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U.S. NUCLEAR REGULATORY COMMISSION

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

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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/S405, 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 on 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

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 1989. The report is published in two separate parts. NUREG-1272, Vol. 4, 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. This report also compiles the status of staff actions resulting from previous Incident Investigation Team (IIT) reports. NUREG-1272, Vol. 4, No. 2, covers nonreactors and presents a review of the events and concerns during 1989 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 1980-1989.

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Abbreviations

ac	alternating current	EED	Engineering Evaluations Department
ACRS	Advisory Committee on Reactor Safeguards	EER	engineering evaluation request
ADS	automatic depressurization system	EFIC	emergency feedwater initiation and control
ADV	atmospheric dump valve	EFO	equipment-forced outage
AEOD	NRC's Office for Analysis and Evaluation of Operational Data	ENS	emergency notification system
AFW	auxiliary feedwater	EOP	emergency operating procedure
AIT	Augmented Inspection Team	EP	emergency power
ANO	Arkansas Nuclear One	EPG	emergency procedure guideline
AO	abnormal occurrence	EPRI	Electric Power Research Institute
AOT	allowed outage time	ERDS	Emergency Response Data System
APS	Arizona Public Service Co.	ESF	engineered safety feature
ASCO	Automatic Switch Company	ESFAS	engineered safety features actuation system
ASME	American Society of Mechanical Engineers		
ASP	accident sequence precursor	FEF	Foreign Event File
ASTM	American Society for Testing and Materials	FOR	forced outage rate
ATOG	anticipated transient operating guideline	FSAR	Final Safety Analysis Report
ATWS	anticipated transient without scram		
B&W	Babcock & Wilcox Co.	GE	General Electric
B&WOG	Babcock & Wilcox Owners Group	GI	generic issue
BNL	Brookhaven National Laboratory	GL	generic letter
BWR	boiling water reactor	gpm	gallons per minute
BWROG	Boiling Water Reactor Owners Group	HOO	Headquarters Operations Officer
CCP	centrifugal charging pump	HPCI	high-pressure coolant injection
CE	Combustion Engineering	HPCS	high-pressure core spray
CEI	Cleveland Electric Illuminating Co.	HVAC	heating, ventilation, and air conditioning
CFR	<i>Code of Federal Regulations</i>		
CP&L	Carolina Power and Light Co.	IAEA	International Atomic Energy Agency
CRDM	control rod drive mechanism	IAP	integrated action plan
CRGR	Committee To Review Generic Requirements	ICS	integrated control system
CY	calendar year	IE	NRC's Office of Inspection and Enforcement
DBA	design-basis accident	IIP	Incident Investigation Program
dc	direct current	IIT	Incident Investigation Team
DE	diagnostic evaluation	IN	information notice
DEP	Diagnostic Evaluation Program	INEL	Idaho National Engineering Laboratory
DET	Diagnostic Evaluation Team	INES	International Nuclear Event Scale
DHR	decay heat removal	INPO	Institute of Nuclear Power Operations
ECCS	emergency core cooling system	IPE	individual plant examination
EDG	emergency diesel generator	IRB	AEOD's Incident Response Branch
EDO	Executive Director for Operations	IRM	intermediate range monitor
		IRS	Incident Reporting System
		IST	inservice testing
		LER	licensee event report
		LOCA	loss-of-coolant accident
		LOSP	loss of station power

Abbreviations (cont.)

LPCI	low-pressure coolant injection	RCIC	reactor core isolation cooling
LWR	light water reactor	RCP	reactor coolant pump
LWRPA	light water reactor plant analyzer	RCS	reactor coolant system
ME	maintenance effectiveness	RI	NRC Region I
MOV	motor-operated valve	RII	NRC Region II
MS	main steam	RIII	NRC Region III
MSIV	main steam isolation valve	RIV	NRC Region IV
NEA	Nuclear Energy Agency	RV	NRC Region V
NED	Nuclear Engineering Department	RES	NRC's Office of Nuclear Regulatory Research
NNI	non-nuclear instrumentation	RHR	residual heat removal
NPRDS	Nuclear Plant Reliability Data System	RPS	reactor protection system
NPSH	net positive suction head	RWCU	reactor water cleanup
NRC	U.S. Nuclear Regulatory Commission	RWST	refueling water storage tank
NRR	NRC's Office of Nuclear Reactor Regulation	SAIC	Science Applications International Corp.
NSSS	nuclear steam supply system	SALP	Systematic Assessment of Licensee Performance
NUBARG	Nuclear Utility Backfitting and Reform Group	SCC	stress corrosion cracking
NUDOCS	Nuclear Documents System	SCE	Southern California Edison Company
NUMARC	Nuclear Management and Resources Council	SCS	Society for Computer Simulation
OE	operating experience	SCSS	Sequence Coding and Search System
OECD/NEA	Organization for Economic Cooperation and Development/Nuclear Energy Agency	SE	significant event
OGC	NRC's Office of the General Counsel	SFRCS	steam feedwater rupture control system
OIG	NRC's Office of the Inspector General	SG or S/G	steam generator
ORNL	Oak Ridge National Laboratory	SI	safety injection
P	pressure	SINET	Shared Information Network
PDP	positive displacement pump	SIS	safety injection system
PI	performance indicator	SMUD	Sacramento Municipal Utility District
PORV	pilot-operated relief valve	SNUPPS	standardized nuclear unit power plant system
PRA	probabilistic risk assessment	SOER	significant operating experience report
PRB	plant review board	SONGS	San Onofre Nuclear Generating Station
PRISIM	plant risk statistical information management	SPDS	safety parameter display system
PSA	probabilistic safety analysis	SRM	source range monitor
PTS	pressurized thermal shock	SSA	safety system actuation
PWG	principal working group	SSES	Susquehanna Steam Electric Station
PWR	pressurized water reactor	SSF	safety system failure
PZR	pressurizer	STA	shift technical advisor
QA	quality assurance	STI	surveillance test interval
RBCCW	reactor building closed cooling water	SW	service water
		TAC	task action commitment
		TAP	Task Action Plan
		TCM	technical committee meeting
		TE	Toledo Edison Co.
		TMI	Three Mile Island
		TS	Technical Specification

Abbreviations (cont.)

TTC	Technical Training Center	VCT	volume control tank
UNIPED	Union of Producers and Distributors of Electrical Energy	W	Westinghouse
		WANO	World Association of Nuclear Operators
		WNP	Washington Nuclear Plant
USI	unresolved safety issue	WPPSS	Washington Public Power Supply System

Executive Summary

The mission of the Office for Analysis and Evaluation of Operational Data (AEOD) is to evaluate safety concerns involving reactor and nonreactor licensed activities, to assess operational events and experience, and to provide technical training to the NRC staff.

The elements that contribute to this mission include operational data analysis, performance indicators, operational data collection and dissemination, diagnostic evaluations, incident investigations, incident response, and technical training.

AEOD also provides support to the Committee To Review Generic Requirements, including oversight of regulatory backfits on licensed power reactors. The office's activities related to nonreactors and technical training are reported separately.

Through many of the activities of AEOD, the overall safety performance of power reactor licensees may be inferred. Comparison of 1989 performance data with data from the preceding 4 years indicates a general leveling off of the previously noted improving trend. The data examined include the performance indicators, the accident sequence precursors, and the number of abnormal occurrences. This apparent leveling off of the improving performance trend may be attributed to a combination of fewer plants on the early learning curve and the completion by the industry of improvements with high return in safety performance.

The U.S. Nuclear Regulatory Commission (NRC) is finding it increasingly difficult to extract lessons from analysis of operational data. The more obvious lessons have been identified, and more complex analysis and greater overall effort are required to generate recommendations for actions. Although licensee event reports remain the principal source of operational data, searches increasingly rely on component failure reports (the Nuclear Power Plant Reliability Data Base) and risk-based analysis (the Accident Sequence Precursor Program). In addition, it has been necessary to increase onsite collection of information through review of records, examination of equipment, and interviews with personnel to supplement and extend data analysis.

Ensuring that previous lessons have a continued effectiveness is evolving into an activity to systematically assess previous NRC actions in addition to following up on AEOD recommendations. Where continued problems are occurring despite previous actions, further recommendations may be necessary.

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 remain unlearned.

AEOD continued developmental activities to improve the safety performance indicator (PI) program. During 1989, cause code trends were implemented as a new PI; a total of eight PIs were monitored. In addition, the staff continued work to develop ways to monitor or trend maintenance effectiveness at nuclear power plants. The staff initiated a Demonstration Project with selected utilities to obtain licensees' perspectives on a proposed maintenance PI. Work also continued on the development of other potential indicators, including one based on safety system unavailability.

In 1989, about 3,100 immediate notifications, slightly less than last year, were telephoned to the NRC Operations Center. In 1989, 197 of these were classified as "Unusual Events" and 13 as "Alerts"; the latter are events during which a substantial degradation of plant safety has occurred or had the potential to occur. None were classified at higher levels of emergencies. AEOD engineers screened about 2,350 licensee event reports to identify events of safety significance.

During 1989, no event was judged to have a sufficiently high level of safety significance to warrant an onsite review by a specially formed Incident Investigation Team. However, NRC regional offices established 19 Augmented Inspection Teams (AITs), supported by headquarters personnel, to help improve the safety of the plants at which AIT inspections were conducted. These inspections consisted of detailed investigations of potentially safety-significant events and their root causes. An example of lessons learned and communicated to licensees from 1989 AIT inspections includes the possible overpressurization of low-pressure systems during testing of valves connecting these systems to higher pressure systems, with the potential for leakage of reactor coolant system water outside the containment building.

In addition, AEOD conducted performance reviews during 1989 at four licensed reactor facilities under the Diagnostic Evaluation Program: (1) the Perry Nuclear Power Plant, (2) the Brunswick Steam Electric Plant, (3) the Arkansas Nuclear One station, and (4) the Palo Verde Nuclear Generating Station. These assessments provided senior management with an independent evaluation of each licensee's overall performance in ensuring safe plant operations and identified

plant-specific performance issues, including substantial management, organizational, and technical issues, requiring licensee corrective actions and improvements.

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 the public health and safety. In 1989, NRC reported four AOs at commercial power reactors to Congress. This number is similar to the number reported in 1987 and in 1988.

AEOD continued its efforts to maintain and improve the exchange of operational experience and ideas with foreign countries as a contribution to nuclear power safety programs worldwide. In 1989, AEOD received about 130 reports of events at foreign reactors. These reports were reviewed to identify issues of importance to U.S. licensees of reactors and to disseminate findings from studies of these events. In addition, during 1989, AEOD prepared and submitted to the international community about 60 reports of U.S. incidents that addressed operational events and generic concerns identified within the U.S. program; a number of these reports have been valuable to the nuclear safety programs of foreign countries. Also, in 1989, the Paris-based Nuclear Energy Agency (NEA) and the NRC, as the responsible U.S. Government agency, finalized an agreement authorizing the NRC to assume responsibility for operation of NEA's Incident Reporting System database, which contains data on worldwide nuclear power plant operating experience, and which was previously located in Europe. AEOD supervises this effort through a management contract with the Oak Ridge National Laboratory.

AEOD's international efforts extend beyond these programs to include exchanges of information and ideas on a variety of topics of international interest as part of NRC's overall international program. For example, the

AEOD staff provided assistance to foreign countries and to the International Atomic Energy Agency in a number of safety-related areas and supported working group sessions of the U.S./U.S.S.R. Joint Coordinating Committee for Civilian Nuclear Reactor Safety.

Work also has continued on the Accident Sequence Precursor Program, which evaluates licensee event reports of plant problems, equipment failures, or other operating incidents, to identify and categorize precursors to potential severe core-damage accidents and to estimate the likelihood that core damage will occur. This quantitative assessment of operational events complements the staff's qualitative evaluation as well as the staff's comparison of significant operational events to safety analyses in support of operation. The numerical quantification provides valuable perspectives regarding areas in which plant safety and reliability could be improved.

For the third consecutive year, the total number of outstanding AEOD recommendations has declined, largely because of expeditious resolution of generic issues by NRC and industry initiatives.

In 1989, AEOD issued 7 special studies, 9 engineering evaluations, and 25 technical review reports analyzing safety issues that arose from study of nuclear plant operating experience. These studies provided the basis for three information notices advising licensees of deficiencies in electrical bus bars, seal cooler flow for the residual heat removal pumps, and a potential for gas binding of the high head safety injection pumps. A study of the failure of steam generator isolation check valves identified an unanalyzed condition that contributed to both an NRC information notice and a generic letter. A report on broken lifting beam bolts in the high-pressure coolant injection Terry turbine provided information for an NRC bulletin on stress corrosion cracking of preloaded bolting.

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 new generic requirements proposed by the NRC staff. This report summarizes the above activities, with the exception of those of the Technical Training Center, which are summarized in a separate report.

The office 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.
- Development and implementation of the agency program on reactor performance indicators for use by senior management.
- Development and implementation of diagnostic evaluations of licensee performance and direction of the diagnostic evaluation teams.
- Development of policy, program requirements, and procedures for NRC incident investigations of significant operational events.
- Identification of operational data needed to support safety analyses, and development of agency-wide reporting of operational data and the methods and systems to retrieve these data.
- Development of a coordinated system for feedback of operational safety information to NRC offices, licensees, and other organizations, as appropriate, and preparation of the quarterly Abnormal Occurrence Report to Congress.
- Development, in consultation with other NRC offices, of the NRC policy for response to incidents and emergencies.
- Development of an agency-wide technical qualification 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 other information reported to the Operations Center to ensure the proper NRC response to reported events.
- Acting as a focal point for coordination of 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 1989 AEOD Annual Report is published in two separate parts: *Power Reactors* and *Nonreactors*. The report on power reactors, Vol. 4, 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's Operations Center. The report includes appendices as follows: Appendix A contains data to support the section on operational experience, Appendix B lists and summarizes 1989 abnormal occurrences, Appendix C lists AEOD reports issued in 1989, Appendix D lists AEOD reports issued from 1980 through 1988, Appendix E presents the status of outstanding recommendations included in AEOD studies, and Appendix F presents

the status of NRC staff actions for three reactor events since 1985 that were investigated by NRC Incident Investigation Teams (IITs). The report on nonreactors, Vol. 4, No. 2, presents a review of the events and

concerns during 1989 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations.

2 Performance Indicators

One aspect of the Commission's efforts to monitor the performance of nuclear power plant licensees is the Performance Indicator (PI) Program, which is under the direction of AEOD.

AEOD currently monitors industry-wide data on eight PIs and evaluates the data to determine performance trends. The eight PIs are (1) the number of unplanned automatic reactor scrams (trips) while a reactor is critical, (2) the number of selected safety system actuations (SSAs), (3) the number of significant events, (4) the number of safety system failures (SSFs), (5) forced outage rates (FORs), (6) the number of equipment-forced outages per 1,000 critical hours, (7) the collective radiation exposure per plant, and (8) cause code trends. The AEOD staff quarterly provides plant-specific data for these eight PIs to the Commission and NRC senior management. The PI reports are also placed in the NRC Public Document Room. Beginning with the issuance of the fourth-quarter calendar year 1989 PI report, the staff will routinely provide plant-specific information to licensee management. Insights provided by trending seven of the eight indicators for the years from 1985 through 1989 are discussed in this section (cause code trends are not included because industry-wide averages are not calculated for this indicator). In calculating industry-wide averages, plants that were in extended shutdowns have been excluded for various calendar quarters. (Table 2.1 provides a list of these plants and the calendar quarters excluded.) Appendix A-1 provides definitions for the eight PIs.

The PI program is a single, coordinated, overall NRC program that provides an additional view of operational performance and enhances the NRC's ability to recognize areas of changing safety performance of operating plants. However, it is only a tool and is to be used in conjunction with other tools, such as the results of routine and special inspections and the Systematic Assessment of Licensee Performance (SALP) Program, for providing input to NRC management decisions regarding the need to adjust plant-specific regulatory programs. PIs for a given plant should be viewed as a set. When viewed as a set, PIs provide an additional input for assessing changes in plant operational performance. PIs focus attention on the need to assess and understand underlying causes of changes based on other available information.

Table 2.1 Plants Excluded From Averages in Performance Indicator Data Because of Extended Outages

Plant	Quarters Excluded
Browns Ferry 1	85-2 through 89-4
Browns Ferry 2	85-4 through 89-4
Browns Ferry 3	85-2 through 89-4
Davis Besse	85-3 through 86-4
Fort St. Vrain	86-1 through 87-1, 88-3, 88-4 & 89-3 through 89-4
Palisades	86-3 through 87-1
Peach Bottom 2	87-2 through 89-2
Peach Bottom 3	87-2 through 89-4
Pilgrim	88-1 through 88-4
Rancho Seco	86-1 through 88-2 & 89-2 through 89-4
Seabrook	85-2 through 89-2
Sequoyah 1	86-1 through 88-4
Sequoyah 2	85-4 through 88-2
Shoreham	85-1 through 85-2 & 85-4 through 89-4

Automatic Reactor Trips

AEOD trends the number of unplanned automatic reactor trips (scrams) per plant while a reactor is critical. Reactor trips are challenges to reactor safety systems and can result from initiating events that range from relatively minor incidents to events that are precursors of accidents.

Figure 2.1 shows the trend in the average number of automatic reactor trips per plant by year. As shown, the average number of trips per plant has steadily decreased for the last 4 years. The decrease in the number of trips was less from 1988 to 1989 than from 1987 compared to 1988. A review of the progress made by the nuclear industry in reducing reactor trips is discussed in NUREG-1275, Vol. 5, and NUREG-1275, Vol. 5, Addendum, which were issued by AEOD in March 1989 and August 1989, respectively (see Appendix C).

The PI trip data for each plant for 1988 and 1989 are provided in Table A-1.1 of Appendix A-1. In addition, tables in Appendix A-2 contain data for 1985 through 1989 on all trips (Table A-2.1), initiating systems

Figure 2.1
Annual Industry Averages, 1985–1989
(Excluding Plants in Extended Shutdown)

† These updated data for some years preceding 1989 differ slightly from previously published data as a result of refinements in data quality.

Figure 2.1 (cont.)
Annual Industry Averages, 1985–1989
(Excluding Plants in Extended Shutdown)

(Table A-2.2), activities in progress at the time of the trips (Table A-2.3), trip causes (Table A-2.4), and trip signals (Table A-2.5). Similar data for new plants are also provided separately in Appendix A-3. In addition, Section 6.2.1 of this report discusses 1989 industry experience for all unplanned reactor trips.

Safety System Actuations

AEOD monitors two types of SSAs: actuations of the emergency core cooling system and actuations of the emergency ac electrical system (actuations caused by low voltage on a safety bus). Monitoring the number of these SSAs provides insights into the more significant challenges to plant safety systems.

Figure 2.1 shows the trend in the average number of SSAs per plant by year. As shown, the number of SSAs has generally decreased for the last 4 years.

Table A-1.2 of Appendix A-1 contains plant-specific data for this PI for 1988 and 1989. Table A-2.6 of Appendix A-2 contains data for 1985 through 1989 on all engineered safety feature (ESF) actuations of which the PI data is a subset. New plant data on ESF actuations are also provided separately in Appendix A-3. The industry-wide data on all ESF actuations occurring in 1989 are also discussed in Section 6.2.2 of this report.

Significant Events

Significant events are those events that the NRC staff identifies as meeting certain selection criteria discussed in this paragraph. The screening process includes a daily review and discussion of selected operating reactor events. Significant events normally 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 a major transient itself; (3) a degradation of fuel integrity, 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 limits of TS; and (7) other events that are considered significant. Table A-1.3 of Appendix A-1 provides the list of significant events for 1989. Table A-1.4 of Appendix A-1 provides plant-specific data for this PI for 1988 and 1989.

Figure 2.1 shows the trend in the average number of significant events per plant by year. As shown, there was a steady decrease in significant events from 1985 through 1987. From 1987 through 1989, the number of significant events has been relatively constant.

Safety System Failures

This indicator includes any event or condition that could prevent the fulfillment of the safety function of structures or systems. Twenty-six safety systems, subsystems, and components are monitored for this indicator. Failures in these systems, subsystems, or components are generally found during testing, special inspections, and engineering design reviews. For a system that consists of multiple redundant subsystems or trains, failure of all trains constitutes a safety system failure. System failures can be indicative of a plant's readiness to respond to anticipated events and postulated accidents. Table A-1.5 of Appendix A-1 provides the plant-specific data for SSFs for 1988 and 1989.

Figure 2.1 shows the trend in the average number of SSFs per plant by year. In 1987, AEOD began to determine the number of SSFs by using a more detailed review of operating events than used previously. This improved procedure for determining the number of SSFs resulted in a higher average rate for SSFs per plant for 1987 through 1989 than for previous years. Thus, comparing only the rate between 1987 and 1989 is appropriate. During this period, the average number of failures per plant decreased from 4.09 to 3.37.

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 (i.e., generator on-line 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 the weekend following the discovery of an off-normal condition. The trend in forced outage rate can provide a perspective on overall plant performance. Plant-specific data for 1988 and 1989 are provided in Table A-1.6 of Appendix A-1.

Figure 2.1 shows the trend in the average forced outage rate per plant by year. As shown, the forced outage rate was relatively constant at about 11 percent for the years 1985 through 1987. There was a notable decrease in 1988; however, the rate increased again in 1989 to about 10 percent.

Equipment-Forced Outages per One Thousand Critical Hours

The equipment-forced outage (EFO) rate is a measure of the number of forced outages caused by equipment failures per 1,000 critical hours. An EFO rate is the inverse of the mean time between forced outages caused by equipment failures. AEOD monitors the EFO rate as an indicator that can provide a perspective on the effects of equipment problems and overall plant performance. Table A-1.7 of Appendix A-1 contains

the data for this PI for 1988 and 1989 for each plant. Table A-1.8 of Appendix A-1 contains the number of critical hours for 1988 and 1989 for each plant.

Figure 2.1 shows the trend in the average EFO rate per 1,000 critical hours per plant by year. As shown, the trend in the average value of the EFO rate declined from 1986 to 1988 and remained relatively the same in 1989.

Collective Radiation Exposure

The collective radiation exposure (man-rem) per plant that AEOD monitors is the total radiation dose reported at a site for the year divided by the total number of units at the site. Some items that may contribute to this indicator are the licensee's efforts to have an effective maintenance program, good design, experienced and well-trained personnel, a good water-chemistry control program, effective decontamination and cleanup practices, good radiation exposure control programs, and an alert health physics staff. Table A-1.9 of Appendix A-1 contains the data provided by the Institute of Nuclear Power Operations (INPO) for each plant from the first quarter of 1988 to the last quarter of 1989. (INPO routinely monitors this radiation exposure data and provides it to AEOD quarterly. To avoid

unnecessary duplication of effort, AEOD uses the data supplied by INPO for the PI program.)

Figure 2.1 shows that the previous downward trend in collective radiation exposure continued in 1989.

Cause Code Trends

Cause codes are intended to identify possible programmatic deficiencies. Cause code trend data are developed using the Sequence Coding and Search System (SCSS) database. This database includes the cause(s) associated with each licensee event report. The cause code data for each plant for the PI program from the first quarter of 1988 through the last quarter of 1989 are shown in Tables A-1.10 through A-1.19. 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) [maintenance problems are further divided into sub-categories of maintenance personnel error (Table A-1.14), personnel error during surveillance or testing (Table A-1.15), maintenance equipment failure (Table A-1.16), and potential maintenance problem (Table A-1.17)]; design, construction, installation, or fabrication problems (Table A-1.18); and equipment (electronic piece-part or environmentally related) failures (Table A-1.19).

3 Development of Performance Indicators

During 1989, the AEOD staff continued its efforts on new performance indicators to (1) incorporate cause codes into the PI report as a new indicator; (2) evaluate the usefulness of corrective actions for measuring the effectiveness of licensee response to problems; (3) develop an indicator of maintenance performance; (4) develop, in conjunction with the NRC Office of Nuclear Regulatory Research (RES), a system unavailability indicator, as well as indirect programmatic performance indicators; and (5) participate in the International Atomic Energy Agency (IAEA) program for the development of performance indicators. Sections 3.1 through 3.5 provide additional details about these tasks.

3.1 Cause Codes

The staff proposal for the use of cause codes as a performance indicator was documented in SECY-89-046 and SECY-89-211. Through staff requirements memoranda dated March 15, 1989, and August 10, 1989, the Commission approved cause code trends as a new performance indicator. The PI report incorporated cause code trends beginning with the second quarter 1989 report. Additional information regarding cause code trends as a performance indicator is contained in the background material of Appendix A-1.

3.2 Corrective Action Trends

Corrective action coding was incorporated into the ongoing Sequence Coding and Search System process beginning January 1, 1989, as a part of the cause code trend development program. Corrective actions appear to provide insights into the different types of problems occurring at nuclear plants and the effectiveness of licensee response to those problems. Evaluation of the usefulness of corrective actions in combination with cause codes for monitoring and improving plant performance began in the third quarter of 1989 and is continuing. Corrective action trends are not being considered as a candidate indicator but as supplemental information to cause codes.

3.3 Development of a Maintenance Performance Indicator

During 1989, the AEOD staff continued the development of an approach for monitoring maintenance effectiveness. The development of and previous validation efforts for a proposed maintenance indicator on the basis of component failure experience have been documented in two technical reports: AEOD/S804B,

"Application of the NPRDS [Nuclear Plant Reliability Data System] for Maintenance Effectiveness Monitoring," issued in January 1989, and the EG&G Idaho, Inc., report, "Maintenance Effectiveness Indicator," issued in October 1989. The former report covered the construction and validation of a maintenance indicator on the basis of failure experiences at boiling water reactors. The latter report extended the approach and validation to pressurized water reactors. In June 1989, the AEOD staff also initiated a "Demonstration Project" with selected utilities in order to obtain licensees' perspectives on the proposed maintenance indicator. The technical work associated with the Demonstration Project was accomplished from September 1989 through March 1990, and a report on the Demonstration Project is scheduled to be issued in 1990. The Demonstration Project is discussed further in Section 3.3.1.

