isolation system (4%), and
6. process and effluent radiation monitoring (4%).

Another 37 systems were involved with 1 to 8 failures each (<4% of the total).

The components and parts involved in events that were reported as *end of life* were largely equipment that the industry expects to replace periodically. The components include smoke detectors, radiation monitors, filters, and certain valves. The parts involved were primarily sensing elements, charcoal filter cartridges, packing, and seals. Because failure of these types of items can be expected, it was not surprising that 64% of the events were detected during normal testing or inspection; another 33% were discovered due to operational abnormalities.

3.7 Vibration

Vibration was reported as the failure mechanism for 165 events at 57 plants. The plants reported from 1 to 11 events each. The events that were classified as *vibration* included two groups of failure. In one group, the vibration resulted in loose parts (nuts becoming unthreaded, coupling coming loose, wiring connections losing contact, etc.). The other group includes events where vibration was stated as the failure cause with no more information given. If it was clearly a vibration-induced fatigue failure, the report was classified as *fatigue*. If not, the event was classified as vibration and is included here.

Two systems each accounted for 15% of the vibration events — the emergency power system and the RHR system. The emergency power system events primarily involved diesel-generator subcomponents including hoses, governors, piping, and control components (switches, relays). The RHR system events involved heat exchangers, piping, valves, valve operators, and fasteners.

In examining the vibration events, the most frequently reported components and parts involved were diesel generators, piping, valves, snubbers, and fasteners (bolts and screws).

Detection of the vibration-related failures was accomplished during routine testing or inspection in 78% of the events, with 21% being discovered due to operational abnormalities. Sixty-one percent of the failures were considered degraded equipment operation, and 39% involved catastrophic failures.

3.8 Stress Corrosion

The study reviewed 110 events in which *stress corrosion* was identified as the mechanism causing the failure. Stress corrosion is different from other mechanisms because it is generally not identified specifically by the operating staff. An LER will often only state that a failure *may* be due to stress corrosion. In other cases, an updated LER will indicate that results from an off-site metallurgical inspection of the failed component identified the specific mechanism as stress corrosion. For this reason, it is likely that a portion of the events reported as corrosion is actually the result of stress corrosion. Also, a large number of the events classified with crack as a failure mechanism (Sect. 6.4) are probably stress-corrosion events.

The 110 events that were identified as due to stress corrosion were reported by 47 plants. The plants reported from 1 to 10 events each in the time frame of the study. The most-reported plant systems subjected to stress corrosion were

1. main steam supply system (15 events, mostly steam generators),
2. reactor coolant cleanup system (14),
3. containment heat removal system (13),
4. ECCS (13), and
5. coolant recirculation system (10).

The primary component subjected to stress corrosion was piping (64 events). More than one-half of these events were stress corrosion of piping welds. Also 12 stress-corrosion events were reported for steam generators. These involved a variety of steam generator parts, including tubing, fasteners, and nozzles.

Detection of stress-corrosion damage was generally accomplished in normal testing and/or inspection (71%), primarily detection of cracking during nondestructive testing.

3.9 Erosion

The study examined 102 events in which *erosion* was identified as the failure mechanism. Also some reports used the term corrosion/erosion, either because plant personnel were unsure of the exact cause of the event or because it was due to the combination of mechanisms. Thirty-nine plants reported one or more erosion events, but two plants together accounted for 38% of the events. Salem 2 (Docket No. 311) reported 24 events, from May 1982 to November 1982, all involving leakage of CFCUs due to tube erosion by silt. Monticello (Docket No. 263) reported 14 cases of piping erosion due to steam since 1976. These plants seem to have specific design problems that have caused the large number of reportable events due to erosion.

The problems that these two plants have experienced represent the two primary types of erosion that have occurred, to a lesser extent, at a large number of plants. The first erosion problem is erosion of service water piping, particularly tubing and elbows, due to contaminants in the service water supply. In addition to silt, other plants have reported problems with erosion in saltwater cooling systems. In most cases, the LERs do not identify a specific contaminant.

The second form of erosion frequently reported is erosion of piping, fittings, and valves due to steam and two-phase water/steam flow. The components primarily affected are relief valves, control valves, and drain lines for steam piping. Problems are also reported for liquid lines at pump discharges.

Detection of erosion problems was primarily via operational abnormalities (66%) or routine inspection and surveillance (27%). The majority of the events resulted in leakage, so the events tended to announce themselves.

3.10 Drift

Of the 7200 events originally extracted from the NOAC LER file for this study, 39% of the failures (2795) were identified as *drift*. A representative number of these drift events were reviewed; due to the similar nature of each failure and the universal use of instrumentation in almost all plant systems, the analysis of the failure characteristics will be addressed separately in this report.

