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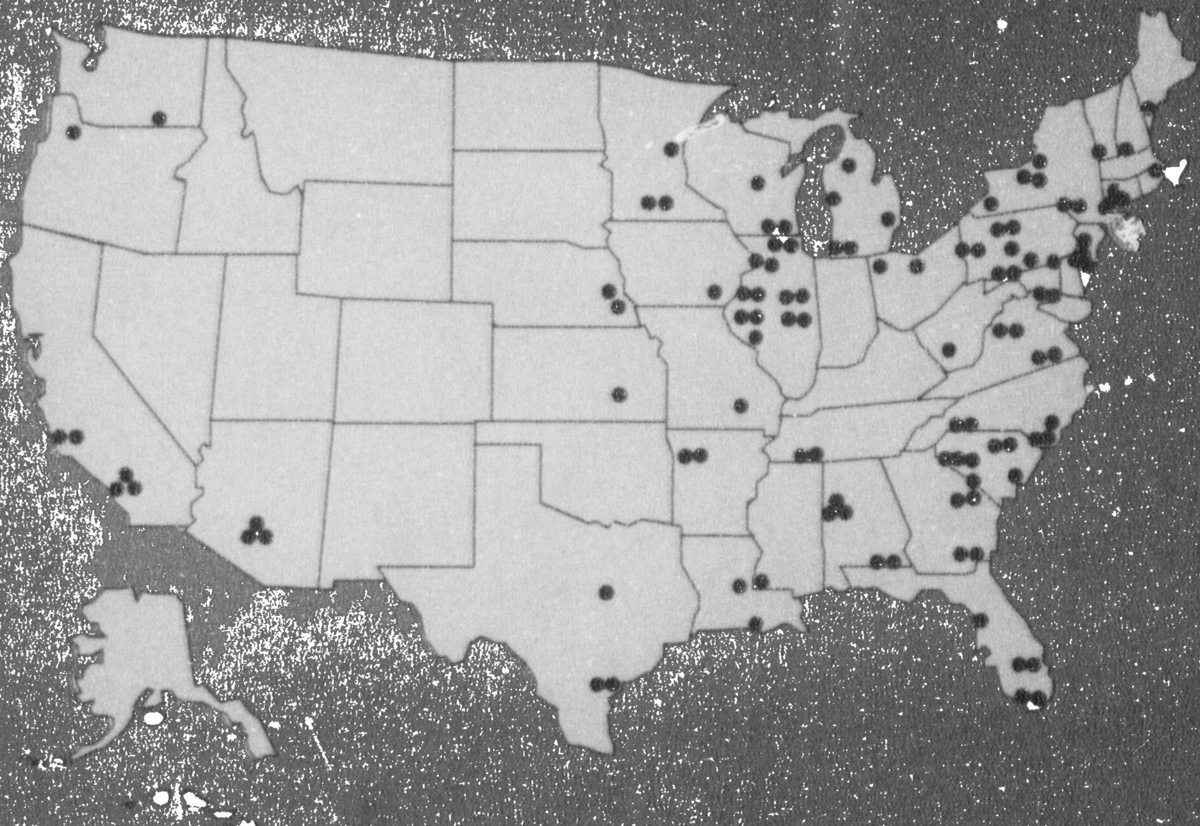
NUREG-1272 VOL. 7, NO. 1

Analysis and Evaluation of Operational Data



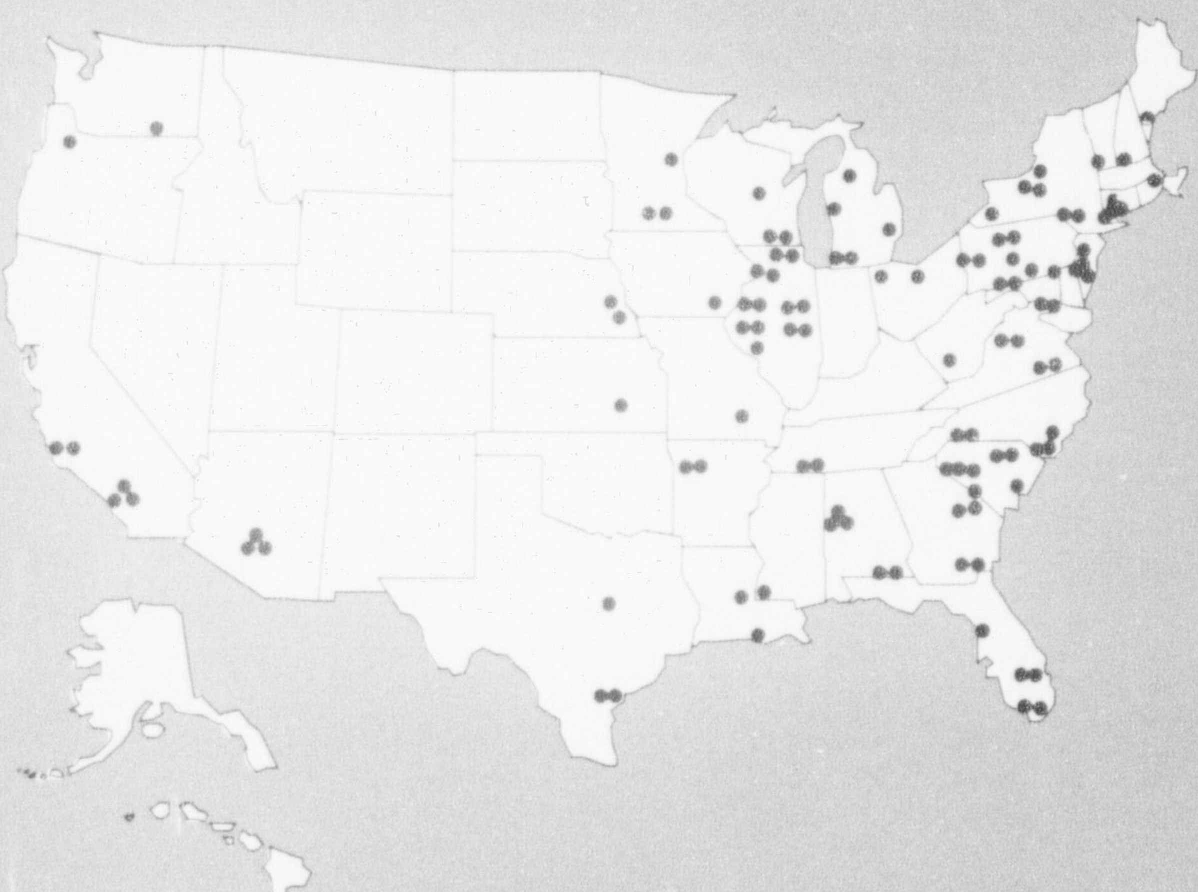
1992 ANNUAL REPORT

POWER REACTORS

U.S. NUCLEAR REGULATORY COMMISSION

O F F I C [illegible] R [illegible] NUREG-1272 VOL. 7, NO. 1

ANALYSIS AND EVALUATION OF OPERATIONAL DATA



1992 ANNUAL REPORT

POWER REACTORS

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O F F I C E   F O R

NUREG-1272
VOL. 7, NO. 1

ANALYSIS AND EVALUATION OF OPERATIONAL DATA

1992 ANNUAL REPORT

POWER REACTORS

U.S. NUCLEAR REGULATORY COMMISSION

JULY 1993

The map on the cover shows the locations of commercial nuclear power plants licensed to operate in the United States.

Previous Reports in Series

The following semiannual or annual reports have been prepared by the Office for Analysis and Evaluation of Operational Data (AEOD).

- Semiannual Report, January – June 1984, AEOD S/405, September 1984
- Semiannual Report, July – December 1984, AEOD/S502, April 1985
- Annual Report 1985, AEOD/S601, April 1986
- *Report to the U.S. Nuclear Regulatory Commission of Analysis and Evaluation of Operational Data* 1986, NUREG-1272, AEOD/S701, May 1987
- *Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data* 1987, Power Reactors, NUREG-1272, AEOD/S804, Vol. 2, No. 1, October 1988
- *Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data* 1987, Nonreactors, NUREG-1272, AEOD/S804, Vol. 2, No. 2, October 1988
- *Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Power Reactors,* NUREG-1272, Vol. 3, No. 1, June 1989
- *Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Nonreactors,* NUREG-1272, Vol. 3, No. 2, June 1989
- *Office for Analysis and Evaluation of Operational Data 1989 Annual Report*, NUREG-1272, Vol. 4, No. 1, July 1990
- *Office for Analysis and Evaluation of Operational Data 1989 Annual Report*, NUREG-1272, Vol 4, No. 2, July 1990
- *Office for Analysis and Evaluation of Operational Data 1990 Annual Report,* NUREG-1272, Vol. 5, No. 1, July 1991
- *Office for Analysis and Evaluation of Operational Data 1990 Annual Report,* NUREG-1272, Vol. 5, No. 2, July 1991
- *Office for Analysis and Evaluation of Operational Data 1991 Annual Report,* NUREG-1272, Vol. 6, No. 1, July 1992
- *Office for Analysis and Evaluation of Operational Data 1991 Annual Report,* NUREG-1272, Vol. 6, No. 2, August 1992

Abstract

The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1992. The report is published in two separate parts. NUREG-1272, Vol. 7, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. NUREG-1272, Vol. 7, No. 2, covers nonreactors and presents a review of the events and concerns during 1992 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Both reports also contain a discussion of the Incident Investigation Team program and summarize both the Incident Investigation Team and Augmented Inspection Team reports. Each volume contains a list of the AEOD reports issued for 1980–1992.

Contents

Contents (cont.)

Contents (cont.)

Appendices

Figures

Figures (cont.)

Tables

Abbreviations

AEOD	Analysis and Evaluation of Operational Data (NRC Office for)
AFW	auxiliary feedwater
AIT	augmented inspection team
ALARA	As Low as Reasonably Achievable
AM	above median
AMF	alternate minimum flow
ANO	Arkansas Nuclear One
ANSI	American National Standards Institute
AO	abnormal occurrence
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
ATI	automatic test insertion
ATWS	anticipated transient without scram
BM	below median
B&W	Babcock & Wilcox Co.
BWR	boiling-water reactor
BWROG	BWR Owners Group
CCDP	conditional core damage probability
CE	Combustion Engineering
CFR	Code of Federal Regulations
C.I.S.	Commonwealth of Independent States
CRE	collective radiation exposure
CRGR	Committee To Review Generic Requirements
CSNI	Committee on the Safety of Nuclear Installations
CY	calendar year
DBR	design-basis review
DE	diagnostic evaluation
DEP	Diagnostic Evaluation Program
DER	design electrical rating
DET	diagnostic evaluation team
E	extensive
ECCS	emergency core cooling system
EDG	emergency diesel generator
EDO	Executive Director for Operations
EFO	equipment-forced outage
EOP	emergency operating procedure
EP	emergency power
EPC	Emergency Procedures Committee
EPG	emergency planning/procedure guideline
EPRI	Electric Power Research Institute
EQ	environmental qualification
ERP	emergency response procedure
ESF	engineered safety feature(s)
ESFAS	engineered safety features actuation system
GE	General Electric Co.
GI	generic issue
GL	generic letter
HOO	headquarters operations officer
HHSI	high-head safety injection
HPCI	high-pressure coolant injection
HPCS	high-pressure core spray
HPI	high-pressure injection
HVAC	heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronics Engineers

IGSSC	intergranular stress corrosion cracking
IIP	Incident Investigation Program
IIT	incident investigation team
ILRT	integrated leak rate test
IN	information notice
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
IRM	intermediate-range monitor
IRS	Incident Reporting System
ITG	Interoffice Task Group
LER	licensee event report
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LTS	long-term solution
LWR	light-water reactor
MCCB	molded-case circuit breaker
MDC	maximum dependable capacity
MOV	motor-operated valve
MR	medium range
MSSV	main steam safety valve
NEA	Nuclear Energy Agency
NMAC	Nuclear Maintenance Assistance Center
NMP-2	Nine Mile Point Unit 2
NMSS	Nuclear Material Safety and Safeguards (NRC Office of)
NPRDS	Nuclear Plant Reliability Data System
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation (NRC Office of)
NSO	nuclear station operator
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resources Council
OECD	Organization for Economic Cooperation and Development
OGC	Office of the General Counsel
ORNL	Oak Ridge National Laboratory
PASNY	Power Authority of the State of New York
PCIV	primary containment isolation valve
PI	performance indicator
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSV	pressurizer safety valve
PWG	principal working group
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
REIRS	Radiation Exposure Information Report System
RES	Nuclear Regulatory Research (NRC Office of)
RHR	residual heat removal
RI	Region I
RO	reactor operator
RPI	rod position indication
RWCU	reactor water cleanup
SALP	Systematic Assessment of Licensee Performance
SCSS	Sequence Coding and Search System
SG	steam generator

SI	safety injection
SLCS	standby liquid control system
SONGS	San Onofre Nuclear Generating Station
SOV	solenoid-operated valve
SRV	safety-relief valve
SSF	safety system failure
STA	shift technical advisor
TMI	Three Mile Island
TS	Technical Specifications
UPS	uninterruptible power supply
WNP-2	Washington Nuclear Project, Unit 2

Executive Summary

General

Since its formation in 1979, one of the primary missions of the NRC Office for Analysis and Evaluation of Operational Data (AEOD) has been to provide a strong, independent capability for the analysis of operational data. The office serves as the focal point for the independent assessment of operational events through the review, analysis, and evaluation of the safety performance of reactors and other facilities. AEOD is also responsible for the NRC's Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and Incident Investigation Program. In addition, AEOD provides administrative and technical support for the Committee To Review Generic Requirements (CRGR). Office activities related to facilities that are not reactors (nonreactors) are reported in the second volume of this annual report, and activities related to the technical training program are published in a separate report.

To perform this mission, the AEOD staff collects, analyzes, and disseminates operational data, assesses trends in performance from these data, and analyzes operating events to provide insights and to improve the understanding of events by providing a risk perspective for events deemed to be significant. Other elements that contribute to this mission are diagnostic evaluations and incident investigations.

The AEOD programs, taken as a whole, constitute the essential independent review and assessment of power reactor safety performance. This independent review and assessment complements the regional and the Office of Nuclear Reactor Regulation (NRR) review of operating events, and provides a quality assurance function through a system of checks and balances that reduces the likelihood that an important safety lesson will be overlooked. AEOD findings and recommendations continue to be addressed through generic correspondence, in the resolution of generic issues, and in initiatives taken by industry.

The CRGR comprises senior NRC managers and is chaired by the Director, AEOD. It reviews proposed new generic requirements, new staff positions for adherence to the provisions of 50.109 and the Commission's Safety Goal Policy, and it gives its views on these requirements to the Executive Director for Operations before those requirements are promulgated. AEOD also oversees NRC's plant-specific backfitting, the training of NRC personnel, and the auditing of NRC practices. The staff also obtains industry feedback for these activities.

Nuclear Reactor Safety Performance

Commercial nuclear power reactors in the United States now have approximately 1,700 reactor years of operating experience. Through the many activities of AEOD, trends in overall safety performance of power reactors may be inferred. The Performance Indicator (PI) and Accident Sequence Precursor (ASP) programs of AEOD have been applied to analyze operational data and related information in a consistent manner in recent years.

The PI Program includes seven indicators: automatic scrams while critical, significant events, equipment-forced outages per 1,000 critical hours, collective radiation exposures, safety system failures, safety system actuations, and forced-outage rate. One of these indicators, safety system actuations, continues to exhibit a significant improving trend. Equipment-forced outage per 1,000 commercial critical hours remained relatively constant from 1988 to 1991, and improved in 1992. The forced-outage rate has been erratic. Scrams, significant events and collective radiation exposures have remained essentially constant for the last 2 years. Safety system failures have stablized within a narrow range.

AEOD performed additional analyses of the PI data related to the leveling off of the PIs. Evaluations of these data over the most recent years lent additional support to the observation that the PIs were stabilizing. This apparent leveling off of performance was identified in several indicators last year, and has expanded this year to include all but one indicator: safety system actuations. Performance indicators are influenced by varying root causes; however, equipment failures during normal operation continue to dominate the causes for scrams. Problems in the feedwater system continued to initiate the greatest number of scrams. Engineered safety features actuations and actual safety system failures are dominated by maintenance activities.

On the basis of these comparative evaluations and the additional PI analysis, it appears that on a nationwide basis, the PIs have stabilized, although additional effort is warranted among the poorer performers.

The Accident Sequence Precursor (ASP) Program quantitatively evaluates operational experience. It serves as one of several tools to ensure that important operating lessons are not overlooked. It uses a rigorous method that integrates actual initiating events, plant conditions, and the reliability of standby safety equipment into an overall quantitative assessment, which is expressed as the conditional core damage probability (CCDP). Examination of ASP evaluations for the last 8 years indicates that some improvement took place during this period.

Radiation Exposures

Collective radiation exposures at reactor sites continue to decline. The average collective dose per reactor was 582 person-centisieverts (cSv) (person-rem) in 1973 and 421 person-cSv in 1987, declining to 257 person-cSv in 1991. Of the six categories of licensees that are required to report collective exposures, for 1991, reactor licensees, by virtue of the large number of employees, had the highest collective exposure (28,527 cSv to 184,829 people), followed by radiographers (2,160 cSv to 6,820 people), manufacturers and distributors (721 cSv to 4,930 people), and fuel fabrication licensees (272 cSv to 11,016 people). Low-level waste disposal (39 cSv to 905 people) and independent spent fuel storage (4 cSv to 41 people) licensees had relatively low collective doses. Among the measures that reduce collective exposures at nuclear power plants are licensee efforts to improve maintenance programs, experienced and well-trained personnel, a good water chemistry control program, effective decontamination and cleanup practices, good fuel cladding integrity, effective radiation exposure control programs, good housekeeping, and an alert health physics staff. Licensee violations of NRC limits on personnel exposures are rare.

Results of AEOD Studies

In 1992, AEOD issued 1 case study, 3 special studies, 3 engineering evaluations, and 10 technical review reports. In these documents, the staff analyzed safety issues that arose from the study of nuclear plant operating experience. These analyses covered a wide range of subjects and varied from a relatively broad evaluation to develop insights from common-cause failure events, to an in-depth review of such component-related problems as safety valves and molded-case circuit breakers, quantitative analysis of the risk associated with operational events and conditions, to studies that contributed to a better understanding of human performance.

The study on pressure locking and thermal binding of gate valves and the one on human performance problems were distributed to all licensees as NUREG reports. The study on dominant corrective actions that would preclude or reduce the likelihood of common-cause failures provided the basis for an NRC information notice. Other studies were considered in the resolution of several generic safety issues.

AEOD studies of operating events have shown the importance of human performance in reactor safety. To obtain additional information, AEOD continued to conduct onsite studies of human performance for selected power reactor events. AEOD Case Study C92-01 described observations and conclusions from 16 human performance event investigations. Some important observations from this study are that licensee response to events has been affected by such factors as (1) control room staffing level, division of responsibility, and degree of teamwork significantly affecting operating crew response to events; (2) some operators acting during events without using a procedure; (3) some operators inappropriately defeating the automatic operation of safety features during valid system demands; (4) a lack of direct-reading, control room instrumentation with the appropriate range causing operators to have difficulty in recognizing and responding to certain shutdown events, when operator actions were required to accomplish the safety functions of disabled automatic safety systems; (5) annunciator and computer alarms being important operator aids in recognizing and responding to events; (6) lack of direct control room indication of flow rate affecting the reactor coolant system inventory, including discharges of safety-relief valves connected to the reactor coolant system, impairing operator response to events, especially during shutdown; and (7) the wide variance in the effectiveness of licensee studies of human performance. While some licensees have not responded aggressively to identified human factors weaknesses, others have initiated worthwhile plant-specific corrective actions because of their human performance studies.

Other AEOD Activities

Performance Indicators Enhancements. During 1992, the AEOD staff conducted a trial program of the enhanced performance indicator methodology which accounts for the effects of the operational phase of the plant, and peer groups for comparing performance. Based on the results of the trial program, the staff recommended, and the Commission approved, the use of the enhanced performance indicator report. The changes are designed primarily to determine (1) operating and shutdown performance separately, (2) performance relative to a group of similar plants rather than to the entire industry, and (3) the statistical significance of the calculated trends and deviations. These modifications allow substantially more information to be conveyed clearly and concisely.

Event Reporting. In September 1992, the NRC published a minor rule change deleting the reporting requirements for a limited set of specifically defined invalid engineered safety features actuations that have been determined to be of little or no significance. In addition, AEOD worked toward resolving the issues raised by public comments on the detailed reporting guidance proposed in draft report NUREG–1022, Rev. 1. As part of this effort, the staff decided to hold a second public meeting on this subject in May 1993 to obtain more public comment on the resolution of the identified issues.

Abnormal Occurrences. AEOD administers the Commission's program for reporting abnormal occurrences (AOs) to Congress. AOs are incidents or events that the Commission determines are significant from the standpoint of public health and safety. In 1992, NRC reported three AOs at nuclear power plants to Congress. The AOs occurred at (1) Shearon Harris, (2) Arkansas Nuclear One Unit 2 and McGuire Units 1 and 2, and (3) Millstone Unit 2. In 1992, AEOD issued four quarterly reports describing several AOs at nonreactor licensees (see NUREG-1272, Vol. 6, No. 2) and various events of interest, and updated information on previously reported AOs. The number of AOs in nuclear power plants since 1988 has remained low, fewer than four per year.

Immediate Notification and Licensee Event Reports. In 1992, 2,276 immediate notifications, primarily from nuclear power plant licensees (2,033 notifications), were telephoned to the NRC Operations Center. Of these 2,276 notifications, 135 were classified as "Unusual Events," 20 as "Alerts," and 1 as a "Site Area Emergency." The latter two categories are events during which, in the licensee's preliminary appraisal, a substantial degradation of plant safety occurred or had the potential to occur. In addition to prompt screening of telephone notifications, AEOD reviewed and evaluated licensee event reports (approximately 1,800) to identify events or aspects of events with safety significance.

Augmented Inspections. During 1992, NRC conducted nine augmented inspection team (AIT) inspections at nuclear power plants. Examples of lessons learned and communicated to licensees from these AIT inspections include potential problems pertaining to inadvertent isolation of the ultimate heat sink, loss of control room annunciator and alarm systems, and power oscillations in a boiling-water reactor where such instabilities had not been expected.

International Operating Experience. AEOD continued to maintain and improve the exchange of operational experience and ideas with foreign countries as a contribution to nuclear power safety programs worldwide. At the end of 1992, 109 of the more than 410 nuclear power plants in commercial operation worldwide were located in the United States. In 1992, AEOD reviewed 158 reports of events at foreign reactors received through the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) Incident Reporting System (IRS). Additional information was obtained through bilateral exchange programs with more than 20 countries. AEOD reviewed these reports to identify issues of importance to the U.S. nuclear power program and to disseminate findings from studies of these events to U.S. licensees. Examples of issues of importance to domestic reactors that were highlighted by these information sources were (1) discovery of reactor vessel control rod penetration cracks, (2) boiling-water reactor wetwell pump suction strainers clogged by insulation, (3) gate valve pressure locking, and (4) water intrusion into the control room.

During 1992, AEOD prepared and submitted to the international community 55 reports of U.S. incidents that addressed operational events and generic concerns. Subjects reported included failures and potential failures of piping caused by corrosion/erosion testing and surveillances of motor-operated valves, pressure locking of motor-operated flexible wedge gate valves, and level instrumentation inaccuracies caused by reference level changes during rapid depressurization.

In addition to these programs and as part of NRC's overall international program, AEOD exchanges information and ideas on a variety of topics of international interest. For example, the AEOD staff provided assistance to foreign countries and to the IAEA in a number of safety-related areas, supported working group sessions of the U.S./C.I.S. Joint Coordinating Committee for Civilian Nuclear Reactor Safety, and served as the principal U.S. technical representative on reactor operating experience to NEA's Committee on the Safety of Nuclear Installations Principal Working Group 1 (PWG-1), "Operating Experience and Human Factors."

Summary

Through review, analysis, and evaluation of domestic and foreign events, AEOD continues to identify and communicate lessons learned from operating plant experience. In the last few years, industry performance has improved significantly; however, it appears that on a nationwide basis, performance indicators have stabilized, although additional effort is warranted among the poorer performers.

1 Introduction

The Office for Analysis and Evaluation of Operational Data (AEOD) was created in 1979 to provide, as one of its primary roles, a strong, independent capability to analyze operational data. This role was strengthened and expanded in 1987 in accordance with the Commission's emphasis on operational safety matters. In addition to its primary role of independent analysis of operational events, AEOD manages the review, analysis, and evaluation of reactor plant safety performance. In May 1987, AEOD also became responsible for the U.S. Nuclear Regulatory Commission's (NRC's) Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and Incident Investigation Program. In addition to managing these programs, AEOD provides administrative and technical support for the NRC Committee To Review Generic Requirements, which is responsible for recommending to the Executive Director for Operations the disposition of proposed new regulations, requests for information, and new requirements to be applied to power reactors. This report summarizes AEOD activities, except for those of the Technical Training Center and analysis of nonreactor licensee performance, which are summarized in separate reports.

AEOD consists of two divisions: the Division of Operational Assessment, which includes the Incident Response Branch, the Diagnostic Evaluation and Incident Investigation Branch, and the Technical Training Center; and the Division of Safety Programs, which includes the Reactor Operations Analysis Branch, the Trends and Patterns Analysis Branch, and the Nonreactor Assessment Staff. AEOD reports directly to the Executive Director for Operations.

AEOD reviews and evaluates operating experience to identify (1) significant events and their associated safety concerns and root causes, (2) trends and patterns displayed by these significant events, (3) the adequacy of corrective actions taken to address the safety concerns, and (4) generic applicability of these events and concerns to other nuclear power plants. These reviews and evaluations include the following specific functions:

- analysis of operational safety data associated with all NRC-licensed activities and identification of safety issues that require new or additional NRC staff actions
- implementation of the agency program on reactor performance indicators for use by senior managers
- development and implementation of diagnostic evaluations of licensee performance and direction of the diagnostic evaluation teams
- development of policy, procedures, and program requirements, and procedures for NRC incident investigations of significant operational events
- identification of operational data needed to support safety analyses and development of agencywide reporting of operational data and the methods and systems to retrieve these data
- analysis of selected operating events using the Accident Sequence Precursor Program to gain insight into significant events and improve understanding of them from a risk perspective
- conduct of studies of the impact of human performance during selected power reactor events
- development of a coordinated system for feedback of operational safety information to NRC offices, licensees, and other organizations, as appropriate
- preparation of the quarterly Abnormal Occurrence Reports to Congress
- development, in consultation with other NRC offices, of the NRC policy for response to incidents and emergencies, as well as assessment of the NRC response capabilities and performance
- tracking of the recommendations and staff actions contained in the AEOD studies and incident team reports until they are resolved
- development of an agencywide technical qualification and training program for a broad range of technical positions within the NRC staff and operation of the NRC's Technical Training Center at Chattanooga, Tennessee, to provide

the technical training needed by NRC personnel

- continuous staffing of the NRC Operations Center to screen reactor and nonreactor events and any other information reported to the center to ensure prompt and appropriate NRC reaction to reported events and incidents
- serving as a focal point for coordinating generic operational safety information and data systems with industry, foreign governments, and other agencies involved with the collection, analysis, and feedback of operational data The 1992 AEOD Annual Report is published in two separate parts: Power Reactors and Nonreactors. The report on power reactors, Vol. 7, No. 1, presents an overview of the operational experience of the nuclear power industry from the NRC perspective, including the trends of some key performance measures. This report also includes the principal findings and issues identified in AEOD studies over the past year and a summary of information from licensee event reports, diagnostic evaluations, and the NRC Operations Center. The report includes appendices as follows:

- Appendix A contains data to support the section on operational experience.
- Appendix B lists and summarizes abnormal occurrences in 1992.
- Appendix C lists AEOD reports issued in 1992.
- Appendix D lists AEOD reports issued from 1980 through 1991.
- Appendix E presents the status of outstanding recommendations included in AEOD studies.
- Appendix F presents the status of NRC staff actions resulting from the findings of NRC incident investigation teams.
- Appendix G presents the status of NRC staff actions involving potential generic issues resulting from the findings of NRC diagnostic evaluation teams.

The report on nonreactors, Vol. 7, No. 2, presents a review of the events and concerns during 1992 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations.

2 Operating Experience Feedback

2.1 Operating Performance

AEOD collects, analyzes, and disseminates a wide range of operational data. A subset of this information (Figure 2.1) has been selected for quarterly review in the NRC Performance Indicator (PI) Program. Enhancements to the program are discussed in Section 4.1. The quarterly operational data collected within the PI Program are presented in Appendix A-1 and other plant operational experience data in Appendix A-2.

The selected industry trends, discussed in this section, were developed from analyzing operational experience data from 1988 through 1992. Figure 2.1 presents the industry averages over the last 5 years for seven specific types of data that AEOD monitors as indicators of plant performance.

The industrywide averages in this section have excluded plants that either (1) have ceased commercial operation or (2) were in extended shutdowns.

2.1.1 Reactor Scrams

Reactor scrams can result from initiating events that range from relatively minor incidents to events that are precursors of accidents. Total reactor scrams (automatic and manual) decreased from 274 in 1988 to 196 in 1991, but were essentially flat for the last 2 years. A total of 195 scrams occurred in 1992, one less than in 1991. The quarterly automatic scram data for each plant during 1991 and 1992 are given in Table A-1.1 of Appendix A-1. The total number of automatic and manual scrams and the scram rate while the reactor was critical for 1988 through 1992 for each plant are listed in Table A-2.1 of Appendix A-2. The combined automatic and manual reactor scram data are summarized in Tables A-2.2 through A-2.5 of Appendix A-2.

Figure 2.2 shows the industry trend in scram causes for the 5-year period. Equipment failures have remained the leading cause of all scrams. The total number of scrams due to equipment failure has remained relatively constant since 1989, ranging from 135 to 126, though the percentage of this cause has increased from 56 to 66 percent as scrams from other causes have decreased. Of scrams caused by equipment failures at operating plants during 1992, 58 percent were initiated by problems in three systems: feedwater (24 percent), main turbine and control (21 percent), and electrical (13 percent). These three systems have been the most significant contributors to scrams.

Figure 2.3 is a pie chart of 1992 operating experience which shows that 50 percent of all scrams occurred during normal plant operation, 16 percent occurred during maintenance activities, and 18 percent occurred during testing while at power.

In summary, 66 percent of all scrams occurring at operating plants during 1992 were caused by equipment failures. Problems in the feedwater system continued to initiate the greatest number of scrams occurring at operating plants.

2.1.1.1 Automatic Reactor Scrams

Though the average number of automatic scrams decreased from 2.3 in 1988 to 1.4 in 1992, the last two annual industry averages for this indicator appear to be leveling off. For the 5-year period, an average of 52 percent of automatic scrams occurred during steady-state plant operations, while 33 percent occurred during maintenance and testing. These two factors remained relatively constant during the period. Automatic scrams during maintenance and testing caused by procedural problems decreased from 21 percent to 4 percent during the period.

2.1.1.2 Manual Reactor Scrams

Manual scram averages were relatively constant for the 5-year period, with an annual average of 44. An average of 56 percent of manual scrams occurred during steady-state plant operations, while 21 percent occurred during maintenance and testing.

Personnel error as a cause of manual reactor scrams decreased slightly from 24 percent in 1998 to 20 percent in 1992. Manual scrams during maintenance and testing caused by personnel errors increased from 13 percent in 1988 to 40 percent in 1992. Personnel errors as a cause of manual scrams during power changes dropped from 25 percent in 1988 to 0 percent in 1992, with an average for the 5-year period of 14 percent.

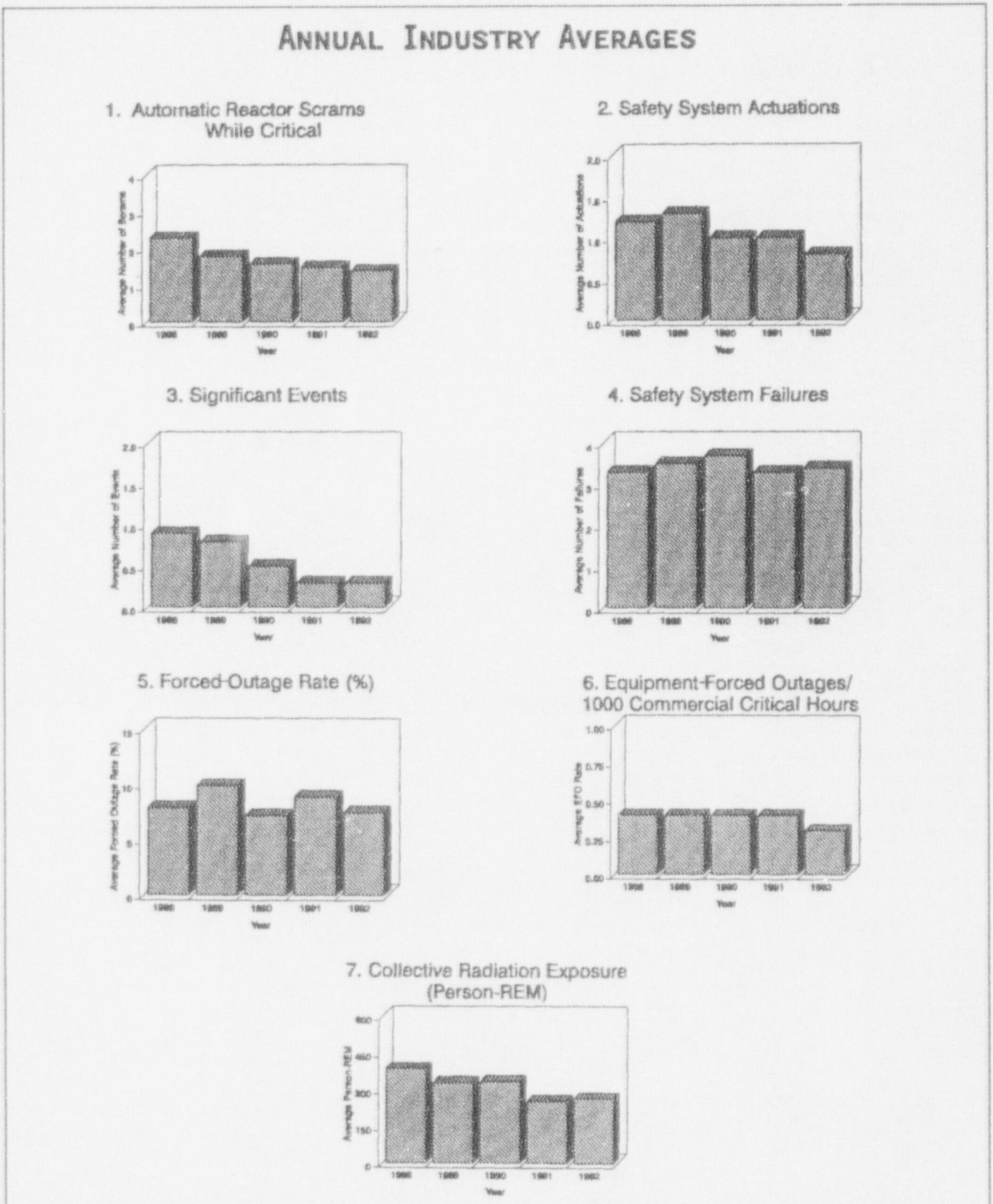

Figure 2.1 Annual Industry Performance Averages by Calendar Year

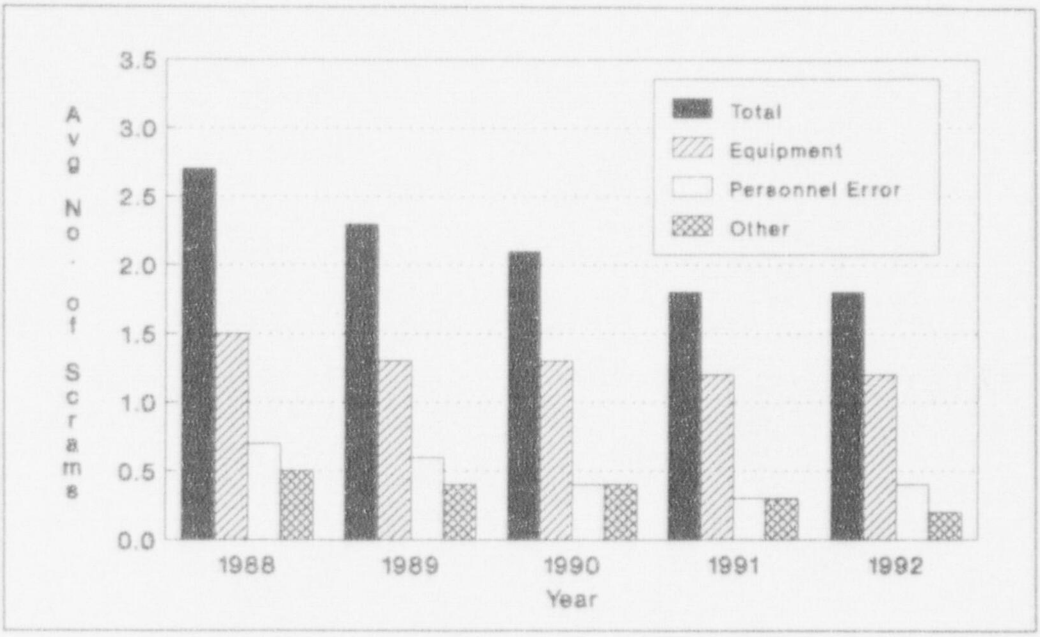

Figure 2.2 Scram Causes—Industry Trend

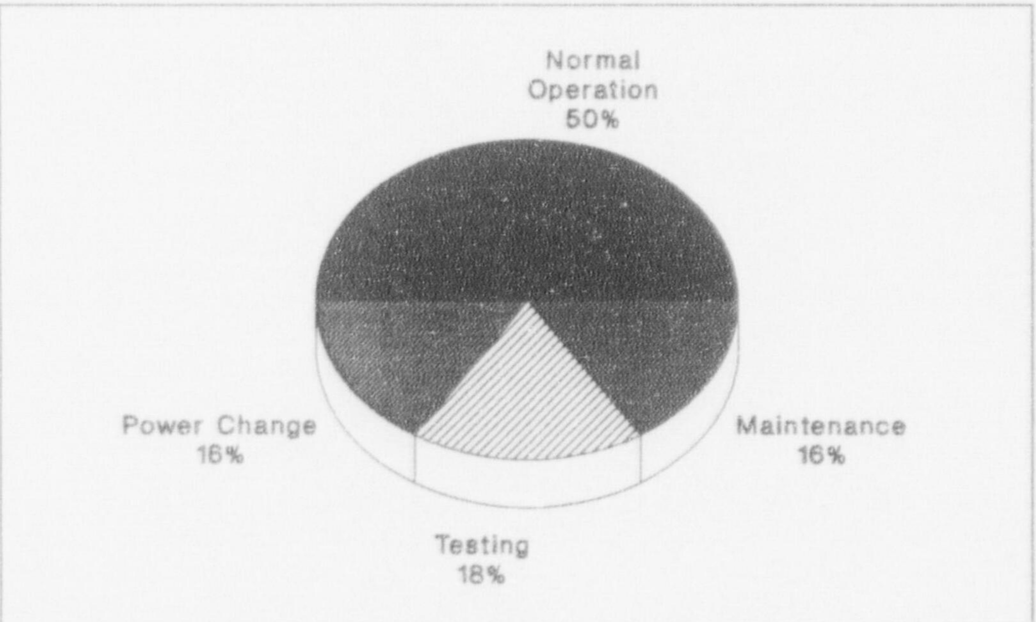

Figure 2.3 1992 Operating Experience Reactor Scram—Activity in Progress

Procedural problems causing manual scrams decreased from 14 percent to 2 percent during the 5-year period. Procedural problems resulting in manual scrams during steady-state operations dropped from 9 percent to 0 percent, during maintenance and testing from 37 percent to 0 percent (average 12 percent), but rose during power changes from 6 percent to 14 percent (average 10 percent).

2.1.2 Engineered Safety Features Actuations

In 1992, the industry experienced a total of 556 valid actuations for all engineered safety features (ESFs), of which 89 actuations were considered as safety system actuations in the PI Program. Data for safety system actuations may be found in Table A-1.2 in Appendix A-1. The total of 556 actuations represented a 22 percent decrease from that in 1991. The most significant decrease (384 to 291) occurred in General Electric (GE) plants. Actuations in Westinghouse plants decreased from 274 in 1991 to 216 in 1992. Combustion Engineering (CE) and Babcock & Wilcox Co. plants experienced smaller decreases in the number of actuations between 1991 and 1992 (10 and 9, respectively). Industry data for all ESF actuations are given in Appendix A-2, Tables A-2.6 through A-2.9.

Figure 2.4 shows the industry trend in all ESF actuations for the 5-year period, as well as actuations of heating, ventilation, and air conditioning (HVAC) systems, emergency power (EP) systems, and emergency core cooling systems (ECCSs). The downward trend of total ESF actuations for the period has been driven primarily by a reduction in HVAC actuations.

Figure 2.5 is a pie chart that reflects 1992 operating experience showing the causes for actuations of HVAC, EP, and ECCSs. "Maintenance" was the most significant contributor (48 percent), followed by "other [than licensed] personnel" (16 percent) and "administrative" (15 percent).

Industry ESF actuation data classified according to the four NSSS vendors are shown in Figures 2.6 through 2.9. Because of a variety of plant-specific operational characteristics, these data should not be used to imply relative safety performance between NSSS vendor designs.

In Figures 2.6 through 2.9, for each NSSS vendor, the "Total" plot includes all ESF events except scrams. Figure 2.10 is a compilation of the individual "Total" plots. The "Valid" events, which are a subset of the "Total," were initiated by a signal from the measurement of a physical parameter that exceeded a setpoint and provided input to the ESF actuation logic in response to an actual plant condition. All other events are considered unnecessary challenges to the safety systems.

The EP and ECCS actuation plots are subsets of the "Total" ESF events. They show the trends of all unplanned actuations of these systems; that is, these plots include actuations beyond those counted in the PI Program for those systems. The plot for HVAC shows the trend in the actuation of those HVAC systems that have a safety function. These HVAC system actuations dominate the reported ESF events. The plot labeled "Other Than ECCS, Emergency Power, and HVAC" accounts for actuations of ESFs other than those shown in their own plots and contains contributions from various systems, such as the auxiliary feedwater system. Reporting of certain HVAC actuations was eliminated in September 1992, which will affect the number of future reports.

For GE plants, an additional plot is provided for the reactor water cleanup (RWCU) system, which is unique to boiling-water reactor designs. As shown in Figure 2.8, the isolation of this system accounts for a significant percentage of ESF events. The number of isolations of the RWCU system in GE plants declined steadily from 1988 to 1990, increased slightly in 1991, and in 1992 returned to the 1990 level. Reporting of invalid RWCU isolations was eliminated in September 1992, which will affect the number of future reports.

The average number of valid ESF actuations per plant per year may have leveled off in the range of 2.1 to 1.7. The number of plants without valid ESF actuations has remained relatively constant during the 5-year period, ranging from 22 to 33 annually.

During the 5-year period, 56 percent of the ESF actuations occurred during plant operations above 15-percent power, while 27 percent occurred during cold-shutdown and refueling modes, and 10 percent occurred during the hot-shutdown mode. The most significant cause of ESF actuations has been

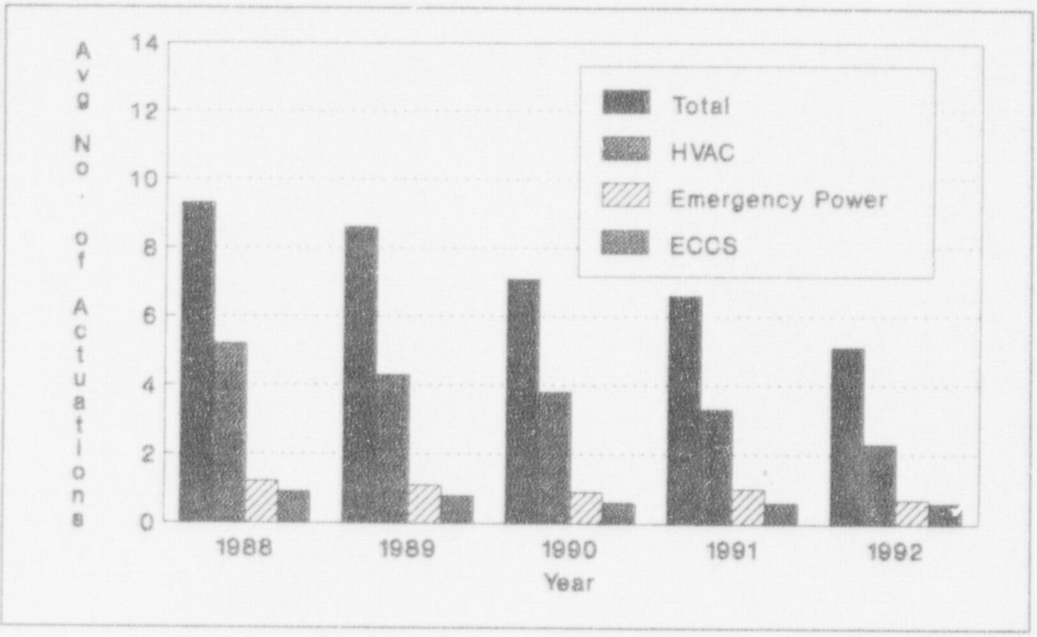

Figure 2.4 All ESF Actuations—Industry Trend

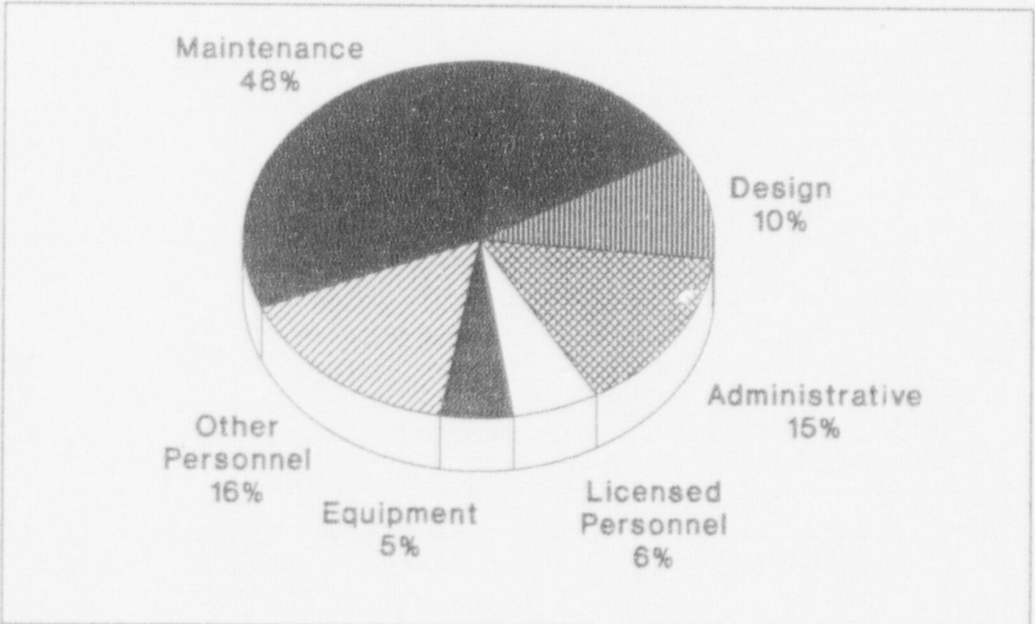

Figure 2.5 1992 Operating Experience—Causes of ESF Actuations

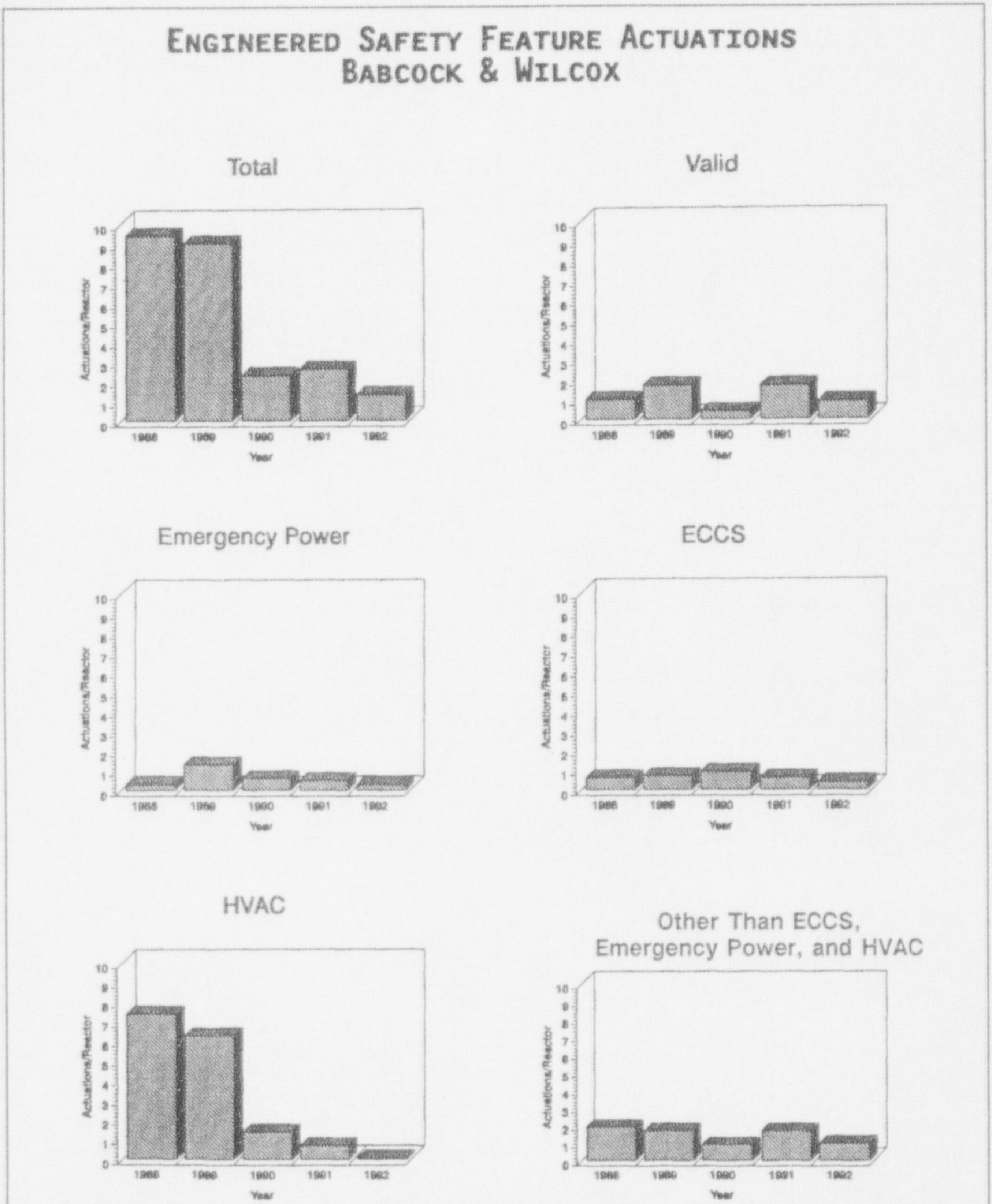

Figure 2.6 Babcock & Wilcox Reactor Engineered Safety Feature Actuations by Calendar Year

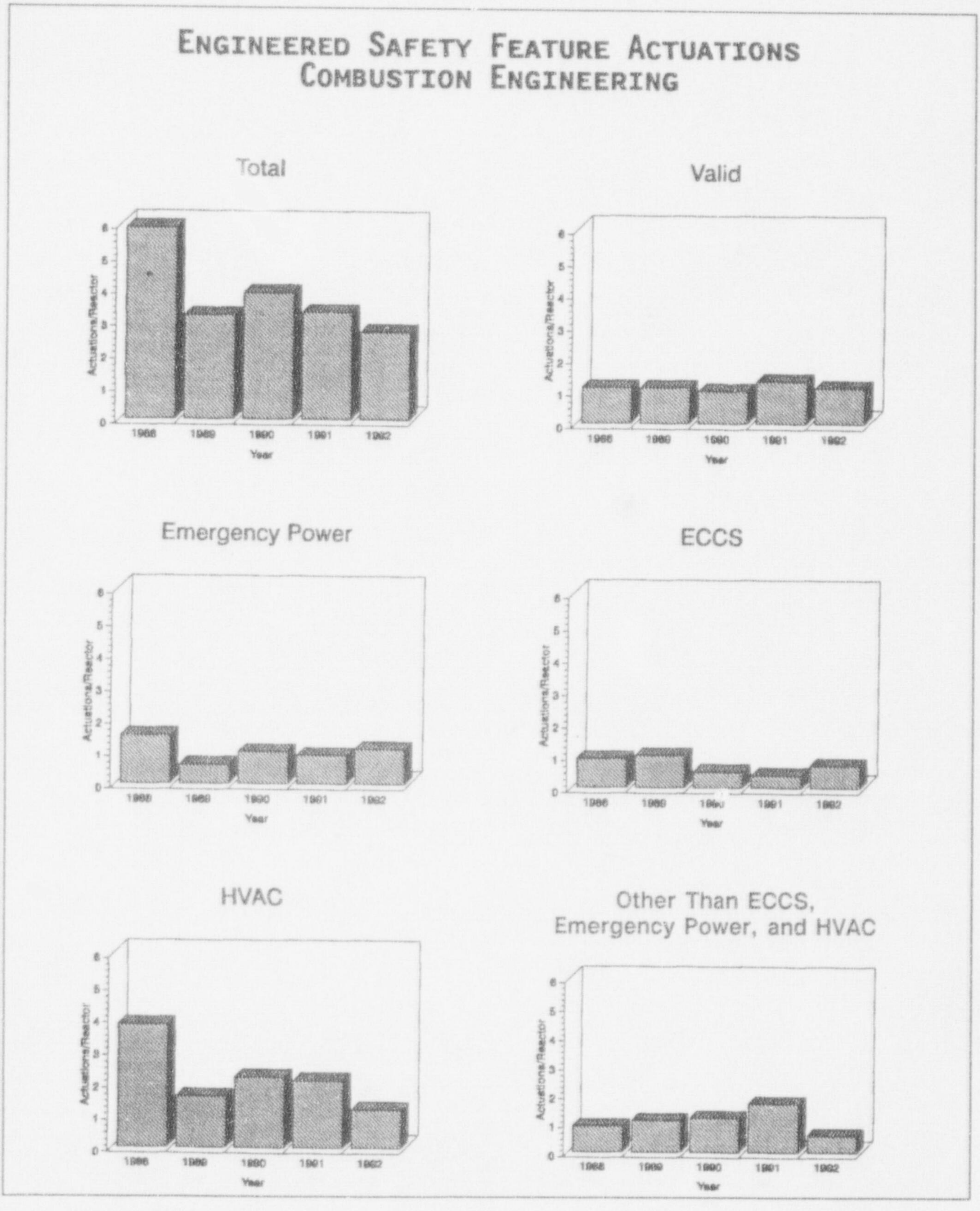

Figure 2.7 Combustion Engineering Reactor Engineered Safety Feature Actuations by Calendar Year

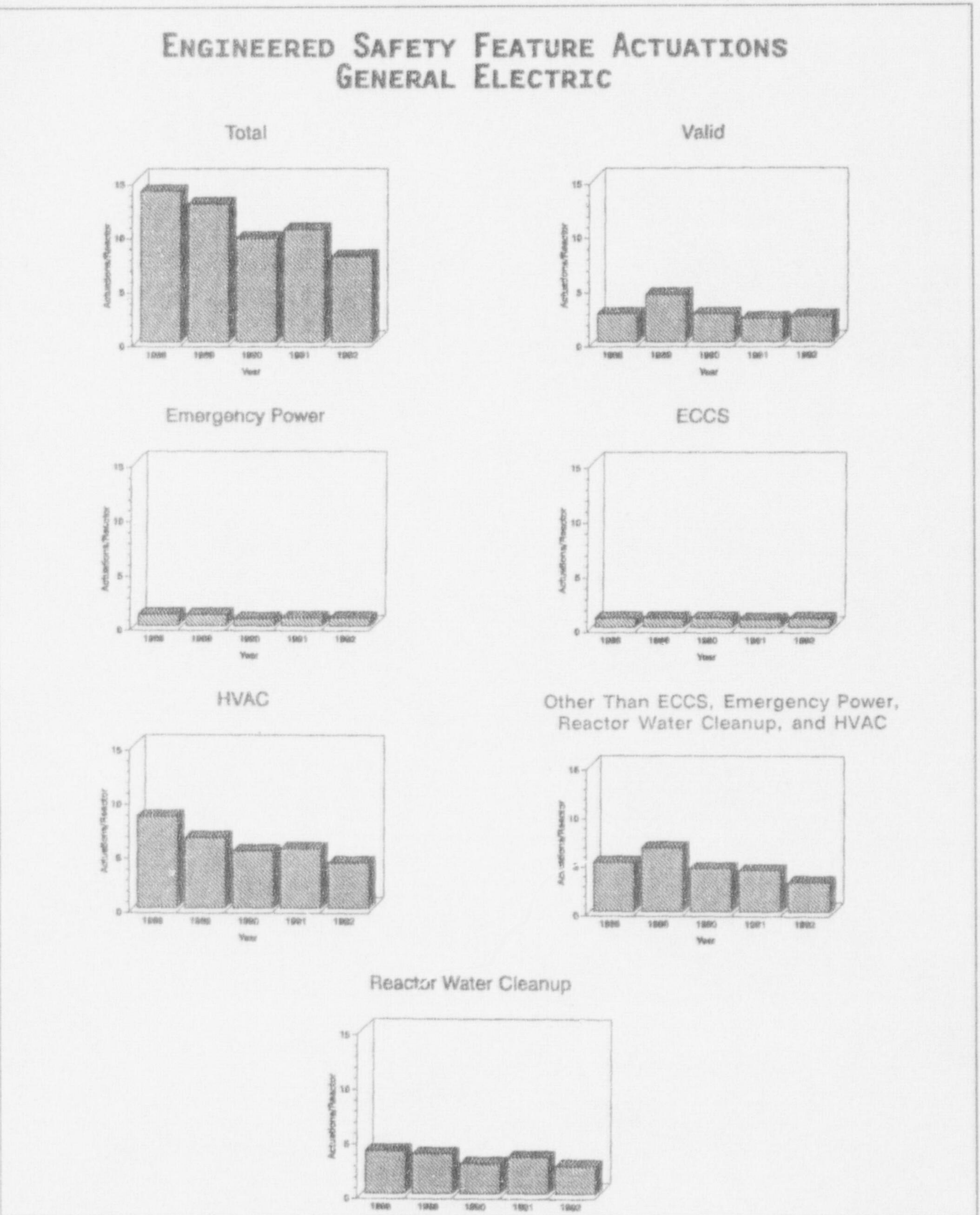

Figure 2.8 General Electric Reactor Engineered Safety Feature Actuations by Calendar Year

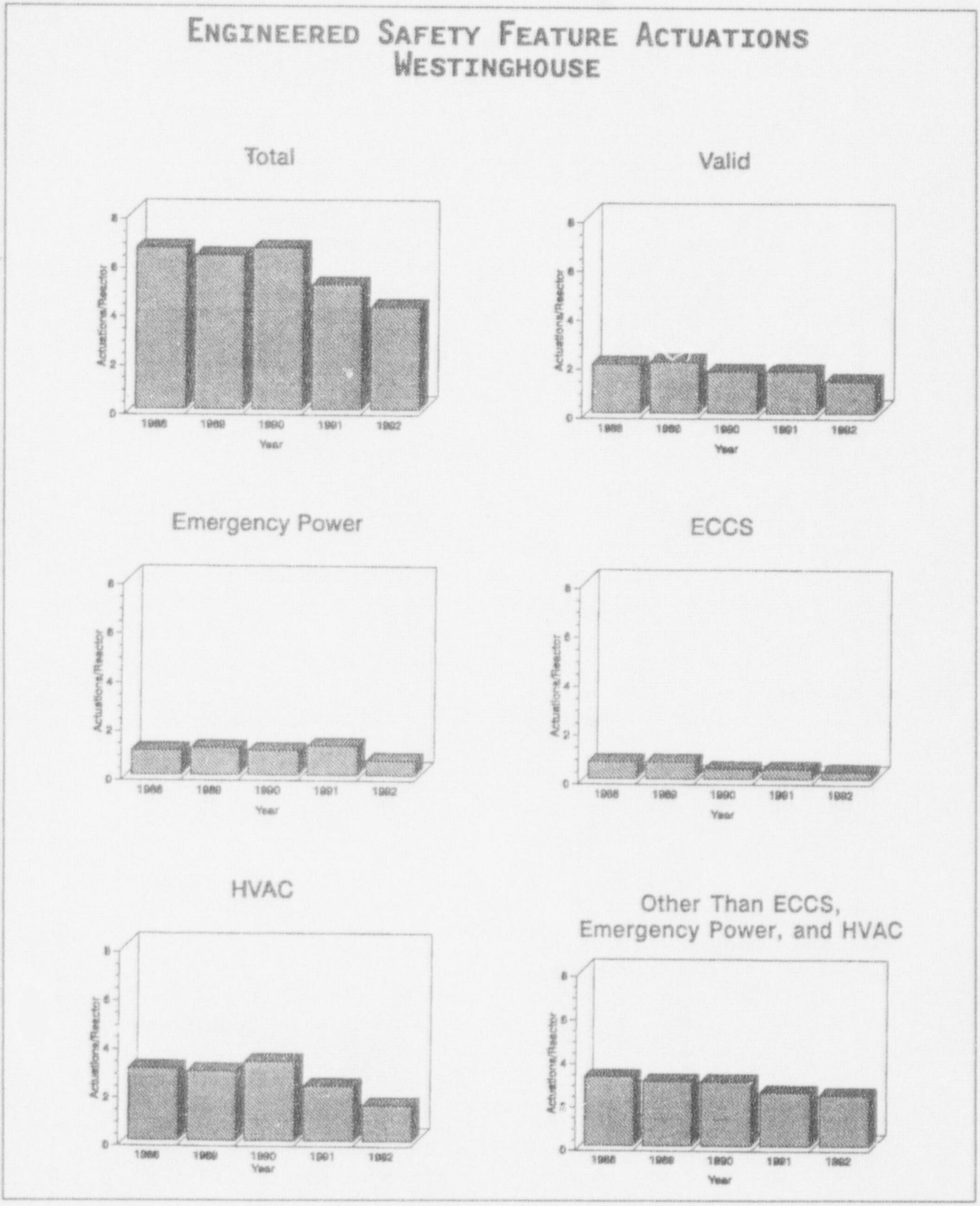

Figure 2.9 Westinghouse Reactor Engineered Safety Feature Actuations by Calendar Year

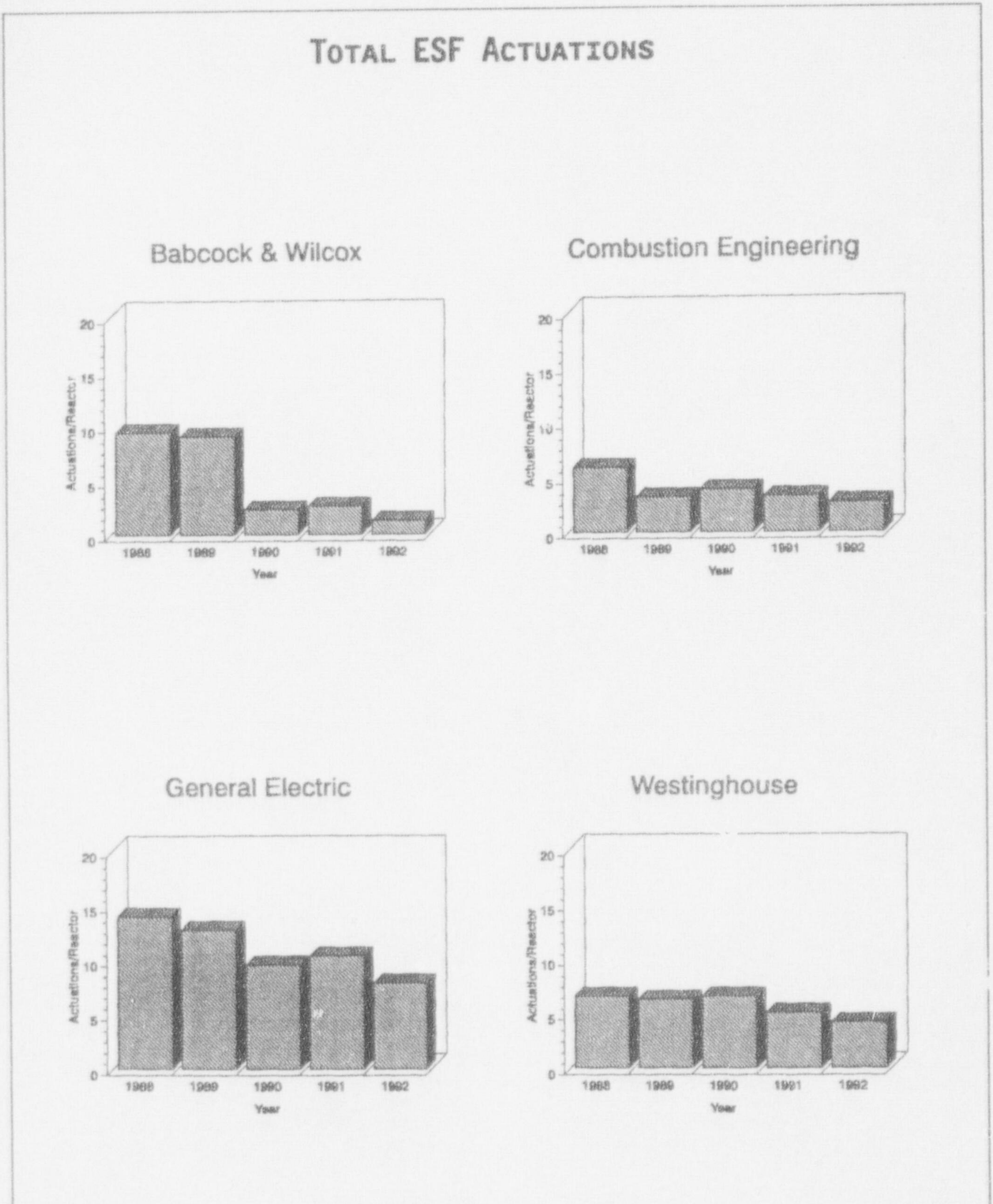

Figure 2.10 Total Reactor Engineered Safety Feature Actuations by Calendar Year

maintenance (relatively constant with an average of 40 percent for the 5-year period), administrative problems (relatively constant with an average of 21 percent), and errors by other than licensed operators (relatively constant with an average of 16 percent). Design, licensed operator errors, and equipment failures averaged 11 percent, 9 percent, and 3 percent, respectively, for the period.