The proposed maintenance indicator uses a methodology that evaluates the NPRDS component failure data for preselected components and signals any increase in the failure frequency that exceeds a predetermined value. These periods of increased failure frequency are trended over time to provide an indication of the overall effectiveness of the licensees' maintenance programs. Figure 3.1 shows a trend for the indicator for a selected plant. The trend in the indicator is examined over several fuel cycles, and periods of peak maintenance activity that coincide with outages are noted for use in evaluating the trend.

3.3.1 Maintenance Indicator Demonstration Project

In response to direction from the Commission in June 1989, the NRC staff (represented by AEOD) initiated a Demonstration Project for the development of maintenance performance indicators. The staff identified candidate utilities on the basis of characteristics such as nuclear steam supply system vendor design, plant age and power rating, utility's organizational size or number of plants operated, and plant location. Through the coordination of the Nuclear Management and Resources Council (NUMARC), six utilities agreed to participate: Commonwealth Edison Company, Duke Power Company, Northeast Utilities, Rochester Gas & Electric Corporation, Southern California Edison Company, and Systems Energy Resources, Incorporated. These industry participants agreed to a project that was limited in scope to the review of the NRC staff's proposed maintenance effectiveness indicator.

Figure 3.1
Maintenance Indicator
(Trend)

The Demonstration Project included the following elements: an NRC presentation on the proposed indicator to the utilities; an NRC plant-specific analysis of component failure records from which the proposed indicator was used to develop a representation of maintenance effectiveness; data review and analysis by the participating utilities; analysis by the Institute of Nuclear Power Operations (INPO); substantive discussions of maintenance management and monitoring techniques with plant staffs; discussions of further development activities by the AEOD staff; and working discussions by the project members to formulate results. Meetings with individual utilities generally involved their maintenance managers and the performance or reliability assessment groups. These meetings, which typically lasted 1–1/2 days, followed the same basic agenda. The project group was well rounded, with representation from utility maintenance, operations, licensing, and performance or reliability assessment operations.

The Demonstration Project participants worked to achieve consensus on issues associated with a performance indicator to monitor the effectiveness of maintenance at domestic nuclear power plants. The AEOD staff believes that some technical consensus was reached on various methods to improve the NRC staff's proposed indicator. These areas of consensus included changes to the indicator construction and calculational techniques, the identification of potential improvements to the NPRDS and, to some extent, use of the indicator.

The NRC staff's proposed indicator is programmatic by design and is premised on a broad view of maintenance as outlined in the NRC Policy Statement and proposed rule* on maintenance and industry guidelines. However, this view of maintenance and the current configuration of the indicator do not correspond in a practical way with traditional maintenance line organizations at the sites. Variations of the indicator to make it more useful for plant staffs were discussed, including component types consistent with reliability-centered maintenance programs and calculating the changes in failures by component type as well as by system. However, the fundamental differences that existed in the definitions of maintenance resulted in disagreement on the characterization of the indicator as a "maintenance" indicator.

*The NRC defined maintenance as the aggregate of those planned and systematic actions required to preserve and promptly restore the operability, reliability, and availability of, or to prevent the failure of, plant structures, systems, and components. The scope of the rule was to cover all systems, including those in the balance of plant. This definition and scope are considerably larger than the traditional definition of maintenance and include many activities outside the scope of a typical plant maintenance manager (53 FR 47822; November 28, 1988).

Potential changes to the indicator to address these concerns were also discussed, including both the indicator construction and calculational techniques.

Overall, the Demonstration Project achieved successful communications between the NRC and industry on their perspectives regarding maintenance monitoring and use of a maintenance effective indicator and stimulated the interest of utilities in this area at the senior management and working levels.

3.4 Development of Other Potential Performance Indicators

During 1989, RES in conjunction with AEOD continued to evaluate other potential candidate indicators, such as safety system unavailability indicators, and indirect programmatic indicators.

3.4.1 Development of a Safety System Unavailability Indicator

RES continued to evaluate candidate indicators of the unavailability of selected, risk-important safety systems. The candidate indicators are approximate measures of the unavailability that selected safety systems at a plant have experienced during recent calendar quarters.

Research performed for the NRC by Brookhaven National Laboratory and Science Applications International Corporation included the analysis of historical data on a few selected safety systems (such as the auxiliary feedwater system) collected from 10 units at 6 sites. The results of this research showed that operational data analyzed in this way can provide indications of high unavailability of selected safety systems. These data, however, are not routinely reported to the NRC. Therefore, research to evaluate the feasibility of alternative data sources is continuing.

3.4.2 Development of Indirect Programmatic Indicators

The objective of this effort is to identify indicators that can provide early signals of safety performance on the basis of licensee technical support programs and organizational effectiveness. During 1989, work was performed in the following areas:

Organizational Factors

A consortium of universities led by the University of Minnesota developed a conceptual model linking organizational factors to safety and tested an initial set of organizational performance indicators. The initial study showed that it may be possible to develop useful indicators. During 1990, candidate indicators will be refined and further evaluated.

Industry-Based Indicators

Communications Technology Applications, Inc., conducted a feasibility analysis that demonstrated a potential for transferring established safety performance indicators from other industries to the nuclear industry. The chemical processing industry was selected as an initial test case because of its analogies to the nuclear industry, its long history of the use of performance indicators, and the availability of data on those indicators (NUREG/CR-5568). During 1990, indicators transferable to nuclear environments may be selected for validation.

Licensee Technical Support Programs Such As Training and Maintenance

Science Applications International Corporation (SAIC) performed qualitative and quantitative analyses to test the relationships to safety of unintended engineering safety feature actuations during testing and maintenance. In addition, SAIC performed analyses to determine whether trending the daily average loss of generation at a plant (maximum dependable capacity minus average daily power) could be used as an indicator of overall safety performance. A plant performance characterization matrix was developed on the basis of daily power loss data and was initially tested with data from seven plants. During 1990, the matrix will be refined and further evaluated.

3.5 International Atomic Energy Agency Performance Indicator Development Program

Since 1985, the International Atomic Energy Agency (IAEA) has been involved in the development and use of performance indicators as part of its Operational Safety Indicator Program. Since 1986, AEOD has provided expert consultants to the IAEA for the development of indicators, as well as evaluations by the Operational Safety Review Team and the Assessment of Safety-Significant Events Team.

Following one of the recommendations from an IAEA workshop in September 1988, the IAEA convened a Consultants Meeting in July 1989 to assess different methodologies that could be used to develop risk-related indicators of safety system unavailability. Attendees of the meeting also discussed and suggested a

general structure of risk-based indicators and identified areas where further development is needed.

The report of the Consultants Meeting was one of the topics discussed at the Technical Committee Meeting (TCM) on the "Use of Numerical Indicators To Enhance Nuclear Power Plant Safety Performance" that was held in Vienna, Austria, in September 1989. AEOD staff participated in this meeting. Committee members also reviewed the progress in the development and use of performance indicators achieved by INPO, together with the Union of Producers and Distributors of Electrical Energy (UNIPED), and recognized these indicators as the most appropriate indicators for international use.* At the same time, some concerns were raised about the relation to safety of some of the INPO/UNIPED indicators. However, the committee members believed that the existing set of performance indicators partly and indirectly covers many of the activities associated with operational safety, although that relationship has not yet been defined.

Committee members also concluded that safety indicators could be developed that would provide early warning of declining safety. It was agreed that such indicators might be developed either from trend analysis or by a probabilistic safety analysis (PSA) approach, that they are plant specific, and that they should not be confused with performance indicators. Committee members also recognized that considerable work is necessary to develop indicators of this kind and that adequate validation is essential before they could be used as a "prevention tool."

With respect to the worldwide interest in the development of such indicators, committee members presented a number of recommendations to the IAEA. One of the recommendations was that the IAEA should convene an Advisory Group Meeting during 1990 to (1) take a closer look into the needs and possible ways of developing plant-specific safety indicators in selected areas and how to validate these indicators, (2) determine how to use tools such as PSA in the development of these indicators, and (3) define in more detail IAEA's role in this area.

*These indicators are also to be used by members of the World Association of Nuclear Operators (WANO).

4 Abnormal Occurrence Reporting

The Office for the Analysis and Evaluation of Operational Data (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 incident, a recurring event, a generic concern, or a series of incidents that the Commission determines is significant from the standpoint of the public health or safety.

Four quarterly AO reports to Congress were published during calendar year (CY) 1989. These reports were for the third and fourth quarters of CY 1988 (designated NUREG-0090, Vol. 11, No. 3, and Vol. 11, No. 4, respectively) and the first and second quarters of CY 1989 (NUREG-0090, Vol. 12, No. 1, and Vol. 12,

No. 2, respectively). The third and fourth quarter reports for CY 1989 (designated NUREG-0090, Vol. 12, No. 3, and Vol. 12, No. 4, respectively) were published during the first quarter of CY 1990.

During CY 1989, four AOs occurred at nuclear power plants. Figure 4.1 shows the number of AOs occurring at nuclear power plants for each calendar year since 1985, and Figure 4.2 shows the history of AOs on a per-plant basis. The figures show a sharp decrease in the number of AOs beginning in 1987 to a level that has stayed about the same for 1988 and 1989.

A summary of AOs at nuclear power plants for 1989 is provided in Appendix B to this report, and a summary of 1989 nonreactor AOs (at both NRC and Agreement State licensees' facilities) is provided in Appendix C to the AEOD annual report on nonreactors (NUREG-1272, Vol. 4, No. 2, *Nonreactors*).

Figure 4.1
U.S. Nuclear Power Plants
Abnormal Occurrences per Year

Figure 4.2
U.S. Nuclear Power Plants
Abnormal Occurrences per Plant by Year

5 Data From the Nuclear Regulatory Commission's Operations Center for 1989

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

The Operations Center received 3,092 notifications in 1989 from licensees, primarily of nuclear power plants, concerning events that must be reported under NRC's prompt reporting requirements. A small subset of these notifications involved events that a licensee classified as 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 to be classified as an emergency.

Table 5.1 shows the number of events classified under the 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 actual or potential substantial degradation of plant safety. For comparison, a "Site Area Emergency" and a "General Emergency" (should they occur) would indicate major failures of systems required for public safety or events with the potential for a major offsite radiological release. Exercises are held periodically to ensure that NRC and licensee response organizations are proficient in dealing with these types of emergencies. In 1989, NRC headquarters and regional offices participated in emergency planning exercises with the South Texas 1 (April 26, 1989), Crystal River 3 (June 21, 1989), Millstone 1 (October 4, 1989), and Browns Ferry 2 (November 1, 1989) power plants. The photograph on this page shows participants in a typical exercise as they evaluate plant status and licensee actions to determine required NRC response and possible guidance to State and local agencies.

Participants in a Typical Emergency Response Exercise
in NRC's Operations Center in Bethesda, Maryland

**Table 5.1 Classification of Events Under
Emergency Plans 1985-1989**

	1985	1986	1987	1988	1989
Unusual Event	312	209	231	212	197
Alert	11	9	9	6	13
Site Area Emergency	0	0	0	0	0
General Emergency	0	0	0	0	0

The number of event reports received by the Operations Center in 1989 is categorized by day, time, and month in Figures 5.1 through 5.3, respectively. The data in the figures represent reports of events related to power reactors and the occasional reports of events related to research reactors, fuel facilities, transportation and materials, and hospitals, as well as reports from other agencies such as the Department of Energy, the Federal Emergency Management Agency, and the Department of Transportation. The maximum and the average numbers of events per day reported to the Operations Center were relatively low on weekends and increased during the work week. More events are reported in the afternoon and early evening than are reported at night or in the morning. The afternoon-early evening peak may be explained by a combination

of the following factors: (1) many reportable events are the result of maintenance or surveillance activities conducted during normal working hours, (2) a greater number of plants are in the eastern and central time zones than in the mountain and Pacific time zones (about 10 to 1), and (3) the length of time allowed for reporting an event to the NRC Operations Center can be up to 4 hours for non-emergencies. The licensees within Region I reported the highest number of events, followed by Regions II, III, IV, and V, in that order. This order approximately reflects the distribution of licensees within the regions (the number of operating nuclear plants in the regions is 30, 33, 28, 9, and 10 in Regions I through V, respectively).

Table 5.2 provides information on the "Alert" event at a fuel facility that was reported in 1989 and the "Alert" events at power facilities that were reported in 1989.

Actions taken by the the NRC Headquarters Operations Officer (HOO) in response to these event notifications ranged from a computer or log entry, followed by appropriate notifications, to establishing emergency conference calls between 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. However, in 1989 none of the event notifications resulted in activation of this plan.

Table 5.2 "Alert" Events Reported in 1989 at a Fuel Facility and at Power Facilities

Name of Facility	Event No.	Date	Description	Duration
Fuel Facility				
Sequoyah Fuels	15704	4/24/89	Hydrogen fluoride leak	6 minutes
Power Facilities				
Braidwood 2	14833	2/23/89	Loss of shutdown cooling while in cold shutdown	28 minutes
North Anna 1	14850	2/25/89	Steam generator (SG) tube leak after reactor trip at 76% power	6 hours 55 minutes
McGuire 1	14963	3/7/89	SG tube rupture at 100% power	19 hours 30 minutes
Dresden	15117	3/25/89	Loss of emergency core cooling system annunciators while at 97% power	5 minutes
Brunswick 2	15461	4/27/89	Nuclear service water pump fire while at 100% power	3 minutes
Clinton	15761	6/1/89	Recirculation pump seal failure with leak greater than 50 gpm while at 44% power	4 minutes
Crystal River 3	15886	6/16/89	Loss of offsite power while at 12% power	3 hours 58 minutes
Zion 2	15950	6/24/89	Loss of control room annunciators while at 99% power	35 minutes
Brunswick 2	16655	9/21/89	Loss of turbine/generator and balance-of-plant annunciators	2 hours 8 minutes
Harris	16805	10/9/89	Generator fire and release of flammable gas; reactor trip from 100% power	3 hours 8 minutes
Braidwood 1	17235	12/1/89	Loss of reactor coolant system inventory via residual heat removal relief valve while in cold shutdown	3 hours 58 minutes
Brunswick 2	17411	12/22/89	Loss of audible annunciators while in refueling	1 hour 7 minutes

Figure 5.1
1989 Operations Center Summary
Average and Maximum Number of Events per Day

Figure 5.2
1989 Operations Center Summary
Average Number of Events by Time of Day

Figure 5.3
1989 Operations Center Summary
Number of Events per Month

6 Operating Experience Databases

The AEOD staff analyzes and evaluates the operational experience of nuclear power plants from the following major data sources: 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 voluntary reports of component failures submitted for the Nuclear Plant Reliability Data System, which is managed by the Institute of Nuclear Power Operations. In addition, AEOD uses plant operating data found in the licensees' monthly operating reports to provide the context for event analysis and as a source of data for various programs. AEOD also uses data on foreign events in comparing and studying reactor operational experience.

The primary source of data on operational events is the licensee event report (LER). In the early 1980s, a major effort was devoted to preparing the rule (10 CFR 50.73) governing the content and submission of LERs. The rule clarified reporting requirements and established a more uniform threshold for event reporting. The threshold included consideration of infrequent events of significance to plant and public safety, as well as the more frequent events of lesser significance that are more conducive to statistical analysis and trending. Since the implementation of this rule in 1984, the events that met the threshold have provided a consistent basis for assessing the performance trends of the industry as a whole and those of individual licensees.

6.1 Growth and Use of U.S. Plant Data

Table 6.1 shows the number of LERs submitted each year from 1985 through 1989 for plants licensed to operate.

Table 6.1 LERs Submitted by Year*

Year	No. of LERs	No. of Units	LERs per Unit
1985	3002	96	31
1986	2866	105	27
1987	2919	110	27
1988	2500	111	23
1989	2362	113	21

*Plants indefinitely or permanently shut down, that is, Dresden Unit 1, Humboldt Bay, Three Mile Island Unit 2, and LaCrosse, are excluded from this table.

AEOD uses the Sequence Coding and Search System (SCSS) for storage and retrieval of LER data. This system, developed in the early 1980s and maintained

under contract at the Oak Ridge National Laboratory, Oak Ridge, Tennessee, contains, on an average, 150 related pieces of data for each LER submitted since 1980. Its primary purpose is to facilitate the storage and retrieval of information about each event (e.g., causal and time aspects of occurrences within the event sequence).

In 1989, AEOD used the LER information from the SCSS database to support certain NRC activities, for example, plant diagnostic evaluations, NRC senior management meetings, and those pertaining to the maintenance performance indicator and a cause code performance indicator. The SCSS database is a primary source of information for AEOD studies. In addition, 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 at the Idaho National Engineering Laboratory (INEL) to support studies for specific kinds of events and to support the NRC Performance Indicator Program. From the INEL Trends and Patterns database, AEOD obtained the performance indicator data for reactor trips, safety system actuations, and safety system failures throughout 1989. The database was also used to obtain performance indicator data for the forced outage rate and equipment-forced outages per 1,000 critical hours. In addition, AEOD used the INEL databases to support the preparation of Commission site visit briefing packages, special studies, and evaluations of selected plants.

6.2 Analysis of Licensee Event Reports Submitted During 1989

As seen in Table 6.1, the number of LERs per plant has generally decreased each year. This decrease is consistent with the reductions of reactor trips, significant events, loss-of-safety-system occurrences, and so forth, which are discussed in other sections of this report.

Analysis of the 2,362 LERs submitted in 1989 indicates a wide range in the number of LERs submitted by each licensee (from 3 LERs to 59 LERs). The safety significance of each LER is assessed by engineering reviews. Variations in LER counts from plant to plant can result from numerous factors, only one of which is an actual difference in safety performance. Thus, the number of LERs submitted by a plant should not be viewed as a measure of the plant's safety performance.

Table 6.2 shows the percentage of 1989 LERs in terms of the individual 10 CFR 50.73 reporting requirements. In addition to reports submitted pursuant to 10 CFR 50.73 requirements, about 9 percent of the total number of LERs were submitted pursuant to other requirements, such as 10 CFR Part 21 or 10 CFR 50.36.

As shown in Table 6.2, a large number of reports was associated with a condition prohibited by Technical Specifications (TSs). Examples of reportable items in this category include a missed surveillance test or, to a much lesser extent, the completion of a plant shutdown

because of the unavailability of required safety components. Another equally large number of reports was associated with engineered safety feature (ESF) actuations (including reactor trips). The next two most frequent LER categories concerned (1) events that did result or could have resulted in a loss of safety function at the system level and (2) unanalyzed conditions.

Reactor trips, ESF actuations, and new plant (plants licensed within the last 2 years) experience in 1989 are discussed in Sections 6.2.1, 6.2.2, and 6.2.3, respectively.

Table 6.2 Percentage of LERs Submitted in 1989 According to 10 CFR 50.73 Requirements

CFR Title 10 Section	Requirement	Percentage of Reports* (Approximate)
50.73(a)(2)(i)	TS shutdown or TS violation	39
50.73(a)(2)(iv)	ESF actuation (including reactor trips)	39
50.73(a)(2)(v)	Real or potential loss of safety system	10
50.73(a)(2)(ii)	Unanalyzed conditions	9
50.73(a)(2)(vii)	Failures in multiple systems	3
50.73(a)(2)(x)	Internal threat	Less than 1
50.73(a)(2)(iii)	External threat	Less than 1
50.73(a)(2)(viii)(B)	Liquid effluent radioactive releases	0
50.73(a)(2)(viii)(A)	Airborne radioactive releases	0

*Percentages may sum to more than 100 percent because an LER can be submitted under more than one reporting requirement and reflects the licensee's categorization of LERs.

6.2.1 Reactor Trip Experience

Table 6.3 shows the total number of unplanned reactor trips (i.e., the performance indicator data plus unplanned manual trips while the reactor is critical) and the reactor trip rate (reactor trips per 1,000 critical hours) at U.S. light water reactors (LWRs) for the years 1985 through 1989.

Table 6.3 Overall LWR Reactor Trip Statistics by Year

Year	Total No. of Reactor Trips	Reactor Trips per 1,000 Critical Hours
1985	525	0.96
1986	457	0.81
1987	406	0.65
1988	274	0.41
1989	244	0.36

The total of 244 reactor trips in 1989 is an improvement over the total of 274 trips in 1988. The number of

reactor trips per year has declined continuously since 1985, one year after initial efforts to systematically track the trend in U.S. LWR unplanned reactor trips began. This trend is in large part attributable to reductions in the number of trips initiated by problems in the main feedwater system, an area that has received a great deal of attention under the reactor trip reduction programs of the nuclear steam supply system (NSSS) vendors. The overall reactor trip rate decreased by about 11 percent from 1988 to 1989. The progress in reactor trip reduction made by the nuclear power industry is discussed in two AEOD reports issued during 1989 (see Appendix C): NUREG-1275, Vol. 5, published in March 1989, covered the period from January 1, 1984, to December 31, 1987. NUREG-1275, Vol. 5, Addendum, published in August 1989, covered the period from January 1, 1988, to March 31, 1989.

The reactor trip rates at new units are typically higher than the rates for mature units. Table 6.4 provides the unplanned reactor trip rates for both automatic and manual trips and for both mature and new units, for the years 1985 through 1989.

**Table 6.4 Unplanned Reactor Trip Rates for Mature and New Units by Year
(Reactor Trips per 1,000 Critical Hours)**

	1985	1986	1987	1988	1989
Overall industry	0.96	0.81	0.65	0.41	0.36
New units	2.37	1.70	1.91	0.81	0.94
Automatic	2.22	1.56	1.48	0.63	0.87
Manual	0.15	0.14	0.43	0.18	0.07
Mature units	0.77	0.70	0.49	0.37	0.34
Automatic	0.68	0.61	0.43	0.31	0.27
Manual	0.09	0.09	0.05	0.06	0.07

In general, units in their first 2 years of operation after receipt of a low-power license are considered new units. Table A-2.1 of Appendix A-2 contains a summary of the trip statistics for individual units for the years 1985 through 1989.

Appendix A-2 also contains summary tables that list statistics for unplanned trips for the population of mature units for each of the major NSSS vendors. Statistics are provided for unplanned reactor trips as a function of the initiating system (Table A-2.2), activity at the time of the trip (Table A-2.3), trip cause (Table A-2.4), and trip signal (Table A-2.5).

6.2.2 Engineered Safety Feature Actuation Experience

During 1989, about 900 events involved the actuation of engineered safety features (ESFs). Table 6.5 summarizes the data for the four principal NSSS vendors and the overall industry averages for 1985 through 1989. The mean number of ESF events decreased by about 8 percent from 1988 to 1989. Plant systems that are designated ESFs vary considerably among the NSSS

vendors; in addition, many ESF events are spurious (i.e., not warranted by a safety condition). Therefore, the values in Table 6.5 should not be used to imply differences in safety performance among the NSSS plant types.

Figure 6.1 shows the overall industry trend plot for ESF events since 1985. The "Total" plot includes all ESF events, except those involving only a reactor trip; it includes events in which a reactor trip occurred in combination with actuation of another ESF. 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 regardless of whether the signal was warranted by a safety condition; all other events are considered "unneeded." The latter are unnecessary challenges to the safety systems. The downward trend in the number of ESF events cannot be attributed to any one factor. Rather, improvements have been across a wide variety of systems, plant activities (i.e., normal operations, testing, maintenance), and causes (i.e., equipment problems, personnel errors, procedural problems).

Table 6.5 Average Number of ESF Events per Unit by Year

	ESF Events per Unit*				
	1985	1986	1987	1988	1989
Westinghouse (W)	12.4	8.8	9.1	6.6	6.4
Combustion Engineering (CE)	17.9	9.2	8.9	5.9	3.6
Babcock & Wilcox (B&W)	7.8	2.6	2.5	8.3	7.5
General Electric (GE)	25.2	22.9	19.1	14.0	12.9
Industry Average	16.7	12.7	12.1	9.0	8.2

*Events with only reactor protection system actuations are not included.

Figure 6.1
Total and Valid Engineered Safety Feature Actuations
Overall Industry Trends

The emergency power (EP) and emergency core cooling system (ECCS) actuation plots in Figure 6.2 are subsets of the “Total” ESF events shown in Figure 6.1. They show the trends of all unplanned actuations of these systems (i.e., actuations beyond those counted in the Performance Indicator Program). The heating, ventilation, and air conditioning (HVAC) plot shows the trend in the actuation of all HVAC systems that serve a safety function; these systems dominate all ESF events. Figure 6.3 shows selected systems involved in 1989 ESF events by NSSS vendor. The HVAC system is the predominant system involved in ESF events (the predominant components involved are radiation and toxic gas monitors). The number of HVAC system actuations has decreased by an average factor of 1.2 each year from 1985 through 1989.

Figures A-2.1 through A-2.4 of Appendix A-2 provide trend plots for each principal NSSS vendor. For General Electric (GE) plants, an additional plot is provided for the reactor water cleanup (RWC) system, which is unique to GE NSSS plants, because that system accounts for a large fraction of all ESF events.

As shown in Table 6.5, each type of NSSS vendor plant showed a decrease in the average number of ESF events per plant in 1989 as compared to 1988. The decrease is generally attributable to reductions in the rates of HVAC and RWC system actuations, while the ECCS and EP rates remained at rather low levels.

The notable differences among the NSSS vendor groups are as follows:

- The average number of ESF events at Babcock & Wilcox (B&W) NSSS plants was rather high for 1988 and 1989 compared to the previous 2 years. The high average number of ESF events in 1989 was a result of a large increase in HVAC system actuations at one plant (Arkansas Nuclear One, Unit 1), most of which occurred in the first half of the year. A combination of a new interpretation of the reporting requirements, together with problems with instrumentation (radiation and toxic gas monitors), resulted in the high number. The problem was resolved and the overall HVAC event rate for all B&W NSSS plants returned to low levels for the rest of the year. A similar combination of a new interpretation of the reporting requirements and instrumentation problems at Rancho Seco Nuclear Generating Station accounted for the high rate in 1988; the Rancho Seco problems were resolved in 1988.
- The Combustion Engineering (CE) NSSS plants experienced a significant decrease in ESF event

rates for 1989 compared to 1988, as well as for 1988 compared to 1987. The reduced number of events at two plants (San Onofre Nuclear Station, Units 1 and 2)—80 in 1987, 33 in 1988, and 13 in 1989—primarily accounted for the overall reduction in these rates. These reductions were due primarily to reductions in HVAC system actuations, principally as a result of design changes made to reduce electromagnetic interference with radiation monitors.