Drift was the reported cause of a system or component failure whenever a safety-related device set point or calibration was found to be outside acceptance criteria delineated in the plant Technical Specifications. To more accurately characterize the incidents reported as drift, further analysis of the reportable events (and the routinely discovered, but not reportable, events) is required to determine whether the cause is true age-related drift or simply the result of administrative factors. Two such factors are (1) Technical Specification acceptance criteria band and (2) instrument maintenance and surveillance practices. On a system basis, reactor protection and main cooling systems contributed about 20% each to the total of 2795 drift events. This would be expected because reactor protection systems are almost totally instrumentation and the main cooling system is the most heavily instrumented. Each has diverse and redundant channels to ensure reliability, which adds to the number and complexity of the subsystems.

No apparent failure trend of instrumentation can be inferred from the data collected. For example, eight plants in commercial operation for about 13 years reported instrument drift events in 2 to 48% of each plant's total LERs; 23 plants in operation about 9 years reported 1.2 to 19.6% of their LERs were caused by instrument drift. Such diversity of data indicates that further study is required to more accurately determine the statistics of true instrument drift (aging) vs the reported events due to administrative requirements (Technical Specifications) or device surveillance, testing, or maintenance practices.

4. SPECIFIC CONCLUSIONS

4.1 Quantity of Data

The search of the NOAC file yielded over 7200 LER abstracts, of which 5890 events were determined to be caused by an identified aging mechanism — either intrinsic to a component or externally applied by the operating environment. The total of 5,893 reports represents 17% of the more than 35,000 LERs (or LER predecessors) in the NOAC data base covering the years 1969 to 1982. Of these, 2795 abstracts identified instrument drift as the cause of a reportable event. The discussion in the rest of this section only addresses the aging-related events other than drift, a total of 3098 events.

4.2 Systems

The 3098 events described in the preceeding paragraph were each associated with one of 68 system classifications. The most frequently reported systems are listed in Table 4.1, with those ten systems representing 53.4% of the events.

Table 4.1. Age-related LERs by system

System designation[a]	System	Number of age-related events involving system	Percent[b]
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.

[b]Based on 3098 age-related events.

The ECCS, emergency generator, containment isolation, and CVCS and liquid poison systems were responsible for 28% of the LERs listing an age-related failure cause. This result is not unexpected as plant Technical Specifications are predominantly concerned with maintaining operability of these key systems. Consequently, almost all failures in these areas are reportable, leading to a large percentage of LERs originating with these four systems.

The next largest percentage of reported events occurred due to service water system failure. Such failures occurred mostly at a component interface with one of the four systems — usually involving heat exchangers or control valves for heat exchangers.

Coolant recirculation system failure events were mostly reactor coolant pump seals and associated cooling water or leakoff piping and controls, as well as reactor coolant system piping degradation. Containment heat removal system failure events mostly involved coolers and associated devices affected by silt or foreign material in the cooling water. Main steam valves (isolation and control) and their controls were the cause of most main steam supply system age-related failure events.

The RHR system events were mainly caused by failure of valves and pumps and associated control devices. The predominant failed devices for containment systems were isolation valves, vacuum breakers, and airlock doors and seals.

4.3 Components and Parts

Tables 4.2 and 4.3 present a ranking of the ten components and parts most frequently identified as failed in the event reports. On a component basis, valves make up 20% of the total failed components, and over one-third (7.8%) were containment isolation valves. Most of the reported events for these types of valves resulted from failure to pass the 10 CFR 50, Appendix J leakage tests required during periodic surveillance testing. In most cases, foreign material or wear on the valve seat was identified as the root cause of leakage. Other types of valves, such as check, control, and drain valves, total slightly over half of all failed valves. Most failures were internal leakage and packing leakage.

Pipe failures comprised over 14% of the age-related incidents reported. The predominant associated failed parts were pipe welds and pipe walls. Most pipe weld failures appeared to be caused by (1) vibration-induced fatigue of inadequately supported piping, (2) temperature-cyling stresses on improperly fitted welds, and (3) weld defects. Pipe wall failures were caused mostly from erosion by the process fluid (wet steam, borated water) or heat-mechanical stress. On a part basis, tubing and pipe failures combined to make up 7.5% of the failed parts — ranking third after welds and miscellaneous subcomponents. Most of these failures were due to vibration-induced cracking at pipe threads or tube fittings.

Pump failures, not including events involving atmosphere monitoring pumps, were responsible for 239 events. Forty-two percent of pump failures involved impellers, wear rings, shafts, bearings, housing, or couplings; 21% resulted from failure of seals or packing. The balance of pump

Table 4.2. Age-related LERs by component

Rank	Component	Number	Percent[a]
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

[a]Based on 3098 age-related events.