2.1.3 Significant Events

Significant events are those events that the NRC staff identifies as meeting certain selection criteria. Significant events typically involve one or more of the following selection criteria: (1) the degradation of important safety equipment; (2) an unexpected plant response to a transient or major transient itself; (3) a degradation of fuel integrity, the primary coolant pressure boundary, or important associated structures; (4) a reactor trip with complications; (5) an unplanned release of radioactivity exceeding plant Technical Specifications (TS) or regulations; (6) operation outside the TS limits; and (7) other events that are considered significant.

Figure 2.1 shows that the trend in the average number of significant events per plant decreased from 1988 through 1991, but remained essentially constant for 1992. Actual events decreased from 87 in 1988, to 30 in 1991, and 28 in 1992, of which five events involved a loss of offsite power. The trend may indicate that it is still improving, but at a lesser rate.

Table A–1.3 of Appendix A–1 provides a list of the significant events for 1992. Table A–1.4 of Appendix A–1 gives plant-specific quarterly data for significant events for 1991 and 1992.

2.1.4 Safety System Failures

The safety system failure (SSF) indicator includes any event or condition that could prevent the fulfillment of the safety function of structures or systems. For this indicator, 26 safety systems, subsystems, and component groups are monitored. Unsatisfactory conditions in these systems, subsystems, or components are generally identified during testing, special inspections, and engineering design reviews rather than on actual demand to operate. For a system that consists of multiple redundant subsystems or trains, inoperability of all trains constitutes an SSF. Safety system failures can be indicative of a plant's readiness to respond to anticipated events and postulated accidents.

Figure 2.1 shows that the industry average number of SSFs per plant has remained essentially constant during the last 5 years. Figure 2.11 shows SSFs broken down by the number of actual and conditional SSFs. Actual SSFs are those failures that challenged the system. Conditional SSFs include those identified by industry design reviews, including those performed during design-basis reconstitution programs. For the 5-year period, the ratio of actual to conditional SSFs has remained essentially constant. Figure 2.12 contains charts of 1992 operating experience that show the causes of SSFs as percentages of actual and conditional SSFs. The three most significant contributors to actual SSFs were maintenance (41 percent), administration (24 percent), and design (17 percent). The three most significant contributors to conditional SSFs were design (37 percent), maintenance (28 percent), and administration (22 percent).

Table A–1.5 of Appendix A–1 provides the quarterly plant-specific data for SSFs for 1991 and 1992. The annual number of SSFs for each plant from 1988 through 1992 is given in Table A–1.6 of Appendix A–1.

Four system groups have been the dominant contributors to SSFs during the 5-year period, accounting for over 60 percent of SSF events. For 1992, the contributions of these groups were as follows: the emergency power systems group (17 percent), the emergency core cooling systems group (15 percent), the containment and containment isolation group (15 percent), and the control room emergency ventilation group (15 percent).

Figure 2.13 shows the number of SSF events per plant for each NSSS vendor. These figures do not indicate any definite trend.

2.1.5 Forced-Outage Rate

The forced-outage rate is the number of forced-outage hours in a period divided by the sum of the unit service hours; that is, generator online hours plus the forced-outage hours. For performance monitoring purposes, forced outages are defined as those outages required to be initiated by the end of

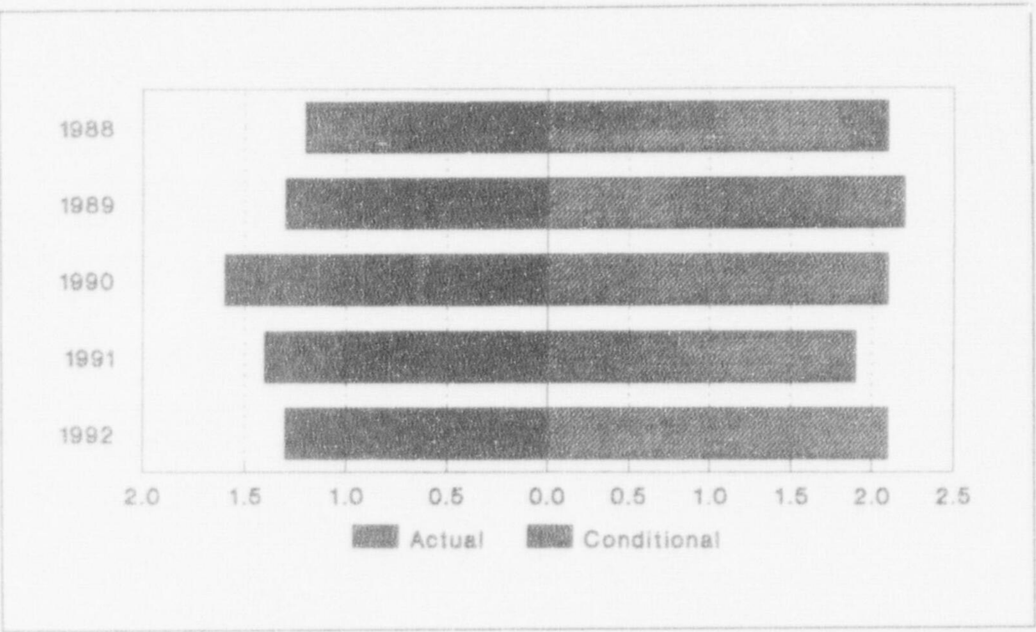

Figure 2.11 Safety System Failures—Industry Trend

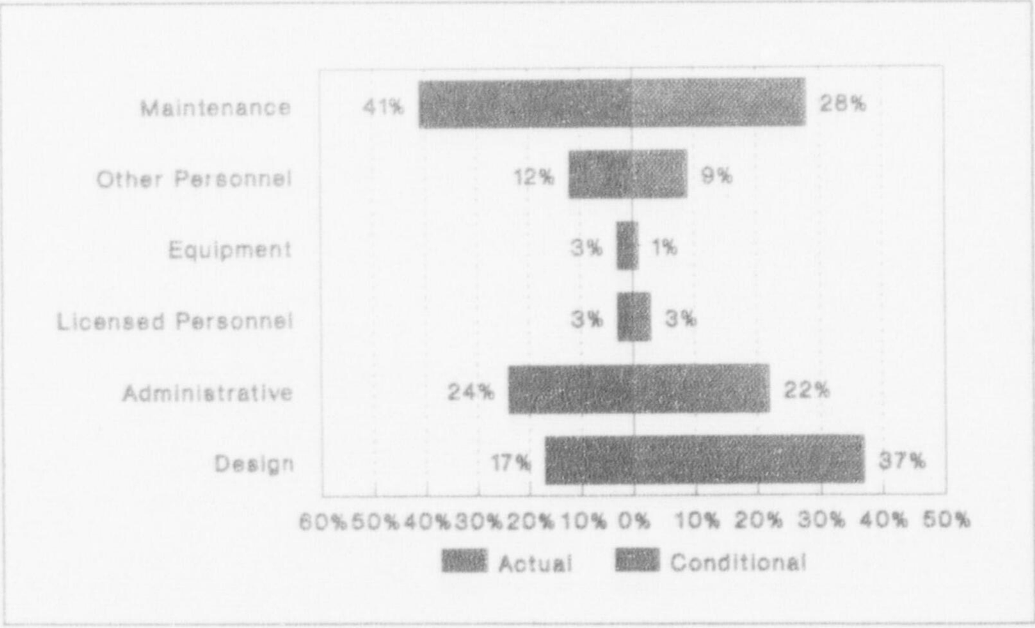

Figure 2.12 1992 Operating Experience—Causes of Safety System Failures

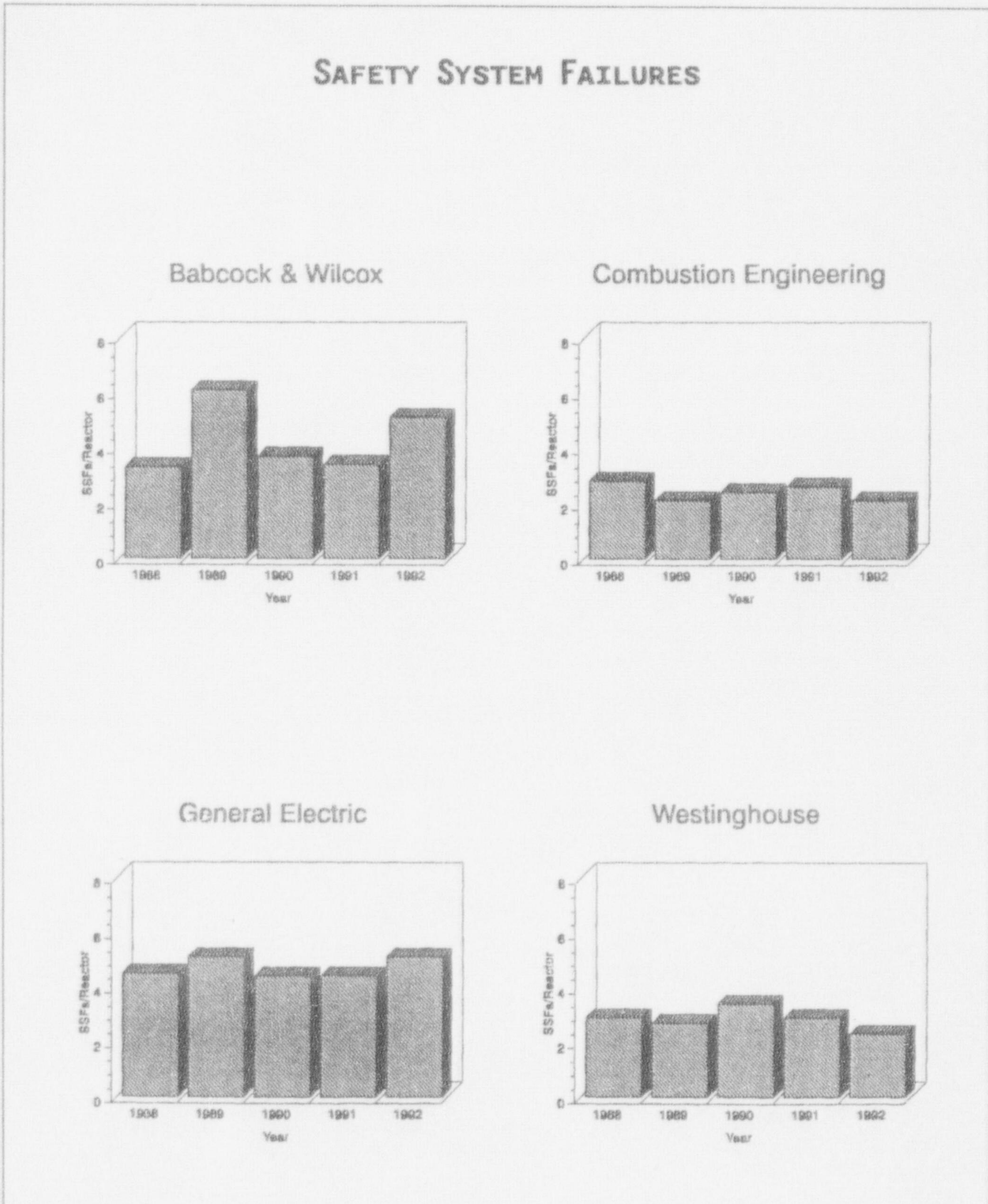

Figure 2.13 Safety System Failures by Calendar Year

the weekend following the discovery of an off-normal condition. The trend in forced-outage rate can provide a perspective on overall plant operating performance. The forced-outage rate, as shown in Figure 2.1 has been erratic; it increased from 1990 to 1991 because of large increases in forced-outage hours at several plants, but dropped slightly in 1992. Overall, the forced-outage rate remains within the range, 7.2 percent to 9.9 percent, that has been seen for the past 5 years. Plant-specific quarterly data are provided in Table A–1.7 of Appendix A–1 for 1991 and 1992.

2.1.6 Equipment-Forced Outages per 1,000 Commercial Critical Hours

The equipment-forced-outage (EFO) indicator is a measure of the number of forced outages caused by equipment failures per 1,000 hours of commercial operation while the reactor is critical. The EFO rate is the inverse of the mean time between forced outages caused by equipment failures. AEOD monitors the EFO rate as an indicator of the effects of equipment problems on overall plant performance.

Figure 2.1 shows that the industry average EFO rate remained relatively constant from 1988 through 1991, and improved in 1992. Table A–1.8 of Appendix A–1 contains the quarterly data for the EFO rate for 1991 and 1992 for each plant.

2.1.7 Cause Codes

The cause codes indicator is intended to identify possible programmatic deficiencies. Cause code trends are developed from data in the Sequence Coding and Search System database. This database includes the causes associated with each event that a licensee reports as required by the regulations in Title 10, Section 50.73 of the *Code of Federal Regulations* (10 CFR 50.73).

Industry averages for this indicator are not calculated. The quarterly cause code data used in the PI Program for each plant for 1991 and 1992 are shown in Tables A–1.10 through A–1.15 of Appendix A–1. The cause code indicator captures the trends for administrative control problems (Table A–1.10); licensed operator errors (Table A–1.11); other personnel errors (Table A–1.12); maintenance problems (Table A–1.13); design, construction, fabrication, or installation problems (Table A–1.14); and equipment (electronic piece part or environmentally related) failures (Table A–1.15).

2.1.8 Power Production Performance

Within the context of its safety mission, the NRC is not generally concerned with the power production performance of the nuclear industry. However, because producing power requires attention to detail and good management, sustained reliable power production can be a positive indicator of safe performance. Power production statistics for the U.S. commercial nuclear industry are presented in Tables A.2–10 and A.2–11 of Appendix A–2. The industry average unit availability factor has risen from 69.8 percent in 1988 to 74.1 percent in 1992.

2.1.9 Statistical Analysis of Some Trends

As part of an assessment of trends in performance indicators (PIs) through 1992, AEOD performed a statistical analysis to evaluate the trends in PIs. Of particular interest was the rate of change (improvement) that could be detected. The PIs considered in this analysis were (1) automatic reactor trips, (2) safety system actuations, (3) significant events, (4) safety system failures, (5) forced-outage rate, and (6) equipment-forced-outage rate.

The four quarterly PIs for a plant were summed to obtain a single number for each year from 1988 through 1992. In the analysis, it was assumed that a plant operated for the entire year, even though it may have a value only for a single quarter. The analysis was performed as follows. First, an exponential model, $y = Ae^{Bx} + C$ was fit to each PI. In this model, y is the PI value, x is a time increment index (with 1988 = 1), and A, B and C are parameters estimated from the data. Statistical tests were then performed to determine if the estimated parameters, A, B, and C were non-zero. If all statistical tests were rejected, the nonlinear model was assumed to fit the data.

If the tests of the hypothesis for a non-zero value for B and C was not rejected, a linear regression model was fit to the data and the hypothesis of zero slope was tested. If this hypothesis was rejected, the linear regression was assumed to fit the data. If this last hypothesis was not rejected, it was assumed that the trend was constant over the 5-year period. In all cases, confidence intervals on the mean were calculated and plotted in the figures.

The results of the regression analyses are summarized in Table 2.1. Automatic reactor trips, safety system actuations, significant events, and equipment forced outage rate have statistically significant exponential model fits, which indicates that a trend is discernible. Figure 2.14 shows the trend for automatic reactor trips. The plots for the other three PIs are similar. Safety system failures and forced-outage rate do not have a nonlinear or linear trend over the 5-year period. That is, they are constant (level) over the time of interest.

Table 2.1
Summary of Regression Analysis Trends for PIs

Performance Indicator	1988–1992
Automatic reactor trips	Slowly decreasing nonlinear trend
Safety system actuations	Slowly decreasing nonlinear trend
Safety system failures	Level
Forced-outage rate	Level
Equipment-forced-outage rate	Slowly decreasing nonlinear trend
Significant events	Slowly decreasing nonlinear trend

Table 2.2 contains the estimates of *A*, *B*, and *C* for the exponential model. Only automatic reactor trips has a value of *C* significantly different from 0. The estimate of *B* for these PIs is negative, indicating a decreasing trend. The percent decrease from one year to the next is about 35 percent for significant events, 25 percent for reactor trips, and about 10 percent for the other two PIs. Reactor trips are asymptotically approaching a value of 1.28.

Table 2.2 Nonlinear Regression Results

Performance Indicator	Estimate of *A*	*B*	*C*
Automatic reactor trips	1.75	–0.538	1.28
Safety system actuations	2.00	–0.128	–
Significant events	3.72	–0.351	–
Equipment-forced-outage rate	0.68	–0.096	–

2.1.10 Summary

In regard to the apparent stabilizing of PIs, AEOD performed additional analyses of the PI data. Evaluations of these data over the most recent years provided additional support for the assertion that the PIs were leveling off. On the basis of these comparative evaluations and the additional PI analysis, it appears that, on a nationwide basis, PIs have stabilized, although additional effort is warranted among the poorer performers.

2.2 Radiation Exposures From Reactors and Nonreactors

2.2.1 Sources of Radiation Exposure

According to the National Council on Radiation Protection and Measurements, the total average effective dose equivalent to a person in the United States is approximately 3.6 millisieverts (mSv) (360 millirem (mrem)) per year. The average person in the United States receives an effective dose equivalent of about 0.5 mSv (50 mrem) per year from medical applications. The whole fuel cycle, including operation of reactors, contributes less than 0.01 mSv (1 mrem) per year. All the other human-controlled sources of radiation combined add up to an effective dose equivalent of approximately 0.06 mSv (6 mrem) per year.

Almost all of the radiation dose from nuclear power plants is occupational dose, that is, the dose to the nuclear power plant employees and their contractors who work at the plant. Because the economics of operating a plant creates a strong impetus to lower exposures and achieve ALARA (As Low As Reasonably Achievable) objectives, utility violations of NRC limits on personnel exposure are rare, and the vast majority of nuclear power plant personnel have annual exposures far below NRC regulatory limits specified in 10 CFR Part 20. The actual mean value has been reduced from 1.9 in 1973 to 1.0 in 1985, and to 0.4 in 1991. The reduction is believed to be primarily the result of the licensees' extensive dose-reduction efforts. Some measures that reduce collective exposure are the licensees' efforts to have an effective maintenance program, experienced and well-trained personnel, a good water chemistry control program, effective decontamination and

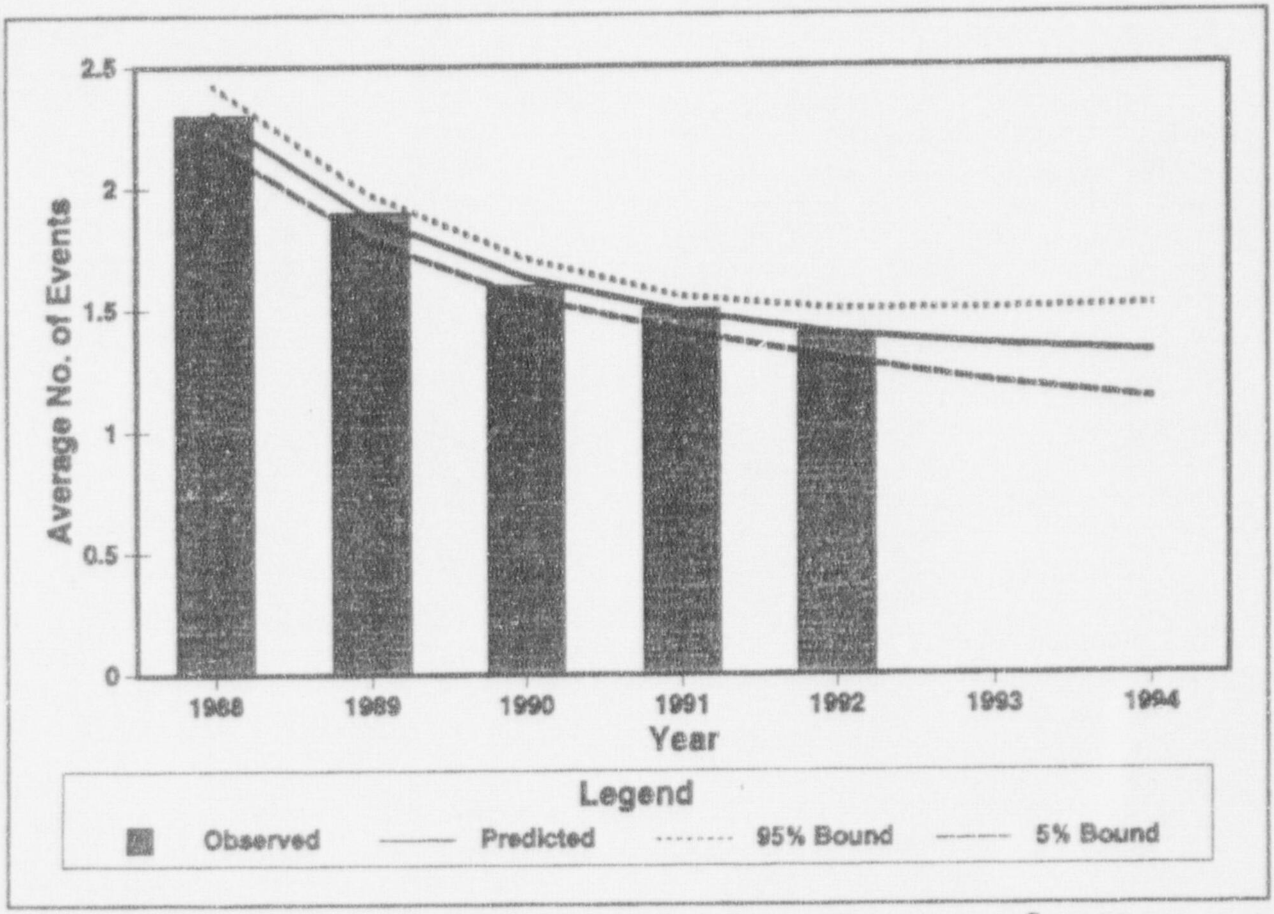

Figure 2.14 Nonlinear Regression Fit for Automatic Reactor Trips ($y = Ae^{Bx} + C$)

cleanup practices, good fuel cladding integrity, effective radiation exposure control programs, good housekeeping, and an alert health physics staff.

2.2.2 Exposures for Reactor and Nonreactor Applications

NRC regulates both reactor and nonreactor applications of nuclear materials. All NRC licensees are required to provide appropriate personnel monitoring equipment to, and require the use of such equipment by, each individual who receives or is likely to receive a dose in any calendar quarter in excess of 25 percent of the allowable limits specified in 10 CFR Part 20. Certain licensees, namely reactor operators and those involved with industrial radiography, manufacturing and distribution of radioactive materials, low-level radioactive waste disposal, and independent spent fuel storage installation and processing, are required to provide annual summaries of exposure data for individuals for whom personnel monitoring had been required. The summary of annual occupational exposure information reported by commercial light water cooled reactors from 1987 to 1991 is shown in Table 2.3. For purposes of comparison, 1973 has also been included.

Figure 2.1 shows that collective occupational radiation exposure reported by commercial light-water-cooled reactors declined from 1988 to 1991. Table A-1.9 of Appendix A-1 shows quarterly radiation exposure data for 1991 and 1992 provided by the Institute of Nuclear Power Operations (INPO) for each plant. While NRC receives radiation-exposure data on an annual basis, INPO routinely receives these data from the plants on a quarterly basis. AEOD uses the INPO data to provide more timely information, without duplicating their effort.

Table 2.3 lists the exposure data for licensee categories for 1991. Of the six categories of licensees that are required to report collective exposures for

Table 2.3 Summary of Annual Occupational Exposure Information Reported by Commercial Light-Water-Cooled Reactors*—1973 and 1987–1991

Year	No. of Reactors Included	Annual Collective Doses (person-rem or person-cSv	No. of Workers With Measurable Doses	Gross Electricity Generated (MW-yr)	Average Dose per worker (rem or cSv)	Average Collective Dose per Reactor (person-rem or person-cSv)	Average No. of Workers With Measurable Doses per Reactor	Average Collective Dose per MW-yr person-rem or person–cSv MW-yr)	Average Electricity Generated per Reactor (MW-yr)	Average Maximum Dependable Capacity Net (MWe)
1973	24	13,962	14,780	7,164.1	0.94	582	616	1.95	299	491
1987	96	40,401	104,334	52,116.3	0.39	421	1,087	0.8	543	877
1988	102	40,878	103,226	59,595.1	0.40	401	1,012	0.7	584	871
1989	107	35,929	108,254	62,223.0	0.33	336	1,012	0.6	582	883
1990	110	36,592	109,650	68,291.7	0.33	333	997	0.5	621	892
1991	111	28,527	98,045	73,448.4	0.29	257	883	0.4	662	895

*Only those reactors are included that had been in commercial operation for at least one full year as of December 31 of each of the indicated years, and all figures are uncorrected for multiple reporting of transient individuals. The Radiation Exposure Information Report System (REIRS) is funded by the Office of Nuclear Regulatory Research (RES).

monitored individuals, the 114 reactor licensees that reported (111 operating), by virtue of the large number of employees, had the highest collective exposure (28,527 cSv or rem to 184,829 people), followed by radiographers (2,160 cSv or rem to 6,820 people), manufacturers and distributors (721 cSv or rem to 4,930 people), and fuel fabrication licensees (272 cSv or rem to 11,016 people). Low-level waste disposal (39 cSv or rem to 905 people) and independent spent-fuel storage (4 cSv or rem to 41 people) licensees had relatively low collective doses.

Although worker occupational exposures have been maintained at a low level, a few overexposures continue to occur. Between 1987 and 1992, licensees reported 17 events of operational overexposure at nuclear power plants. Table 2.4 lists these data, which show that 19 individuals received exposures that exceeded the quarterly limits specified in 10 CFR Part 20.

The reported overexposures attributed to hot particles are identified in Table 2.4 with an asterisk. Hot-particle exposure is not an important contributor to the total occupational dose and has no discernible ill effects; however, doses of this type can exceed prescribed (10 CFR Part 20) exposure limits.

Members of the public may be exposed to radioactive effluents or to direct radiation from the plant. Even though these doses are trivial compared with doses from nature and medical applications, the effluents that contribute to these doses are regulated by the NRC. Exposures from the transportation of radioactive materials to and from the site are negligible in comparison to the doses from the effluents. Nonoccupational doses have declined relative to occupational doses. In 1975, nonoccupational collective exposures were approximately 6.5 percent of occupational doses. By 1990, the nonoccupational collective doses were less than 0.2 percent of occupational exposures. The calculated annual offsite dose commitments are reported annually in NUREG/CR-2850, No. 12, "Population Dose Commitments Due to Radioactive Releases From Nuclear Power Plant Sites in 1990."

2.2.3 Comparison of Overexposures for Reactor and Nonreactor Applications

Overexposures in nonreactor applications are discussed in the AEOD Annual Report, "Nonreactors," NUREG-1272, Vol. 7, No. 2. A summary of the data on the number of reports from, and the number of individuals overexposed in, NRC licensed facilities for reactors and nonreactors for the years 1987 through 1991 is given in Table 2.5. Data for Agreement State licensees are not included in this table because they are not readily available. As can be seen, every year the number of events and the number of individuals overexposed in nonreactor applications exceeded those exposed at reactor sites.

All overexposures reported are occupational overexposures, which are violations of NRC regulations. Although radiation doses in excess of that prescribed for a patient do occur, they are not measured against any fixed exposure limit because NRC regulations do not limit the radiation doses to patients.

Data on the number of individuals with measurable exposures are not readily available for all groups of NRC and Agreement State nonreactors; but they are available for NRC licensed radiographers, the licensee category having the largest number of overexposures of employees. The number of overexposures and the number of workers with measurable doses for reactors and NRC-licensed radiographers are shown in Table 2.6. As can be seen, the rate of overexposures of radiographers is greater by more than a factor of 10 than that for personnel working at a reactor site. The special radiological problems of industrial radiography have been recognized for a long time. The NRC has provided a special guidance and training document, NUREG/CR-0024, "Working Safely in Gamma Radiography," for radiographers for the purpose of reducing overexposures. In addition AEOD is preparing a videotape on good safety practices in industrial radiography. The tape is entitled "Taking Control: Safety Procedures for Industrial Radiography" and is scheduled to be completed in 1993.

Data are also available for fuel-fabrication and processing licensees. These categories of licensees reported relatively few overexposures (from 0 to 3 annually between 1987 and 1991), but had an exposure rate that, in general, exceeded that for reactors.

The number of overexposures reported annually by reactor, industrial radiography, and fuel-fabrication

Table 2.4 Annual Exposure Data for Certain Categories of NRC Licensees for 1991*

Category	No. of Licensees Reporting	No. of Monitored Individuals	No. of Workers With Measurable Doses	Collective Dose (person-rem or person-cSv)	Average Individual Dose (rem or cSv)	Average Measurable Dose per Worker (rem or cSv)
Reactors	114	184,829	93,519	28,527	0.15	0.29
Industrial radiography	248	6,820	4,649	2,160	0.31	0.46
Manufacturing & distribution	58	4,930	1,956	721	0.15	0.37
Low-level waste disposal	2	905	147	39	0.04	0.27
Independent spent fuel storage	2	41	24	4	0.10	0.17
Fuel fabrication & processing	11	11,702	3,929	378	0.03	0.10

*The Radiation Exposure Information Report System (REIRS) is funded by RES. Data for 1992 were not readily available when this report was published.

Table 2.5 Operational Overexposure Reports From Reactors—1987-1992

Date	Reactor	Type of Exposure	Exposure (cSv or rem)
02/13/87	J.A. FitzPatrick	Extremity	30
04/09/87	Trojan	Extremity	9.6
05/12/87	H.B. Robinson 1	Whole body	1.3
05/30/87	Yankee Rowe	Skin	10.5*
10/08/87	Crystal River 3	Whole body	1.8
05/16/88	Farley	Whole body	1.25
05/18/88	Waterford 3	Skin (2)	17.6, 22.2*
05/27/88	Surry 1	Whole body	3.3
11/02/88	Arkansas 1	Whole body	4.5
11/23/88	Arkansas 1	Skin	61*
12/09/88	Wolf Creek	Skin	12.5*
09/25/89	Three Mile Island 2	Extremity (2)	55, 13
03/08/90	J.A. FitzPatrick	Extremity	49
1991	None		
03/26/92	Wolf Creek	Skin	15.9*
03/28/92	North Anna 2	Skin	21.4*
04/01/92	Catawba 1	Skin	8.7*
12/21/92	Palo Verde 1	Whole body	1.9

*Dose from hot particle.

Table 2.6 Overexposure Events at Reactor and NRC Nonreactor Licensees—1987–1991

Type of Licensee	1987 No. of Reports	1987 No. of People	1988 No. of Reports	1988 No. of People	1989 No. of Reports	1989 No. of People	1990 No. of Reports	1990 No. of People	1991 No. of Reports	1991 No. of People
Reactors	5	5	6	7	1	2	1	1	0	0
Medical & academic	4	4	6	6	3	3	3	3	3	3
Radiography	2	2	3	3	4	4	6	7	2	2
Commercial & industrial	2	2	3	3	1	1	1	1	1	1
Fuel cycle	1	2	1	1	0	0	0	0	0	0
Other	2	2	3	4	0	0	0	0	1	1

Table 2.7 Overexposure Rate at Reactor and NRC Radiography Licensees—1987–1991

Year	Reactors: No. of Workers With Measurable Doses	Reactors: No. of Workers Overexposed	Reactors: Over-exposures per 1,000 Workers	Radiography Licensees: No. of Workers With Measurable Doses	Radiography Licensees: No. of Workers Overexposed	Radiography Licensees: Over-exposures per 1,000 Workers
1987	104,330	5	0.05	4,454	2	0.44
1988	103,227	7	0.07	4,223	3	0.71
1989	108,253	2	0.02	4,352	14	3.2
1990	109,702	1	0.01	4,458	12	2.6
1991	93,519	0	0	4,649	2	0.43

and processing licensees over the 5-year period 1987 through 1991 represents a rate that is small; in no case was the overexposure rate more than 0.3 percent of the number of workers with measurable doses.

2.2.4 Summary

As a result of measures taken by utilities and other licensees, worker occupational overexposures have been reduced to a point where one or two are reported annually by nuclear power plants.

2.3 Accident Sequence Precursor Program

The NRC established the Accident Sequence Precursor (ASP) Program in the summer of 1979 under contract at the Nuclear Operations Analysis Center at Oak Ridge National Laboratory to provide a structured and systematic means of evaluating the safety significance of nuclear plant operating experience. In the program, selected licensee event reports (LERs) of plant problems, equipment failures, or other operational incidents at nuclear plants are evaluated. An accident sequence precursor is defined as an operational event, or events, or a plant condition that is an important part of a postulated nuclear plant core-damaging accident sequence.

Under the ASP Program, operational occurrences that involve portions of postulated core-damage sequences are identified and evaluated. Plant equipment and human responses that could affect the progression of an accident are evaluated, including the actual failures that have occurred along with the probabilities for postulated additional failures that could occur. Event tree models and probabilistic risk assessment techniques are used to provide a quantitative estimate of the significance of the reported data, hence, a perspective on event importance. The event trees model plant responses to challenges such as transients, loss-of-coolant accidents (LOCAs), or loss of offsite power (LOOP).

Accident sequences considered in the ASP Program are those associated with inadequate core cooling. Accident precursors are important elements in such sequences. Such precursors can be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition involving inadequate core cooling. The precursor method couples and evaluates seemingly disparate elements of operational experience with random failures assumed for other branches of the event tree models being evaluated. All actual or potentially concurrent failures, degradations, or outages of safety systems or related plant systems are accounted for, to the extent possible, in these evaluations.

Commercial nuclear power reactors in the United States now have approximately 1,700 reactor-years of operating experience. The precursor program utilizes information from this accumulating and valuable nuclear plant experience data to provide an ongoing assessment of operating experience.

The precursor events from the ASP Program form a unique database of historical system failures, multiple losses of redundancy, and potential core-damage initiators. Several of the precursor events involved failure of equipment caused by factors or conditions or phenomenology that affected the ability of safety equipment to perform its function. These mechanistic failures are different from "random" failure or unavailability of equipment. The precursor assessments and results provide an indication of how well nuclear plant designs and capabilities cope with actual operational events.

Table 2.8 lists precursor events with estimated conditional core damage probabilities (CCDPs) greater than 1.0E–6 that occurred in 1991. That is, given the event or condition, the estimated probability that the event or condition could have led to core damage was greater than 1 in 1 million. Figure 2.15 shows the number of such precursors by the year of occurrence.

Four events involved unavailability or potential unavailability of high- or low-pressure safety injection at pressurized-water reactors. In two of these cases, hydrogen gas could have potentially made the high-pressure injection pumps inoperable. It should be noted that in 1990, several ASP events also involving gas entrainment occurred.

In 1991, 6 of the 27 ASP events were losses of offsite power. Losses of offsite power were also important ASP events in previous years. ASP analyses indicated that the loss of offsite power with the highest CCDP occurred at Yankee Rowe on June 15, 1991. In this event, caused by lightning, offsite power was lost for 24 minutes. All of the emergency diesel generators (EDGs) operated as designed. In another loss of offsite power event at Vermont Yankee, offsite power was lost for 13 hours. The EDGs worked properly and a tie line from off site was available through operator actions to power one-half of the emergency equipment at the site.

In addition, four events involved unavailability of the EDGs that would be required to respond to loss-of-offsite-power events. In three of these events (Perry Unit 1 on March 14, 1991; Millstone Unit 2 on August 21, 1991; and Surry Unit 2 on July 15, 1991), two diesel generators were potentially inoperable for some period of time.

In 1991, unavailability of equipment needed to mitigate the consequences of anticipated events (e.g., small-break LOCA, LOOP) made a significant contribution to the ASP precursors.

Additional information and detailed analysis are provided in NUREG/CR–4674, "Precursors to Potential Core Damage Accidents," Vols. 1 through 16.

Table 2.8 Precursor Events Occurring in 1991

Plant	LER Number	Date	CCDP	Description
Harris 1	400/91-008	04/03/91	6.3E-3	HPI unavailable for one refueling cycle because of inoperable alternate miniflow lines
Millstone 3	423/91-011	04/10/91	8.6E-4	Both trains of HPI inoperable because of relief valve failure
Yankee Rowe	029/91-002	06/15/91	6.1E-4	LOOP caused by lightning strike
Perry 1	440/91-009	03/14/91	5.3E-4	Two EDGs inoperable
Arkansas 2	368/91-012	05/15/91	4.8E-4	Both normal service water trains fouled by debris
Nine Mile Point 2	410/91-017	08/13/91	3.8E-4	Loss of five nonsafety uninterruptible power supplies
Peach Bottom 3	278/91-017	09/24/91	3.3E-4	Control wiring for ADS/relief valves found damaged
Vermont Yankee	271/91-009	04/23/91	2.9E-4	Extended LOOP
McGuire 1	369/91-001	02/11/91	2.6E-4	Switchyard breaker test resulting in LOOP
Zion 2	304/91-002	03/21/91	2.1E-4	LOOP with one EDG out of service
Millstone 2	336/91-009	08/21/91	2.1E-4	Both EDGs unavailable and unit shut down
Oconee 1	269/91-010	09/19/91	1.2E-4	Potential for hydrogen entrainment in HPI pumps
Pilgrim 1	293/91-024	10/30/91	1.2E-4	LOOP and RCIC trip
FitzPatrick	333/91-014	08/05/91	9.5E-5	Hydraulic pressure locking of two LPI valves
Comanche Peak 1	445/91-012	03/26/91	6.2E-5	Potential charging pump unavailability due to hydrogen voids
Brunswick 1	325/91-018	07/18/91	6.0E-5	LOFW with degraded HPCI system
Seabrook	443/91-008	06/27/91	4.4E-5	LOOP
FitzPatrick	333/91-006	05/07/91	2.0E-5	Trip with both LPCI trains inoperable
Oconee 3	287/91-007	07/03/91	1.8E-5	Reactor trip due to LOFW plus degraded emergency feedwater

Table 2.8 (cont.)

Plant	LER Number	Date	CCDP	Description
Hatch 1	321/91-001	01/18/91	1.1E-5	LOFW with degraded HPCI and RCIC failed
Zion 2	304/91-004	06/11/91	1.0E-5	Main feedwater pump trip with one failed AFW pump
Harris 1	400/91-010	06/03/91	6.6E-6	Reactor trip breaker fails to open on trip
Salem 1	272/91-030	09/20/91	4.4E-6	Failure of both PORVs because of leaking actuators
Surry 2	280/91-017	07/15/91	2.9E-6	Both EDGs for Unit 2 inoperable for 13 hours
San Onofre 1	206/91-014	08/07/91	2.1E-6	Inoperable VCT level transmitters
Diablo Canyon 2	323/91-003	09/01/91	2.1E-6	Containment sump isolation valves and containment spray pumps de-energized during hot shutdown
Indian Point 2	247/91-001	01/07/91	2.0E-6	Reactor trip and AFW pump failure

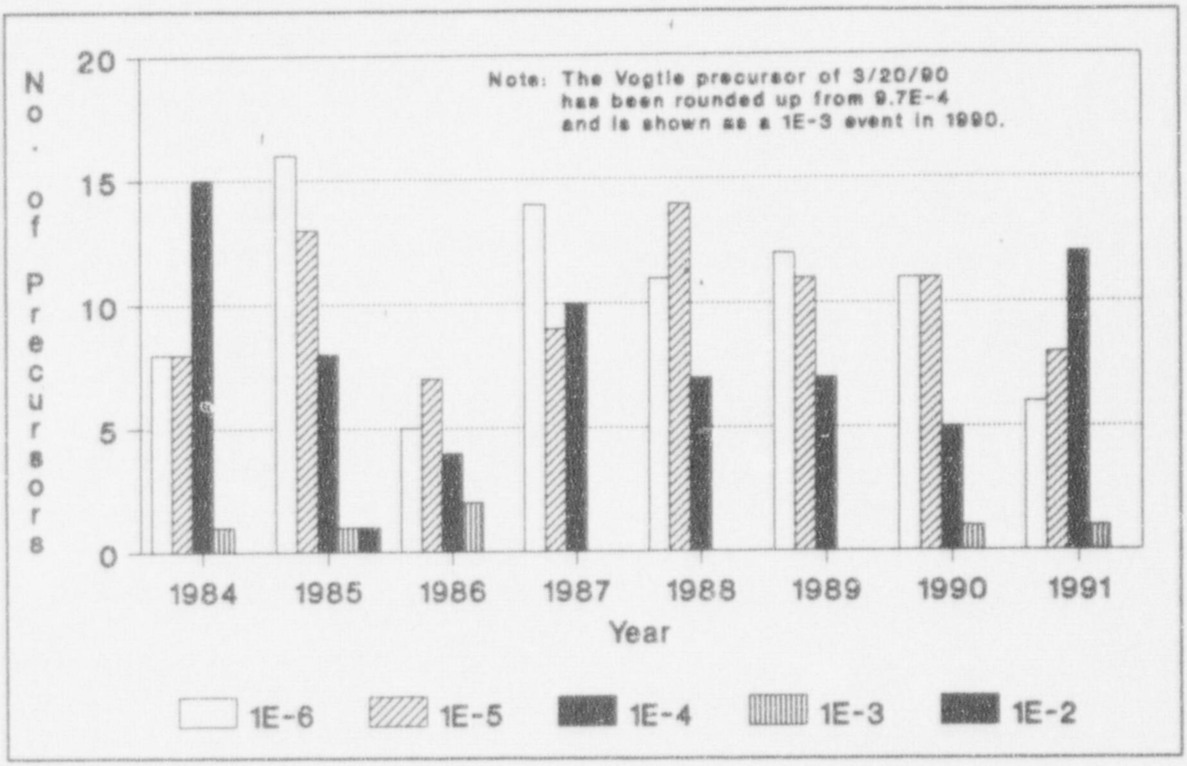

Figure 2.15 Number of Precursor Events—1984–1991

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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ERRATA SHEET

Report Number:	NUREG-1272, Vol. 7, No. 1
Report Title:	Office for Analysis and Evaluation of Operational Data 1992 Annual Report: Power Reactors
Date Published:	July 1993
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3 Results of AEOD Studies

3.1 Evaluations

In 1992, the AEOD staff continued to analyze and evaluate operating experience, publishing studies of safety and safety-relief valve reliability, molded-case circuit breakers, pressure locking and thermal binding of gate valves, inadequate management control of snubber surveillance, insights from common-mode failures, and several technical reports describing equipment problems. The results of these studies are summarized in the rest of this section.

Considerable effort was expended on the quantitative analysis of risk associated with operational events and conditions, and on better understanding human performance. These efforts are described in Sections 2.3 and 3.1.1, respectively.

In the evaluation of operational experience, the AEOD staff reviews a broad variety of operating data. These data include reports submitted by licensees to the NRC in compliance with 10 CFR 50.72 ("Immediate Notification Requirements for Operating Nuclear Power Reactors"), 10 CFR 50.73 ("Licensee Event Report System"), and the database of component failures in the Nuclear Plant Reliability Data System, a system managed by the Institute of Nuclear Power Operations (INPO).

Other operational experience reviewed includes 10 CFR Part 21 reports ("Reporting of Defects and Noncompliance"), NRC regional inspection reports, preliminary notifications of events or unusual occurrences that the NRC issued, and data on foreign reactor events.

AEOD case studies involve substantive, in-depth analyses of significant safety issues that are identified through the review of this operating experience. These studies document the bases for AEOD recommendations for regulatory or industry actions. A case study report goes through a rigorous peer review process to ensure its technical adequacy before it is published. Special studies document accelerated investigations and contain suggestions or recommendations for regulatory actions that are to be completed expeditiously.

Engineering evaluations document assessments of significant operating events and contain suggestions for remedial actions, if appropriate.

Technical reviews document AEOD studies of an issue that the staff concludes has little safety significance. These studies typically conclude that the licensees' or industries' planned or scheduled corrective actions are adequate.

3.1.1 Operating Experience Feedback Report—Human Performance in Operating Events

Case Study AEOD/C92-01—Published as NUREG-1275, Volume 8

AEOD began a program in 1990 to conduct onsite, in-depth studies of human performance that affected reactor safety during selected power reactor events. The purpose of the program is to identify the factors that have contributed to good operator performance during events, as well as the factors that have hindered performance, and to feed this information back to industry.

Each study was conducted by a multi-disciplinary team, led by an AEOD staff member, with additional NRC headquarters, regional, and Idaho National Engineering Laboratory personnel. The studies focused on those factors that helped or hindered operator performance. The team usually spent 1 to 3 days on site interviewing plant personnel and gathering records. Individual reports of each site study were prepared and distributed within the NRC, the site involved in the study, industry groups, and the public. This case study describes generic observations and conclusions drawn from 16 such studies.

These events represented an estimated one-fourth to one-third of the events that significantly challenged operating crews during this 2-1/2-year period. Six studies were performed in 1990, seven in 1991, and three in 1992. The 1992 studies are discussed in Section 3.2 of this report. The 1990 and 1991 studies are discussed in AEOD annual reports of their respective years. Of these 16 events, 9 occurred at pressurized-water reactors and 7 occurred at boiling-water reactors (BWRs). Ten events occurred while the reactor was at power, and six occurred while the plant was in a standby or shutdown mode. Fifteen separate sites were visited. Five studies were performed as part of augmented inspection team inspections, while eleven were performed solely under AEOD auspices.

The events represented a wide variety of event or accident scenarios, including stuck-open safety-relief valve, reactor trip with safety injection, reactor scram due to positive reactivity insertion, reactor scram due to control rod withdrawal, pressurizer spray valve failure, partial loss of instrument air in the containment, turbine building pipe rupture, loss of shutdown cooling, excess steam demand, main steam isolation, defeat of reactor water cleanup isolation, relief-valve lifting, loss of annunciators and plant computer, and loss of electrohydraulic fluid.

This study summarized each event and the findings drawn, observations discerned from multiple events, and conclusions concerning overall human performance. These fell into four groups: control room organization, procedures, human-machine interface, and industry initiatives. Finally, the categorization of events by latent factors compared the similarities among the events. The primary observations and conclusions of the special study included the following.

Control Room Organization

Control room staffing level, division of responsibility, and degree of teamwork significantly affected crew response to events.

Control room management was overburdened during emergencies when task, supervision, and technical oversight were not appropriately allocated.

The use of the "dual role" shift technical advisor impaired crew performance by overloading other senior reactor operators when one senior reactor operator assumed the shift technical advisor role. The dual-role shift technical advisors sometimes lacked independent "fresh eyes" because of involvement in shift activities. Other tasks, such as notifications, also detracted from the shift technical advisor's safety function.

Teamwork during events improved human performance in complex, high-stress situations. Training and teamwork were shown to be useful in increasing the effectiveness of knowledge-based performance.

In this study, the staff concluded that an examination of control room staffing and structure versus emergency functions would result in better utilization of shift resources and allocation of tasks so that no individuals were overburdened. This would be especially worthwhile with regard to the dual-role shift technical advisor function.

Procedures

Some operators acted during events without using a formal procedure. Procedure content, ease of use, and management policy and practices influenced procedure use. Procedure problems were key contributors in the less-successful events, but not during the more-successful events when the procedures were accurate and complete and management required their use.

Operators experienced difficulty in applying knowledge to unusual plant conditions during events, which resulted in delays in recognizing and responding to events. Some knowledge-based performance is necessary in every event so that personnel recognize the significance of the situation, initiate use of the appropriate abnormal operating procedures or emergency operating procedures, and follow those procedures to respond to events.

Preconditioning from past experience, training, or management direction strongly affected how operators recognized and responded to events and in some cases led operators to doubt valid indications or take inappropriate actions. Preconditioning to bypass engineered safety features actuation has a high potential for harm.

In two events, operators inappropriately defeated the automatic operations of engineered safety features during valid system demands. Some licensees have not provided criteria and sufficient guidance that limits the defeating of engineered safety features. Improper defeating of engineered safety features in two events during a 4-month period showed that the NRC and industry efforts to appropriately control engineered safety features have not been completely effective and that further action would have a high safety return in the reduction of risk of operator error.

Human-Machine Interface

A lack of appropriately ranged, direct-reading control room instrumentation to monitor reactor pressure, temperature, and level caused operators to have difficulty in recognizing and responding to shutdown events, when operator actions were required to accomplish the safety functions of disabled automatic safety systems.

Annunciator and computer alarms are important operator aids in recognizing and responding to events. Operators failed to recognize conditions that were off-normal, but that were not alarmed during events.

Industry Initiatives

The effectiveness of individual licensee studies of human performance during operating events varies widely. While some licensees have initiated worthwhile plant-specific corrective actions because of their followup of these events, other have missed such opportunities.

Industry groups are engaged in many efforts to improve human performance and human reliability. These efforts have resulted in improvements in plant performance, procedures, and programs. With the perceived reduction in the number of events caused by equipment failures, INPO and other industry groups and human-performance experts agree that a key to continued improvement in plant performance and safety is improved human performance.

3.1.2 Safety and Safety-Relief-Valve Reliability

Special Report AEOD/S92–02

The AEOD staff analyzed 1,100 events that had occurred from January 1981 to December 1989 in which a safety valve or a safety-relief valve (SRV) malfunctioned during operation or during surveillance testing. Safety valves should open within setpoint pressure tolerance limits to relieve overpressure conditions and should reclose to maintain system boundary integrity. The systems of interest were the reactor coolant system and the main steam system. The valves involved were Crosby and Dresser pressurizer safety valves (PSVs), Crosby and Dresser main steam safety valves (MSSVs), and Target Rock two-stage SRVs.

PSVs, SRVs, and MSSVs are usually allowed plus or minus 1-percent tolerance on either side of the setpoint pressure. Most accident and transient analyses assume operation within the tolerance. However, approximately [illegible] percent of all the reported safety-valve malfunctions were attributed to the condition called "setpoint drift," where the valves do not meet the assumed setpoint tolerance. The safety significance of safety-valve or SRV malfunction was shown to be the degradation of overpressure protection for higher-than-required setpoints or a potential loss-of-coolant event for lower-than-required setpoints. Significant complications can occur during a post-scram transient when a safety valve or SRV lifts unexpectedly, whether or not it reseats properly, or when it lifts for cause and fails to reseat.

The AEOD staff concluded that safety-valve performance could be improved if the owners established a program similar to that used previously by the SRV Owners Group to identify and correct SRV malfunctions. Other suggestions include development of standard practices for maintenance and testing of safety valves and SRVs in order to eliminate testing-induced errors and establish effective corrective actions.

The study was forwarded to the Office of Nuclear Reactor Regulation (NRR) for action. NRR has incorporated it into a proposed generic safety issue "Spring-Actuated Safety and Relief Valve Reliability" and requested that the Office of Nuclear Regulatory Research (RES) prioritize the issue. RES scheduled this work to begin in January 1993.

3.1.3 Review of Operational Experience With Molded-Case Circuit Breakers in U.S. Commercial Nuclear Power Plants

Special Study AEOD/S92–03

During the NRC staff's diagnostic evaluation at Zion in June 1990, the evaluation team observed that the Zion licensee had no program for routine testing of molded-case circuit breakers (MCCBs). Because of the failure to routinely test safety-related MCCBs, the team was concerned about the potential degradation of these components, which could occur without detection, and whose failure to perform their design interrupt function could impair the performance of safety-related systems. This current interrupt function is to protect safety-related components against damage from overcurrent conditions.

As a followup action to the Zion diagnostic evaluation, AEOD was tasked to review the failure history of safety-related MCCBs to determine whether regulatory action was required to provide assurance that licensees were adequately addressing MCCB reliability in safety-related applications. This review covered MCCB failures reported to the Nuclear Plant Reliability Data System (NPRDS) between January 1985 and December 1990. AEOD examined 345 reports of failures of safety-related MCCBs out of a total population of 21,000 MCCBs.

The results of this review of operational experience indicate that, in spite of the various manufacturer recommendations and industry standards identifying good practice regarding MCCB testing and maintenance that are available to licensees, licensees of many plants do not perform specific testing or maintenance of safety-related MCCBs.

The analysis of MCCB failure modes showed that most of the failure-rate data fell below the estimated range for the generic MCCB failure rate contained in Institute of Electrical and Electronics Engineers (IEEE) Standard 500–1984, with no discernible trend. Because of this overall behavior, coupled with the relatively low failure rates, the failure history of safety-related MCCBs over the 6 years reviewed did not, in general, exhibit any notable declining performance trend. Moreover, a review of licensee event reports for this period did not show any recurrent, industrywide safety concerns. Consequently, operational experience does not support any specific regulatory initiative at this time.