- GE NSSS plants accounted for more than 50 percent of the 1989 ESF events, averaging 12.9 events per plant. Forty-nine percent of these events involved the HVAC system.
- The number of ESF events at Westinghouse (W) NSSS plants remained relatively constant from 1988 to 1989, with an average of only 6.4 ESF events per plant in 1989. Again, HVAC systems were the systems involved in most of the events reported.

Table A-2.6 in Appendix A-2 lists the number of unplanned ESF events for the years 1985 through 1989. In addition, Appendix A-2 provides tables of data for the number of ESF events for each principal NSSS vendor as a function of cause (Table A-2.7), associated activity (Table A-2.8), selected system (Table A-2.9), and actuation signal (Table A-2.10).

Figure 6.4 summarizes the causes for 1989 ESF actuations for each principal NSSS vendor, and Figure 6.5 shows the ongoing activities associated with the 1989 ESF actuations for each principal NSSS vendor.

In summary, the dominant contributors to unplanned ESF actuations are associated with HVAC systems, and in particular, the two components that initiate them—radiation and toxic gas monitors.

6.2.3 New Plant Experience

As previously mentioned, plants operating in the first 2 years after receipt of an operating license are considered to be new plants. Tables A-3.1 through A-3.5 and Figures A-3.1 and A-3.2 of Appendix A-3 contain detailed data on new plant experience.

Although there is a wide range of individual performance, new plants have typically experienced higher numbers of reactor trips and ESF events in their first 2 years of operation than mature plants. Since the new plants account for less than 10 percent of the industry, the high number of events per plant only slightly affects the overall industry rate.

Figure 6.2
Engineered Safety Feature Actuations of Selected Systems
Overall Industry Trends

Figure 6.3
Selected ESF Systems 1989
by NSSS Vendor

Figure 6.4
Causes of 1989 Actuations
by NSSS Vendor

Figure 6.5
Activities: 1989 ESF Actuations
by NSSS Vendor

Four units achieved initial criticality in 1989: Limerick Generating Station, Unit 2; Vogtle Nuclear Plant, Unit 2; Seabrook Nuclear Station; and South Texas Project, Unit 2. The number of reactor trips and ESF events at these new plants was consistent with new plant data from previous years. Data on reactor trips and ESF events for these four units, and others, are provided in Appendix A-2, Tables A-2.1 and A-2.6, respectively.

6.3 Accident Sequence Precursor (ASP) Program

The Accident Sequence Precursor (ASP) Program was established at the Nuclear Operations Analysis Center at Oak Ridge National Laboratory in the summer of 1979 to provide a means of evaluating in a structured manner the significance of operational experience. Under this program, which is administered and directed by AEOD, operational experience reported by the commercial nuclear power plants in the United States was reviewed to identify and categorize precursors to potential severe core damage accidents. Under this program, evaluations are made of licensee event reports (LERs) submitted on plant problems, equipment failures, or other operational incidents that are required to be reported by the operators of the nuclear plants. Precursor events might be one of the following:

- One or more plant safety systems failing on demand or found failed during test or maintenance
- Two or more safety systems or subsystems found degraded
- Occurrence of an initiator, such as
 - A loss of offsite power
 - A loss-of-coolant accident
 - A steamline break
 - Any transient or plant upset that proceeds in an off-normal manner

Operational events for the years 1969 through 1988 have been reviewed, except those events occurring in 1982-1983. The operational events selected in the ASP Program form a unique database of historical system failures, multiple losses of redundancy, and infrequent accident initiators. These events are useful in identifying significant weaknesses in design and operation, for use in analysis of industry performance, and for use in probabilistic risk assessment-related studies.

Under the ASP Program, operational occurrences that involve portions of postulated core damage sequences are identified and evaluated. Event tree models and probabilistic risk assessment techniques are used to put the reported data in perspective for evaluation. The event trees model plant equipment that could affect, or could be used to mitigate, the event being evaluated, as well as human actions. This method allows quantitative estimates of the significance of the event in terms of a conditional core damage probability.

The breakdown of precursors in 1984 through 1988 by conditional core damage probability is shown in Figure 6.6. In 1985, there was one precursor with conditional core damage probability in the $1E-2$ range and one precursor with this probability in the $1E-3$ range. The 1985 precursor with conditional core damage probability in the range of $1E-2$ was an operational event at Davis-Besse Nuclear Power Station involving a complete loss of feedwater. The precursor with conditional core damage probability in the $1E-3$ range was an operational event at Edwin I. Hatch Nuclear Plant, Unit 1, involving a stuck-open safety relief valve and a loss of the high-pressure coolant injection system and the reactor isolation cooling system.

In 1986, there were two precursors with conditional core damage probability in the $1E-3$ range. The precursors were operational events at Catawba Nuclear Station, Unit 1 (LER 413/86-31), and at Turkey Point Plant, Unit 3 (LER 250/86-39). The event at Catawba Unit 1 was a small-break loss-of-coolant accident involving a guillotine rupture of the letdown line. The event at Turkey Point Unit 3 involved a reactor trip with a stuck-open pressurizer relief valve. There were no precursors with conditional core damage probability greater than $1E-3$ in 1987 or 1988.

In 1988, four of the seven precursors with the highest conditional core damage probability (i.e., 1988 precursors having an estimated conditional core damage probability greater than $1E-4$) involved common mode failures; another event involved potential common mode failures. These data illustrate the importance of common mode failures to reactor safety and the need for continued vigilance in the areas of maintenance, inspection, and testing of safety equipment.

The precursor evaluations can help show whether plant designs and capabilities can ensure that actual operational events will be coped with adequately. Additionally, the precursor evaluations can help show where plant reliability or response capability could be improved. The ASP Program provides a continuous, ongoing assessment of plant capabilities.

Figure 6.6
Number of Precursors and Associated
Conditional Core Damage Probabilities

6.4 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. Subsequent to the accident at Three Mile Island Nuclear Station (TMI) in 1979, international agencies also developed an Incident Reporting System (IRS) for the exchange of information on events of particular safety significance. Accordingly, AEOD continued its efforts to maintain and improve the exchange of operational experience information and ideas with foreign communities. The results of these efforts have provided valuable data for AEOD studies and valuable support for regulatory actions.

Also subsequent to the accident at TMI in 1979, AEOD contracted with the Nuclear Operations Analysis Center at Oak Ridge National Laboratory (ORNL), Oak Ridge, Tennessee, to develop a computer-based Foreign Event File (FEF). The FEF, developed in 1980, is limited to events at foreign light water reactors with power output greater than 200 MWe. The historical data from this file are used in conjunction with U.S. experience when assessing potential safety concerns.

6.4.1 Review of Foreign Events

As noted in Section 6.4, the operating experience of foreign reactors is a source of information that is important to U.S. nuclear programs. About 130 IRS reports from foreign reactors were received and reviewed during 1989. Thus, AEOD routinely reviews reports of events from foreign reactors submitted to the IRS of both the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development's Nuclear Energy Agency (NEA), as well as 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 1989 are (1) premature decomposition of boron carbide plates in spent fuel storage racks, (2) gas accumulation in the reactor vessel head area, (3) safety system failure caused by mercury-wetted relays, and (4) the failure of diesel generators to start. AEOD also developed important findings based on foreign events in several of its studies. AEOD disseminates, within the NRC, reports of selected foreign reactor events of particular interest to the U.S. program.

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

At the end of 1989, 113 of the more than 420 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 a large number of reports (about 60) to NEA's IRS. The United States and 12 other countries (Belgium, Canada, Finland, France, the Federal Republic of Germany, Italy, Japan, the Netherlands, Spain, Sweden, Switzerland, and the United Kingdom) contribute to this system. The United States, through the NEA, also participates in IAEA's IRS, which broadens the operational experience base to include all countries with nuclear power programs, except Taiwan. The reports submitted by the United States addressed various individual operational events and generic concerns identified within the U.S. program. Such reports were based primarily on information earlier communicated to U.S. plant operators by NRC information notices, bulletins, and AEOD reports (on such subjects as pipe wall thinning, service water system failures, and safety injection pipe failure).

A number of the foreign countries indicated that the U.S. reports have proved valuable to their programs. For example, at a meeting of an NEA Principal Working Group (PWG), foreign representatives cited over 25 specific U.S. experiences that resulted in some actions being taken in their country's program. These U.S. experiences included (1) unexpected pressurizer surge line movement at Trojan Nuclear Plant as a result of thermal stratification, (2) pipe wall thinning in various systems, (3) broken retaining block studs on Anchor Darling check valves, (4) inadequacies in the design of dc motor-operated valves documented in a generic report, and (5) service water system failures and degradation documented in a generic report.

AEOD also serves as the principal U.S. technical representative on operating experience to NEA's PWG-1, "Operating Experience and Human Factors." In September 1989, PWG-1 met jointly (annual meeting) with nine representatives from IAEA countries to discuss significant operating events. PWG-1 also met separately with representatives from various NEA member countries to discuss significant operating events and data. The United States cited more than 10 foreign IRS reports that were particularly relevant to understanding several incidents experienced by U.S. nuclear plants. These exchanges continue to be important to nuclear power safety programs worldwide.

6.4.3 Maintenance and Operation of the NEA IRS Database

In 1989, the NEA and the NRC, as the responsible U.S. Government agency, finalized an agreement authorizing the NRC to assume the responsibility for the management and operation of NEA's IRS database. This database contains information on nuclear power plant operating experience that has been reported to the NEA IRS Program by participating countries. This work, which is performed under a contract with ORNL, includes the processing and encoding of all IRS events received from the NEA, and the quarterly transmittal of multiple copies of a computerized update to the database to NEA for further distribution to the member countries.

6.4.4 International Nuclear Event Scale

The AEOD staff participated in several international meetings and workshops associated with the development and implementation of the International Nuclear Event Scale (INES). The scale was developed as a joint program of the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). The INES is intended to facilitate communications with the public regarding events

occurring at nuclear facilities. The scale consists of seven levels or classifications, which range from "minor anomaly" (Level 1) to "major accident" (Level 7). The classification of events into the appropriate level requires consideration of criteria associated with the offsite impact of the event, the onsite impact, and degradation of the defense-in-depth design philosophy.

Although the NRC staff has decided that implementation of the INES or a similar scale is not beneficial for the U.S. nuclear industry, the AEOD staff continues to monitor implementation of the INES in other countries as well as to participate in the IAEA/NEA programs associated with the scale. The decision by the NRC staff not to implement a numerical scale such as the INES was based on the existence of well-established systems in the United States associated with the dissemination of information regarding events at commercial nuclear facilities. Introduction of a new scale was perceived to provide little benefit while requiring major efforts by the NRC, licensees, the media, and the public, and contains the potential for an adverse effect on the existing systems for public information and emergency response by adding complexities and opportunities for confusion. Various Commission papers and memoranda originated by AEOD provide documentation of the IAEA/NEA programs and the NRC decision process on this issue.

7 Results of AEOD Studies

In 1989, AEOD did not publish any case studies but issued a number of special studies, engineering evaluations, and technical reviews in addition to routine reports. The results of some of the more significant of these studies are summarized in the following sections.

In the evaluation of operational experience, 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 (NPRDS), 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 issued by the NRC, 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 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 licensee's planned or scheduled corrective actions are adequate.

7.1 Maintenance Problems at Nuclear Plants

(Special Study Report AEOD/S901, Revision 1)

The AEOD staff obtained a measure of operating nuclear plant maintenance deficiencies by reviewing operational experience reported to the NRC from 1985 to 1988. This review focused on existing studies and reports prepared by AEOD, generic communications

issued by the NRC to licensees, feedback documents issued by INPO, and licensee event reports (LERs) issued by licensees of operating domestic nuclear plants. These documents were reviewed to identify maintenance-related problems and deficiencies, the number of plants involved, the systems and components affected, and the human and organizational errors associated with maintenance. The review attempted to ascertain if there had been a discernible trend in the area of maintenance. The study also addressed estimates of the magnitude of and trends in the cost of identified maintenance deficiencies to the commercial nuclear industry.

Of the 70 AEOD case studies, special studies, and engineering evaluations issued since 1985, the review identified 15 (about 20 percent of all study reports) that specifically addressed maintenance-related problems. The AEOD staff concluded that the maintenance-related problems found in systems, such as the service water system, the instrument air system, and other safety- and non-safety-related systems, and in components, such as motor-operated valves (MOVs), inverters, circuit breakers, and pumps, are widespread in the industry. Problems reflect deficiencies in quality control, procedures, planning, communication, and training.

Since 1985, the NRC has issued 80 bulletins and information notices specifically addressing maintenance deficiencies (approximately 20 percent of about 400 of these documents). The AEOD staff's review of these documents provided conclusions similar to those obtained by the review of the study reports; that is, problems addressed in these generic communications included deficiencies in maintenance procedures, inadequate post-maintenance testing, and maintenance problems with MOVs, emergency diesel generators, pumps, valves, relays, and so forth. Similar findings were identified in the review of generic feedback documents issued by INPO.

AEOD reports and NRC bulletins and information notices characterize the broad scope of maintenance deficiencies and the significant NRC staff effort to alert licensees to maintenance problems. Some of these documents are based on operating experience over several years; others are based on one or more safety-significant events. Simple counts of the documents cannot meaningfully be used to establish trends. To ascertain trends in reported maintenance deficiencies, the AEOD staff relied on data from operating reactors reported in LERs contained in the Sequence Coding and Search System. This effort identified a gradually improving trend in the number of maintenance-related

problems in the nuclear industry in the 4-year period beginning with 1985. It should be noted that LERs focus on safety-related equipment and systems, and hence, trends based on LERs are primarily associated with such equipment and systems.

Improving trends in the forced outage rate and equipment-forced outages and a marked improvement in the scram rate support the view that overall plant maintenance is improving.

Hence, the AEOD staff concludes that although maintenance problems continue to occur in a broad range of systems and equipment, recent operating experience points to an improving trend in safety performance that can be attributed to improvements in maintenance.

A bounded estimate for the costs related to maintenance deficiencies associated with the costs for replacement power for forced outages was developed from the Shared Information Network (SINET) database. The upper bound of this estimate shows steadily increasing costs for replacement power from 1984 to 1987, reaching an annual level of \$960 million in 1987. The upper bound for 1988 is currently projected at \$720 million per year, a decrease from 1987. The lower bound estimates are \$120 million in 1987 and a projected \$80 million in 1988.

This report provides additional insights for ongoing maintenance team inspections, as well as a basis for modification of the inspection modules. The results provide useful data on the quality of maintenance at operating plants and serve as supporting data for a regulatory guide on maintenance. Lastly, the information regarding financial matters provides additional perspective on current maintenance costs.

7.2 Problems With Oils, Greases, Solvents, and Other Chemical Materials

(Engineering Evaluation Report AEOD/E901)

The study of problems with oils, greases, solvents, and other chemical materials was prompted by an event at Sequoyah Unit 1 where a sealant was applied inside a hydraulic actuator for a diesel generator and ultimately clogged the small oil passages in the actuator. The licensee's subsequent investigation indicated that the sealant had been improperly cured. Since all the diesel generators had undergone similar maintenance about the same time period, the potential existed for a common mode failure of all emergency power. The sealant will not be used in the actuator in the future.

These types of deficiencies are generally not reported in LERs unless they cause a total system failure. Con-

sequently, most of the 26 events discussed in the report were accumulated from routine reading of inspection reports.

Three other events besides the Sequoyah event were cited in the report. At Rancho Seco, many of the connectors in the safety system actuation circuitry were cleaned in the same timeframe using a Freon-based solvent. Subsequent testing resulted in incorrect voltage readings and spurious module trips. After considerable investigation, the licensee determined that the commercial-grade solvent contained contaminants that degraded the connections. As a result of this discovery, the licensee cleaned all the affected connectors with a refined Freon-based solvent.

At North Anna, piston pin bushing extrusion in the emergency diesel generators had occurred for several years. The licensee finally identified the lube oil, which did not have a foam inhibitor, as the cause of the chronic problem. The type of lube oil was not specified by the manufacturer. The inappropriate oil is no longer stored on site.

At Farley, greases having different bases (calcium or lithium) were used to lubricate the MOVs in spite of specific warnings in the manufacturer's manual and an independent industry notification.

Several steps are involved in handling chemical materials at a plant. They include identification and purchase of the appropriate material, receipt inspection, storage control, specific identification in work orders or maintenance procedures, correct use of the material, and inspection of the completed maintenance activity. Deficiencies in these aspects of the process are discussed in the report and supported by specific operational examples.

The problems associated with handling chemical materials are largely due to personnel errors. Administrative procedures generally exist for controlling these materials. Because of the potential for common mode failure, AEOD has forwarded the report to the NRC Office of Nuclear Reactor Regulation (NRR) for consideration of development of an inspection module for specific monitoring of licensees' activities in this area.

7.3 Fires and Explosive Mixtures Resulting From Introduction of Hydrogen Into Plant Air Systems

(Engineering Evaluation Report AEOD/E902)

On January 6, 1989, hydrogen gas was inadvertently introduced into the plant air systems at the H. B. Robinson facility, resulting in two fires in the turbine building and the presence of explosive mixtures within

the auxiliary building. On learning of the event, the AEOD staff searched the LER database for similar events and found eight additional events.

In the engineering evaluation report, AEOD suggested that the information in the report and information in a previous study, AEOD Case Study C501, "Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution System in Nuclear Power Plants," be considered by the NRC Office of Nuclear Regulatory Research (RES) in addressing the use of hydrogen gas in nuclear plants as part of Generic Issue (GI)-106, "Piping and Use of Highly Combustible Gases in Vital Areas." AEOD also suggested that NRR issue an information notice to inform licensees of the hydrogen gas intrusion event at the H. B. Robinson plant.

Subsequently, RES did consider this issue, and the work scope on GI-106 was modified to address the information provided in the engineering evaluation report. NRR considered issuing an information notice on the issue but did not do so because INPO issued a significant event report on the event on April 11, 1989.

7.4 On-Demand Malfunctions of HPCI and RCIC

(Engineering Evaluation Report AEOD/E904)

Despite many studies and efforts for improvement made by the NRC and the industry, failures of the high-pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems have occurred in the past and seem to continue to occur at an undesirable frequency. The performance of either system has been of continuing regulatory concern because the systems are single-train, non-redundant systems that are needed for high-pressure coolant makeup whenever the feedwater system is unavailable. It is estimated that the feedwater system will be unavailable on two occasions a year (one failure initiating the scram and one failure occurring as a consequence of the scram) after a reactor scram on low reactor vessel water level at a typical boiling water reactor (BWR). Under these conditions, the feedwater system may be difficult to restart, and if the HPCI and RCIC systems do not respond, drastic action such as reactor depressurization is required.

Because many statistical studies on the reliability of the HPCI and RCIC systems have been conducted, the AEOD staff thought that a compilation of actual malfunctions of the HPCI and RCIC systems as they respond to an operational transient might be more useful for identifying and directing regulatory attention. The most frequent operational transient for challenging

the HPCI and RCIC systems is the analyzed occurrence of a loss of main feedwater flow to the reactor core. A less frequent operational challenge is the post-scram reactor depressurization transient.

The HPCI and RCIC systems can inject water into a BWR core at full operating pressure. The HPCI system is an emergency core cooling system (ECCS) installed to mitigate the consequences of a small-break loss-of-coolant accident that would not depressurize the reactor to the low-pressure ECCS injection pressure. However, HPCI is most commonly automatically initiated on low reactor vessel water level when feedwater flow is reduced or interrupted. The HPCI system can also be used to depressurize the vessel after a reactor scram by injecting cold water and using reactor steam to drive the HPCI turbine. Because this system is designed to mitigate the effects of an accident, it has much greater pumping capacity than the RCIC system (by about a factor of 10).

For BWRs with both HPCI and RCIC systems, the Nuclear Documents System (NUDOCS) subject/Boolean search was used to find events involving an operational transient. The specific search terms were "docket number" and "LER with HPCI," or "RCIC," or "scram." With this search, all LERs at a specific plant involving the HPCI or the RCIC system or a reactor scram should have been retrieved. For the 22 plants in the study, approximately 40 to 90 LERs for each plant matched this search criterion. The term "scram" was used because all operational loss-of-feedwater transients would involve a reactor scram. Each LER was reviewed to determine if the event involved an operating transient together with the unavailability or subsequent malfunction of the HPCI or the RCIC system. Challenges that resulted in both HPCI and RCIC system successes were not tabulated in this study.

In reviewing the data, AEOD found that the HPCI and/or the RCIC system could fail to perform its intended function in three ways as follows:

1. Failure—The system was unable to perform its required function throughout the operating transient.
2. Successful Recovery—An initially faulted system regained, by any means, the ability to perform successfully its required function in response to the operating event.
3. Technically Inoperative—The system was declared to be or was technically inoperative before the initiating transient but nevertheless successfully performed its specific function throughout the transient.

The study found that all the malfunctions of the HPCI or the RCIC system following an operational transient

were essentially equivalent; that is, there was no statistically significant difference in the reliability of the two systems.

The study also found that the majority of the on-demand malfunctions of the HPCI or the RCIC system during an operational transient occurred at four plants in Region II: Hatch Units 1 and 2 and Brunswick Units 1 and 2. In addition, each of the six known malfunctions of both the HPCI and RCIC systems during an operational transient occurred at these four plants. Finally, the last six HPCI or RCIC system malfunctions and the most recent failure of both systems during a transient occurred at these plants.

The report found that the reported malfunctions at these plants dominated all trends for the remaining BWRs in the study.

AEOD provided a copy of the report to Region II management for their consideration in taking action on deficient performance.

7.5 Electrical Bus Bar Failures

(Engineering Evaluation Report AEOD/E905)

Failures of medium-voltage electrical bus bars occurred at the Palo Verde, Kewaunee, Millstone, and Sequoyah nuclear facilities. These failures, principally involving 4160- and 6900-volt ac buses, resulted in bus bar electrical faults and fires, electrical power system undervoltage conditions, plant transients, and reactor trips.

The failure of the bus bars was attributed to the cracking of Noryl bus bar insulation (bus sleeving), combined with the accumulation of moisture or debris in the bus bar housings. Cracked insulation resulting from high ambient temperatures and contamination from the use of "black" bus bar joint compound was observed on bus bars that were manufactured by Westinghouse, General Electric, and the Calvert Company and that were insulated with Noryl flame-retardant insulation. Cracked Noryl insulation in the presence of moisture or debris provided undesired phase-to-phase or phase-to-ground electrical tracking paths, which resulted in catastrophic failure of the buses. Bus failures led to loss of power to electrical loads connected to the buses, caused subsequent reactor trips, initiated unnecessary challenges to plant safety systems, and provided opportunities for equipment malfunctions and operator errors.

As a result of the analysis of information on the Palo Verde occurrence, AEOD initiated searches of available LERs to determine generic applicability of the

insulation failure at that plant. Several different database search techniques were used to identify the specific electrical power system failures described in this report. First, NUDOS text searches of LERs issued since 1984 were conducted to identify those LERs containing the word "Noryl." Sequence Coding and Search System (SCSS) searches were conducted to identify LERs containing the name of an electrical bus manufacturer and the term "Noryl." The SCSS search was then broadened to include all reports of electrical insulation problems.

The findings and conclusions of AEOD/E905 were the following:

- The potential exists for Noryl-insulated medium-voltage buses to experience catastrophic failure if they are not periodically inspected and maintained free of moisture and debris. Failures occurred at the Palo Verde, Kewaunee, Millstone, and Sequoyah plants.
- Bus failures, in turn, cause subsequent reactor trips, initiate unnecessary challenges to plant safety systems, and provide opportunities for equipment malfunctions and operator errors.
- Noryl-insulated buses should be inspected periodically at nuclear power facilities to determine if the insulation has cracked.
- Enhanced maintenance programs for medium-voltage electrical buses should be developed and implemented by utilities to prevent the buildup of water and debris in the bus housings.
- Bus bar enclosures around Noryl-insulated buses should be assessed to determine the need for modifications to the enclosures to prevent intrusion and accumulation of water or debris.

AEOD transmitted AEOD/E905 and a proposed information notice on this subject to NRR. The information notice was subsequently issued as NRC Information Notice 89-64, "Electrical Bus Bar Failures," on September 7, 1989.

7.6 Failure of Steam Generator Isolation Check Valve

(Engineering Evaluation Report AEOD/E906)

Since 1979, the NRC has issued a number of generic communications pertaining to faulty check valves. The Nuclear Plant Reliability Data System (NPRDS) contains reports of more than 286 check valve failures, 22 of which can be associated with potential intersystem leakage.

Recently, the nuclear industry has initiated several efforts to improve the performance of check valves. Activities of the Nuclear Industry Check Valve Group and the standards-writing American Society of Mechanical Engineers Operations and Maintenance Committee OM-22 are especially noteworthy. The NRC is keenly interested in the success of these efforts.

As part of this ongoing effort, AEOD issued AEOD/E906, which discusses the failure of the steam generator (S/G) isolation check valves at Calvert Cliffs.

On March 17, 1988, a licensee inspection showed that both S/G isolation check valves at Unit 2 had been damaged and were incapable of performing their isolation function. These check valves were replaced, together with their identical counterparts at Unit 1, with heavier check valves with an improved seating material.

To prevent recurrence of this type of failure, the licensee committed to reverse leak-test the check valves every refueling outage or, as an alternative, to disassemble and inspect one valve per refueling outage. If the inspected valve shows signs of wear indicating degradation leading toward failure, the other check valve will be disassembled and inspected. The valves are partially stroked for positive flow verification during a monthly surveillance test to verify the operability of the auxiliary feedwater actuation system.