Table 4.3. Age-related LERs by part

Rank	Part	Number	Percent[a]
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

[a]Based on 3098 age-related events.

events were failures of belts, mounting bolts, gaskets, and miscellaneous associated parts.

One notable component that appeared to fail frequently was building atmosphere radiation monitors — containment, drywell, auxiliary building, etc. Although not always specified as to manufacturer or model, the vane-type air pumps used in these monitors appeared to fail from wear caused by foreign material in the sampled air stream.

4.4 Failure Mechanisms

Among the other aging mechanisms identified in Table 4.4, *wear* was identified in almost 9% of the reported events as the failure mechanism responsible for the component or part failure. Corrosion of components made up 7% of the LERs, not including electrical contact corrosion in relays and switches, which accounted for 3.5% of the identified failure causes. Debris or contamination originating within a system was termed contamination, internal (6.5%); externally generated contamination (e.g., marine life and construction dust) was termed contamination, external (5.8%). The distinction was made to identify age-related degradation of components that generate contamination (internal) as differing from infrequent or one-time environmentally induced contamination (external), which is not necessarily time-related. For example, many failures of smoke detectors were attributed to construction dust — not a normal operating situation. *Fatigue* was identified in 5.5% of the abstracts as the failure mechanism of the affected part. No further definition of such failures was possible from the reports, as laboratory determination of thermal fatigue or mechanical fatigue on a microscopic basis is not generally made except in the case of major components.

Table 4.4. Failure mechanism percentages

Mechanism	Number	Percent[a]
Drift	2795	47.4
Wear	522	8.9
Corrosion	414	7.0
Crud (internal)	382	6.5
Crud (external)	331	5.6
Fatigue	324	5.5
Crack	259	4.4
End of life	226	3.8
Corrosion contacts	204	3.5
Vibration	165	2.8
Stress corrosion	110	1.9
Erosion	102	1.7
Miscellaneous (oxidation, friction, stress, hardening, high temperature)	59	1.0
TOTAL	5893	100

[a]Percentage of events examined in this study.

4.5 Method of Detection

Timely detection of age-related degradation is a key factor in maintaining the readiness of safety-related systems to perform their function when required. Of the 3098 events determined to be age-related (not including drift), over 64% of the failures were detected by routine testing and surveillance performed in accordance with the plant Technical Specifications or maintenance program. Operating personnel detected 28% of the reportable failure events during normal operational checks and inspections.

For LERs identifying *drift* as a failure cause (2795), a survey of a representative number of abstracts indicated about 80% of the events were detected by scheduled surveillance testing, about 20% of the *drift* events were detected as operational abnormalities by plant operators. These percentages indicate that detection of age-related degradation of many components is accomplished for the most part by plant surveillance testing and instrument maintenance programs.

4.6 Severity

Review of the 3098 age-related LER abstracts indicated about 62% of the failure severities were judged degraded; 38% were deemed catastrophic (Table A.2). No events could be judged to indicate incipient failure due to the nature of an LER — an occasion whenever plant operation falls outside the limits delineated in the respective Technical Specifications. Such events occur only upon discovery of degraded or catastrophic failure of a component — incipient failures are normally detected by routine preventative maintenance (PM) activities and are corrected but not reported. Because such events are not required to be reported, the LERs do not provide an indication of incipient failure detection.

When applying a degraded or catastrophic severity to *instrument* or *relief valve drift* failures, it should be noted that a reportable failure of such devices may not necessarily be classified as catastrophic. The degree of drift is administratively defined in the Technical Specification acceptance criteria; deviation outside these criteria is cause for an LER, but it is not necessarily a device failure.

5. GENERAL CONCLUSIONS

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 nonstatistical 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 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.

6. RECOMMENDATIONS FOR FURTHER STUDY

6.1 Data Sources

This study provides an overview of reportable occurrences resulting from identified age-related failure mechanisms. Note that the frequency of component age-related failures throughout a given nuclear power plant is believed to be much higher than indicated in this study. Because LERs were and are issued only on the occasion of plant operation outside limits contained in a plant's Technical Specifications, failures in systems not important to safety or safety-related are generally not reported. Additionally, the new LER rules scheduled to go into effect in January 1984 will reduce the number of reports generated and consequently the amount of aging data available. However, the Nuclear Plant Reliability Data System (NPRDS) is expected to see increased participation by all commercial nuclear plants under the management of the Institute for Nuclear Power Operation (INPO). This system may hold promise because it is a more wide-ranging and detailed reporting system, which contains useful data on aging of plant components and systems. The NOAC is presently subscribing to and evaluating the usefulness of the NPRDS for contributing to the NPAR Program of aging assessment.