3.1.4 Pressure Locking and Thermal Binding of Gate Valves

Special Study Report AEOD/S92–07

In July 1991, at FitzPatrick, the inboard injection valve for the low-pressure coolant injection (LPCI) system became pressure locked following a hydrostatic test of the piping between the inboard and outboard injection valves. The test pressure had pressurized the injection valve bonnet cavity, and the cavity did not completely depressurize during the subsequent 10 hours before the valve operability test. As a result, the motor-operated valve (MOV) failed to operate. The motor operator burned out because the force required to open the MOV exceeded motor capability. Although the valve pressure locking was revealed during a hydrostatic test, the licensee determined that the cause was a design problem that was shared by the LPCI inboard injection valves in both loops and both inboard injection valves of the core spray system. The problem could prevent the operation of all four low-pressure emergency core cooling systems.

The nuclear industry has been aware for many years of the potential for gate valve inoperability caused by pressure locking or thermal binding. The staff found that valve binding caused by pressure locking or thermal binding is a common-mode failure mechanism. This not only can prevent a gate valve from opening on demand, but may damage the motor winding and could render redundant trains of safety systems, or multiple safety systems, inoperable. In spite of numerous generic communications issued in the past by both the NRC and industry, pressure locking and thermal binding continue to occur in gate valves installed in safety-related systems of both boiling-water reactors and pressurized-water reactors. The previous generic communications have not led to effective industry action to fully identify, evaluate, and correct the problem.

The AEOD staff identified (1) conditions when the failure mechanisms have occurred, (2) the spectrum of safety systems that have been subjected to the failure mechanisms, and (3) conditions that may introduce the failure mechanisms under both normal and accident conditions. The valve binding is a result of inadequate design considerations under specific system conditions. Most valve binding events occurred during plant evolutions, system transients, or unusual system alignments. Hence, the inadequacy in design or installation will not necessarily be found during plant startup testing or regular surveillance testing. The staff concluded that comprehensive system evaluations and analyses, including consideration of plant system conditions and ambient conditions during all modes of plant operations, are needed to identify the valves susceptible to binding and determine the effect on safety system function.

As a result of the study, the NRC is preparing an information notice and plans to conduct workshops for the NRC staff and industry. NRC inspectors plan to verify the adequacy of licensees' evaluations and corrective actions on the potential binding mechanisms of safety-related gate valves. The NRC also plans to request that the Nuclear Management and Resources Council (NUMARC) develop specific industry guidance on approaches to analyze and remedy the problem.

3.1.5 Inadequate Management Control of Snubber Surveillance

AEOD Engineering Evaluation Report AEOD/E92-01

The AEOD staff evaluated 33 licensee event reports from 28 plants involving management control deficiencies associated with snubber surveillance. These deficiencies were (1) removal of a snubber from a system (component) when the system was required to be operable and (2) omission of a snubber from the surveillance program.

The removal of snubbers from systems when the systems were required to be operable left the systems in conditions outside the design bases, while the omission of the snubbers from the surveillance programs compromised snubber operability. These were significant deficiencies in management control of snubber surveillance that could have posed a hazard to the safety of plant operation had they remained uncorrected.

Most deficiencies were discovered when the plant staff was performing special reviews or inspections initiated for some designated programs other than snubber surveillance and the problem had existed for some time. This suggested that snubber surveillance programs may not be adequate to ensure snubber operability as required by plant Technical Specifications and may represent a generic problem.

3.1.6 Insights From Common-Mode Failure Events

Engineering Evaluation Report AEOD/E92-02

Common-mode failures were studied because they are a major uncertainty in the bottom-line value in probabilistic risk assessment. Forty-four licensee event reports that described actual or potential common-cause failures were analyzed. Most of the events studied occurred in 1990 and were limited to those situations judged not recoverable (in the event of a coincident accident) or not self-revealing during normal plant operation. In addition, miscalibration events were not considered in the study on the basis of perceived low safety significance. Sixteen of the events reviewed were identified as precursors under the Accident Sequence Precursor Program and thus exemplify the potential importance of this issue.

The intent of this work was to identify dominant corrective actions that would preclude or reduce the likelihood of common-cause failures at operating nuclear power plants. Each of the events was reviewed against a set of eight corrective actions. The events fell into three roughly equal groups—(1) maintenance, (2) design/installation, and (3) environmental qualification (EQ) and train separation-related failures. Examination of the events indicated that staggered testing might have prevented about one-half of the maintenance-related events. The use of equipment with larger design margins or of comprehensive system tests that enveloped all operating conditions might have prevented many of the design/installation events. The use of diverse equipment and equipment having larger design margins might have prevented many of the EQ failure events.

The effectiveness or practicality of applying any of these corrective actions to all important safety components or systems at operating plants was not explored. However, information from this study is being considered by RES in its evaluation of Generic Issue 145, "Actions To Reduce Common Cause Failures." The NRC staff is preparing an information notice informing the licensees of these results.

3.1.7 Enhanced Setpoint Testing Procedures for Pressurizer Safety Valves at Oconee and Catawba

Technical Review Report AEOD/T92–01

Pressurizer safety valves prevent overpressurization of the reactor coolant system during transients and accidents that involve a mismatch between the primary heat source and the secondary heat sink. The impact of an increased pressurizer safety valve setpoint for a pressurized-water reactor would be the reduction of margin between the peak transient reactor coolant system pressure and the pressure vessel safety limit.

Duke Power Company, licensee for Catawba and Oconee, presented information developed while investigating the apparent excessive setpoint drift of three pressurizer safety valves. The licensee postulated nine factors that could have caused the deviations and evaluated them. The four factors deemed more important were valve leakage, lack of control of the "jack-and-lap" process, improper trending, and ring adjustment without retest. The licensee then took steps to correct errors introduced by each factor, including the less important ones.

The AEOD staff concluded that the licensee had demonstrated that leakage can change the effective seat area of a safety valve causing it to lift at a lower apparent setpoint, that attempts to correct leakage without retesting the valve can result in a different setpoint, and that carefully controlled testing procedures provide a more precise setpoint determination.

3.1.8 BWR–5 and –6 Events Applicable to Laguna Verde

Technical Review Report AEOD/T92–02

Selected significant events from January 1, 1988, to May 31, 1991, for Grand Gulf, River Bend, Nine Mile Point Unit 2, LaSalle, and Washington Nuclear Plant Unit 2 (WNP–2) were reviewed. The events for these plants were chosen because the plants are similar to Laguna Verde, a Mexican BWR–5 with a Mark II containment. This study was initiated and prepared by a Mexican National Commission for Nuclear Safety and Safeguards staff member who was the recipient of a National Institute for Nuclear Research fellowship. The purpose was to identify issues, problems, or lessons learned from U.S. operating experience that might be applicable to Laguna Verde.

Eight events were identified with potential significance for Laguna Verde. These events involved (1) an unanalyzed condition when handling large loads over irradiated fuel; (2) turbine stop-valve closure resulting in reactor scram; (3) overheating in diesel generator excitation control cabinets; (4) primary containment purge line crack due to failures in the containment nitrogen-inerting system; (5) lockup of the rod pattern control system when 12 control rods drifted out from their original position because of high differential pressure in the control rod drive system; (6) failures in the emergency reactor building heating, ventilation, and air conditioning system; (7) high-pressure core spray cracks; and (8) deficiencies in the emergency operating procedure graphs for the pressure-suppression limit curve, heat capacity temperature-limit curve, and primary containment pressure-limit curve. A summary of each event was provided in the study.

3.1.9 Solenoid-Operated Valves and Related Equipment – A Status Report

Technical Review Report AEOD/T92–03

Following several safety-significant events involving the simultaneous failures of multiple solenoid-operated valves (SOVs), AEOD began a systematic analysis of operational events involving these valves. It performed a case study that showed that nuclear plants were susceptible to common-mode SOV failures caused by deficiencies in SOV design, application, maintenance, surveillance testing, and manufacturing. AEOD's case study, C90–01, "Solenoid-Operated Valve Problems in Light Water Reactors," was issued as NUREG–1275, Volume 6, in February 1991. In September 1991, the NRC issued Generic Letter 91–15, Operating Experience Feedback Report, "Solenoid-Operated Valve Problems at U.S. Light Water Reactors," forwarding the case study to licensees.

Technical review report AEOD/T92–03 presents a status of recently performed, ongoing, and planned activities associated with SOVs and related equipment. Industry activities being undertaken in response to the concerns raised in the case study AEOD/C90–01 (NUREG–1275, Volume 6) are described, including initiatives being taken by professional societies such as the American Society of Mechanical Engineers and IEEE, and actions being taken by individual licensees, major SOV manufacturers, and industry groups.

3.1.10 Recent Solenoid-Operated Valve Experiences Involving Maintenance and Testing Deficiencies

Technical Review Report AEOD/T92-04

This technical review report presents recent SOV operating experience and the AEOD staff's analysis of common-mode failures at Salem Unit 2 and Peach Bottom Units 2 and 3. At the Salem plant, a catastrophic failure of turbine and generator equipment occurred that resulted in an outage of about 6 months. The Peach Bottom event involved repetitive failures of safety systems. The events described in the report demonstrate the need for prudent preventive maintenance and surveillance testing.

Prudent maintenance and surveillance testing can reduce the likelihood of common-mode SOV failures. However, maintenance and surveillance testing alone cannot take the place of good design. Furthermore, maintenance and surveillance testing should not be relied on to overcome design and application errors.

The AEOD staff concluded that the events at Salem Unit 2 and Peach Bottom Units 2 and 3 were examples of situations where less-than-adequate surveillance testing and maintenance of SOVs resulted in the reduction of plant safety margins and/or significant financial burden. In recognition of the fact that highly reliable, non-obtrusive SOV diagnostic and monitoring equipment is not available, prudent preventive maintenance, surveillance testing, and SOV replacement strategies can be used to reduce the likelihood of common-mode SOV failures.

3.1.11 Errors in Effective Reactor Trip Settings or Monitoring Associated With Excore Detectors

Technical Review Report AEOD/T92-05

The AEOD staff examined errors in effective reactor trip settings or reactor monitoring associated with excore monitors that were reported in 26 licensee event reports (LERs) covering power ascension as well as full-power operation. A wide range of human errors contributed to the observed mistakes in the effective reactor trip settings. These errors included miscommunication with the vendor, procedure errors, administrative oversights in maintaining the monitors, as well as technician errors. Industry generic communications have addressed most of these causal factors, which generally reflect site-specific deficiencies. On several occasions, during random supervisory observations in the control room, mismatches in power indications were detected and corrective actions were subsequently initiated. The safety analyses in the LERs generally did not indicate a significant safety issue. This topic has been reviewed extensively in six industry generic communications from 1988 through 1990. No additional action is warranted at this time.

3.1.12 Water Intrusion Into Sensitive Control Room Equipment

Technical Review Report AEOD/T92-06

The AEOD staff reviewed operational events of the past 13 years (1980 to 1992) that resulted in the intrusion of water into the main control rooms of nuclear power plants. The main control room houses sensitive control equipment; therefore, even a small amount of water could cause electrical shorts in circuitry of control equipment, causing spurious actuation or failures of safety-related systems.

The review indicated that reportable events involving water intrusion into control rooms are infrequent. The events, although having potentially significant effects, are readily identified by operators who then quickly stop the water intrusion. In about half of the events, there were no significant systems interactions. All three of the systems interaction events involved water entering analog trip units at BWRs.

Generic communications regarding the BWR events have been issued previously. Similarly, generic communications regarding other problems identified in the study, such as design or maintenance deficiencies of watertight penetration seals and drains and events involving inadvertent fire suppression system actuation, have been issued previously.

On the basis of the review of operating experience and related generic communications, the staff concluded that no further NRC generic communications or AEOD actions were warranted.

3.1.13 Inoperability of the Standby Liquid Control System During Surveillance Testing at Nine Mile Point Unit 2

Technical Review Report AEOD/T92-07

During review of the quarterly surveillance testing procedure, the licensee for Nine Mile Point Unit 2 discovered that the standby liquid control system (SLCS) would not have been capable of performing its safety function if an automatic actuation signal were received during the test. The AEOD staff determined that an actuation signal during the test could result in either injection of demineralized water instead of sodium pentaborate into the reactor coolant system or possible draining of the sodium pentaborate tank contents onto the floor.

The AEOD staff reviewed and described in the report the test methodology, potential operability problems, and applicability to BWRs. It reviewed the generic aspects of SLCS operability during surveillance testing and determined that only four other BWR plants were required to have automatic actuation for the SLCS. It concluded that the licensees' procedural enhancements would minimize the impact of surveillance testing. Further investigation was not required.

3.1.14 Emergency Diesel Generator Start Frequency

Technical Review Report AEOD/T92-08

The AEOD staff reviewed the frequency of emergency diesel generator (EDG) starts from 1988 through 1991. Although the number of starts varied significantly among the licensees, there was no significant difference in EDG reliability between groups of EDGs with the most frequent starts and those that were started least. The results of this technical review and also AEOD/T92-10 were sent to RES for its use in resolving Generic Issue B-56, "Diesel Reliability."

3.1.15 Review of Manual Valve Failures

Technical Review Report AEOD/T92-09

The AEOD staff conducted a review of manual valve failures in light-water reactors. It obtained data on these failures using the NRC Nuclear Document System, Sequence Coding and Search System, and Nuclear Plant Reliability Data System. Results indicated there were 20 plant systems with 20 or more reported failures for the same kind of manual valve.

No events were discovered that severely compromised plant safety. In addition, the review of manual valve failures indicated, in general, that large valves (45.7 cm (18 in.)) fail by normal wearout, 15.2-cm (6-in.) valves fail as a result of random mechanical failures, and 5-cm (2-in.) valves fail because of packing problems.

On the basis of this evaluation, the staff concluded that current industry practices with respect to maintenance and surveillance of manual valves were adequate.

3.1.16 Prospective Trend of Low-Reliability Emergency Diesel Generators

Technical Review Report AEOD/T92-10

The AEOD staff reviewed EDG start data to identify EDGs with less than 95-percent reliability to start and load-run; this accounted for approximately 7 percent of the EDGs in any given year. The analyses showed that in almost all cases, start and load reliability of these problem EDGs improved in succeeding years to match the industry average. However, yearly EDG rotation occurs; after 1 year, a new set of low-reliability EDGs was identified. In each succeeding year, the group of problem EDGs was replaced by a new group, with few repeat offenders.

The staff did not identify any EDGs with consistently ongoing problems, and no station appears to have an inordinate number of low-reliability EDGs.

3.1.17 Usefulness and Effectiveness of Design-Basis Reviews

At the request of one of the Commissioners, the AEOD staff performed an analysis of LERs that reported problems discovered during design-basis reviews (DBRs) to determine the usefulness and effectiveness of these reviews. The staff assumed that such usefulness and effectiveness were related to the number of significant problems discovered during DBRs and reported in LERs.

The AEOD staff screened approximately 14,000 LERs submitted during the 5-1/2–year period from 1987 through the middle of 1992 and identified approximately 2,000 candidate LERs that reported design problems. As a result of its review of candidate LER abstracts, the staff identified approximately 800 DBR LERs that were the result of formal or informal DBR programs. The number of DBR LERs per site ranged from 1 to 33. The total number of DBR LERs for the oldest unit was used as the total for the site.

To determine the influence of plant age, plant age groups were established with approximately one-third of the sites in each group (Group A, Group B, and Group C). This grouping was based on commercial operation dates: 1974 and earlier, 1975 through 1983, and 1984 and later, respectively. The groupings were established in this manner only to keep the number of plants in each group constant.

The median number of DBR LERs reported per site was nine. Categories were developed to describe the frequency of DBR LERs as follows:

Below median (BM):	5 or less
Median range (MR):	6–12
Above median (AM):	13-24
Extensive (E):	25 or more

To ensure that an adequate base of DBR LERs would be used to evaluate the usefulness and effectiveness of the licensees' programs, the staff selected sites from those that were categorized as "AM" and "E" and performed a detailed review of LER abstracts for all sites in these two categories. Each of the DBR LERs was assumed as having a level of significance determined as either "higher significance" or "lesser significance." A more detailed characterization of the higher-significance DBR LERs was provided for two sample sites in Group A and one sample site from Group B. Each sample site had submitted 25 or more DBR LERs.

Generally, most of the DBR LERs reported one or more of the following: (1) an unanalyzed condition in accordance with 10 CFR 50.73(a)(2)(ii), (2) a design error or inadequacy, or (3) a situation where a plant was outside its design (including licensing) basis.

The staff reviewed DBR LERs from the three sample sites and determined the number and percentage of higher-significance DBR LERs. For each of these LERs, the staff documented the regulatory criteria violated and the characterization of the significance. They then reviewed DBR LERs for the remainder of the sites in the AM and E categories for significance level, using the same criteria as those used in the review for the three sample sites. They tabulated the number and percentage of higher-significance DBR LERs, but did not develop further details to characterize each reported event.

The study showed that the older plant age grouping, Group A, had 40 percent of the DBR LERs; Group B had 34 percent; and the newer plant age grouping, Group C, had 26 percent. The distribution of all frequency categories for the Group A and Group B plants was similar. For the Group C plants, the BM frequency category distribution was much higher (46 percent), while the AM and E frequency category distributions were significantly lower. For AM category sites, 28 percent (average) of the DBR LERs were evaluated as having a higher significance level, while the E category sites averaged 36 percent, a somewhat higher distribution percentage. The overall average for the two categories was 32 percent.

Because of the large percentage of higher-significance DBR LERs from plants with an assumed E or AM category of DBR LER reporting, the staff concluded that, for these sites, DBR programs were useful and effective. No conclusions could be reached regarding effectiveness and usefulness for sites with little reporting. The staff also found that the number of plants with DBR programs was related to the plant age grouping.

3.1.18 Summary of Evaluations

Evaluations for calendar year 1992 covered a wide range of subjects and varied from a relatively broad evaluation to develop insights from common-cause failure events to indepth reviews of component-related problems (e.g., gate valves, safety valves and safety-relief valves, and molded-case circuit breakers).

In "Insights From Common-Mode Failure Events" (E92-02), the AEOD staff took a broad look at common-cause failures reported in LERs. The results of this evaluation will be considered by RES in resolving Generic Issue 145, "Actions To Reduce Common Cause Failures," and will be provided to licensees in an NRC information notice.

In "Review of Operational Experience With Molded Case Circuit Breakers in U.S. Commercial Nuclear Power Plants" (S92-03), the AEOD staff concluded safety-related molded-case circuit-breaker failure rates were not significant and did not indicate any recurrent, industrywide safety concerns. However, the study on gate valves, "Pressure Locking and Thermal Binding of Gate Valves" (S92-07), showed that although the nuclear industry has been aware of this phenomenon for many years, numerous industry and agency generic communications have not led to effective industry action to identify, evaluate, and correct this problem.

The study "Safety and Safety-Relief Valve Reliability" (S92-02) involved analyses in which a safety valve or safety-relief valve malfunctioned. The AEOD staff determined that valve performance could be improved through development of standard practices for maintenance and testing. NRR incorporated the report in a new proposed generic issue, "Spring-Actuated Safety and Relief Valve Reliability."

In "Inadequate Management Control of Snubber Surveillance" (E92-02), the AEOD staff concluded that significant deficiencies in management control of snubber surveillance activities could have posed a potential degradation to safe plant operation.

In the study "Usefulness and Effectiveness of Design Basis Reviews," performed at the request of one of the Commissioners, the staff analyzed LERs submitted from 1982 through the middle of 1992 in which design problems were reported. For sites that submitted 13 or more LERs, the staff concluded that design-basis review programs were useful and effective. No conclusions could be drawn for sites submitting fewer LERs.

Thus, as a result of its studies, AEOD continues to find both new and previously identified problems. AEOD studies also provide information that is helpful to the agency and industry in correcting problems.

3.2 Analyses of Human Performance in Operating Events

Studies of operating events have repeatedly shown the importance of human performance in regard to reactor safety. To obtain additional information, AEOD continued a program of conducting onsite, indepth studies of human performance during selected power reactor events. The primary goal of the program is to identify factors that influence human performance during events.

Under this program, teams of NRC staff and contractor specialists study selected events at plant sites shortly after the events occur. Interviews with the people involved in the event, reviews of instrument and computer readings, log-book entries, and observations of simulator reenactments provide the basis for these studies. Individual reports of each site study are prepared and distributed within the NRC, the site involved in the study, industry groups, and the public. Case Study AEOD/C92-01, "Human Performance in Operating Events," described generic observations and conclusions drawn from the 16 AEOD human-performance studies conducted between 1990 and 1992. See Section 5.0 of AEOD/C92-01. Three human-performance studies were completed during 1992 (event date in parentheses):

(1) Prairie Island Unit 2—Loss of Shutdown Cooling (2/20/92)

(2) LaSalle County Unit 2—Reactor Water Cleanup System Isolation Bypass (4/20/92)

(3) Fort Calhoun—Loss of Instrument Inverter and Subsequent Loss of Coolant (7/03/92)

Prairie Island Unit 2 Event

At 11:10 p.m. on February 20, 1992, an insufficient water level in the reactor coolant system (RCS) caused a loss of shutdown cooling. The operators responded promptly and initiated recovery procedures to restore water level in the reactor vessel and reestablish shutdown cooling flow. On February 21, 1992, NRC Region III sent an augmented inspection team (AIT) to investigate the event.

On February 20, 1992, Prairie Island Unit 2 was 2 days into a refueling outage. Late on the day shift, reactor vessel draining to midloop had started and then been halted for a shift change. The evening shift (6:00 p.m. to 6:00 a.m.) conducted beginning-of-shift briefings and reestablished draining. The two reactor operators (ROs) conducting the draindown were extra personnel from another shift used to supplement the normal duty shift. The extra ROs were in communication with operators in the containment building to accomplish the draindown.

Newly installed electronic level instrumentation was considered to be operable during the activity. When the draindown started, the electronic level instrument display on the control room emergency response computer system was off-scale high. A tygon tube was the only instrument providing usable level information during the draindown. To obtain actual level within the system, tygon tube level measurements were corrected using manual calculations for the nitrogen overpressure effects. Nitrogen overpressure was maintained to facilitate the draindown.

A systems engineer was on duty to provide assistance with the draindown and also to perform a preoperational check on the electronic instrumentation when it was indicating on-scale. After approximately 2 hours of draining, at 9:30 p.m., the electronic instrumentation was still off-scale high. The systems engineer conferred with an instrument technician and decided to leave the control room to investigate the level transmitter valve lineup in the containment building. This effort was interrupted by the announcement that shutdown cooling was lost. The systems engineer returned to the control room.

At 10:55 p.m., the draindown ROs were having difficulty calculating actual level and became concerned about reactor vessel water level. A containment building operator was sent to open a vent in the suction line of the residual heat removal (RHR) system to check for air (nitrogen). One of the draindown ROs decided nitrogen pressure was higher than it should have been at this point in the draindown and opened a reactor head vessel vent to vent off some of the excess pressure. The containment operator reported back that nothing but air was coming from the vent on the RHR suction line. He was ordered to close the vent and drain valves. Electronic level had suddenly changed from off-scale to an indication of about 220 m (723 ft)—12.7 cm (5 in.) below midloop—and a low level alarm was received. On the basis of interview data, the indicated level was as low as 220 m (722 ft 6.5 in.)—25.4 cm (10 in.) below midloop. Alarms on the emergency response computer system for RHR pump low suction pressure, low motor-amps, and low flow were received at 11:08 p.m. At 11:10 p.m., the shift manager ordered the running 22 RHR pump stopped.

The shift supervisor took direct command of the operations and entered Abnormal Procedure D2 AOP1, "Loss of Coolant While in a Reduced Inventory Condition," which directed the starting of a charging pump to raise the reactor vessel water level. The operators were monitoring RCS temperature using available incore thermocouples. The temperature was about 55 °C (133 °F) at the time of the running RHR pump trip. One entry condition for Emergency Operating Procedure (UP) 2E–4, "Core Cooling Following Loss of RHR Flow," required RCS temperature to be at 88 °C (190 °F). However, operators observed from the rate of level increase and heatup that the actions of the abnormal procedure were insufficient to mitigate the transient before the entry conditions of the emergency procedure were reached. The emergency procedure was immediately implemented when the temperature reached 88 °C (190 °F). The 21 RHR pump was aligned to the refueling water storage tank and started to inject water to the reactor vessel. Reactor vessel level was promptly regained. The 21 RHR pump was then stopped, realigned for shutdown cooling, and restarted. A peak temperature of 105 °C (221 °F) was reached before shutdown cooling was reestablished and the plant was returned to pre-event conditions.

Forty-two people were evacuated from the containment, with the exception of two members of the operations staff. The control room staff directed these two to stay in the containment to continue monitoring tygon tube level and to be available to operate valves for the draindown. Operators verified that containment integrity was intact as directed by the emergency procedure.

Prairie Island Unit 2 findings:

- Procedures and training did not provide sufficient direction in nitrogen pressure control. The significance of round-off errors during water level calculations was not recognized by the ROs and had not been addressed during training. As a result, incorrect information was used for the draindown.

- There was uncertainty as to who had responsibility and authority to make the decision to hold or stop draindown activity. The shift supervisors assumed the ROs were experienced and did not require continual supervision. An apparent hesitation by the draindown crew to communicate some concerns to the supervisors may have occurred because the ROs were not working with their normal crews.

- The draindown ROs lacked awareness of how higher nitrogen pressures affected the draining process.

- A questioning attitude regarding the response of the electronic display indicators was lacking even though the procedure stated that the displays should be operable.

- It would have been appropriate to hold or stop the draindown because of discrepancies and uncertainties regarding water level, but this was not done.

- A human-machine interface issue was identified when the local operator had difficulty reading the level correctly in the tygon tube because of parallax problems, poor lighting, and impaired tube visibility where the tube penetrated the floor.

LaSalle County Unit 2 Event

At 8:47 a.m. on April 20, 1992, shutdown of the reactor water cleanup (RWCU) system caused the lifting of an RWCU regenerative heat exchanger relief valve for 3-1/2 minutes, because an operator erroneously bypassed the automatic RWCU system isolation.

Several weeks earlier, an RWCU system isolation had occurred because of a spurious RWCU high-differential flow signal. Both RWCU containment isolation valve motors had failed because of thermal expansion effects on the valve limit switch settings. Licensee management had criticized the operators for allowing the spurious isolation. The motors had to be replaced, and a testing program was established to verify the limit switch settings as the plant power level increased.

On April 20, Unit 2 was at 20-percent power. The nuclear station operator (NSO) shut down the RWCU system, as part of the verification of limit switch settings, by closing the system return valve before stopping the RWCU pumps, which was in reverse order to that stated in the procedure substeps. About a minute later, the RWCU high-differential flow alarmed, indicating the start of a 45-second delay timer preceding isolation of the RWCU system.

The NSO did not want to abort the test and obtained the shift foreman's permission to bypass the automatic engineered safety feature (ESF) closure of the RWCU containment isolation valves. The NSO removed keys from other front control board switches and gave them to a second NSO. The second NSO used them to bypass the RWCU system isolation, but reported a continuing RWCU differential flow of 360 L/min (95 gal/min).

About 3 minutes later, the operators determined that the alarm was not spurious. An equipment attendant identified flow through an RWCU regenerative heat exchanger relief valve. A third NSO found the level in the reactor building equipment drain tank increasing, while the 360 L/min (95 gal/min) RWCU differential flow continued. The NSO asked the shift control room engineer and the shift foreman how they wanted to isolate the RWCU. Both agreed to allow the automatic RWCU system isolation, despite the precaution in the special test procedure that valve operation without thermal overload protection (as in the case of automatic operation) could damage the motor or the valve if the limit switch settings had drifted because of thermal expansion. The operators returned the RWCU bypass key switch to normal, allowing the RWCU system to isolate automatically, thus terminating the loss of inventory from the RWCU system through the open relief valve.

LaSalle County Unit 2 findings:

- The operators lacked understanding of the required order of performance of procedural directions.

- The special test procedure did not address response to an automatic isolation signal.

- While the alarm response procedure for the RWCU high-differential flow alarm did not address determination of alarm validity or criteria for use of ESF bypass keys, teamwork with auxiliary operators was a positive factor in verifying its validity.

- There was no direct RWCU relief valve discharge flow indication in the control room, and other instruments used to diagnose this event were located on different panels.
- Control room operators performed recovery actions without consulting applicable procedures because of their frequent revision and their level of detail.

Fort Calhoun Event

At 11:36 p.m. on July 3, 1992, a non-safety-related inverter was returned to service following repairs. When connected to its bus, the inverter output voltage oscillated and caused an electrical supply breaker to electrical panel AI-50 to trip open on a high-current condition. Electrical panel AI-50 supplied various instrumentation and components in the plant, including the control circuitry for the main turbine. When power was lost, the circuitry caused the main turbine control valves to close to protect the main turbine. With the turbine control valves closed, the heat sink for the RCS was temporarily lost, resulting in an increase in RCS pressure. The reactor and turbine tripped at approximately 16,550 kilopascals (kPa) (2,400 pounds per square inch absolute (psia)). As pressure continued to increase, the power-operated relief valves (PORVs), main steam safety valves, and a pressurizer code safety valve opened to reduce RCS pressure. The PORVs shut at 16,200 kPa (2,350 psia). When pressure reached approximately 12,000 kPa (1,750 psia), a pressurizer code safety valve closed and RCS pressure increased to approximately 13,270 kPa (1,925 psia). At this point, pressure began to drop rapidly. The operator shut the PORV block valves when the pressurizer quench tank level and pressure were observed to rise and the PORV/code safety valve tailpipe acoustic flow alarm was received. The pressure drop continued and safety injection, containment isolation, and ventilation actuation signals were received. All safety systems functioned as designed. The open pressurizer code safety valve partially closed at approximately 6,895 kPa (1,000 psia). The licensee declared an Alert when the safety valve stuck open.

The operators implemented the emergency operating procedures (EOPs) and secured the four reactor coolant pumps. The plant was subsequently cooled down, using natural circulation and shutdown cooling, to cold-shutdown conditions. A Region IV AIT investigated the event and issued the AIT report on August 6, 1992.

Fort Calhoun findings:

- The operations staff quickly diagnosed the plant status and took appropriate actions in a timely manner.
- A number of factors contributed to the successful operator response including the following: loss of coolant from the RCS was included in simulator training, EOPs were upgraded and provided sufficient guidance, emergency planning actions were practiced weekly in simulator training sessions, and control room organization and staffing provided a sufficient number of personnel with appropriate assignment of responsibilities.
- The event revealed a number of areas in which the technical content of EOPs could be improved.

4 Other AEOD Activities

4.1 Performance Indicator Enhancements

In 1992, the AEOD staff conducted a trial program of the enhanced performance indicator (PI) operational cycle/peer group methodology. Three enhanced quarterly PI reports were produced and distributed to the Interoffice Task Group (ITG) on Performance Indicators that was formed to evaluate the proposed methodology, as well as to NRC senior managers. The results of the trial program were documented in late 1992 in SECY-92-425, "Performance Indicator Program Peer Group and Operating Cycle Phase Enhancements," in which the staff recommended that the Commission approve the use of the enhanced PI report. Approval by the Commission is anticipated in early 1993.

The nine peer groups, listed below, are based on design and regulatory issues and are appropriate for comparing the overall performance of licensees operating similar plants in a similar regulatory environment.

<u>Boiling Water Reactor Peer Groups</u>

General Electric Pre-Three Mile Island
General Electric Post-Three Mile Island

<u>Pressurized Water Reactor Peer Groups</u>

All Babcock and Wilcox
Combustion Engineering Without Core Protection Calculators
Combustion Engineering With Core Protection Calculators
Small Westinghouse
Westinghouse Older 3-Loop
Westinghouse Older 4-Loop
Westinghouse New 3-Loop and 4-Loop

The five operational cycle phases, listed below, provide the capability to monitor and analyze various aspects of plant performance, such as the length of refueling outages, the number of days of non-refueling outages in a cycle, performance during startups, and problems during refueling.

(1) refueling outage
(2) startup from refueling
(3) power operations
(4) pre-refueling operations
(5) non-refueling outages

The ITG assessed the enhanced format and found it to be more informative than the current format. The changes in the calculational methods are designed primarily to determine (1) operating and shutdown performance separately, (2) performance relative to a group of similar plants rather than the entire industry, and (3) the statistical significance of the calculated trends and deviations. These modifications allow substantially more information to be conveyed clearly and concisely.

4.2 Licensee Event Reporting

The primary source of data on operational events is the licensee event report (LER). In the early 1980s, NRC published a rule (10 CFR 50.73) governing the content and submittal of LERs. The rule, which became effective in January 1984, clarified reporting requirements and established a more uniform threshold for the reporting of events by licensees. The threshold requires reporting of certain infrequent events of significance to plant and public safety and the more-frequent events of less significance. The latter events are more amenable to statistical analysis and developing trends.

Safety performance is only one of several factors that cause the number of LERs submitted by individual licensees to vary. Therefore, the staff does not base the assessment of safety performance of a plant on the number of LERs that have been submitted. Rather, judgments about safety performance are based on an evaluation of the number and significance of operational events. This section provides an overview of the reporting process as a major source of operational data. The average number of LERs per plant per year has consistently declined as shown in Table 4.1. The number of LERs submitted by individual licensees in 1992 ranged from a low of 2 to a high of 53; the average in 1992 for the industry was 16 LERs per plant. Approximately 94 percent of all LERs submitted in 1992 were submitted in response to the reporting requirements of 10 CFR 50.73; the remaining 6 percent were submitted in response to 10 CFR Part 21 or 10 CFR 50.36 or as voluntary reports. Table 4.2 shows the percentage of LERs submitted to meet the requirements in a specific section of 10 CFR 50.73.

Table 4.1 LERs Submitted by Year*

Year	No. of LERs	No. of Units	LERs per Unit
1987	2,895	111	26
1988	2,479	110	23
1989	2,356	112	21
1990	2,128	111	19
1991	1,858	111	17
1992	1,774	111	16

*Counts for 1987 through 1991 do not include the Dresden Unit 1, Humboldt Bay Unit 3, and Three Mile Island Unit 2 plants. Counts also do not include the Fort St. Vrain plant after August 29, 1989; the LaCrosse plant after April 30, 1987; the Rancho Seco plant after June 7, 1989; the Shoreham plant after June 6, 1987; the Yankee Rowe plant after February 26, 1992; and the San Onofre Unit 1 plant after November 30, 1992. Canceled, proprietary, and safeguards LERs were excluded from all counts.

4.2.1 U.S. Operational Experience Databases

AEOD uses the Sequence Coding and Search System (SCSS) for storing and retrieving LER information. This system, developed in the early 1980s and maintained under contract at the Oak Ridge National Laboratory (ORNL), contains, on an average, 150 items of information for each LER submitted since 1980 (40,000 LERs). The LER descriptive text is coded into computer-searchable sequences, with each sequence identified by categories such as components, systems, personnel errors, causes, and corrective actions. Coding the LER in sequences facilitates searches. The SCSS, given a series of failures or errors for an event or event type, can identify previous similar events to support trend analysis.

In 1992, the AEOD staff used the LER information from the SCSS database to support certain NRC activities, such as plant diagnostic evaluations, NRC senior management meetings, and activities pertaining to the cause-code performance indicator. The SCSS database is also a primary source of information for AEOD studies. The Office of Nuclear Reactor Regulation, the Office of Nuclear Regulatory Research, and the NRC regional offices use the SCSS as a source of information on operating experience.

AEOD also maintains data on LERs at the Idaho National Engineering Laboratory (INEL) to support studies for specific kinds of events and to support the NRC Performance Indicator (PI) Program. From the INEL Trends and Patterns database, AEOD obtained the PI data for reactor trips, safety system actuations, safety system failures, forced-outage rate, and equipment-forced outages per 1,000 critical hours through 1992. In addition, AEOD used the INEL databases to prepare special studies, evaluations of selected plants, and briefing packages for Commission site visits.

The Nuclear Plant Reliability Data System (NPRDS) is a proprietary database containing more than 619,000 component engineering records and 142,000 component failure records from commercial nuclear power plants. Industry provides the data to the Institute of Nuclear Power Operations (INPO), which manages and directs the development of the database. NPRDS is the NRC's primary source of industrywide component failure data. Since 1982,

Table 4.2 Percentage of LERs Submitted in 1992 by 10 CFR 50.73 Requirements

CFR Title 10 Section	Requirement	Percentage of LERs
50.73(a)(2)(i)	Technical Specifications (TS) shutdown or TS violation	41
50.73(a)(2)(iv)	Engineered safety feature actuation (including reactor trips)	34
50.73(a)(2)(v)	Real or potential loss of safety system	12
50.73(a)(2)(ii)	Unanalyzed conditions	11
50.73(a)(2)(vii)	Failures in multiple systems	2
50.73(a)(2)(x)	Internal threat	<1
50.73(a)(2)(iii)	External threat	<1
50.73(a)(2)(viii)(A)	Airborne radioactive releases	0
50.73(a)(2)(viii)(B)	Liquid effluent radioactive releases	0

AEOD has conducted a program to annually evaluate the capability of the system to meet NRC's needs for component failure and reliability data. AEOD reports its findings from these evaluations to the Commission.

4.2.2 Event Reporting Guidance

In September 1991, the staff issued a draft report, NUREG-1022, Rev. 1, "Event Reporting Systems 10 CFR 50.72 and 50.73," for public comment. In this report, the staff updated and consolidated reporting guidance issued since the 1983 publication of the rule on immediate notifications and on LERs, 10 CFR 50.72 and 10 CFR 50.73, respectively.

The public comments on the draft report indicated that the consolidation of reporting requirements was helpful, but that the clarifications included new and different guidance in major areas. The commenters expressed concerns that the guidance, contrary to the stated intent, would result in significant increases in reporting with no apparent safety benefit. The 37 comment letters generally expressed the view that draft NUREG-1022, Rev. 1, should be revised further, allowing continuing interaction with the public to reach clarifications that would benefit both the NRC and the industry. Areas of concern included apparent increased reporting of engineered safety feature actuations, events and conditions outside the plant design, operations and conditions prohibited by Technical Specifications, and internal and external threats. The staff met with interested parties on May 7, 1992, and a consensus was achieved regarding about half of the significant issues. At the end of 1992, the staff was preparing for another meeting for further discussion of the remaining issues.

4.2.3 Changes in Event Reporting Requirements

In September 1992, the NRC published a minor rule change modifying event reporting requirements pursuant to 10 CFR 50.72 and 50.73. The intent of this rulemaking was to reduce reporting requirements for certain events that have been determined to be of little or no safety significance. This rule change, which became effective in October 1992, deleted reporting requirements for a limited set of specifically defined invalid engineered safety feature actuations. The staff estimated that this rule change would reduce the number of LERs by about 150 per year among the 109 operating units: some respondents, in their comments on the proposed rule, estimated greater reductions.

4.3 Abnormal Occurrence Reporting

AEOD prepares the quarterly "Report to Congress on Abnormal Occurrences" (AOs), NUREG-0090, and the associated *Federal Register* notices and, after staff coordination, sends them to the Executive Director for Operations and, subsequently, to the Commission for review and approval. An AO may be an individual event, a generic concern, or a series of incidents that the Commission determines is significant from the standpoint of public health or safety.

Four quarterly AO reports to Congress were published during calendar year (CY) 1992. These reports were for the fourth quarter of CY 1991 (NUREG-0090, Vol. 14, No. 4) and the first, second, and third quarters of CY 1992 (NUREG-0090, Vol. 15, No. 1; Vol. 15, No. 2; and Vol. 15, No. 3; respectively). The fourth-quarter report for CY 1992 (NUREG-0090, Vol. 15, No. 4) was published in the first part of CY 1993.

During CY 1992, three AOs occurred at nuclear power plants. Table 4.3 shows the number of AOs occurring at nuclear power plants for each calendar year since 1987. The number of AOs remained low throughout this period. A summary of 1992 reactor AOs is provided in Appendix B to this report.

A summary of CY 1992 nonreactor AOs (at both NRC and Agreement States licensee facilities) is provided in Appendix D to the AEOD Annual Report on Nonreactors (NUREG-1272, Vol. 7, No. 2).

Table 4.3 U.S. Nuclear Power Plant Abnormal Occurrences (AOs) per Year

Year	No. of AOs
1987	3
1988	3
1989	4
1990	1
1991	0
1992	3

4.4 Evaluation of the Nuclear Plant Reliability Data System

AEOD continued its annual evaluation of NPRDS and the capability of the system to meet NRC needs for component reliability data. In July 1992, AEOD issued its report "Nuclear Plant Reliability Data System (NPRDS)" as a Commission paper designated SECY-92-248. The 1991 NPRDS evaluation focused on issues identified by the NRC staff during routine use of the database and on an INPO/industry review group's recommendations for changes to NPRDS.

In its evaluation, AEOD reported that NPRDS data continued to be to a useful source of supplemental information for making general assessments of component failures and problems (e.g., assessing industry's experience with a specific component or model, deciding whether a generic problem exists, preparing for inspections, and studying component aging). However, NPRDS was of limited use when component failure rate and reliability data were required (such as for resolving generic issues, performing probabilistic risk assessments (PRAs), conducting reliability analyses, and developing risk-based Technical Specifications and risk-based performance indicators). For these areas, users wanted the types of reliability and unavailability data typically needed for PRAs. One limitation in all applications of NPRDS noted in the report was the continued need for more complete reporting of failures.

In response to the NRC staff's concerns outlined in SECY-91-407, a supplement to the 1990 NPRDS evaluation report, and to similar views expressed by other organizations, INPO established in late 1991 a task force, the Industry Group To Review Long-Term Equipment Performance Data Needs. INPO invited NRC management to participate in this group. In March 1992, the group presented to INPO its final recommendations for changes to NPRDS. Those final recommendations for PRA applications encompassed most of NRC's needs (demand data, operating hours, failures on demand, failures while running, unavailability data, and out-of-service records for like-for-like component replacements). In its 1991 NPRDS evaluation report, AEOD informed the Commission that the NRC staff was generally satisfied that the industry group's recommendations for improvements in NPRDS would meet the current and projected needs of both the industry and the NRC.

During the summer and fall of 1992, the industry group's recommendations were presented to the NPRDS Users Group, the Analysis and Engineering Industry Review Group, and lastly to INPO management for final approval. At the end of 1992, INPO wrote to AEOD informing it of (1) the changes that were approved by INPO for implementation during the next 2 years, (2) the recommendations that were not approved, and (3) the recommended changes that were still being considered.

INPO approved adding room coolers, dampers, and cooler isolation valves for rooms with certain safety-related equipment to the reportable scope of components and adding about 250 application codes for important components. The new recommendations also encourage the reporting of any documented failure-cause analysis if it has been performed and the reporting of information on the as-found conditions of components in the failure reports. A recommendation that was approved by INPO will be useful to NRC aging studies and will provide more accurate data on inservice life and mean time between failure by reporting like-for-like component replacement data. Now, such replacements are not reportable to NPRDS. INPO has approved reporting out-of-service and new engineering records for like-for-like component changeouts and describing the condition (new or refurbished/rebuilt) of each component when placed in service. In addition, INPO has approved expanding the system effect codes to include effects of failure on mechanical/electrical train and/or on instrument channel and expanding the failure-cause description codes. The approved recommendations also included some deletions from the reportable scope: control rods; heater failures that do not affect operation of a reportable component; valves in vent, drain, and test lines less than 7.6 cm (3 in.) in diameter in non-safety-related systems; main turbine blade cracking that does not affect operation of the turbine; and postulated design failures. INPO will continue initiatives begun in 1991 and 1992 to improve the accuracy and consistency of failure reporting.

INPO did not approve the recommendation for reporting the actual time to repair a failure by adding maintenance start and stop time fields to reports. INPO is still reviewing changes that were proposed in support of PRA work and maintenance rule implementation. These are probably the most important recommendations to NRC users. AEOD wrote to INPO expressing its concern that these recommended changes were being put off indefinitely. AEOD noted that these recommended changes would address data needs of the industry and the NRC in the following areas: PRA, individual plant examinations, reliability, maintenance rule implementation, and risk-based regulation.

AEOD will continue to work with INPO through the NPRDS Users Group to ensure that the changes made to NPRDS, to the extent feasible and practical, meet NRC and industry needs.

4.5 International Exchange of Information

Since the United States first began using commercial nuclear power, around the late 1950s, the growing use of nuclear power outside the United States and the world's growing recognition of the worldwide impact of nuclear events, regardless of the country of origin, led to the development of cooperative agreements by which information on operating nuclear power plant events is shared by the international community. After the accident at Three Mile Island (TMI) Nuclear Station in 1979, international agencies developed an Incident Reporting System (IRS) for the exchange of information on events of particular safety significance. Consistent with this spirit of international cooperation, AEOD continued its efforts to maintain and improve the exchange of information and ideas about operational experience with foreign communities. The results of these efforts have provided valuable data for AEOD studies and valuable support for regulatory actions.

4.5.1 Review of Foreign Events

The NRC received and reviewed 158 IRS reports from foreign reactors during 1992. AEOD routinely reviews these reports submitted to the IRS from nuclear power reactors in member countries of both the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) in addition to information obtained through bilateral exchange programs with over 20 countries.

Examples of issues of importance to domestic reactors that were highlighted by these information sources in 1992 are (1) discovery of reactor vessel control rod penetration cracking, (2) BWR wetwell pump suction strainers clogged by insulation, (3) gate valve pressure locking, and (4) water intrusion into the control room. AEOD has developed important findings that were partly based on information about foreign events in several of its studies.

AEOD disseminates, within the NRC, reports of selected foreign reactor events of particular interest to the staff regulating the U.S. program. It identifies significant foreign events that could be applicable to U.S. plants and transmits reports of these events to interested parties within the NRC.

4.5.2 U.S. Reports Submitted for the Incident Reporting System

At the end of 1992, 109 of the more than 410 nuclear power plants in commercial operation worldwide were located in the United States. In support of the international exchange of operating experience, AEOD prepared and submitted 55 reports to the NEA IRS. The United States and 12 other countries (Belgium, Canada, Finland, France, Germany, Italy, Japan, the Netherlands, Spain, Sweden, Switzerland, and the United Kingdom) are members of the NEA. The NEA member countries also participate in the IAEA IRS, which broadens the operational experience base to include all countries with nuclear programs, except Taiwan. The reports submitted by the United States addressed individual operational events and various generic concerns identified within the U.S. operational experience feedback program. These reports were based primarily on information communicated to U.S. plant operators by NRC information notices, bulletins, and generic letters on such subjects as failures, and potential failures, of piping caused by erosion/corrosion, testing and surveillance of motor-operated valves, pressure locking of motor-operated flexible wedge gate valves, and level instrumentation inaccuracies caused by reference level changes during rapid depressurization.

In addition to these programs and as part of NRC's overall international program, AEOD exchanges information and ideas on a variety of topics of international interest. For example, the AEOD staff provided assistance to foreign countries and to the IAEA in a number of safety-related areas. AEOD also serves as the principal U.S. technical representative on reactor operating experience to NEA's Committee on the Safety of Nuclear Installations (CSNI) Principal Working Group 1 (PWG-1), "Operating Experience and Human Factors."

At the PWG-1 meeting, the members endorsed continued use of the modified software for future update and transmittal of the IRS database being maintained by NRC at ORNL. AEOD presented information on the types of backfitting in the United States, as well as backfitting items, effected in 1991 as a result of operating experience, emphasizing that the licensing basis for each plant takes precedence over current requirements. Other topics discussed by AEOD were pressure locking of double-disk and flexible-wedge gate valves, the Accident Sequence Precursor (ASP) Program, and several recent operating events. In its presentation on the ASP Program, AEOD emphasized that the most important event identified this year was that at Shearon Harris Unit 1, where tests during a refueling outage revealed that both relief valves on the piping to the refueling water storage tank had failed due to water hammer. This made the high-head safety injection system unavailable, making the conditional core damage probability about 0.006. Other operating events discussed by AEOD were loss of shutdown cooling at Prairie Island Unit 2; depressurization, reactor scram, and emergency core cooling system injection at Crystal River Unit 3; failure of a pressurizer safety valve to reseat at Fort Calhoun; and the August 1991 event involving transformer failure and subsequent common-mode loss of instrument power at Nine Mile Point Unit 2.

At this meeting, members praised the continuing cooperation between the NEA and IAEA and the significant use made of IRS data by the member countries of the two agencies. The group agreed that the future PWG-1 agenda will include general discussions on technical subjects of interest to all participants and that it will continue to provide feedback to members on the usefulness of IRS data and other issues of most concern to the member countries. The group stressed the importance of continuing NEA and IAEA cooperation in the use of a unified IRS database by the world community.

At the meeting, a number of foreign countries indicated that the U.S. reports had proved valuable to their programs. Foreign representatives cited various U.S. experiences with broad generic implications that resulted in actions being taken in their country's nuclear programs. The United States also cited various foreign IRS reports that were pertinent to ongoing U.S. studies, provided worthwhile information, and were particularly relevant to understanding several incidents experienced at U.S. nuclear power plants. These exchanges continue to be important to nuclear power safety programs worldwide.

4.5.3 Maintenance and Operation of the Incident Reporting System Database

In 1989, OECD and the NRC, as the responsible U.S. Government agency, finalized an agreement whereby NRC assumed the responsibility for managing and operating the NEA IRS database. This database contains information on nuclear power plant operating experience derived from reports from NEA members and also those submitted to the IAEA by its member countries under similar reporting guidelines. Thus, the database contains information on worldwide nuclear power plant experience. In 1992, this activity continued at ORNL under an NRC contract, and includes the processing and encoding of all IRS events received from the NEA and the quarterly transmittal of multiple copies of a computerized update of the database to the NEA for further distribution to the member countries and to the IAEA.

In September 1992, during its eleventh meeting, PWG-1 met jointly with representatives from IAEA countries to discuss, in addition to significant operating events, the recently completed modifications to the IRS database. Earlier in 1992, as a contribution to the world nuclear community, NRC funded the development of the IRS database at ORNL. The specifications were formulated by the AEOD staff with input from the NEA and the IAEA representatives. The purpose of these improvements was to facilitate efficient data searches by various fields and/or key words, and search-within-search capability. In August, a set of ready-to-use trial diskettes was forwarded to the NEA and the IAEA member countries. The modified IRS database software contains all the short-term improvements previously requested by PWG-1. These improvements have made the database more user friendly, increased text and date field search capabilities, as well as improved print and display formats. At the PWG-1 meeting, the members endorsed continued use of the modified software for future update of the IRS database.

4.6 Data From the Nuclear Regulatory Commission's Operations Center for 1992

The NRC's Operations Center in Bethesda, Maryland, provides a focal point for NRC communications with Commission licensees, State agencies, and other Federal agencies about operating events in the commercial nuclear sector. The Operations Center is staffed 24 hours a day by an NRC headquarters operations officer (HOO), who is a reactor systems specialist trained to receive, evaluate, and respond to events reported to the center.

As shown in Table 4.4, the Operations Center received 2,276 notifications in 1992, primarily from nuclear power plant licensees, about events that must be reported under NRC's prompt reporting requirements. A small subset of these notifications involved events classified by licensees into one of the four classes of emergencies: Unusual Event, Alert, Site Area Emergency, and General Emergency. The remainder of the notifications involved events that did not meet the threshold for classification.

Table 4.5 shows the number of each type of event classified under licensee emergency plans for the past 5 years. An Unusual Event represents a condition that is of no immediate threat to the public health, and an Alert indicates substantial actual or potential degradation of plant safety. A Site Area Emergency or a General Emergency indicates a major failure of one or more systems required for public safety or an event with the potential for a major offsite radiological release.

The increase in the number of Alerts in 1992 was partly attributed to an increase in loss-of-control-room-annunciator events (eight in 1992 compared to two in 1991) and an increase in events at fuel facilities (three in 1992 compared to one in 1991). Table 4.6 lists the events reported in 1992 that were classified as an Alert or higher. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," was often the basis cited in licensee emergency action level schemes for classifying a loss of annunciators as an Alert. Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," issued in August 1992, endorses a Nuclear Management and Resources Council (NUMARC) methodology, NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," as an alternative to NUREG-0654. Many of the loss-of-annunciator events listed in Table 4.6 would be classified as Unusual Events using the NUMARC classification guidance. Many licensees are revising their emergency plans and associated emergency action level schemes to conform with the NUMARC guidance.

Exercises are held periodically to ensure that NRC's and the licensee's response organizations are proficient in dealing with each type of emergency. In 1992, NRC headquarters and regional offices participated in emergency planning exercises with the River Bend power plant (February 26, 1992), the Quad Cities power plant (April 29, 1992), the Vogtle power plant (May 19, 1992), the Arkansas Nuclear One power plant (June 24, 1992), and the Maine Yankee power plant (October 21, 1992). The following photographs (Figures 4.1 through 4.4) show participants in a typical exercise as they evaluate plant status and licensee actions to determine what NRC response is required and what guidance to offer State and local agencies.

Actions taken by the NRC HOO in response to these notifications of events ranged from a computer or log entry, followed by appropriate notifications, to establishing emergency conference calls among the HOO, the licensee, and the senior NRC regional and headquarters staff members. For very significant events, these conference calls would result in activation of the agency's Incident Response Plan. In the Standby mode, the NRC closely monitors the event and prepares to rapidly enter the Initial Activation mode, if necessary. The NRC entered the Standby mode once in 1992, for an event at Fort Calhoun involving a stuck-open primary relief valve. Under some circumstances the NRC may send specialists to the Operations Center to monitor an event, even though no safety consequences are projected on the basis of existing plant conditions. The NRC traditionally referred to this informal response mode as "enhanced readiness." On July 1, 1992, the NRC formalized this enhanced readiness

Table 4.4 Total Events Reported to the Operations Center in 1992

	Power Reactor	Fuel Facility	Non-Power	Hospital	Transport/ Materials	Other	Total
Non-Emergency Event	1,886	59	7	58	51	59	2,120
Unusual Event	130	3	2	0	0	0	135
Alert	17	3	0	0	0	0	20
Site Area Emergency	0	1	0	0	0	0	1
General Emergency	0	0	0	0	0	0	0
Total	2,033	66	9	58	51	59	2,276

Table 4.5 Classification of Events Under Licensee Emergency Plans—1988–1992

Event	1988	1989	1990	1991	1992
Unusual Event	212	197	151	170	135
Alert	6	13	10	9	20
Site Area Emergency	0	0	1	2	1
General Emergency	0	0	0	0	0

state as the "Monitoring mode." As shown in Table 4.6, the NRC entered the enhanced readiness or Monitoring mode for nine events in 1992. Some of these events were the result of a common cause, such as Hurricane Andrew, which affected or had the potential to affect three different sites.

From the exercises and events that occurred this year, many issues that could affect response activities were identified. Some of the more significant concepts include

- the need for backup communications channels which are not susceptible to damage from natural phenomena (e.g., hurricanes)
- the importance of control room annunciation for plant operations
- the importance of emergency planning for toxic chemical hazards
- the impact that insurance compensation could have on State and local government response capabilities

These issues will require analysis and review. As a result of this evaluation, the agency's response procedures may require further changes.

4.7 Incident Investigation Program

The Incident Investigation Program (IIP) ensures that NRC investigations of significant events are timely, thorough, well coordinated, and formally administered. The scope of the IIP includes investigations of significant operational events involving reactor and nuclear materials activities licensed by the NRC. Under the IIP, the NRC responds to an operational event according to its safety significance. To investigate an event of potentially major safety significance, the Executive Director for Operations (EDO) establishes an incident investigation team (IIT) to investigate an event of less safety significance, the cognizant NRC regional administrator may establish an augmented inspection team (AIT). Both IITs and AITs are assigned to determine the circumstances and causes of an operational event and to assess the safety significance of the event so that appropriate followup actions can be taken.

Administration of the NRC's incident investigation activities is prescribed in NUREG-1303, "Incident Investigation Manual." As stated in that report, AEOD is responsible for the overall administration of the IIP, while NRR is responsible for maintaining the procedures for an AIT response. During 1992, no reactor IITs were established. The status of staff actions that the EDO assigned to various NRC offices associated with IIT reports is given in Appendix F.

Table 4.6 Site Area Emergency and Alert Events Reported in 1992 at NRC-Licensed Facilities

Name	Event No.	Date	Description	Duration*	Response
Site Area Emergency					
Fuel Facility					
Sequoyah Fuels	24616	11/17/92	Release of nitric acid fumes off site	1 hr, 3 min	Monitoring
Alerts					
Power Facilities					
Quad Cities 2	22681	01/25/92	Loss of control room annunciators	0 hr, 0 min	NA
Quad Cities 1	22818	02/14/92	Loss of control room annunciators	1 hr, 14 min	NA
Nine Mile Point 2	23078	03/23/92	Loss of control room annunciators, loss of offsite power	3 hr, 1 min	Enhanced
Dresden 3	23173	04/04/92	Loss of control room annunciators	7 hr, 30 min	NA
Quad Cities 1	23190	04/07/92	Loss of control room annunciators	0 hr, 14 min	NA
Quad Cities 1	23211	04/10/92	Loss of control room annunciators	0 hr, 5 min	NA
Robinson 2	23241	04/15/92	CO_2 release in cable penetration room	1 hr, 18 min	Enhanced
Oyster Creek	23394	05/03/92	Fire in switchyard, loss of offsite power	16 hr, 15 m	Enhanced
Palo Verde 3	23396	05/04/92	Loss of control room annunciators	15 hr, 3 min	Enhanced
Zion 2	23448	05/13/92	Spray-down of containment	0 hr, 31 min	NA
Sequoyah 2	23622	06/10/92	Excessive reactor coolant system leakage ($>$265 L/min ($>$70 gal/min))	0 hr, 0 min	NA
Dresden 2	23768	07/01/92	Loss of control room annunciators	7 hr, 35 min	Monitoring
Fort Calhoun	23790	07/04/92	Stuck-open primary relief valve	17 hr, 10 min	Standby
Peach Bottom 3	23791	07/04/92	Loss of offsite power as a result of transformer fire	4 hr, 15 min	Monitoring
Crystal River 3	23988	08/04/92	Fire in a battery charger lasting $>$10 minutes	0 hr, 24 min	NA
Turkey Point 3 & 4	24104	08/23/92	Hurricane Andrew, loss of offsite power, degraded fire protection	7 days, 15 hr, 57 min**	Monitoring
Robinson 2	24338	09/30/92	Release of CO_2 into plant vital area	5 hr, 26 min	NA
Fuel Facilities					
Sequoyah Fuels	23383	05/01/92	Release of UF_6 at the facility (in the UF_4 building)	2 hr, 34 min	Enhanced at region
B&W Fuels, Lynchburg	23879	07/15/92	Sounding of emergency evacuation alarm	1 hr, 10 min	NA
B&W Fuels, Lynchburg	24086	08/21/92	Sounding of radiation alarms	1 hr, 32 min	NA

*Time from event time to termination of emergency class.
**From declaration of Unusual Event through Alert until downgraded to Unusual Event.
Notes: The NRC established a new response mode, called monitoring mode, on July 1, 1992.
NA means not applicable.