On October 29, 1988, the licensee found that one of the recently installed heavier S/G isolation check valves at Unit 1 was leaking.

In this report, AEOD concluded that a failed S/G isolation check valve could lead to a more severe steamline break accident than that analyzed in the Final Safety Analysis Report (FSAR). The failure of this check valve may permit both steam generators to communicate during a postulated steamline break scenario (see Figure 7.1). Thus, it is possible that both steam generators may release steam (blow down) simultaneously through a common break in the steamline. With both steam generators blowing down, steam generator differential pressure may not be sufficient to isolate auxiliary feedwater flow to the ruptured steam generator. The continuation of this flow during a blowdown of two steam generators may result in a more rapid and prolonged cooldown and a greater insertion of positive core reactivity than that assumed in the FSAR when there is a single active failure (the most reactive control element assembly in its fully withdrawn position after the reactor trip).

The report indicates that licensees need to ensure that these check valves are capable not only of passing

steam to drive the auxiliary feedwater pump turbines, but also of preventing a simultaneous blowdown of the steam generators. Inservice testing of these check valves should be conducted regularly for both the forward- and reverse-flow directions. NRC Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," which addresses the need for back-flow testing of check valves, appears to be an appropriate step toward resolving this concern. In addition to testing, periodic disassembly of check valves to inspect their internal components could provide valuable information related to check valve performance.

This information is being considered for inclusion in a possible information notice on check valves.

7.7 Diversion of Seal Cooler Flow for RHR Pumps

(Engineering Evaluation Report AEOD/E907)

This study was prompted by a single-failure vulnerability in the residual heat removal (RHR) system reported at Haddam Neck. The licensee identified this problem during a probabilistic risk assessment study of the emergency core cooling system. The RHR system is used for long-term cooling following a loss-of-coolant accident (LOCA). Cooling water for the RHR pump seal coolers comes from a supply header off the cooling water inlet lines to each RHR heat exchanger. Normally, the component cooling water system cools the heat exchangers and seals, but during an accident the component cooling water is isolated and service water is valved in to provide cooling.

If one of the service water valves did not open during alignment for recirculation cooling following a LOCA, one branch (part of a train) of service water would have to provide cooling water to both RHR heat exchangers, one directly and one via a small interconnecting line. The increased flow in the small interconnecting line would result in an increased pressure drop in this line and therefore in reduced flow to the RHR pump seal coolers as this flow is tapped off this small line.

A survey of service water systems for about 30 pressurized water reactors (PWRs) indicated that most plants do not use service water to cool the emergency core cooling system pumps directly or they have a different piping arrangement to preclude this situation. The potential for the flow diversion problem depends on the actual piping configuration and the valve alignment for the auxiliary water system under normal and accident conditions. This type of deficiency generally is

Figure 7.1
Anticipated Valve Lineup During
a Steam Line Break at Calvert Cliffs Plant

undetected for extended periods because it is not considered in periodic surveillance tests.

The licensee added check valves in the small interconnecting line to eliminate the problem. Information Notice 89-71, "Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-of-Coolant Accident," was issued on October 19, 1989, to alert other licensees to this potential problem.

7.8 Excessive Valve Body Erosion at Brunswick

(Engineering Evaluation Report AEOD/E908)

Excessive valve body erosion was discovered in low-pressure coolant injection (LPCI) system injection valve E11-F017B at Brunswick Unit 1 during a planned disassembly and inspection as part of the licensee's long-term maintenance program for the plant. The valve body had erosion pockets with a minimum wall thickness of 1.7 inches, compared to an original wall thickness of 3.5 to 4 inches. The same valve in the other LPCI train had similar erosion damage. AEOD had previously issued reports in 1983, 1984, 1987, and 1988 that addressed various erosion issues. The AEOD staff therefore decided to perform a broad overview study of erosion.

The database used for the review included AEOD Reports E315, E416, E807, and C801; events from the Sequence Coding and Search System (SCSS); and events from the Nuclear Plant Reliability Data System (NPRDS) equipment failure records. More than 30 new events not considered in previous AEOD studies were obtained from the SCSS and more than 140 new events were identified by the NPRDS.

This study indicated that a broad range of erosion-induced problems existed, including cavitation-induced valve body erosion, valve assembly degradation as a result of vibration, pipe support damage as a result of vibration, piping erosion downstream of throttled valves, severe degradation of pumps in the service water system and the residual heat removal (RHR) service water system, low-flow-related damage to pumps in other systems, and corrosion/erosion fouling that may adversely affect the heat transfer capability of heat exchangers. The NRC had issued two information notices, one bulletin, and one generic letter to address these problem areas. AEOD found that two ongoing efforts were also addressing erosion. The BWR Owners Group had established an effort to cover "target" valves in systems such as the RHR, high-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), high-pressure core spray (HPCS), and RHR

service water systems. The BWR Owners Group will concentrate on valve cavitation but will not address pump degradation resulting from low-flow conditions. The second effort involves the response by licensees to Generic Letter 89-13 pertaining to service water systems.

This overview study concluded that PWR plants had not been reviewed in a similar manner as BWR plants and that low-flow-induced degradation of pumps was not being addressed by the industry. This information was transmitted to NRR with a suggestion that NRR address these issues. NRR requested that the Nuclear Management and Resources Council (NUMARC) approach the PWR Owners Group to review erosion. NRR also sent a memorandum to RES to request that it recommend to the American Society of Mechanical Engineers (ASME) that erosion monitoring of pumps be considered for inclusion in the ASME Code.

7.9 Operator Actions During Operational Events

(Engineering Evaluation Report AEOD/E909)

This study was prompted by the recent steam generator tube rupture accident at McGuire and the extended primary system blowdown at Braidwood during RHR system cooling. The events cited in the report were found by routine review of inspection reports.

Several related events occurred at McGuire. One was the blocking of the safety injection (SI) actuation signal while a depressurization transient was in progress. The SI signal was blocked after discussion between the two reactor operators and the senior reactor operator regarding conditions and the safety of plant personnel inside the primary containment at the time.

The operators manually increased charging flow to maintain pressurizer level following a steam generator tube rupture at McGuire and consequently used abnormal operating procedures to recover from this event instead of emergency operating procedures specifically developed by the owners group for steam generator tube rupture. The overall mitigative strategy deviated substantially from the owners group guidelines, and plant cooldown was substantially delayed because the licensee was concerned about the boron concentration and the system temperature and pressure limits imposed by the operating procedures used in the recovery.

The "Diagnostic Evaluation Team Report for McGuire Nuclear Station," dated March 8, 1988, noted: "The practice of giving the operators greater flexibility in some procedures could carry over into the use of this practice in other procedures which require strict

verbatim step-by-step compliance to assure safe operation of the plant.”

NRC Inspection Report 50-247/88-07, dated May 2, 1988, on operator training at Indian Point noted that “the instructor was not teaching the procedural requirement that the EOPs are to be implemented as written...[and] the second simulator instructor stated that they teach ‘good operating philosophy’ and should not be forced to teach policy.” A similar concept was noted in Inspection Report 50-293/87-53, dated December 14, 1987, for an event at Pilgrim: “Discussions with watch engineers indicate that based on their experience and plant knowledge, they are prepared to take prudent actions, without spending substantial time sifting through and interpreting the interrelated procedures which would cover the full spectrum of operator actions during a station blackout.”

The issue of operator guidance, Generic Issue 125.II.10, “Hierarchy of Impromptu Operator Actions,” was raised after the Davis-Besse event in 1985.

The results of this study support the development of such guidance to cover unanticipated situations. AEOD sent the report, in which it endorsed further development in this area, to RES. Subsequently, RES agreed to address this issue in its Long Term Human Factors Research Program.

7.10 Potential for Gas Binding of High Head Safety Injection Pumps Resulting From Inservice Testing of VCT Outlet Isolation Valves

(Engineering Evaluation Report AEOD/E910)

At the Trojan nuclear plant, personnel discovered that by opening one manual bypass valve around the volume control tank (VCT) outlet isolation valves during testing, both high head safety injection pumps could be rendered inoperable as a result of hydrogen gas binding if a safety injection signal occurred while the bypass valve was open. Upon completion of testing, should the manual bypass valve be inadvertently left in the open position, both trains of high head safety injection could be unknowingly rendered inoperable for an extended period.

At Trojan, three charging pumps have the capability of taking suction from the VCT (see Figure 7.2). One pump is a positive displacement pump. The other two pumps are high head centrifugal charging pumps, which are engineered safeguards components, providing high head safety injection during a loss-of-coolant accident

or a main steamline break. On receipt of a safety injection signal, both high head centrifugal charging pumps are started, the normal charging path isolates the VCT, and flow from the charging pumps is aligned to the boron injection tank while seal injection flow to the reactor coolant pumps is still being maintained. Additionally, valves MO 112D and MO 112E open to supply charging pump suction from the refueling water storage tank, and valves MO 112B and MO 112C close to isolate the VCT from the charging pumps.

Valve 8484 is normally locked closed, and valve 8482 is normally open to provide a flow path for the water from the return heat exchanger for the reactor coolant pump seal. During tests at Trojan, valve 8484 is opened to provide a flow path around the VCT outlet isolation valves. This alignment allows valves MO 112B and MO 112C to be time opened and closed while the VCT still supplies the charging pump suction. However, this valve configuration also defeats the automatic capability to isolate the VCT on receipt of a safety injection signal. If VCT hydrogen cover gas pressure was sufficiently high, suction to the pumps could be preferentially provided from the VCT, even though the pump suctions were aligned to the refueling water storage tank. Hydrogen gas could then be entrained into the high head safety injection pumps, rendering both of them inoperable. For a large-break loss-of-coolant accident (LOCA), with an initial VCT pressure of 15 to 45 psig, the licensee calculated that the pumps would become inoperable within 1 to 19 minutes of initiation of the safety injection signal.

The findings and conclusions of AEOD/E910 were the following:

- The potential exists for a common mode failure to disable both high head safety injection pumps by opening a manual bypass valve around the VCT outlet isolation valves during testing.
- This situation can be exacerbated by inadvertent failure to close the valve after testing is completed. Should the manual bypass valve be inadvertently left in the open position, both trains of high head safety injection could be unknowingly rendered inoperable for an extended period.
- Inservice test procedures, surveillance test procedures, and operating procedures should be reviewed to determine if precautions should be added to inform plant personnel of the potential to disable the high head safety injection pumps when opening the manual bypass valve around the VCT outlet isolation valves. Alternatively, the procedures could be modified to ensure

Figure 7.2
Charging System at Trojan Plant

pump suction is taken from the refueling water storage tank rather than from the VCT through the bypass valve.

AEOD transmitted AEOD/E910 and a proposed information notice supplement on this subject to NRR. The supplement was issued as Supplement 2 to NRC Information Notice 88-23, "Potential for Gas Binding of High-Pressure Safety Injection Pumps During A Loss-of-Coolant Accident," on January 31, 1990.

7.11 Inadvertent Reactor Trips Due to RCS Flow Instrumentation Maintenance Activities

(Technical Review Report AEOD/T902)

A review of recent plant operating events showed that reactor trips had been inadvertently generated as a result of maintenance activities on reactor coolant system (RCS) flow transmitters at several Westinghouse pressurized water reactor plants while the plants were at power. These reactor trips occurred at Trojan on August 16, 1988; at Sequoyah Unit 2 on May 23, 1988; at Salem Unit 2 on April 21, 1988; and at Diablo Canyon Unit 1 on January 8, 1988.

In each case, the reactor trips were attributed to personnel error. Technicians failed to follow established procedures and generally did not understand that with the plant operating at higher power levels, even careful manipulation of RCS flow transmitter isolation, calibration, and vent valves could create sufficient perturbations in sensed RCS flow to cause inadvertent reactor trips.

Flow transmitter sensitivity to maintenance activities is a result of system design. Three flow transmitters in each RCS loop share a common "high side" pressure tap. During maintenance activities on a single transmitter, perturbations induced at that transmitter can be propagated to all three transmitters, resulting in the generation of a reactor trip signal.

Unplanned reactor trips cause unnecessary challenges to plant safety systems and provide opportunities for unexpected equipment malfunctions and operator errors. Accordingly, both the NRC and the nuclear industry have focused increased attention on the issue of unplanned reactor trips at U.S. plants. Nuclear steam supply system vendor owners groups also have established programs to achieve reactor trip reductions.

The findings and conclusions of AEOD/T902 were the following:

- A search of the licensee event report database revealed that four inadvertent reactor trips had occurred in 1988 at Westinghouse plants as a result of sensed low RCS flow during maintenance of RCS flow transmitters.
- The inadvertent RCS low-flow trips were attributed to personnel errors, procedural inadequacy, improper instrumentation design configuration, and a general lack of understanding of the sensitivity of RCS flow transmitters with common sensing lines to valve manifold manipulations.
- Corrective actions taken by involved utilities included revising appropriate procedures to include precautions and detailed guidance for instrument manifold manipulations, incorporation of "wet calibration" techniques, and the use of signs at valve manifold locations to caution personnel about the sensitivity of the transmitter to valve manipulations.
- No inadvertent reactor trips resulting from maintenance activities on steam generator level or pressurizer pressure instrumentation sharing common sensing lines were reported between January 1987 and December 1988.

On February 7, 1989, a letter from the Director of the Division of Safety Programs (AEOD) transmitted AEOD/T902 to the chairman of a Westinghouse owners group for consideration in its Trip Reduction and Assessment Program.

7.12 Design Deficiency of Safety Injection Block Switch

(Technical Review Report AEOD/T904)

Point Beach Unit 1 has only one "block/unblock" selector switch to block the automatic safety injection (SI) of both trains to permit normal cooldown and depressurization during hot-to-cold-shutdown operation. This selector switch is a Westinghouse Type OT2V6, three-position, cam-operated, and spring-return-to-neutral selector switch with four Westinghouse Type OT2A contact blocks: two top contact blocks for blocking SI, one for each train, and two bottom contact blocks for unblocking SI, one for each train.

If the top blocks stick in the closed position, automatic SI of both trains will be prevented. This block switch controls SI actuation by low steamline pressure or low pressurizer pressure. Even if this switch failed, automatic SI actuation on high containment pressure would still occur.

Furthermore, the SI master relay will energize independent annunciator lights “SI Blocked Train A” and “SI Blocked Train B.” This activation will alert the operator to the need for manual SI operation. AEOD discussed the issue with Westinghouse, Pittsburgh. Subsequently, Westinghouse alerted other licensees who use similar switches about the problem and recommended the use of separate block/unblock switches for each SI train.

7.13 Broken Lifting Beam Bolts in HPCI Terry Turbine

(Technical Review Report AEOD/T906)

The NRC has long been concerned with incidents of failed, degraded, or wasted fasteners in systems that are essential for coping with design-basis accidents or in closures to the primary pressure boundary. As a result of this ongoing concern, the NRC issued Office of Inspection and Enforcement (IE) Bulletin 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” and IE Information Notice 82-06, “Failure of Steam Generator Primary Side Manway Closure Studs.”

More recently (as reported in Licensee Event Report 333/87-03), a licensee maintenance inspection of the high-pressure coolant injection (HPCI) Terry turbine during the 1987 refueling outage at the James A. FitzPatrick Nuclear Power Plant revealed that six of the eight bolts that hold the turbine throttle valve lifting beam together were broken. The remaining two bolts were badly cracked. The broken bolts did not render the turbine inoperable because two dowel pins and the two remaining badly cracked bolts kept the lifting beam intact. If the lifting beam had not remained intact, turbine control would have been lost, causing the HPCI system to be inoperable.

A metallurgical evaluation by the licensee of three failed American Society for Testing and Materials (ASTM) A193, Grade B6 (Type 410 stainless steel), 5/8-inch-diameter by 1-3/4-inch-long bolts led to the following determinations:

- The bolts failed by an intergranular failure mode, most likely stress corrosion cracking (SCC).
- Hardness measurements showed the bolts were much harder than expected. The higher level of hardness was thought to have been a major contributor to the cause of failure. Laboratory tempering at 1100°F for 1 hour resulted in a drop in hardness of more than 10 points Rockwell C, from

the upper 30s for the as-received bolts to the mid-20s for the laboratory-retempered bolts.

- Copper was detected in the thread root regions of the broken bolts. It is believed that the copper may have come from a copper-bearing antiseizure compound used on the bolts. The use of such compounds causes localized pitting corrosion, which, in turn, may have acted as the origin sites for the stress corrosion cracks in the Type 410 stainless steel. The bolt vendor recommended that when bolts are heat treated as specified for ASTM A193, Grade B6 bolts, the use of a lubricant such as nonmetal-bearing petroleum jelly be used as an antiseizure compound. This recommendation for bolt lubricants will be incorporated into the FitzPatrick HPCI maintenance procedures.

A Nuclear Documents System (NUDOCS) data search did not identify any similar degradations of throttle valve lifting beams or SCC in any components of the HPCI system. Therefore, it appears these bolt degradations were due solely to plant-specific deficiencies, and industry action on HPCI lifting beam bolts would not be appropriate at this time.

The bolt failures described in this technical review report were subsequently discussed in NRC Bulletin 89-02, “Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design,” to document similar bolting failures in swing check valves.

7.14 Operating Events Involving Dampers

(Technical Review Report AEOD/T909)

This study was initiated because licensees appeared to be submitting licensee event reports involving dampers regularly. These devices are used throughout the plant to provide system realignment for emergency ventilation in the control room, the containment building, the diesel generator building, the auxiliary building, and other locations. They are also used as barriers to prevent the spread of fire through the ventilation system.

A search of the NUDocs identified about 250 documents with the word “damper” or “dampers” in the title or text for the period 1986 through about the first quarter of 1989. About 130 independent references were divided between ventilation concerns (60 percent) and fire barrier problems (40 percent).

Ventilation dampers include louvered, butterfly, blade, and bladder types that can be air actuated or motor driven. Fire dampers are generally a curtain type with a

fusible link that is heat sensitive. The damper is closed, after the link melts, by gravity or is spring assisted.

The inability of fire dampers to close under design air-flow conditions was identified in 1984 and was subsequently discussed in Information Notice 86-50, "Inadequate Testing To Detect Failures of Pneumatic Components or Systems," dated June 18, 1986. Damper installation problems included interferences, upside-down installation, installation in the middle of a duct instead of at the wall, and no installation at all. The Office of Nuclear Reactor Regulation (NRR) is developing an information notice delineating the inadequacies in fire damper design and application and in surveillance testing.

Problems with ventilation dampers include mispositioning, single failures, absence of backdraft dampers, and actuator problems. Problems with dampers in ventilation systems could go undetected because damper positions are not monitored during surveillance tests. NRR is considering the need for developing a generic letter on control room emergency ventilation systems to address these types of problems.

7.15 Debris in Containment Recirculation Sumps

(Technical Review Report AEOD/T916)

During cleanup activities after a refueling outage at the Trojan nuclear power plant, debris was discovered in the containment recirculation sump. Items found in the sump included insulation, metal wire, pipe fittings, fabricated steel, duct tape, and a piece of muslin material.

The containment recirculation sump and associated screens, shelves, and curbs are designed to provide a source of water, with a minimum acceptable amount of debris, to the containment spray system and the residual heat removal system during the recirculation phase of a loss-of-coolant accident. Debris in the containment sump could be drawn into the suction of the containment spray pumps and the residual heat removal pumps during the recirculation phase. This circumstance potentially could cause a partial or complete loss of one or more of the containment spray, residual heat removal, or safety injection systems.

As a result of the identified concerns, AEOD initiated this study to investigate the frequency of occurrence and the generic applicability of debris accumulation in containment recirculation sumps. In addition to the occurrence at Trojan, the study evaluated seven licensee event reports (LERs) that were identified through a

Sequence Coding and Search System (SCSS) review of the LER database from 1980 to 1989. In particular, the search focused on foreign material in (1) the containment, (2) equipment drains, (3) the decay heat removal system, and (4) the containment spray system.

Except for occurrences at Surry Units 1 and 2, none of the other LERs involve actual accumulation of debris in containment sumps. At Surry, foreign objects identified and removed from the sumps included nuts, bolts, wire, metal pieces, and wood. The licensee determined that the sources of the debris were early plant modifications made when adequate foreign material exclusion programs were not being implemented. The events at Surry were specifically addressed in NRC Information Notice 88-87, "Pump Wear and Foreign Objects in Plant Piping Systems," issued on November 16, 1988.

At the Trojan facility, immediate corrective actions have been completed, and long-term programmatic actions are in progress.

On the basis of these findings, no further AEOD action on this matter is warranted at this time.

The event at the Trojan plant was included as an abnormal occurrence in NUREG-0090, Vol. 12, No. 3 (*Report to Congress on Abnormal Occurrences: July-September 1989*).

7.16 Check Valve Failure Rates From NPRDS Data

(Technical Review Report AEOD/T918)

The Advisory Committee on Reactor Safeguards requested information about check valve failure rates. The approach selected to develop the check valve failure rate was to use the Nuclear Plant Reliability Data System (NPRDS) procedure to determine failure rates for a single component group. The failure rate in failures per million hours and the failure rate trend in equation form were developed for three groups of check valves on the basis of inlet size. The groups were (1) all sizes of inlets, (2) inlets 2.0 inches or greater, and (3) inlets less than 2.0 inches. The failure rate and the failure rate trend for each group were as follows:

Check Valve Size Group	Failures per Million Hours	Best Estimate, Failure Rate Trend
All sizes of inlets	5.34	$y = -0.05x + 7.03$
Inlets 2.0 inches or greater	5.48	$y = -0.06x + 7.36$
Inlets less than 2.0 inches	4.77	$y = -0.02x + 5.61$

Figure 7.3 is a line plot of the failure data for check valves with an inlet size of 2.0 inches or greater. This inlet size was selected to envelope anticipated pipe diameters for safety systems. The data covers the period from January 1984 through July 1988.

The data shows there is little difference in failure rate (failures per million hours) for the check valve size groups and the failure rate exhibits a small downward slope with time.

7.17 Delaminating Foil Insulation in Primary Containment

(Technical Review Report AEOD/T923)

On March 5, 1988, during a walkdown inspection of the primary containment following commencement of a refueling outage at Susquehanna Steam Electric Station Unit 2 (SSES-2), large quantities of loose aluminum foil were found throughout the drywell. The licensee concluded that loose aluminum foil could become debris during the blowdown phase of a loss-of-coolant accident and could block emergency core cooling system (ECCS) pump suction strainers, resulting in inadequate net positive suction head, thus causing pump damage.

The aluminum foil came from surface delamination of fibrous insulation blankets made by Alpha Associates, Inc. The aluminum foil fiberglass cloth insulation had been used extensively on valve bodies, pipe hangers, and other areas difficult to insulate in the SSES-2 drywell. The exact method of delamination could not be determined but could be the result of extended exposure to operating temperatures, humidity, radiation, or some combination of these and other factors. Although the rated service temperature of the insulation was 500°F, normal operating temperatures in the drywell do not exceed 135°F.

At SSES-2, the licensee estimated that 5,000 square feet of this insulation was used in more than 300 locations within the drywell and that 50 percent of the insulation had undergone some degradation. Because of the apparent extensive breakdown of the aluminum foil laminating adhesive in the SSES-2 drywell, the licensee believed that continued exposure to operating conditions would have resulted in further delamination.

A different insulation material is used at SSES-1, and no insulation delamination has been observed at this unit.

On March 21, 1988, the foil delamination problem was reported pursuant to 10 CFR Part 21. Subsequently, NRC Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris," was issued on May 19, 1988. This information notice noted that representatives of Alpha Associates, Inc., had indicated that other companies supplied the same or similar insulation and that its use might be widespread.

The event was included in Appendix C ("Other Events of Interest") of NUREG-0090, Vol. 11, No. 2 (*Report to Congress on Abnormal Occurrences: April-June 1988*).

On March 18, 1988, the licensee for Unit 1 of the Millstone Nuclear Power Station specified in LER 88-004 a similar concern for potential fouling of its ECCS strainers. In this case, an engineering evaluation showed that the area of the existing strainers was insufficient to pass the debris from fibrous insulation in the drywell. At Millstone Unit 1, the same insulation as that used at SSES-2 is used without an aluminum foil facing.

Many publications have addressed the potential problem of debris blocking containment emergency sump screens in pressurized water reactors or residual heat removal pump strainers in boiling water reactors. For example, Regulatory Guide 1.82, "Sumps for ECCS & Containment Spray Systems," was published in 1974, and Generic Issue A-43, "Containment Emergency Sump Performance," was initiated in 1979 and resolved in 1986. In view of the extent of information on containment debris and the number of years it has been available, it was not reassuring to receive two new LERs on the subject. As both LERs indicated potential deficiencies with insulation blankets from the same manufacturer, and the manufacturer supposed that its insulation or similar insulation may be widely used throughout the nuclear industry, AEOD believed that searches for other instances of recent potential insulation breakdown would be appropriate.

The available search systems, the Sequence Coding and Search System, the Nuclear Plant Reliability Data System, and the Nuclear Documents System, were used to determine if any other recent events of insulation deficiencies in primary containment had been reported. None of the searches revealed any additional ongoing insulation deficiencies in primary containment. Therefore, it appears that these two LERs represent isolated cases rather than an ongoing generic deficiency.

Figure 7.3
Check Valve Failure Rates
(Two-Inch Inlet Size and Greater)

8 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 non-reactor activities licensed by the NRC. Under the IIP, the NRC responds to an operational event according to its safety significance. For an event of potentially major safety significance, the Executive Director for Operations (EDO) establishes an Incident Investigation Team (IIT) to investigate the event, and for an event of less safety significance, the cognizant NRC Regional Administrator may establish an Augmented Inspection Team (AIT) to investigate the event. 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.

During 1985, three IITs were established to investigate incidents at Davis-Besse Nuclear Power Station (June 9), San Onofre Nuclear Station (November 21), and Rancho Seco Nuclear Generating Station (December 26). The status of staff actions assigned by the EDO to the various NRC offices associated with these IIT reports is given in Appendix F. Appendix F, which is a new appendix to the AEOD Annual Report, has been added as a means of recording, tracking, and following up those actions assigned by the EDO regarding IIT findings. It is anticipated that Appendix F will be included in future AEOD Annual Reports.