Another effort presently active is the IPRD study. This study utilizes maintenance records of six nuclear power plants to develop reliability data on selected plant components — pumps, valves, diesel generators, batteries, and chargers. The IPRDS does not limit its information only to safety-related components, but includes balance-of-plant equipment, and so provides comprehensive data on these components that may yield more definitive aging information.

The IPRDS may be useful to follow the history of a selected plant component — valve, pump, diesel generator, or battery — to determine if incipient failures were detected (and mitigated), or if not, did a degraded failure occur as a result. Such a study may provide some correlation with a corresponding reportable occurrence event (LER) or an NPRDS failure report.

Other sources of age-related failure data include:

1. plant PM records,
2. ISI reports,
3. in-plant surveillance reports,
4. reports of unscheduled maintenance, and
5. monthly operating reports.

More and more utilities have computers to track these records; such data bases may be useful in extracting additional age-degradation failures. However, such detailed records are not generally available outside the operating utility.

6.2 Failure and Component-Aging Signature

Most failure studies have tended to concentrate on a generic component, that is, valve, and extended the failure rates (for a generic component) to all such components, regardless of type, service, process fluid, operating frequency, material of construction. To develop true aging-mechanism data on components, future work should include characterization of all the age effects acting on each discrete part of a component in similar service. For example, a feedwater pump shaft seal has a substantially different operating environment from the same seal in a decay heat pump; a containment isolation gate valve exercised only once every 10 to 12 months experiences different operating demands than the same valve in a raw service water system.

It is important that study of age-related degradation of components consider the environment for each component (and part) to accurately define the detection and mitigation methods to be used. It may be possible to utilize IPRDS data to identify failure modes of specific components in a specific operating environment, and so characterize the aging history of all such specific components.

The NPAR program includes an analysis of eight key nuclear plant components — motor-operated valves, check valves, auxiliary feedwater pumps, diesel generators, snubbers, batteries, chargers, and inverters — to characterize all the time-dependent effects acting on these components. Additionally, methods of age degradation detection and monitoring will be identified and evaluated for overall contribution to plant safety through mitigation of aging effects.

APPENDICES

Appendix A

LICENSEE EVENT REPORTS

A.1. History

Certain events that occur at nuclear power plants in the United States must be reported to the Nuclear Regulatory Commission (NRC); these events are designated Reportable Occurrences (ROs) and are reported using the NRC's Licensee Event Report (LER) form. Reporting requirements for nuclear facility licensees are delineated in *Code of Federal Regulations*, Title 10: Energy, Parts 20, 40, 50, 70, and 73 (Ref. 1), and are described in detail in NRC *Regulatory Guide No. 1.16 (Ref. 2)*. Events to be reported are specified in the licensee's Technical Specifications and/or license provisions. While all reports from licensees are designated ROs, LER is the accepted term for these reports since 1976.

To collect, collate, store, retrieve, and evaluate information concerning licensee events in a timely manner, the NRC and its predecessor, the Atomic Energy Commission (AEC), have, over the years, maintained a computer-based data file of information extracted from licensee reports. One portion of this data bank, now maintained at the Nuclear Operations Analysis Center (NOAC) and known as the LER file, provides a centralized source of data to be used for qualitative assessment of the nature and extent of off-normal events in the nuclear industry and as an index of source information to which users could refer for more detail. To provide completeness and consistency in the event data, the NRC developed a standard data entry sheet (LER form) and issued instructions for its preparation in October 1974. Issued in July 1977, NUREG-0161 (Ref. 3) incorporated a number of revisions and clarifications to make the reports more useful.

Prior to the use of NUREG-0161, licensee events were usually reported by letter. Because there was no standard format required for these early occurrence reports, the completeness of the data varied considerably from licensee to licensee. Use of the LER form became mandatory in January 1976, and data reported after that time are generally more complete and uniform. However, the types of events reported are not totally consistent because Technical Specifications vary from one plant to another.

A.2. Content

Events that constitute operation outside the plant Technical Specifications are required to be reported. Events that result in damage to major pieces of equipment and/or cause an extended unit outage are generally reported. Events involving non-safety-related equipment are not required to be reported, but some licensees report such events and state that the report is for information only.

An LER form is prepared for each RO. Figure A.1 is a facsimile of the transcription form used. The primary purpose of the LER form (and its

ORNL–DWG 83–5563 ETD

NRC FORM 366
(7-77)

U. S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT

EXHIBIT A

CONTROL BLOCK: (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 — LICENSEE CODE (2) LICENSE NUMBER (3) LICENSE TYPE (4) CAT (5)

CON'T 0 1 — REPORT SOURCE (6) DOCKET NUMBER (7) EVENT DATE (8) REPORT DATE (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2

0 3

0 4

0 5

0 6

0 7

0 8

0 9 — SYSTEM CODE (11) CAUSE CODE (12) CAUSE SUBCODE (13) COMPONENT CODE (14) COMP. SUBCODE (15) VALVE SUBCODE (16)

(17) LER/RO REPORT NUMBER — EVENT YEAR; SEQUENTIAL REPORT NO.; OCCURRENCE CODE; REPORT TYPE; REVISION NO.