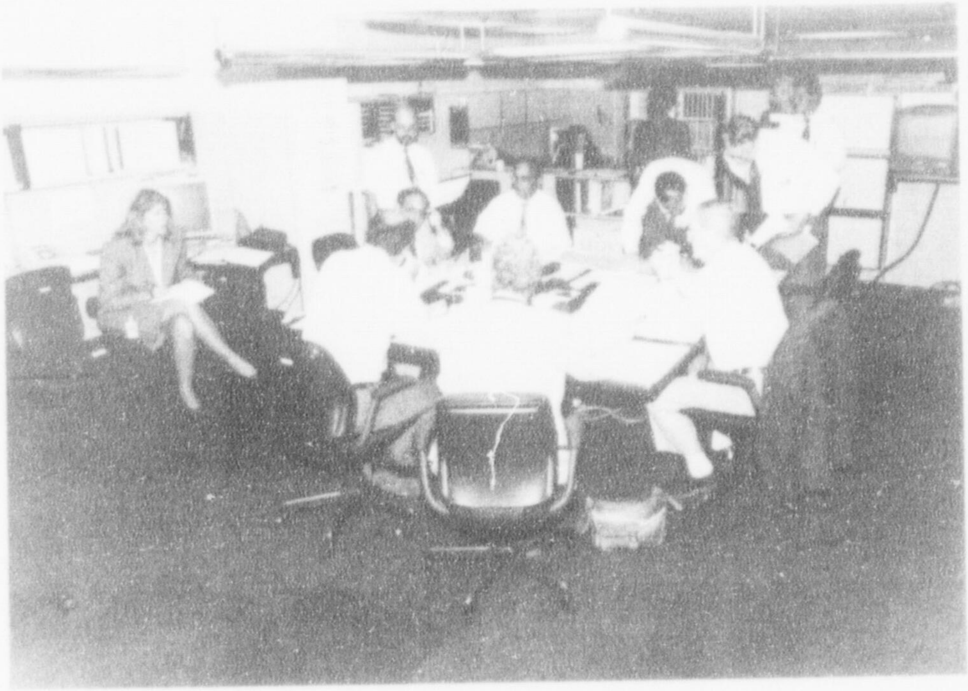

Figure 4.1 Reactor Safety Team

Figure 4.2 Protective Measures Team

Figure 4.3 Protective Action Support Team Member
(U.S. Department of Agriculture)

Figure 4.4 Executive Team

4.8 Augmented Inspection Team

During 1992, nine AIT inspections were conducted at power-generation facilities (see Table 4.7). These AITs helped to improve plant safety by conducting detailed investigations of the problems experienced at the plant and identifying their root causes. NRC regional administrators are responsible for identifying needed actions on the basis of the AIT findings. In addition, the directors of NRR and NMSS are responsible for reviewing AIT reports for generic safety implications, initiating followup actions, and tracking issues affecting more than one plant, as appropriate. AEOD independently reviews AIT reports to provide additional assurance that potential generic lessons are learned and communicated to the industry. Thus, industrywide safety is enhanced by including the significant lessons learned from the AIT inspections, along with those from engineering studies and review of operating experience, in generic communications to licensees.

Examples of lessons learned and communicated to licensees from events investigated by AITs in 1992 include the following:

- On February 21, 1992, at Nine Mile Point Unit 1, the ultimate heat sink was inadvertently isolated when all gates to the service water system inlet bay were closed. The event occurred during post-maintenance testing while the reactor was shut down.

  As a result of this AIT investigation, the NRC issued Information Notice (IN) 92-49, "Recent Loss or Severe Degradation of Service Water Systems," July 2, 1992. This IN alerted licensees to problems involving the loss or potential loss of safety-related heat-transfer capability in service water systems.

- On August 15, 1992, Washington Nuclear Power Unit 2 experienced power oscillations during startup. On recognizing the power oscillations, the plant operators manually initiated a reactor scram.

  As a result of this AIT investigation, the NRC issued IN 92-74, "Power Oscillations at Washington Nuclear Power Unit 2," November 10, 1992. This IN alerted licensees to problems associated with power oscillations in a power-flow operating region where instability had not been specifically predicted.

4.9 Diagnostic Evaluation Program

The Diagnostic Evaluation Program (DEP) provides for an independent assessment of licensee annunciators performance at reactor facilities selected by the EDO. This assessment augments information provided by the Systematic Assessment of Licensee Performance (SALP) Program, the Performance Indicator Program, and the various inspections performed by the NRC headquarters and regional office staffs. The assessment is independent in the sense that the administration and management of the program are independent of NRC's licensing, inspection, and enforcement processes.

When a diagnostic evaluation (DE) is approved for a specific reactor facility, the EDO authorizes and establishes a diagnostic evaluation team (DET). The DET consists of expert technical staff members from headquarters and the regional offices, experienced resident inspectors, and contractors, if appropriate. The DET manager and team members selected will not have had recent significant involvement in the licensing, inspection, or enforcement process at the facility where the DE is to be performed. The evaluation process involves observations of plant and corporate activities, indepth technical reviews, employee interviews, equipment walkdown inspections, and programmatic reviews in a number of functional areas important to safety. Areas evaluated generally include maintenance, surveillance and testing, management effectiveness, operations, engineering, and quality programs.

No DEs were performed during 1992. A DET was established in early 1993 to evaluate performance at South Texas Project Units 1 and 2. Results from the South Texas DE will be documented in the 1993 AEOD Annual Report.

4.10 Committee To Review Generic Requirements

The Committee To Review Generic Requirements (CRGR) reviews all generic requirements proposed by the NRC staff that involve one or more classes of power reactors. The CRGR consists of senior managers from various headquarters program offices and, on a rotational basis, from one of the NRC regional offices. While performing the CRGR

Table 4.7 Reactor Events for Which AIT Inspections Were Conducted in 1992

Event Date	Plant	Event
02/21/92	Nine Mile Point 1	Inadvertent isolation of the ultimate heat sink
03/23/92	Nine Mile Point 2	Loss of offsite power and control room annunciation
05/04/92	Palo Verde 3	Loss of plant annunciator and alarm systems
07/03/92	Ft. Calhoun 1	Loss of reactor coolant greater than 151 L/min (40 gal/min)
08/15/92	Washington Nuclear Power 2	Unusual event and manual scram due to power oscillations
08/27/92	LaSalle 2	Scram without feedwater trip
10/17/92	Callaway	Unreported failure of all control room annunciators
10/19/92	Oconee 2	Reactor trip with loss of offsite power
12/13/92	Salem 2	Unreported failure of control room annunciators

review function, a CRGR member expresses an individual professional opinion about each issue considered rather than representing the view of his or her respective office. The members of the CRGR determine whether proposed new generic requirements have sufficient merit in terms of safety and are justified in terms of cost (where appropriate) before reaching a consensus recommendation about each issue considered. Each independent CRGR recommendation is given to the EDO to consider. The Director, AEOD, serves as the CRGR Chairman, and the AEOD staff provides support for all the Committee's activities. The Director, AEOD, and the CRGR staff also oversee plant-specific activities of the NRC staff in the headquarters program offices and the regional offices.

The CRGR held 22 meetings (including 1 public meeting) in 1992. During these meetings, it considered the following 34 issues:

- possible reduction or elimination of existing NRC regulations without reducing safety
- proposed rule to decouple site-suitability issues from plant design in siting nuclear power plants
- supplement to a generic letter on determining seismic adequacy of equipment in older nuclear power plants
- generic letter on installation of digital-based safety systems under 10 CFR 50.59
- revision to a regulatory guide on recording and reporting occupational radiation exposure
- rule amendment (proposed and final) to supplement the list of cask designs approved for use under a general license for dry storage of power reactor spent fuel
- supplement to a generic letter to relax existing NRC staff position on life testing of reactor trip breakers
- revision to a regulatory guide on quality assurance program requirements
- final rule amendment to incorporate periodic ASME Code update and to specify augmented reactor vessel weld inspections and containment valve leakage test requirements
- EDO questions regarding staff proposal for rulemaking on the reactor coolant pump seal issue (and related station blackout topics)
- generic letter on augmented inspection of General Electric Company Mark I and Mark II containments for drywell and torus corrosion
- proposed rule amendment involving both increases in and reductions of existing fitness-for-duty program requirements
- regulatory guide on radiation dose to the embryo or fetus
- regulatory guide on monitoring criteria and methods for calculating occupational dose
- regulatory guide on planned special exposures
- proposed relaxation of a regulatory guide position on Category I neutron flux monitoring systems in BWRs
- advance notice of proposed rulemaking on severe accident plant performance criteria

- proposed rule amendment relating to emergency response exercise requirements
- proposed regulatory guide, standard review plan, and branch technical positions relating to license renewal
- generic letter on effects of noncondensible gases on the accuracy of water level instrumentation in BWRs
- consideration of improved standard technical specifications for trial use in existing operating plants
- supplement to a bulletin regarding loss of fill oil in Rosemount transmitters
- generic letter on piping and use of combustible gases in vital areas of nuclear power plants
- generic letter on availability and adequacy of design-basis information
- proposed rule amendment on reactor operator requalification examination requirements
- generic letter on Thermo-Lag fire barriers
- proposed rule amendment to require submittal of nuclear transaction data in computer-readable form
- proposed regulatory guide endorsing use of industry (NUMARC) guidance for implementing the maintenance rule
- proposed revision to NRC regulatory analysis guidelines
- generic letter on allowed modifications of administrative controls in existing technical specifications relating to emergency and security plans
- safety analysis of topical report on allowed relaxation of existing technical specification limits on steam relief valve setpoints
- proposed rule amendment to incorporate by reference Subsections IWE and IWL of Section XI, ASME Code
- proposed rule amendment to allow reduced random test rate in licensees' fitness-for-duty programs
- generic letter regarding allowed modification of existing technical specifications to reduce surveillance testing during power operations

During January–April 1992, the CRGR conducted a special review of NRC regulations to determine which, if any, existing NRC requirements could be reduced or eliminated without reducing safety. The special review was conducted at the direction of the Commission in response to a request by the President. The CRGR held six meetings, including one public meeting, to consider this matter. As a result of this effort, eight areas were identified for rulemaking action that could significantly reduce regulatory burden without affecting the safety of nuclear power plants. The Commission approved expedited rulemaking in each of the eight areas.

Appendix A

Data on Plant Operational Experience

A–1 Performance Indicator Program and Data

A–2 Other Plant Operational Experience Data

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Tables (cont.)

Appendix A-1

Performance Indicator Program and Data

Appendix A-1

Performance Indicator Program and Data

Introduction

The NRC program for monitoring performance indicators (PIs) for operating commercial nuclear power reactors includes the following eight indicators: (1) the number of unplanned automatic scrams while a reactor is critical, (2) the number of selected safety system actuations, (3) the number of significant events, (4) the number of safety system failures, (5) forced outage rates, (6) the number of equipment-forced outages per 1000 commercial critical hours, (7) the collective radiation exposure per plant, and (8) cause-code trends.

The data for significant events are provided by the NRC's Office of Nuclear Reactor Regulation (NRR), and the data for collective radiation exposure are obtained from the Institute of Nuclear Power Operations (INPO). The data for cause-code trends are obtained from the Sequence Coding and Search System database maintained at the Oak Ridge National Laboratory (ORNL). The data for the remaining five PIs are obtained from Trends and Patterns databases maintained at the Idaho National Engineering Laboratory (INEL).

This appendix provides the background of the development of the PI program and the definitions of the PIs. It also provides tables containing PI data for 1991 and 1992 and a listing of the significant events that occurred during 1992.

Background

Since May 1985, an interoffice task group has worked to develop an NRC program for using quantitative indicators of nuclear power plant safety performance. In July and August of 1986, the group conducted a trial program for 50 operating plants, testing 17 prospective performance indicators. For the most part, this trial program used data through calendar year 1985. The group then selected eight performance indicators as candidate PIs for initial implementation. After considering industry comments, the staff deleted one of the candidate PIs, the corrective maintenance backlog.

In October 1986, the NRC prepared a prototype report by expanding the trial program to include data for the first half of 1986 on 100 operating reactors. The staff discussed the recommended program, the task group report, and the prototype report in a Commission paper designated SECY-86-317, "Performance Indicators," dated October 28, 1986. The staff briefed the Commission on the recommended program in November 1986. The Commission approved the implementation of the program in December 1986, instructing the staff to delete the enforcement action index from the set of indicators. Since February 1987, the AEOD staff has provided quarterly PI reports to the Commission and to NRC senior managers. The reports are also placed in the NRC's Public Document Room. Beginning with the issuance of the PI report for the fourth quarter of calendar year 1989, the staff routinely provides plant-specific information extracted from each PI report to licensee managers.

The Commission approved the use of cause-code trends in the PI report in SECY 89-211, dated August 10, 1989. At that time, the Commission did not approve the use of cause-code deviations, but instructed the staff to assess the validity of comparing plants to their nuclear steam supply system (NSSS) average, in light of plant-to-plant variations within NSSS groups. Early in the effort to develop suitable peer groups for comparison of plant performance, it was found that a plant's operating phase could also have an effect on the occurrence of reportable events. To address this issue, the staff initiated a study to identify phases of operation in which the frequency of reportable events varies significantly. The result was the development of the operational cycle/peer group methodology. An interoffice task group was formed and a trial program implemented in 1992 to assess the proposed changes. This enhancement to the PI program was sent to the

Commission for approval in SECY-92-425, "Performance Indicator Program—Peer Group and Operating Cycle Phase Enhancements," dated December 23, 1992.

Definitions of the Indicators

Automatic Scrams While Critical

This indicator is the number of all unplanned automatic scrams that occur while the reactor is critical; a reactor scram means any actuation of the reactor protection system that results in control rod motion. The PI program also monitors the number of automatic scrams that occur while the reactor is critical at power levels equal to or below 15 percent and the number of automatic scrams per 1000 critical hours that occur while the reactor is above 15-percent power.

Safety System Actuations

This indicator is the number of manual and automatic actuations (including safety system logic actuations) of certain emergency core cooling systems (ECCSs) plus actuations of the emergency ac power systems that were caused by loss of power to a vital bus.

For pressurized-water reactors, only actuations of the high-pressure injection system, the low-pressure injection system, or safety injection tanks are counted. For boiling-water reactors, only actuations of the high-pressure coolant injection system, the low-pressure coolant injection system, the high-pressure core spray systems, or the low-pressure core spray system are counted. Actuations of the reactor core isolation cooling system are not counted.

Significant Events

This indicator is the number of events that the NRC staff identifies as meeting certain selection criteria. Examples of these events include the degradation of important safety equipment; an unexpected plant response to a transient or a major transient itself; a reactor trip with complications; or a degradation of fuel integrity, the primary coolant pressure boundary, or important associated structures.

Safety System Failures

This indicator includes any event or condition that could prevent the fulfillment of the safety function of structures or systems. The AEOD staff monitors 26 safety systems, subsystems, and components for this indicator. If a system consists of multiple redundant subsystems or trains, failure of all trains constitutes a safety system failure. Failure of one of two or more trains is not counted as a safety system failure.

Forced Outage Rate

This indicator is the number of forced outage hours in a period divided by the sum of the forced outage hours and the generator on-line hours. This indicator is used only for plants that are in commercial operation.

Equipment-Forced Outages per 1000 Commercial Critical Hours

This indicator is the number of forced outages caused by equipment failures per 1000 critical hours of commercial reactor operation. It is the inverse of the mean time between forced outages caused by equipment failures. This indicator is used only for plants that are in commercial operation.

Collective Radiation Exposure

This indicator is the total radiation dose accumulated by unit personnel. With the exception of the Indian Point and Millstone nuclear plants, unit values at multiunit sites are obtained by dividing the station total by the number of units contributing to the exposure. The Indian Point and Millstone sites report individual unit values.

Cause-Code Trends

This indicator captures the plant's trends for administrative control problems; licensed operator errors; other personnel errors; maintenance problems; design, construction, installation, or fabrication problems; and equipment failures (electronic piece-part or environmentally related failures).

Table A–1.1 Number of Automatic Scrams While the Reactor Is Critical—Quarterly PI Data

Plant Name	Calendar Year–Quarter							
	91–1	91–2	91–3	91–4	92–1	92–2	92–3	92–4
Arkansas 1	1	1	0	0	0	0	0	0
Arkansas 2	1	0	0	0	0	0	0	0
Beaver Valley 1	0	0	2	0	0	0	0	1
Beaver Valley 2	0	0	0	1	0	0	0	0
Big Rock Point	0	0	0	0	0	2	0	0
Braidwood 1	0	0	0	1	0	0	0	0
Braidwood 2	0	0	1	1	2	0	1	1
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	1	1	0	1	1	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	1	0	1	0	2	0	0	0
Brunswick 2	1	0	0	1	1	0	0	0
Byron 1	0	0	0	0	1	0	0	0
Byron 2	0	0	0	1	0	0	0	0
Callaway	0	0	0	1	1	1	1	0
Calvert Cliffs 1	0	0	0	1	0	0	0	1
Calvert Cliffs 2	0	1	0	0	0	0	1	0
Catawba 1	0	1	2	1	0	0	0	0
Catawba 2	0	1	0	0	1	0	0	1
Clinton 1	0	0	0	0	2	0	0	0
Comanche Peak 1	3	0	1	0	1	1	0	0
Cook 1	0	1	0	0	0	0	0	1
Cook 2	1	0	1	1	0	0	1	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	0	0	0	3	1	0	0	1
Davis–Besse	0	0	0	0	1	0	0	0
Diablo Canyon 1	1	2	0	0	1	1	0	0
Diablo Canyon 2	0	0	0	0	0	0	0	0
Dresden 2	1	1	1	1	0	0	0	0
Dresden 3	0	0	1	0	0	0	0	2
Duane Arnold	1	1	0	0	0	0	1	1
Farley 1	0	2	2	0	0	0	0	1
Farley 2	0	2	1	0	1	3	0	0
Fermi 2	0	1	0	0	0	0	0	0
FitzPatrick	0	0	0	0	0	0	0	0
Fort Calhoun	0	0	0	0	0	1	2	0
Ginna	0	0	0	0	2	0	0	0
Grand Gulf	0	3	2	1	0	2	1	0
Haddam Neck	0	0	0	0	1	0	0	0
Harris	0	1	0	0	0	0	2	0

Table A-1.1 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	3	0	2	0	1	1	1	0
Hatch 2	2	0	0	0	0	1	0	1
Hope Creek	1	1	0	0	0	0	0	0
Indian Point 2	1	0	1	0	1	1	1	0
Indian Point 3	2	0	0	0	0	0	2	0
Kewaunee	0	0	0	1	0	0	1	0
LaSalle 1	0	1	0	0	1	0	0	0
LaSalle 2	0	0	1	1	0	0	1	1
Limerick 1	0	1	0	0	0	0	0	0
Limerick 2	0	0	0	0	0	0	0	0
Maine Yankee	0	2	0	2	0	0	0	0
McGuire 1	2	0	0	0	0	1	1	0
McGuire 2	0	0	0	1	1	2	2	0
Millstone 1	0	0	0	0	0	0	0	1
Millstone 2	1	0	0	0	0	0	0	0
Millstone 3	0	1	0	0	0	0	0	2
Monticello	1	2	1	0	0	0	0	0
Nine Mile Pt. 1	1	0	2	1	1	2	1	0
Nine Mile Pt. 2	0		1	2	0	0	1	1
North Anna 1	0	0	1	0	0	0	0	0
North Anna 2	0	0	1	0	1	0	1	0
Oconee 1	0	1	0	1	0	2	0	1
Oconee 2	0	0	0	0	0	0	0	1
Oconee 3	0	1	1	1	2	1	1	0
Oyster Creek	0	0	1	0	0	2	2	0
Palisades	0	0	2	1	0	0	4	1
Palo Verde 1	0	0	1	1	0	0	1	1
Palo Verde 2	0	0	1	0	2	0	0	1
Palo Verde 3	0	0	0	2	1	0	0	0
Peach Bottom 2	0	0	1	0	0	1	2	0
Peach Bottom 3	1	0	1	0	0	1	1	1
Perry	0	0	0	0	0	0	1	0
Pilgrim	0	0	0	0	0	0	0	2
Point Beach 1	0	2	0	0	0	0	0	1
Point Beach 2	0	0	0	1	0	0	0	0
Prairie Island 1	0	0	1	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0
Quad Cities 1	0	0	0	1	1	0	0	0
Quad Cities 2	0	0	0	0	0	0	0	0
River Bend	0	0	0	0	2	0	0	1

Table A-1.1 (cont.)

Plant Name	Calendar Year–Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	0	0	1	0	0	0	1	0
Salem 1	0	1	0	0	0	0	0	0
Salem 2	0	0	0	1	0	2	1	0
San Onofre 1	0	0	0	1	0	0	0	0
San Onofre 2	1	0	0	0	0	1	1	0
San Onofre 3	1	0	0	0	0	1	0	0
Seabrook	1	2	1	0	0	0	1	1
Sequoyah 1	0	0	0	0	0	1	0	2
Sequoyah 2	0	0	0	1	1	1	1	1
South Texas 1	0	1	0	2	1	0	0	0
South Texas 2	2	1	0	1	1	0	0	0
St. Lucie 1	0	0	2	0	0	0	1	0
St. Lucie 2	0	0	0	0	0	0	1	0
Summer	0	0	0	0	0	2	0	0
Surry 1	0	0	0	0	0	1	0	0
Surry 2	0	0	1	1	0	0	0	0
Susquehanna 1	0	0	1	0	0	0	0	1
Susquehanna 2	0	0	1	0	0	0	0	0
Three Mile Island 1	0	0	2	0	0	0	1	0
Trojan	1	0	0	0	0	1	2	0
Turkey Point 3	0	0	0	0	0	0	0	0
Turkey Point 4	0	0	0	0	1	0	0	0
Vermont Yankee	1	2	0	0	1	0	0	0
Vogtle 1	0	0	0	0	0	0	0	0
Vogtle 2	2	1	0	0	1	1	0	0
Wash. Nuclear 2	0	0	0	1	0	0	0	0
Waterford 3	0	2	1	1	0	0	0	0
Wolf Creek	0	0	0	0	1	0	0	1
Yankee-Rowe	0	1	1	0	0	PSD	PSD	PSD
Zion 1	0	0	0	1	0	0	0	0
Zion 2	1	1	0	0	0	0	0	0
Total	36	42	46	41	39	38	45	32

Note: PSD means the plant is permanently shut down.

Table A-1.2 Number of Safety System Actuations—Quarterly PI Data

Plant Name	Calendar Year–Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	0	0	0	0	0	0	0	0
Arkansas 2	0	0	0	0	0	0	1	0
Beaver Valley 1	0	0	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	1	0	0
Big Rock Point	0	1	0	0	0	0	0	1
Braidwood 1	0	0	0	0	0	0	0	1
Braidwood 2	0	0	0	0	0	0	0	0
Browns Ferry 1	0	1	0	0	0	0	0	0
Browns Ferry 2	0	0	0	0	0	0	0	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	0	0	1	0	2	0	0	0
Brunswick 2	1	1	0	3	1	0	0	0
Byron 1	0	0	0	1	0	0	0	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	1	0	0	0	0	0	0	0
Catawba 1	0	0	4	0	0	0	0	0
Catawba 2	0	0	0	0	0	0	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	2	1	2	0	0	0	0	0
Cook 1	0	1	0	0	0	0	0	0
Cook 2	0	0	0	0	0	0	0	0
Cooper Station	0	1	0	2	0	0	0	0
Crystal River 3	0	1	0	2	1	0	0	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	2	1	0	0	0	0	0	0
Diablo Canyon 2	0	0	0	1	0	0	0	0
Dresden 2	0	0	0	0	0	0	0	0
Dresden 3	0	0	0	0	0	0	0	0
Duane Arnold	0	0	1	0	0	1	1	0
Farley 1	0	0	1	0	0	0	0	2
Farley 2	0	0	0	0	0	1	0	0
Fermi 2	0	1	0	0	1	0	0	2
FitzPatrick	0	0	0	0	0	0	0	0
Fort Calhoun	0	1	0	0	0	0	1	0
Ginna	2	0	0	0	0	0	0	1
Grand Gulf	0	2	2	0	0	0	0	0
Haddam Neck	0	0	0	0	0	0	0	0
Harris	0	0	0	0	0	0	0	0

Table A-1.2 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	2	0	1	0	0	0	2	0
Hatch 2	0	1	0	0	0	1	0	1
Hope Creek	0	2	1	0	1	0	0	0
Indian Point 2	2	1	0	0	1	0	0	0
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	1
LaSalle 2	0	0	0	0	3	0	0	0
Limerick 1	0	1	0	0	0	0	0	0
Limerick 2	1	0	0	0	0	0	0	0
Maine Yankee	0	0	0	0	1	0	0	0
McGuire 1	2	0	0	0	0	0	0	0
McGuire 2	0	1	0	0	0	0	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	1	0
Millstone 3	0	0	0	0	0	0	0	0
Monticello	0	1	2	0	0	0	0	0
Nine Mile Pt. 1	0	0	0	0	0	0	0	0
Nine Mile Pt. 2	0	1	0	1	3	0	2	1
North Anna 1	0	2	1	0	0	0	0	0
North Anna 2	0	1	1	0	0	1	1	0
Oconee 1	0	1	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	2
Oconee 3	0	0	1	0	2	0	0	0
Oyster Creek	0	0	2	0	0	2	1	0
Palisades	1	0	0	1	1	3	0	0
Palo Verde 1	1	0	0	1	0	1	0	0
Palo Verde 2	0	0	0	0	1	1	0	1
Palo Verde 3	0	1	1	3	1	1	0	0
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	0	0	0	0	0	0	1	1
Perry	0	0	0	0	0	0	1	0
Pilgrim	0	0	0	2	0	1	1	1
Point Beach 1	0	1	0	0	0	1	0	0
Point Beach 2	0	0	0	0	0	0	0	1
Prairie Island 1	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	0	0	2	1	0	0
River Bend	0	0	0	0	0	0	0	0

Table A-1.2 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	0	0	0	0	0	0	2	0
Salem 1	0	0	2	0	0	1	0	0
Salem 2	0	0	0	0	1	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	0	0	0	0
San Onofre 3	0	0	0	0	1	0	0	0
Seabrook	0	1	1	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	1	0	1
Sequoyah 2	0	0	0	0	0	0	1	1
South Texas 1	5	1	0	0	0	0	0	0
South Texas 2	1	0	0	1	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	1	1	0	0	0	1
Surry 1	0	0	1	0	0	0	0	0
Surry 2	0	0	1	0	0	0	0	0
Susquehanna 1	0	0	1	0	0	0	0	0
Susquehanna 2	0	0	0	0	1	0	0	0
Three Mile Island 1	0	0	0	0	0	0	0	0
Trojan	0	0	1	0	1	0	0	0
Turkey Point 3	0	0	0	0	0	0	1	0
Turkey Point 4	1	0	0	0	1	0	1	0
Vermont Yankee	0	2	0	0	0	1	0	0
Vogtle 1	0	0	0	0	0	0	0	0
Vogtle 2	0	0	1	0	0	1	0	0
Wash. Nuclear 2	0	0	0	2	0	0	1	0
Waterford 3	0	0	1	1	0	0	0	1
Wolf Creek	0	0	0	0	0	0	0	0
Yankee-Rowe	0	2	0	0	0	PSD	PSD	PSD
Zion 1	0	1	0	2	0	0	1	3
Zion 2	1	0	0	0	0	0	0	0
Total	25	33	31	24	26	20	20	23

Note: PSD means the plant is permanently shut down.

Table A-1.3 Descriptions of Significant Events for 1992

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Brunswick 1 & 2	04/04/92	GE	II	Fraudulent or deficient bolts were installed in seismic walls and missile shields in the building housing the emergency diesel generators.
Callaway	10/17/92	W	III	Improper maintenance resulted in a complete loss of annunciators, which was not recognized by plant personnel.
Dresden 2	09/18/92	GE	III	The wrong control rod was inserted during a routine operation. In response, the instructions were rewritten to reflect the actual rather than the planned rod movement. The five individuals involved agreed not to report the rod mispositioning.
Fort Calhoun	07/03/92	CE	IV	The failure of a pressurizer code safety valve to reseat initiated a LOCA with the potential to degrade the reactor coolant pressure boundary.
Hatch 1 & 2	04/24/92	GE	II	A single failure of a thermostat in the intake structure ventilation system could result in the loss of all three ventilation fans and subsequent failure of the service water pumps and RHR pumps for both units. Room temperature could exceed 121 °C (250 °F).
Indian Point 2	04/13/92	W	I	A reactor trip was initiated by a high SG level turbine trip, which resulted from operator actions in response to a condenser low hotwell level. Both motor-driven auxiliary feed-feedwater pumps failed to automatically start.
LaSalle 2	08/27/92	GE	III	Following a reactor trip, the feedwater pump turbines could not be tripped automatically or manually. The MSIVs were closed manually by operators to secure the pumps.
Millstone 2	07/06/92	CE	I	A design deficiency in the electronic surveillance systems and a single failure design vulnerability in the dc control system could result in the loss of one dc bus (single failure), cause a LOCA (by opening the PORVs) and disable mitigating ECCS.
Nine Mile Pt. 1	02/21/92	GE	I	During postmaintenance testing, the intake canal became isolated from Lake Ontario. This loss of the ultimate heat sink existed for only a few minutes.
Nine Mile Pt. 2	03/23/92	GE	I	A loss of offsite power occurred in conjunction with a loss of all control room annunciators.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Oconee 2	10/19/92	B&W	II	A plant trip and loss of offsite power occurred with complications.
Palisades	02/05/92	CE	III	Actuating valves for the main steam isolation valves were not environmentally qualified (unqualified valves were inadvertently installed).
Perry	03/28/92	GE	III	Three of four steam line isolation valves were found to have leakage in excess of the Technical Specification.
Prairie Island 2	02/20/92	W	III	A loss of shutdown cooling occurred during reactor vessel draindown. Core exit temperature rose approximately 90 degrees in 20 minutes. No other performance indicators were involved.
River Bend	10/01/92	GE	IV	Poor radiation practices during the refueling outage resulted in the exposure of workers due to an unposted high radiation area and the transfer of low-level radioactive material off site to an unauthorized recipient. No other performance indicators were involved.
Robinson 2	08/22/92	W	II	The loss of the startup transformer caused a reactor trip and a complete loss of offsite power.
Robinson 2	08/24/92	W	II	Debris was found in the "B" SI recirculation line restricting flow. The licensee declared both SI pumps inoperable and initiated a plant shutdown.
Sequoyah 1	08/14/92	W	II	The licensee discovered that an insufficient quantity of ice was maintained in the ice condenser units during Cycles 4 and 5.
Sequoyah 2	08/14/92	W	II	The licensee discovered that an insufficient quantity of ice was maintained in the ice condenser units during Cycle 4.
Sequoyah 1 & 2	10/26/92	W	II	Water entrained in the control air system resulted in turbine runbacks at both units and a scram at Unit 1.
Turkey Point 3&4	08/24/92	W	II	Unusual event was declared as a result of a hurricane. All offsite power was lost for several days. All communications were lost. The fire protection system was lost. Other site damage included loss of physical security system, resulting in an upgrade to Alert.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Vogtle 1	11/18/92	W	II	Failure of EDG 1A during surveillance testing ultimately revealed a fabrication error involving the air distributor. This rendered the EDG technically inoperable since installation. No other performance indicators were involved.
Wash. Nuclear 2	02/25/92	GE	V	Both trains of the containment atmosphere control system were inoperable because a drain line for the scrubber, which is part of the hydrogen recombiner subsystem, was connected to an RHR discharge line causing drainage problems.
Wash. Nuclear 2	08/15/92	GE	V	Corewide power oscillations occurred while returning to 100% power following drywell leakage investigation and isolation.

Table A-1.4 Number of Significant Events—Quarterly PI Data

	Calendar Year–Quarter							
Plant Name	**91–1**	**91–2**	**91–3**	**91–4**	**92–1**	**92–2**	**92–3**	**92–4**
Arkansas 1	0	0	0	0	0	0	0	0
Arkansas 2	0	0	0	0	0	0	0	0
Beaver Valley 1	0	0	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	0	0
Big Rock Point	0	1	0	0	0	0	0	0
Braidwood 1	0	0	0	0	0	0	0	0
Braidwood 2	0	0	0	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	0	0	0	0	0	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	0	0	0	0	0	1	0	0
Brunswick 2	0	0	0	0	0	1	0	0
Byron 1	0	0	0	0	0	0	0	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	0	1
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	0	0	0	0	0	0	0	0
Catawba 1	0	0	0	0	0	0	0	0
Catawba 2	0	0	0	0	0	0	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	0	0	0	0	0	0	0	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	0	0	0	0	0	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	0	0	0	1	0	0	0	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	1	0	0	0	0	0	0	0
Diablo Canyon 2	0	0	0	0	0	0	0	0
Dresden 2	0	0	1	0	0	0	1	0
Dresden 3	0	0	1	0	0	0	0	0
Duane Arnold	0	0	0	0	0	0	0	0
Farley 1	0	0	0	0	0	0	0	0
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	0	0	0	0	0	0	0	0
FitzPatrick	1	2	0	0	0	0	0	0
Fort Calhoun	0	1	0	0	0	0	1	0
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	0	0	0	0	0	0
Haddam Neck	0	0	0	0	0	0	0	0
Harris	0	2	0	0	0	0	0	0

Table A-1.4 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	1	0	0	0	0	1	0	0
Hatch 2	0	0	0	0	0	1	0	0
Hope Creek	0	0	0	0	0	0	0	0
Indian Point 2	0	0	0	0	0	1	0	0
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	0
LaSalle 2	0	0	0	0	0	0	1	0
Limerick 1	0	0	0	0	0	0	0	0
Limerick 2	0	0	0	0	0	0	0	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	1	0	0	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	1	0
Millstone 3	0	0	1	0	0	0	0	0
Monticello	0	0	1	0	0	0	0	0
Nine Mile Pt. 1	0	0	0	0	1	0	0	0
Nine Mile Pt. 2	0	0	1	0	1	0	0	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	1	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	1
Oconee 3	1	0	0	0	0	0	0	0
Oyster Creek	1	0	0	0	0	0	0	0
Palisades	0	0	0	0	1	0	0	0
Palo Verde 1	0	0	0	0	0	0	0	0
Palo Verde 2	0	0	0	0	0	0	0	0
Palo Verde 3	0	0	0	0	0	0	0	0
Peach Bottom 2	0	0	0	0	0	0	0	0
Peach Bottom 3	0	0	1	0	0	0	0	0
Perry	1	0	0	0	1	0	0	0
Pilgrim	0	0	0	0	0	0	0	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	1	0	0	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	0	0	0	0	0	0
River Bend	0	0	0	0	0	0	0	1

Table A-1.4 (cont.)

	Calendar Year–Quarter							
Plant Name	**91–1**	**91–2**	**91–3**	**91–4**	**92–1**	**92–2**	**92–3**	**92–4**
Robinson 2	1	0	0	0	0	0	2	0
Salem 1	0	0	0	0	0	0	0	0
Salem 2	0	0	0	1	0	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	0	0	0	0
San Onofre 3	0	0	0	0	0	0	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	0	1	1
Sequoyah 2	0	0	0	0	0	0	1	1
South Texas 1	0	0	0	0	0	0	0	0
South Texas 2	0	0	0	0	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	0	0	0	0	0	0	0	0
Surry 2	0	0	0	0	0	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	0
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	0	0	0	0	0	0
Trojan	1	1	0	0	0	0	0	0
Turkey Point 3	0	0	0	0	0	0	[illegible]	0
Turkey Point 4	0	0	0	0	0	0	1	0
Vermont Yankee	0	1	0	0	0	0	0	0
Vogtle 1	0	0	0	0	0	0	0	1
Vogtle 2	1	0	0	0	0	0	0	0
Wash. Nuclear 2	0	0	0	0	1	0	1	0
Waterford 3	0	0	0	0	0	0	0	0
Wolf Creek	0	0	0	0	0	0	0	0
Yankee-Rowe	1	1	0	0	0	PSD	PSD	PSD
Zion 1	0	0	0	0	0	0	0	0
Zion 2	1	0	0	0	0	0	0	0
Total	12	9	7	2	6	5	11	6

Note: PSD means the plant is permanently shut down.

Table A-1.5 Number of Safety System Failures—Quarterly PI Data

Plant Name	Calendar Year–Quarter							
	91–1	91–2	91–3	91–4	92–1	92–2	92–3	92–4
Arkansas 1	0	1	1	2	0	1	0	0
Arkansas 2	1	3	2	3	0	0	1	0
Beaver Valley 1	0	3	0	3	1	0	0	1
Beaver Valley 2	0	1	0	0	0	1	0	0
Big Rock Point	0	0	0	0	2	0	0	0
Braidwood 1	1	0	1	0	2	0	1	0
Braidwood 2	0	0	1	0	1	0	1	0
Browns Ferry 1	0	2	0	0	0	0	0	1
Browns Ferry 2	0	3	1	0	0	1	0	2
Browns Ferry 3	0	2	0	0	0	0	0	1
Brunswick 1	2	2	2	0	0	2	2	0
Brunswick 2	1	2	1	1	0	2	3	0
Byron 1	0	0	1	1	0	0	1	0
Byron 2	0	0	1	0	0	0	1	0
Callaway	0	1	0	0	0	0	0	0
Calvert Cliffs 1	1	0	0	2	1	0	0	0
Calvert Cliffs 2	1	1	0	0	2	0	1	0
Catawba 1	3	1	3	2	2	0	0	1
Catawba 2	3	1	2	2	2	0	0	1
Clinton 1	0	0	1	0	0	0	0	0
Comanche Peak 1	1	0	0	2	0	2	0	0
Cook 1	0	0	2	0	0	0	2	0
Cook 2	0	0	1	0	0	0	2	0
Cooper Station	0	0	1	1	3	3	3	1
Crystal River 3	0	1	0	2	0	2	1	0
Davis-Besse	0	0	0	0	1	0	0	0
Diablo Canyon 1	1	0	0	2	0	1	3	1
Diablo Canyon 2	0	0	2	2	1	0	3	0
Dresden 2	0	2	4	3	1	1	1	1
Dresden 3	0	2	1	3	1	2	1	5
Duane Arnold	0	0	3	0	2	1	1	2
Farley 1	0	0	0	0	0	0	0	0
Farley 2	0	1	0	0	0	0	0	0
Fermi 2	2	1	1	1	1	2	0	0
FitzPatrick	0	4	4	7	7	4	0	2
Fort Calhoun	2	3	3	0	2	1	0	0
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	0	0	0	3	0	1
Haddam Neck	5	0	1	2	3	3	0	0
Harris	2	1	1	0	0	0	0	0

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	1	0	1	2	1	2	0	0
Hatch 2	0	0	1	0	1	1	0	1
Hope Creek	0	3	2	0	1	2	0	1
Indian Point 2	0	1	1	0	0	1	0	0
Indian Point 3	0	0	1	1	0	1	0	1
Kewaunee	2	0	0	1	0	0	0	1
LaSalle 1	0	0	1	3	1	2	0	0
LaSalle 2	1	1	0	1	0	2	2	0
Limerick 1	1	3	0	1	1	1	1	0
Limerick 2	0	1	1	1	5	3	0	0
Maine Yankee	2	0	0	1	2	1	0	1
McGuire 1	3	3	2	2	0	1	0	0
McGuire 2	2	2	2	2	0	2	0	0
Millstone 1	0	2	4	1	5	1	1	1
Millstone 2	1	0	1	1	1	1	2	0
Millstone 3	1	3	0	2	3	0	3	0
Monticello	0	3	1	0	0	0	0	2
Nine Mile Pt. 1	1	0	1	1	2	0	0	0
Nine Mile Pt. 2	0	0	1	0	1	0	0	1
North Anna 1	1	0	0	0	0	0	1	0
North Anna 2	0	0	0	0	0	1	1	0
Oconee 1	1	2	2	0	1	0	4	5
Oconee 2	1	3	2	0	3	0	4	5
Oconee 3	1	2	2	0	1	0	3	5
Oyster Creek	0	1	0	0	0	1	0	0
Palisades	0	0	1	0	8	0	0	0
Palo Verde 1	1	0	0	0	0	1	0	0
Palo Verde 2	1	0	0	0	0	1	0	0
Palo Verde 3	1	0	0	0	0	1	1	0
Peach Bottom 2	0	2	0	2	2	1	0	0
Peach Bottom 3	1	2	4	0	1	1	0	2
Perry	2	0	0	4	2	1	1	1
Pilgrim	4	0	2	2	1	1	1	1
Point Beach 1	0	1	0	1	0	1	1	2
Point Beach 2	0	0	1	1	0	0	0	2
Prairie Island 1	0	0	0	0	3	0	0	0
Prairie Island 2	0	0	0	0	3	0	0	0
Quad Cities 1	2	3	2	3	1	5	4	4
Quad Cities 2	3	1	1	1	1	5	5	2
River Bend	1	2	0	0	2	0	0	4

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	1	0	1	0	0	5	2	0
Salem 1	2	2	1	0	0	1	1	1
Salem 2	1	1	0	1	0	1	0	0
San Onofre 1	0	0	1	0	0	0	0	0
San Onofre 2	2	0	0	1	2	0	0	0
San Onofre 3	1	0	0	0	1	0	0	0
Seabrook	0	1	0	0	1	1	0	0
Sequoyah 1	0	3	0	0	1	0	3	1
Sequoyah 2	0	2	0	0	1	1	2	1
South Texas 1	2	0	0	2	0	0	0	0
South Texas 2	1	0	0	1	0	1	0	0
St. Lucie 1	0	0	0	1	0	0	0	0
St. Lucie 2	0	1	0	0	0	0	0	0
Summer	0	0	1	0	0	0	0	0
Surry 1	0	2	3	0	1	1	3	0
Surry 2	0	3	2	0	3	1	3	0
Susquehanna 1	1	0	0	2	0	0	2	0
Susquehanna 2	0	0	0	2	0	1	1	0
Three Mile Island 1	0	0	0	1	0	0	0	0
Trojan	3	1	2	1	0	1	2	0
Turkey Point 3	0	0	0	0	1	1	0	1
Turkey Point 4	0	0	0	0	1	0	0	0
Vermont Yankee	2	1	0	0	3	2	0	0
Vogtle 1	0	1	2	0	0	1	0	0
Vogtle 2	0	1	1	1	0	1	0	0
Wash. Nuclear 2	2	0	4	2	9	8	3	1
Waterford 3	0	1	0	0	0	0	0	0
Wolf Creek	0	0	2	1	0	1	0	0
Yankee-Rowe	1	0	0	1	0	PSD	PSD	PSD
Zion 1	0	1	0	1	0	1	2	0
Zion 2	1	1	0	0	0	2	4	0
Total	81	101	98	93	112	104	91	67

Note: PSD means the plant is permanently shut down.

Table A-1.6 Safety System Failures

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Arkansas 1	B&W	5	7	5	4	1	22	4
Arkansas 2	CE	4	2	2	9	1	18	4
Beaver Valley 1	W	0	0	0	6	2	8	2
Beaver Valley 2	W	0	0	4	1	1	6	1
Big Rock Point	GE	0	1	1	0	2	4	1
Braidwood 1	W	0	0	1	2	3	6	1
Braidwood 2	W	1	2	0	1	2	6	1
Browns Ferry 1	GE	8	8	2	2	1	21	4
Browns Ferry 2	GE	6	11	2	4	3	26	5
Browns Ferry 3	GE	7	8	2	2	1	20	4
Brunswick 1	GE	16	4	6	6	4	36	7
Brunswick 2	GE	12	4	6	5	5	32	6
Byron 1	W	0	0	1	2	1	4	1
Byron 2	W	0	0	0	1	1	2	0
Callaway	W	2	1	2	1	0	6	1
Calvert Cliffs 1	CE	0	6	4	3	1	14	3
Calvert Cliffs 2	CE	0	6	2	2	3	13	3
Catawba 1	W	4	3	9	9	3	28	6
Catawba 2	W	2	3	9	8	3	25	5
Clinton 1	GE	7	3	5	1	0	16	3
Comanche Peak 1	W	NYC	NYC	5	3	2	10	3
Cook 1	W	2	1	2	2	2	9	2
Cook 2	W	2	0	3	1	2	8	2
Cooper Station	GE	1	5	2	2	10	20	4
Crystal River 3	B&W	3	9	3	3	3	21	4
Davis-Besse	B&W	3	2	1	0	1	7	1
Diablo Canyon 1	W	1	1	2	3	5	12	2
Diablo Canyon 2	W	1	2	1	4	4	12	2
Dresden 2	GE	6	5	2	9	4	26	5
Dresden 3	GE	0	5	1	6	9	21	4
Duane Arnold	GE	4	7	0	3	6	20	4
Farley 1	W	1	3	0	0	0	4	1
Farley 2	W	2	2	0	1	0	5	1
Fermi 2	GE	5	6	4	5	3	23	5
FitzPatrick	GE	5	12	4	15	13	49	10
Fort Calhoun	CE	5	5	4	8	3	25	5

Footnotes at end of table.

Table A–1.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Fort St. Vrain*	GA	1	3	PSD	PSD	PSD	4	2
Ginna	W	0	1	1	0	0	2	0
Grand Gulf	GE	2	2	7	0	4	15	3
Haddam Neck	W	6	8	18	8	6	46	9
Harris	W	8	2	2	4	0	16	3
Hatch 1	GE	5	2	3	4	3	17	3
Hatch 2	GE	5	1	4	1	3	14	3
Hope Creek	GE	6	3	6	5	4	24	5
Indian Point 2	W	2	2	1	2	1	8	2
Indian Point 3	W	1	2	3	2	2	10	2
Kewaunee	W	1	1	2	3	1	8	2
LaSalle 1	GE	2	5	4	4	3	18	4
LaSalle 2	GE	3	5	1	3	4	16	3
Limerick 1	GE	4	16	5	5	3	33	7
Limerick 2	GE	NYC	6	8	3	8	25	6
Maine Yankee	CE	3	0	3	3	4	13	3
McGuire 1	W	7	12	12	10	1	42	8
McGuire 2	W	4	12	9	8	2	35	7
Millstone 1	GE	4	7	7	7	8	33	7
Millstone 2	CE	1	3	5	3	4	16	3
Millstone 3	W	4	2	4	6	6	22	4
Monticello	GE	0	6	4	4	2	16	3
Nine Mile Pt. 1	GE	2	2	5	3	2	14	3
Nine Mile Pt. 2	GE	8	0	1	1	2	12	2
North Anna 1	W	2	4	3	1	1	11	2
North Anna 2	W	2	2	2	0	2	8	2
Oconee 1	B&W	3	7	6	5	10	31	6
Oconee 2	B&W	4	6	5	6	12	33	7
Oconee 3	B&W	5	7	5	5	9	31	6
Oyster Creek	GE	6	3	2	1	1	13	3
Palisades	CE	4	2	7	1	8	22	4
Palo Verde 1	CE	6	2	1	1	1	11	2
Palo Verde 2	CE	4	2	2	1	1	10	2
Palo Verde 3	CE	3	3	2	1	2	11	2
Peach Bottom 2	GE	4	10	10	4	3	31	6
Peach Bottom 3	GE	3	7	6	7	4	27	5

Footnotes at end of table.

Table A–1.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Perry	GE	10	5	12	6	5	38	8
Pilgrim	GE	2	6	2	8	4	22	4
Point Beach 1	W	4	5	4	2	4	19	4
Point Beach 2	W	3	2	2	2	2	11	2
Prairie Island 1	W	1	1	0	0	3	5	1
Prairie Island 2	W	2	1	0	0	3	6	1
Quad Cities 1	GE	4	3	5	10	14	36	7
Quad Cities 2	GE	3	1	6	6	13	29	6
River Bend	GE	1	1	4	3	6	15	3
Robinson 2	W	6	3	0	2	7	18	4
Salem 1	W	4	5	4	5	3	21	4
Salem 2	W	4	2	9	3	1	19	4
San Onofre 1	W	3	9	3	1	0	16	3
San Onofre 2	CE	6	0	0	3	2	11	2
San Onofre 3	CE	6	0	1	1	1	9	2
Seabrook	W	1	2	1	1	2	7	1
Sequoyah 1	W	5	2	7	3	5	22	4
Sequoyah 2	W	6	2	2	2	5	17	3
South Texas 1	W	9	1	2	4	0	16	3
South Texas 2	W	0	1	1	2	1	5	1
St. Lucie 1	CE	0	0	0	1	0	1	0
St. Lucie 2	CE	0	0	1	1	0	2	0
Summer	W	1	3	2	1	0	7	1
Surry 1	W	7	2	4	5	5	23	5
Surry 2	W	7	3	3	5	7	25	5
Susquehanna 1	GE	4	1	7	3	2	17	3
Susquehanna 2	GE	4	1	6	2	2	15	3
Three Mile Island 1	B&W	0	1	1	1	0	3	1
Trojan	W	6	10	14	7	3	40	8
Turkey Point 3	W	6	2	4	0	3	15	3
Turkey Point 4	W	6	3	4	0	1	14	3
Vermont Yankee	GE	0	7	1	3	5	16	3
Vogtle 1	W	1	2	3	3	1	10	2
Vogtle 2	W	NYC	2	2	3	1	8	2
Wash. Nuclear 2	GE	5	11	9	8	21	54	11
Waterford 3	CE	0	1	2	1	0	4	1

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Wolf Creek	W	2	4	4	3	1	14	3
Yankee-Rowe	W	3	0	1	2	0	6	1
Zion 1	W	2	4	4	2	3	15	3
Zion 2	W	1	2	2	2	6	13	3
Total All Plants**		375	403	402	373	374	1927	
Total All W Plants		145	139	178	149	120	731	
Total All GE Plants		164	190	162	161	187	864	
Total All CE Plants		42	32	36	39	31	180	
Total All B&W Plants		23	39	26	24	36	148	

*Fort St. Vrain, a high-temperature gas reactor designed by General Atomic (GA) Corporation, ceased all operations on August 18, 1989.
**This total includes Fort St. Vrain.

Note: NYC means the plant is not yet critical; PSD means the plant is permanently shut down.

Table A-1.7 Forced Outage Rate (Percent) — Quarterly PI Data

	Calendar Year-Quarter							
Plant Name	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	13	1	1	0	0	0	0	0
Arkansas 2	2	1	0	7	24	35	0	0
Beaver Valley 1	17	0	11	41	0	0	0	26
Beaver Valley 2	0	0	0	2	0	1	0	0
Big Rock Point	0	23	2	0	0	11	12	14
Braidwood 1	100	0	1	11	4	0	0	0
Braidwood 2	0	10	5	4	6	3	1	1
Browns Ferry 1	100	100	100	100	100	100	100	100
Browns Ferry 2	100	99	3	2	0	2	6	0
Browns Ferry 3	100	100	100	100	100	100	100	100
Brunswick 1	20	40	17	7	9	65	0	0
Brunswick 2	10	42	0	0	14	30	0	0
Byron 1	0	0	0	6	3	0	0	0
Byron 2	0	0	0	12	0	6	0	0
Callaway	0	0	0	2	2	6	1	0
Calvert Cliffs 1	19	0	12	11	1	0	11	3
Calvert Cliffs 2	0	33	0	0	16	11	12	3
Catawba 1	5	13	5	1	0	0	27	21
Catawba 2	0	9	11	5	2	0	0	5
Clinton 1	28	0	0	9	24	0	0	3
Comanche Peak 1	7	41	1	5	2	4	4	7
Cook 1	0	7	4	0	0	0	0	1
Cook 2	3	0	25	6	0	0	93	86
Cooper Station	0	0	0	0	6	0	0	0
Crystal River 3	0	3	2	38	5	13	1	3
Davis-Besse	0	0	0	11	2	0	0	0
Diablo Canyon 1	5	6	0	0	4	2	0	1
Diablo Canyon 2	0	0	0	0	6	0	0	0
Dresden 2	47	9	12	77	43	0	8	11
Dresden 3	0	0	7	0	0	17	0	15
Duane Arnold	8	4	0	0	0	0	5	3
Farley 1	0	5	4	1	0	0	0	5
Farley 2	0	10	1	0	2	10	0	2
Fermi 2	2	11	0	0	2	4	0	5
FitzPatrick	19	74	54	37	100	0	0	0
Fort Calhoun	9	0	20	8	0	4	38	0
Ginna	0	1	1	0	4	5	0	0
Grand Gulf	0	15	12	6	10	7	3	0
Haddam Neck	14	0	0	7	10	0	0	0
Harris	0	6	0	0	6	0	12	0

Table A-1.7 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	15	0	8	0	4	3	4	1
Hatch 2	17	0	0	0	1	3	0	12
Hope Creek	21	4	0	0	0	6	0	4
Indian Point 2	5	0	7	8	6	1	6	0
Indian Point 3	12	11	0	18	6	0	32	13
Kewaunee	0	0	0	1	0	0	3	2
LaSalle 1	0	5	0	4	4	0	0	0
LaSalle 2	0	0	13	5	0	6	16	3
Limerick 1	0	18	0	14	0	100	4	0
Limerick 2	2	0	0	0	0	0	0	10
Maine Yankee	17	35	0	7	7	12	9	5
McGuire 1	4	21	0	14	37	50	2	0
McGuire 2	0	0	3	6	9	5	5	0
Millstone 1	7	0	33	100	79	6	46	1
Millstone 2	12	62	43	61	19	0	0	0
Millstone 3	25	8	73	100	40	9	2	20
Monticello	5	10	2	0	4	0	0	0
Nine Mile Pt. 1	17	0	5	16	49	79	42	0
Nine Mile Pt. 2	3	13	49	8	8	96	24	5
North Anna 1	0	15	15	0	0	0	0	0
North Anna 2	0	0	8	2	2	0	2	0
Oconee 1	0	2	14	2	3	20	0	6
Oconee 2	0	0	0	0	0	2	0	11
Oconee 3	0	2	1	43	8	1	4	14
Oyster Creek	0	0	14	5	0	9	4	0
Palisades	10	0	7	6	1	0	24	10
Palo Verde 1	0	0	6	5	25	0	2	4
Palo Verde 2	0	0	11	0	10	0	0	2
Palo Verde 3	0	30	4	18	2	9	0	0
Peach Bottom 2	0	38	3	21	5	17	19	6
Peach Bottom 3	7	29	9	0	1	0	19	28
Perry	0	18	1	15	5	26	4	8
Pilgrim	0	14	0	25	6	14	0	15
Point Beach 1	0	2	0	4	0	3	0	1
Point Beach 2	0	0	0	4	0	0	0	0
Prairie Island 1	0	0	5	0	0	0	6	30
Prairie Island 2	0	0	0	0	0	0	0	0
Quad Cities 1	0	56	0	9	13	7	5	0
Quad Cities 2	4	18	16	8	0	0	0	0
River Bend	9	1	0	35	43	0	20	13

Table A-1.7 (cont.)

Plant Name	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
	Calendar Year-Quarter							
Robinson 2	0	0	4	0	0	0	38	0
Salem 1	0	12	4	0	6	40	52	14
Salem 2	0	0	0	59	100	26	23	3
San Onofre 1	26	3	0	6	0	0	0	0
San Onofre 2	3	34	0	0	20	4	2	0
San Onofre 3	3	27	0	0	0	19	0	0
Seabrook	3	10	18	0	0	0	0	8
Sequoyah 1	7	0	0	0	14	26	0	8
Sequoyah 2	2	0	0	11	3	3	6	0
South Texas 1	100	5	3	12	4	0	0	98
South Texas 2	7	4	0	60	9	0	0	2
St. Lucie 1	0	1	3	0	0	0	14	0
St. Lucie 2	0	0	0	0	0	0	6	20
Summer	0	0	0	5	0	0	0	0
Surry 1	0	0	0	0	12	1	2	0
Surry 2	0	87	18	15	0	0	0	0
Susquehanna 1	0	0	1	0	0	42	0	3
Susquehanna 2	0	0	10	0	7	0	0	2
Three Mile Island 1	0	0	2	0	0	0	2	0
Trojan	31	0	0	0	0	8	6	57
Turkey Point 3	0	0	0	1	1	16	36	0
Turkey Point 4	0	0	0	2	8	0	41	0
Vermont Yankee	5	13	0	0	1	0	0	0
Vogtle 1	0	0	0	0	0	9	3	0
Vogtle 2	4	5	0	0	2	3	0	0
Wash. Nuclear 2	0	61	100	17	25	0	24	0
Waterford 3	0	7	2	2	3	1	2	0
Wolf Creek	0	0	0	100	56	0	0	1
Yankee-Rowe	1	10	2	0	0	PSD	PSD	PSD
Zion 1	100	50	0	41	2	0	28	2
Zion 2	19	81	2	0	0	84	0	0
Average	10	13	8	13	11	11	9	8

Note: PSD means the plant is permanently shut down.

Table A-1.8 Equipment Forced Outages/1000 Commercial Critical Hours—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	1.60	0.00	0.45	0.00	0.00	0.77	0.00	0.00
Arkansas 2	0.80	0.55	0.00	0.97	0.61	0.00	0.00	0.00
Beaver Valley 1	0.55	0.00	1.24	0.00	0.00	0.00	0.00	0.61
Beaver Valley 2	0.00	0.00	0.00	0.46	0.00	0.80	0.00	0.00
Big Rock Point	0.00	0.00	0.00	0.00	0.00	1.00	1.61	1.04
Braidwood 1	0.00	0.00	0.45	0.97	0.47	0.00	0.00	0.00
Braidwood 2	0.00	1.51	0.58	1.17	0.48	0.51	0.00	0.46
Browns Ferry 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Browns Ferry 2	0.00	2.19	0.95	0.48	0.00	0.46	0.47	0.00
Browns Ferry 3	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Brunswick 1	1.34	0.00	1.05	0.00	0.99	2.01	0.00	0.00
Brunswick 2	0.51	0.00	0.00	0.00	0.53	2.03	0.00	0.00
Byron 1	0.00	0.00	0.00	0.78	0.47	0.00	0.00	0.00
Byron 2	0.00	0.00	0.00	0.51	0.00	1.38	0.00	0.00
Callaway	0.00	0.00	0.00	0.46	0.00	0.99	0.46	0.00
Calvert Cliffs 1	0.00	0.00	0.56	1.01	0.00	0.00	1.00	0.46
Calvert Cliffs 2	0.00	2.75	0.00	0.00	0.54	1.53	2.04	0.00
Catawba 1	0.60	5.17	1.41	0.46	0.00	0.00	4.20	0.00
Catawba 2	0.00	2.48	0.51	1.55	0.93	0.00	0.00	0.00
Clinton 1	1.52	0.00	0.00	0.49	1.80	0.00	0.00	0.46
Comanche Peak 1	1.11	1.18	0.46	3.09	0.00	0.47	0.47	2.94
Cook 1	0.00	0.56	0.00	0.00	0.00	0.00	0.00	0.00
Cook 2	0.00	0.00	0.60	0.48	0.00	0.00	0.00	0.00
Cooper Station	0.00	0.00	0.00	0.00	0.48	0.00	0.00	0.00
Crystal River 3	0.00	0.00	0.00	1.43	0.48	0.00	0.54	0.46
Davis-Besse	0.00	0.00	0.00	1.61	0.00	0.00	0.00	0.00
Diablo Canyon 1	1.33	1.48	0.00	0.00	0.47	0.00	0.00	0.79
Diablo Canyon 2	0.00	0.00	0.00	0.00	0.48	0.00	0.00	0.00
Dresden 2	3.91	0.50	2.01	1.97	0.00	0.00	0.48	0.50
Dresden 3	0.00	0.00	0.63	0.00	0.00	1.30	0.00	1.55
Duane Arnold	0.53	0.51	0.00	0.00	0.00	0.00	0.48	0.47
Farley 1	0.00	1.00	0.00	0.46	0.00	0.00	0.00	1.32
Farley 2	0.00	1.01	0.00	0.00	0.64	0.84	0.00	0.00
Fermi 2	0.48	2.18	0.00	0.00	0.00	0.47	0.00	0.00
FitzPatrick	0.61	1.69	0.00	0.00	0.00	0.00	0.00	0.00
Fort Calhoun	0.00	0.00	1.13	0.96	0.00	0.71	1.41	0.00
Ginna	0.00	0.81	0.00	0.00	0.50	0.83	0.00	0.00
Grand Gulf	0.00	1.02	0.50	0.47	0.00	0.00	0.46	0.00
Haddam Neck	0.00	0.00	0.00	2.36	2.27	0.00	0.00	0.00
Harris	0.00	0.00	0.00	0.00	0.00	0.00	1.92	0.00

Table A-1.8 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	1.06	0.00	0.55	0.00	0.00	0.48	0.47	0.00
Hatch 2	0.62	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Hope Creek	2.33	0.00	0.00	0.00	0.00	0.96	0.00	0.00
Indian Point 2	0.00	0.00	1.05	0.94	1.88	0.00	0.94	0.00
Indian Point 3	1.55	0.00	0.45	0.00	0.48	0.00	0.98	0.00
Kewaunee	0.00	0.00	0.00	0.00	0.00	0.00	0.46	0.46
LaSalle 1	0.00	0.78	0.00	0.47	0.47	0.00	0.00	0.00
LaSalle 2	0.00	0.00	1.03	0.92	0.00	1.55	0.53	0.00
Limerick 1	0.46	1.06	0.00	0.52	0.00	0.00	0.00	0.00
Limerick 2	1.55	0.00	0.00	0.00	0.00	0.00	0.00	0.45
Maine Yankee	1.10	0.69	0.00	0.95	0.00	0.60	0.50	0.45
McGuire 1	0.96	1.15	0.00	1.78	0.73	0.00	0.46	0.00
McGuire 2	0.00	0.00	0.93	0.96	3.44	1.52	0.95	0.00
Millstone 1	0.00	0.00	0.00	0.00	2.06	0.00	0.00	0.00
Millstone 2	1.02	1.18	1.47	1.02	0.56	0.00	0.00	0.00
Millstone 3	0.00	0.57	1.68	0.00	0.00	0.00	0.46	1.52
Monticello	0.00	0.00	0.46	0.00	0.47	0.00	0.00	0.00
Nine Mile Pt. 1	0.96	0.00	0.53	0.53	0.90	4.03	2.16	0.00
Nine Mile Pt. 2	0.61	0.00	0.78	0.48	0.69	5.44	0.53	0.47
North Anna 1	0.00	0.51	0.53	0.00	0.00	0.00	0.00	0.00
North Anna 2	0.00	0.00	0.96	0.46	0.74	0.00	0.46	0.00
Oconee 1	0.00	0.46	2.57	0.46	0.47	1.14	0.00	0.00
Oconee 2	0.00	0.00	0.00	0.00	0.00	0.93	0.00	0.49
Oconee 3	0.00	0.93	0.46	3.13	0.49	0.46	3.34	0.50
Oyster Creek	0.00	0.00	1.04	0.00	0.00	1.10	0.93	0.00
Palisades	2.05	0.00	0.96	0.48	1.13	0.00	2.06	1.00
Palo Verde 1	0.00	0.00	0.50	0.00	1.22	0.00	0.00	0.46
Palo Verde 2	0.00	0.00	1.01	0.00	1.04	0.00	0.00	0.00
Palo Verde 3	0.00	1.56	0.47	0.00	0.46	0.00	0.00	0.00
Peach Bottom 2	0.00	0.78	0.46	1.09	0.48	1.63	1.20	1.78
Peach Bottom 3	0.49	0.00	0.60	0.00	0.93	0.00	1.07	1.24
Perry	0.00	0.55	0.45	1.04	0.00	1.89	0.46	0.49
Pilgrim	0.00	0.00	0.00	0.00	0.49	0.00	0.00	0.77
Point Beach 1	0.00	1.78	0.00	0.00	0.00	1.10	0.00	0.46
Point Beach 2	0.00	0.00	0.00	1.72	0.00	0.00	0.00	0.00
Prairie Island 1	0.00	0.00	0.47	0.00	0.00	0.00	0.48	0.00
Prairie Island 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Quad Cities 1	0.00	0.00	0.00	0.49	0.52	0.00	0.54	0.00
Quad Cities 2	0.48	0.00	0.54	0.00	0.00	0.00	0.00	0.00
River Bend	0.00	0.46	0.00	1.35	0.96	0.00	1.94	0.52

Table A–1.3 (cont.)