During 1989, no event was judged to have a sufficiently high level of safety significance to warrant an IIT investigation. However, 19 AIT inspections were conducted during 1989 (see Table 8.1).

These AIT inspections helped to improve the safety of the plants at which they were conducted by providing detailed investigations of the problems experienced at the plant and identifying their root causes. Regional Administrators are responsible for identifying needed actions on the basis of the AIT findings. In addition, the Directors of the Office of Nuclear Reactor Regulation and the Office of Nuclear Material Safety and Safeguards are responsible for reviewing AIT reports for generic safety implications and initiating followup actions and tracking issues on a multiplant basis, as appropriate. The Office for Analysis and Evaluation of Operational Data independently reviews AIT reports to provide additional assurance that lessons are learned and communicated to the industry. Thus, industry-wide safety is enhanced by including the significant lessons learned from the AIT inspections, along with engineer-

ing studies and reviews of operating experience, in generic communications to licensees.

Examples of lessons learned and communicated to licensees from events investigated by AITs in 1989 include the following:

- On January 20, 1989, at Arkansas Nuclear One, Unit 1, an event involving reactor coolant system (RCS) leakage caused excessive temperatures in emergency core cooling system piping located outside containment. A failed check valve in the high-pressure injection system caused the RCS leakage. As a result of the AIT evaluation of this event, NRC Information Notice (IN) 89-36 was issued. This notice alerted licensees to the potential for RCS leakage outside containment, thus exposing emergency core cooling system piping to fluid temperatures in excess of design temperatures.
- On March 3, 1989, at Palo Verde Nuclear Generating Station, Unit 3, an event involving multiple equipment failures occurred following a large load rejection, which resulted in a reactor trip and actuation of engineered safety features. As a result of the AIT evaluation of this event, NRC IN 89-38 was issued. This notice alerted licensees to potential failures of main steamline atmospheric dump valves (ADVs) and deficiencies found in the licensee's inservice testing program. These problems may have contributed to the multiple ADV failures at Palo Verde.
- On April 23 and May 5, 1989, at Comanche Peak Steam Electric Station, Unit 1, a combination of procedural errors and malfunctioning check valves resulted in backflow of high temperature water from the steam generators through the auxiliary feedwater system to the boron sulfate storage tank. The high temperature water caused both thermal stress in excess of allowable stresses and damage to piping supports. As a result of the AIT evaluation of this event, NRC IN 89-62 was issued. This notice alerted licensees to the potential malfunctioning of Borg-Warner pressure seal bonnet check valves caused by the misalignment of the valve disk as a result of deficient vendor procedures.
- On September 5, 1989, at William B. McGuire Nuclear Station, Unit 2, an event occurred involving overpressurization of the containment spray system and subsequent failure of a containment spray heat exchanger head gasket,

Table 8.1 Plants at Which AITs Conducted Inspections in 1989

Event Date	Plant & Unit	Event	Report Date	Report Number
01/03/89	Oconee 1	Electrical fire and loss of forced coolant flow	02/10/89	50-269/89-03 50-270/89-03 50-287/89-03
01/20/89	Arkansas 1	Coolant loss outside of containment	02/17/89	50-313/89-03 50-368/89-03
03/02/89	LaSalle 1 & 2	Unit 2 transformer fault that resulted in Unit 1 trip and equipment malfunction	03/21/89	50-373/89-07 50-374/89-07
03/03/89	Palo Verde 3	Multiple equipment failures following load rejection	04/19/89	50-528/89-13 50-529/89-13 50-530/89-13
03/07/89	McGuire 1	Steam generator tube rupture	04/10/89	50-369/89-06
04/12/89	Pilgrim 1	Unexpected opening of reactor core isolation cooling valve	05/08/89	50-293/89-80
04/18/89 & 04/19/89	Braidwood 1 & 2	Inattentive licensee employees	05/26/89	50-456/89-14 50-457/89-14
04/19/89	River Bend	Freeze plug failure in service water system that caused loss of spent fuel and shutdown cooling	05/16/89	50-458/89-20
04/23/89 & 05/05/89	Comanche Peak 1	Intrusion of hot water into auxiliary feedwater system	07/10/89 05/05/89	50-445/89-30 50-446/89-30
06/22/89	Seabrook 1	Reactor operation outside bounds of test procedure	08/17/89	50-443/89-82
08/14/89	D.C. Cook 2	Loss of safety system redundancy that resulted in loss of control room instrumentation	09/15/89	50-315/89-25 50-316/89-25
08/16/89	H.B. Robinson 2	Inadequate net positive suction head of auxiliary feedwater pumps	09/15/89	50-261/89-20
08/21/89	Nine Mile Point 1	Contamination of sub-basement by leaking drums	10/02/89	50-220/89-80
09/05/89	McGuire 2	Refueling water storage tank water spill into auxiliary building	10/03/89	50-369/89-31 50-370/89-31
10/23/89	Dresden 2 & 3	Inoperable high-pressure coolant injection systems	11/21/89	50-237/89-23 50-249/89-22
11/21/89	Palisades	Spurious opening of pilot-operated relief valve and failure of motor-operated block valve	01/08/90	50-255/89-33
12/01/89	Braidwood 1	Stuck-open residual heat removal pump relief valve	12/28/89	50-456/89-30
12/07/89	Oyster Creek	Possible failure of reactor protection system to scram on low condenser vacuum	02/14/90	50-219/89-81
12/21/89 & 12/26/89	Prairie Island 2	Reactor trip and partial loss of offsite power	02/09/90	50-282/89-32 50-306/89-32

resulting in the leakage into the auxiliary building of 10,000 to 15,000 gallons of water from the RCS and the refueling water storage tank. As a result of the AIT evaluation of this event, NRC IN 89-73 was issued. This notice alerted licensees to the potential for overpressurization of low pressure systems during testing of valves connecting these low pressure systems to higher pressure systems.

- On February 21, 1989, at Dresden Nuclear Power Station, Unit 2, the temperature was greater than normal in the high-pressure coolant injection (HPCI) room. The abnormal heat load was caused by feedwater leaking through uninsulated HPCI piping to the condensate storage tank. On

October 23, 1989, leakage had increased sufficiently to raise the temperature between the injection valve and the HPCI pump discharge valve to 275°F and at the discharge of the HPCI pump to 246°F. Pressure in the HPCI piping was 47 psia. On the basis of the temperature gradient and the pressure in the piping, the AIT concluded that feedwater leaking through the injection valve had flashed to steam and displaced some of the water in the piping system. As a result of the AIT evaluation of this event, NRC IN 89-80 was issued. This notice alerted licensees to the potential for water hammer, thermal stratification, and steam binding in HPCI piping as a result of the leakage of feedwater into the system.

9 Diagnostic Evaluation Program

The Diagnostic Evaluation Program (DEP) was established in 1987 to provide an assessment of licensee performance at selected reactor facilities. The DEP evaluates the involvement of licensee management and staff in ensuring safe plant operations, the effectiveness of their actions, and the root causes of safety-related performance problems. The DEP supplements the assessment of licensee performance provided by the Systematic Assessment of Licensee Performance Program, the Performance Indicator Program, and the routine and special inspections performed by the NRC headquarters and regional offices. The DEP also enables NRC senior management to make more informed decisions concerning the need for NRC and licensee actions to improve plant safety.

When a diagnostic evaluation is approved for a specific reactor facility, a diagnostic evaluation team (DET) is authorized and established by the NRC Executive Director for Operations. The DET consists of expert technical staff members from headquarters and the regional offices, experienced resident inspectors, and contractors, if appropriate. Team members are selected to provide an unbiased and independent assessment of plant performance. The evaluation process involves observations of plant and corporate activities, in-depth 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 involvement, operation, engineering, and quality programs.

During 1989, four reactor facility performance reviews were conducted under the DEP. DETs were established to evaluate performance at the Perry Nuclear Power Plant, the Brunswick Steam Electric Plant, Arkansas Nuclear One, and the Palo Verde Nuclear Generating Station. Summaries of the major findings from these diagnostic evaluations follow:

Perry Nuclear Power Plant Diagnostic Evaluation

The Perry plant consists of a 1,205-megawatt electric (MWe) boiling water reactor in Perry, Ohio, operated by the Cleveland Electric Illuminating Company (CEI). Perry began commercial operation in 1987.

The diagnostic evaluation (DE), conducted from February 13 to March 10, 1989, concluded that the licensee's management had effectively implemented the management, organizational, and programmatic plans for the transition from construction to operations. Even though

a capable management team, a strong technical staff, and relatively comprehensive program documents had been put in place at the start of the first operating cycle, a significant number of equipment problems emerged during the cycle that significantly affected plant performance, management attention, and technical resources. Many of these problems involved original design deficiencies affecting both safety-related and nonsafety-related equipment that were not corrected as part of the preoperational and startup test programs. Consequently, these problems led to reactor scrams, burdened or distracted the operators, and caused safety-related equipment to fail.

The DET concluded that the root causes of Perry's performance problems during the first operating cycle were (1) pressure to place and maintain Perry in productive operation; (2) the emergence of a significant number of equipment design problems during startup testing and, later, in the operating cycle; (3) management's acceptance of continuation of operations with an elevated number of equipment problems while permanent engineering resolutions were pursued; (4) delayed management attention to procedural and personnel performance problems caused by priority attention to technical or hardware problems and external issues; and (5) human relations weaknesses.

Although extensive solutions for Perry's equipment problems were developed by the engineering and technical support groups, several of the remedies were not implemented during the cycle. Accordingly, end-of-cycle plant performance was still being negatively affected by several continuing equipment problems. The DET further found that although many of the equipment problems were scheduled to be corrected by repair or modification during the first refueling outage, a number of these problems would not be corrected before the second cycle began because of the length of the outage, incomplete engineering analysis, or unavailability of replacement parts. Based on the Cycle 1 decisions and the scope of the first refueling outage, the DET concluded that CEI management was willing to support continued plant operation with reduced equipment reliability over the short term. At the same time, the team concluded that CEI management sought a high level of plant availability through a high level of equipment reliability over the long term.

Progress in reducing personnel errors was mixed during the first cycle. In the operations area, the DET concluded that management was generally aware of, but acted too slowly on, the underlying human performance issues that led to the inattention to detail and procedural adherence problems of operators during the

second quarter of 1988. Poor communication between crews and management, low operator morale, heavy administrative workload on shift supervision, and weak accountability by supervisors for crew performance were identified by the licensee and corroborated by the DET. The team found Perry management's evaluation of the crew performance problems to be thorough and the corrective actions for reducing personnel errors to be effective. However, not all of the licensee's recommendations to address the underlying causes were effectively implemented. Consequently, many of the problems that adversely affected operator morale during 1988 were still present at the time of the DE.

The response by Perry management to these human performance issues was almost totally reactive rather than proactive. This was not uncharacteristic of Perry management's response to other personnel issues. This response adversely affected plant performance during the first cycle. The DET concluded that management's priority attention to technical or hardware problems and external issues such as Centerior (parent company of CEI) rate case hearings, union negotiations, and Centerior affiliation efforts (merger of Davis-Besse and Perry) delayed attention to procedural and personnel performance problems. The DET also found that weaknesses in the human relations area had negatively affected plant personnel performance, contributed to operator errors and union organizing attempts, and increased resignations and reduced worker morale in selected areas at the station. Accordingly, organizational performance was still negatively affected at the end of the first cycle by personnel issues.

Overall, however, the DET concluded that progress had been made during this cycle by Perry management in addressing the causes of the problems that adversely affected performance, as evidenced by the improving trends in most performance indicators over the last two quarters. Except for the human relations area, plant management had effective self-analysis and performance-monitoring programs, which enabled the plant staff to develop an early and detailed awareness of technical, programmatic, and performance weaknesses.

Additionally, Perry management had demonstrated the capability to solve difficult problems effectively once they became a management priority. Improvements involving station teamwork, personal accountability, equipment design and maintenance, and increased staff experience with equipment operations, testing, and procedures were also having a positive effect toward the end of the cycle.

Notwithstanding these improvements, the DET determined that some areas needed additional attention by

management to increase the rate of progress and ensure continued success. These areas included the need to resolve expeditiously continuing equipment problems affecting plant and personnel performance, to take more timely action to address human relations weaknesses to ensure long-term improvements in personnel performance, and to take prompt actions to establish an effective Independent Safety Engineering Group in order to facilitate improved problem resolution. Because of heavy reliance on contractor support, the DET was concerned that an excessive reduction in contractors in selected areas after the first refueling outage could affect the licensee's ability to accomplish engineering support tasks.

CEI responded to the findings and issues raised by the DET in a detailed and thorough manner. In general, the response included a discussion regarding the accuracy and validity of each finding or issue and a list of actions that had been taken or are planned to address the concern.

Brunswick Steam Electric Plant Diagnostic Evaluation

The Brunswick plant is composed of two 790-MWe boiling water reactors in Southport, North Carolina, operated by the Carolina Power and Light Co. (CP&L). Brunswick began commercial operation in the mid-1970s.

The DE, which was conducted from April 10 to May 5, 1989, concluded that the root causes of poor plant performance during the past few years were (1) inadequate corporate management, coincident with a period of past site management weakness; (2) CP&L management's failure to clearly define and communicate site goals, priorities, and expectations; (3) cultural issues, including the failure of CP&L management to review adequately and understand Brunswick's declining level of performance, and the lack of individual accountability and teamwork; (4) an ineffective root cause determination and corrective action program; and (5) an ineffective engineering design and technical support program.

The DET noted that many of the current problems at Brunswick were previously addressed in the 1982 Brunswick Improvement Program. Although short-term performance improvements were seen after implementation of this program in 1983, there was a general failure on the part of CP&L corporate and site management to ensure that improvements were sustained and broadened. Corporate oversight, leadership, and direction were lacking in the mid-1980s during a period of continued weaknesses in site management. The result, in part, was a subsequent decline in overall

performance and a general inability on the part of CP&L to evaluate Brunswick's performance accurately.

As a result of extensive reassignments of CP&L managers at the corporate office and at Brunswick, many managers in the Nuclear Generation Group had held their present positions for less than 6 months. This management team was in the process of defining and communicating safety performance goals and individual performance expectations and norms. Although the site-corporate relationship had not been clearly defined, site managers were implementing actions to improve the organizational culture by establishing individual accountability, making commitments to "do it right" at all levels within the Brunswick organization, and striving to improve communication and teamwork. However, these initial actions were not viewed with optimism by working-level CP&L personnel because these initiatives had not been reinforced by incorporation into site directives, procedures, and other administrative controls. In addition, there was a lack of confidence that these initiatives would succeed in view of management's failure to implement similar initiatives in the past.

Additionally, CP&L was conducting an organizational analysis to review the company's environment, its internal and external customers, and the manner in which business was conducted. The results of the organizational analysis were expected to be made known in the fall of 1989 and were expected to result in additional personnel and/or organizational changes at the site and at the corporate level of management. These anticipated changes and the associated internal review process were creating an increased level of anxiety in the company, but no more than would normally be expected for an organization undergoing changes of this magnitude and duration.

The DET concluded that although many of the changes discussed by CP&L managers were considered to be positive, only a few had been formally defined and implemented. The team found that several of the fundamental weaknesses addressed in the 1982 Brunswick Improvement Plan were still present and, thus, had not been adequately resolved. Specifically, the corrective action program lacked provisions for effective identification of problems and determination of root causes. Also, the implementation of quality program activities was weak. Audits were either ineffective in identifying weaknesses in areas such as engineering design or failed to receive adequate attention.

The DET found major weaknesses in the engineering design and technical support areas. For example, significant problems were identified during the design

review of the service water system, and several of these problems were similar to the findings identified during CP&L's 1987 self-assessment of the high-pressure coolant injection system. Longstanding weaknesses in plant configuration control and engineering support activities, as evidenced by these reviews, indicated that equipment problems would continue to be prevalent at Brunswick. Thus, the basis for confidence in the reliability and operability of safety and nonsafety equipment under credible off-normal conditions was brought into question. Persistent equipment problems have fostered an attitude of "living with" or "working around" chronic hardware deficiencies. Frequently, CP&L had focused its efforts on the "easy fix" rather than on timely and effective root cause determination and on permanent solutions to problems.

Overall, the DET concluded that recent management changes and initiatives were having a positive effect. Of particular note was the involvement and presence of senior site managers in the plant. The new management team was competent and capable of making the changes necessary to develop a safety culture and improve Brunswick's overall performance. The DET determined that several areas needed additional management attention, including (1) implementation of an effective corporate oversight program to provide leadership and direction and to monitor and assess accurately Brunswick performance; (2) definition of site safety goals, priorities, and expectations which are effectively communicated to and understood at all levels; (3) implementation and monitoring the effectiveness of actions to establish the desired culture at Brunswick; (4) implementation of an effective corrective action program having a lower threshold for problem identification and effective measures for root cause determination; and (5) implementation of an integrated program to correct engineering design and technical support weaknesses involving both equipment failures and weaknesses in support activities such as configuration control and safety evaluations.

The licensee responded positively to the DET's findings, committing to an Integrated Action Plan (IAP) that integrated actions resulting from both the DET report and CP&L initiatives. Under the plan, those activities that are expected to provide the most rapid and sustained improvement in Brunswick's safety performance are to be documented and tracked. The licensee offered to provide periodic updates regarding progress in implementing the IAP.

Arkansas Nuclear One Diagnostic Evaluation

The Arkansas Nuclear One (ANO) station consists of two pressurized water reactors of about 850 MWe each, located in Russellville, Arkansas, and operated by the Arkansas Power & Light Co. One unit has been in

commercial operation since 1974 and the other since 1980.

The DE, which was conducted from August 21 to September 15, 1989, found that ANO had several substantial management, organizational, and technical problems that were caused by a number of longstanding deficiencies, which over a period of time became more evident.

The DET concluded that the root causes of ANO's performance problems were (1) weak corporate leadership, oversight, and involvement, coincident with a period of site management weaknesses, that had resulted in poor cooperation and teamwork among ANO organizations and a lack of accountability and sense of ownership within the staff; (2) the lack of a clearly documented plant design basis that adversely affected corrective action programs; (3) inadequate maintenance and engineering support to the plant that adversely affected plant performance; and (4) a sense of complacency regarding plant performance and a willingness to live with material and equipment problems, both of which resulted from various cultural issues and influences, including a lack of outside experience with high industry performance standards, a minimum compliance with regulations versus a safety approach to problems, weak self-assessment and performance-monitoring efforts, and previous assessments by outside organizations that had not been sufficiently critical.

At the time of the DE, Arkansas Power & Light Co. and ANO were still in a period of organizational transition. The DET found that the structural changes were substantially complete but that members of the new organization were still learning to work together as a team and to deal with the numerous issues confronting them.

The DET found that corporate involvement in the oversight of plant activities had improved as indicated by the Nuclear Vice President's role and involvement in the management and organizational changes. The new site management team was more visible in the plants, was emphasizing personal accountability, and was committed to improved teamwork and communications. However, the DET found that some of these initiatives were not fully implemented and not all of the commitments had been effectively communicated to plant personnel. For example, clear guidance for entering Technical Specification limiting conditions for operations for maintenance activities had not been provided to the plant operators, and incorrect interpretations were identified.

The functional areas of maintenance, operations, and engineering support continued to be adversely affected by longstanding design-basis and configuration-control

problems. Although significant progress had been made, weaknesses still existed in the engineering design and technical support areas. For example, several design and operational problems were identified during the DET's review of the service water system, and many of these were similar to findings identified during ANO's 1989 self-assessment of the Unit 1 decay heat removal system. Programs existed or had been identified to address these problems, but in many cases their implementation had met with delays as a result of resource and scheduling conflicts.

Corrective and preventive maintenance support was found to be weak, which, among other things, adversely affected the plant material condition and operating performance. These maintenance weaknesses were caused by material control problems, including the lack of qualified spare parts; poor tracking and trending; poor communications and coordination with engineering and operations personnel; weak root cause determinations; and inefficiencies in the planning and scheduling of maintenance work.

The DET found that the new management team had taken steps to address the cultural issues and influences that had adversely affected plant performance in the past. For example, the new managers from outside the ANO organization were implementing policies and establishing performance standards that were having a positive effect on the staff. The licensee had also made progress in its efforts to improve communications with the NRC, for example, by emphasizing safety over the regulatory aspects of its operability determinations.

Assessments by outside organizations had not been sufficiently critical of the performance of ANO. The licensee's weak self-assessment and root cause analysis efforts, combined with the results of these outside assessments, contributed to a sense of complacency on the part of the licensee regarding its overall performance and the adequacy of any corrective actions. Although progress had been achieved, there was still evidence that plant staff and managers were willing to accept poor material conditions, marginally functional equipment, and degraded system performance. Repair efforts seemed to focus on restoring equipment to service at the expense of aggressive and effective corrective actions.

Strengths were noted in the experience level and the number of licensed operators, as well as in management's commitment to a quality training program. In addition, the DET concluded that, overall, the recent management changes and initiatives were having a positive effect, but that a number of areas needed increased management attention. These areas included (1) identification and resolution of equipment problems with the highest safety significance and potential impact on plant operations and operator performance;

(2) resolution of the plant design-basis and as-built configuration problems; (3) establishment of resource commitments and priorities to resolve more expeditiously longstanding maintenance, engineering, and material control problems; (4) implementation of better performance monitoring, self-assessment, and root cause analysis efforts; and (5) increased emphasis on teamwork, communications, and accountability among ANO organizations.

The NRC staff has evaluated the licensee's response to the DET findings.

Palo Verde Nuclear Generating Station Diagnostic Evaluation

The Palo Verde station has three 1,304-MWe pressurized water reactors that began commercial operation in 1986–1988. The units are located in Wintersburg, Arizona, and are operated by the Arizona Public Service Co. (APS).

The DE, conducted from November 6 to December 8, 1989, found that Palo Verde had several substantial management, organizational, and technical problems that were caused by a number of longstanding deficiencies, which over a period of time became more evident. Subsequent to licensing, operational events and the licensee's inadequate response to certain problems evidenced deficiencies in the areas of leadership, management involvement, teamwork, resource utilization, communications, accountability, creativity, technical expertise, ownership, motivation, work planning, work control, establishment of work priorities, problem identification, problem resolution, and corrective action.

The DET concluded that the root causes of Palo Verde's performance problems were (1) insufficient technical and management depth to support startup and operation of a three-unit facility; (2) the focusing during startup of management and technical resources on the next unit on line at the expense of the operational units, resulting in a backlog of technical and programmatic issues; and (3) the 1987 reorganization, which compounded management deficiencies rather than contributed to improvement.

At the time of the diagnostic evaluation, Palo Verde was still in a period of organizational renewal and transition, in part as a result of the licensee's bringing in a senior site management team with prior operations experience, reacting to findings of the NRC and industry groups, and initiating improvements to achieve excellence. However, a number of functional areas, in particular maintenance and engineering, needed increased attention to foster greater emphasis on management, ownership, and urgency.

The DET found positive attributes in engineering support, including a quality staff in the Engineering Evaluations Department (EED) and strong support for the Engineering Excellence Program and other improvement programs. Strengths included the excellent document retrieval capabilities and the design of electrical components and systems. However, engineering support was often untimely and inadequate as a result of management instability and deficiencies, strained resources, and poor communications. The engineering departments had expanded over the last 2 years, and thus many of the supervisory personnel were in their positions less than 1 year. Unclear and unrealistic guidance resulted in confused roles and responsibilities. The DET found systems engineers had too broad a work scope, and the anticipated work scope of the resident nuclear engineering group included overlapping duties and authorities that were similar to those already provided by the onsite and the corporate nuclear engineering departments.

Although it was the goal of the corporate Nuclear Engineering Department to reduce its reliance on contractor personnel support, the licensee had to increase greatly the number of contract personnel in an attempt to reduce the backlog of work, which was discovered in July 1989 to require approximately 400,000 man-hours of effort to resolve. The engineering evaluation request (EER) system was ineffective and was "overwhelming" the system engineers with paperwork. The team also reviewed many open EERs and determined that much of the time spent by engineering personnel to resolve various issues was a misuse of resources. The licensee was not effective in screening out frivolous or irrelevant EERs or in rerouting them to other organizations if they were not engineering related.

A new integrated, department-wide work prioritization system was being developed and implemented at the time of the DET site visit but was not yet effective. Because of outages at all three units, most of the EED's efforts involved reacting to whichever plant submitted the initial assistance request or to whomever applied the most pressure to obtain engineering support.

Although general housekeeping at the facility was a positive attribute, maintenance tasks were not always being performed in a timely, superior, and coordinated manner as a result of ineffective management, strained resources, and inadequate procedures and programs. A lack of coordination of maintenance activities resulted in delays in the performance of work and interfered with proper root cause analysis. Inadequate planning for and management of outages were factors in the lengthy refueling outages for Units 1 and 3, which exceeded 230 days each. No central person or organization was responsible for managing the outages. In addition, there was little pre-staging of parts for outages,

few tangible plans to deal with emergent work, and no cutoff date for submission of maintenance or modification activities to the outage schedule. Maintenance work orders were inconsistent in their level of detail and often contained errors. In addition, work packages and reference materials were inaccurate, cumbersome, and difficult to use, resulting in an increased probability of maintenance errors. Maintenance management also was not sufficiently aware of the personnel deficiency reporting systems and equipment failure data trending programs to trend maintenance activities effectively and to initiate corrective actions as appropriate.

Motor-operated valve (MOV) setpoints were not being adequately controlled because trained technicians had trouble using the MOV database drawing. Additional weaknesses were identified in the MOV program, including a lack of baseline data for many MOVs, a lack of detailed procedures for MOV disassembly and re-assembly, and a lack of proper documentation for MOV setpoints. Internal responses to industry experience reports in this area were incomplete and untimely.