ACTION TAKEN (18) FUTURE ACTION (19) EFFECT ON PLANT (20) SHUTDOWN METHOD (21) HOURS (22) ATTACHMENT SUBMITTED (23) NPRD-4 FORM SUB. (24) PRIME COMP. SUPPLIER (25) COMPONENT MANUFACTURER (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0

1 1

1 2

1 3

1 4

1 5 — FACILITY STATUS (28) % POWER (29) OTHER STATUS (30) METHOD OF DISCOVERY (31) DISCOVERY DESCRIPTION (32)

1 6 — ACTIVITY RELEASED (33) CONTENT OF RELEASE (34) AMOUNT OF ACTIVITY (35) LOCATION OF RELEASE (36)

1 7 — PERSONNEL EXPOSURES NUMBER (37) TYPE (38) DESCRIPTION (39)

1 8 — PERSONNEL INJURIES NUMBER (40) DESCRIPTION (41)

1 9 — LOSS OF OR DAMAGE TO FACILITY (43) TYPE (42) DESCRIPTION

2 0 — PUBLICITY ISSUED (44) DESCRIPTION (45) NRC USE ONLY

NAME OF PREPARER ____________________ PHONE: ____________________

Fig. A.1. Licensee Event Report.

subsequent addition to the computer data bank) is to provide a rapid means of wide dissemination of the information, aiding in identification of generic problems and identification of areas for further evaluation. The LER form is intended to be a meaningful abstract of the event, not a substitute for an in-depth report.

Each LER form contains a description of the event and probable consequences, a description of the cause and corrective action, and certain other encoded data. For most events, a supplementary report is included with the LER form. If additional data become available after an LER is issued, a revised LER is submitted for the event by the licensee.

Segments of the LER reporting format that provide information for assessment of aging of components and systems in nuclear power plants include:

1. plant docket number and event date,
2. event description and probable consequences (less than 100 words),
3. cause description and corrective action (less than 100 words),
4. system code,
5. cause code,
6. cause subcode,
7. component code,
8. component subcode,
9. valve subcode (if applicable),
10. action taken,
11. effect on plant,
12. shutdown method,
13. method of discovery, and
14. discovery description.

With time, the format of the LER abstract has been modified to provide additional data in a more convenient form. Events prior to 1976 may not have a unique identification number because the licensee may not have assigned one to each event. Also, the date given in the LER abstract is often the date of the report and not the date of the event. This was done to aid in retrieval of the original document because letters are filed by letter date. Abstracts prepared prior to 1981 may contain data from the supplemental report that is attached to some LERs. Since then, only data from the LER form are included.

A.3. Data Bases — RECON and SCSS

Since 1967, the Nuclear Safety Information Center (NSIC), now NOAC, has abstracted and stored LERs and their generic predecessors on a computer file. From 1973 through 1981, the NRC also stored LER data on a computer file that was discontinued in 1982. The NOAC data base is now the official file for the NRC.

The LER file is available through the RECON computer program maintained by the Department of Energy (DOE). RECON (an acronym for REmote CONsole) signifies that the system can be accessed by users located at various sites across the United States. It is a computerized, interactive, on-line information retrieval system developed to provide rapid and easy access to energy-related data bases. There are over 40 files available at this time; they are routinely accessed by over 600 users. One of the data files is the NOAC file that includes the LER data, as well as other information on nuclear power plant safety, operating experience, licensing data, and design.

The LERs received at NOAC are reviewed by the technical staff and prepared for entry into the computer file. Keywords assigned to each LER enable a user to search the computer file for selected areas of interest. The abstract, which contains the descriptive material provided on the LER form, and selected encoded data are then entered into the computer and transferred to RECON. Searches on RECON use Boolean logic to combine keywords and other keyed fields to obtain an output for a desired subject matter.

As the number of LERs on the data base increased, a need for a more advanced search system was recognized. Keywords are useful for searches of general types. Detailed searching, such as for system interaction studies or analysis of system failures, was not feasible. In 1980, the NRC Office for Analysis and Evaluation of Operational Data (AEOD) initiated the development of a new LER data system at NOAC. This new system, Sequence Coding and Search System (SCSS), reduces the descriptive text of the LER and its supplemental report to a coded sequence that is both computer readable and searchable. This system provides a structured format for detailed coding of component, system, and unit effects as well as personnel actions. The SCSS reduces the event to time-ordered occurrences and relates each occurrence to one or more prior occurrences and/or future occurrences as appropriate. The component, system, quantity, cause, and failure effect of each occurrence are coded. SCSS is still under development; however, at this time most LERs with 1981 and 1982 event dates are on the computer. Selected LERs prior to 1981 may be processed at a later date.