Plant Name	Calendar Year–Quarter							
	91–1	91–2	91–3	91–4	92–1	92–2	92–3	92–4
Robinson 2	0.00	0.00	0.47	0.00	0.00	0.00	1.45	0.00
Salem 1	0.00	0.67	0.50	0.00	0.46	0.00	0.00	1.42
Salem 2	0.00	0.00	0.00	2.11	0.00	0.00	0.56	0.45
San Onofre 1	0.00	0.75	0.00	0.50	0.00	0.00	0.00	0.00
San Onofre 2	0.48	0.69	0.00	0.00	0.00	0.00	0.00	0.00
San Onofre 3	0.48	0.57	0.00	0.00	0.00	1.24	0.00	0.00
Seabrook	0.48	0.50	3.77	0.00	0.00	0.00	0.00	0.88
Sequoyah 1	0.49	0.00	0.00	0.00	0.00	1.21	0.00	0.97
Sequoyah 2	0.46	0.00	0.00	0.50	0.58	0.90	0.46	0.45
South Texas 1	0.00	0.48	0.47	0.00	0.00	0.00	0.00	0.00
South Texas 2	0.97	0.00	0.00	2.23	0.98	0.00	0.00	0.46
St. Lucie 1	0.00	0.46	1.39	0.00	0.00	0.00	0.50	0.00
St. Lucie 2	0.00	0.00	0.00	0.00	0.00	0.00	0.47	0.00
Summer	0.00	0.00	0.00	1.79	0.00	0.00	0.00	0.00
Surry 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Surry 2	0.00	15.66	0.54	1.04	0.00	0.00	0.00	0.00
Susquehanna 1	0.00	0.00	0.48	0.00	0.00	0.00	0.00	0.46
Susquehanna 2	0.00	0.00	0.49	0.00	0.48	0.00	0.00	0.00
Three Mile Island 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Trojan	0.71	0.00	0.00	0.00	0.00	0.00	0.96	1.06
Turkey Point 3	0.00	0.00	0.00	0.46	0.46	0.54	0.00	0.00
Turkey Point 4	0.00	0.00	0.00	0.70	1.47	0.00	0.00	0.00
Vermont Yankee	0.96	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Vogtle 1	0.00	0.00	0.00	0.00	0.00	0.00	0.47	0.00
Vogtle 2	1.04	0.47	0.00	0.00	0.00	0.77	0.00	0.00
Wash. Nuclear 2	0.00	0.00	0.00	1.62	0.60	0.00	2.05	0.00
Waterford 3	0.00	2.34	0.46	0.46	0.52	0.58	0.53	0.00
Wolf Creek	0.00	0.00	0.00	0.00	0.96	0.00	0.00	0.00
Yankee–Rowe	0.00	0.51	0.46	0.00	0.00	PSD	PSD	PSD
Zion 1	0.00	0.89	0.00	1.51	0.00	0.00	0.94	0.00
Zion 2	1.13	2.17	0.48	0.00	0.00	2.81	0.00	0.00
Average	0.34	0.59	0.38	0.49	0.35	0.43	0.41	0.26

Note: PSD means the plant is permanently shut down.

Table A-1.9 Collective Radiation Exposure (Person-Rem)—Quarterly PI Data[c]

	Calendar Year-Quarter							
Plant Name	**91-1**	**91-2**	**91-3**	**91-4**	**92-1**	**92-2**	**92-3**	**92-4**
Arkansas 1	108	45	10	12	184	129	76	9
Arkansas 2	108	45	10	12	184	129	76	91
Beaver Valley 1	20	201	13	15	60	62	3	17
Beaver Valley 2	20	201	13	15	60	62	3	2
Big Rock Point	12	28	12	139	181	53	12	20
Braidwood 1	69	38	42	126	6	5	54	49
Braidwood 2	69	38	42	126	6	5	54	49
Browns Ferry 1	43	26	24	24	40	44	51	37
Browns Ferry 2	43	26	24	24	40	44	51	37
Browns Ferry 3	43	26	24	24	40	44	51	37
Brunswick 1	117	33	61	177	27	49	230	121
Brunswick 2	117	33	61	177	27	49	230	121
Byron 1	3	2	53	76	70	24	2	4
Byron 2	3	2	53	76	70	24	2	4
Callaway	9	5	5	4	73	255	4	5
Calvert Cliffs 1	25	19	7	15	10	127	31	7
Calvert Cliffs 2	25	19	7	15	10	127	31	7
Catawba 1	12	130	7	82	7	5	166	20
Catawba 2	12	130	7	82	7	5	166	20
Clinton 1	170	13	12	22	190	183	19	12
Comanche Peak 1	2	11	1	119	9	6	6	153
Cook 1	12	5	8	10	47	55	132	12
Cook 2	12	5	8	10	47	55	132	12
Cooper Station	17	12	14	362	18	21	20	13
Crystal River 3	11	6	3	84	9	393	15	5
Davis-Besse	3	4	110	99	2	5	7	9
Diablo Canyon 1	145	5	97	25	4	4	102	105
Diablo Canyon 2	145	5	97	25	4	4	102	105
Dresden 2	100	66	117	220	136	46	43	76
Dresden 3	100	66	117	220	136	46	43	76
Duane Arnold	55	76	45	21	269	170	30	21
Farley 1	149	167	6	2	92	126	15	169
Farley 2	149	167	6	2	92	126	15	169
Fermi 2	14	161	18	17	11	13	81	124
FitzPatrick	87	85	73	48	244	187	144	99
Fort Calhoun	16	13	14	9	175	53	17	10
Ginna	241	73	9	6	35	210	10	5
Grand Gulf	23	42	15	24	35	422	14	13
Haddam Neck	27	11	26	532	170	10	10	12
Harris	69	139	[illegible]	10	14	12	49	139

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	86	209	58	227	31	43	57	143
Hatch 2	86	209	58	227	31	43	57	143
Hope Creek	298	22	19	30	27	20	140	248
Indian Point 2	618	775	32	43	25	21	17	18
Indian Point 3	6	21	4	4	5	156	21	12
Kewaunee	69	138	3	3	80	32	2	1
LaSalle 1	202	100	43	59	294	36	42	211
LaSalle 2	202	100	43	59	294	36	42	211
Limerick 1	11	26	7	7	29	110	14	15
Limerick 2	11	26	7	7	29	110	14	15
Maine Yankee	41	17	18	20	400	40	11	14
McGuire 1	6	13	30	132	140	45	4	4
McGuire 2	6	13	30	132	140	45	4	4
Millstone 1	28	290	42	18	43	17	25	13
Millstone 2	29	118	16	13	39	149	491	494
Millstone 3	134	3	8	8	4	9	2	1
Monticello	39	385	20	22	43	26	17	28
Nine Mile Pt. 1	61	22	31	19	63	168	52	29
Nine Mile Pt. 2	61	22	31	19	63	168	52	18
North Anna 1	259	28	16	12	218	58	8	5
North Anna 2	259	28	16	12	218	58	8	5
Oconee 1	81	10	66	28	75	11	101	35
Oconee 2	81	10	66	28	75	11	101	35
Oconee 3	81	10	66	28	75	11	101	35
Oyster Creek	398	661	45	80	63	66	62	465
Palisades	163	5	8	5	264	14	6	8
Palo Verde 1	37	62	12	91	59	51	17	54
Palo Verde 2	37	62	12	91	59	51	17	54
Palo Verde 3	37	62	12	91	59	51	17	54
Peach Bottom 2	210	34	55	219	29	37	63	137
Peach Bottom 3	210	34	55	219	29	37	63	137
Perry	33	60	18	26	88	549	11	20
Pilgrim	43	336	166	59	53	40	36	144
Point Beach 1	9	59	6	58	5	58	4	61
Point Beach 2	9	59	6	58	5	58	4	61
Prairie Island 1	1	44	3	2	32	1	3	69
Prairie Island 2	1	44	3	2	32	1	3	69
Quad Cities 1	143	33	33	45	249	37	59	233
Quad Cities 2	143	33	33	45	249	37	59	233
River Bend	43	26	27	31	63	393	221	33

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	154	15	13	12	21	298	22	10
Salem 1	152	14	9	50	65	136	13	3
Salem 2	152	14	9	50	65	136	13	2
San Onofre 1	44	35	43	21	79	12	5	12
San Onofre 2	44	35	43	21	79	12	5	12
San Onofre 3	44	35	43	21	79	12	5	12
Seabrook	2	1	74	4	2	1	91	135
Sequoyah 1	9	5	7	336	85	136	5	6
Sequoyah 2	9	5	7	336	85	136	5	6
South Texas 1	43	46	10	72	2	5	10	111
South Texas 2	43	46	10	72	2	5	10	3
St. Lucie 1	7	7	6	214	9	96	10	7
St. Lucie 2	7	7	6	214	9	96	10	7
Summer	3	2	63	227	7	13	4	27
Surry 1	23	190	17	26	130	122	17	12
Surry 2	23	190	17	26	130	122	17	12
Susquehanna 1	113	89	26	24	125	94	69	80
Susquehanna 2	113	89	26	24	125	94	69	80
Three Mile Island 1	7	7	19	164	6	8	11	8
Trojan	22	322	95	145	17	11	2	54
Turkey Point 3	149	142	135	43	12	17	19	116
Turkey Point 4	149	142	135	43	12	17	19	116
Vermont Yankee	24	23	29	35	227	74	26	43
Vogtle 1	6	3	59	112	65	71	4	7
Vogtle 2	6	3	59	112	65	71	4	7
Wash. Nuclear 2	34	254	56	41	48	455	66	50
Waterford 3	106	231	5	2	18	3	56	133
Wolf Creek	3	4	39	262	57	4	3	5
Yankee-Rowe	4	7	6	23	73	PSD	PSD	PSD
Zion 1	41	20	6	20	109	277	30	104
Zion 2	41	20	6	20	109	277	30	104
Total	8254	8391	3581	8139	8537	9104	4977	6849

*The data in this table were obtained from the Institute of Nuclear Power Operations.

Note: PSD means the plant is permanently shut down.

Table A-1.10 Cause Codes—Administrative Control Problems—Quarterly PI Data

	Calendar Year-Quarter							
Plant Name	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	0	3	1	3	0	2	0	1
Arkansas 2	2	6	2	3	1	1	1	2
Beaver Valley 1	2	4	3	2	2	1	1	1
Beaver Valley 2	0	1	0	0	1	2	1	0
Big Rock Point	0	0	0	0	2	4	2	2
Braidwood 1	1	0	2	1	2	0	1	2
Braidwood 2	0	0	2	1	1	1	1	2
Browns Ferry 1	3	3	1	0	0	0	0	1
Browns Ferry 2	1	6	1	0	0	0	1	2
Browns Ferry 3	1	2	1	1	0	0	0	2
Brunswick 1	6	1	2	0	2	4	3	1
Brunswick 2	3	0	3	2	1	2	3	1
Byron 1	1	0	1	2	1	0	3	0
Byron 2	1	0	1	1	1	0	4	0
Callaway	0	1	1	3	3	0	1	0
Calvert Cliffs 1	0	0	1	0	1	1	1	1
Calvert Cliffs 2	0	1	0	1	2	0	2	0
Catawba 1	3	2	3	2	1	1	2	0
Catawba 2	2	1	3	5	1	1	0	0
Clinton 1	2	0	0	3	3	1	1	1
Comanche Peak 1	5	2	0	6	3	5	1	3
Cook 1	1	2	1	1	0	1	1	0
Cook 2	2	0	2	2	1	1	2	1
Cooper Station	3	0	2	4	3	3	1	2
Crystal River 3	0	2	1	5	1	3	3	2
Davis-Besse	1	0	1	2	1	0	2	0
Diablo Canyon 1	2	0	5	2	1	3	3	3
Diablo Canyon 2	0	0	4	2	2	1	2	2
Dresden 2	1	4	11	2	6	7	2	7
Dresden 3	2	4	3	2	7	5	1	5
Duane Arnold	0	1	2	1	2	1	1	2
Farley 1	1	2	0	2	2	0	2	0
Farley 2	0	1	0	2	4	5	0	0
Fermi 2	1	2	1	3	1	2	1	0
FitzPatrick	2	2	4	4	8	8	2	7
Fort Calhoun	4	2	2	6	5	6	3	2
Ginna	1	0	0	0	0	0	0	0
Grand Gulf	0	2	3	1	0	5	1	1
Haddam Neck	7	1	0	3	6	1	1	1
Harris	4	5	2	1	2	2	3	0

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	1	2	0	4	4	3	4	0
Hatch 2	2	5	0	2	3	3	4	5
Hope Creek	3	3	1	0	1	1	4	0
Indian Point 2	0	0	0	1	1	1	0	0
Indian Point 3	0	0	1	1	1	1	1	4
Kewaunee	0	2	1	2	0	1	0	2
LaSalle 1	4	2	0	3	2	0	0	6
LaSalle 2	3	1	1	0	3	1	0	3
Limerick 1	5	5	1	2	1	6	0	2
Limerick 2	6	7	2	0	1	7	0	0
Maine Yankee	1	1	0	0	1	2	0	3
McGuire 1	3	2	5	3	4	3	1	0
McGuire 2	2	2	6	3	2	1	0	0
Millstone 1	2	6	2	2	3	2	1	1
Millstone 2	4	0	0	0	4	2	1	0
Millstone 3	4	4	2	2	6	5	5	5
Monticello	3	3	2	0	3	0	3	1
Nine Mile Pt. 1	2	1	2	0	3	0	1	0
Nine Mile Pt. 2	4	4	3	1	4	3	4	0
North Anna 1	3	2	3	0	3	2	2	3
North Anna 2	2	3	4	1	2	2	2	3
Oconee 1	0	1	1	0	1	4	3	2
Oconee 2	0	1	1	0	3	1	1	1
Oconee 3	4	0	1	0	3	1	1	1
Oyster Creek	2	0	1	1	0	3	0	0
Palisades	5	3	1	3	10	3	2	0
Palo Verde 1	2	0	1	0	1	2	0	0
Palo Verde 2	2	0	1	2	1	2	0	1
Palo Verde 3	2	0	2	1	1	2	2	0
Peach Bottom 2	6	6	6	1	2	0	2	4
Peach Bottom 3	3	6	4	3	3	0	0	4
Perry	4	1	1	2	1	6	0	4
Pilgrim	2	3	3	2	3	2	4	2
Point Beach 1	0	0	1	1	1	3	0	1
Point Beach 2	0	0	0	2	0	1	1	2
Prairie Island 1	1	2	2	0	3	0	3	1
Prairie Island 2	1	1	2	0	2	0	3	1
Quad Cities 1	6	3	1	2	3	4	2	2
Quad Cities 2	3	2	1	1	7	3	1	1
River Bend	3	2	1	5	2	2	4	3

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	1	1	3	0	2	3	1	1
Salem 1	6	4	2	1	2	0	3	0
Salem 2	3	3	1	5	2	2	1	0
San Onofre 1	6	1	2	4	0	0	1	0
San Onofre 2	3	0	0	2	1	1	1	1
San Onofre 3	2	1	2	2	1	2	0	1
Seabrook	1	1	0	0	3	2	8	2
Sequoyah 1	2	7	5	1	3	0	0	5
Sequoyah 2	5	6	6	3	3	1	2	5
South Texas 1	5	3	1	1	2	2	4	5
South Texas 2	3	0	0	0	2	1	2	3
St. Lucie 1	1	1	2	3	1	1	0	0
St. Lucie 2	1	2	1	0	0	1	2	0
Summer	1	2	2	0	0	0	0	1
Surry 1	2	2	4	1	3	3	1	0
Surry 2	2	2	3	0	3	0	1	0
Susquehanna 1	1	0	1	1	2	0	3	1
Susquehanna 2	4	3	1	3	1	0	2	1
Three Mile Island 1	0	0	1	3	1	0	1	0
Trojan	6	7	6	4	6	5	7	4
Turkey Point 3	1	0	5	3	3	2	2	1
Turkey Point 4	1	0	2	2	4	1	4	1
Vermont Yankee	4	4	0	1	5	3	0	2
Vogtle 1	1	1	1	5	0	1	0	1
Vogtle 2	2	1	2	1	0	5	2	1
Wash. Nuclear 2	2	5	7	3	10	10	7	5
Waterford 3	1	3	1	0	1	1	3	2
Wolf Creek	4	5	2	3	3	2	1	0
Yankee-Rowe	1	0	0	1	2	PSD	PSD	PSD
Zion 1	5	4	2	3	2	2	5	5
Zion 2	5	3	4	1	1	3	3	2
Total	247	221	208	191	248	219	189	175

Note: PSD means the plant is permanently shut down.

Table A-1.11 Cause Codes—Licensed Operator Errors—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	0	0	0	0	0	0	0	0
Arkansas 2	1	0	0	0	0	1	0	0
Beaver Valley 1	0	0	1	1	0	0	1	0
Beaver Valley 2	0	0	0	0	1	1	0	0
Big Rock Point	0	1	0	0	0	1	0	0
Braidwood 1	0	0	0	0	0	1	0	0
Braidwood 2	0	0	0	0	1	0	0	0
Browns Ferry 1	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	0	1	0	0	0	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	0	0	0	0	0	1	0	1
Brunswick 2	0	0	1	0	0	0	0	0
Byron 1	0	0	0	2	0	0	0	0
Byron 2	0	0	0	1	0	0	0	0
Callaway	0	0	0	0	1	1	0	0
Calvert Cliffs 1	0	0	1	0	0	0	0	0
Calvert Cliffs 2	1	0	0	1	0	0	0	0
Catawba 1	2	0	0	0	2	0	1	0
Catawba 2	1	1	0	0	2	0	0	2
Clinton 1	1	0	0	0	0	1	0	0
Comanche Peak 1	1	0	1	3	2	0	0	0
Cook 1	1	0	0	0	0	0	0	0
Cook 2	1	0	0	0	0	0	1	0
Cooper Station	0	0	0	2	0	0	0	0
Crystal River 3	0	0	0	2	0	0	1	2
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	1	0	2	0	0	0	0	1
Diablo Canyon 2	0	0	1	1	0	0	0	0
Dresden 2	0	0	2	0	0	0	0	1
Dresden 3	0	0	1	0	0	0	0	1
Duane Arnold	1	0	0	1	0	0	0	0
Farley 1	0	0	1	0	1	0	1	0
Farley 2	0	0	0	0	2	2	0	1
Fermi 2	0	0	0	2	0	1	0	0
FitzPatrick	0	0	0	1	0	0	2	1
Fort Calhoun	0	1	0	1	1	2	1	0
Ginna	0	0	0	0	1	0	0	0
Grand Gulf	1	0	1	0	0	1	0	0
Haddam Neck	2	0	0	1	0	0	0	0
Harris	0	0	0	0	0	0	2	0

Table A-1.11 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	0	0	0	1	2	1	0	1
Hatch 2	1	0	0	1	1	1	2	2
Hope Creek	1	0	0	0	0	0	0	0
Indian Point 2	0	0	0	0	0	2	0	0
Indian Point 3	0	0	0	0	0	0	2	0
Kewaunee	0	1	0	1	1	0	1	0
LaSalle 1	0	0	0	0	0	0	0	3
LaSalle 2	0	0	0	0	0	0	0	1
Limerick 1	1	0	0	2	0	0	0	0
Limerick 2	0	1	0	0	0	1	0	0
Maine Yankee	0	1	0	0	0	0	0	0
McGuire 1	0	0	1	0	0	1	0	0
McGuire 2	0	1	0	0	1	0	0	0
Millstone 1	0	1	0	0	0	0	1	1
Millstone 2	0	1	0	0	0	0	1	0
Millstone 3	0	0	1	0	0	0	2	2
Monticello	1	1	0	0	1	0	0	0
Nine Mile Pt. 1	0	0	0	0	0	0	2	1
Nine Mile Pt. 2	0	1	0	1	1	0	0	1
North Anna 1	1	1	0	1	0	0	0	0
North Anna 2	1	2	0	0	0	2	1	1
Oconee 1	0	1	0	1	2	0	0	0
Oconee 2	0	1	0	0	3	0	0	0
Oconee 3	1	0	1	0	2	0	0	0
Oyster Creek	0	0	1	0	0	2	0	0
Palisades	0	0	0	0	0	1	0	0
Palo Verde 1	0	0	0	1	3	0	0	0
Palo Verde 2	0	0	2	1	1	0	0	0
Palo Verde 3	0	0	0	1	1	0	0	2
Peach Bottom 2	0	4	1	0	0	0	4	0
Peach Bottom 3	2	1	0	0	0	0	0	1
Perry	3	0	0	0	0	1	0	1
Pilgrim	0	1	0	1	0	1	0	0
Point Beach 1	0	1	1	0	1	1	0	0
Point Beach 2	0	0	1	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	1	0	0	0
Quad Cities 1	1	0	1	0	1	0	1	0
Quad Cities 2	0	0	2	0	0	0	0	0
River Bend	0	0	0	1	0	0	1	1

Table A-1.11 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	1	1	1	0	0	0	0	1
Salem 1	0	1	0	0	0	1	0	0
Salem 2	0	1	0	0	0	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	1	0	0	0
San Onofre 3	0	1	0	0	2	0	0	0
Seabrook	0	1	0	0	0	0	0	1
Sequoyah 1	0	0	0	0	1	0	0	0
Sequoyah 2	0	0	0	0	0	2	0	1
South Texas 1	3	0	0	1	0	0	1	1
South Texas 2	1	0	0	2	0	0	0	1
St. Lucie 1	1	1	0	0	1	0	0	0
St. Lucie 2	0	0	0	1	0	0	0	0
Summer	0	0	0	0	1	0	0	0
Surry 1	0	0	0	0	0	0	0	0
Surry 2	0	0	1	0	1	0	1	0
Susquehanna 1	0	0	1	0	0	0	0	0
Susquehanna 2	1	0	0	0	0	0	0	0
Three Mile Island 1	0	0	1	2	0	0	0	0
Trojan	0	0	0	0	0	0	0	1
Turkey Point 3	0	0	1	0	0	0	0	0
Turkey Point 4	0	0	0	0	0	0	0	0
Vermont Yankee	0	0	0	0	1	0	0	0
Vogtle 1	1	0	0	4	0	1	0	0
Vogtle 2	1	0	0	0	1	1	1	0
Wash. Nuclear 2	0	2	0	0	0	0	1	0
Waterford 3	0	1	1	0	0	0	1	0
Wolf Creek	0	2	0	2	2	1	0	1
Yankee-Rowe	0	0	0	0	0	PSD	PSD	PSD
Zion 1	0	0	0	0	0	1	1	2
Zion 2	0	1	0	0	0	1	0	0
Total	36	35	30	45	47	35	34	37

Note: PSD means the plant is permanently shut down.

Table A-1.12 Cause Codes—Other Personnel Errors—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	2	1	0	0	0	0	0	0
Arkansas 2	3	2	0	0	1	1	1	2
Beaver Valley 1	4	1	1	1	0	1	0	0
Beaver Valley 2	0	1	2	0	0	0	0	1
Big Rock Point	0	1	1	2	2	1	0	0
Braidwood 1	2	1	0	1	3	0	0	0
Braidwood 2	0	0	2	1	1	0	1	0
Browns Ferry 1	2	2	0	0	0	2	0	0
Browns Ferry 2	0	3	0	0	0	3	1	0
Browns Ferry 3	0	2	0	1	0	2	0	1
Brunswick 1	3	1	2	1	2	1	2	1
Brunswick 2	2	0	1	1	2	1	4	2
Byron 1	0	0	0	0	0	2	0	0
Byron 2	0	0	0	0	0	3	0	0
Callaway	0	1	0	0	1	0	0	0
Calvert Cliffs 1	0	1	0	2	0	0	0	1
Calvert Cliffs 2	0	3	0	0	1	0	2	0
Catawba 1	2	2	3	6	1	1	1	0
Catawba 2	2	2	1	2	0	0	1	0
Clinton 1	1	0	0	1	1	0	0	0
Comanche Peak 1	3	1	1	0	1	2	1	2
Cook 1	1	1	1	0	1	0	0	3
Cook 2	2	0	0	1	1	0	0	2
Cooper Station	3	0	0	5	1	0	0	0
Crystal River 3	1	0	0	3	1	2	2	0
Davis-Besse	0	0	1	1	2	1	1	0
Diablo Canyon 1	4	2	2	0	0	1	1	1
Diablo Canyon 2	0	0	2	2	0	2	0	2
Dresden 2	1	1	4	2	2	1	0	1
Dresden 3	0	1	3	0	2	2	1	1
Duane Arnold	0	0	1	0	0	0	1	1
Farley 1	0	2	0	0	0	0	1	1
Farley 2	0	2	0	0	0	1	1	0
Fermi 2	0	4	2	1	0	0	1	2
FitzPatrick	2	1	4	1	2	4	4	4
Fort Calhoun	1	3	0	2	0	4	1	2
Ginna	1	0	0	0	0	0	0	0
Grand Gulf	0	0	0	0	1	4	0	1
Haddam Neck	1	1	2	1	1	2	1	0
Harris	4	3	1	0	0	1	4	0

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	4	1	2	3	1	0	0	0
Hatch 2	2	1	2	2	3	1	2	2
Hope Creek	2	1	0	0	1	1	1	1
Indian Point 2	3	0	1	0	0	3	0	0
Indian Point 3	0	0	1	0	0	2	1	1
Kewaunee	1	0	0	1	0	1	0	1
LaSalle 1	2	1	0	2	0	0	0	2
LaSalle 2	1	2	2	1	0	0	0	1
Limerick 1	3	2	2	1	1	3	1	1
Limerick 2	2	4	1	2	2	3	1	2
Maine Yankee	1	0	0	0	2	0	0	0
McGuire 1	1	2	2	1	1	2	1	1
McGuire 2	1	4	3	0	2	2	0	0
Millstone 1	0	0	0	0	1	1	0	0
Millstone 2	2	0	0	0	1	0	0	0
Millstone 3	2	2	2	0	2	0	1	2
Monticello	1	1	1	0	1	0	0	1
Nine Mile Pt. 1	0	1	1	0	0	0	0	0
Nine Mile Pt. 2	1	3	1	0	4	2	1	1
North Anna 1	0	2	2	1	1	0	0	0
North Anna 2	0	0	2	1	0	2	0	0
Oconee 1	0	2	0	0	1	1	1	3
Oconee 2	0	1	0	1	1	0	1	3
Oconee 3	1	0	1	1	3	0	1	3
Oyster Creek	0	0	0	0	0	1	0	1
Palisades	0	2	2	0	5	2	1	0
Palo Verde 1	1	0	0	0	2	1	1	0
Palo Verde 2	0	2	0	0	0	0	0	1
Palo Verde 3	0	0	1	2	1	0	0	0
Peach Bottom 2	1	3	3	2	1	2	2	2
Peach Bottom 3	0	4	0	1	2	2	1	1
Perry	0	0	4	5	0	0	2	3
Pilgrim	0	3	2	1	0	0	2	2
Point Beach 1	0	2	1	1	1	2	0	0
Point Beach 2	0	2	0	3	1	0	1	2
Prairie Island 1	0	4	0	0	1	0	1	2
Prairie Island 2	0	3	0	0	1	0	1	1
Quad Cities 1	1	0	0	1	1	3	3	2
Quad Cities 2	0	0	0	2	3	1	2	1
River Bend	0	2	2	0	1	2	1	3

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	0	1	2	0	0	0	3	2
Salem 1	4	2	3	1	2	0	2	0
Salem 2	1	1	1	1	1	0	4	0
San Onofre 1	2	0	1	1	0	0	0	1
San Onofre 2	1	0	2	1	1	0	0	0
San Onofre 3	1	0	2	4	3	0	0	0
Seabrook	1	2	2	0	1	1	2	1
Sequoyah 1	1	3	4	2	2	1	1	2
Sequoyah 2	5	3	4	3	3	1	3	2
South Texas 1	3	1	0	1	1	0	3	1
South Texas 2	0	1	0	1	0	0	1	0
St. Lucie 1	1	0	2	1	1	1	0	0
St. Lucie 2	0	2	0	0	0	1	0	0
Summer	1	0	0	0	1	2	0	0
Surry 1	1	5	2	1	2	3	0	0
Surry 2	1	4	1	1	3	2	0	0
Susquehanna 1	0	0	0	0	1	1	2	0
Susquehanna 2	1	2	0	0	0	0	2	1
Three Mile Island 1	0	0	1	0	0	0	0	1
Trojan	2	2	2	1	2	1	2	1
Turkey Point 3	0	0	2	0	1	1	0	2
Turkey Point 4	1	0	0	0	1	0	0	1
Vermont Yankee	1	0	0	0	4	2	0	0
Vogtle 1	2	0	1	1	0	1	0	1
Vogtle 2	2	0	1	0	0	2	1	0
Wash. Nuclear 2	1	2	1	1	1	3	2	2
Waterford 3	1	4	0	1	0	0	2	1
Wolf Creek	0	0	2	1	0	1	0	0
Yankee-Rowe	0	0	1	1	1	PSD	PSD	PSD
Zion 1	2	2	3	2	0	2	2	3
Zion 2	2	1	3	0	1	1	0	2
Total	118	141	121	99	111	112	95	101

Note: PSD means the plant is permanently shut down.

Table A-1.13 Cause Codes—Maintenance Related—Quarterly PI Data

Plant Name	Calendar Year–Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	2	5	0	2	0	3	0	1
Arkansas 2	7	7	0	4	2	1	2	3
Beaver Valley 1	7	7	4	3	3	2	2	2
Beaver Valley 2	0	1	2	1	2	4	2	2
Big Rock Point	0	2	1	4	3	2	1	2
Braidwood 1	2	1	3	3	4	0	2	4
Braidwood 2	1	1	4	3	2	2	3	3
Browns Ferry 1	5	8	0	0	1	4	2	1
Browns Ferry 2	8	11	1	0	1	5	3	2
Browns Ferry 3	4	7	0	1	2	4	2	3
Brunswick 1	7	7	6	3	7	7	5	2
Brunswick 2	3	3	7	3	4	4	6	3
Byron 1	1	0	1	2	0	2	1	0
Byron 2	1	0	1	3	0	4	2	0
Callaway	0	1	1	3	1	2	2	0
Calvert Cliffs 1	0	2	2	3	1	1	1	1
Calvert Cliffs 2	0	5	0	1	3	1	5	0
Catawba 1	5	4	5	8	2	2	5	1
Catawba 2	3	3	5	6	4	1	1	1
Clinton 1	3	0	1	4	5	2	1	2
Comanche Peak 1	8	3	2	7	4	6	2	5
Cook 1	2	2	3	2	2	3	4	3
Cook 2	4	0	2	3	3	2	3	3
Cooper Station	3	1	2	7	4	2	1	2
Crystal River 3	0	5	1	6	2	3	6	3
Davis-Besse	1	0	2	5	3	0	3	0
Diablo Canyon 1	5	2	5	2	1	4	4	4
Diablo Canyon 2	0	0	4	3	3	1	1	4
Dresden 2	6	7	17	6	8	9	8	8
Dresden 3	3	5	9	3	10	7	3	7
Duane Arnold	2	0	2	2	1	3	2	2
Farley 1	0	5	0	3	1	0	1	2
Farley 2	0	3	0	2	2	5	1	0
Fermi 2	3	9	2	5	2	2	2	2
FitzPatrick	3	2	9	3	4	12	5	6
Fort Calhoun	3	5	0	5	6	7	5	1
Ginna	2	2	1	1	3	2	0	0
Grand Gulf	1	3	2	2	1	8	3	2
Haddam Neck	6	1	4	6	4	3	0	1
Harris	7	8	3	3	1	1	6	0

Table A-1.13 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	8	5	5	12	7	6	8	2
Hatch 2	9	9	3	7	8	7	9	7
Hope Creek	6	7	1	1	4	3	4	0
Indian Point 2	5	1	6	2	6	5	1	0
Indian Point 3	3	1	2	1	2	5	2	3
Kewaunee	0	2	1	4	6	3	3	4
LaSalle 1	6	6	4	4	3	3	0	5
LaSalle 2	4	5	6	1	5	5	3	2
Limerick 1	8	8	4	7	3	7	2	2
Limerick 2	7	11	5	4	4	8	1	2
Maine Yankee	2	2	0	2	3	2	0	3
McGuire 1	3	2	4	3	3	2	1	2
McGuire 2	2	4	5	2	4	3	1	1
Millstone 1	1	8	3	2	4	2	2	3
Millstone 2	5	0	1	1	4	2	3	0
Millstone 3	5	5	5	2	6	4	5	8
Monticello	3	6	2	0	3	0	3	2
Nine Mile Pt. 1	4	2	3	0	5	2	3	0
Nine Mile Pt. 2	4	6	4	1	7	6	1	4
North Anna 1	5	5	6	3	7	2	2	3
North Anna 2	2	4	7	2	7	7	1	3
Oconee 1	0	2	0	1	4	3	3	3
Oconee 2	0	1	0	1	5	0	2	2
Oconee 3	1	1	1	1	6	0	3	2
Oyster Creek	0	0	1	2	0	6	2	3
Palisades	5	4	2	4	6	5	3	1
Palo Verde 1	3	1	1	0	3	3	1	0
Palo Verde 2	2	2	2	3	0	2	0	1
Palo Verde 3	3	0	3	2	1	1	2	1
Peach Bottom 2	4	11	9	6	5	5	5	5
Peach Bottom 3	3	8	7	3	3	4	1	5
Perry	6	2	4	6	6	7	2	6
Pilgrim	5	8	4	3	2	3	4	2
Point Beach 1	0	5	3	4	2	3	0	2
Point Beach 2	0	2	0	5	1	1	1	4
Prairie Island 1	2	5	1	0	3	0	5	2
Prairie Island 2	2	3	1	0	4	0	4	1
Quad Cities 1	8	4	4	8	7	6	6	7
Quad Cities 2	3	2	6	6	10	9	3	3
River Bend	2	4	3	3	3	4	7	8

Table A-1.13 (cont.)

Plant Name	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
	Calendar Year-Quarter							
Robinson 2	0	1	4	0	3	4	5	3
Salem 1	13	7	7	3	5	4	2	3
Salem 2	4	4	2	6	6	5	4	0
San Onofre 1	5	3	4	5	0	0	0	1
San Onofre 2	4	2	3	4	4	3	1	0
San Onofre 3	3	2	3	6	5	3	1	0
Seabrook	2	2	2	1	4	4	6	5
Sequoyah 1	3	7	5	3	6	4	3	6
Sequoyah 2	6	6	5	2	5	4	6	8
South Texas 1	8	4	2	3	2	2	8	4
South Texas 2	5	3	0	1	2	4	3	4
St. Lucie 1	2	1	2	3	2	2	0	0
St. Lucie 2	1	3	1	0	0	1	3	0
Summer	1	1	2	1	1	2	0	2
Surry 1	2	7	7	2	6	3	2	0
Surry 2	3	8	8	2	7	3	4	1
Susquehanna 1	3	2	1	3	3	3	3	2
Susquehanna 2	4	5	2	5	2	1	3	3
Three Mile Island 1	0	0	3	2	1	0	1	1
Trojan	5	5	7	5	9	5	7	4
Turkey Point 3	1	0	3	3	4	2	2	4
Turkey Point 4	1	0	0	2	6	1	4	3
Vermont Yankee	5	3	0	2	8	4	0	2
Vogtle 1	4	1	1	6	0	4	1	2
Vogtle 2	5	2	3	3	1	6	2	1
Wash. Nuclear 2	4	5	8	5	9	9	6	5
Waterford 3	2	10	1	2	1	2	5	3
Wolf Creek	3	5	3	6	5	2	1	1
Yankee-Rowe	1	0	2	1	2	PSD	PSD	PSD
Zion 1	5	6	3	3	1	6	5	7
Zion 2	6	2	4	0	1	4	2	3
Total	370	409	341	342	395	376	307	280

Note: PSD means the plant is permanently shut down.

Table A-1.14 Cause Codes—Design/Fabrication/Installation—Quarterly PI Data

	Calendar Year-Quarter							
Plant Name	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	0	2	1	2	0	2	0	0
Arkansas 2	3	3	3	1	0	0	1	0
Beaver Valley 1	1	4	1	1	1	0	0	2
Beaver Valley 2	0	0	0	1	1	0	0	1
Big Rock Point	1	0	0	0	1	2	1	0
Braidwood 1	0	0	1	0	1	0	2	0
Braidwood 2	0	0	1	0	0	0	2	0
Browns Ferry 1	0	1	0	0	1	0	0	1
Browns Ferry 2	0	1	2	0	1	1	0	2
Browns Ferry 3	0	1	0	0	1	0	0	3
Brunswick 1	2	1	2	0	1	4	4	1
Brunswick 2	1	2	2	3	2	4	3	1
Byron 1	0	0	1	0	0	0	2	0
Byron 2	0	0	1	0	1	0	2	0
Callaway	0	0	0	0	0	1	0	0
Calvert Cliffs 1	1	0	0	1	1	1	0	1
Calvert Cliffs 2	1	1	0	0	2	1	0	1
Catawba 1	2	0	4	1	2	0	1	1
Catawba 2	3	0	4	1	1	0	1	1
Clinton 1	0	0	0	2	2	1	0	0
Comanche Peak 1	1	2	1	0	1	2	1	0
Cook 1	0	0	1	0	1	0	1	0
Cook 2	0	0	1	0	2	1	1	0
Cooper Station	1	0	1	3	1	3	2	1
Crystal River 3	1	0	0	4	1	6	2	2
Davis-Besse	0	0	0	1	0	1	0	0
Diablo Canyon 1	1	0	0	2	2	1	2	1
Diablo Canyon 2	0	0	0	1	1	2	2	2
Dresden 2	0	3	5	5	2	2	2	4
Dresden 3	0	2	3	2	3	4	1	3
Duane Arnold	0	1	2	0	3	3	3	1
Farley 1	0	0	0	0	0	0	0	1
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	2	0	0	0	1	2	0	0
FitzPatrick	1	3	6	8	12	7	1	1
Fort Calhoun	2	1	4	3	6	4	5	0
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	1	1	0	0	4	0	0
Haddam Neck	2	0	1	4	4	2	1	0
Harris	0	1	1	1	2	0	1	0

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	1	2	3	1	1	3	0	0
Hatch 2	1	2	4	0	0	3	3	2
Hope Creek	1	2	0	2	1	0	0	2
Indian Point 2	0	1	2	0	0	2	2	0
Indian Point 3	0	0	1	0	1	3	0	2
Kewaunee	2	0	1	2	1	3	0	0
LaSalle 1	1	0	0	1	1	1	1	1
LaSalle 2	1	0	0	0	0	1	2	0
Limerick 1	0	1	0	0	0	5	1	0
Limerick 2	0	1	0	1	0	5	1	0
Maine Yankee	0	0	1	1	2	0	1	0
McGuire 1	2	1	2	1	2	1	1	0
McGuire 2	2	2	3	1	0	1	0	0
Millstone 1	3	4	3	2	9	2	1	0
Millstone 2	1	0	0	1	2	2	2	0
Millstone 3	3	1	3	2	1	4	2	1
Monticello	0	3	1	0	1	1	0	2
Nine Mile Pt. 1	1	0	2	2	0	0	1	0
Nine Mile Pt. 2	2	1	1	0	1	0	1	0
North Anna 1	0	1	0	0	1	0	1	0
North Anna 2	0	0	0	0	1	0	2	0
Oconee 1	1	4	2	2	0	1	3	5
Oconee 2	1	5	2	2	0	1	3	5
Oconee 3	2	5	2	2	0	2	3	5
Oyster Creek	1	0	0	1	1	1	1	1
Palisades	0	1	1	1	12	0	3	0
Palo Verde 1	1	0	1	2	3	2	1	3
Palo Verde 2	1	0	1	2	1	3	1	2
Palo Verde 3	1	1	2	3	2	3	0	2
Peach Bottom 2	2	3	1	1	0	0	3	0
Peach Bottom 3	1	2	2	0	0	0	3	2
Perry	1	1	0	6	2	0	1	0
Pilgrim	2	2	1	2	2	1	1	1
Point Beach 1	0	1	1	0	0	0	1	1
Point Beach 2	0	1	1	0	0	0	2	2
Prairie Island 1	1	1	0	0	3	2	0	0
Prairie Island 2	1	0	0	0	2	1	0	0
Quad Cities 1	2	2	2	3	1	4	2	1
Quad Cities 2	2	1	2	5	1	5	1	1
River Bend	2	2	4	1	2	0	0	1

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	0	0	2	0	1	2	1	3
Salem 1	5	3	3	3	2	3	4	2
Salem 2	6	2	3	3	2	3	2	1
San Onofre 1	3	0	0	0	0	1	0	0
San Onofre 2	1	0	1	2	2	0	0	0
San Onofre 3	1	0	1	0	1	0	0	0
Seabrook	0	2	2	0	1	0	2	1
Sequoyah 1	0	2	4	3	2	1	2	0
Sequoyah 2	0	2	4	2	2	1	2	0
South Texas 1	1	1	1	2	0	0	1	1
South Texas 2	1	0	1	2	0	0	0	1
St. Lucie 1	0	0	0	0	1	0	0	0
St. Lucie 2	0	1	0	1	0	1	1	0
Summer	0	0	0	1	0	1	1	0
Surry 1	0	2	0	0	0	0	0	0
Surry 2	0	3	0	0	0	0	0	0
Susquehanna 1	0	1	2	2	3	1	3	1
Susquehanna 2	0	3	2	2	2	0	2	0
Three Mile Island 1	0	0	0	0	1	0	0	0
Trojan	2	3	3	0	3	5	3	2
Turkey Point 3	0	0	1	1	0	0	0	2
Turkey Point 4	0	0	1	1	0	0	0	0
Vermont Yankee	0	4	0	0	1	1	0	1
Vogtle 1	0	0	1	1	0	1	0	1
Vogtle 2	1	0	0	1	0	1	0	0
Wash. Nuclear 2	1	8	2	3	4	9	4	1
Waterford 3	1	3	2	0	1	0	0	2
Wolf Creek	1	0	2	1	0	0	0	0
Yankee-Rowe	0	0	0	1	0	PSD	PSD	PSD
Zion 1	1	2	0	0	0	1	3	2
Zion 2	1	1	0	0	0	1	4	2
Total	95	127	140	129	147	157	130	96

Note: PSD means the plant is permanently shut down.

Table A-1.15 Cause Codes—Equipment Failures (Electronic Piece-Part or Environmentally Related Failures)—Quarterly PI Data

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Arkansas 1	0	0	0	0	0	0	1	0
Arkansas 2	0	0	0	0	0	0	1	0
Beaver Valley 1	0	0	2	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	0	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	2	0	1	0	0	0	0	0
Braidwood 2	0	0	1	0	1	0	0	0
Browns Ferry 1	0	0	0	0	0	1	0	0
Browns Ferry 2	1	0	0	1	1	1	0	0
Browns Ferry 3	0	0	0	0	0	1	0	0
Brunswick 1	0	1	0	1	1	0	0	0
Brunswick 2	0	0	0	1	1	0	0	0
Byron 1	1	1	0	0	0	1	0	0
Byron 2	1	1	0	1	0	2	0	0
Callaway	0	0	0	1	0	1	0	0
Calvert Cliffs 1	0	0	0	0	0	0	0	0
Calvert Cliffs 2	0	0	1	0	0	0	0	0
Catawba 1	0	1	0	0	0	0	0	0
Catawba 2	0	1	0	0	0	0	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	0	1	1	0	0	1	0	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	1	0	0	0	0	0	0	0
Cooper Station	0	0	1	0	0	0	0	0
Crystal River 3	0	0	0	0	0	0	1	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	1	0	0	0	0	1	0
Diablo Canyon 2	0	0	0	0	0	0	1	0
Dresden 2	0	0	1	0	0	0	1	0
Dresden 3	0	0	1	0	1	0	0	1
Duane Arnold	0	0	2	0	0	0	1	0
Farley 1	0	0	1	0	0	0	0	1
Farley 2	0	1	1	0	0	0	0	0
Fermi 2	0	1	0	0	0	0	0	0
FitzPatrick	0	0	0	0	0	0	0	0
Fort Calhoun	0	0	0	0	0	0	0	0
Ginna	1	0	0	0	1	0	0	1
Grand Gulf	0	1	1	1	0	1	0	0
Haddam Neck	0	0	0	0	0	1	0	0
Harris	0	0	0	0	0	0	0	0

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter 91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Hatch 1	3	0	1	1	0	0	0	0
Hatch 2	4	1	0	1	0	1	1	0
Hope Creek	0	0	2	0	0	0	0	1
Indian Point 2	0	0	1	0	0	0	0	0
Indian Point 3	0	0	0	1	0	0	1	0
Kewaunee	0	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	0
LaSalle 2	0	0	0	1	0	0	0	0
Limerick 1	1	0	0	1	0	0	1	0
Limerick 2	0	0	0	1	0	0	1	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	0	0	0	0	0	0	0	0
McGuire 2	0	0	1	1	0	0	1	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	0	1	0	0	0	0	0	0
Monticello	0	0	1	0	0	0	0	0
Nine Mile Pt. 1	0	0	0	0	0	0	0	0
Nine Mile Pt. 2	1	0	0	0	0	0	1	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	0	0	2	0	0	0	0	0
Oconee 1	0	0	0	0	0	0	1	0
Oconee 2	0	0	0	0	0	0	1	0
Oconee 3	0	1	0	0	0	0	1	0
Oyster Creek	0	0	0	0	0	0	1	0
Palisades	0	0	0	0	0	0	0	0
Palo Verde 1	0	1	1	0	0	0	0	0
Palo Verde 2	0	0	0	0	0	0	0	0
Palo Verde 3	0	0	0	1	0	0	0	0
Peach Bottom 2	0	0	1	0	0	1	1	0
Peach Bottom 3	0	0	1	0	0	0	1	1
Perry	0	0	0	0	1	0	0	0
Pilgrim	0	0	0	1	0	0	0	1
Point Beach 1	0	1	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	1	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0
Quad Cities 1	1	0	0	0	0	1	0	0
Quad Cities 2	1	0	0	0	0	0	0	0
River Bend	1	0	0	0	1	0	1	0

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter							
	91-1	91-2	91-3	91-4	92-1	92-2	92-3	92-4
Robinson 2	1	0	0	0	0	0	1	0
Salem 1	1	1	1	0	0	0	0	0
Salem 2	0	2	1	0	0	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	0	0	1	0	0	0	0	0
San Onofre 3	0	0	1	0	0	0	0	0
Seabrook	0	0	0	0	0	0	0	1
Sequoyah 1	1	0	0	0	0	0	0	0
Sequoyah 2	0	0	0	0	0	0	0	0
South Texas 1	0	3	0	0	0	0	0	0
South Texas 2	0	1	1	0	1	0	0	2
St. Lucie 1	0	0	0	0	0	0	1	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	0	1	1	0	0	0	0
Surry 1	0	0	0	0	0	0	0	0
Surry 2	0	0	1	0	0	0	0	0
Susquehanna 1	0	0	2	1	1	0	0	0
Susquehanna 2	0	1	1	1	1	0	1	0
Three Mile Island 1	0	0	0	0	0	0	0	0
Trojan	1	0	1	1	0	0	1	0
Turkey Point 3	0	0	0	0	0	0	0	0
Turkey Point 4	0	0	0	1	0	0	0	0
Vermont Yankee	0	1	0	0	1	0	0	0
Vogtle 1	1	0	2	0	0	1	1	0
Vogtle 2	3	0	1	0	0	1	1	0
Wash. Nuclear 2	0	0	1	1	0	0	0	0
Waterford 3	0	1	0	0	0	1	0	0
Wolf Creek	0	1	0	0	1	0	0	0
Yankee-Rowe	0	1	0	0	0	PSD	PSD	PSD
Zion 1	0	0	0	0	0	0	0	0
Zion 2	0	1	0	0	0	0	0	0
Total	27	28	41	21	13	16	26	9

Nsote: PSD means the plant is permanently shut down.

Appendix A-2

Other Plant Operational Experience Data

Appendix A-2

Other Plant Operational Experience Data

This appendix presents selected licensee event report (LER) and plant operational experience data. This information is referenced in Section 2 of this report.

Tables A-2.1 through A-2.5 present data regarding reactor scrams. Tables A-2.6 through A-2.9 provide data regarding engineered safety feature actuations. Tables A-2.10 and A-2.11 present information regarding annual unit operating factors and reactor critical hours.

Note that in Tables A.2-2 through A.2-5, because of roundoff of some of the individual entries in the columns under the heading, "Scrams/1000 Critical Hours," the sum of the individual entries may not equal the total shown for that column.

Table A–2.1 Total Automatic and Manual Reactor Scrams While the Reactor Is Critical and the Reactor Scram Rates for 1988 Through 1992

		1988				1989				1990				1991				1992			
Plant Name	Vendor	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate	Total Scrams	Auto	Man	Total Scram Rate
Arkansas 1	B&W	1	1		0.16	5	5		0.83	0				2	2		0.25	0			
Arkansas 2	CE	2	1	1	0.33	2	2		0.30	4	2	2	0.49	1	1		0.14	0			
Beaver Valley 1	W	3	3		0.42	4	4		0.68	1	1		0.12	2	2		0.40	1	1		0.12
Beaver Valley 2	W	4	4		0.48	1	1		0.16	1	1		0.15	1	1		0.11	0			
Big Rock Point	GE	3	2	1	0.47	1	1		0.14	0				0				3	2	1	0.63
Braidwood 1	W	3	2	1	0.52	2	2		0.36	5	5		0.64	1	1		0.19	0			
Braidwood 2	W	14	10	4	2.92	3	3		0.39	1	1		0.14	2	2		0.30	4	4		0.48
Browns Ferry 1	GE	0				0				0				0				0			
Browns Ferry 2	GE	0				0				0				4	2	2	0.86	2	2		0.24
Browns Ferry 3	GE	0				0				0				0				0			
Brunswick 1	GE	2	2		0.30	0				1	1		0.17	2	2		0.33	2	2		0.79
Brunswick 2	GE	2	1	1	0.35	1		1	0.17	6	5	1	1.01	2	2		0.38	1	1		0.42
Byron 1	W	3	3		0.46	1		1	0.11	4	4		0.56	0				1	1		0.11
Byron 2	W	6	5	1	0.69	0				2	1	1	0.30	1	1		0.12	1		1	0.14
Callaway	W	6	5	1	0.73	2	1	1	0.27	4	4		0.54	1	1		0.11	3	3		0.41
Calvert Cliffs 1	CE	3	2	1	0.47	0				0				1	1		0.15	1	1		0.20
Calvert Cliffs 2	CE	2	2		0.26	0				0				1	1		0.22	5	1	4	0.63
Catawba 1	W	0				3	1	2	0.40	0				4	4		0.63	0			
Catawba 2	W	9	4	5	1.39	3	2	1	0.47	1	1		0.17	1	1		0.15	2	2		0.24
Clinton 1	GE	3	2	1	0.41	5	1	4	1.18	2	1	1	0.41	2		2	0.28	3	2	1	0.50
Comanche Peak 1	W	NYC				NYC				9	6	3	1.70	5	4	1	0.91	5	2	3	0.70
Cook 1	W	3	3		0.36	2	2		0.32	0				1	1		0.13	1	1		0.17
Cook 2	W	0				1	1		0.15	3	3		0.60	3	3		0.37	1	1		0.32
Cooper Station	GE	3	2	1	0.50	3	3		0.45	1	1		0.14	0				0			
Crystal River 3	B&W	2	2		0.27	1	1		0.23	0				4	3	1	0.56	2	2		0.30
Davis-Besse	B&W	1	1		0.47	2	2		0.23	2	2		0.40	0				1	1		0.11
Diablo Canyon 1	W	5	5		0.88	1	1		0.14	4	3	1	0.47	4	3	1	0.56	2	2		0.27
Diablo Canyon 2	W	3	2	1	0.48	4	1	3	0.49	0				0				0			
Dresden 2	GE	0				2	2		0.28	3	2	1	0.50	4	4		0.76	0			
Dresden 3	GE	1	1		0.16	3	3		0.41	1	1		0.13	1	1		0.19	2	2		0.35
Duane Arnold	GE	1	1		0.15	5	4	1	0.72	6	4	2	0.90	3	2	1	0.36	2	2		0.28
Farley 1	W	1	1		0.13	1	1		0.13	1		1	0.12	5	4	1	0.72	1	1		0.14

Footnotes at end of table.

Table A–2.1 (cont.)