In the area of quality programs, positive attributes included recent personnel and programmatic changes in quality assurance (QA) and a good vendor QA program. However, in some instances, the DET found deficiencies that were not being identified and corrective actions that were ineffective or slowly implemented as a result of programmatic weaknesses and insufficient support from line organizations. The audit program was compliance based, and was thus limited in scope and effectiveness. Numerous examples were found that illustrated that the line organization was not identifying problems, was slow in determining root causes, and was not fully addressing root causes in resolving deficiencies.

The Plant Review Board (PRB) was not functioning effectively and suffered from a lack of direction. Also, the PRB was making slow progress in developing its improvement program. Many of the DET assessments of this group were similar to those made by a contractor assessment of the PRB in the third quarter of 1989, indicating that weaknesses continue to exist.

In the operations area, positive attributes included knowledgeable and professional licensed and auxiliary

operators and good communications within the operations group at the working level. A strength was the use of mockups and computers in training. However, system misalignment events were being caused by programmatic weaknesses and inattention to detail, as evidenced by the breakdown of the independent verification process, the lack of maintenance of the system status prints and valve logs, and operator inattention. Communications between operations management and the staff were not effective, as evidenced by inconsistent guidance and poor implementation by the staff of the three units of two policy letters that were issued to reemphasize normal practices. The DET was concerned with the ability of the licensee to continue to apply resources necessary to develop, maintain, and improve the training programs because of the extensive use of contractors and their upcoming release from the facility.

In several instances, implementation of surveillance testing was hampered by the licensee's lack of attention to detail during the planning, performance, and review activities. The licensee had not made adequate progress in response to industry guidance on check valve testing.

Overall, the DET concluded that the new management team understood the major management, organizational, and technical issues that affected plant performance and had begun implementing improvement initiatives. At the time of the diagnostic evaluation, APS had more than 50 major initiatives in various stages of development and implementation. Although some progress by the licensee in resolving the issues was evident, many initiatives have not been in place long enough to achieve the desired results. In addition, the rate at which the licensee was resolving major issues was being limited by a number of factors, including insufficient top level improvement program integration; a lack of systematic and complete programs, plans, and implementing strategies for all issues; insufficient management oversight of improvement efforts; and organizational instability, uncertainty, and insecurity.

The licensee is developing its response to the DET's findings.

Diagnostic Team Members at the Palo Verde Plant

10 Light Water Reactor Plant Analyzer

AEOD is developing a plant analyzer that will be used to

- Evaluate the severity of recorded light water reactor (LWR) plant transients and accidents and those events resulting from plausible variations in transient scenarios.
- Train individuals at the Technical Training Center (TTC) in Chattanooga, Tennessee. The plant analyzer will enable trainees to understand and visualize plant response by providing information and display beyond that available in the control room.
- Train the NRC Operations Center staff. The plant analyzer also will be used as a diagnostic tool by safety team engineers.

10.1 Purpose of the LWR Plant Analyzer (LWRPA)

Every LWR plant differs in many respects from all other plants, even those of the same type (e.g., PWRs) and from the same manufacturer. Only a full-scope (and plant-specific) simulator can attempt to simulate faithfully all the plant features. Full-scope simulators are very useful for training that involves control room indicators, control devices, and annunciators. On the other hand, engineers surveying the safety of a variety of plants may not be too interested in faithful simulation of the plant's normal operational transients. When conducting "what if" studies, engineers are interested in severe transients in which the focus shifts from the particular control system response to the response of the plant protection system, emergency core cooling systems, and other systems built into the plant as part of the defense-in-depth concept. Safety engineers are also interested in testing the validity of the emergency operating procedures, which operators are trained to follow.

The LWRPA was designed to assist safety engineers in their work and to provide training for all types of engineers by focusing attention on the plant response to the assumed failures of plant components or systems or assumed operator errors rather than on the control room activities. The LWRPA foregoes simulation of the exact behavior of a particular plant during normal operation. Although the LWRPA currently models only a few typical LWRs, changes in setpoints can be introduced interactively to approximate more closely a particular plant. A user can interactively approximate the desired prevailing "initial condition" of a plant just before a postulated event, initiate an event, introduce a

great variety of component malfunctions and operator errors, and view the consequences of the event on a computer color monitor.

The LWRPA has the ability to display in real time most, if not all, of the variables that provide an understanding of the underlying processes in pressurized water reactors (PWRs) and boiling water reactors (BWRs). Time histories of up to 24 selected variables can be displayed on the color monitor, and time histories of the same number of variables can be saved for off-line plotting in the desired scale and units. Facilities also are available for alphanumeric output. Multiple layers of menus in conjunction with the use of a mouse pointing device expedite selections, changes during the run, and other functions.

10.2 Capability of the LWRPA

As the name "LWR plant analyzer" implies, it is able to analyze transients in PWRs and BWRs. Thermal-hydraulic analysis is based on the following conservation equations: three for mass (liquid, vapor, and non-condensable gas); two for energy, thus allowing for full non-equilibrium; and one momentum equation based on the drift flux principle, thus allowing for co-current and counter-current flow of fluid components. RETACT, the predecessor of the LWRPA, was extensively verified using data obtained from a variety of test facilities and their recorded plant transients and accidents.

RETACT was originally designed to address only thermal hydraulics and heat transfer within the primary coolant system and steam generators in PWRs, and the reactor and the steamlines in BWRs. The rest of the plant was simulated by Singer-Link. A plant analyzer, on the other hand, must be capable of addressing more of the plant than RETACT addressed and must be user-friendly. As many transients of interest take several hours of real time to evolve, fast running time is essential, preferably at relatively inexpensive work stations.

Currently, the LWRPA addresses thermal hydraulics and heat transfer in system components bounded by the main feedwater discharge valves and the turbine stop valve. Although the steam turbine is not simulated, facilities exist for specifying loads, load change rates, and turbine trip. Steam dump or turbine bypass systems are addressed. On the feedwater side, the steam generator or reactor vessel level controllers determine the feedwater flow rates.

Other factors addressed by the LWRPA include the following: reactor kinetics and control rods, steamline

facilities exist for specifying loads, load change rates, and turbine trip. Steam dump or turbine bypass systems are addressed. On the feedwater side, the steam generator or reactor vessel level controllers determine the feedwater flow rates.

Other factors addressed by the LWRPA include the following: reactor kinetics and control rods, steamline pressure control, pressurizer heater and spray control, pressurizer level control, pump speed in BWRs (for power control), the reactor protection system, all types of emergency core cooling systems, control of the main steam isolation valves, control of the turbine flow valves, the auxiliary feedwater flow control, and a simple containment.

The boron injection control system and the long-term reactivity effects as a result of fission gas (such as xenon) decay are not currently simulated. Plant startup operations and any operations at very low power also are not addressed unless they result from a reactor trip.

Metric units are used internally and in all outputs of the LWRPA. However, to facilitate interaction, nearly all inputs provided interactively before and during the run are specified in engineering units. Plots selected for hard copy can be shown either in metric units or engineering units.

The photographs illustrate two of several modes available for displaying the computed results.

As of January 1, 1990, the following plants have been modeled: R.E. Ginna (a two-loop Westinghouse PWR), Three Mile Island Unit 1 (a Babcock & Wilcox [B&W] PWR), Washington Nuclear Plant Unit 1/ Washington Public Power Supply System (a B&W PWR, simulated at the TTC), and Peach Bottom Unit 2 (a General Electric [GE] BWR/4). In preparation are the Standardized Nuclear Unit Power Plant System (a four-loop Westinghouse PWR, simulated at the TTC) and Black Fox (a GE BWR/6, also simulated at the TTC). One plant of Combustion Engineering design also will be modeled during 1990.

Stan Fabric of AEOD's Division of Operational Assessment Looks at One Type of Display of the LWR Plant Analyzer

Another Type of Display of the LWR Plant Analyzer

11 Committee To Review Generic Requirements

All generic requirements proposed by the NRC staff involving one or more classes of power reactors must be reviewed by the Committee To Review Generic Requirements (CRGR). The CRGR is made up of senior managers from various headquarters program offices and, on a rotational basis, one of the NRC regional offices. CRGR members do not represent their offices in the CRGR review function but instead express their own individual views in arriving at a consensus, an independent CRGR recommendation for the NRC Executive Director for Operations (EDO) as to whether proposed new generic requirements have sufficient merit in terms of safety and are justified in terms of cost. The Director, Office for Analysis and Evaluation of Operational Data (AEOD), serves as the CRGR Chairman, and the AEOD staff provides support for all the committee's activities.

There were 22 meetings of the CRGR in calendar year 1989. During these meetings, the CRGR addressed some 45 issues. Major issues considered by the CRGR during this period included the final rule on standardization and licensing of advanced reactors; the final rule on fitness for duty; the final rule on dry storage of spent fuel at reactor sites; the final rule on access authorization; the backfit appeal regarding the use of nuclear heat in boiling water reactor hydrostatic testing (first effort of this type); the proposed rule on maintenance; the final policy statement on maintenance; the proposed final resolution of Unresolved Safety Issue (USI) A-40, "Seismic Design Criteria"; and a number of bulletins and generic letters, including those pertaining to the prevention of inadvertent criticality during refueling of pressurized water reactors, pipe wall thin-

ning, Mark I containment venting, service water system problems, motor-operated valve testing, and guidance on implementation by licensees of individual plant examinations for severe accident vulnerability.

Other important initiatives of the CRGR and the CRGR staff during calendar year 1989 included the following:

1. Maintaining contact with industry to understand potential issues, including visiting a reactor site (Fort Calhoun Station) to obtain general feedback and meeting periodically with industry organizations to consider industry comments (from the Nuclear Management and Resources Council, the Nuclear Utility Backfitting and Reform Group, and the Boiling Water Reactor Owners Group)
2. Improving the clarity of generic communications to ensure that backfitting considerations are understood by licensees
3. Initiating CRGR Charter revisions to improve clarity and, as a result, to enhance the quality and completeness of review packages provided by the NRC staff
4. Reviewing critically the results of implementation of the backfit rule regarding proposed new generic requirements

The Director, AEOD, also oversees plant-specific backfitting activities of the NRC staff in the headquarters program offices and the regional offices.

Appendix A

Data on Plant Operational Experience

A-1 Performance Indicator Program and Data

A-2 Other Plant Operational Experience Data

A-3 New Plant Operational Data

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Appendix A-1

Performance Indicator Program and Data

Appendix A-1

Performance Indicator Program and Data

Introduction

The NRC Performance Indicator (PI) Program for Operating Commercial Nuclear Power Plants includes eight indicators: (1) the number of unplanned automatic scrams (trips) while a reactor is critical, (2) the number of selected safety system actuations (SSAs), (3) the number of significant events, (4) the number of safety system failures (SSFs), (5) forced outage rates (FORs), (6) the number of equipment-forced outages per 1,000 critical hours, (7) the collective radiation exposure per plant, and (8) cause code trends.

The data for five of these PIs - Items 1, 2, 4, 5, and 6, above—are available from the Trends and Patterns database maintained at the Idaho National Engineering Laboratory (INEL). The data for significant events (Item 3 above) are provided by the NRC's Office of Nuclear Reactor Regulation (NRR), and the data for collective radiation exposure (Item 7 above), collected on a quarterly basis, are obtained from the Institute of Nuclear Power Operations (INPO). The data for cause code trends (Item 8 above) are available from the Sequence Coding and Search System database maintained at the Oak Ridge National Laboratory (ORNL).

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 1988 and 1989, a listing of the significant events that occurred during 1989, and a table of critical hours for each plant for 1988 and 1989.

Since February 1987, AEOD has provided quarterly PI reports to the Commission and to NRC senior management. A copy of each report is also placed in the NRC's Public Document Room. Beginning with the issuance of the fourth-quarter calendar year 1989 PI report, the staff will routinely provide plant-specific information to licensee management.

Background

Since May 1986, 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 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 possible

candidate PIs for initial implementation. Subsequently, the staff deleted one of these, the corrective maintenance backlog PI, after considering industry comments about the proposed candidate PIs.

In October 1986, the NRC prepared a prototype report by expanding the trial program data to include 100 operating reactors and data for the first half of 1986. The staff documented the recommended program, the task group report, and the prototype report in 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. Both INPO and the NRC adopted common definitions for four indicators, but the values reported differ as a result of differences in the treatment of the reactor populations and differences in data compilation. The trends in performance shown by use of the indicators are consistent despite differences in specific values.

As mentioned in Section 3.1, during 1989 the staff proposed and the Commission approved the use of cause codes from licensee event reports (LERs) as an addition to the existing PI program. The staff recommendation was based on a trial program for the use of cause codes as a basis for a new PI. The trial program, which followed earlier efforts by the Office of Nuclear Regulatory Research, was completed in 1988. It included (1) manually coding the cause and corrective action information in LERs for events that occurred during the first 6 months of 1988, (2) developing a method to use, for a 3.5-year period, the cause data contained in the Sequence Coding and Search System to broaden the database to support trend analyses, (3) evaluating the validity of cause code data, and (4) evaluating the effectiveness of the use of cause codes as a basis for a new PI.

The results of the trial program showed that the cause code information provided additional perspective beyond that available from the indicators in the PI program in use at the time. The cause codes were of value for performance monitoring and diagnosis of performance problems, thereby providing another screening tool. For performance monitoring, the cause codes could identify plant performance outliers that were not identified by the existing set of indicators. For

diagnostic purposes, the cause codes provided a programmatic view of the trends in performance. The initiation of corrective action coding, to supplement cause codes, provided an area for future development. Cause code trends were incorporated in the PI report in August 1989 (the 89-2 PI report*).

Definitions of the Indicators (Also see Section 2 for further discussion of the indicators.)

Automatic Scrams While Critical (Scrams)

This indicator includes the unplanned automatic scrams (reactor protection system logic actuations) while the reactor is critical. This PI is similar to another indicator, "Unplanned Automatic Scrams While Critical," used by INPO. In addition, the number of automatic scrams while critical below 15-percent power per 1,000 critical hours and the number of automatic scrams while critical below 15-percent power are monitored.

Safety System Actuations (SSAs)

This indicator includes manual and automatic actuations (safety system logic actuations) of emergency core cooling systems (ECCSs), as well as actuations of emergency ac power systems as a result of loss of power to a vital bus. This PI is similar to another indicator, "Unplanned Safety System Actuations," used by INPO.

Significant Events (SEs)

This indicator includes events that the NRC staff identifies as meeting certain selection criteria. Examples of such events include degradation of important safety equipment; unexpected plant response to a transient; a major transient; discovery of a major condition not considered in the plant safety analysis; and degradation of fuel integrity, primary coolant pressure boundary, or important associated structures.

**Performance Indicators for Operating Commercial Nuclear Power Reactors*, Report for Second Quarter 1989, Data through June 30, 1989, Parts I and II, issued by AEOD in August 1989.

Safety System Failures (SSFs)

This indicator includes any event or condition that could prevent the fulfillment of the safety function of structures or systems. Twenty-six safety systems, subsystems, and components are monitored for this indicator.

Forced Outage Rate (FOR)

This indicator is similar to the one used by INPO and in the NRC "Grey Book" (NUREG-0020). It is the number of forced outage hours in a period divided by the sum of forced outage hours and generator on-line hours. This indicator is used only for plants that are in commercial operation.

Equipment-Forced Outages per 1,000 Critical Hours (EFO Rate)

This indicator is the inverse of the mean time between forced outages caused by equipment failures. The mean time is equal to the number of hours the reactor is critical in a period divided by the number of forced outages caused by equipment failures in that period. 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 Indian Point and Millstone, unit values at multi-unit 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. This indicator is identical to the one used by INPO.

Cause Code Trends

This indicator captures the plant's trends for licensed operator errors; other personnel errors; equipment failures (electronic piece-part or environmentally related failures); design, construction, installation, or fabrication problems; administrative control problems; and maintenance problems. Maintenance problems are further divided into subcategories of maintenance personnel error, test or calibration personnel error, maintenance equipment failure, and potential maintenance problems.

Table A-1.1 Number of Automatic Trips While the Reactor Is Critical—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	0	0	0	1	1	0	3
Arkansas 2	0	0	0	1	0	1	0	1
Beaver Valley 1	0	3	0	0	2	1	0	1
Beaver Valley 2	1	1	2	0	1	0	0	0
Big Rock Point	0	0	0	2	0	0	1	0
Braidwood 1	0	0	1	1	1	0	1	0
Braidwood 2	0	4	3	3	0	1	2	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	2	0	0	0	0
Brunswick 2	0	0	0	1	0	0	0	0
Byron 1	0	1	2	0	0	0	0	0
Byron 2	1	1	2	1	0	0	0	0
Callaway	2	2	1	0	0	1	0	0
Calvert Cliffs 1	0	0	2	0	0	0	0	0
Calvert Cliffs 2	1	1	0	0	0	0	0	0
Catawba 1	0	0	0	0	1	0	0	0
Catawba 2	1	3	0	0	2	0	0	0
Clinton 1	0	1	0	1	0	1	0	0
Cook 1	1	0	0	2	2	0	0	0
Cook 2	0	0	0	0	0	0	1	0
Cooper Station	1	0	1	0	1	0	1	1
Crystal River 3	1	0	0	1	0	1	0	0
Davis-Besse	0	0	0	1	1	1	0	0
Diablo Canyon 1	1	0	4	0	0	0	0	1
Diablo Canyon 2	1	0	1	0	0	1	0	0
Dresden 2	0	0	0	0	1	0	1	0
Dresden 3	0	0	0	1	2	1	0	0
Duane Arnold	0	0	1	0	2	1	1	0
Farley 1	0	0	0	1	0	0	0	1
Farley 2	0	0	0	0	0	2	0	3
Fermi 2	1	2	1	0	1	0	0	1
FitzPatrick	0	0	0	0	0	0	0	2
Fort Calhoun	0	0	0	0	0	0	0	0
Fort St. Vrain	0	1	0	0	0	0	0	NLI*
Ginna	1	1	0	0	0	1	0	0
Grand Gulf	3	0	2	1	0	1	2	1
Haddam Neck	2	1	0	0	0	0	0	0
Hatch 1	1	2	1	1	0	0	0	0

Footnotes at end of table.

Table A-1.1 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	1	3	1	0	0	0	1	0
Hope Creek	0	1	1	2	0	0	1	1
Indian Point 2	1	1	0	2	1	0	0	1
Indian Point 3	1	1	0	1	0	0	0	0
Kewaunee	1	2	0	0	0	0	0	1
LaSalle 1	0	0	0	0	1	0	0	0
LaSalle 2	1	0	0	0	0	0	1	0
Limerick 1	0	1	0	0	0	0	0	0
Limerick 2	NYC**	NYC	NYC	NYC	NYC	NYC	0	1
Maine Yankee	1	0	1	1	1	1	0	0
McGuire 1	2	1	0	0	0	0	1	0
McGuire 2	1	0	0	0	2	1	0	0
Millstone 1	1	0	0	0	0	2	0	1
Millstone 2	0	0	0	1	0	0	0	0
Millstone 3	1	1	0	2	0	1	0	0
Monticello	0	0	0	1	0	1	0	1
Nine Mile Point 1	0	0	0	0	0	0	0	0
Nine Mile Point 2	3	3	1	0	0	2	1	3
North Anna 1	2	0	1	0	1	0	1	1
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	1	0	1	0	1	0
Oconee 2	0	0	1	0	2	1	0	0
Oconee 3	0	0	0	2	1	0	1	0
Oyster Creek	0	0	0	0	0	2	2	0
Palisades	0	0	0	0	0	0	1	0
Palo Verde 1	0	2	3	0	1	0	0	0
Palo Verde 2	0	0	0	1	1	0	1	1
Palo Verde 3	0	0	0	0	1	0	0	0
Peach Bottom 2	0	0	0	0	0	1	1	2
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry 1	1	7	0	0	0	0	0	0
Pilgrim	0	0	0	0	1	1	1	1
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	1	0	0	1	0	1	0
Prairie Island 1	0	0	0	0	0	0	1	0
Prairie Island 2	0	0	0	0	0	1	0	2
Quad Cities 1	0	0	0	1	0	1	0	0
Quad Cities 2	2	0	0	0	0	1	0	1
Rancho Seco	0	1	0	2	1	0	NLI	NLI
River Bend	3	0	2	0	2	0	1	1

Footnotes at end of table.

Table A-1.1 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	2	0	0	3	0	0	0
Salem 1	1	0	1	0	2	1	0	0
Salem 2	0	4	2	1	2	1	0	0
San Onofre 1	0	0	0	0	0	0	1	0
San Onofre 2	0	0	0	0	0	0	0	0
San Onofre 3	0	0	0	0	1	1	0	0
Seabrook	NYC	NYC	NYC	NYC	NYC	0	0	0
Sequoyah 1	0	0	0	2	1	0	0	1
Sequoyah 2	0	5	0	0	0	3	1	0
Shearon Harris	1	0	0	0	5	0	0	1
Shoreham	0	0	0	0	0	0	0	0
South Texas 1	1	0	3	0	2	0	1	0
South Texas 2	NYC	NYC	NYC	NYC	0	3	5	1
St. Lucie 1	1	1	1	0	0	0	2	0
St. Lucie 2	0	0	0	0	0	1	0	0
Summer	1	2	1	0	0	1	1	1
Surry 1	1	0	1	0	0	0	1	0
Surry 2	0	1	1	0	0	0	2	0
Susquehanna 1	1	1	0	0	2	0	0	1
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	0	1	0	0	0	1
Trojan	1	0	2	1	0	0	1	0
Turkey Point 3	0	0	0	0	1	0	0	0
Turkey Point 4	0	0	1	0	0	0	0	1
Vermont Yankee	0	2	1	0	0	0	0	0
Vogtle 1	2	1	3	0	0	1	0	1
Vogtle 2	NYC	NYC	NYC	NYC	0	3	1	2
Wash. Nuclear 2	1	0	0	0	1	1	2	0
Waterford 3	1	1	0	1	0	0	1	0
Wolf Creek	0	0	0	0	2	0	0	0
Yankee-Rowe	2	1	0	0	0	1	1	0
Zion 1	1	1	2	0	1	0	0	0
Zion 2	0	0	0	2	0	0	0	0
Total	58	71	55	44	59	48	47	43

*NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

**NYC means the plant is not yet critical.

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

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	0	0	0	0	1	0	0	2
Arkansas 2	2	1	1	0	0	4	0	0
Beaver Valley 1	0	1	0	0	0	1	0	1
Beaver Valley 2	2	0	0	0	1	1	0	0
Big Rock Point	1	0	0	1	0	0	0	0
Braidwood 1	1	0	0	1	0	1	0	1
Braidwood 2	1	0	0	0	0	0	0	0
Browns Ferry 1	1	1	0	3	1	0	0	0
Browns Ferry 2	0	0	0	2	1	0	0	0
Browns Ferry 3	0	0	0	1	0	0	0	0
Brunswick 1	0	0	0	1	0	0	0	1
Brunswick 2	0	0	0	1	1	3	0	2
Byron 1	0	0	0	0	0	0	0	0
Byron 2	0	0	0	0	1	0	0	0
Callaway	1	0	0	1	0	2	0	0
Calvert Cliffs 1	0	4	0	0	2	0	0	0
Calvert Cliffs 2	0	0	0	0	0	0	0	0
Catawba 1	1	0	0	0	2	0	0	0
Catawba 2	2	0	0	0	2	0	0	0
Clinton 1	0	0	1	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	1	2	2	0	0	2	0	0
Crystal River 3	0	0	0	2	0	4	0	1
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	1	1	0	0	0	0	1
Diablo Canyon 2	0	0	2	1	0	0	0	0
Dresden 2	0	0	0	0	1	0	0	0
Dresden 3	0	0	0	0	2	0	0	0
Duane Arnold	0	0	0	2	2	0	2	0
Farley 1	0	0	0	0	0	0	0	1
Farley 2	0	0	0	0	0	1	0	0
Fermi 2	1	2	0	0	1	0	1	1
FitzPatrick	0	0	0	1	0	0	0	1
Fort Calhoun	0	0	0	1	0	0	0	0
Fort St. Vrain	0	0	0	0	0	0	0	NLI*
Ginna	0	1	1	0	0	2	1	0
Grand Gulf	1	0	0	1	1	0	0	0
Haddam Neck	0	0	0	0	0	1	0	0
Hatch 1	0	0	1	2	0	0	0	0

Footnotes at end of table.

Table A-1.2 (cont.)

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	0	1	1	0	0	0	1	0
Hope Creek	0	2	2	2	0	1	0	0
Indian Point 2	1	0	0	0	0	0	0	0
Indian Point 3	0	0	0	1	1	0	0	0
Kewaunee	2	0	0	0	0	0	0	0
LaSalle 1	0	0	0	0	0	0	0	1
LaSalle 2	0	0	0	0	1	1	0	0
Limerick 1	0	0	0	0	0	0	0	0
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	0	2
Maine Yankee	0	0	1	2	0	0	0	0
McGuire 1	1	0	0	1	1	0	1	0
McGuire 2	0	1	0	0	0	0	0	0
Millstone 1	0	0	0	0	0	1	0	0
Millstone 2	3	0	0	1	0	1	0	0
Millstone 3	1	0	0	0	1	0	0	2
Monticello	0	0	0	0	0	0	0	0
Nine Mile Point 1	1	0	0	0	1	0	0	0
Nine Mile Point 2	3	0	1	2	3	1	0	0
North Anna 1	0	0	1	0	1	1	0	0
North Anna 2	0	0	1	0	0	1	0	0
Oconee 1	0	0	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	0	0	0	0	0	0
Oyster Creek	0	0	0	1	0	1	0	0
Palisades	0	0	0	0	0	0	0	2
Palo Verde 1	1	0	1	0	0	0	1	0
Palo Verde 2	2	0	1	0	2	0	2	0
Palo Verde 3	0	1	0	0	1	0	0	0
Peach Bottom 2	1	0	1	1	0	0	0	1
Peach Bottom 3	0	0	2	0	0	0	0	0
Perry 1	0	1	0	1	0	0	0	0
Pilgrim	0	0	1	0	1	2	1	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	2	0	0	1	0	0	1
Prairie Island 1	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	1
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	1	0	1	0	0	0	0
Rancho Seco	1	1	0	1	1	0	NLI	NLI
River Bend	1	0	2	0	0	2	0	1

Footnotes at end of table.