As discussed in Chap. 1, study personnel reviewed 4461 event abstracts in detail. The purpose of the review was to identify events that were age-related. The review process involved elimination of nonaging effects, consolidation of information from multiple abstracts concerning a single event, and disregard for events involving other than commercial power reactors. 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. Each record contains:

1. accession number,
2. plant docket number,
3. report number,
4. event date (or report date if event date was not available),
5. system,
6. component,
7. specific part,
8. failure mechanism,
9. severity,
10. method of detection, and
11. brief event description (including additional specific information concerning age, materials, and environment).

Items 1—5 are standard information contained in every event report. Items 6—10 required technical judgment on the part of the reviewer to accurately classify and characterize the components, failure mechanism, severity, and method of detection. The comment field, item 11, was provided to allow inclusion of additional pertinent information.

A discussion of the coding rationale for items 6—11 follows.

1. Component and specific part (items 6 and 7). The component field lists the component whose function was affected by the failure, while the part represents the discrete specific item that failed. The intent was to be as specific as the information would allow in identifying failed components and parts. The list of component/part codes used in the study is given in Table A.1.
2. Failure mechanism (item 8). The age-related failure mechanisms are discussed in Chap. 3. It should be noted that the NOAC keywords (Table 3.1) do not necessarily correspond to each failure mechanism used in the study. For example, an event extracted using the keyword *wear* may be coded as *erosion* if such distinction could be made from the abstract. Additionally, some keywords (e.g., *crack* and *crud*) are symptoms of an age effect, while others actually are age effects. The failure mechanism was specifically identified for the part that failed, when possible.
3. Severity (item 9). The failure severity definitions used in the study are those used by the In-Plant Reliability Data (IPRD) study[4] being conducted at ORNL (Table A.2). Only the terms *degraded* and *catastrophic* as applied to the specific failed part were applicable in this study. The major component containing the affected part may operate in an incipient (or degraded) mode, but the occasion for issuance of an LER is the failure of the part that places the plant in an operating condition outside the Technical Specifications. Therefore, LERs do not report incipient failures.
4. Method of detection (item 10). The codes and definitions listed in Table A.3 indicate the method by which the failure was made known to plant operating personnel. Assignment of these codes from LER abstracts was not always straightforward because such information is often not directly available.
5. Event description/comments (item 11). A short statement is included of the event as it relates to the overall system or plant, for example, "power range channel failed low" or "containment pressure sensor failed." Comments mention any pertinent information given that is relevant, such as:

1. number of components,
2. number of failures,
3. age of failed part,
4. age of failed component,
5. service life of failed part,
6. operating environment — past or present,
7. material of failed part, and
8. other age-related information as appropriate.

Table A.1. Component/part codes used in the aging project

Code	Description
ACC	Accumulator
ADRY	Air dryer, absorption
AHU	Air handling/conditioning unit
ALAR	Alarm
AMP	Amplifier
BY	Battery
BRG	Bearing/bushing
BEL	Bellows
BLT	Belt
BIS	Bistable
BOLT	Bolt
BRK	Brake
VCBR	Breaker, vacuum
BSH	Brush
BUS	Bus
CBL	Cable/wire
CAP	Capacitor
CCD	Card, circuit
CHAR	Charcoal
BYC	Charger, battery
CBA	Circuit breaker, ac
CBD	Circuit breaker, dc
CBX	Circuit breaker, unknown type
CUE	Cleanup equipment, miscellaneous
CTC	Clutch
CL	Coil
CCL	Coil, cooling
CMP	Compressor
CPU	Computer
COND	Condenser
ICON	Condenser, ice
CON	Connector
CNTR	Contactor/contacts
CRE	Control rod
CRA	Control rod assembly
CRD	Control rod drive
PMC	Controller
CLR	Cooler
CHX	Cooler (heat exchanger)
CTW	Cooling tower
CPLG	Coupling
CRN	Crane
DMP	Damper/louver
DET	Detector
DPM	Diaphragm
DISC	Disc, valve

Table A.1 (continued)