Plant Name	Vendor	1988 Total Scrams	1988 Auto	1988 Man	1988 Total Scram Rate	1989 Total Scrams	1989 Auto	1989 Man	1989 Total Scram Rate	1990 Total Scrams	1990 Auto	1990 Man	1990 Total Scram Rate	1991 Total Scrams	1991 Auto	1991 Man	1991 Total Scram Rate	1992 Total Scrams	1992 Auto	1992 Man	1992 Total Scram Rate
Farley 2	W	0				6	5	1	0.83	1	1		0.15	4	3	1	0.47	7	4	3	0.98
Fermi 2	GE	5	4	1	0.94	4	2	2	0.67	2	2		0.27	2	1	1	0.30	3		3	0.42
FitzPatrick	GE	0				2	2		0.25	5	4	1	0.79	0				0			
Fort Calhoun	CE	0				1		1	0.13	1		1	0.18	0				3	3		0.52
Fort St. Vrain*	GA	4	1	3	1.05	0				PSD				PSD				PSD			
Ginna	W	2	2		0.26	1	1		0.15	6	6		0.81	0				2	2		0.26
Grand Gulf	GE	6	6		0.71	5	4	1	0.71	5	4	1	0.72	6	6		0.73	3	3		0.41
Haddam Neck	W	3	3		0.49	0				2		2	0.71	0				1	1		0.14
Harris	W	3	1	2	0.46	6	6		0.86	0				1	1		0.14	3	2	1	0.46
Hatch 1	GE	5	5		0.83	0				4	2	2	0.67	5	5		0.74	4	3	1	0.47
Hatch 2	GE	7	5	2	1.10	1	1		0.15	2	2		0.23	2	2		0.30	3	2	1	0.43
Hope Creek	GE	5	4	1	0.71	2	2		0.29	4	4		0.50	2	2		0.27	1		1	0.14
Indian Point 2	W	4	4		0.53	2	2		0.35	0				2	2		0.42	3	3		0.35
Indian Point 3	W	4	3	1	0.55	1		1	0.19	2	1	1	0.36	2	2		0.26	2	2		0.37
Kewaunee	W	3	3		0.39	1	1		0.13	0				1	1		0.14	2	1	1	0.26
LaSalle 1	GE	0				1	1		0.16	2	2		0.24	1	1		0.15	1	1		0.15
LaSalle 2	GE	2	1	1	0.30	1	1		0.15	2	2		0.32	3	2	1	0.36	3	2	1	0.49
Limerick 1	GE	1	1		0.12	0				0				1	1		0.12	0			
Limerick 2	GE	NYC				1	1		0.51	2	2		0.26	0				1		1	0.11
Maine Yankee	CE	3	3		0.43	2	2		0.24	0				4	4		0.53	1		1	0.14
McGuire 1	W	4	3	1	0.59	2	1	1	0.28	3	3		0.62	2	2		0.32	2	2		0.29
McGuire 2	W	2	1	1	0.27	3	3		0.43	1		1	0.17	3	1	2	0.35	5	5		0.80
Millstone 1	GE	1	1		0.12	3	3		0.41	2	1	1	0.25	1		1	0.32	1	1		0.17
Millstone 2	CE	1	1		0.14	0				2	1	1	0.31	3	1	2	0.58	0			
Millstone 3	W	5	4	1	0.69	2	1	1	0.30	7	2	5	0.89	1	1		0.34	3	2	1	0.46
Monticello	GE	1	1		0.11	2	2		0.30	1	1		0.12	4	4		0.57	0			
Nine Mile Point 1	GE	0				0				4	2	2	1.19	4	4		0.57	4	4		0.77
Nine Mile Point 2	GE	9	7	2	1.99	6	6		1.15	2	1	1	0.42	3	3		0.43	2	2		0.35
North Anna 1	W	4	3	1	0.50	3	3		0.60	1	1		0.11	1	1		0.15	0			
North Anna 2	W	0				0				1	1		0.14	1	1		0.12	2	2		0.27
Oconee 1	B&W	1	1		0.11	3	2	1	0.41	1	1		0.13	2	2		0.27	3	3		0.40
Oconee 2	B&W	1	1		0.14	3	3		0.41	0				0				1	1		0.14

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1988 Total Scrams	1988 Auto	1988 Man	1988 Total Scram Rate	1989 Total Scrams	1989 Auto	1989 Man	1989 Total Scram Rate	1990 Total Scrams	1990 Auto	1990 Man	1990 Total Scram Rate	1991 Total Scrams	1991 Auto	1991 Man	1991 Total Scram Rate	1992 Total Scrams	1992 Auto	1992 Man	1992 Total Scram Rate
Oconee 3	B&W	2	2		0.28	2	1	1	0.26	3	2	1	0.34	4	3	1	0.59	4	4		0.59
Oyster Creek	GE	0				5	4	1	1.00	3	1	2	0.38	1	1		0.19	4	4		0.53
Palisades	CE	0				1	1		0.17	2	1	1	0.39	3	3		0.44	5	5		0.75
Palo Verde 1	CE	5	5		0.87	1	1		0.66	1	1		0.24	2	2		0.26	2	2		0.33
Palo Verde 2	CE	1	1		0.17	3	3		0.71	1		1	0.19	2	1	1	0.30	3	3		0.35
Palo Verde 3	CE	0				1	1		0.83	2	2		0.24	3	2	1	0.47	1	1		0.14
Peach Bottom 2	GE	0				4	4		0.75	0				2	1	1	0.36	4	3	1	0.65
Peach Bottom 3	GE	0				0				3	1	2	0.38	2	2		0.37	4	3	1	0.52
Perry	GE	8	8		1.15	0				1	1		0.17	1		1	0.12	1	1		0.15
Pilgrim	GE	0				5	4	1	0.89	2	1	1	0.28	0				2	2		0.27
Point Beach 1	W	0				0				0				2	2		0.26	1	1		0.13
Point Beach 2	W	1	1		0.13	2	2		0.28	0				1	1		0.13	0			
Prairie Island 1	W	0				1	1		0.11	1	1		0.13	1	1		0.13	0			
Prairie Island 2	W	0				3	3		0.38	5	5		0.64	0				0			
Quad Cities 1	GE	1	1		0.12	3	1	2	0.45	1	1		0.14	1	1		0.20	1	1		0.16
Quad Cities 2	GE	3	2	1	0.48	2	2		0.24	2	2		0.32	1		1	0.13	0			
River Bend	GE	5	5		0.60	4	4		0.66	3	3		0.44	0				3	3		0.86
Robinson 2	W	3	3		0.52	3	3		0.70	2	2		0.35	1	1		0.14	1	1		0.17
Salem 1	W	3	2	1	0.43	3	3		0.48	3	3		0.50	1	1		0.15	0			
Salem 2	W	7	7		1.17	4	3	1	0.52	2	2		0.37	1	1		0.14	3	3		0.58
San Onofre 1	W	0				3	1	2	0.84	2	1	1	0.48	2	1	1	0.35	0			
San Onofre 2	CE	0				1		1	0.19	1	1		0.13	2	1	1	0.35	2	2		0.24
San Onofre 3	CE	1		1	0.17	2	2		0.24	1	1		0.16	1	1		0.12	1	1		0.15
Seabrook	W	NYC				1		1	5.14	4	4		0.72	5	4	1	0.75	3	2	1	0.42
Sequoyah 1	W	2	2		5.27	2	2		0.23	3	3		0.46	0				4	3	1	0.51
Sequoyah 2	W	5	5		0.96	4	4		0.63	2	2		0.29	1	1		0.12	4	4		0.56
South Texas 1	W	4	4		0.77	3	3		0.52	8	6	2	1.45	3	3		0.48	1	1		0.16
South Texas 2	W	NYC				10	9	1	2.22	4	4		0.67	5	4	1	0.78	3	1	2	0.35
St. Lucie 1	CE	3	3		0.40	2	2		0.24	1		1	0.18	3	2	1	0.42	1	1		0.12
St. Lucie 2	CE	0				2	1	1	0.30	1	1		0.15	0				4	1	3	0.66
Summer	W	4	4		0.66	5	3	2	0.69	0				0				2	2		0.23
Surry 1	W	2	2		0.53	2	1	1	0.47	2	1	1	0.30	0				2	1	1	0.28

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1988 Total Scrams	1988 Auto	1988 Man	1988 Total Scram Rate	1989 Total Scrams	1989 Auto	1989 Man	1989 Total Scram Rate	1990 Total Scrams	1990 Auto	1990 Man	1990 Total Scram Rate	1991 Total Scrams	1991 Auto	1991 Man	1991 Total Scram Rate	1992 Total Scrams	1992 Auto	1992 Man	1992 Total Scram Rate
Surry 2	W	3	2	1	0.60	2	2		1.33	3		3	0.38	2	2		0.33	0			
Susquehanna 1	GE	2	2		0.24	4	3	1	0.61	0				1	1		0.12	1	1		0.15
Susquehanna 2	GE	0				0				2	2		0.24	1	1		0.14	1		1	0.14
Three Mile Island 1	B&W	1	1		0.15	1	1		0.11	1	1		0.14	2	2		0.26	1	1		0.11
Trojan	W	4	4		0.68	1	1		0.18	1	1		0.17	1	1		0.71	4	3	1	0.83
Turkey Point 3	W	0				1	1		0.17	2	2		0.38	1		1	0.44	0			
Turkey Point 4	W	1	1		0.20	2	1	1	0.48	3	2	1	0.44	0				1	1		0.14
Vermont Yankee	GE	3	3		0.36	0				3	3		0.40	3	3		0.36	1	1		0.13
Vogtle 1	W	9	6	3	1.32	5	2	3	0.59	4	1	3	0.56	0				1		1	0.12
Vogtle 2	W	NYC				7	6	1	1.14	4	2	2	0.55	3	3		0.35	2	2		0.28
Wash. Nuclear 2	GE	2	1	1	0.32	4	4		0.58	2	1	1	0.34	2	1	1	0.45	3		3	0.52
Waterford 3	CE	4	3	1	0.60	3	1	2	0.41	3	3		0.37	5	4	1	0.71	0			
Wolf Creek	W	0				2	2		0.23	4	4		0.56	0				2	2		0.26
Yankee-Rowe	W	3	3		0.40	3	2	1	0.37	1	1		0.19	2	2		0.32	0			
Zion 1	W	4	4		0.59	1	1		0.19	2	2		0.39	1	1		0.21	0			
Zion 2	W	2	2		0.29	0				4	3	1	1.28	2	2		0.36	0			
Total**		270	225	45	0.40	242	195	47	0.36	232	176	56	0.33	196	165	31	0.27	195	154	41	0.27
Total All W Plants		151	126	25	0.48	125	99	26	0.38	126	97	29	0.37	84	74	10	0.24	88	71	17	0.25
Total All GE Plants		81	68	13	0.41	79	65	14	0.39	79	60	19	0.34	67	55	12	0.29	66	50	16	0.30
Total All CE Plants		25	21	4	0.24	21	16	5	0.26	20	13	7	0.23	31	24	7	0.30	29	21	8	0.29
Total All B&W Plants		9	9	0	0.20	17	15	2	0.34	7	6	1	0.15	14	12	2	0.27	12	12	0	0.23

*Fort St. Vrain, a high-temperature gas reactor designed by General Atomic (GA) Corporation, ceased all operations on August 18, 1989.
**This total includes Fort St. Vrain.

Note: NYC means the plant is not yet critical; PSD means the plant is permanently shut down.

Table A-2.2 Reactor Scram Initiating Systems

Westinghouse Reactors

System	Total Scrams						Scrams/1000 Critical Hours[1]				
	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Feedwater	47	31	42	12	31	163	0.15	0.09	0.12	0.03	0.09
RPS	27	10	14	5	9	65	0.09	0.03	0.04	0.01	0.03
Turbine	18	24	13	14	8	77	0.06	0.07	0.04	0.04	0.02
Electrical	14	9	11	19	10	63	0.04	0.03	0.03	0.05	0.03
Control Rod Drive	10	14	8	9	3	44	0.03	0.04	0.02	0.03	0.01
Main Generator	13	10	9	8	7	47	0.04	0.03	0.03	0.02	0.02
Main Steam	4	14	10	2	2	32	0.01	0.04	0.03	0.01	0.01
Support	6	6	9	6	9	36	0.02	0.02	0.03	0.02	0.03
RCS	5	4	8	7	3	27	0.02	0.01	0.02	0.02	0.01
Condensate	7	3	2	2	6	20	0.02	0.01	0.01	0.01	0.02
Total	151	125	126	84	88	574	0.48	0.38	0.37	0.24	0.25

[1]Critical hours: 1988 = 313,019.0; 1989 = 328,494.6; 1990 = 340,879.4; 1991 = 348,341.5; and 1992 = 358,555.9.

General Electric Reactors

System	Total Scrams						Scrams/1000 Critical Hours[2]				
	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Feedwater	16	13	17	8	9	63	0.08	0.06	0.07	0.03	0.04
RPS	10	5	4	7	9	35	0.05	0.02	0.02	0.03	0.04
Turbine	18	25	17	11	14	85	0.09	0.12	0.07	0.05	0.06
Electrical	3	7	8	9	8	35	0.02	0.03	0.03	0.04	0.04
Control Rod Drive	2	0	3	0	0	5	0.01	0.00	0.01	0.00	0.00
Main Generator	12	3	6	9	4	34	0.06	0.01	0.03	0.04	0.02
Main Steam	4	12	8	4	8	36	0.02	0.06	0.03	0.02	0.04
Support	6	5	9	7	5	32	0.03	0.02	0.04	0.03	0.02
RCS	4	3	3	4	4	18	0.02	0.01	0.01	0.02	0.02
Condensate	6	6	4	8	5	29	0.03	0.03	0.02	0.03	0.02
Total	81	79	79	67	66	372	0.41	0.39	0.34	0.29	0.30

[2]Critical hours: 1988 = 199,293.3; 1989 = 204,482.1; 1990 = 231,776.7; 1991 = 230,335.2; and 1992 = 221,772.5.

Table A-2.2 (cont.)

Combustion Engineering Reactors											
	Total Scrams						Scrams/1000 Critical Hours[3]				
System	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Feedwater	9	7	4	9	4	33	0.09	0.09	0.05	0.09	0.04
RPS	3	1	3	4	0	11	0.03	0.01	0.03	0.04	0.00
Turbine	0	2	2	5	7	16	0.00	0.02	0.02	0.05	0.07
Electrical	3	3	3	2	5	16	0.03	0.04	0.03	0.02	0.05
Control Rod Drive	0	3	2	2	1	8	0.00	0.04	0.02	0.02	0.01
Main Generator	4	1	1	2	2	10	0.04	0.01	0.01	0.02	0.02
Main Steam	1	1	2	0	2	6	0.01	0.01	0.02	0.00	0.02
Support	1	1	2	2	2	8	0.01	0.01	0.02	0.02	0.02
RCS	2	2	0	2	2	8	0.02	0.02	0.00	0.02	0.02
Condensate	2	0	1	3	4	10	0.02	0.00	0.01	0.03	0.04
Total	25	21	20	31	29	126	0.24	0.26	0.23	0.30	0.29

[3]Critical hours: 1988 = 102,723.3; 1989 = 80,825.6; 1990 = 85,829.4; 1991 = 103,924.1; and 1992 = 100,068.1.

Babcock & Wilcox Reactors											
	Total Scrams						Scrams/1000 Critical Hours[4]				
System	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Feedwater	5	3	2	3	5	18	0.11	0.06	0.04	0.06	0.09
RPS	0	3	0	0	0	3	0.00	0.06	0.00	0.00	0.00
Turbine	2	5	0	1	2	10	0.04	0.10	0.00	0.02	0.04
Electrical	0	4	0	0	2	6	0.00	0.08	0.00	0.00	0.04
Control Rod Drive	1	1	4	2	1	9	0.02	0.02	0.08	0.04	0.02
Main Generator	0	1	0	2	1	4	0.00	0.02	0.00	0.04	0.02
Main Steam	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Support	0	0	0	1	1	2	0.00	0.00	0.00	0.02	0.02
RCS	0	0	1	2	0	3	0.00	0.00	0.02	0.04	0.00
Condensate	1	0	0	3	0	4	0.02	0.00	0.00	0.06	0.00
Total	9	17	7	14	12	59	0.20	0.34	0.15	[illegible]	0.23

[4]Critical hours: 1988 = 45,489.4; 1989 = 49,977.5; 1990 = 48,234.5; 1991 = 52,746.4; and 1992 = 52,945.6.

Table A-2.3 Activities at Time of Reactor Scram

Westinghouse Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[1] 1988	1989	1990	1991	1992
Normal Operation	62	71	78	53	47	311	0.20	0.22	0.23	0.15	0.13
Testing	33	25	19	11	13	101	0.11	0.08	0.06	0.03	0.04
Maintenance	16	11	8	8	11	54	0.05	0.03	0.02	0.02	0.03
Raising Power	19	6	7	5	6	43	0.06	0.02	0.02	0.01	0.02
Reducing Power	12	6	9	4	6	37	0.04	0.02	0.03	0.01	0.02
Troubleshooting	8	4	4	3	5	24	0.03	0.01	0.01	0.01	0.01
Calibration	1	1	1	0	0	3	0.00	0.00	0.00	0.00	0.00
Special Testing	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	151	125	126	84	88	574	0.48	0.38	0.37	0.24	0.25

[1]Critical hours: 1988 = 313,019.0; 1989 = 328,494.6; 1990 = 340,879.4; 1991 = 348,341.5; and 1992 = 358,555.9.

General Electric Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[2] 1988	1989	1990	1991	1992
Normal Operation	37	34	48	40	31	190	0.19	0.17	0.21	0.17	0.14
Testing	16	21	14	9	15	75	0.08	0.10	0.06	0.04	0.07
Reducing Power	10	7	8	9	9	43	0.05	0.03	0.03	0.04	0.04
Raising Power	10	8	5	4	5	32	0.05	0.04	0.02	0.02	0.02
Maintenance	7	3	3	4	5	22	0.04	0.01	0.01	0.02	0.02
Troubleshooting	0	3	0	1	1	5	0.00	0.01	0.00	0.00	0.00
Calibration	1	2	1	0	0	4	0.01	0.01	0.00	0.00	0.00
Special Testing	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	81	79	79	67	66	372	0.41	0.39	0.34	0.29	0.30

[2]Critical hours: 1988 = 199,293.3; 1989 = 204,482.1; 1990 = 231,776.7; 1991 = 230,335.2; and 1992 = 221,772.5.

Table A-2.3 (cont.)

Combustion Engineering Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[3] 1988	1989	1990	1991	1992
Normal Operation	10	12	11	20	17	70	0.10	0.15	0.13	0.19	0.17
Maintenance	6	2	0	3	5	16	0.06	0.02	0.00	0.03	0.05
Reducing Power	2	3	4	3	2	14	0.02	0.04	0.05	0.03	0.02
Testing	3	0	2	2	3	10	0.03	0.00	0.02	0.02	0.03
Raising Power	3	3	1	0	1	8	0.03	0.04	0.01	0.00	0.01
Troubleshooting	1	1	2	3	1	8	0.01	0.01	0.02	0.03	0.01
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Calibration	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Special Testing	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	25	21	20	31	29	126	0.24	0.26	0.23	0.30	0.29

[3]Critical hours: 1988 = 102,723.3; 1989 = 80,825.6; 1990 = 85,829.4; 1991 = 103,924.1; and 1992 = 100,068.1.

Babcock & Wilcox Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[4] 1988	1989	1990	1991	1992
Normal Operation	4	6	3	6	3	22	0.09	0.12	0.06	0.11	0.06
Testing	1	4	3	2	3	13	0.02	0.08	0.06	0.04	0.06
Maintenance	0	4	0	1	3	8	0.00	0.08	0.00	0.02	0.06
Raising Power	2	0	0	2	2	6	0.04	0.00	0.00	0.04	0.04
Calibration	1	2	1	1	0	5	0.02	0.04	0.02	0.02	0.00
Reducing Power	1	0	0	2	0	3	0.02	0.00	0.00	0.04	0.00
Troubleshooting	0	1	0	0	1	2	0.00	0.02	0.00	0.00	0.02
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Special Testing	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	9	17	7	14	12	59	0.20	0.34	0.15	0.27	0.23

[4]Critical hours: 1988 = 45,489.4; 1989 = 49,977.5; 1990 = 48,234.5; 1991 = 52,746.4; and 1992 = 52,945.6.

Table A-2.4 Reactor Scram Causes

Westinghouse Reactors

System	Total Scrams						Scrams/1000 Critical Hours[1]				
	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Equipment	83	71	68	55	53	330	0.27	0.22	0.20	0.16	0.15
Personnel Error	43	31	30	16	23	143	0.14	0.09	0.09	0.05	0.06
Procedure	17	9	6	3	5	40	0.05	0.03	0.02	0.01	0.01
Unknown	3	7	12	2	0	24	0.01	0.02	0.04	0.01	0.00
Natural Phenomenon	3	6	5	5	2	21	0.01	0.02	0.01	0.01	0.01
Other	1	0	4	3	3	11	0.00	0.00	0.01	0.01	0.01
SG Level	1	1	1	0	2	5	0.00	0.00	0.00	0.00	0.01
Total	151	125	126	84	88	574	0.48	0.38	0.37	0.24	0.25

[1]Critical hours: 1988 = 313,019.0; 1989 = 328,494.6; 1990 = 340,879.4; 1991 = 348,341.5; and 1992 = 358,555.9.

General Electric Reactors

System	Total Scrams						Scrams/1000 Critical Hours[2]				
	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Equipment	48	47	48	44	47	234	0.24	0.23	0.21	0.19	0.21
Personnel Error	17	19	12	12	12	72	0.09	0.09	0.05	0.05	0.05
Procedure	11	5	5	2	0	23	0.06	0.02	0.02	0.01	0.00
Unknown	1	4	7	3	0	15	0.01	0.02	0.03	0.01	0.00
Natural Phenomenon	4	2	1	3	5	15	0.02	0.01	0.00	0.01	0.02
Other	0	2	6	3	2	13	0.00	0.01	0.03	0.01	0.01
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	81	79	79	67	66	372	0.41	0.39	0.34	0.29	0.30

[2]Critical hours: 1988 = 199,293.3; 1989 = 204,482.1; 1990 = 231,776.7; 1991 = 230,335.2; and 1992 = 221,772.5.

Table A-2.4 (cont.)

Combustion Engineering Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[3] 1988	1989	1990	1991	1992
Equipment	12	12	14	20	24	82	0.12	0.15	0.16	0.19	0.24
Personnel Error	6	2	2	4	4	18	0.06	0.02	0.02	0.04	0.04
Procedure	5	3	0	2	1	11	0.05	0.04	0.00	0.02	0.01
Unknown	2	3	2	1	0	8	0.02	0.04	0.02	0.01	0.00
Natural Phenomenon	0	1	1	2	0	4	0.00	0.01	0.01	0.02	0.00
Other	0	0	1	2	0	3	0.00	0.00	0.01	0.02	0.00
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	25	21	20	31	29	126	0.24	0.26	0.23	0.30	0.29

[3]Critical hours: 1988 = 102,723.3; 1989 = 80,825.6; 1990 = 85,829.4; 1991 = 103,924.1 and 1992 = 100,068.1.

Babcock & Wilcox Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[4] 1988	1989	1990	1991	1992
Equipment	5	5	5	7	5	27	0.11	0.10	0.10	0.13	0.09
Personnel Error	2	11	1	3	3	20	0.04	0.22	0.02	0.06	0.06
Procedure	1	0	0	2	4	7	0.02	0.00	0.00	0.04	0.08
Unknown	1	1	1	2	0	5	0.02	0.02	0.02	0.04	0.00
Natural Phenomenon	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	9	17	7	14	12	59	0.20	0.34	0.15	0.27	0.23

[4]Critical hours: 1988 = 45,489.4; 1989 = 49,977.5; 1990 = 48,234.5; 1991 = 52,746.4; and 1992 = 52,945.6.

Table A-2.5 Reactor Scram Signals

Westinghouse Reactors											
	Total Scrams						Scrams/1000 Critical Hours[1]				
System	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Turbine Trip	37	25	27	25	24	138	0.12	0.08	0.08	0.07	0.07
Low SG Level	36	33	32	13	21	135	0.12	0.10	0.09	0.04	0.06
Manual	25	27	29	12	17	110	0.08	0.08	0.09	0.03	0.05
High Negative Flux Rate	13	10	7	7	2	39	0.04	0.03	0.02	0.02	0.01
Low RCS Flow	8	4	2	8	2	24	0.03	0.01	0.01	0.02	0.01
Other	6	1	8	3	4	22	0.02	0.00	0.02	0.01	0.01
Loss of Power	3	1	1	5	2	12	0.01	0.00	0.00	0.01	0.01
High Reactor Power	2	3	2	1	3	11	0.01	0.01	0.01	0.00	0.01
Stm/Feed Flow Mismch.	1	3	3	2	2	11	0.00	0.01	0.01	0.01	0.01
IRM	4	1	2	1	2	10	0.01	0.00	0.01	0.00	0.01
Safety Injection	3	3	1	1	1	9	0.01	0.01	0.00	0.00	0.00
High RCS Temperature	3	4	1	0	0	8	0.01	0.01	0.00	0.00	0.00
RCP Breaker Open	3	0	2	1	1	7	0.01	0.00	0.01	0.00	0.00
PZR Low Pressure	1	0	3	1	0	5	0.00	0.00	0.01	0.00	0.00
Rx Trip Breaker Open	2	1	2	0	0	5	0.01	0.00	0.01	0.00	0.00
High SG Level	1	1	1	1	0	4	0.00	0.00	0.00	0.00	0.00
Over Power ΔTemp.	0	2	0	1	1	4	0.00	0.01	0.00	0.00	0.00
RCP Under Voltage	1	0	0	0	2	3	0.00	0.00	0.00	0.00	0.01
Low RCS Pressure	0	1	0	1	1	3	0.00	0.00	0.00	0.00	0.00
MSIV Closure	0	2	0	1	0	3	0.00	0.01	0.00	0.00	0.00
SRM	1	0	1	0	0	2	0.00	0.00	0.00	0.00	0.00
PZR High Pressure	0	0	1	0	1	2	0.00	0.00	0.00	0.00	0.00
P-7 Permissive	0	2	0	0	0	2	0.00	0.01	0.00	0.00	0.00
Low Steamline Pressure	0	1	1	0	0	2	0.00	0.00	0.00	0.00	0.00
Low Neg. Rx Power	1	0	0	0	0	1	0.00	0.00	0.00	0.00	0.00
High RCS Pressure	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.00
Low Condenser Vacuum	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.00
Total	151	125	126	84	88	574	0.48	0.38	0.37	0.24	0.25

[1]Critical hours: 1988 = 313,019.0; 1989 = 328,494.6; 1990 = 340,879.4; 1991 = 348,341.5, and 1992 = 358,555.9.

Table A-2.5 (cont.)

General Electric Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[2] 1988	1989	1990	1991	1992
Turbine Trip	30	29	22	29	19	129	0.15	0.14	0.09	0.13	0.09
Manual	13	14	19	11	16	73	0.07	0.07	0.08	0.05	0.07
Low Rx Water Level	16	10	13	8	11	58	0.08	0.05	0.06	0.03	0.05
High Reactor Power	7	10	6	6	8	37	0.04	0.05	0.03	0.03	0.04
MSIV Closure	3	7	7	3	4	24	0.02	0.03	0.03	0.01	0.02
IRM	4	3	2	4	2	15	0.02	0.01	0.01	0.02	0.01
High RCS Pressure	1	2	4	2	0	9	0.01	0.01	0.02	0.01	0.00
Other	2	1	4	1	1	9	0.01	0.00	0.02	0.00	0.00
Loss of Power	1	1	0	2	1	5	0.01	0.00	0.00	0.01	0.00
MS Line Radiation	1	1	0	1	2	5	0.01	0.00	0.00	0.00	0.01
Low Condenser Vacuum	1	1	1	0	1	4	0.01	0.00	0.00	0.00	0.00
High Rx Water Level	0	0	1	0	1	2	0.00	0.00	0.00	0.00	0.00
High Trip Volume	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
ATWS	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
Total	81	79	79	67	66	372	0.41	0.39	0.34	0.29	0.30

[2]Critical hours: 1988 = 199,293.3; 1989 = 204,482.1; 1990 = 231,776.7; 1991 = 230,335.2; and 1992 = 221,772.5.

Combustion Engineering Reactors

System	Total Scrams 1988	1989	1990	1991	1992	Total	Scrams/1000 Critical Hours[3] 1988	1989	1990	1991	1992
Manual	4	5	7	7	8	31	0.04	0.06	0.08	0.07	0.08
Turbine Trip	5	5	1	6	7	24	0.05	0.06	0.01	0.06	0.07
Low SG Level	7	4	1	5	2	19	0.07	0.05	0.01	0.05	0.02
Low DNBR	3	3	5	2	5	18	0.03	0.04	0.06	0.02	0.05
Other	1	1	1	4	2	9	0.01	0.01	0.01	0.04	0.02
PZR High Pressure	1	0	2	1	1	5	0.01	0.00	0.02	0.01	0.01
High RCS Pressure	1	1	0	0	1	3	0.01	0.01	0.00	0.00	0.01
Loss of Power	1	0	0	1	1	3	0.01	0.00	0.00	0.01	0.01
Safety Injection	1	0	0	2	0	3	0.01	0.00	0.00	0.02	0.00
High SG Level	0	1	0	1	0	2	0.00	0.01	0.00	0.01	0.00
MSIV Closure	0	0	2	0	0	2	0.00	0.00	0.02	0.00	0.00
IRM	0	0	0	1	0	1	0.00	0.00	0.00	0.01	0.00
High Reactor Power	0	0	1	0	0	1	0.00	0.00	0.01	0.00	0.00
Low RCS Flow	0	0	0	1	0	1	0.00	0.00	0.00	0.01	0.00
RCP Under Voltage	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
Low RCS Pressure	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.01
Low Steamline Pressure	0	1	0	0	0	1	0.00	0.01	0.00	0.00	0.00
Total	25	21	20	31	28	125	0.24	0.26	0.23	0.30	0.28

[3]Critical hours: 1988 = 102,723.3; 1989 = 80,825.6; 1990 = 85,829.4; 1991 = 103,924.1; and 1992 = 100,068.1.

Table A-2.5 (cont.)

Babcock & Wilcox Reactors											
	Total Scrams						Scrams/1000 Critical Hours[a]				
System	1988	1989	1990	1991	1992	Total	1988	1989	1990	1991	1992
Turbine Trip	2	7	0	2	4	15	0.04	0.14	0.00	0.04	0.08
High RCS Pressure	3	4	2	2	1	12	0.07	0.08	0.04	0.04	0.02
Other	2	2	0	5	3	12	0.04	0.04	0.00	0.09	0.06
Low RCS Pressure	1	0	2	2	1	6	0.02	0.00	0.04	0.04	0.02
Manual	0	1	1	1	0	3	0.00	0.02	0.02	0.02	0.00
High Reactor Power	1	0	2	0	0	3	0.02	0.00	0.04	0.00	0.00
Stm/Feed Flow Mismch.	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
RCP Under Voltage	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
Low DNBR	0	0	0	1	0	1	0.00	0.00	0.00	0.02	0.00
High SG level	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.02
High RCS Temperature	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
Power/Flow Imbalance	0	0	0	1	0	1	0.00	0.00	0.00	0.02	0.00
Loss of Power	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.02
Low Condenser Vacuum	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.02
Total	9	17	7	14	12	59	0.20	0.34	0.15	0.27	0.23

[a]Critical hours: 1988 = 45,489.4; 1989 = 49,977.5; 1990 = 48,234.5; 1991 = 52,746.4; and 1992 = 52,945.6.

Table A-2.6 Engineered Safety Features Actuations

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Arkansas 1	B&W	2	49	4	3	2	60	12.00
Arkansas 2	CE	4	7	3	4	1	19	3.80
Beaver Valley 1	W	11	11	11	7	6	46	9.20
Beaver Valley 2	W	7	15	17	2	6	47	9.40
Big Rock Point	GE	0	0	0	1	2	3	0.60
Braidwood 1	W	15	11	13	4	4	47	9.40
Braidwood 2	W	11	1	7	2	3	24	4.80
Browns Ferry 1	GE	40	18	13	3	5	79	15.80
Browns Ferry 2	GE	39	17	7	11	6	80	16.00
Browns Ferry 3	GE	27	15	7	3	2	54	10.80
Brunswick 1	GE	13	14	15	11	13	66	13.20
Brunswick 2	GE	14	17	21	17	7	76	15.20
Byron 1	W	5	3	2	5	1	16	3.20
Byron 2	W	2	2	6	2	1	13	2.60
Callaway	W	7	14	4	0	4	29	5.80
Calvert Cliffs 1	CE	5	2	4	1	0	12	2.40
Calvert Cliffs 2	CE	0	0	1	2	0	3	0.60
Catawba 1	W	4	9	2	6	3	24	4.80
Catawba 2	W	13	4	3	6	1	27	5.40
Clinton 1	GE	10	17	4	4	3	38	7.60
Comanche Peak 1	W	NYC	NYC	14	15	5	34	11.33
Cook 1	W	0	0	0	1	0	1	0.20
Cook 2	W	1	1	2	2	0	6	1.20
Cooper Station	GE	22	14	7	18	7	68	13.60
Crystal River 3	B&W	6	7	1	6	3	23	4.60
Davis-Besse	B&W	4	1	7	3	0	15	3.00
Diablo Canyon 1	W	12	8	4	7	4	35	7.00
Diablo Canyon 2	W	17	5	6	3	1	32	6.40
Dresden 2	GE	7	13	8	21	17	66	13.20
Dresden 3	GE	8	5	4	7	15	39	7.80
Duane Arnold	GE	13	19	16	9	7	64	12.80
Farley 1	W	2	1	0	1	3	7	1.40
Farley 2	W	0	1	1	0	1	3	0.60
Fermi 2	GE	24	21	6	16	6	73	14.60
FitzPatrick	GE	4	11	16	7	10	48	9.60
Fort Calhoun	CE	10	2	8	5	7	32	6.40

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Fort St. Vrain*	GA	5	6	PSD	PSD	PSD	11	5.50
Ginna	W	5	10	6	6	5	32	6.40
Grand Gulf	GE	6	11	11	5	7	40	8.00
Haddam Neck	W	0	1	2	0	0	3	0.60
Harris	W	8	6	3	4	1	22	4.40
Hatch 1	GE	9	8	10	31	17	75	15.00
Hatch 2	GE	12	5	6	7	13	43	8.60
Hope Creek	GE	15	16	14	19	11	75	15.00
Indian Point 2	W	2	3	22	31	25	83	16.60
Indian Point 3	W	1	1	0	1	1	4	0.80
Kewaunee	W	13	9	7	3	5	37	7.40
LaSalle 1	GE	15	10	5	11	6	47	9.40
LaSalle 2	GE	7	11	8	6	12	44	8.80
Limerick 1	GE	28	27	12	15	7	89	17.80
Limerick 2	GE	NYC	7	14	6	4	31	7.75
Maine Yankee	CE	3	0	1	0	1	5	1.00
McGuire 1	W	12	7	2	3	3	27	5.40
McGuire 2	W	6	2	1	4	3	16	3.20
Millstone 1	GE	6	4	4	7	3	24	4.80
Millstone 2	CE	5	1	6	1	1	14	2.80
Millstone 3	W	6	5	7	6	2	26	5.20
Monticello	GE	3	20	11	7	2	43	8.60
Nine Mile Pt. 1	GE	4	8	10	4	2	28	5.60
Nine Mile Pt. 2	GE	36	18	13	15	18	100	20.00
North Anna 1	W	1	4	0	6	0	11	2.20
North Anna 2	W	3	4	4	2	3	16	3.20
Oconee 1	B&W	0	0	1	2	0	3	0.60
Oconee 2	B&W	0	1	0	0	1	2	0.40
Oconee 3	B&W	0	2	1	2	2	7	1.40
Oyster Creek	GE	10	7	2	1	5	25	5.00
Palisades	CE	10	3	6	4	9	32	6.40
Palo Verde 1	CE	3	4	3	6	2	18	3.60
Palo Verde 2	CE	5	3	2	1	5	16	3.20
Palo Verde 3	CE	1	7	1	9	4	22	4.40
Peach Bottom 2	GE	14	10	14	23	8	69	13.80
Peach Bottom 3	GE	12	3	10	7	5	37	7.40

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Perry	GE	20	15	15	11	7	68	13.60
Pilgrim	GE	17	21	10	15	13	76	15.20
Point Beach 1	W	0	2	3	7	3	15	3.00
Point Beach 2	W	2	5	2	0	2	11	2.20
Prairie Island 1	W	3	25	7	5	3	43	8.60
Prairie Island 2	W	0	2	4	0	3	9	1.80
Quad Cities 1	GE	6	7	9	11	6	39	7.80
Quad Cities 2	GE	13	2	3	5	9	32	6.40
River Bend	GE	17	23	22	11	8	81	16.20
Robinson 2	W	2	1	0	0	1	4	0.80
Salem 1	W	0	11	35	25	17	88	17.60
Salem 2	W	2	12	41	26	18	99	19.80
San Onofre 1	W	2	3	2	1	0	8	1.60
San Onofre 2	CE	26	7	6	9	1	49	9.80
San Onofre 3	CE	7	6	2	1	4	20	4.00
Seabrook	W	3	7	9	3	4	26	5.20
Sequoyah 1	W	22	11	13	2	6	54	10.80
Sequoyah 2	W	16	5	10	2	6	39	7.80
South Texas 1	W	22	9	6	12	9	58	11.60
South Texas 2	W	1	14	8	10	4	37	7.40
St. Lucie 1	CE	2	0	3	1	0	6	1.20
St. Lucie 2	CE	0	1	2	1	1	5	1.00
Summer	W	2	5	4	3	3	17	3.40
Surry 1	W	4	13	4	1	2	24	4.80
Surry 2	W	4	12	0	2	2	20	4.00
Susquehanna 1	GE	12	18	7	6	11	54	10.80
Susquehanna 2	GE	11	12	1	11	4	39	7.80
Three Mile Island 1	B&W	0	1	2	2	1	6	1.20
Trojan	W	17	9	10	6	6	48	9.60
Turkey Point 3	W	9	5	3	2	4	23	4.60
Turkey Point 4	W	5	4	3	4	4	20	4.00
Vermont Yankee	GE	2	14	8	10	5	39	7.80
Vogtle 1	W	14	6	4	8	3	35	7.00
Vogtle 2	W	NYC	14	4	5	8	31	7.75
Wash. Nuclear 2	GE	10	16	13	12	8	59	11.80
Waterford 3	CE	7	5	10	5	4	31	6.20

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992	Total	Average Yearly Rate
Wolf Creek	W	19	6	12	5	3	45	9.00
Yankee-Rowe	W	5	2	2	1	0	10	2.00
Zion 1	W	6	4	2	5	10	27	5.40
Zion 2	W	3	0	7	1	4	15	3.00
Total All Plants**		938	914	771	717	556	3896	
Total All W Plants		327	325	341	265	216	1474	
Total All GE Plants		506	474	356	384	291	2011	
Total All CE Plants		88	48	58	50	40	284	
Total All B&W Plants		12	61	16	18	9	116	

*Fort St. Vrain, a high-temperature gas reactor designed by General Atomic (GA) Corporation, ceased all operations on August 18, 1989.

**This total includes Fort St. Vrain.

Note: NYC means the plant is not yet critical; PSD means the plant is permanently shut down.

Table A-2.7 Engineered Safety Feature Actuations of Selected Systems*

Westinghouse Reactors

System	1988	1989	1990	1991	1992	Total
ECCS	36	35	21	22	15	129
Emergency Power	52	56	51	61	29	249
HVAC	147	151	170	119	80	667
Total	235	242	242	202	124	1045

General Electric Reactors

System	1988	1989	1990	1991	1992	Total
ECCS	35	32	34	29	32	162
Emergency Power	42	40	26	29	30	167
HVAC	315	242	197	205	154	1113
RWCU	145	136	102	125	96	604
Total	537	450	359	388	312	2046

Combustion Engineering Reactors

System	1988	1989	1990	1991	1992	Total
ECCS	13	15	7	6	10	51
Emergency Power	22	9	15	13	17	76
HVAC	57	24	33	31	18	163
Total	92	48	55	50	45	290

Babcock & Wilcox Reactors

System	1988	1989	1990	1991	1992	Total
ECCS	2	4	6	4	3	19
Emergency Power	1	9	4	3	2	19
HVAC	1	44	10	5	1	61
Total	4	57	20	12	6	99

*ECCS – systems include high pressure coolant injection, low pressure core spray, low pressure coolant injection, low pressure safety injection, high pressure safety injection, high pressure core spray, accumulators, and isolation condensers.
Emergency Power – systems include all diesel starts.
RWCU – reactor water cleanup system.
HVAC – systems include standby gas treatment, containment fan cooling, containment combustible gas control, containment purge, reactor building environmental control, drywell environmental control, shield annulus return and exhaust, access corridors environmental control, auxiliary building environmental control, fuel building environmental control, radwaste building environmental control, control building environmental control, emergency onsite power supply building environmental control, turbine building environmental control, and plant exhaust.

Table A-2.8 Engineered Safety Feature Actuation Activities

Westinghouse Reactors

Activity	1988	1989	1990	1991	1992	Total
Normal Operation	174	180	213	184	144	895
Testing	101	98	87	55	56	397
Maintenance	52	44	41	26	16	179
Unknown	0	2	0	0	0	2
Other	0	1	0	0	0	1
Total	327	325	341	265	216	1474

General Electric Reactors

Activity	1988	1989	1990	1991	1992	Total
Normal Operation	251	253	214	239	186	1143
Testing	155	146	75	83	71	530
Maintenance	100	71	67	62	34	334
Other	0	4	0	0	0	4
Unknown	0	0	0	0	0	0
Total	506	474	356	384	291	2011

Combustion Engineering Reactors

Activity	1988	1989	1990	1991	1992	Total
Normal Operation	48	25	29	29	19	150
Maintenance	23	11	11	12	11	68
Testing	17	12	18	9	10	66
Unknown	0	0	0	0	0	0
Other	0	0	0	0	0	0
Total	88	48	58	50	40	284

Babcock & Wilcox Reactors

Activity	1988	1989	1990	1991	1992	Total
Normal Operation	9	33	8	10	4	64
Testing	2	22	3	5	4	36
Maintenance	1	6	5	3	1	16
Unknown	0	0	0	0	0	0
Other	0	0	0	0	0	0
Total	12	61	16	18	9	116

Table A-2.9 Engineered Safety Feature Actuation Causes

Westinghouse Reactors

Cause	1988	1989	1990	1991	1992	Total
Equipment	141	150	143	131	109	674
Personnel Error	107	100	91	71	56	425
Procedure	42	39	44	16	21	162
Unknown	19	18	36	18	9	100
Other	15	12	23	14	14	78
Natural Phenomenon	3	6	4	15	7	35
Total	327	325	341	265	216	1474

General Electric Reactors

Cause	1988	1989	1990	1991	1992	Total
Equipment	232	200	135	173	139	879
Personnel Error	149	151	98	107	89	594
Procedure	67	62	47	33	32	241
Other	17	25	41	34	14	131
Unknown	23	33	29	29	12	126
Natural Phenomenon	18	3	6	8	5	40
Total	506	474	356	384	291	2011

Combustion Engineering Reactors

Cause	1988	1989	1990	1991	1992	Total
Equipment	46	21	17	21	18	123
Personnel Error	26	14	19	17	17	93
Procedure	12	6	6	1	4	29
Unknown	3	3	11	3	0	20
Other	1	2	5	5	1	14
Natural Phenomenon	0	2	0	3	0	5
Total	88	48	58	50	40	284

Babcock & Wilcox Reactors

Cause	1988	1989	1990	1991	1992	Total
Equipment	4	38	4	8	4	58
Personnel Error	7	13	9	8	3	40
Procedure	1	2	3	1	1	8
Unknown	0	6	0	1	0	7
Other	0	1	0	0	1	2
Natural Phenomenon	0	1	0	0	0	1
Total	12	61	16	18	9	116

Table A-2.10 Annual Unit Operating Factors for 1992

Plant Name	Docket	Availability Factor[1]	Service Factor[2]	Maximum Dependable Capacity Factor[3]	Design Electrical Rating Capacity Factor[4]
Arkansas 1	313	80.7	80.7	79.3	78.0
Arkansas 2	368	72.8	72.8	73.0	68.7
Beaver Valley 1	334	93.6	93.6	88.5	85.9
Beaver Valley 2	412	83.6	83.6	78.4	76.9
Big Rock Point	155	53.5	53.5	46.1	42.9
Braidwood 1	456	81.3	81.3	72.7	72.7
Braidwood 2	457	95.0	95.0	89.0	89.0
Browns Ferry 1	259	0.0	0.0	0.0	0.0
Browns Ferry 2	260	95.7	95.7	89.7	89.7
Browns Ferry 3	296	0.0	0.0	0.0	0.0
Brunswick 1	325	28.3	28.3	27.1	25.3
Brunswick 2	324	25.1	25.1	19.0	17.4
Byron 1	454	99.3	99.3	92.6	91.3
Byron 2	455	80.0	80.0	72.0	71.0
Callaway	483	82.0	82.0	81.9	78.7
Calvert Cliffs 1	317	56.1	56.1	56.8	55.4
Calvert Cliffs 2	318	89.5	89.5	90.9	88.8
Catawba 1	413	72.2	72.2	70.9	69.9
Catawba 2	414	94.3	94.3	93.5	92.2
Clinton 1	461	66.3	66.3	60.4	60.2
Comanche Peak 1	445	79.1	79.1	68.8	68.8
Cook 1	315	64.8	64.8	55.7	55.7
Cook 2	316	19.5	19.5	14.9	15.0
Cooper Station	298	96.0	96.0	92.8	91.1
Crystal River 3	302	75.5	75.5	73.5	73.1
Davis-Besse	346	99.5	99.5	99.3	96.1
Diablo Canyon 1	275	82.3	82.?	79.0	78.1
Diablo Canyon 2	323	98.5	98.5	96.9	94.1
Dresden 2	237	84.5	84.5	61.7	59.9
Dresden 3	249	61.4	61.4	45.1	43.9
Duane Arnold	331	81.0	81.0	75.9	72.6
Farley 1	348	81.0	81.0	79.2	77.6
Farley 2	364	79.6	79.6	74.7	74.2
Fermi 2	341	79.9	79.9	79.0	76.4
FitzPatrick	333	0.0	0.0	0.0	0.0
Fort Calhoun	285	64.7	64.7	60.4	60.4

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Docket	Availability Factor[1]	Service Factor[2]	Maximum Dependable Capacity Factor[3]	Design Electrical Rating Capacity Factor[4]
Ginna	244	85.8	85.8	84.4	84.4
Grand Gulf	416	81.5	81.5	81.4	74.4
Haddam Neck	213	79.2	79.2	78.9	75.9
Harris	400	74.0	74.0	71.6	68.4
Hatch 1	321	96.2	96.2	94.6	90.3
Hatch 2	366	75.9	75.9	69.8	68.1
Hope Creek	354	78.9	78.9	77.9	75.2
Indian Point 2	247	96.7	96.7	95.5	91.0
Indian Point 3	286	59.8	59.8	56.2	56.2
Kewaunee	305	87.5	87.5	87.7	83.8
LaSalle 1	373	74.3	74.3	70.9	68.1
LaSalle 2	374	66.6	66.6	63.5	61.1
Limerick 1	352	69.6	69.6	67.2	67.2
Limerick 2	353	97.4	97.4	91.6	91.6
Maine Yankee	309	75.5	75.5	70.9	70.1
McGuire 1	369	77.9	77.9	75.5	72.2
McGuire 2	370	69.9	69.9	68.4	65.4
Millstone 1	245	67.3	67.3	62.9	62.3
Millstone 2	336	36.3	36.3	35.4	35.4
Millstone 3	423	71.9	71.9	65.8	64.9
Monticello	263	97.1	97.1	94.6	93.0
Nine Mile Pt. 1	220	57.6	57.6	54.2	53.4
Nine Mile Pt. 2	410	58.9	58.9	54.5	54.3
North Anna 1	338	82.3	82.3	70.6	67.3
North Anna 2	339	82.4	82.4	79.2	79.4
Oconee 1	269	85.3	85.3	84.5	80.7
Oconee 2	270	80.9	80.9	80.0	76.4
Oconee 3	287	75.5	75.5	73.3	70.0
Oyster Creek	219	85.0	85.0	84.5	79.3
Palisades	255	71.7	71.7	75.9	68.8
Palo Verde 1	528	68.4	68.4	66.4	63.8
Palo Verde 2	529	95.0	95.0	94.2	90.6
Palo Verde 3	530	78.8	78.8	78.2	75.2
Peach Bottom 2	277	66.2	66.2	61.2	60.6
Peach Bottom 3	278	84.2	84.2	79.0	76.8
Perry	440	72.7	72.7	70.0	68.5

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Docket	Availability Factor[1]	Service Factor[2]	Maximum Dependable Capacity Factor[3]	Design Electrical Rating Capacity Factor[4]
Pilgrim	293	84.3	84.3	80.6	82.4
Point Beach 1	266	84.4	84.4	84.6	82.5
Point Beach 2	301	85.3	85.3	86.1	84.0
Prairie Island 1	282	77.9	77.9	79.1	75.1
Prairie Island 2	306	74.2	74.2	73.3	69.1
Quad Cities 1	254	70.1	70.1	61.7	60.1
Quad Cities 2	265	64.0	64.0	57.7	56.2
River Bend	458	36.6	36.6	33.6	33.6
Robinson 2	261	66.2	66.2	67.7	66.0
Salem 1	272	58.0	58.0	54.5	54.1
Salem 2	311	53.0	53.0	48.6	48.2
San Onofre 1	206	91.3	91.3	76.3	76.3
San Onofre 2	361	93.5	93.5	93.6	93.6
San Onofre 3	362	74.4	74.4	72.0	72.0
Seabrook	443	80.3	80.3	77.9	78.0
Sequoyah 1	327	88.1	88.1	84.8	82.9
Sequoyah 2	328	80.1	80.1	73.8	72.1
South Texas 1	498	68.7	68.7	66.1	66.1
South Texas 2	499	97.3	97.3	94.1	94.1
St. Lucie 1	335	96.5	96.5	96.9	98.0
St. Lucie 2	389	75.1	75.1	73.7	74.5
Summer	395	97.1	97.1	96.7	95.1
Surry 1	280	80.1	80.1	76.1	75.5
Surry 2	281	96.4	96.4	93.7	92.8
Susquehanna 1	387	74.8	74.8	70.0	69.3
Susquehanna 2	388	81.1	81.1	78.3	77.8
Three Mile Island 1	289	99.5	99.5	100.5	96.4
Trojan	344	66.9	53.0	47.5	46.1
Turkey Point 3	250	67.2	67.2	58.4	56.2
Turkey Point 4	251	87.5	81.3	79.3	76.2
Vermont Yankee	271	87.4	87.4	84.4	82.7
Vogtle 1	424	97.0	97.0	96.7	97.0
Vogtle 2	425	81.7	81.7	79.7	80.3
Wash. Nuclear 2	397	62.7	62.7	59.7	58.9
Waterford 3	382	82.1	82.1	80.7	78.6
Wolf Creek	482	85.8	85.8	85.5	82.6

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Docket	Availability Factor[1]	Service Factor[2]	Maximum Dependable Capacity Factor[3]	Design Electrical Rating Capacity Factor[4]
Yankee-Rowe	029	0.0	0.0	0.0	0.0
Zion 1	295	49.1	49.1	45.0	45.0
Zion 2	304	65.4	65.4	58.7	58.7
Industry Average		74.1	73.9	70.6	69.0

[1]Availability Factor = $\frac{\text{Unit Available Hours x 100}}{\text{Period Hours}}$

[2]Service Factor = $\frac{\text{Unit Service Hours x 100}}{\text{Period Hours}}$

[3]Maximum Dependable Capacity Factor = $\frac{\text{Net Electrical Energy Generated x 100}}{\text{Period Hours x MDC Net}}$

[4]Design Electrical Rating Capacity Factor = $\frac{\text{Net Electrical Energy Generated x 100}}{\text{Period Hours x DER}}$

Unit Available Hours	The total clock hours in the report period during which the unit operated on line or was capable of such operation.
Unit Service Hours	The total clock hours in the report period during which the unit operated with breakers closed to the station bus. These hours added to the total outage hours experienced by the unit during the report period shall equal the hours in the report period.
Net Electrical Energy Generated	Gross electrical output of the unit measured at the output terminals of the turbine generator during the reporting period, minus the normal station service electrical energy utilization. If this quantity is less than zero, a negative number should be recorded.
Period Hours	For units in power ascension at the end of the period, the gross hours from the beginning of the period or the first electrical production, whichever comes last, to the end of the period. For units in commercial operation at the end of the period, the gross hours from the beginning of the period or of commercial operation, whichever comes last, to the end of the period or decommissioning, whichever comes first.
Maximum Dependable Capacity Net (MDC Net) (Net MWe)	Maximum dependable capacity gross less the normal station service loads.
Design Electrical Rating (DER) (NET MWe)	The nominal net electrical output of the unit specified by the utility and used for the purpose of plant design.

Table A-2.1 All Licensed Reactor Critical Hours by Year

Plant Name	Vendor	1988	1989	1990	1991	1992
Arkansas 1	B&W	6156.6	5999.1	6500.2	8149.8	7137.8
Arkansas 2	CE	6032.0	6610.1	8246.6	7341.1	6454.2
Beaver Valley 1	W	7066.7	5887.6	8155.9	5029.2	8226.7
Beaver Valley 2	W	8283.8	6307.[illegible]	6790.5	8732.9	7421.0
Big Rock Point	GE	6[illegible]94.2	6920.8	6759.0	7460.5	4790.5
Braidwood 1	W	5746.1	5586.8	7830.1	5352.9	7237.0
Braidwood 2	W	4756.1	7618.1	6904.2	6727.1	8395.9
Browns Ferry 1	GE	0.0	0.0	0.0	0.0	0.0
Browns Ferry 2	GE	0.0	0.0	0.0	4646.3	8496.0
Browns Ferry 3	GE	0.0	0.0	0.0	0.0	0.0
Brunswick 1	GE	6660.7	5749.3	5948.2	6061.3	2517.9
Brunswick 2	GE	5645.8	5779.9	5926.6	5236.2	2378.3
Byron 1	W	6485.1	8742.7	7144.2	7242.7	8731.4
Byron 2	W	8676.0	7060.4	6667.1	8502.0	7101.5
Callaway	W	8202.1	7481.6	7365.0	8734.1	7289.2
Calvert Cliffs 1	CE	6398.5	1806.6	1924.5	6687.0	5050.2
Calvert Cliffs 2	CE	7827.1	1718.4	0.0	4651.0	7924.1
Catawba 1	W	7070.3	7485.1	6348.9	6372.7	6396.3
Catawba 2	W	6496.8	6448.2	6047.5	6699.6	8348.8
Clinton 1	GE	7399.4	4244.3	4826.8	7079.5	6025.3
Comanche Peak 1	W	NYC	NYC	5302.5	5488.8	7103.3
Cook 1	W	8433.8	6169.8	6944.8	7754.3	5752.1
Cook 2	W	2715.5	6580.9	4958.9	8053.2	3169.4
Cooper Station	GE	5967.9	6672.9	6953.3	6898.8	8466.7
Crystal River 3	B&W	7457.3	4274.4	5591.1	7187.2	6684.2
Davis-Besse	B&W	2126.7	8547.1	4966.6	7054.6	8759.2
Diablo Canyon 1	W	5682.3	7189.1	8504.3	7197.4	7297.6
Diablo Canyon 2	W	6190.7	8136.8	7432.9	7486.1	8672.9
Dresden 2	GE	6974.7	7252.5	5958.8	5279.9	7553.4
Dresden 3	GE	6346.3	7311.6	7453.4	5356.0	5689.3
Duane Arnold	GE	6609.9	6921.1	6641.2	8277.5	7192.9
Farley 1	W	7428.3	7613.4	8695.9	6987.0	7210.4
Farley 2	W	8784.0	7205.2	6501.1	8480.1	7157.6
Fermi 2	GE	5325.8	6002.4	7420.8	6746.0	7139.5
FitzPatrick	GE	6060.6	8086.8	6356.0	4675.2	0.0
Fort Calhoun	CE	6510.0	7816.5	5622.4	8030.0	5791.6

Footnotes at end of table.

Table A-2.11 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992
Fort St. Vrain*	GA	3798.1	3331.9	PSD	PSD	PSD
Ginna	W	7679.2	6648.5	7393.2	7591.6	7633.7
Grand Gulf	GE	8498.1	7005.5	6911.1	8230.3	7349.0
Haddam Neck	W	6177.0	5883.3	2824.5	6693.2	7039.6
Harris	W	6585.1	6962.6	7848.6	7141.9	6580.8
Hatch 1	GE	6008.8	8760.0	5939.6	6790.3	8566.3
Hatch 2	GE	6359.2	6495.8	8684.7	6778.8	7004.9
Hope Creek	GE	7089.5	6813.9	8020.0	7379.8	7094.3
Indian Point 2	W	7491.8	5644.2	5837.0	4762.7	8625.4
Indian Point 3	W	7312.7	5352.0	5511.3	7668.5	5397.0
Kewaunee	W	7755.6	7436.2	7700.5	7306.0	7726.0
LaSalle 1	GE	5931.1	6114.8	8475.3	6747.1	6568.3
LaSalle 2	GE	6648.2	6693.0	6343.1	8445.6	6077.7
Limerick 1	GE	8476.3	5784.5	6002.8	8177.2	6240.4
Limerick 2	GE	NYC	1961.8	7727.3	7029.0	8784.0
Maine Yankee	CE	6949.7	8210.0	6215.9	7585.4	6950.9
McGuire 1	W	6783.8	7210.8	4807.9	6327.6	6862.8
McGuire 2	W	7313.5	6943.4	5937.3	8561.3	6214.9
Millstone 1	GE	8661.6	7377.3	8021.0	3099.9	5983.6
Millstone 2	CE	6953.1	6027.7	6551.5	5141.0	3204.0
Millstone 3	W	7196.3	6716.1	7909.2	2962.2	6490.8
Monticello	GE	8768.7	6679.1	8487.3	7075.6	8566.3
Nine Mile Pt. 1	GE	0.0	0.0	3365.8	6987.8	5206.0
Nine Mile Pt. 2	GE	4525.4	5206.2	4800.4	6971.8	5648.4
North Anna 1	W	8019.5	5023.1	8748.4	6697.6	7242.3
North Anna 2	W	8734.9	6918.9	7012.2	8601.6	7308.2
Oconee 1	B&W	8769.0	7371.0	7774.7	7287.5	7586.1
Oconee 2	B&W	6989.2	7385.8	7505.7	8760.0	7229.3
Oconee 3	B&W	7229.7	7682.9	8730.6	6740.6	6803.2
Oyster Creek	GE	5789.0	5015.2	7804.6	5297.6	7545.8
Palisades	CE	4990.4	6050.6	5143.1	6845.5	6686.0
Palo Verde 1	CE	5762.9	1522.0	4198.1	7598.9	6116.5
Palo Verde 2	CE	5750.0	4226.0	5376.2	6718.5	8479.6
Palo Verde 3	CE	8369.7	1209.5	8168.5	6418.0	7010.0
Peach Bottom 2	GE	0.0	5330.5	7173.4	5553.3	6130.1
Peach Bottom 3	GE	0.0	801.4	7844.1	5359.2	7696.3

Footnotes at end of table.

Table A-2.11 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992
Perry	GE	6939.2	4997.0	5880.4	8054.6	6629.7
Pilgrim	GE	0.3	5613.8	7195.5	5759.9	7498.0
Point Beach 1	W	7847.7	7728.3	7423.8	7622.9	7492.8
Point Beach 2	W	7707.8	7243.6	7738.8	7645.2	7546.1
Prairie Island 1	W	7835.6	8740.7	7840.4	7988.3	6850.8
Prairie Island 2	W	7813.9	7852.4	7785.[illegible]	8760.0	6538.2
Quad Cities 1	GE	8477.9	6621.4	7318.1	5030.2	6249.8
Quad Cities 2	GE	6292.8	8434.7	6304.6	7794.5	5692.6
River Bend	GE	8280.0	6051.7	6835.2	7642.4	3487.4
Robinson 2	W	5791.5	4262.0	5674.7	7131.0	5867.4
Salem 1	W	6937.1	6276.4	6055.0	6636.8	5581.8
Salem 2	W	5992.9	7650.0	5350.5	7259.9	5149.4
San Onofre 1	W	3817.7	3582.8	4162.8	5790.3	8022.0
San Onofre 2	CE	8286.3	5227.0	7692.8	5732.7	8242.0
San Onofre 3	CE	5930.8	8251.6	6297.7	8270.3	6701.5
Seabrook	W	NYC	194.4	5524.9	6646.0	7137.9
Sequoyah 1	W	379.5	8671.3	6576.8	6882.1	7794.0
Sequoyah 2	W	5202.1	6343.8	6940.8	8537.1	7204.5
South Texas 1	W	5172.3	5750.7	5533.8	6238.9	6121.5
South Texas 2	W	NYC	4514.1	6004.6	6441.4	8594.0
St. Lucie 1	CE	7554.3	8290.1	5569.7	7151.0	8561.0
St. Lucie 2	CE	8784.0	6626.9	6691.4	8760.0	6039.9
Summer	W	6067.7	7276.2	7346.2	7265.5	8553.1
Surry 1	W	3755.2	4272.2	6723.4	8760.0	7140.8
Surry 2	W	5028.3	1504.3	7973.7	6035.8	8478.8
Susquehanna 1	GE	8289.7	6592.5	6769.1	8622.5	6747.2
Susquehanna 2	GE	6156.9	6916.4	8197.5	7119.1	7255.8
Three Mile Island 1	B&W	6760.9	8717.2	7165.6	7566.7	8745.8
Trojan	W	5925.3	5423.2	5810.5	1409.0	4797.0
Turkey Point 3	W	5408.1	5806.6	5283.7	2252.1	6034.2
Turkey Point 4	W	5050.1	4147.1	6802.7	1426.3	7226.1
Vermont Yankee	GE	8404.4	7416.2	7522.8	8265.0	7742.8
Vogtle 1	W	6822.3	8413.0	7170.5	7180.4	8563.0
Vogtle 2	W	NYC	6134.5	7325.7	8455.0	7253.5
Wash. Nuclear 2	GE	6310.9	6857.8	5908.9	4406.5	5758.0
Waterford 3	CE	6624.5	7232.6	8131.0	6993.7	6856.6

Footnotes at end of table.

Table A-2.11 (cont.)

Plant Name	Vendor	1988	1989	1990	1991	1992
Wolf Creek	W	6117.6	8715.3	7096.1	6294.6	7612.4
Yankee-Rowe	W	7486.8	8137.2	5390.7	6331.6	0.0
Zion 1	W	6747.9	5268.3	5097.0	4652.6	4605.3
Zion 2	W	7004.6	8333.9	3122.7	5544.4	5758.7
Total All Plants**		664323.1	667111.7	706720.0	735347.2	733342.1
Total All W Plants		313019.0	328494.6	340879.4	348341.5	358555.9
Total All GE Plants		199293.3	204482.1	231776.7	230335.2	221772.5
Total All CE Plants		102723.3	80825.6	85829.4	103924.1	100068.1
Total All B&W Plants		45489.4	49977.5	48234.5	52746.4	52945.6

*Fort St. Vrain, a high-temperature gas reactor designed by General Atomic (GA) Corporation, ceased all operations on August 18, 1989.
**This total includes Fort St. Vrain.