Table A-1.2 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	0	0	0	1	0	0	0
Salem 1	0	0	0	0	1	1	0	0
Salem 2	0	1	0	0	1	0	0	1
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	0	0	0	1
San Onofre 3	1	0	0	0	1	0	0	0
Seabrook	0	0	1	0	0	0	0	0
Sequoyah 1	0	1	0	0	0	0	0	0
Sequoyah 2	1	0	1	0	2	0	0	0
Shearon Harris	0	1	0	1	0	0	0	0
Shoreham	0	1	1	0	0	0	0	0
South Texas 1	4	0	1	2	1	0	0	0
South Texas 2	NYL	NYL	NYL	0	3	3	1	0
St. Lucie 1	1	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	1	0	1	0	0	1	0
Surry 1	0	0	1	0	2	2	0	1
Surry 2	1	1	0	0	1	2	1	0
Susquehanna 1	0	0	0	0	0	0	1	0
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	0	0	0	0	0	1
Trojan	0	0	1	0	0	0	0	0
Turkey Point 3	0	0	0	0	0	2	0	0
Turkey Point 4	0	0	0	0	0	1	1	0
Vermont Yankee	0	0	0	0	2	0	0	0
Vogtle 1	0	0	0	1	0	0	0	0
Vogtle 2	NYL	NYL	NYL	NYL	1	0	1	0
Wash. Nuclear 2	0	0	0	0	1	2	0	0
Waterford 3	0	0	0	0	1	0	0	2
Wolf Creek	0	0	0	0	0	0	0	0
Yankee-Rowe	1	1	0	1	0	0	1	0
Zion 1	0	0	1	0	0	0	0	0
Zion 2	0	0	0	1	0	0	0	0
Total	43	30	31	42	52	47	17	29

*NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

**NYL means the plant is not yet licensed.

Table A-1.3 Descriptions of Significant Events for 1989

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Arkansas 1	01/20/89	B&W	IV	A reactor trip occurred with multiple complications and leakage of a boundary check valve.*
Arkansas 1	09/12/89	B&W	IV	A discrepancy was found between plant electrical drawings and as-built electrical components and equipment.
Arkansas 1	12/05/89	B&W	IV	The reactor coolant system (RCS) boundary was found to be degraded. A deteriorated gasket led to severe boric acid corrosion of a control rod drive mechanism (CRDM) flange. Corrosion was severe enough to cause two of eight bolts on the CRDM to loosen.
Arkansas 2	04/18/89	CE	IV	An extraction steam pipe ruptured because of erosion, resulting in a reactor trip with complications.
Arkansas 2	06/26/89	CE	IV	A safety injection system (SIS) check valve, which isolates the low-pressure SIS from the RCS, malfunctioned. This resulted in the potential for overpressurizing the low-pressure SIS and an interfacing system loss-of-coolant accident.
Arkansas 2	09/12/89	CE	IV	A discrepancy was found between plant electrical drawings and as-built electrical components and equipment.
Braidwood 1	04/18/89	W	III	Inattentive employees were observed in the control room and at the auxiliary building radiological control exit.*
Braidwood 1	12/01/89	W	III	RCS water (68,000 gallons) leaked through the suction relief valve on the "B" train of the residual heat removal (RHR) system.*
Braidwood 2	04/18/90	W	III	Inattentive employees were observed in the control room and at the auxiliary building radiological control exit.*
Browns Ferry 1	01/31/89	GE	II	The licensee determined that upon loss of instrument air, the containment isolation valve (which is a butterfly valve) would fail in the open position, thus causing the loss of one of the containment isolation barriers.
Browns Ferry 1	02/08/89	GE	II	All parts of the emergency equipment cooling water system were not seismically qualified.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Browns Ferry 2	01/31/89	GE	II	The licensee determined that upon loss of instrument air, the containment isolation valve (which is a butterfly valve) would fail in the open position, thus causing the loss of one of the containment isolation barriers.
Browns Ferry 2	02/08/89	GE	II	All parts of the emergency equipment cooling water system were not seismically qualified.
Browns Ferry 3	01/31/89	GE	II	The licensee determined that upon loss of instrument air, the containment isolation valve (which is a butterfly valve) would fail in the open position, thus causing the loss of one of the containment isolation barriers.
Browns Ferry 3	02/08/89	GE	II	All parts of the emergency equipment cooling water system were not seismically qualified.
Brunswick 2	06/17/89	GE	II	Offsite power was lost for about 10 hours. All safety systems (including emergency diesel generators) functioned properly.
Bryon 2	01/16/89	W	III	Because of an incorrect procedure, a combustible gas mixture filled the accumulator.
Calvert Cliffs 2	03/01/89	CE	I	The turbine-driven auxiliary feed pump throttle trip valve failed, resulting in a control room fire.
Calvert Cliffs 2	05/05/89	CE	I	Boric acid buildup was found on the pressurizer heaters on the bottom of the pressurizer.
Clinton 1	06/01/89	GE	III	Rebuilt inboard and outboard seals on a recirculation pump failed.
Cook 2	08/14/89	W	III	Failure of a 120-V ac instrument bus caused a reactor trip. A partial loss of control room instrumentation occurred and one train of the plant protection system failed.*
Crystal River 3	01/17/89	B&W	II	A reactor coolant pump shaft failed.
Crystal River 3	06/16/89	B&W	II	Loop and Class 1E electrical power distribution anomalies were found.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Crystal River 3	10/26/89	B&W	II	The licensee determined that high-pressure injection line flow instrumentation was not sufficiently accurate for operators to correctly balance loop injection flows in accordance with the plant emergency operating procedures.
Diablo Canyon 1	01/17/89	W	V	The potential existed for the same problem that occurred at Diablo Canyon 2 on 1/17/89.
Diablo Canyon 2	01/17/89	W	V	An unrecognized inoperability of the auxiliary feedwater (AFW) system occurred as a result of isolating one of two steam supplies to AFW pump 2-1.
Dresden 2	01/30/89	GE	III	Balls were missing in two control rod drive accumulator charging water check valves.
Dresden 2	10/23/89	GE	III	Leakage of feedwater into the high-pressure coolant injection system through the injection valves resulted in water hammer and the potential of thermal stratification and steam binding.*
Dresden 3	10/23/89	GE	III	Because of design similarities, the potential existed for the same problem described in the 10/23/89 Dresden 2 event.*
Fermi 2	01/04/89	GE	III	The anticipated transient without scram system recirculation pump trip breaker failed to open.
Grand Gulf	08/14/89	GE	II	The failure of the condenser boot seal resulted in a reactor trip, with complications. One main steam isolation valve failed to close on manual and automatic demand. One control rod failed to insert beyond position 08. The scram discharge volume failed to drain.
Haddam Neck	11/17/89	W	I	About 450 fuel rods failed because they were abraded and pierced by post-maintenance debris left in the vessel.
Indian Point 2	04/08/89	W	I	The redundant trains of service water system cables experienced steam damage.
Indian Point 2	05/07/89	W	I	The licensee determined that a single failure of a selector switch for a voltmeter could result in the loss of all ac buses.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
LaSalle 1	03/02/89	GE	III	A fault on the Unit 2 station auxiliary transformer resulted in a Unit 1 reactor trip, with subsequent malfunctions of equipment, such as the instrument air system, the process computer, the alternate reactor protection system (RPS) breakers, and a recirculation flow control valve.*
LaSalle 2	03/02/89	GE	III	A fault on the Unit 2 station auxiliary transformer resulted in a Unit 1 reactor trip with subsequent malfunctions of equipment, such as the instrument air system, the process computer, the alternate RPS breakers, and a recirculation flow control valve.*
Limerick 1	02/10/89	GE	I	Excessive corrosion of the fuel cladding was caused by crud-induced localized corrosion.
McGuire 1	03/07/89	W	II	A steam generator tube rupture occurred in the "B" steam generator with a maximum leak rate of about 540 to 600 gpm.*
McGuire 2	09/05/89	W	II	Overpressurization of the containment spray system resulted in a leak of reactor coolant and refueling water.*
Millstone 1	03/22/89	GE	I	The licensee determined that a high-energy line break inside containment could degrade containment integrity. The reactor building closed cooling water system would not remain a closed system.
Millstone 1	05/29/89	GE	I	New inboard and outboard seals on a recirculation pump failed.
Nine Mile Point 2	01/25/89	GE	I	Penetrations were found without seals and with inadequate seals. As a result, internal flooding could bypass watertight dams and affect operability of engineered safety features cabling.
Nine Mile Point 2	02/23/89	GE	I	The licensee determined that loss of offsite power with a single failure in one emergency diesel generator (EDG) could lead to loss of a second EDG as a result of a failure to isolate service water (SW) trains and subsequent EDG undercooling. This possibility is caused by a control logic design flaw; the two SW loops should be automatically isolated.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
North Anna 1	01/07/89	W	II	The instrument air system experienced contamination and degradation as a result of water and oil.
North Anna 1	02/25/89	W	II	A steam generator tube plug failed. The primary-to-secondary leak rate reached about 74 gpm. The plug failure was caused by primary water stress corrosion cracking.
North Anna 2	01/07/89	W	II	The instrument air system experienced contamination and degradation as a result of water and oil.
Oconee 1	01/03/89	B&W	II	A switchgear electrical fire, a loss of forced coolant flow, and an excessive cooldown rate were experienced.*
Oconee 1	06/07/89	B&W	II	An unanalyzed plant condition was discovered that could have resulted in the loss of all ac power to safety-related loads.
Oconee 2	06/07/89	B&W	II	An unanalyzed plant condition was discovered that could have resulted in the loss of all ac power to safety-related loads.
Oconee 3	1/12/89	B&W	II	The reactor building cooling unit was declared inoperable when two of the three units were found to be fouled, resulting in a heat transfer rate that was below safety analyses assumptions.
Oconee 3	06/07/89	B&W	II	An unanalyzed plant condition was discovered that could have resulted in the loss of all ac power to safety-related loads.
Palisades	11/21/89	CE	III	A pressurizer power-operated relief valve (PORV) (by Target Rock) spuriously opened when the motor-operated block valve was opened. The block valve could not be closed to isolate the open PORV.*
Palo Verde 3	03/03/89	CE	V	Following a load reject, Unit 3 experienced several equipment failures: failure of fast transfer to reactor coolant pumps, failure of atmospheric dump valves to open on loss of condenser, failure of several drain valves, and leakage in the reactor coolant pump seat.*
Palo Verde 3	09/06/89	CE	V	The licensee discovered improperly set ring settings on the Unit 3 steam safety valves. The licensee shut down Unit 2 but did not find a similar problem.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Perry 1	02/28/89	GE	III	Temperatures in excess of 330° F were calculated to have existed in the upper portion of the drywell. The region involved contains about 73 safety-related cables and 35 snubbers. The cables and snubbers had been environmentally qualified for 195° F and 300° F, respectively.
Pilgrim	01/10/89	GE	I	The licensee determined that upon loss of air, the air accumulators, which are undersized, deplete their air supply in about 50 minutes. This event could cause one of the containment isolation valves to fail in the open position, thus causing loss of one of the containment isolation barriers.
Pilgrim	04/12/89	GE	I	A reactor core isolation cooling (RCIC) injection block valve was inadvertently opened during a test, and an RCIC check valve leaked feedwater back through RCIC suction piping and out a relief valve.*
Point Beach 1	11/07/89	W	III	Circuit breakers were found without instantaneous trip devices to provide circuit protection against high-amperage short-duration faults. A seismic event could create fault conditions that could damage redundant batteries.
Point Beach 2	11/07/89	W	III	Circuit breakers were found without instantaneous trip devices to provide circuit protection against high-amperage short-duration faults. A seismic event could create fault conditions that could damage redundant batteries.
Quad Cities 1	04/17/89	GE	III	During testing of electromatic relief valve D, the valve failed to close. Operators manually tripped the reactor per procedures. The reactor cooldown rate exceeded the Technical Specification limit of 100° F during the first 2 hours of the blowdown.
Quad Cities 1	06/09/89	GE	III	Incorrect valves and valves installed backwards were found on the instrument air line. The six 18-inch butterfly valves would not close if instrument air was lost while the drywell was being purged. Containment integrity could not be ensured.
Quad Cities 2	06/09/89	GE	III	Incorrect valves and valves installed backwards were found on the instrument air line. The six 18-inch butterfly valves would not close if instrument air was lost while the drywell was being purged. Containment integrity could not be ensured.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Rancho Seco	01/31/89	B&W	V	Auxiliary feedwater system overpressurization occurred as a result of turbine/pump overspeed.
Rancho Seco	02/14/89	B&W	V	Reactor coolant pump seal injection line check valve was found to be inoperable.
River Bend	04/19/89	GE	IV	A freeze plug failure in the service water system caused loss of spent fuel cooling and shutdown cooling.*
River Bend	12/01/89	GE	IV	A second instance of multiple main steam isolation valve (MSIV) failures at River Bend occurred after corrective action was taken. MSIV solenoid valves failed to release.
Robinson 2	01/07/89	W	II	Hydrogen from the main generator cooling system was introduced to plant and instrument air systems as a result of personnel error.
Robinson 2	08/16/89	W	II	The net positive suction head was inadequate for the auxiliary feedwater pumps.*
Salem 1	05/20/89	W	I	A loss of RHR occurred as a result of inadvertent accumulator nitrogen discharge.
San Onofre 1	01/08/89	W	V	Fasteners on thermal shield support blocks were found broken.
San Onofre 1	03/02/89	W	V	A design deficiency was found in the EDG load sequencer logic.
Summer	05/28/89	W	II	The pressurizer safety valve opened spuriously.
Summer	08/25/89	W	II	The pressurizer code safety valve lifted spuriously, causing a rapid depressurization of the RCS.
Surry 1	01/26/89	W	II	Degraded power cable insulation was found on the inside recirculation spray pumps.
Surry 1	01/31/89	W	II	Numerous motor-operated valves (MOV) and operators were deficient, rendering them inoperable. These deficiencies were found during an engineering evaluation. Numerous safety systems may have been inoperable.

*Denotes events investigated by NRC Augmented Inspection Teams (AITs). See Section 8 and Table 8.1.

Table A-1.3 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Surry 1	03/10/89	W	II	Unqualified safety-related parts supplied by an equipment manufacturer other than the original manufacturer were discovered.
Surry 2	01/26/89	W	II	Degraded power cable insulation was found on the inside recirculation spray pumps.
Surry 2	01/31/89	W	II	Numerous MOVs and operators were deficient, rendering them inoperable. These deficiencies were found during an engineering evaluation. Numerous safety systems may have been inoperable.
Surry 2	03/10/89	W	II	Unqualified safety-related parts supplied by an equipment manufacturer other than the original manufacturer were discovered.
Trojan	04/09/89	W	V	An RHR isolation valve was inoperable as a result of a wiring error, resulting in degradation of pressure boundary protection.
Trojan	07/12/89	W	V	A number of items and debris were found in the containment sump, and the inner sump screen was missing.
Turkey Point 3	01/16/89	W	II	The licensee found leaks in three reactor core thimble guide tubes.
Vogtle 2	03/09/89	W	II	Two RHR cold-leg injection valves failed while a flow path to the refueling water storage tank was open.
Zion 1	10/22/89	W	III	Indications of surface and subsurface cracks were found by ultrasonic testing of the transition cone upper-girth weld of the steam generator.

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

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	0	0	0	2	1	0	1	1
Arkansas 2	0	0	1	0	0	2	1	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	0	0	0	0	0	0	0
Braidwood 1	1	0	0	0	0	1	0	1
Braidwood 2	1	0	0	0	0	1	0	0
Browns Ferry 1	0	0	0	0	2	0	0	0
Browns Ferry 2	0	0	0	0	2	0	0	0
Browns Ferry 3	0	1	0	0	2	0	0	0
Brunswick 1	1	0	1	1	0	0	0	0
Brunswick 2	3	0	1	1	0	1	0	0
Byron 1	0	0	1	0	0	0	0	0
Byron 2	0	0	0	0	1	0	0	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	1	0	0	0	0	0	0
Calvert Cliffs 2	0	1	0	0	1	1	0	0
Catawba 1	0	1	0	0	0	0	0	0
Catawba 2	1	1	0	0	0	0	0	0
Clinton 1	0	0	0	1	0	1	0	0
Cook 1	0	0	1	0	0	0	0	0
Cook 2	0	0	1	0	0	0	1	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	1	1	0	0	1	1	0	1
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	1	0	0	0
Diablo Canyon 2	0	0	0	1	1	0	0	0
Dresden 2	0	1	0	1	1	0	0	1
Dresden 3	0	0	0	0	0	0	0	1
Duane Arnold	0	0	0	1	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	1	1	0	1	0	0	0
FitzPatrick	0	0	1	1	0	0	0	0
Fort Calhoun	0	1	0	0	0	0	0	0
Fort St. Vrain	0	1	0	0	0	0	0	NLI*
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	1	0	0	0	1	0
Haddam Neck	0	1	0	0	0	0	0	1
Hatch 1	0	0	0	0	0	0	0	0

Footnotes at end of table.

Table A-1.4 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	0	0	0	0	0	0	0	0
Hope Creek	0	0	0	0	0	0	0	0
Indian Point 2	2	1	0	0	0	2	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	1	0	0	0
LaSalle 2	1	0	0	0	1	0	0	0
Limerick 1	0	0	0	2	1	0	0	0
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	0	0
Maine Yankee	0	0	1	0	0	0	0	0
McGuire 1	0	0	0	2	1	0	0	0
McGuire 2	0	0	0	1	0	0	1	0
Millstone 1	0	1	0	0	1	1	0	0
Millstone 2	0	1	0	1	0	0	0	0
Millstone 3	1	1	0	0	0	0	0	0
Monticello	0	0	0	0	0	0	0	0
Nine Mile Point 1	0	0	0	1	0	0	0	0
Nine Mile Point 2	1	0	0	0	2	0	0	0
North Anna 1	0	0	0	0	2	0	0	0
North Anna 2	0	0	0	0	1	0	0	0
Oconee 1	0	0	0	0	1	1	0	0
Oconee 2	0	1	0	0	0	1	0	0
Oconee 3	0	0	1	0	1	1	0	0
Oyster Creek	0	0	1	1	0	0	0	0
Palisades	0	0	1	0	0	0	0	1
Palo Verde 1	0	0	1	0	0	0	0	0
Palo Verde 2	0	0	0	0	0	0	0	0
Palo Verde 3	0	0	0	0	1	0	1	0
Peach Bottom 2	0	1	1	0	0	0	0	0
Peach Bottom 3	0	1	1	0	0	0	0	0
Perry 1	0	2	1	0	1	0	0	0
Pilgrim	0	0	0	0	1	1	0	0
Point Beach 1	0	1	0	0	0	0	0	1
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	2	0	0
Quad Cities 2	0	0	0	0	0	1	0	0
Rancho Seco	1	0	0	0	2	0	NLI	NLI
River Bend	0	0	1	0	0	1	0	1

Footnotes at end of table.

Table A-1.4 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0	0	0	0	1	0	1	0
Salem 1	0	0	0	0	0	1	0	0
Salem 2	0	1	0	0	0	0	0	0
San Onofre 1	0	0	0	2	2	0	0	0
San Onofre 2	0	0	0	1	0	0	0	0
San Onofre 3	0	0	0	1	0	0	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	0	0	0
Sequoyah 2	0	0	0	0	0	0	0	0
Shearon Harris	0	0	1	0	0	0	0	0
Shoreham	0	0	0	0	0	0	0	0
South Texas 1	0	1	0	0	0	0	0	0
South Texas 2	NYL	NYL	NYL	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	1	1	0
Surry 1	0	1	0	0	3	0	0	0
Surry 2	1	1	1	1	3	0	0	0
Susquehanna 1	0	0	0	0	0	0	0	0
Susquehanna 2	1	0	0	0	0	0	0	0
Three Mile Island 1	1	0	0	0	0	0	0	0
Trojan	0	0	1	0	0	1	1	0
Turkey Point 3	0	0	0	0	1	0	0	0
Turkey Point 4	0	0	0	0	0	0	0	0
Vermont Yankee	0	0	1	0	0	0	0	0
Vogtle 1	1	0	0	0	0	0	0	0
Vogtle 2	NYL	NYL	NYL	NYL	1	0	0	0
Wash. Nuclear 2	2	0	1	0	0	0	0	0
Waterford 3	0	1	0	0	0	0	0	0
Wolf Creek	0	0	0	2	0	0	0	0
Yankee-Rowe	0	0	0	0	0	0	0	0
Zion 1	0	0	0	1	0	0	0	1
Zion 2	0	0	0	0	0	0	0	0
Total	20	26	23	25	42	22	9	11

*NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

**NYL means the plant is not yet licensed.

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

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	0	2	2	1	0	3	2
Arkansas 2	1	1	0	2	1	0	0	1
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	0	0	0	1	0	0	0
Braidwood 1	0	0	0	0	0	0	0	0
Braidwood 2	0	0	1	0	1	0	0	1
Browns Ferry 1	2	0	2	1	1	2	3	1
Browns Ferry 2	2	0	2	1	2	3	3	1
Browns Ferry 3	2	1	2	1	1	2	3	1
Brunswick 1	2	5	4	5	2	0	0	1
Brunswick 2	3	3	3	2	2	1	2	0
Byron 1	0	0	0	0	0	0	0	0
Byron 2	0	0	0	0	0	0	0	0
Callaway	1	0	0	1	0	0	0	0
Calvert Cliffs 1	0	0	0	0	1	1	1	2
Calvert Cliffs 2	0	0	0	0	0	1	1	2
Catawba 1	2	0	0	2	1	0	1	0
Catawba 2	1	1	0	1	1	0	2	0
Clinton 1	1	3	0	3	2	1	0	0
Cook 1	0	0	1	1	0	0	0	0
Cook 2	0	1	0	1	0	0	0	0
Cooper Station	0	0	1	0	4	2	0	0
Crystal River 3	0	0	0	0	2	0	1	1
Davis-Besse	1	0	2	0	0	1	0	0
Diablo Canyon 1	0	0	0	1	1	0	0	0
Diablo Canyon 2	0	0	0	1	2	0	0	0
Dresden 2	0	4	2	0	2	1	1	2
Dresden 3	0	0	0	0	1	1	0	1
Duane Arnold	2	2	0	0	4	0	1	1
Farley 1	0	1	0	0	0	0	1	2
Farley 2	1	1	0	0	0	0	0	1
Fermi 2	4	1	3	0	3	0	2	2
FitzPatrick	2	1	1	3	4	3	2	4
Fort Calhoun	0	2	1	2	0	1	1	1
Fort St. Vrain	0	0	1	0	1	0	2	NLI*
Ginna	0	0	0	0	0	0	0	0
Grand Gulf	0	0	1	1	0	2	1	0
Haddam Neck	3	2	0	1	2	4	1	2
Hatch 1	1	1	1	4	1	1	0	0

Footnotes at end of table.

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	3	2	0	2	0	1	0	0
Hope Creek	0	2	3	2	0	3	0	0
Indian Point 2	0	1	0	1	1	0	1	0
Indian Point 3	0	1	0	0	0	0	0	0
Kewaunee	0	1	0	0	1	0	0	0
LaSalle 1	0	0	2	0	1	1	1	0
LaSalle 2	0	2	2	0	1	1	1	3
Limerick 1	2	1	0	1	8	4	3	1
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	3	2
Maine Yankee	0	1	0	1	0	0	0	0
McGuire 1	0	0	3	4	2	0	6	2
McGuire 2	0	1	1	1	1	0	5	2
Millstone 1	1	1	1	1	2	2	0	2
Millstone 2	1	0	0	0	0	0	0	1
Millstone 3	3	0	0	1	0	0	0	1
Monticello	0	0	0	0	1	3	0	1
Nine Mile Point 1	2	0	0	1	0	1	0	1
Nine Mile Point 2	1	1	4	2	0	0	0	0
North Anna 1	1	0	0	1	1	1	0	0
North Anna 2	0	1	0	1	0	1	0	0
Oconee 1	1	1	0	0	3	3	1	0
Oconee 2	1	2	0	0	2	3	1	0
Oconee 3	1	1	2	0	3	3	1	0
Oyster Creek	2	0	3	1	3	0	0	0
Palisades	0	1	1	2	1	0	0	0
Palo Verde 1	2	1	2	0	0	1	0	0
Palo Verde 2	2	0	2	0	0	1	0	0
Palo Verde 3	1	0	2	0	0	2	0	0
Peach Bottom 2	1	0	2	0	3	3	0	3
Peach Bottom 3	0	0	2	0	1	0	2	6
Perry 1	1	2	4	4	3	1	0	1
Pilgrim	1	0	1	0	2	0	3	1
Point Beach 1	1	0	2	1	1	3	0	1
Point Beach 2	0	0	2	1	0	1	0	1
Prairie Island 1	1	0	0	0	0	0	0	0
Prairie Island 2	1	0	0	1	0	0	0	0
Quad Cities 1	1	2	1	0	1	1	0	1
Quad Cities 2	2	3	0	0	0	1	0	1
Rancho Seco	0	1	0	0	4	0	NLI	NLI
River Bend	0	0	0	1	1	1	0	0

Footnotes at end of table.