Code	Description
DR	Door/cover/hatch
DRN	Drain
DRY	Dryer
DUCT	Duct
EDR	Eductor
EJR	Ejector
ELBO	Elbow
DSL	Engine, diesel
ENG	Engine, other type
XXX	Entire system
EXC	Exciter
FCU	Fan cooler unit
FAN	Fan/blower
FAS	Fastener
FLTI	Filter (instrumentation and control)
FLTP	Filter (process)
FLA	Fuel assembly
FLE	Fuel element/rod
FHE	Fuel handling equipment
FU	Fuse
GSKT	Gasket
GR	Gear
GEN	Generator
MG	Generator, motor
GOV	Governor
HANG	Hanger
HX	Heat exchanger
HTTR	Heat tracing
HTRE	Heater, electric
HTR	Heater, other type
HOSE	Hose
HSNG	Housing (pump/motor)
HYD	Hydrant
IMPL	Impeller (vanes)
IND	(Indicator)
ISL	Insulation
INT	Integrator
INL	Interlock
INV	Inverter
MSF	Miscellaneous structural features
MSC	Miscellaneous subcomponent
MON	Monitor
MOT	Motor
MOTS	Motor starter
NZL	Nozzle
ORNG	O-ring
PACK	Packing, valve

Table A.1 (continued)

Code	Description
PND	Penetration, electrical
PNC	Penetration, equipment access
PNO	Penetration, other/unknown type
PNA	Penetration, personnel access
PNP	Penetration, process piping
PIPE	Pipe (any size or material)
PIST	Piston
PLG	Plug
JX	Power supply, electric
JXU	Power supply, uninterruptible
PZB	Pressurizer
PE01	Primary element, analyzer
PE02	Primary element, conductivity
PE03	Primary element, current
PE04	Primary element, fire/smoke
PE05	Primary element, flux/neutron
PE06	Primary element, level
PE10	Primary element, moisture/humidity
PE11	Primary element, other/unknown type
PE13	Primary element, pressure
PE14	Primary element, radiation
PE15	Primary element, speed/frequency
PE16	Primary element, temperature
PE17	Primary element, torque/force
PE18	Primary element, vibration
PE19	Primary element, voltage
PMPA	Pump, axial
PMPB	Pump, centrifugal
PMPC	Pump, diaphragm
PMPK	Pump, jet
PMPX	Pump, other type
PMPE	Pump, reciprocating
PMPZ	Pump, unknown type
PMPH	Pump, vane type
PB	Push button
RCB	Recombiner
RECO	Recorder
RECT	Rectifier
REL9	Relay
REL2	Relay, overcurrent
REL3	Relay, overvoltage
REL4	Relay, time delay
REL5	Relay, undercurrent/underpower
REL6	Relay, undervoltage
RSSR	Resistor
RPD	Rupture disc
SCN	Screen

Table A.1 (continued)

Code	Description
SCRU	Screw
SEAL	Seal
SL	Sensing line
SEP	Separator
SEQ	Sequencer
SHFT	Shaft/stem
SLV	Sleeve
SNB	Snubber
SOCW	Socket weld
SOL	Solenoid
SPRG	Spring
SG	Steam generator
STNR	Strainer
SFR	Structural framing and foundation
SPT	Support
SWF	Switch, flow
SWL	Switch, level
SWM	Switch, manual
SWX	Switch, other type
SWS	Switch, overspeed
SWZ	Switch, position/limit
SWP	Switch, power
SWPR	Switch, pressure
SWTP	Switch, temperature
SWTS	Switch, test
SWB	Switch, torque/force
TK	Tank
TEE	Tee (pipe fitting)
TB	Terminal block
TOT	Totalizer/integrator
TD	Transducer
XFMR	Transformer, other type
XPWT	Transformer, power
TRF	Transmitter, flow
TBL	Transmitter, level
TRPR	Transmitter, pressure
TRS	Transmitter, speed/frequency
TBG	Tubing
TBN	Turbine
TCHG	Turbocharger
ZZZ	Unknown component
VOB	Valve operator, electric/servo
VOG	Valve operator, explosive/squib
VOC	Valve operator, hydraulic
VOX	Valve operator, other type
VOK	Valve operator, piston
VOD	Valve operator, pneumatic

Table A.1 (continued)

Code	Description
VOE	Valve operator, solenoid
VOZ	Valve operator, unknown type
VLVS	Valve seat
VABP	Valve, bypass
VACH	Valve, check
VACN	Valve, control
VADR	Valve, drain
VAIS	Valve, isolation
VAPR	Valve, pressure reducing
VARF	Valve, relief
VART	Valve, root (instrument)
VASP	Valve, sample
VAZZ	Valve, unknown function or type
VAVN	Valve, vent
VSL	Vessel
RPV	Vessel, reactor
WALL	Wall/bulkhead
WRNG	Wear ring
WELD	Weld

Table A.2. Aging study codes for failure severity

Code	Description
C — catastrophic	The part is completely unable to perform its function. For example, a pump is frozen and will not operate due to a seized bearing (part); a valve fails to change position on demand due to a clogged pilot solenoid valve.
D — degraded	The part operates at less than its specified performance level. A degraded part failure does not normally bring about failure of the related component to perform its intended function; for example, leakage from a pump seal or partial or slow position change of a valve.
I — incipient	The component performs within its design envelope but exhibits characteristics that, if left unattended, will probably develop into a degraded or catastrophic failure. For example, in a pump, a leak at the mechanical seal (degraded seal), or excessive vibration, noise, or overheating (degraded bearing), is classified as incipient pump failure. Note that an incipient failure does not normally bring about failure of the component to perform its intended function.