Note: NYC means the plant is not yet critical; PSD means the plant is permanently shut down.

Appendix B

Summary of 1992 Abnormal Occurrences (Reactors)

1

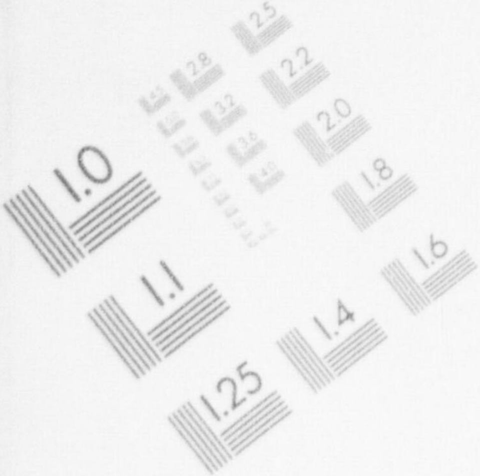
IMAGE EVALUATION
TEST TARGET (MT-3)

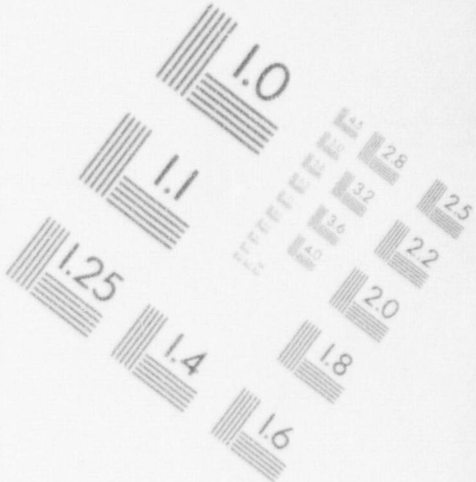
150mm

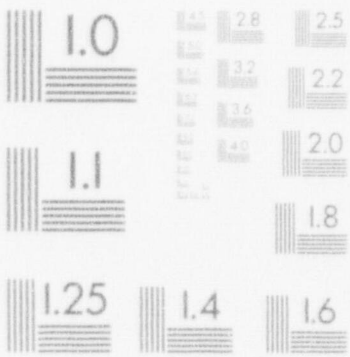
6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

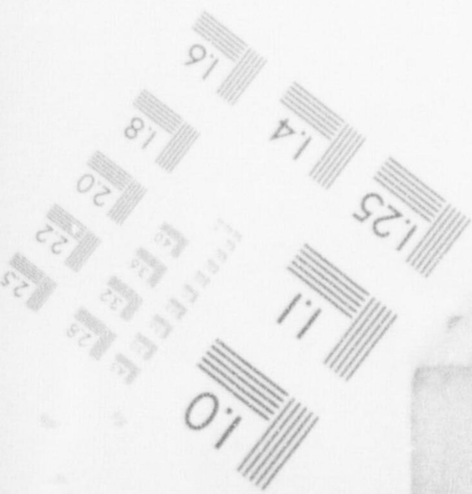


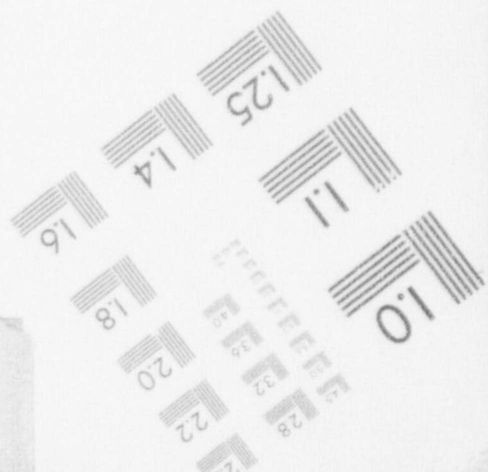




150mm

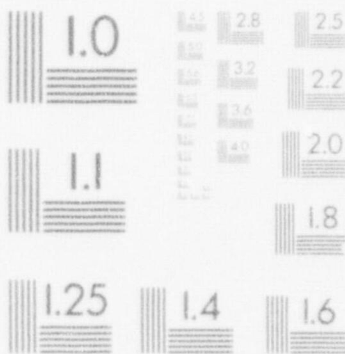
6″





1

IMAGE EVALUATION TEST TARGET (MT-3)

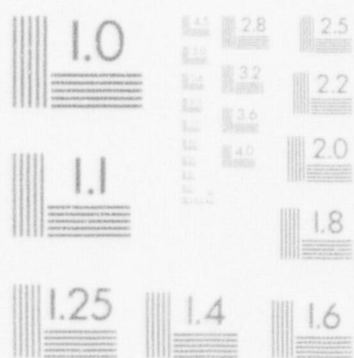


150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



150mm

6″

NUREG-0090, Volume 15, No. 2

Report No. 92-4

Loss of High-Head Safety Injection Capability at Shearon Harris Nuclear Power Plant

When Shearon Harris Unit 1 was shut down for refueling on April 3, 1991, the licensee identified degradated piping and relief valves used in the high-head safety injection (HHSI) system that would have adversely affected the performance of the HHSI system during the previous operating cycle should it have been required to operate and deliver water to the primary coolant system. The HHSI flow rate required to mitigate design-basis accidents assumed in the licensing analysis would not have been attained because a significant amount of flow would have been diverted by the piping failure and early opening of the relief valve. The event was subsequently analyzed under the NRC Accident Sequence Precursor Program and found to have potential safety significance because the conditional core damage probability was 6E-3.

When the NRC determined that the event had potential safety significance, a special inspection team was sent to the plant to evaluate the event, including the licensee's corrective actions and the actual safety significance of the event. The team learned that the degraded piping and relief valves were part of the alternate minimum flow (AMF) system, which protects the charging/safety injection pumps by providing a minimum flow path via relief valves to the refueling water storage tank whenever the pumps operate against a primary system pressure head that is greater than the pump discharge pressure. Without the AMF system, pump damage from lack of flow could preclude the pumps from performing their safety functions. The AMF system, which was installed as part of the original facility before it received an operating license, was deficient. The physical layout of the AMF piping permitted air to be trapped on both sides of the relief valves. The trapped air could result in water hammer events during operational transients. The team determined that recurrent water hammer events over the past 6 years during engineered safety feature testing and operation when air was trapped in the system had degraded the AMF relief valves and piping to the extent that the HHSI function would not be performed.

Licensee corrective actions included (1) repairing the damage and revising plant procedures to prevent recurrence, (2) revising procedures to eliminate the air, and (3) modifying the system and performing additional testing to ensure that the problem did not recur. The team identified the following AMF system unanalyzed potential design weaknesses: water hammer events upstream and downstream of the relief valves, AMF system piping transient or water hammer loads, and relief valve chatter and setpoint drift. Similar damage to AMF system components was previously identified at other facilities.

Report No. 92–12

Operation With Degraded Steam Generator Tubes at Arkansas Nuclear One Unit 2 and McGuire Nuclear Station Units 1 and 2

The licensees for Arkansas Nuclear One (ANO), Unit 2, and McGuire Nuclear Station, Units 1 and 2, notified the NRC that certain steam generator (SG) tubes in these plants were degraded, so that structural integrity could not be retained for the full range of plant conditions. Since the SG tubes constitute more than half of the primary coolant pressure boundary, their integrity is necessary to prevent the loss of primary coolant and the release of radioactive fission products. Requirements for periodic inservice inspection of the SG tubes have been established in the plant Technical Specifications (TS) to ensure that they retain structural integrity under all conditions.

In March 1992, ANO Unit 2 was shut down when a primary-to-secondary coolant leak was detected that was approximately half of its TS limit. The licensee conducted an eddy current inspection of the SG tubing and identified the location of the leak. A review of the eddy current test data from April 1991 showed that this tube had a flaw-like indication at that time. Six additional tubes were identified that were incorrectly analyzed in 1991. To determine the degradation mechanism, the licensee removed several tube sections from the SG for examination. The examination showed circumferential intergranular stress corrosion cracking (IGSCC) that began on the outer surface of the tubes, completely encircled them, and extended deep within the tube wall.

In January 1992, McGuire Unit 1 was shut down after the primary-to-secondary coolant leak rate became excessive. The licensee identified the source and location of the leak using eddy current techniques. Review of the eddy current test data from the October 1991 inspection showed that this tube had a flaw indication. The tube was not removed from service as required by the plant TS because of improper classification and administrative handling. In addition, an indication was found that had not been identified in the inspection of October 1991.

Since McGuire Unit 2 was in a scheduled refueling outage, the licensee decided to characterize the degradation mechanism of the SG tube by removing several tube sections from the same location in the Unit 2 SGs for analysis. The eddy current indications in these sections exhibited characteristics similar to those in the Unit 1 SGs. In April 1992, the licensee reported the preliminary results of the examination to the NRC. One tube was found to contain axially oriented IGSCC, beginning on the outer surface of the tube, which reduced its burst pressure to below safe limits.

In May 1992, McGuire Units 1 and 2 were shut down for further examination of SG tubing. Several tube sections were removed from the Unit 1 SGs, including the tube that had leaked and was subsequently plugged in January 1992. Destructive examination of the January 1992 tube segment showed an axially oriented stress corrosion crack that started at the outer surface of the tube. The structural integrity of this tube was not adequate.

It should be noted that a previous SG tube failure, in this case a SG tube rupture, occurred at McGuire Unit 1 in March 1989 and was reported as an abnormal occurrence in NUREG–0090, Vol. 12, No. 1.

The missed eddy current indications at ANO Unit 2 were attributed to (1) a lack of training of the eddy current data analysts, (2) a lack of a performance demonstration test by the data analysts using actual site data, and (3) inherent difficulties in analyzing signals at the location where the defect was found. The licensee took corrective actions and plans to perform a midcycle inspection of SG tubes beginning at the end of April 1993.

The primary-to-secondary leakage observed at McGuire Unit 1 in January 1992 was attributed primarily to a SG tube that had not been removed from service because its flaw indication had been misclassified. The licensee took corrective actions to address the causes of the missed indication.

NUREG–0090, Volume 15, No. 4

Report No. 92–13

Engineered Safety Features Actuation System Design Deficiency—Single Failure Vulnerability at Millstone Power Station Unit 2

In July 1992, during a planned outage at Millstone Nuclear Power Station, Unit 2, the licensee was preparing to replace two vital inverters. The plant uses four inverters, two on each vital dc bus, to power two trains of the engineered safety features actuation system (ESFAS). The control circuitry is contained in four sensor cabinets and two actuation cabinets. Operators removed power from one actuation train, causing a false loss of normal power signal and a false start signal for the emergency core cooling system. The effect of this action was similar in consequence to the complete loss of one of the two vital dc buses.

One emergency diesel generator (EDG) started and electrically connected to the bus, while the second did not start because it was out of service for maintenance. After the one EDG started, the safety loads failed to sequence onto the ac bus because of a continuous false load shed signal. Operators recovered from the event by stopping the EDG and restoring power to one of the sensor cabinets. As a result of this action, the false loss of power signal and thus the load shed signal were removed.

The event occurred because the unblocking feature of the automatic test insertion (ATI) system had caused the continuous load shed signal. The ATI system, a continuous on-line logic tester that is common to both trains, was still energized and permitted the spurious loss of power signal to continue to shed the loads. The design deficiency in the on-line testing feature could have prevented both EDGs from accepting emergency loads under certain single-failure conditions.

The root cause of the event was a failure to correctly transfer design package requirements into the plant modification, which identified the proper sequence for replacing and turning on the inverters. However, when the work order was prepared, the planned sequence was not followed.

The licensee also determined that the ESFAS could cause the following unintended automatic actions (not related to the ATI modification) under certain power supply failure conditions: (1) with loss of power to either one of the two dc vital buses, both the safety injection actuation signal and the sump recirculation actuation signal would be simultaneously initiated, thereby tripping all low-pressure injection pumps and opening one of the containment sump outlet valves; (2) with loss of power to the sensor cabinets in one actuation train, both containment sump outlet valves would open; if this occurred during a loss-of-coolant accident, high pressure in the containment could shut both refueling water storage tank check valves, thereby inhibiting flow to all emergency coolant injection pumps; and (3) loss of all dc power to one actuation train would cause a power- operated relief valve in the other train to open, and loss of control power to the sensor cabinets in a single actuation train would generate spurious high pressurizer pressure signals that would cause the relief valves in both trains to open; both cases would result in a loss of primary coolant.

It should be noted that the licensee was aware of plant design vulnerabilities before this event. In addition, in 1990 the licensee discovered a longstanding technical specification interpretation that had permitted indefinite operation of an emergency electrical bus on the non-safety-related backup supply. The licensee took corrective actions and demonstrated satisfactory operation of the ESFAS.

Appendix C

Reports Issued During 1992 (Reactors)

AEOD Reports Issued in 1992

Case Studies

Date	Title	No.	Author
12/92	Human Performance in Operating Events	C92-01	J. Kauffman G. Lanik R. Spence E. Trager

Special Study Reports

Date	Title	No.	Author
07/92	Office for Analysis and Evaluation of Operational Data 1991 Annual Report, Power Reactors	NUREG-1272, Vol. 6, No. 1	
09/92	Operating Experience Feedback Report—Experience With Pump Seals Installed in Reactor Coolant Pumps Manufactured by Byron Jackson	NUREG-1275, Vol. 7	
12/92	Operating Experience Feedback Report—Human Performance in Operating Events	NUREG-1275, Vol. 8	
	Not issued	S92-01	
04/92	Safety and Safety/Relief Valve Reliability	S92-02	M. Wegner
06/92	Review of Operational Experience With Molded Case Circuit Breakers in U.S. Commercial Nuclear Power Plants	S92-03	J. Houghton W. Leschek P. O'Reilly D. Rasmuson
	Not issued	S92-04	
	Not issued	S92-05	
	Not issued	S92-06	
12/92	Pressure Locking and Thermal Binding of Gate Valves	S92-07	C. Hsu

Engineering Evaluations

Date	Title	No.	Author
05/92	Inadequate Management Control of Snubber Surveillance	E92-01	C. Hsu
06/92	Insights From Common-Mode Failure Events	E92-02	S. Israel

Technical Review Reports

Date	Title	No.	Author
01/92	Enhanced Setpoint Testing Procedures for Pressurizer Safety Valves at Oconee and Catawba	T92-01	M. Wegner
01/92	BWR 5 and 6 Events Applicable to Laguna Verde	T92-02	J. Kauffman N. Casas
06/92	Solenoid-Operated Valves and Related Equipment—A Status Report	T92-03	H. Ornstein
06/92	Recent Solenoid-Operated Valve Experiences Involving Maintenance and Testing Deficiencies	T92-04	H. Ornstein
06/92	Errors in Effective Reactor Trip Settings or Monitoring Associated With Excore Instrumentation	T92-05	S. Israel
09/92	Water Intrusion Into Sensitive Control Room Equipment	T92-06	J. Kauffman
09/92	Inoperability of the Standby Liquid Control System During Surveillance Testing at Nine Mile Point Unit 2	T92-07	L. Gundrum
10/92	Emergency Diesel Generator Start Frequency	T92-08	T. Cintula
11/92	Review of Manual Valve Failures	T92-09	S. Salah
12/92	Prospective Trend of Low Reliability Emergency Diesel Generators	T92-10	T. Cintula

Appendix D

Reports Issued From 1980 Through 1991 (Reactors)

Table D–1

Reports Issued in CY 1991

Special Study Reports

Date	Title	No.	Author
07/91	Office for Analysis and Evaluation of Operational Data 1990 Annual Report, Power Reactors	NUREG–1272, Vol. 5, No. 1	
09/91	Performance of Emergency Diesel Generators in Restoring Power to Their Associated Safety Buses—A Review of Events Occurring at Power	S91–01	T. Cintula

Engineering Evaluations

Date	Title	No.	Author
02/91	A Review of Water Hammer Events After 1985	E91–01	E. Brown

Technical Review Reports

Date	Title	No.	Author
02/91	Causes of Incorrect System Flows	T91–01	S. Israel
02/91	Incorrect Rotation of PDP	T91–02	T. Cintula
03/91	Overloaded Emergency Buses	T91–03	S. Israel
04/91	Turbine Overspeed Trip Due to Steam Valve Leakage and Condensate	T91–04	C. Hsu
05/91	Setpoint Testing of Pressurizer Safety Valves With Water-Filled Loop Seals	T91–05	M. Wegner
06/91	Deficiencies in External Flood Protection	T91–06	S. Israel
07/91	Evaluation of Partial Loss of Station Power Events at Prairie Island Unit No. 2 on December 21 and December 26, 1989	T91–07	F. Manning

Table D-2

Reports Issued in CY 1990

Case Studies

Date	Title	No.	Author
10/90	Operating Experience Feedback Report—Solenoid Operated Valve Problems at U.S. Light Water Reactors (Reprinted as NUREG-1275, Vol. 6)	C90-01	H. Ornstein

Special Study Reports

Date	Title	No.	Author
07/90	Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Power Reactors	NUREG-1272, Vol. 4, No. 1	
03/90	Review of Thermal Stratification Operating Experience	S902	T. Su
08/90	Recurrence of Important Safety Issues Reported in LERs	S90-01	S. Israel

Engineering Evaluations

Date	Title	No.	Author
02/90	Failures of Electrical Supply and Power Generation Equipment Which Disrupted Plant Function at Nuclear Power Plants	E90-01	M. Wegner
02/90	Crosby Low Pressure Relief Valves	E90-02	S. Israel
05/90	Overpressurization of Auxiliary Feedwater Systems	E90-03	C. Hsu
04/90	Swelling and Cracking in Hafnium Control Rods	E90-04	M. Wegner
05/90	Operational Experience on Bus Transfer	E90-05	S. Mazumdar
07/90	Potential for Residual Heat Removal System Pump Damage	E90-06	C. Hsu
07/90	Effects of Internal Flooding of Nuclear Power Plants on Safety Equipment	E90-07	T. Su
09/90	Low Temperature Overpressure Protection: Testing PORVs With the Alternate Pneumatic Supply	E90-08	S. Israel S. Salah
10/90	Additional Factors Affecting the Lift Setpoint of Pressurizer Safety Valves	E90-09	L. Padovan
12/90	Evaluation of Boiling Water Reactor Mode Switch Events	E90-10	W. Jones

Technical Review Reports

Date	Title	No.	Author
01/90	PNO's Issued in First Quarter of 1989	T90-01	R. Dennig T. Wolf

Table D–2 (cont.)

Technical Review Reports (cont.)			
Date	Title	No.	Author
01/90	Insights Regarding Commonwealth Edison Plant Root-Cause Determinations Related to Maintenance Effectiveness (Proprietary)	T90–02	N. Thomasson
03/90	Improper Installation of Heat Shrinkable Tubing	T90–03	S. Mazumdar
03/90	Reverse (Backward) Acting Valve Manual Handwheels	T90–04	T. Cintula
03/90	Association Between Nuclear Plant Utilization and Incentive Regulation by Station Public Utility Commissions	T90–05	S. Stern
05/90	Aquatic Life in Emergency Cooling Ponds	T90–06	L. Padovan
05/90	Reversed Sensing Lines Connections	T90–07	B. Kaufer
06/90	Turbine Bypass Malfunctions	T90–08	B. Kaufer
06/90	Inadvertent Partial Draining of Condensate Storage Tanks	T90–09	T. Cintula
07/90	Evaluation of Maintenance Trends at Five Selected Sites (Proprietary)	T90–10	P. O'Reilly
07/90	Evaluation of Safety Equipment Outages For Significance at Zion (Revised)	T925A	F. Manning
08/90	Effect of High Energy Line Breaks on Chilled Water Systems at Nuclear Power Plants	T90–11	L. Padovan
09/90	Loss of Offsite Power To Comply With NRC Regulations	T90–12	T. Cintula
10/90	Corrosion and Failure of Service Water Pump Impeller Snap Rings	T90–13	C. Hsu
10/90	Seal Problems in Boric Acid Transfer Pumps	T90–14	S. Israel
10/90	Salem 1 and 2 Evaluation of Operating Experience (Proprietary)	T90–15	P. O'Reilly
11/90	Impact of Pipe Liner Failure of Pump Operation	T90–16	S. Israel
12/90	Inadvertent Containment Spray Actuations	T90–17	M. Harper

Table D–3

Reports Issued in CY 1989

Case Studies

Date	Title	No.	Author
	None		

Special Study Reports

Date	Title	No.	Author
06/89	Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Power Reactors	NUREG–1272, Vol. 3, No. 1	
03/89	Operating Experience Feedback Report—Technical Specifications	NUREG–1275, Vol. 4	P. O'Reilly G. Plumlee
03/89	Operating Experience Feedback Report—Progress in Scram Reduction	NUREG–1275, Vol. 5	L. Bell P. O'Reilly
08/89	Operating Experience Feedback Report—Progress in Scram Reduction	NUREG–1275, Vol. 5 Addendum	L. Bell
01/89	Application of the NPRDS for Effectiveness Monitoring (Appendices A and B are proprietary)	S804B	P. O'Reilly T. Wolf P. Cross-Prather
02/89	Maintenance Programs at Nuclear Power Plants (Table 2 is proprietary)	S901 Revision 1	M. Chiramal S. Israel M. Wegner S. Stern

Engineering Evaluations

Date	Title	No.	Author
02/89	Problems With Oils, Greases, Solvents and Other Chemical Materials	E901	S. Israel
03/89	Fire and Explosive Mixtures Resulted From Introduction of Hydrogen Into Plant Air Systems	E902	H. Ornstein
	Not issued	E903	
04/89	On Demand Malfunctions of HPCI and RCIC	E904	T. Cintula
06/89	Electrical Bus Bar Failures	E905	M. Padovan
08/89	Failure of Steam Generator Isolation Check Valve	E906	T. Cintula
09/89	Diversion of Seal Cooler Flow for RHR Pumps	E907	S. Israel
10/89	Excessive Valve Body Erosion at Brunswick	E908	E. Brown
12/89	Operator Actions During Operational Events	E909	S. Israel

Table D-3 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
12/89	Potential for Gas Binding of High Head Safety Injection Pumps Resulting From Inservice Testing of VCT Outlet Isolation Valves	E910	M. Padovan

Technical Review Reports			
Date	Title	No.	Author
01/89	Millstone Unit 1—Safety/Relief Valve Discharge Line Vacuum Breakers Failed Open	T901	T. Su
02/89	Inadvertent Reactor Trips Due to RCS Flow Instrumentation Maintenance Activities	T902	M. Padovan
03/89	Generic Implication of Browns Ferry Fire on November 2, 1987	T903	T. Su
04/89	Design Deficiency of Safety Injection Block Switch	T904	S. Mazumdar
04/89	Failure of 4160V GE Magneblast Breaker To Trip Open	T905	S. Mazumdar
04/89	Broken Lifting Beam Bolts in HPCI Terry Turbine	T906	T. Cintula
04/89	Component Degradation Due to Indiscriminate Painting	T907	M. Padovan
	A nonreactor report; See NUREG-1272, Vol. 4, No. 2	T908	
05/89	Operating Events Involving Dampers	T909	S. Israel
06/89	Investigation of Cracked Control Rod Drive Seal Housings at Palisades	T910	W. Jones
06/89	Evaluation of Individually Reported Safety System LERs for Their Combined Significance	T911	F. Manning
06/89	Selected Maintenance Rework	T912	S. Israel
07/89	Comparison of the Proposed Maintenance Effectiveness (ME) Indicator With Catawba and Farley Nuclear Plants Regarding Inspections (Proprietary)	T913	N. Thomasson T. Wolf M. Harper
09/89	Overview of Design/Installation Fabrication Errors in 1988	T914	S. Israel
09/89	EDG Ground Fault Detection and Trip Circuit at Perry Unit 1	T915	S. Mazumdar
09/89	Debris in Containment Recirculation Sumps	T916	M. Padovan
	Not issued (refer to E908)	T917	
09/89	Check Valve Failure Rates From NPRDS Data	T918	E. Brown

Table D-3 (cont.)

Date	Technical Review Reports (cont.) Title	No.	Author
09/89	Failure of Overcurrent Protective Device at Palisades Unit 1	T919	S. Mazumdar
	Not issued	T920	
10/89	Inadequate Capacity of 4160V Switchgear at FitzPatrick	T921	S. Mazumdar
11/89	Failure of HPCI Turbine Due to High Moisture in Lube Oil	T922	C. Hsu
11/89	Delaminating Foil Insulation in Primary Containment	T923	T. Cintula
	Not issued	T924	
12/89	Evaluation of Safety Equipment Outages for Significance at Zion	T925	F. Manning
12/89	Evaluation of Two Beaver Valley 2 Nuclear Plant Equipment Degradation Events for Their Combined Significance	T926	F. Manning
12/89	Follow-up on Steam Binding of AFW Pumps	T927	C. Hsu
12/89	Inadequate Overpressure Protection for Auxiliary Steam Headers at the Oconee Plants	T928	S. Salah

Table D-4

Reports Issued in CY 1988

Case Studies

Date	Title	No.	Author
08/88	Service Water System Failures and Degradations in Light Water Reactors	C801	P. Lam E. Leeds

Special Study Reports

Date	Title	No.	Author
03/88	Significant Events That Involved Procedures	S801	E. Trager
03/88	Operational Experience Feedback Evaluation Rancho Seco Nuclear Generating Station, Restart	S802	G. Plumlee
06/88	AEOD Concerns Regarding the Power Oscillation Event at LaSalle 2 (BWR-5)	S803	J. Kauffman
08/88	Preliminary Results of the Trial Program for Maintenance Performance Indicators	S804A	
09/88	Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data—1987 Power Reactors (NUREG-1272, Vol. 2, No. 1)	S804	

Engineering Evaluations

Date	Title	No.	Author
04/88	BWR Overfill Events Resulting in Steam Line Flooding	E801	J. Kauffman
05/88	Design and Operating Deficiencies in Control Room Emergency Ventilation Systems	E802	S. Israel
08/88	Inadequate NPSH in High Pressure Safety Injection Systems in PWRs	E803	S. Israel
08/88	Reliability of Recirculation Pump Breaker During an ATWS	E804	T. Su
09/88	Potential LOCA Due to Energized Uncovered Pressurizer Heaters	E805	T. Cintula
10/88	Loss of Decay Heat Removal Due to Rapid Refueling Cavity Pumpdown	E806	M. Padovan
10/88	Pump Damage Due to Low Flow Cavitation	E807	C. Hsu
12/88	Operational Experience Review of Potential Large Openings in Containment	E808	T. Cintula

Table D-4 (cont.)

Technical Review Reports			
Date	**Title**	**No.**	**Author**
01/88	Perry Nuclear Power Plant Unit 1—Unexpected MSIVs Closure and Reopening	T801	T. Su
05/88	Summary of Early Operational Experience of Foreign Commercial Nuclear Reactors (Proprietary)	T803	P. O'Reilly
05/88	"Precursor" Operational Events That Occurred From November 1, 1987, Through March 1988	T804	F. Manning
05/88	Insights From Significant Events in 1987	T805	S. Israel
05/88	Recent Operational Experience Trends at Fermi 2	T806	T. Wolf
06/88	Recent Operational Experience Trends at Indian Point 2	T807	T. Wolf
06/88	A Technical Basis for Granting Test Frequency Relief	T808	G. Plumlee
06/88	Blocked Thimble Tubes/Stuck Incore Detector	T809	M. Wegner
07/88	An Analysis of NPRDS Data for Hatch Plant (Proprietary)	T810	T. Wolf P. Cross-Prather
11/88	Degradation of Ice Condenser Containment Functional Capability	T811	F. Manning

Incident Investigation Program Reports			
Date	**Title**	**No.**	**Author**
02/88	Incident Investigation Manual	NUREG-1303	

Table D–5

Reports Issued in CY 1987

Case Studies

Date	Title	No.	Author
03/87	Air Systems Problems at U.S. Light Water Reactors	C701	H. Ornstein

Engineering Evaluations

Date	Title	No.	Author
01/87	Potential Containment Airlock Window Failure Due to Radiation	E701	S. Israel
03/87	MOV Failure Due to Hydraulic Lockup From Excessive Grease in Spring Pack	E702	E. Brown
03/87	Loss of Offsite Power Due to Unneeded Actuation of Startup Transformer Protection Differential Relay	E703	F. Ashe
03/87	Discharge of Primary Coolant Outside of Containment at PWRs While on RHR Cooling	E704	S. Israel
03/87	RWCU System Automatic Isolation and Safety Considerations	E705	N. Thomasson
03/87	Inadequate Mechanical Blocking of Valves	E706	T. Cintula
03/87	Design and Construction Problems at Operating Nuclear Plants	E707	C. Hsu
08/87	Depressurization of Reactor Coolant Systems at PWRs	E708	S. Israel
08/87	Auxiliary Feedwater Pump Trips Due to Low Suction Pressure	E709	C. Hsu
10/87	Inadequate NPSH in Low-Pressure Safety Systems in PWRs	E710	S. Israel

Program Support Reports

Date	Title	No.	Author
07/87	Operational Experiences at Newly Licensed Nuclear Power Plants	NUREG-1275, Vol. 1	R. Dennig
09/87	Trends and Patterns Program Report—Operational Experience Feedback on Main Feedwater Flow Control and Main Feedwater Flow Bypass Valves and Valve Operators	P701	G. Plumlee

Technical Review Reports

Date	Title	No.	Author
01/87	Compression Fitting Failures	T701	H. Ornstein

Table D-5 (cont.)

Technical Review Reports (cont.)

Date	Title	No.	Author
03/87	Leaking Pulsation Dampener Leads to Loss of Charging System	T702	T. Cintula
03/87	Potential for Loss of Emergency Feedwater Caused by Pump Runout During Certain Transients	T703	M. Wegner
03/87	Pressurizer Code—Safety Valve Reliability	T704	M. Wegner
05/87	Occurrence of Events Involving Wrong Unit/Wrong Train/Wrong Component—Update Through 1986	T705	E. Trager
06/87	Recent Events Involving Turbine Runbacks at PWRs	T706	E. Leeds
08/87	Undetected Loss of Reactor Water	T707	S. Israel
08/87	Problems With High Pressure Safety Injection Systems in Westinghouse PWRs	T708	S. Israel
10/87	Recent New Plant Operational Experience	T709	T. Wolf
11/87	Heating Ventilating, and Air Conditioning System Problems	T710	M. Chiramal
11/87	Unplanned Criticality Events at U.S. Power Reactors Similar to That at Oskarshamm Unit 3 on 07/30/87	T712	T. Wolf
12/87	Mispositioning of "Reverse Acting" Valve Controllers	T713	J. Stewart

Special Study Reports

Date	Title	No.	Author
05/87	Report to the U.S. Regulatory Commission on Analysis and Evaluation of Operational Data—1986	NUREG-1272	
12/87	Operating Experience Feedback Report—Air Systems Problems	NUREG-1275, Vol. 2	H. Ornstein
05/87	Loss of Decay Heat Removal Function at Pressurized Water Reactors With Partially Drained Reactor Coolant Systems	S702	H. Ornstein

Table D–6

Reports Issued in CY 1986

Case and Special Studies

Date	Title	No.	Author
08/86	Operational Experience Involving Turbine Overspeed Trips	C602	C. Hsu
12/86	A Review of Motor-Operated Valve Performance	C603	E. Brown
12/86	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	C604	M. Chiramal
12/86	Operational Experience Involving Losses of Electrical Inverters	C605	F. Ashe
01/86	Trends and Patterns Program Plan—FY86–FY88	P601	R. Dennig
08/86	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985	P602	L. Bell
08/86	Trends and Patterns Report of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants	P603	M. Harper
08/86	Trends and Patterns Report of the Operational Experience of Newly Licensed U.S. Nuclear Power Reactors	P604	T. Wolf
04/86	AEOD Annual Report for 1986	S601	J. Heltemes
05/86	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	S602	J. Crooks
06/86	Adequacy of the Scope of IE Bulletin 86–01	S603	E. Leeds

Engineering Evaluations and Technical Reviews

Date	Title	No.	Author
05/86	Core Damage Precursor Event at Trojan	E514 Revision 1	D. Zukor
01/86	Deficient Operator Actions Following Dual Function Valve Failures	E601	E. Leeds
01/86	Unexpected Criticality Due to Incorrect Calculation and Failure To Follow Procedures	E602	E. Leeds
02/86	Delayed Access to Safety Related Areas During Plant Operation	E603	T. Cintula
03/86	Spurious System Isolations Due to the Panalarm Model 86 Thermocouple Monitor	E604	E. Leeds

Table D-6 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
04/86	Lightning Events at Nuclear Power Plants	E605	M. Chiramal
05/86	Loss of Safety Injection Capability at Indian Point Unit 2	E606	R. Tripathi
07/86	Degradation or Loss of Charging Systems With Swing Pump Designs	E607	F. Ashe
07/86	Reexamination of Water Hammer Occurrences	E608	E. Leeds
08/86	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	E609	P. Lam
08/86	Loss of Low Pressure Coolant Injection Loop Selection Logic at Millstone Unit 1	E610	E. Leeds
10/86	Deficiencies in Seismic Anchorage for Electrical and Control Panels	E611	N. Thomasson
12/86	Emergency Diesel Generator Component Failures Due to Vibration	E612	C. Hsu
12/86	Localized Rod Cluster Control Assembly Wear at PWR Plants	E613	E. Brown
01/86	Pressure Sensitive Temperature Switch Results in Spurious Actuation of Fire Suppression System	T601	T. Cintula
04/86	Emergency Diesel Generator Cooling Water System Design Deficiencies at Maine Yankee and Haddam Neck	T602	E. Leeds
04/86	Inadvertent Pump Suction Transfer and Potential Auxiliary Feedwater Pump Cavitation at Davis-Besse	T603	R. Tripathi
05/86	Events Resulting From Deficiencies in Labeling and Identification Systems	T604	E. Trager
06/86	Failure of Main Steam Safety Valves To Properly Reseat	T605	R. Freeman
08/86	Inadvertent Recirculation Actuation Signals at Combustion Engineering Plants	T606	T. Cintula
09/86	Occurrence of Events Involving Wrong Units/Wrong Train/Wrong Component—Update Through June 1986	T607	E. Trager
11/86	Hydrogen Fire and Failure of Detection System	T608	M. Chiramal
12/86	Foreign Material and Debris in Safety-Related Fluid Systems	T609	E. Leeds
12/86	ADS/RCIC System Interaction Events at River Bend Unit 1	T610	E. Leeds
12/86	Denied Access Due to Negative Room Pressure	T611	T. Cintula

Table D-6 (cont.)

Incident Investigation Program Reports			
Date	Title	No.	Author
12/86	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	T612	R. Tripathi
01/86	Loss of Power and Water Hammer Event at San Onofre, Unit 1 on November 21, 1986	NUREG-1190	
02/86	Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985	NUREG-1195	
08/86	Incident Investigation Manual*		
12/86	Incident Investigation Manual, Revision 1*		

*Superseded by NUREG-1303 ("Incident Investigation Manual"), published 2/88. (See Table D-4.)

Table D-7

Reports Issued in CY 1985

Case and Special Studies			
Date	Title	No.	Author
09/85	Licensee Event Report System, Evaluation of First Year Results and Recommendations for Improvements	NUREG-1022, Supp. No. 2	
06/85	Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	C501	H. Ornstein
09/85	Overpressurization of Emergency Core Cooling in Boiling Water Reactors	C502	P. Lam
12/85	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	C503	H. Ornstein
12/85	Loss of Safety System Function Events	C504	E. Trager
07/85	Feedwater Transient Incidents in Westinghouse PWRs	P501	R. Dennig
06/85	Trends and Patterns Analysis of 1981 Through 1983 LER Data (NUREG/CR-4129)	P502	B. Brady
08/85	Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Reactors—January 1 Through June 30, 1984	P503	T. Wolf
08/85	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1984	P504	L. Bell
03/85	Review of Operational Experience From Non-Power Reactors	S501	D. Zukor
04/85	AEOD Semiannual Report for July-December 1984	S502	J. Heltemes
09/85	Evaluation of Recent Valve Operator Motor Burnout Events	S503	E. Brown

Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
01/85	Motor-Operated Valve Failures Due to Hammering Problem	E501	M. Chiramal
01/85	Failure of Residual Heat Removal Suppression Pool Cooling Valve To Operate	E502	C. Hsu
03/85	Partial Failures of Control Rod Systems To Scram	E503	M. Chiramal
03/85	Loss or Actuation of Various Safety-Related Equipment Due to Removal of Fuses or Opening of Circuit Breakers	E504	F. Ashe

Table D-7 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
03/85	Service Water System Air Release Valve Failures	E505	S. Salah
05/85	Valve Stem Susceptibility to Intergranular Stress Corrosion Cracking Due to Improper Heat Treatment	E506	C. Hsu
05/85	Electrical Interaction Between Units During Loss of Offsite Power Event of August 21, 1984 at McGuire Units 1 and 2	E507	M. Chiramal
05/85	Nuclear Plant Operating Experience Involving Safety System Disturbances Due to Bumped Electro-Mechanical Components	E508	S. Rubin
07/85	Salem Unit 2 Depressurization Event	E509	R. Freeman
07/85	Disabling of a Shared Diesel Generator Set Due to Electrical Power Supply Arrangement for Support Auxiliaries	E510	F. Ashe
08/85	Closure of Emergency Core Cooling System Minimum Flow Valves	E511	E. Leeds
09/85	Failure of Safety-Related Pumps Due to Debris	E512	R. Freeman
09/85	High Pressure Core Spray System Relief Valve Failures	E513	S. Salah
10/85	Core Damage Precursor Event at Trojan	E514	D. Zukor
12/85	Inadvertent Actuation of Safety System Due to Cross Talk	E515	M. Chiramal
01/85	Failure of Automatic Protection for Boron Dilution Event at Callaway Unit 1	T501	R. Freeman
03/85	Comparative Analysis of Recent Feedline Water Hammer Events at Maine Yankee, Calvert Cliffs, Palisades, and Salem	T502	E. Leeds
05/85	Pressurizer Level Instrumentation of Combustion Engineering Reactor Units	T503	M. Chiramal
05/85	Loss of Instrument Air and Subsequent Pressure Transient at Callaway Unit 1	T504	R. Freeman
07/85	Beaver Valley Component Cooling Water Pump Damage	T505	C. Hsu
07/85	Primary System Release Due to Pressurizer Degas Relief Valve Lifting	T506	T. Cintula
08/85	Standby Liquid Control System Pressure Relief Valves Lift at a Pressure Lower Than Reactor Coolant Pressure	T507	E. Brown
08/85	Browns Ferry Nuclear Plant High Pressure Coolant Injection System Performance Assessment	T508	E. Leeds

Table D-7 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
08/85	Inadequate Surveillance Testing Procedures for Degraded Voltage and Undervoltage Relays Associated With 4160-Volt Emergency Buses	T509	F. Ashe
09/85	Xenon Induced Power Oscillations at Catawba	T510	R. Freeman
09/85	Technicians Perform Work on Wrong Control Rod Drive Mechanism	T511	E. Trager
10/85	Incorrect Plugging of Steam Generator Tubes	T512	R. Freeman
11/85	Flooding of Safety-Related Valves in Pits	T513	D. Zukor
11/85	Potential Loss of Component Cooling Water Due to Maladjustment of Relief Valves	T514	D. Zukor
12/85	Residual Heat Removal Service Water Booster Pump Air Binding at Brunswick Unit 1	T515	S. Salah
12/85	High Pressure Coolant Injection Overspeed Trip Loss Events and Subsequent Damage Due to Water Hammer	T516	E. Trager

Date	Title	No.	Author
	Incident Investigation Program Reports		
07/85	Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985	NUREG-1154	

Table D–8

Reports Issued in CY 1984

Case and Special Studies

Date	Title	No.	Author
02/84	Licensee Event Report System, Description of System and Guidelines for Reporting	NUREG–1022, Supp. No. 1	
03/84	Low Temperature Overpressure Events at Turkey Point Unit 4	C401	W. Lanning
06/84	Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors	C402	M. El-Zeftawy
05/84	Hatch Unit 2 Plant Systems Interaction Event on August 25, 1982	C403	S. Rubin
07/84	Steam Binding of Auxiliary Feedwater Pumps	C404	W. Lanning
02/84	Operating History Overview for Diesel Generators in Nuclear Service	P401	R. Dennig M. Chiramal
03/84	AEOD Trends and Patterns Program Plan	P402	R. Dennig
05/84	AEOD Trends and Patterns Evaluation Report, Preliminary Assessment of LER Reporting Under 10 CFR 50.73	P403	F. Hebdon
03/84	LER Data on Personnel Errors	P404	F. Hebdon
11/84	Draft Trends and Patterns Analysis of Feedwater Transients at Westinghouse PWRs	P405	M. Harper
11/84	Trends and Patterns Analysis of Reactor Scrams (Pilot Study)	P406	L. Bell
01/84	Human Error in Events Involving Wrong Unit or Wrong Train	S401	E. Trager
07/84	Pressure Locking of Flexible Disk Wedge Type Gate Valves	S402	S. Rubin
06/84	Annual Report of U.S. NRC Participation in the Nuclear Energy Agency Incident Reporting System During 1983	S403	J. Crooks
06/84	Analysis of Foreign IRS Reports Submitted During CY 1983	S404	D. Zukor
09/84	Semiannual Report on AEOD Activities	S405	J. Heltemes
10/84	Application of Risk Perspectives: A Procedures Guide	S406	P. Lam

Engineering Evaluations and Technical Reviews

Date	Title	No.	Author
01/84	Temporary Loss of All AC Power Due to Relay Failure in Diesel Generator Load Shedding Circuitry at Fort St. Vrain	E401	M. Chiramal

Table D-8 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
01/84	Water Hammer in Boiling Water Reactor High-Pressure Coolant Injection Systems	E402	S. Rubin
01/84	Deficiency in Automatic Switch Company (ASCO) Spare Parts Kits for Scram Pilot Solenoid Valves	E403	F. Ashe
02/84	Failures in the Upper Head Injection System	E404	D. Zukor
03/84	Common Mode Failure of HPCI Steam Flow Isolation Capability at Browns Ferry	E405	M. El-Zeftawy
03/84	Mechanical Snubber Failure	E406	C. Hsu
03/84	Initiation and Indication Circuitry for High Pressure Coolant Injection Systems	E407	F. Ashe
03/84	Load Reduction Transient at the Salem Unit 2 on January 14, 1982	E323 Revision 1	N. Trehan
04/84	Reversed Differential Pressure Instrument Sensing Lines	E408	S. Rubin
05/84	Operating Experience Involving Air in Instrument Sensing Lines	E409	S. Salah
05/84	Operational Experiences Involving Standby Gas Treatment Systems That Illustrate Potential Common Cause Failure or Degradation Mechanisms	E410	F. Ashe
05/84	Failure of Anti-Cavitation Device in Residual Heat Removal Service Water Heat Exchanger Outlet Valve	E411	C. Hsu
05/84	Adverse System Interaction With Domestic Water Systems	E412	T. Cintula
05/84	Natural Circulation in Pressurized Water Reactors	E413	W. Lanning
05/84	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	E414	P. Lam
06/84	Overcooling Transient	E415	E. Imbro
06/84	Erosion in Nuclear Power Plants	E416	E. Brown
07/84	Loosening of Flange Bolts on Residual Heat Removal Heat Exchanger Leading to Primary to Secondary Side Leakage	E417	C. Hsu
07/84	Feedwater Transients During Startup at Westinghouse Plants	E418	D. Zukor
07/84	Failures of Fischer-Porter Transmitters Used in Safety Related Systems	E419	M. Chiramal
08/84	Operational Experiences Involving Shorted Lamp Sockets of Indication Lights	E420	M. Chiramal

Table D-8 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
08/84	Loss of Pressurizer Heaters During Precore Hot Functional Testing	E421	T. Cintula
08/84	High Pressure Coolant Injection System Performance at Hatch Units 1 and 2	E422	T. Wolf
09/84	Failure of Large Hydraulic Snubbers To Lock Up	E423	E. Brown
10/84	Failure of Anchor Bolt on Diesel Generator Day Tank at Davis-Besse Unit	E424	C. Hsu
10/84	High Pressure Coolant Injection System Lockout at Vermont Yankee	E425	M. Chiramal
10/84	Single Failure Vulnerability of Power-Operated Relief Valve Actuation Circuitry for Low Temperature Overpressure Protection	E426	E. Imbro
11/84	Licensee Event Reports That Address Situations That Potentially Could Result in Overloading Electrical Equipment in the Emergency Power System or Prevent Operation of the Onsite Power System Sequencer	E427	F. Ashe
03/84	Failures of Containment Air Monitors at Farley Units 1 and 2	T401	D. Zukor
03/84	Chemical Contamination of Primary and Secondary Systems in Light Water Reactors	T402	M. El-Zeftawy
03/84	Setpoint Drift of Barton Model 288 Switches	T403	M. Chiramal
04/84	Cable Fire and Loss of Control Power to Engineered Safeguards Valves	T404	M. Chiramal
04/84	Cold Weather Events 1983–1984	T405	T. Cintula
04/84	Improper Spare Parts Procurement Event at Grand Gulf Unit	T406	T. Wolf
04/84	Failure of a 4 kV Circuit Breaker To Trip	T407	M. Chiramal
05/84	Diesel Generator Inoperability Due to Overheating of Ventilation Cowling	T408	M. Chiramal
05/84	Multiple Failure of Bell and Howell Dual Potentiometer Modules That Occurred at the Fort Calhoun Nuclear Station	T409	F. Ashe
05/84	Failure of Injection Valve for the High Pressure Coolant Injection System To Open During a Surveillance Test	T410	E. Brown
06/84	Contamination of the Nitrogen System at Sacramento Municipal Utility District	T411	M. El-Zeftawy
06/84	Failure of an Access Door Between the Drywell and the Wetwell	T412	T. Wolf

Table D-8 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
06/84	Failure of Fire Damper in Safeguards Ventilation System	T413	W. Lanning
07/84	Station Operating Restrictions for Loss or Out-of-Service Power Transformers Through Which Electrical Power Is Supplied to the Emergency Buses	T414	F. Ashe
07/84	Destruction of Charging Pump	T415	W. Lanning
08/84	Loss of Engineered Safety Feature Auxiliary Feedwater Pump Capability at Trojan on January 22, 1983	T416	D. Zukor
08/84	Excessive Cooldown Rate Event at LaSalle Unit 1	T417	S. Salah
08/84	Events Involving Fires or Other Related Abnormalities in Motor Control Centers With Aluminum Bus Bars	T418	F. Ashe
08/84	Contamination of Snubber Bleed Screw and Lockup Poppet Valve	T419	C. Hsu
08/84	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System To Open Against Operating Reactor Pressure	T420	P. Lam
08/84	Design Deficiency in Standby Gas Treatment System	T421	M. Chiramal
08/84	Inoperability of Safety Injection Pump at Salem Unit 1 on October 17, 1983	T422	D. Zukor
10/84	Inoperability of Helium Circulator Overspeed Trip Channels Due to Impedance Variations in Speed Sensing Cables Exposed to Steam Leak	T423	E. Imbro
11/84	Fire Water Main Leakage Into 4 kV Switchgear Room at San Onofre Unit 1	T424	T. Cintula

Table D–9

Reports Issued in CY 1983

Case and Special Studies			
Date	Title	No.	Author
09/83	Licensee Event Report System, Description of System and Guidelines for Reporting	NUREG–1022	
09/83	Potentially Damaging Failure Modes of High and Medium Voltage Electrical Equipment	NUREG/ CR–3122	M.Chiramal
04/83	Failures of Class 1E Safety-Related Switchgear Circuit Breakers To Close on Demand	C301	M. Chiramal
07/83	Report on the Implications of the Anticipated Transient Without Scram Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience	P301	J. Crooks

Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
01/83	Fuel Degradation at Westinghouse Plants	E301	D. Zukor
04/83	Update to AEOD/E301 (Fuel Degradation at Plants)	E301 Revision 1	D. Zukor
01/83	Potential Loss of Service Water Flow Resulting From a Loss of Instrument Air	E302	E. Imbro
02/83	Valve Flooding Event at Surry	E303	D. Zukor
03/83	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems To Prevent Flooding of Vital Equipment in Safety-Related Compartments	E304	T. Cintula
04/83	Inoperable Motor-Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	E305	E. Brown F. Ashe
04/83	Cooldown During Loss of Control Room Test at McGuire Unit 1	E306	D. Zukor
04/83	Degradation of Safety-Related Batteries Due to Cracking of Battery Cell Cases and/or Other Possible Aging-Related Mechanisms	E307	F. Ashe
04/83	Cracks and Leaks in Small-Diameter Piping	E308	E. Brown
04/83	The Potential for Water Hammer During the Restart Residual Heat Removal Pumps at BWR Nuclear Power Plants	E309	S. Rubin
04/83	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E310	T. Cintula
04/83	Loss of Salt Water Flow to the Service Water Heat Exchangers for 23 Minutes at Calvert Cliffs Unit 2	E311	T. Cintula

Table D-9 (cont.)

Date	Engineering Evaluations and Technical Reviews (cont.) Title	No.	Author
05/83	Operability of Target Rock Safety Relief Valves in the Safety Mode With Pilot Valve Leakage	E312	J. Pellet
06/83	Potential Contamination of the Spent Fuel Pool and Primary Reactor System	E313	E. Brown
06/83	Loss of All Three Charging Pumps Due to Empty Common Reference Leg in the Liquid Level Transducers for the Volume Control Tank at St. Lucie 1	E314	T. Cintula
07/83	Misuse of Valve Resulting in Vibration and Damage to the Valve Assembly and Pipe Supports	E315	E. Brown
07/83	Frozen Ice Condenser Intermediate Deck Doors	E316	D. Zukor
08/83	Loss of High Pressure Injection System	E317	N. Trehan
08/83	Biofouling at Salem Units 1 and 2	E318	E. Imbro
09/83	Loss of Drywell Torus Pressure Differential During Residual Heat Removal Pump Flow Testing at Cooper	E319	S. Rubin
09/83	Power-Operated Relief Valve Actuation Resulting in Safety Injection Actuation at Calvert Cliffs	E320	E. Imbro
09/83	Three Similar Events of a Loss of Shutdown Cooling Flow at Combustion Engineering Plants	E321	T. Cintula
09/83	Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting at Peach Bottom Unit 2	E322	C. Hsu
09/83	Load Reduction Transient at Salem Unit 2 on January 14, 1982	E323	N. Trehan
09/83	Review of Events Involving Failures of Power Supply in Instrumentation and Control Systems	E324	M. Chiramal
11/83	Vapor Binding of Auxiliary Feedwater Pumps at Robinson Unit 2	E325	W. Lanning
11/83	Steam Voiding in Oconee Unit 3 on June 13, 1975: A Precursor Event to the TMI-2 Accident	E326	H. Ornstein
11/83	Gaseous Releases From Waste Gas Disposal System	E327	N. Trehan
11/83	Human Factors Involvement in Events at Oconee Units 1, 2, and 3	NT304	K. Black
08/83	Human Factors Contributions to Accident Sequence Precursor Events	N305	E. Trager
01/83	Diesel Generator Load Sequencer Design Deficiency—LER 82-025/OIT	T301	M. Chiramal
02/83	Postulated Loss of Auxiliary Feedwater System Resulting From a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	T302	E. Imbro

Table D-9 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	**Title**	**No.**	**Author**
03/83	Seat Degradation in Henry Pratt Butterfly Valves	T303	E. Brown
03/83	Cause of Containment Isolation Valve F042A To Close	T304	S. Salah
03/83	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Intrusion at Sequoyah Unit 1	T305	E. Imbro
04/83	Scram Discharge Volume Level Switch Failure at Hatch Unit 2	T306	J. Pellet
04/83	Condensate Demineralizer Resin Migration Through the Plant Vent and the Standby Gas Treatment System at Pilgrim Unit 1	E307	J. Pellet
04/83	Undetectable Failure in Westinghouse Solid State Protection System	T308	M. Chiramal
04/83	Air in Reactor Water Cleanup System Instrument Sensing Lines at Brunswick Unit 2	T309	S. Salah
04/83	Blocking of Automatic Safety Injection Signals	T310	M. Chiramal
05/83	Rod Control Urgent Failure on June 25, 1982, at Surry Unit 2	T311	N. Trehan
05/83	Failure of 5 kV Cable Terminations	T312	M. Chiramal
05/83	Capped Containment Pressure Sensing Lines	T313	S. Rubin
05/83	Improper Size of Inlet Piping to Primary Safety Valves	T314	E. Imbro
05/83	Events Involving Losses of or Perturbations in a Single 120 Volt AC Vital Power Supply Inverter and Attendant Distribution Bus Which Resulted in Inadvertent Actuations of Safety Systems	T315	F. Ashe
05/83	Thermal Non-Repeatability Problem With Barton Models 763 and 764 Electronic Transmitters	[illegible]6	M. Chiramal
06/83	Problems With Diesel Driven Containment Spray P[illegible] at Zion Unit 2 on December 16, 1982	[illegible]317	D. Zukor
06/83	Failure of Recirculation Spray Service Water Motor-Operated Valves	T318	D. Zukor
06/83	Design Deficiency in Control Circuits of Feedwater Isolation Valves and Boron Injection Tank Recirculation Valves	T319	M. Chiramal
06/83	Inadvertent Safety Injections Attributed to Personnel Error at Summer	T320	F. Ashe
06/83	Check Valve Installed Backwards in Instrument Air Line to the Power-Operated Relief Valve at Surry Unit 2	T321	D. Zukor
06/83	Gouges in Main Coolant System Piping at Diablo Canyon on April 19, 1983	T322	D. Zukor

Table D-9 (cont.)

Date	Title	No.	Author
	Engineering Evaluations and Technical Reviews (cont.)		
06/83	Turbine Trip Bypass Delay at Grand Gulf Unit 1	T323	S. Salah
07/83	Events Involving Two or More Simultaneously Dropped Rod Control Cluster Assemblies	T324	F. Ashe
08/83	Leakage in Static-O-Ring Pressure Switches	T325	M. Chiramal
08/83	Safety Relief Valve Corrosion at a Foreign Reactor	T326	E. Brown
08/83	Auxiliary Feedwater Header Problems at Babcock & Wilcox Plants	T327	H. Ornstein
08/83	Two of Three Emergency Core Cooling System Accumulators Inoperable at Surry Unit 1	T328	D. Zukor
08/83	Leak in Reactor Water Cleanup System "B" Regenerative Heat Exchanger Relief Line	T329	C. Hsu
08/83	Steam Generator Tube Rupture at Oconee Unit 2	T330	M. El-Zeftawy
08/83	Review of Events at Operating Nuclear Plants Involving Plant Computers	T331	M. Chiramal
10/83	Reactor Vessel Drainage	T332	S. Salah
10/83	Degradation of Saltwater Cooling System at San Onofre Unit 1 Due to a Loss of Instrument Air	T333	H. Ornstein
11/83	Reactor Vessel Drainage at Grand Gulf Unit 1	T334	S. Salah
11/83	Simultaneous Safety Injection Actuation Signal and Recirculation Actuation Signal at San Onofre Unit 3	T335	T. Cintula
11/83	Design Deficiency Resulting in Isolation of Both Loops of the Emergency Condenser System at Nine Mile Point Unit 1	T336	M. Chiramal
11/83	Water Hammer in the Main Feedwater System Resulting in a Feedwater Line Crack at Maine Yankee	T337	E. Imbro
11/83	Water Leak Through Containment Spray Block Valves at San Onofre 1	T338	D. Zukor
11/83	Redundant Emergency Core Cooling System Pump Room Air Coolers Out of Service for 22 Hours at Calvert Cliffs Unit 1	T339	T. Cintula
12/83	Evaluation of Control Rod Mismanipulation Event at Hatch Unit 2	T340	T. Wolf
12/83	Corrosion of Carbon Steel Pipe in Service Water Headers	T341	E. Brown

Table D-10

Reports Issued in CY 1982

Case Studies

Date	Title	No.	Author
01/82	Safety Concern Associated With Reactor Vessel Level Instrumentation in Boiling Water Reactors	C201	M. Chiramal
02/82	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	C202	E. Imbro
05/82	Survey of Valve Operator Related Events Occurring During 1978, 1979 and 1980	C203	E. Brown
07/82	San Onofre Unit 1 Loss of Salt Water Cooling Event on March 10, 1980	C204	H. Ornstein
08/82	Abnormal Transient Operating Guidelines as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	C205	J. Pellet
10/82	Inadvertent Loss of Reactor Coolant Events at the Sequoyah Nuclear Plant, Units 1 and 2	C206	W. Lanning

Engineering Evaluations

Date	Title	No.	Author
01/82	Methodology for Vital Area Determination	E201	W. Lanning
01/82	Loss of High Pressure Injection Lube Oil Cooling at Rancho Seco	E202	J. Pellet
01/82	Inadvertent Isolation of Containment Fan Units at Salem Unit 1	E203	W. Lanning
01/82	Effects of Fire Protection System Actuation on Safety Related Equipment	E204	M. Chiramal
02/82	Potential Consequences of Heavy Load Drop Accidents in LWRs	E20[illegible]	M. El-Zeftawy
02/82	Load Reduction Transient on January 14, 1982, at Salem Unit 2	E206	N. Trehan
02/82	LER 50-336/81-26: Investigation of the Relative Frequency of Valve Overtravel Anomalies That Could Result in a Potential Centrifugal Pump Runout Exceeding Net Positive Suction Head	E207	E. Imbro
02/82	An Observed Difference in Lift Setpoint for Steam Generator and Pressurizer Safety Valves	E208	T. Cintula
02/82	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback Unit 1 on April 13, 1979)	E209	M. Chiramal
02/82	Inadequate Switchgear Cooling at Beaver Valley Unit 1	E210	W. Lanning

Table D-10 (cont.)

Engineering Evaluations (cont.)			
Date	**Title**	**No.**	**Author**
02/82	Repetitive Failures of Emergency Feedwater Flow Valves at Arkansas Unit 2 Because of Valve Operator Hydraulic Problems	E211	T. Cintula
02/82	Spurious Trip of the Generator Lockout Relay Associated With a Diesel Generator Unit	E212	F. Ashe
02/82	Trip of Two Inservice Auxiliary Feedwater Pumps From Low Suction at Zion Unit 2 on December 11, 1981	E213	D. Zukor
03/82	Duane Arnold Loss of River Water System Loop	E214	T. Wolf
03/82	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	E215	E. Imbro
03/82	A Recently Evaluated Preoperational Test Precursor of the TMI-2 Accident	E216	H. Ornstein
03/82	Scram Pilot Solenoid Valve Failures Due to Low Voltage—Grand Gulf Unit 1	E217	M. Chiramal
03/82	Potential for Air Binding or Degraded Performance of BWR Residual Heat Removal System Pumps During the Recirculation Phase of a Loss-of-Coolant Accident	E218	S. Rubin
04/82	Proposed Circular: Contamination of Air Serving Safety Related Equipment	E219	H. Ornstein
04/82	Water in the Fuel Oil Tank at Surry Power Station Unit 2	E220	N. Trehan
04/82	Indian Point Unit 2 Flooding Event	E221	W. Lanning
05/82	Loss of Reserve Station Service Transformer "B" on January 18, 1982, at Surry Unit 2	E222	N. Trehan
05/82	Inadvertent Loss-of-Coolant Events at Sequoyah Units 1 and 2	E223	W. Lanning
05/82	Generic Concerns Associated With the Ginna Steam Generator Tube Rupture Event	E224	W. Lanning
06/82	Degradation of BWR Scram Pilot Solenoid Valves Due to Abnormal Power Supply Voltage	E225	M. Chiramal
06/82	Inoperability of Instrumentation Due to Extreme Cold Weather	E226	M. Chiramal
06/82	Failure of Engineered Safety Features Manual Initiation Pushbutton Switches	E227	F. Ashe
06/82	Repetitive Overspeed Trips of the Steam Driven Emergency Feedwater Pump on Initial Start at Arkansas Nuclear One, Unit 2	E228	E. Imbro

Table D-10 (cont.)