Table A.3. Aging study method of detection categories

Code	Description
F — operational abnormality	Condition manifested (by means other than audio/visual alarms) during normal plant operation.
H — AE/vendor notification	Condition became known as a result of architect-engineer or vendor notification.
I — routine testing/inspection	Condition discovered during scheduled surveillance activities or other normal investigative duties of plant personnel.
K — maintenance	Condition discovered during the performance of maintenance. (This is often applicable to subcomponent failures, the identity of which is not evident.)
L — special tests/inspection	Condition discovered by an audio and/or visual alarm during normal plant operation.
R — review of procedures/test results	Condition discovered during review or audit of plant procedures, records, or test results by plant or contractor personnel.

The study found that information desired for statistical evaluation of aging effects (equipment age, service life, environment) was seldom available. This reflects the intent of the LER system as a regulatory instrument, rather than an engineering data collection system.

References

1. Title 10, "Energy," *Code of Federal Regulations* (10 CFR), Parts 20, 40, 50, 70, and 73.
2. "Reporting of Operating Information — Appendix A, Technical Specifications," *NRC Regulatory Guide 1.16*, Rev. 4 (August 1975).
3. *Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File*, NUREG-0161, July 1977.
4. R. J. Borkowski et al., *The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report*, NUREG/CR-2641, ORNL/TM-8271, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., July 1982.

Appendix B

COMPARISON WITH ACCIDENT SEQUENCE PRECURSOR STUDY EVENTS

B.1. Discussion

This appendix discusses the event selection process conducted in the Accident Sequence Precursor (ASP) program and compares that process with the approach taken in the aging study.

The initial ASP program examined ~19,400 Licensee Event Reports (LERs) from the period 1969—1979.[1] The selection criteria for the initial screening of these LERs are listed in Sect. 2 of this appendix. Application of the criteria resulted in selection of 529 events for detailed review. The detailed review identified 169 events considered to be precursors to 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.

Phase I of the aging study involved review of events from 1969 to 1982. Approximately 34,000 abstracts for operational events in that time frame are available at the Nuclear Operations Analysis Center (NOAC). Utilizing the keywords available in the NOAC abstract file, 7256 events were selected for individual review. Of these events, ~2795 events were due to drift effects (e.g., instrument drift and relief valve set point drift). These events were categorized briefly, without in-depth review. Of the rest of the events, 3098 were identified as occurring due to age-related failure during the detailed aging study review. Eleven events were common to both studies. These events are summarized in Table B.1. 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. This is illustrated in events 2, 7, 8, and 10 listed in Table B.1.

This discussion emphasizes the importance of a point illustrated by the aging study information. Routine monitoring and surveillance testing is the most important defense against safety function degradation. Monitoring of operating equipment and periodic surveillance testing of standby equipment can and does identify age-related incipient or degraded failures. If the industry utilizes effective surveillance programs, aging problems can be handled even though component designs can never totally eliminate individual aging failures.

Table B.1. 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.

B.2. Criteria for Selection of LERs for Detailed Review as Precursors

Identification of those 1969—1979 LERs that required a detailed review as precursors was made based on an examination of the abstract of each LER. Approximately 19,400 LER abstracts were examined, and specific LERs were chosen if any of the following criteria were met:

1. any failure to function of a system that should have functioned as a consequence of an off-normal event or accident;
2. any instance where two or more failures occurred;

3. all events that resulted in or required initiation of safety-related equipment (except events that only required trip and when trip was successful);
4. all complete losses of off-site power and any less frequent off-normal initiating events or accidents;
5. any event or operating condition that was not enveloped by or proceeded differently from the plant design bases; and
6. any other event that, based on the reviewer's experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Reference

1. J. W. Minarick and C. A. Kukielka, *Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report*, NUREG/CR-2497 (ORNL/NSIC-182), Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., June 1982.

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16. ABSTRACT *(200 words or less)* The results of a study utilizing the Oak Ridge National Laboratory (ORNL) Nuclear Operations Analysis Center (NOAC) computer files of operating experience reports [licensee event reports (LERs), abnormal occurrences, etc.] are summarized. 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 5800 events attributable to age-related failures. Of these events, 2795 were attributable to instrument drift, which are addressed separately in the report. The remaining events (3098) 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 an 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. The study is a part of the overall Nuclear Regulatory Commission research on nuclear power plant aging effects.

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