Date	Engineering Evaluations (cont.) Title	No.	Author
06/82	Potential for Flooding in Control Room at San Onofre Units 2 and 3	E229	T. Cintula
07/82	Water in the Fuel Oil Tank at Surry Power Station, Unit 2—Additional Information	E230	N. Trehan
07/82	Millstone Unit 2 Loss of Shutdown Cooling Due to Trip of Low Pressure Safety Injection Pump	E231	M. Chiramal
07/82	Potential Deficiency in the Sigma Lumigraph Indicators Model Number 9270	E232	F. Ashe
07/82	Carbon Dioxide Systems Used for Fire Protection in or Adjacent to Critical Areas	E233	M. Chiramal
08/82	Failure in a Section of 4 kV Bus Cable Manufactured by Okonite	E234	F. Ashe
08/82	Wiring Error in Handswitch for Solenoid Control Valves Associated With High-Pressure Coolant Injection System Steam Condensing Mode Pressure Control Valve at Duane Arnold	E235	S. Rubin
08/82	Brunswick Steam Electric Plant Unit 2 Loss of Residual Heat Removal Service Water on January 16, 1982	E236	T. Wolf
08/82	Power-Operated Relief Valve Failure at Robinson	E237	E. Brown
08/82	Water in the Lube Oil in Safety Injection Pump IA-A at Sequoyah—LER 81-076	E238	N. Trehan
09/82	Main Steam Isolation Valve Closures and Pressurizer Safety Valve Actuations at St. Lucie Unit 1 on December 19, 1981	E239	T. Cintula
09/82	Preliminary Account of Events Associated With a Reactor Trip at Hatch Unit 2 on August 25, 1982	E240	S. Rubin
10/82	Emergency Diesel Generator System Problems at FitzPatrick	E241	M. Chiramal
10/82	Fuel Assembly Degradation While in the Spent Fuel Storage Pool	E242	E. Brown
10/82	Plant Trip Followed by a Safety Injection Due to Loss of "A" Cooling Tower Pump at Palisades on February 4, 1982	E243	T. Cintula
10/82	Loss of Residual Heat Removal System Event at Pilgrim Nuclear Power Station on December 21, 1981	E244	T. Wolf
10/82	Failure of Westinghouse Type SC-1 No. 1876-072 Relays	E245	F. Ashe
10/82	Events Involving Loss of Electrical Inverters Including Attendant Inverters to Vital Instrument Buses	E246	F. Ashe

Table D-10 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
10/82	Engineering Evaluation of Turbine/Reactor Trip at Rancho Seco on August 7, 1981	E247	J. Pellet
11/82	Engineering Evaluation Report on McGuire Overpressurization Event of August 27, 1981	E248	D. Zukor
11/82	Engineering Evaluation Memorandum—Licensee Reporting of the Turbine/Reactor Trip at Rancho Seco on August 7, 1981	E249	H. Ornstein
11/82	Quad Cities Unit 2 Loss of Auxiliary Electrical Power Event on June 22, 1982	E250	M. Chiramal
11/82	Salem Unit 2 Loss of Vital Bus No. 2A	E251	M. Chiramal
11/82	Potential Control Logic Problem Resulting in Inoperable Auto-Start of Diesel Generator Units Under the Conditions of Loss-of-Coolant Accident and Loss of Station Power (LOSP)	E253	F. Ashe
11/82	Review of Prairie Island Unit 1 LER 82–015–O1T on Diesel Generator Operability	E254	M. Chiramal
11/82	Failure of the Vent Line on the Common Discharge of the Two Motor-Driven Auxiliary Feedwater Pumps at San Onofre Unit 2 From an Improper Valve Lineup	E255	T. Cintula
11/82	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E256	T. Cintula
12/82	Insufficient Net Positive Suction Head for Charging Pump Service Water Pumps at Surry Nuclear Power Station	E257	D. Zukor

Table D-11

Reports Issued in CY 1981

Case Studies

Date	Title	No.	Author
03/81	Report on the St. Lucie Unit 1 Natural Circulation Cooldown on June 11, 1980	C101	E. Imbro
03/81	Robinson Reactor Coolant System Leak on January 29, 1981	C102	W. Lanning
03/81	AEOD Safety Concerns Associated With Pipe Breaks in the BWR Scram System	C103	S. Rubin
04/81	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	C104	M. Chiramal
12/81	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	C105	E. Imbro

Engineering Evaluations

Date	Title	No.	Author
01/81	Degradation of Internal Appurtenances in LWR Piping	E101	E. Brown
01/81	Sequoyah Unit 1 Loss of Annunciation	E102	M. Chiramal
02/81	Davis-Besse Nuclear Power Station, Unit 1—Engineered Safety Features Actuation System (ESFAS)	E103	M. Chiramal
03/81	Engineering Evaluation of Feedwater Transient and System Pipe Break at Turkey Point 3	E104	S. Sands
03/81	Water Hammer During Restart of Residual Heat Removal Pumps	E105	J. Huang
03/81	Water Hammer in the Steam Condensing Mode of the Residual Heat Removal System Operation	E106	J. Huang
04/81	Peach Bottom Unit 3 Occurrence on February 25, 1981	E107	F. Ashe
04/81	Hatch Units 1 and 2—Alternate Offsite Source Interlock With Emergency Diesel Generators	E108	M. Chiramal
04/81	Potential Common-Mode Failure of Diesel Generators	E109	M. Chiramal
04/81	Requirements of the Preferred or Offsite Power System	E110	F. Ashe
05/81	Evaluation of High Pressure Safety Injection Pump Operability Without Service Water	E111	E. Imbro
06/81	Inoperability of Instrumentation Due to Extreme Cold Weather	E112	M. Chiramal
06/81	Deliberate Pump Trip at Browns Ferry Unit 2 on April 6, 1981	E113	W. Lanning

Table D-11 (cont.)

Engineering Evaluations (cont.)

Date	Title	No.	Author
06/81	Control System Failures That Could Cause or Exacerbate Nuclear Power Plant Accidents	E114	F. Ashe
07/81	Additional Information on Events at TMI-2 During Preoperational Testing (September 5-12, 1977)	E115	H. Ornstein
07/81	Failure of B Phase Main Transformer and Subsequent Fire in the Transformer Area—North Anna Unit 2	E116	M. Chiramal
07/81	Events at TMI-2 During Preoperation Testing	E117	H. Ornstein
07/81	Setpoint Drift Occurrences for the Barton Model 288 Instrument	E118	F. Ashe
07/81	Loss of Residual Heat Removal Capability at Brunswick Units 1 and 2	E119	E. Imbro
08/81	Ignition of Gaseous Waste Decay Tank at San Onofre Unit 1—July 17, 1981	E120	H. Ornstein
08/81	Crystal River 3 Engineered Safeguards Relay Failures	E121	M. Chiramal
09/81	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of Integrated Control System Nonnuclear/ Instrumentation	E122	H. Ornstein
09/81	Immediate Action Memo: Common Cause Failure Potential at Rancho Seco—Desiccant Contamination of Air Lines	E123	H. Ornstein
09/81	Review of Information on Purge Valves	E124	E. Brown
10/81	Engineering Evaluation Report on Shutdown Cooling System Heat Exchanger Failures at Oyster Creek, August 1981	E125	G. Lanik
10/81	Event Sequences Not Considered in the Design of Emergency Bus Control Logic	E126	F. Ashe
10/81	Pressure Boundary Degradation Due to Pump Seal Failure at Arkansas Nuclear One	E127	W. Lanning
11/81	Inoperable Teledyne Solenoid Valves	E128	F. Ashe
12/81	Brunswick Unit 2 Diesel Generator Jacket Water Temperature Control Valve and Manual Bypass Valve	E129	M. Chiramal
12/81	Davis Besse LER 79-062 on Auxiliary Feedwater System Pressure Switches	E130	M. Chiramal
12/81	High Circulating Current Associated With Inverter Output Due to Lack of Circuit Tuning	E131	F. Ashe
12/81	Abnormal Wear Encountered on Aloyco Swing Check Valves Installed in the Low Pressure Safety Injection System at Palisades	E132	T. Cintula

Table D-11 (cont.)

Date	Title	No.	Author
	Engineering Evaluations (cont.)		
04/81	Inadequacies in Periodic Testing of Combustion Engineering PWR Reactor Protection System	E133	M. Chiramal

Table D–12

Reports Issued in CY 1980

Case Studies

Date	Title	No.	Author
07/80	Report on the Browns Ferry Unit 3 Partial Failure To Scram Event on June 28, 1980	C001	S. Rubin
09/80	Report on the Interim Equipment and Procedures at Browns Ferry Unit 3 To Detect Water in the Scram Discharge Volume	C002	G. Lanik
10/80	Report on Loss-of-Offsite-Power Event at Arkansas Nuclear One, Units 1 and 2	C003	W. Lanning
11/80	AEOD Actions Concerning the Crystal River Unit 3 Loss of Nonnuclear Instrumentation and Integrated Control System Power on February 26, 1980	C004	H. Ornstein
12/80	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	C005	E. Imbro

Engineering Evaluations

Date	Title	No.	Author
03/80	Crystal River Nuclear Power Plant Decay Heat Closed Cycle Cooling Water Pumps/DCP–1A and DCP–1B	E001	H. Ornstein
05/80	BWR Jet Pump Integrity	E002	S. Rubin
06/80	Comparison of Reactor Coolant Pump Events Contained in LERs, NPRDS, RECON, and Plant Records	E003	E. Brown
07/80	Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants, January 1, 1972 to April 30, 1978	E004	H. Ornstein
07/80	Operational Restrictions for Class 1E 120V AC Vital Instrument Buses	E005	M. Chiramal
08/80	Loss of Residual Heat Removal at Beaver Valley, LER 80–031	E006	W. Lanning
08/80	Potential for Unacceptable Interaction Between the Control Rod Drive System and Nonessential Control Air System at Browns Ferry	E007	S. Rubin
08/80	Operational Restrictions During Surveillance Testing of Emergency Diesel Generators	E008	M. Chiramal
08/80	Failures of Containment Isolation Valves at Zion	E009	W. Lanning
08/80	Tie Breaker Between Redundant Class 1E Buses—Point Beach Units 1 and 2	E010	M. Chiramal

Table D-12 (cont.)

Engineering Evaluations (cont.)

Date	Title	No.	Author
08/80	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	E011	E. Imbro
09/80	Salem Unit 1—Solenoid Valve of Containment Fan Coil Unit Service Water Flow Control Valve	E012	M. Chiramal
09/80	Excessive Main Feedwater Transient	E013	J. Creswell
10/80	Transient at Crystal River Unit 3—September 30, 1980	E014	H. Ornstein
10/80	January 3, 1977, Quad Cities Unit 1 Loss-of-Air Event and Its Effects on Scram Capability	E015	G. Lanik
10/80	Flow Blockage in Essential Equipment at ANO Caused by *Corbicula* sp. (Asiatic Clams)	E016	E. Imbro
10/80	Engineering Evaluation of Steam Generator Overfill	E017	W. Lanning
12/80	Potential Failure of BWR Backup Scram (Mode Switch in Shutdown) Capability	E018	M. Chiramal
12/80	Davis Besse Unit 1—Emergency Core Cooling System Actuation During Hot Shutdown on December 5, 1980	E019	M. Chiramal
12/80	Internal Appurtenances in LWRs	E020	E. Brown

Appendix E

Status of AEOD Recommendations

Appendix E

Status of AEOD Recommendations

This section summarizes the year-end status of all AEOD recommendations that are either new or still outstanding since the last report. At the beginning of 1992, 15 AEOD recommendations were outstanding. During 1992, one recommendation was resolved, and two new recommendations were added. Therefore, as of December 31, 1992, 16 AEOD recommendations were outstanding. Table E-1 presents the status of AEOD recommendations at the end of 1992.

AEOD's tracking system ensures that all formal AEOD recommendations are tracked until they are resolved. At this time, no outstanding issues involving AEOD recommendations warrant the attention of NRC's Executive Director for Operations.

The majority of current issues have been assigned a high priority, and many are included in NRC's generic issues program.

In addition to implementing the formal recommendations that are tracked and listed in this appendix, NRC program offices routinely implement additional actions that are based on AEOD suggestions included in engineering evaluations and other reports. AEOD does not formally track or close out suggestions.

Information about each recommendation that is outstanding or that was resolved during 1992, including a description and status for each, follows.

Table E-1
Status of AEOD Recommendations – 1992

Status of AEOD Recommendations	12/31/92
Added since previous report	2
Resolved	5
Currently outstanding	12
Included in unresolved safety issues, generic issues, and Three Mile Island Action Plan items	4
Addressed by NRC and industry	8
Office of Nuclear Regulatory Research prioritization of issues	
High priority	10
Medium priority	1
Low priority	0
No priority	1
Functional areas addressed by issues	
Procedures, training, etc.	3
Hardware modification	6
Equipment testing	2
Other	1

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C101

Responsible AEOD Engineer: E. Trager

Title or Subject: "St. Lucie Natural Circulation Cooldown"

Recommendation 1: Provide a supply of cooling water to reactor coolant pump (RCP) seals that will not be disabled by a single failure (study Recommendation 4e).

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	K. Shaukat	High

Status: Active. This recommendation was incorporated into task 2 of Generic Issue (GI) 23, "Reactor Coolant Pump Seal Failures." During 1988 and 1989, the Office of Nuclear Regulatory Research (RES) completed the technical findings and cost benefit analyses for the proposed resolution of GI 23. During 1990, the RES staff coordinated the proposed resolution package with the Office of Nuclear Reactor Regulation (NRR), AEOD, and the Office of the General Counsel (OGC). The Committee To Review Generic Requirements (CRGR) reviewed this package and recommended modifying it to encourage industry comments to facilitate a decision on final resolution. This package was issued for public comment, along with specific questions for possible alternative solutions, in April 1991. Public comments were received by November 1991. The Executive Director for Operations approved initiating rulemaking in October 1992. Final resolution of GI 23 is scheduled for December 1994.

Recommendation Source: Case Study AEOD/C501 (NUREG/CR-3551)

Responsible AEOD Engineer: H. Ornstein

Title or Subject: "Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants"

Recommendation 2: Require protection to prevent hydrogen explosions or fires in areas containing, or impacting operation of, safety-related equipment.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	C. Graves	Medium

Status: Active. GI 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," was originally given a low priority. The priority of this issue was reevaluated, and in December 1987 the issue was assigned a medium priority.

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RES issued a contract to Idaho National Engineering Laboratory in July 1988 to address large hydrogen leaks in pressurized water reactors (PWRs) and to quantify the risk from hydrogen leaks. Indian Point Unit 2 was selected as a reference plant for GI 106, and the probabilistic risk assessment team visited the plant for 2 weeks to assess the risk from hydrogen. Other PWRs were examined for similarity to Indian Point Unit 2. A scoping analysis was also performed for risks from hydrogen at boiling water reactors (BWRs) and hydrogen storage facilities.

RES recommended issuance of a generic letter recommending installation of excess flow check valves and/or use of guarded piping. CRGR disapproved the recommendation. Hydrogen concerns are being reconsidered by NRR and RES due to additional insight gained from the INEL study and recent foreign data. A generic communication is being prepared to alert licensees to the findings of GI 106 and the recent foreign data. The hydrogen issue is to be addressed by GI 167, "Hydrogen Storage Facilities."

Recommendation Source: Case Study AEOD/C502

Responsible AEOD Engineer: W. Jones

Title or Subject: "Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors"

Recommendation 1: Disable the non-safety-related air operator associated with the testable isolation check valve on the injection line in the emergency core cooling systems.

Recommendation 2: Perform leakage testing of the testable isolation check valve before plant startup after each refueling outage or following maintenance, repair, or replacement of the valve.

Recommendation 3: Reduce human errors in maintenance and surveillance testing activities.

Recommendation 4: Study how to reduce the frequency of surveillance testing of the isolation barriers of the emergency core cooling systems during power operation.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/RPSIB	G. Burdick	High

Status: Resolved. The NRC has published three reports about this issue: NUREG-5604, "Assessment of ISLOCA Risk-Methodology and Application to a Babcock and Wilcox Nuclear Power Plant"; NUREG/CR-3744, "Assessment of ISLOCA Risk-Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant"; and NUREG/CR-5745, "Assessment of ISLOCA Risk-Methodology and Application to a Combustion Engineering Plant." Based on the resolution of GI 105, "Interfacing System LOCA at LWRs," on June 21, 1993, these recommendations are resolved.,

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Recommendation Source: Case Study AEOD/C701 (NUREG–1275, Vol. 2)

Responsible AEOD Engineer: H. Ornstein

Title or Subject: "Air Systems Problems at U.S. Light Water Reactors"

Recommendation 5: All operating plants should be required to perform gradual loss of instrument air system pressure tests.

Responsible Office/Div/Br	Contact	Priority
NRR/DEST/SPLB	W. LeFave	High

Status: Active. Information Notice 87–28, "Air System Problems at U.S. Light Water Reactors," and Supplement 1 thereto have been issued. NRR issued Generic Letter (GL) 88–14, "Instrument Air Supply Problems Affecting Safety-Related Equipment." GL 88–14 requires licensees to carry out Recommendations 1 through 4 of this study (now resolved), whereas Recommendation 5 was not addressed. Some plants have performed limited gradual loss of instrument air tests. AEOD is working with industry (American Society of Mechanical Engineers) to formulate a performance standard to ensure that licensees achieve reliable performance of their plant instrument air systems. Resolution is expected by the end of 1993.,

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Recommendation Source: 1. Memorandum, C. Michelson to H. R. Denton, dated July 15, 1980
2. Memorandum, C. Michelson to H. R. Denton, dated August 27, 1980

Responsible AEOD Engineer: W. Raughley

Title or Subject: "Operational Restrictions for Class 1E 120 V ac Vital Instrument Buses"

"Tie Breaker Between Redundant Class 1E Buses—Point Beach Nuclear Plant Units 1 and 2"

Recommendation 1: Interconnection between redundant safety-related electrical load groups should comply with requirements of Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems."

Responsible Office/Div/Br	Contact	Priority
NRR/DRP/PD2–1	S. Hoffman	High
NRR/DRP/PD1–3	A. Johnson	High

Status: Active. This recommendation was initially pursued under GI 128, "Electrical Power Reliability." GI 128 was reviewed and approved by the CRGR in July

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1989, but the ACRS did not agree to the proposed resolution. This recommendation and a number of generic safety issues directly related to onsite electrical power systems are near resolution (NUREG–0933, "A Prioritization of Generic Safety Issues"). These issues include GI A–30, "Adequacy of Safety-Related DC Power Supplies"; GI 48, "LCOs for Class 1E Vital Instrument Bus in Operating Reactors"; and GI 49, "Interlocks and LCOs for Redundant Class 1E Tie Breaker." GL 91–06, "Resolution of Generic Issue A–30, 'Adequacy of Safety-Related DC Power Supplies,' " was issued in April 1991, and GL 91–11, "Resolution of Generic Issues 48, 'LCOs for Class 1E Vital Instrument Buses,' and 49, 'Interlocks and LCOs for Class 1E Tie Breakers,' " was issued in July 1991. Licensees had 180 days to respond. NRR has evaluated the licensee responses to GL 91–06 for GI A–30 and has issued closeout letters to most plants (i.e., less than 10 percent are not closed). Responses to GL 91–11 are under review.

Recommendation Source: Special Study Report AEOD/S803

Responsible AEOD Engineer: J. Kauffman

Title or Subject: "AEOD Concerns Regarding the Power Oscillation Event at LaSalle (BWR-5)"

Recommendation 2: Analysis of thermohydraulic/neutronic oscillations should be undertaken to verify the adequacy of the resolution of GIs B–19, "Thermal-Hydraulic Stability," and B–59, "(N–1) Loop Operation in BWRs and PWRs," and anticipated transient without scram (ATWS) mitigation.

Responsible Office/Div/Br	Contact	Priority
NRR/DEST/SRXB	L. Phillips	N/A

Status: Active. The NRR staff has reviewed the report NEDO–31960, "Long Term Stability Solutions Licensing Methodology," and its Supplement 1 submitted by the BWR Owners Group (BWROG) in 1991 and 1992, describing long-term solutions (LTSs) to exclude or detect and suppress power oscillations, and has sent a safety evaluation to CRGR for review, expected in April 1993. The staff approves concept options I–A, II, III, and III–A, the methodologies used to develop exclusion regions and analyze option II and the algorithms of options III and III–A. Option I–D is still under review and options I–B and I–C were not developed by BWROG. Initial discussions with BWROG on hardware and software design for the options have begun. Full installation in plants is expected to extend to 1995–1997.

Accompanying the safety evaluation report is a generic letter requesting licensees to (1) select and implement LTSs and (2) augment the current interim actions of Bulletin 88–07, Supplement 1, "Power Oscillations in Boiling Water

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Reactors (BWRs)." The augmented areas include training, procedures, monitoring, and power distribution control. The augmentation request evolved, in part, from an instability event at Washington Nuclear Project, Unit 2, on August 15, 1992. The instability occurred outside of current generic exclusion regions and was caused primarily by excessive radial and axial power peaking distributions resulting from inappropriate control rod patterns during startup, and by hydraulic effects from a core loading mixing previous cycle 8x8 fuel and new—more two-phase flow-resistant and less stable--9x9 fuel assemblies.

The staff is still reviewing the ATWS/stability issues. Its evaluation report is expected in April 1993. The staff will consider both the BWROG February 1992 report NEDO-32047, "ATWS Rules Issues Relative to BWR Core Thermal-Hydraulic Stability," and the December 1992 report NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," as well as the work by Brookhaven National Laboratory on large oscillations described in NUREG/CR-5186, "BWR Stability Analysis With the BNL Engineering Plant Analyzer." The staff has thus far concluded that for some ATWS events, large oscillations are possible and may lead to melting of a small fraction of the fuel. However, primary system and containment integrity will be maintained, via approved emergency operating procedures if necessary, and the radiological consequences will remain within 10 CFR Part 100 limits.

Recommendation Source: Case Study AEOD/C90-01 (NUREG-1275, Vol. 6)

Responsible AEOD Engineer: H. Ornstein

Title or Subject: "Solenoid-Operated Valve Problems at U.S. Light Water Reactors"

Recommendation 1: Licensees should review solenoid-operated valve (SOV) design specifications and actual operating conditions to verify proper design and service conditions.

Recommendation 2: Licensees should implement SOV maintenance programs to replace or refurbish SOVs on a timely basis.

Recommendation 3: The training of the licensees' operation and maintenance personnel should emphasize the importance of surveillance testing, root-cause failure analysis, and timely repair or replacement.

Recommendation 4: Licensees should verify the use of qualified SOVs in all safety-related applications.

Recommendation 5: Licensees should consider staggered maintenance and testing of SOVs and also consider use of diverse SOVs (different design or manufacturer).

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Responsible Office/Div/Br	Contact	Priority
NRR/DOEA/EAB	T. J. Carter	High

Status: Active. The case study was issued in December 1990, and issued as NUREG-1275, Vol. 6, "Operating Experience Feedback Report—Solenoid-Operated Valve Problems," in February 1991. Generic Letter 91-15, "Operating Experience Feedback Report, Solenoid-Operated Valve Problems," was issued in September 1991 to alert licensees to the issues presented in the case study. AEOD has been working with the Electric Power Research Institute/Nuclear Maintenance Assistance Center (EPRI/NMAC) to develop an SOV maintenance guide. An EPRI/NMAC SOV maintenance guide was issued. EPRI/NMAC has held two SOV workshops in which AEOD participated. AEOD has also been working with the Institute of Electrical and Electronics Engineers (IEEE) and the American Society of Mechanical Engineers (ASME) to formulate industry consensus documents on SOVs. IEEE is printing an update to its SOV good practice guidance. ASME has formed an operations and maintenance (O&M) committee to write an SOV performance testing guide. Resolution is anticipated in 1994.

Recommendation 6: An industry group such as the Institute of Nuclear Power Operations (INPO) should initiate an SOV failure-feedback program.

Responsible Office/Div/Br	Contact	Priority
NRR/DOEA/EAB	T. J. Carter	High
AEOD/DSP/ROAB	H. Ornstein	

Status: Resolved. INPO has worked with EPRI/NMAC to make SOV failure data available to industry on an as-needed, but not on a periodic, basis. No further activities are expected in this area.

Recommendation Source: Special Study AEOD S92-02

Responsible AEOD Engineer: C. Hsu

Title or Subject: "Pressure Locking and Thermal Binding of Gate Valves"

Recommendation 1: Licensees should evaluate all safety-related gate valves to determine potential susceptibility to pressure locking or thermal binding. The evaluation should employ indepth engineering analyses to cover all plant operating and accident modes.

Recommendation 2: For those valves identified as potentially susceptible to the binding mechanisms, licensees should implement effective valve modifications and appropriate procedures to prevent the binding from occurring.

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Responsible Office/Div/Br	Contact	Priority
NRR/DE/EMEB	T. Scarborough	High

Status: Active. The special study was issued in December 1992. NRR plans to conduct an NRC staff workshop and provide guidance in Temporary Instruction 2515/109 for GL 89-10 ("Safety-Related Motor-Operated Valve Testing and Surveillance") Part 2 inspections. NRC inspectors plan to verify the adequacy of licensees' evaluations and corrective actions on the potential binding mechanisms of safety-related gate valves. NRR also plans to discuss this issue in a planned public workshop during 1993 about GL 89-10 activities. Since past industry efforts have not resulted in resolution of the issue, in 1993 NRR also plans to request that the Nuclear Management and Resources Council develop specific industry guidance on approaches to analyze and remedy the problem.

Appendix F

Status of NRC Staff Actions for Reactor Events Investigated by Incident Investigation Teams

Appendix F

Status of NRC Staff Actions for Reactor Events Investigated by Incident Investigation Teams

In accordance with NRC Management Directive 8.3, "NRC Incident Investigation Program," dated August 12, 1992, on receipt of an incident investigation team (IIT) report, the Executive Director for Operations (EDO) shall identify and assign NRC office responsibility for potential industry-generic and plant-specific actions resulting from the investigation that are safety significant and warrant additional attention or action. Office directors designated by the EDO as having responsibility for the resolution of issues or concerns are responsible for providing written status reports on the disposition of assigned actions. In addition, followup actions associated with the IIT report do not necessarily include all licensee actions, nor do they cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement actions. These items are expected to be defined and implemented through the normal organizational structure and procedures.

This appendix summarizes the disposition and/or status for each of the action items that the EDO assigned to the various NRC offices as a result of the findings associated with the completed incident investigations at reactor facilities.

For the San Onofre, Vogtle, and Nine Mile Point Unit 2 IIT investigations, the appendix addresses the status of the staff actions that had not been documented as resolved in the 1991 AEOD Annual Report.

AEOD IIT Action Tracking System

Action Source:	**IIT Report on San Onofre Nuclear Generating Station (SONGS) Unit 1 Event of November 21, 1985 (Reference 1)**

Item 7: Adequacy of emergency notifications and NRC response

Action (b): Evaluate the need for changes in NRC policy or guidance regarding the use of the emergency notification system (ENS) line, the use of NRC personnel as ENS communicators, and possible approaches to improve the ability to determine overall plant status. (Responsible Office: AEOD)

Disposition: Ongoing

The staff is continuing to develop the site-specific information system as information is compiled into plant data books. Information related to PWR facilities has been completed, and information related to BWR facilities is being organized. The hard-copy information will then be computerized into the emergency response data system. Completion is expected by December 31, 1993.

Reference: 1. NUREG-1190, "Loss of Power and Water Hammer Event at San Onofre, Unit 1, on November 21, 1985," dated January 1986.

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Action Source: **IIT Report on Vogtle Unit 1 Report of March 20, 1990 (Reference 1)**

Item 1: Adequacy of shutdown risk management

Action (a): Review existing regulatory guidance related to shutdown risk control and issue such new guidance as may be needed. Include in the assessment of shutdown risk management: normal and standby electrical systems and sources, including switchyard equipment; normal and alternate cooling systems; special alternate plans for loss of forced circulation; fission product barriers, including primary and containment systems and special activities such as movement of heavy loads or construction activities. (Responsible Office: Office of Nuclear Reactor Regulation (NRR))

Disposition: Ongoing

The staff integrated this action into its continuing interoffice evaluation of safety risks during shutdown and low-power operation.

The staff is developing a generic letter addressing resolution of the issues identified in its evaluation of shutdown and low-power operations pursuant to 10 CFR 50.54(f). Completion is expected by September 30, 1993.

Action (b): Continue to develop shutdown risk analysis methodology and review the effectiveness of alternate cooling methods for loss of forced circulation. Issue new guidance as appropriate. (Responsible Office: Office of Nuclear Regulatory Research (RES))

Disposition: Ongoing

RES is developing risk methods for operating modes other than full power, including the shutdown mode. Completion is expected by December 31, 1993.

RES completed its review of alternate cooling methods as documented in NUREG/CR-5855. In a memorandum of August 27, 1991 (Reference 5), it provided recommendations regarding generic communications of guidance to licensees in planning options in the event of loss of residual heat removal.

Action (c): Review the present regulatory requirements, such as standard technical specifications for shutdown conditions, and revise as needed the results of Action (a) above. Develop guidance documents such as emergency operating procedures, accident management procedures, and plant Technical Specifications, as necessary. (Responsible Office: NRR)

Disposition: Ongoing

The staff integrated this action into its continuing interoffice evaluation of safety risks during shutdown and low-power operation.

The staff is developing a generic letter addressing resolution of the issues identified in its evaluation of shutdown and low-power operations pursuant to 10 CFR 50.54(f). Completion is expected by September 30, 1993.

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Item 3: Adequacy of diesel generator instrument and control systems

Action (a): Evaluate the need for reexamining emergency diesel generator annunciators and control panels, including provisions for alarm printout. Consider the need for reexamining local sequencer panels. (Responsible Office: NRR)

Disposition: Resolved

On June 3, 1991, NRR issued Information Notice 91-34, "Potential Problems in Identifying Causes of Emergency Diesel Generator Malfunctions." No further action is deemed warranted. (Reference 9)

Action (b): Evaluate the need for additional guidance and increased emphasis on procedures and training for emergency diesel and local sequencers including response to malfunctions. (Responsible Office: NRR)

Disposition: Resolved

On January 16, 1992, the staff prepared a draft information notice (IN), "Potential Problems Regarding Protective Trips That Are Bypassed During Accident Conditions and Improving Operator Action." However, NRR's internal review of this draft IN showed that the information it contained was already known to the industry and that no generic communication was needed. No further action is planned. (Reference 9)

Item 4: Adequacy of emergency preparedness

Action (a): Evaluate and revise as necessary, the guidance in NUREG-0654 to classify events that could occur during cold shutdown and loss-of-electrical-power events. Evaluate the NRC guidance to licensees on classification procedures and revise as appropriate. Evaluate the guidance to licensees for personnel accountability during outages. Revise and follow up as appropriate. Evaluate guidance to licensees regarding the availability of notification systems (and alternates) during a loss-of-power event. Consider the priorities and requirements for notifying offsite authorities. Follow up as appropriate. (Responsible Office: NRR)

Disposition: Ongoing

The staff has integrated this action into its evaluation of safety risks during shutdown and low-power operation. It has received draft guidelines from the Nuclear Management and Resources Council (NUMARC) and is reviewing and commenting on them. The staff will analyze the results of the evaluations performed for Item 1 and the guidelines from NUMARC and determine the need for and develop, as appropriate, new or revised guidance pertaining to emergency preparedness (NUREG- 0654). Completion is expected by December 31, 1993.

References:

1. NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operation at Vogtle Unit 1 on March 20, 1990," dated June 1990.

2. Memorandum from J. Taylor to NRC staff, "Staff Actions Resulting From the Investigation of the March 20, 1990, Incident at Vogtle Unit 1 (NUREG-1410)," dated June 21, 1990.

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References (cont.)

3. NUREG-1272, Volume 5, No. 1, "Office for Analysis and Evaluation of Operational Data 1990 Annual Report," dated July 1991.

4. Policy Issue (Information) from J. Taylor to the Commissioners, "Evaluation of Shutdown and Low Power Risk Issues (SECY-91-283)," dated September 9, 1991.

5. Memorandum from D. E. Solberg to M. A. Caruso, "Completion of Section III.D., 'Evaluate Decay Heat Removal Methods' of Staff Plan for Evaluating Risks During Shutdown and Low Power Operations," dated August 27, 1991.

6. Information Notice 90-25, "Loss of Vital Power With Subsequent Reactor Coolant System Heat-up," dated March 11, 1991.

7. Memorandum from T. Murley to J. Taylor, "Closure of Issue 3 of the NRR Staff Action Plan Resulting From the Vogtle Incident Investigation Team (IIT) (NUREG-1410)," dated March 17, 1991.

8. Memorandum from C. H. Berlinger to J. L. Blaha, "WITS Items No. 90-163 and 90-273—Closure," dated August 29, 1991.

9. Memorandum from D. S. Hood to C. Harwood, "Closure of TAC M77007 Regarding Vogtle IIT Action Item 3," dated February 18, 1993.

10. NUREG/CR-5855, "Thermal-Hydraulic Processes During Reduced Inventory Operation With Loss of Residual Heat Removal," dated April 1992.

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Action Source: **IIT Report on Nine Mile Point Unit 2 Transformer Failure and Loss of Instrument Power on August 13, 1991 (Reference 1)**

Item 1: Adequacy of uninterruptible power supply installations

Action (a): Evaluate the need to review the adequacy of the design of safety-related and non-safety-related uninterruptible power supplies (UPSs) regarding design vulnerabilities. (Responsible Office: NRR)

Disposition: Resolved

The staff evaluated the need to review the adequacy of designs of safety and non-safety-related UPSs regarding design vulnerabilities (Reference 2). It evaluated the safety significance of this issue in view of (1) the redundancy of the systems using UPSs, (2) the issuance of the maintenance rule, (3) NRC resources needed to conduct such a review, (4) the uniqueness of plant-specific UPS designs, and (5) the fact that the NRC had provided information on UPS design vulnerability to licensees through IN 91-64, "Site Area Emergency Resulting From a Loss of Non-Class 1E Uninterruptible Power Supplies." The results of the evaluation formed the basis for NRR determining that design adequacy reviews could most effectively be addressed by issuing a supplement to IN 91-64 giving additional detailed information about the vulnerability of the UPS design. Thus, on October 7, 1992, the staff issued Supplement 1 to IN 91-64 that (1) included before and after schematic diagrams showing the design changes made at Nine Mile Point Unit 2 (NMP-2) to reduce vulnerability to UPS power output loss, (2) provided additional information on intervals for replacing the control logic battery packs, and (3) notified licensees that other designs could have similar vulnerabilities.

Action (b): Evaluate the actions taken by the licensee at NMP-2 to address design and maintenance issues for the UPS. (Responsible Office: Region I)

Disposition: Ongoing

Region I staff has reviewed and closed 12 of the licensee's 14 corrective actions (References 3, 4, 5). One issue, the completion of a detailed root-cause investigation by a failure-prevention consultant, has been used by the licensee to support its positions and required no specific regional followup. The remaining issue involved licensee review of procedures to be used on loss of power to various plant electrical buses. The licensee deferred action on this issue in its original response to Office of Inspection and Enforcement Bulletin 79-27 because of the prelicensing status of the plant. The licensee had completed a failure modes and effects analysis, and NRC review of the evaluation concluded that the plant could achieve cold shutdown following the loss of any single power bus and that clear and unambiguous indication of an undervoltage condition is available to alert operators to the loss of power. The remaining item was the licensee's reevaluation of plant procedures for mitigating the effects of loss of bus power. This item was completed by the licensee in mid-December 1992 in conjunction with its development of station blackout procedures.

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Item 2: Adequacy of instrumentation and emergency operating procedure integration

Action (a): Audit the emergency planning guidelines (EPGs) for instrumentation associated with manual operator actions for critical safety functions. (Responsible Office: NRR)

Disposition: Ongoing

The staff determined that emergency operating procedures (EOPs) rather than EPGs should be audited to most effectively address this action item (Reference 6). The staff has audited the EOPs at four plants considered to represent Westinghouse, General Electric, Babcock & Wilcox, and Combustion Engineering units. It has prepared a detailed compilation of actions and instruments from these EOPs. As expected, the audits showed that some instruments required for manual operator actions were not classified as Type A. (Regulatory Guide 1.97 requires that instruments be classified as Type A if they are needed by operators during design-basis events to take specified manual actions for which no automatic control is provided and if they are required for safety systems to accomplish their safety functions. Type A instruments should meet Category I requirements unless otherwise accepted by the staff. Category I requirements include environmental and seismic qualifications and redundant Class 1E power supplies.) The staff will now review the use of these instruments to determine if any are needed for design-basis events and should be upgraded to meet Category I requirements. After completing this work, the staff will prepare a final report that will list instruments associated with manual operator actions for the control of critical safety functions and will list any instruments that should be upgraded to meet Category I requirements.

As of September 1992, the staff completed audits of the four plants. This action item was originally scheduled to be completed on December 31, 1992. However, to provide a more meaningful final report, the staff decided to perform reviews to determine if instruments identified in the EOPs for manual operator actions during design-basis events are properly classified as Type A. Accordingly, the staff requested approval to extend the scheduled completion date to March 31, 1993.

Action (b): Review the vulnerability to a loss of power for critical safety instrumentation identified in Item 2(a) above. (Responsible Office: NRR)

Disposition: Ongoing

Work to address this issue will depend on the results of Action Item 2(a) (Reference 6). The staff will not act further if it finds that all instruments listed in the EOPs for manual operator actions during design-basis events are qualified to Category I requirements. However, if the results of Action Item 2(a) indicate that this is not the case, the staff will review the power supplies and vulnerability to a loss of power for the specific instruments that are not qualified to Category I requirements and identified in the EOPs for manual operator actions during design-basis events. The staff will conclude this work by submitting a final report in which it will assess the vulnerability to loss of power of any instruments identified as a result of Action Item 2(a) that are not qualified to Category I requirements and identified in the EOPs for manual operator actions during design-basis events. On reviewing the conclusions in the final report, the staff will determine if it needs to take additional actions.

This action item was originally scheduled to be completed on February 28, 1993. However, the staff cannot begin work on this item until it completes Action Item 2(a).

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Item 2 (cont.)

Accordingly, it requested approval to extend the scheduled completion date to May 31, 1993, to reflect the 3-month extension in the completion date for Action Item 2(a).

Action (c): Evaluate the need to provide an alternate rod position indication (RPI) or safety-grade power for BWRs. (Responsible Office: NRR)

Disposition: Ongoing

The staff met with the BWR Owners Group (BWROG) on December 9, 1992, to discuss the industry's position on the need to provide an alternate RPI or safety-grade power for BWRs. The BWROG will submit a position paper, which the staff will review before preparing its position. The staff will prepare a safety evaluation scheduled for August 1993 to support its position and will determine the need for further action accordingly.

Item 3: Adequacy of emergency operating procedures and associated training

Action (a): Evaluate the need to review the adequacy of BWR emergency procedure guidelines (EPGs) for prioritizing control of critical safety functions and the adequacy of the guidance for stabilizing decreasing reactor pressure. (Responsible Office: NRR)

Disposition: Resolved

The NRR staff has completed its evaluation of the need to review the adequacy of BWR EPGs with regard to prioritizing the control of critical safety functions (References 7, 8). In its evaluation the staff considered information from Region I and the BWROG Emergency Procedures Committee (EPC). The staff and the BWROG-EPC concluded that it is not prudent to prioritize actions within the EPGs. Specifically, the EPGs were purposely developed for symptom-based procedures so that an initiating event would not need to be identified to determine which procedure should be entered. Operator actions, therefore, are appropriate irrespective of the initiating event or the sequence in which subsequent events occur. The operator actions specified are consistent with the manner in which control room operators actually operate, including concurrent or parallel performance of actions. The priority with which key parameters are addressed is purposely not specified within the EPGs. To preassign a priority entails presupposing the event sequence. Prioritizing the control of key parameters is best determined by the trained operating crew on duty.

The NRR staff has completed an evaluation of the need to review the adequacy of BWR EPGs with regard to the guidance for stabilizing decreasing reactor pressure. In this evaluation the staff also considered information from the BWROG-EPC. The NRR staff and the BWROG-EPC concluded that some additional guidance could enhance operator response. Specifically, the BWROG-EPC will clarify what "stabilize reactor pressure" means in the appendices to the EPGs. The EPGs themselves will not be revised. This clarification to the appendices will be done in the next revision to the EPGs (Rev. 5), scheduled for the beginning of calendar year 1993.

Action (b): Evaluate the need to review the adequacy of training programs and associated emergency operating procedures (EOPs) regarding training for a loss of annunciators combined with a scram or other combinations of events. (Responsible Office: NRR)

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Item 3 (cont.)

Disposition: Ongoing

The staff is currently considering the impact of the recent loss of annunciator power supply on the proposed disposition of Generic Issue 76, "Instrumentation and Control Power Interactions." In addition, the staff is preparing a Commission paper on design certification and licensing policy issues for passive and evolutionary plant designs in which it expects to recommend that the Commission approve a position that the design of control room annunciator systems in future plants be more robust. Finally, NRR is also evaluating the recent loss-of-annunciator events as part of the NMP-2 event followup actions to determine the need for any additional generic action beyond the above efforts.

In the interim, NRR will continue to conduct EOP inspections, training inspections, and operator licensing activities to assess the adequacy of the facility's programs and performance. During these activities, NRR will monitor how the licensees are dealing with loss-of-control-room-annunciator events in their training programs and EOPs (Reference 9).

Item 4: Adequacy of regulatory guidance regarding non-safety-related equipment and instrumentation required for accidents

Action (a): Evaluate the need to provide additional regulatory guidance that conveys the staff expectations regarding maintenance on important non-safety-related equipment. (Responsible Office: RES)

Disposition: Ongoing

The maintenance rule, 10 CFR 50.65, requires licensees to monitor the effectiveness of maintenance efforts on structures, systems, and components within the scope of the rule against goals that the licensee establishes. The scope of the maintenance rule covers a great deal of non-safety-related equipment, including the equipment that failed at NMP-2. Monitoring activities under the maintenance rule are, in general, performance based. Goal setting, monitoring, and subsequent evaluation and feedback efforts are the central focus of the maintenance rule and the forthcoming regulatory guide. The regulatory guide that is being written for the maintenance rule will be consistent with the rule and provide guidance to licensees to implement the rule.

Under the provisions of the maintenance rule, licensees would be expected to take corrective actions in response to failure to meet previously established goals. For non-safety-related equipment, such as the UPS that failed at NMP-2, there is now industrywide experience to indicate to licensees that, depending on their goals, monitoring of such equipment, along with other preventive maintenance measures, should be undertaken. In NUREG-1455, the staff speculated that the maintenance rule's provisions would probably not have prevented the incident described therein. This would be true for the initial occurrence. However, the maintenance rule and the regulatory guide are expected to

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Item 4 (cont.)

encourage licensees to be more sensitive to precursor events and more thorough in applying industry experience to their maintenance efforts. It is expected that licensees would learn from events such as the one at NMP-2 and adjust their maintenance practices for equipment such as the UPS accordingly. Therefore, the number of similar incidents involving this type of equipment would be expected to be minimized.

The maintenance rule is performance oriented as will be the regulatory guide. Licensees are to be allowed flexibility in their maintenance practices as long as the results are acceptable. Thus, direct prescriptive instructions to licensees regarding basic maintenance practices, including such practices as the control of drawings and manuals, are not part of the maintenance rule and will not be contained in the regulatory guide.

The regulatory guide for the maintenance rule is to be completed and issued by June 30, 1993 (Reference 10).

Item 5: Shift coping

Action (a): Evaluate the need to review the adequacy of control room staffing during the simultaneous implementation of emergency operating procedures (EOPs) and emergency response procedures (ERPs) by normal shift crews. (Responsible Office: NRR)

Disposition: Ongoing

NRR has completed an evaluation of the need to review the adequacy of control room staffing during simultaneous implementation of EOPs and ERPs by normal shift crews (Reference 11). For this evaluation, NRR particularly considered the need to review the adequacy of control room staffing when a minimum shift crew would need to simultaneously implement EOPs, the emergency plan, and other functions necessary to respond to an event. The NRR evaluation involved the review of preliminary findings from RES research on the bases used by licensees to determine staffing levels required for plant operations, NRR staff and industry survey results, and results from AEOD's evaluation of the effectiveness of control room organizations. NRR has concluded that there is a need for further review of the adequacy of control room staffing and is reviewing this issue. On completion of this review, NRR will prepare a SECY paper to inform the Commission of its position and recommended actions necessary to address the adequacy of control room staffing. The staff anticipates that this review will be completed in 1993.

Action (b): Incorporate into the ongoing review of shift technical advisor (STA) implementation consideration of the integration of the STA function into the shift crew during command changes. (Responsible Office: NRR)

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Item 5 (cont.)

Disposition: Ongoing

The NRR staff discussed the lack of STA participation in shift turnover activities at some facilities in SECY 92-026, "Implementation of the Shift Technical Advisor at Nuclear Power Plants," January 21, 1992. The staff continues to study the integration of the STA function in order to ensure that timely engineering expertise in response to plant transients or abnormal conditions is available and has incorporated consideration of the integration of the STA function and the shift crew during command changes into the ongoing review of STA implementation. The staff has this issue under review. On completion of this review, the staff will prepare a SECY paper to inform the Commission of its position and recommended actions to address use of the STA. The staff anticipates that this review will be completed in 1993.

Action (c): Evaluate the actions being taken by the licensee at Nine Mile Point to address shift coping issues. (Responsible Office: NRR)

Disposition: Resolved

Licensee actions in this area were reviewed during a procedure and training actions team inspection (50-410/92-27) (Reference 4) and an NRC requalification examination and program. The licensee has implemented changes to relieve the shift supervisor of dual emergency responsibilities, such as integration of the STA into crew emergency response and use of the assistant shift supervisor as EOP implementer. These changes have been confirmed by the NRC staff. This issue is closed.

Item 6: Condensate booster pump injections at BWR design plants (Part 1)

Action (a): Consider the need for actions by the licensee at NMP-2 to address condensate booster pump injections including the need for automated booster pump trip, anticipatory procedural guidance, and mass and heat balance calculations. (Responsible Office: Region I)

Disposition: Resolved

The licensee took two specific corrective actions to ensure operator awareness of similar situations through training and review of the specific event.

The two specific actions taken by Niagara Mohawk were inspected by Region I (50-410/92-27) (Reference 4). This inspection included evaluating (1) training materials developed for providing guidance in anticipating and avoiding a reactor vessel overfill condition and (2) the effectiveness of the training for preventing reactor vessel overfill. The staff concluded that the specific NUREG-1455 training has been implemented. Operators have been sensitized to the potential for reactor vessel overfill by the condensate booster pumps at reduced reactor coolant system pressure.

Region I has also reviewed information submitted by the licensee and the effectiveness of the licensee's training in preventing advertent booster pump injections (Part 1, above), and has concluded that requiring a modification in this area would be an unnecessary

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Item 6 (cont.)

backfit. Review of procedures and training by Region I (50–410/92-27) (Reference 4) showed that operators have been adequately sensitized to prevent inadvertent injection; therefore, this issue is closed.

Item 7: Adequacy of plant-specific operating and recovery procedures

Action (a): Evaluate licensee corrective actions with respect to the procedures discussed above. Include consideration of the need for the scram procedure to segregate and make a distinction between immediate actions and supplemental action in accordance with American National Standards Institute/American Nuclear Society (ANSI/ANS) 3.2-1982 as discussed in Section 5.6.1 of the IIT report. (Responsible Office: Region I)

Disposition: Resolved

The licensee provided five specific corrective actions dealing with procedural improvements and related training.

The five specific licensee actions have been reviewed by Region I and are closed. In addition to the specified licensee actions, an NRC inspection team (Procedures and Training Actions, IR 50–410/92-27) (Reference 4) reviewed licensee procedures for compliance with station administrative requirements and technical adequacy. The team identified a number of weaknesses, and regional review of licensee actions in this area is continuing. Also, the licensee's quality assurance organization is reviewing station compliance with ANSI 3.2-1981, and a number of licensee procedures are being rewritten and reformatted as necessary to distinguish between immediate and subsequent operator actions. Followup by inspectors of licensee actions in both matters has been entered in the Region I followup tracking system. This issue is broader than that noted by the IIT, and planned followup is considered part of the Master Inspection Program for the unit. The licensee has resolved the specific procedural problems identified by the IIT, and the issue is closed.

References:

1. NUREG-1455, "Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1991," dated October 1991.
2. Memorandum from T. Murley to J. Taylor, "Closure of Item 1.a of NRR Staff Action Plan Resulting From Investigation of the August 13, 1991, Incident at Nine Mile Point Nuclear Station, Unit 2 (NUREG-1455)," dated December 4, 1992.
3. Memorandum from T. Martin to J. Taylor, "Status Report on Staff Actions Resulting from the Investigation of the August 13, 1991, Incident at Nine Mile Point Unit 2 (NUREG-1455)," dated January 12, 1993.
4. NRC Inspection Report No. 50–410/92-27, dated November 12, 1992.
5. NRC Inspection Report No. 50–410/92-21, dated November 13, 1992.

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References (cont.)

6. Memorandum from T. Murley to J. Taylor, "Second 6-Month Status Report Regarding NRR Actions Resulting From Investigation of the August 13, 1991, Incident at Nine Mile Point Nuclear Station, Unit 2 (NUREG-1455)," dated December 28, 1992.

7. Memorandum from T. Murley to J. Taylor, "Closure of Item 3.a of NRR Staff Action Plan Resulting From Investigation of the August 13, 1991, Incident at Nine Mile Point Nuclear Station,Unit 2 (NUREG-1455)," dated December 15, 1992.

8. Letters from C. Tully, BWROG, to A. Thadani, USNRC, dated September 22, 1992, and November 25, 1992, "BWR Owner's Group Response to Comments Resulting From the Investigation of the August 13, 1992, Incident at Nine Mile Point Unit 2 (NUREG-1455)."

9. Memorandum from T. Murley to E. Jordan, "Adequacy of Emergency Operating Procedures and Training Associated With Loss of Control Room Annunciators," dated August 10, 1992.

10. Memorandum from E. Beckjord to J. Taylor, "Staff Actions Resulting From the Investigation of the August 13, 1991 Incident at Nine Mile Point 2 (NUREG-1455)," dated February 3, 1992.

11. Memorandum from T. Murley to J. Taylor, "Closure of Items 5.a and 5.b of NRR Staff Action Plan Resulting From Investigation of the August 13, 1991, Incident at Nine Mile Point Nuclear Station, Unit 2 (NUREG-1455)," dated December 15, 1992.

Appendix G

Status of NRC Staff Actions Involving Potential Generic Issues Resulting From Diagnostic Evaluation Team Findings

Appendix G

Status of NRC Staff Actions Involving Potential Generic Issues Resulting From Diagnostic Evaluation Team Findings

In accordance with "NRC Diagnostic Evaluation Program, Handbook 8.7," dated June 7, 1991, on receipt of a diagnostic evaluation team (DET) report, the Executive Director for Operations (EDO) will assign NRC office responsibility for generic and plant-specific staff actions resulting from the diagnostic evaluation (DE). Office directors designated by the EDO as having responsibility for resolving issues or concerns are responsible for providing written status reports on the disposition of assigned actions. The Director, AEOD, will maintain a status of the staff actions involving generic issues. The status of these issues will be compiled in the AEOD Annual Report.

This appendix summarizes the disposition or status or both for each of the generic NRC staff action items that the EDO assigned to the various NRC offices as a result of findings associated with each of the completed DEs. It includes only those staff actions for which the resolutions were still ongoing at the end of calendar year 1992.

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Action Source: **Memorandum from J. Taylor to Office Directors and Region III Administrator, "Staff Actions Resulting From the Diagnostic Evaluation at Zion Nuclear Station," dated September 4, 1990.**

Item 1: The licensee had no program for testing molded-case circuit breakers.

Action: Review the failure history of safety-related MCCBs to determine if a bulletin or other form of generic communication should be required. (Responsible Office: AEOD)

Disposition: Resolved

AEOD completed a special study entitled "Review of Operational Experience With Molded Case Circuit Breakers in U.S. Commercial Nuclear Power Plants," AEOD/S92-03, which was issued July 1, 1992. AEOD concluded that the operational experience with molded case circuit breakers (MCCBs) did not support a specific regulatory initiative. (Reference 1)

Notwithstanding the conclusion of this study, the Office of Nuclear Reactor Regulation (NRR) issued Information Notice (IN) 92-51, "Misapplication and Inadequate Testing of Molded-Case Circuit Breakers," on July 9, 1992. This IN reports that licensees, when determining the MCCB design parameters for motor loads, occasionally underestimate or neglect to consider the inrush transient current occurring during the first few cycles after a motor is started. Often only the locked-rotor current is considered in selecting the appropriate MCCB. The IN then adds that "testing of properly applied and set MCCBs in accordance with industry recommended practices should provide reasonable assurance that the MCCBs' instantaneous trip performance is acceptable for safety-related applications." (Reference 2)

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Action Source:	**Memorandum from J. Taylor to Office Directors and Region I Administrator, "Staff Actions Resulting From the Diagnostic Evaluation at Oyster Creek Nuclear Generating Station," dated February 14, 1991.**
Item 1:	Adequacy of the torus vacuum-breaker system.
Action:	Assess the generic implications regarding the capability to satisfy both the vacuum breaker and containment isolation functions in similar Mark I plants. (Responsible Office: NRR)
Disposition:	Resolved NRR has completed a preliminary evaluation of this item and concluded that although it is generic, it is of a low safety significance and will not be actively pursued at this time.

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Action Source: **Memorandum from J. Taylor to Office Directors and Region I Administrator, "Staff Actions Resulting From Diagnostic Evaluation at FitzPatrick Nuclear Station," dated December 3, 1991.**

Item 1: Site evacuation and fire alarms in the main control room were too loud.

Action: Assess the need for generic communication or inspection based on NUREG-0700. (Responsible Office: NRR)

Disposition: Resolved

The FitzPatrick Detailed Control Room Design Review (DCRDR) Summary Report and Implementation Schedule was submitted on February 28, 1986. At that time, six human engineering deficiencies concerning control room sound level were identified. A further engineering study was required to determine the best combination of audible alarm volume and control room ambient sound levels. As a result of the study, Power Authority of the State of New York (PASNY) decided to install carpeting in the control room. This was completed several years before the DE.

Following the DE, the resident inspector monitored PASNY's weekly test of the fire, evacuation, and station alarms. The auditory signal intensities measured in the control room by PASNY during the alarm test were in the range of 78 to 81 decibels. Section 6.2.2.6, "Signal Intensity," of NUREG-0700, "Guidelines for Control Room Design Reviews," specifies that auditory signal intensities should not exceed 90 decibels, except for evacuation signals, which may be up to 115 decibels. PASNY's test results are within these guidelines. The test results were documented in a routine resident inspector report (Reference 3).

The staff concluded that no generic communications or generic inspections were necessary because control room noise was not a problem at FitzPatrick at the time of the DE (Reference 4).

Item 6: The RCIC/HPCI turbine exhaust steam line vacuum breaker isolation valves were neither being reated as primary containment isolation valves (PCIVs), nor included in the licensee's inservice testing program.

Action: Evaluate the safety significance of these valves and consider what generic action (if any) may be appropriate. (Responsible Office: NRR)

Disposition: Resolved

By letter dated January 16, 1992, the NRC staff requested PASNY to provide information regarding the design specifications and surveillance requirements for the reactor core isolation cooling/high-pressure coolant injection (RCIC/HPCI) vacuum breaker isolation valves. By letter dated February 18, 1992, PASNY provided the NRC with design specifications for the RCIC/HPCI turbine exhaust steam lines and committed to perform a detailed engineering analysis of the containment isolation aspects of these lines before startup from the 1992 refueling outage. PASNY completed the engineering analysis and concluded that the subject isolation valves—one manual and one motor-operated gate

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valve on each HPCI/RCIC vacuum breaker line—are not PCIVs. PASNY concluded that the PCIVs for the HPCI/RCIC vacuum breaker lines are provided by the HPCI/RCIC turbine exhaust check valves, which are leak tested in accordance with the Technical Specifications. The vacuum breaker lines are designed to seismic and quality assurance Category I requirements. The HPCI and RCIC vacuum breaker line configurations prevent the performance of an outward leakage test except during an integrated leak rate test (ILRT).

On the basis of its review of the design and configuration of the 5-cm (2-in.) HPCI and 3.8-cm (1 1/2-in.) RCIC vacuum breaker lines, NRR has concluded that the vacuum breaker isolation valves are not PCIVs. The containment isolation for the RCIC/HPCI vacuum breaker lines is provided by the redundant (two) RCIC/HPCI turbine exhaust check valves, which are tested for leakage in accordance with the Technical Specifications. However, to ensure no leakage through the vacuum breaker valve packing or bonnets, NRR recommended that PASNY perform a soap bubble leak test of these valves during each ILRT PASNY agreed with this recommendation and committed to perform this test during each ILRT.

No generic action is required on this item because (1) no other licensee is believed to have this design and (2) there is no regulatory basis to require the soap bubble tests. This was merely a commitment agreed to by PASNY. This action is documented in a routine resident inspector report. (Reference 5)

Item 11: The licensee had been installing "refurbished" Rosemount pressure transmitters (Model Number 1153) of the type listed in Bulletin 90-01. However, in several instances, the serial number had not been updated to indicate the refurbishment. In addition, the licensee records did not provide evidence of the refurbishment.

Action: Evaluate the need for updating existing generic information regarding Rosemount pressure transmitters. (Responsible Office: NRR)

Disposition: Resolved

NRR evaluated the need for updating generic information regarding Rosemount pressure transmitters. Since the issuance of NRC Bulletin 90-01, NRR has been evaluating data gathered in response to the bulletin, made site visits to facilities, and conducted numerous meetings with representatives from industry, Nuclear Management and Resources Council (NUMARC), and Rosemount. At the request of the staff, NUMARC surveyed the industry and prepared a report covering the performance history and failures identified as related to the loss of fill oil in the population of transmitters installed in nuclear plants. Brookhaven National Laboratory assisted NRR in evaluating the NUMARC survey data and conclusions. On assessing the analyses, evaluations, and historical data related to the loss of fill oil, NRR requested that reactor licensees take further actions in a supplement to NRC Bulletin 90-01. (Reference 6)

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References:

1. AEOD/S92–03, "Review of Operational Experience With Molded Case Circuit Breakers in U.S. Commercial Nuclear Power Plants," dated July 1, 1992.
2. Information Notice 92–51, "Misapplication and Inadequate Testing of Molded-Case Circuit Breakers," dated July 9, 1992.
3. Resident Inspector Report 50–333/92–11, dated July 29, 1992.
4. Memorandum from T. E. Murley to J. M. Taylor (NRC), "Status of NRR Actions Resulting From the Diagnostic Evaluation at James A. FitzPatrick Nuclear Plant," dated December 3, 1992.
5. Resident Inspector Report 50–333/92–15, dated October 6, 1992.
6. NRC Bulletin 90–01, Supplement 1, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," dated December 22, 1992.

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The annual report of the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data (AEOD) is devoted to the activities performed during 1992. The report is published in two separate parts. NUREG–1272, Vol. 7, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. NUREG–1272, Vol. 7, No. 2, covers nonreactors and presents a review of the events and concerns during 1992 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. Each volume contains a list of the AEOD reports issued for 1984–1992.

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