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	1	2	2	0	1	1	1
Salem 1	1	1	0	1	2	2	0	0
Salem 2	0	2	2	1	0	0	0	1
San Onofre 1	1	1	0	1	4	0	2	0
San Onofre 2	1	1	2	1	0	0	0	0
San Onofre 3	1	1	2	1	0	0	0	0
Seabrook	0	0	0	1	1	0	0	1
Sequoyah 1	2	1	1	0	1	1	0	1
Sequoyah 2	5	0	2	0	1	1	0	1
Shearon Harris	2	3	3	0	0	0	1	0
Shoreham	0	0	0	0	0	0	0	1
South Texas 1	4	4	1	2	0	0	0	0
South Texas 2	NYL	NYL	NYL	0	1	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	1	0	0	2	0	1	0
Surry 1	0	2	2	2	2	0	1	0
Surry 2	0	0	2	4	2	0	1	1
Susquehanna 1	0	1	1	1	0	0	1	0
Susquehanna 2	2	0	1	0	1	0	0	0
Three Mile Island 1	0	0	0	0	0	0	0	1
Trojan	0	2	3	0	0	2	2	1
Turkey Point 3	0	2	3	1	1	0	0	1
Turkey Point 4	1	2	2	1	1	0	1	1
Vermont Yankee	1	0	0	0	3	1	2	0
Vogtle 1	0	1	0	0	3	0	0	0
Vogtle 2	NYL	NYL	NYL	NYL	2	0	0	0
Wash. Nuclear 2	1	3	0	0	1	4	3	2
Waterford 3	0	0	0	0	1	0	0	1
Wolf Creek	0	0	0	1	0	1	2	1
Yankee-Rowe	2	0	0	2	0	0	0	0
Zion 1	1	1	0	1	1	1	0	1
Zion 2	0	0	0	1	0	1	0	1
Total	91	89	101	90	121	84	78	77

*NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

**NYL means the plant is not yet licensed.

Table A-1.6 Forced Outage Rate (Percentage)—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	2	0	0	38	77	21	1	7
Arkansas 2	0	3	21	1	8	29	3	0
Beaver Valley 1	0	4	1	9	2	4	0	82
Beaver Valley 2	4	2	5	0	8	41	14	0
Big Rock Point	8	17	8	6	0	0	3	8
Braidwood 1	NYCO*	NYCO	11	7	4	4	2	0
Braidwood 2	NYCO	NYCO	NYCO	18	0	2	4	0
Browns Ferry 1	100	100	100	100	100	100	100	100
Browns Ferry 2	100	100	100	100	100	100	100	100
Browns Ferry 3	100	100	100	100	100	100	100	100
Brunswick 1	0	0	9	5	0	26	7	0
Brunswick 2	0	20	0	4	0	12	0	0
Byron 1	0	4	4	0	1	0	0	0
Byron 2	1	4	2	1	0	8	0	8
Callaway	6	5	1	0	0	3	2	0
Calvert Cliffs 1	1	0	4	2	4	0	0	0
Calvert Cliffs 2	1	6	0	0	11	0	0	0
Catawba 1	15	0	16	0	9	6	1	12
Catawba 2	60	7	8	3	9	18	1	0
Clinton 1	0	10	2	15	0	85	21	0
Cook 1	1	0	0	6	1	0	4	0
Cook 2	0	0	0	0	0	0	5	0
Cooper Station	17	0	5	0	12	0	3	6
Crystal River 3	5	0	0	2	0	59	45	13
Davis-Besse	0	0	0	15	6	1	0	0
Diablo Canyon 1	4	0	4	0	0	0	0	1
Diablo Canyon 2	3	0	33	0	0	4	9	11
Dresden 2	0	0	0	0	9	0	2	4
Dresden 3	0	0	0	0	7	4	0	0
Duane Arnold	0	0	3	100	23	5	3	45
Farley 1	0	2	0	1	0	0	0	6
Farley 2	0	0	0	0	0	13	7	3
Fermi 2	2	0	61	8	28	0	2	20
FitzPatrick	0	0	0	42	0	0	0	14
Fort Calhoun	0	0	0	0	0	9	5	0
Fort St. Vrain	4	42	0	0	100	29	48	NLI**
Ginna	17	4	1	0	1	12	14	0
Grand Gulf	11	0	7	1	0	0	11	3
Haddam Neck	0	0	0	0	0	0	0	0
Hatch 1	2	44	2	18	0	0	0	0

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	6	45	2	1	0	0	1	5
Hope Creek	0	5	3	11	0	0	3	3
Indian Point 2	2	2	7	4	1	0	0	2
Indian Point 3	1	7	0	43	0	8	0	5
Kewaunee	0	3	4	0	0	0	0	1
LaSalle 1	0	0	7	0	4	0	0	0
LaSalle 2	10	0	3	4	0	0	34	15
Limerick 1	0	14	0	0	0	0	0	0
Limerick 2	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
Maine Yankee	1	0	9	34	11	1	0	12
McGuire 1	2	2	0	0	27	43	2	0
McGuire 2	1	0	3	0	2	1	0	0
Millstone 1	2	0	0	6	0	11	0	5
Millstone 2	0	12	0	2	0	0	0	0
Millstone 3	3	16	0	23	13	15	0	10
Monticello	0	0	0	1	0	5	0	2
Nine Mile Point 1	100	100	100	100	100	100	100	100
Nine Mile Point 2	NYCO	19	23	0	0	12	1	49
North Anna 1	37	0	6	0	10	0	2	17
North Anna 2	0	0	0	0	0	0	0	0
Oconee 1	0	0	2	0	11	0	1	6
Oconee 2	0	3	0	0	2	7	0	10
Oconee 3	0	24	0	1	4	0	0	1
Oyster Creek	0	0	37	100	85	55	24	9
Palisades	29	5	0	61	34	0	3	69
Palo Verde 1	69	9	60	0	30	100	0	0
Palo Verde 2	0	0	0	9	17	0	17	41
Palo Verde 3	0	0	20	0	31	0	0	97
Peach Bottom 2	0	0	0	0	0	6	8	18
Peach Bottom 3	0	0	0	0	0	0	0	2
Perry 1	7	35	14	0	2	0	1	0
Pilgrim	0	0	0	0	29	42	17	7
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	0	1	0	0	3	2	1	0
Prairie Island 1	0	0	3	0	0	0	1	0
Prairie Island 2	0	0	0	4	0	1	0	8
Quad Cities 1	0	14	0	4	0	13	6	3
Quad Cities 2	7	0	24	1	3	4	5	3
Rancho Seco	100	0	0	32	64	9	NLI	NLI
River Bend	9	0	6	2	9	77	6	2

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	47	8	23	14	7	14	43	91
Salem 1	5	0	8	0	12	71	0	12
Salem 2	0	8	7	72	23	14	0	0
San Onofre 1	0	0	0	0	0	87	25	3
San Onofre 2	0	0	3	0	34	28	0	23
San Onofre 3	14	0	6	0	3	12	9	0
Seabrook	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
Sequoyah 1	100	100	100	87	3	0	0	3
Sequoyah 2	100	69	0	0	0	16	5	0
Shearon Harris	9	0	0	5	6	0	0	2
Shoreham	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
South Texas 1	NYCO	NYCO	20	8	13	0	8	15
South Texas 2	NYCO	NYCO	NYCO	NYCO	NYCO	0	17	34
St. Lucie 1	1	1	8	0	0	0	1	1
St. Lucie 2	0	0	0	0	0	2	5	0
Summer	2	11	3	0	28	19	20	8
Surry 1	4	0	26	100	100	100	8	1
Surry 2	5	45	0	0	0	0	34	47
Susquehanna 1	7	12	0	0	22	0	4	2
Susquehanna 2	0	4	0	0	7	0	0	0
Three Mile Island 1	3	0	19	18	0	0	0	2
Trojan	6	0	10	20	0	0	13	0
Turkey Point 3	58	0	0	99	45	0	1	0
Turkey Point 4	19	35	3	0	0	7	24	24
Vermont Yankee	0	1	11	0	0	0	0	0
Vogtle 1	26	6	5	9	12	3	2	5
Vogtle 2	NYCO	NYCO	NYCO	NYCO	NYCO	4	1	5
Wash. Nuclear 2	32	0	7	10	4	0	11	0
Waterford 3	5	5	0	2	3	0	6	2
Wolf Creek	31	0	0	0	2	0	0	0
Yankee-Rowe	4	2	0	0	1	4	14	0
Zion 1	1	0	7	11	31	0	9	0
Zion 2	0	0	0	11	21	1	0	3

*NYCO means the plant is not yet in commercial operation.

**NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

Table A-1.7 Equipment-Forced Outages/1000 Critical Hours—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	0.46	0.00	0.00	2.38	1.92	0.57	0.91	0.00
Arkansas 2	0.00	0.95	1.13	0.00	0.50	1.29	0.00	0.95
Beaver Valley 1	1.30	1.43	0.00	0.50	0.47	0.00	0.00	6.72
Beaver Valley 2	1.12	0.92	0.47	0.00	0.59	1.99	0.00	0.00
Big Rock Point	1.00	4.43	0.48	0.95	0.00	0.00	0.87	0.49
Braidwood 1	NYCO*	NYCO	0.00	0.95	1.21	0.51	0.00	0.00
Braidwood 2	NYCO	NYCO	NYCO	1.98	0.00	0.46	0.00	0.00
Browns Ferry 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Browns Ferry 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Browns Ferry 3	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Brunswick 1	0.00	0.00	0.49	1.04	0.00	0.66	0.00	0.00
Brunswick 2	0.00	2.04	0.00	0.46	0.00	0.52	0.00	0.00
Byron 1	0.00	0.67	1.33	0.00	0.47	0.00	0.00	0.00
Byron 2	0.46	1.40	0.92	0.46	0.00	0.49	0.00	0.49
Callaway	0.00	1.59	0.00	0.00	0.00	1.08	0.00	0.00
Calvert Cliffs 1	0.00	0.00	0.47	1.06	1.37	0.00	0.00	0.00
Calvert Cliffs 2	0.00	0.49	0.00	0.00	0.57	0.00	0.00	0.00
Catawba 1	1.05	0.00	1.19	0.00	3.10	0.99	0.46	1.52
Catawba 2	5.30	3.20	1.44	1.39	3.28	0.00	0.45	0.00
Clinton 1	0.00	0.71	0.46	0.52	0.00	4.69	1.10	0.00
Cook 1	0.00	0.00	0.00	0.95	0.00	0.00	0.00	0.00
Cook 2	0.00	0.00	0.00	0.00	0.00	0.00	0.47	0.00
Cooper Station	0.77	0.00	0.47	0.00	0.52	0.00	0.00	0.48
Crysal River 3	0.52	0.00	0.00	0.00	0.00	3.58	0.00	0.00
Davis-Besse	0.00	0.00	0.00	2.15	0.51	0.46	0.00	0.00
Diablo Canyon 1	0.00	0.00	0.52	0.00	0.00	0.00	0.00	0.00
Diablo Canyon 2	0.47	0.00	1.59	0.00	0.00	0.51	0.98	0.50
Dresden 2	0.00	0.53	0.00	0.00	0.00	0.00	0.46	0.00
Dresden 3	0.00	0.00	0.00	0.00	1.47	1.29	0.00	0.00
Duane Arnold	0.00	0.00	0.48	5.80	1.72	0.95	0.00	0.77
Farley 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Farley 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.92
Fermi 2	0.00	0.00	3.02	0.00	0.53	0.00	0.00	0.00
FitzPatrick	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.06
Fort Calhoun	0.00	0.00	0.00	0.00	0.00	0.00	0.47	0.00
Fort St. Vrain	0.93	1.31	0.00	0.00	0.00	1.52	0.86	NLI**
Ginna	0.86	0.47	0.46	0.00	0.55	0.00	0.52	0.00
Grand Gulf	1.52	0.00	0.93	0.00	0.00	0.00	0.50	0.46
Haddam Neck	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Hatch 1	0.47	0.00	0.47	1.90	0.00	0.00	0.00	0.00

Footnotes at end of table.

Table A-1.7 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	0.00	0.00	0.00	0.48	0.00	0.00	0.64	1.68
Hope Creek	0.00	1.06	0.46	1.50	0.57	0.00	0.56	0.93
Indian Point 2	0.60	0.51	0.00	0.92	0.55	0.00	0.00	0.61
Indian Point 3	0.93	0.60	0.00	1.57	0.00	4.64	0.00	0.00
Kewaunee	0.68	1.04	0.00	0.00	0.00	0.00	0.00	0.46
LaSalle 1	0.00	0.00	0.00	0.00	0.48	0.00	0.00	0.00
LaSalle 2	0.50	0.00	0.46	0.00	0.00	0.00	0.00	0.00
Limerick 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Limerick 2	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
Maine Yankee	0.46	0.00	0.96	3.94	1.02	0.00	0.00	0.54
McGuire 1	0.93	0.93	0.00	0.00	0.63	0.80	0.46	0.00
McGuire 2	0.92	0.00	2.52	0.45	0.93	0.46	0.00	0.00
Millstone 1	0.46	0.00	0.00	0.47	0.00	1.12	0.00	0.47
Millstone 2	0.00	0.66	0.00	0.00	0.00	0.00	0.00	0.00
Millstone 3	0.00	1.07	0.00	0.57	0.53	3.40	0.52	1.00
Monticello	0.00	0.00	0.00	0.46	0.00	0.00	0.00	0.00
Nine Mile Point 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Nine Mile Point 2	NYCO	1.56	1.68	0.00	0.00	0.99	0.54	1.52
North Anna 1	0.69	0.00	0.46	0.00	0.75	0.00	0.54	0.54
North Anna 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Oconee 1	0.00	0.00	0.46	0.00	1.84	0.00	0.00	0.48
Oconee 2	0.00	0.00	0.51	0.00	0.93	2.62	0.00	1.54
Oconee 3	0.00	1.80	0.84	0.91	0.96	0.00	0.45	1.66
Oyster Creek	0.00	0.00	0.70	0.00	15.75	0.94	1.09	0.00
Palisades	0.00	0.48	0.00	2.48	0.69	0.00	0.46	3.69
Palo Verde 1	1.72	1.00	2.05	0.00	0.66	0.00	0.00	0.00
Palo Verde 2	0.00	0.00	0.00	0.49	0.68	0.00	1.83	2.82
Palo Verde 3	0.00	0.00	0.00	0.00	1.81	0.00	0.00	9.67
Peach Bottom 2	0.00	0.00	0.00	0.00	0.00	1.47	0.96	1.06
Peach Bottom 3	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.25
Perry 1	1.31	1.96	1.02	0.00	0.80	0.00	0.65	0.00
Pilgrim	0.00	0.00	0.00	0.00	0.00	0.00	0.54	0.00
Point Beach 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Point Beach 2	0.00	0.46	0.00	0.00	0.47	0.00	0.50	0.00
Prairie Island 1	0.00	0.00	0.75	0.45	0.00	0.00	0.00	0.00
Prairie Island 2	0.00	0.00	0.00	0.47	0.00	0.66	0.00	0.98
Quad Cities 1	0.46	2.07	0.00	0.46	0.00	2.03	1.22	1.17
Quad Cities 2	0.97	0.00	0.59	0.00	0.48	1.41	0.49	0.00
Rancho Seco	0.00	0.00	0.00	2.59	3.32	0.00	NLI	NLI
River Bend	0.99	0.00	0.48	0.51	1.81	14.42	0.95	0.48

Footnotes at end of table.

Table A-1.7 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0.84	0.99	1.17	2.29	1.13	0.53	0.00	0.00
Salem 1	3.53	0.54	0.49	0.00	1.07	8.97	0.00	0.00
Salem 2	0.46	2.43	2.08	6.57	1.68	1.55	0.00	0.58
San Onofre 1	0.00	0.00	0.00	0.00	0.00	4.81	1.78	0.59
San Onofre 2	0.00	0.00	0.46	0.00	0.69	0.63	0.00	1.48
San Onofre 3	0.52	0.00	0.00	0.00	0.47	1.04	0.00	0.00
Seabrook	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
Sequoyah 1	0.00	0.00	0.00	5.27	0.00	0.00	0.00	0.46
Sequoyah 2	0.00	3.82	0.00	0.00	0.00	0.59	0.47	0.00
Shearon Harris	1.00	0.00	0.00	0.59	1.44	0.00	0.00	4.05
Shoreham	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO
South Texas 1	NYCO	NYCO	4.81	0.00	1.77	0.00	1.28	0.60
South Texas 2	NYCO	NYCO	NYCO	NYCO	NYCO	0.00	2.61	2.59
St. Lucie 1	0.46	0.46	2.95	0.00	0.00	0.00	0.00	0.45
St. Lucie 2	0.00	0.00	0.00	0.00	0.00	0.64	0.47	0.00
Summer	0.47	1.02	0.00	0.00	0.00	0.56	1.62	1.46
Surry 1	0.47	0.00	0.69	0.00	0.00	0.00	0.00	0.46
Surry 2	0.48	0.81	0.00	0.00	0.00	0.00	0.00	0.85
Susquehanna 1	0.00	0.51	0.00	0.00	1.16	0.00	0.00	0.00
Susquehanna 2	0.00	4.09	0.00	0.45	0.50	0.00	0.00	0.00
Three Mile Island 1	0.47	0.00	1.07	1.63	0.00	0.00	0.00	0.46
Trojan	0.49	0.00	1.11	0.56	0.00	0.00	1.00	0.00
Turkey Point 3	3.01	0.00	0.00	92.17	0.81	0.00	0.45	0.00
Turkey Point 4	0.56	1.36	0.55	0.00	0.00	1.78	2.31	1.62
Vermont Yankee	0.00	0.99	1.00	0.00	0.00	0.00	0.00	0.00
Vogtle 1	1.23	0.97	0.93	2.03	1.03	0.46	0.92	0.47
Vogtle 2	NYCO	NYCO	NYCO	NYCO	NYCO	1.05	0.45	1.46
Wash. Nuclear 2	1.97	0.00	0.53	0.50	0.49	0.00	1.08	0.00
Waterford 3	0.00	1.29	0.00	0.00	0.48	0.00	1.05	0.96
Wolf Creek	1.90	0.00	0.00	0.00	0.95	0.00	0.00	0.00
Yankee-Rowe	0.94	0.47	0.00	0.00	0.53	0.94	0.52	0.00
Zion 1	0.77	0.00	0.48	0.00	1.31	0.00	1.28	0.00
Zion 2	0.00	0.00	0.00	2.33	1.15	0.46	0.00	0.45

*NYCO means the plant is not yet in commercial operation.

**NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

Table A-1.8 Number of Critical Hours—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	2152	2183	1402	419	520	1744	2208	1527
Arkansas 2	1028	1056	1767	2181	2000	1553	2001	1057
Beaver Valley 1	771	2091	2191	2014	2119	2109	1510	149
Beaver Valley 2	1785	2163	2128	2209	1695	502	1902	2209
Big Rock Point	1997	226	2076	2096	2063	1678	1143	2037
Braidwood 1	86	1557	1996	2108	1648	1961	1494	484
Braidwood 2	192	829	1968	1807	1127	2152	2131	2209
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	1531	2137	2031	962	0	1519	2070	2161
Brunswick 2	24	1467	2001	2154	2160	1939	1681	0
Byron 1	2184	1494	1499	1309	2143	2183	2208	2209
Byron 2	2170	2141	2171	2194	806	2021	2208	2025
Callaway	2078	1883	2187	2055	2138	927	2208	2209
Calvert Cliffs 1	2184	194	2139	1881	1455	352	0	0
Calvert Cliffs 2	1358	2052	2208	2209	1766	0	0	0
Catawba 1	1898	2183	1681	1308	1289	2020	2197	1979
Catawba 2	378	1875	2080	2164	1526	505	2208	2209
Clinton 1	1875	1407	2181	1936	51	426	1817	1950
Cook 1	2131	2183	2010	2109	1810	0	2151	2209
Cook 2	2184	532	0	0	395	1863	2114	2209
Cooper Station	1297	332	2130	2209	1913	512	2164	2084
Crystal River 3	1913	2183	2208	1153	1016	279	1216	1763
Davis-Besse	1661	0	0	465	1962	2168	2208	2209
Diablo Canyon 1	1532	0	1942	2209	2160	2183	2208	638
Diablo Canyon 2	2117	2183	1258	632	2160	1946	2035	1996
Dresden 2	2184	1882	2208	700	929	2183	2177	1964
Dresden 3	2066	133	2208	1939	2040	1548	2208	1516
Duane Arnold	2184	2162	2091	173	1741	2103	1785	1293
Farley 1	2041	982	2208	2198	2160	2183	2016	1254
Farley 2	2184	2183	2208	2209	1995	959	2082	2169
Fermi 2	1134	1247	994	1950	1870	2183	1488	461
FitzPatrick	1802	2183	1386	690	2160	2183	1854	1890
Fort Calhoun	2184	2183	2143	0	1479	2022	2107	2209
Fort St. Vrain	2149	1530	119	0	193	1971	1168	NLI*
Ginna	1166	2111	2193	2209	1806	708	1925	2209
Grand Gulf	1970	2183	2154	2191	1829	1025	1987	2166
Haddam Neck	258	1502	2208	2209	2160	2183	1540	0
Hatch 1	2144	1210	2128	527	2160	2183	2208	2209

Footnotes at end of table.

Table A-1.8 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	616	1471	2178	2095	2160	2183	1559	594
Hope Creek	1045	1891	2159	1994	1758	2183	1798	1075
Indian Point 2	1673	1963	1692	2164	1811	0	2206	1627
Indian Point 3	2158	1670	2208	1277	817	215	2208	2112
Kewaunee	1478	1932	2137	2209	1208	1825	2208	2195
LaSalle 1	1730	0	1992	2209	2086	2183	1846	0
LaSalle 2	1983	2183	2159	323	1246	2183	1372	1892
Limerick 1	2184	1875	2208	2209	258	1110	2208	2209
Limerick 2	NYC**	NYC	NYC	NYC	NYC	NYC	541	1421
Maine Yankee	2168	2183	2091	508	1970	2172	2208	1861
McGuire 1	2146	2141	2208	289	1584	1256	2162	2209
McGuire 2	2166	1349	1590	2209	2149	2165	421	2209
Millstone 1	2155	2183	2208	2116	2160	890	2200	2128
Millstone 2	1037	1526	2208	2183	826	1560	2208	1433
Millstone 3	1359	1869	2208	1760	1900	882	1926	2009
Monticello	2184	2183	2208	2194	2160	2074	1157	1288
Nine Mile Point 1	0	0	0	0	0	0	0	0
Nine Mile Point 2	1448	1279	1782	16	0	2020	1869	1317
North Anna 1	1459	2183	2169	2209	1334	0	1840	1849
North Anna 2	2135	2183	2208	2209	1205	1297	2208	2209
Oconee 1	2184	2183	2193	2209	1090	2183	2016	2082
Oconee 2	797	2003	1980	2209	2144	1147	2152	1943
Oconee 3	2184	1667	1184	2195	2094	2183	2203	1204
Oyster Creek	2184	2183	1422	0	64	1069	1842	2040
Palisades	1574	2083	931	403	1445	2183	2152	271
Palo Verde 1	580	1999	976	2208	1522	0	0	0
Palo Verde 2	1202	311	2208	2029	1475	44	1643	1064
Palo Verde 3	2184	2184	1794	2208	1106	0	0	103
Peach Bottom 2	0	0	0	0	0	1359	2082	1890
Peach Bottom 3	0	0	0	0	0	0	0	801
Perry 1	1526	1530	1953	1931	1255	0	1534	2209
Pilgrim	0	0	0	0	969	1259	1855	1531
Point Beach 1	2184	1247	2208	2209	2160	1151	2208	2209
Point Beach 2	2184	2164	2208	1152	2144	2183	2004	912
Prairie Island 1	2102	2183	1341	2209	2160	2183	2189	2209
Prairie Island 2	1286	2183	2208	2137	2088	1521	2208	2036
Quad Cities 1	2184	1934	2208	2152	2160	1967	1640	854
Quad Cities 2	2061	380	1708	2144	2103	2124	2045	2163
Rancho Seco	4	1970	2029	1542	903	1452	NLI	NLI
River Bend	2017	2183	2104	1976	1656	208	2104	2084

Footnotes at end of table.

Table A-1.8 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1189	2016	1714	873	888	1886	1250	238
Salem 1	849	1839	2040	2209	1862	223	2208	1983
Salem 2	2184	2061	1444	304	1783	1939	2208	1720
San Onofre 1	1069	0	1354	1395	0	208	1687	1688
San Onofre 2	1826	2092	2160	2209	1449	1584	1518	676
San Onofre 3	1936	697	1090	2209	2110	1924	2009	2209
Seabrook	NYC	NYC	NYC	NYC	NYC	194	0	0
Sequoyah 1	0	0	0	380	2111	2183	2208	2169
Sequoyah 2	0	785	2208	2209	429	1687	2142	2086
Shearon Harris	1994	2183	700	1708	2078	2183	2208	493
Shoreham	0	0	0	0	3	0	0	0
South Texas 1	384	1181	1735	1873	1129	2183	783	1656
South Texas 2	NYC	NYC	NYC	NYC	411	1414	1918	771
St. Lucie 1	2163	2165	1017	2209	2160	2132	1789	2209
St. Lucie 2	2184	2183	2208	2209	742	1560	2116	2209
Summer	2149	1956	1832	131	1588	1779	1854	2055
Surry 1	2119	194	1443	0	0	0	2088	2184
Surry 2	2080	1242	1706	0	0	0	327	1177
Susquehanna 1	1909	1964	2208	2209	1721	539	2145	2187
Susquehanna 2	1560	245	2143	2209	1987	2183	1720	1027
Three Mile Island 1	2122	1869	932	1837	2160	2183	2208	2166
Trojan	2057	287	1803	1778	2160	120	1001	2142
Turkey Point 3	995	2183	2208	22	1231	167	2199	2209
Turkey Point 4	1780	1466	1804	0	0	562	1730	1855
Vermont Yankee	2184	2017	1994	2209	985	2014	2208	2209
Vogtle 1	1620	2069	2148	985	1944	2172	2175	2122
Vogtle 2	NYC	NYC	NYC	NYC	83	1793	2199	2059
Wash. Nuclear 2	1520	904	1895	1992	2028	770	1850	2209
Waterford 3	2084	777	2173	1590	2101	2183	1907	1042
Wolf Creek	1581	2183	2208	146	2115	2183	2208	2209
Yankee-Rowe	2117	2148	2208	1014	1891	2122	1915	2209
Zion 1	1306	1362	2099	1981	1527	2183	1559	0
Zion 2	2184	2183	2208	430	1734	2183	2208	2209

*NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

**NYC means the plant is not yet critical.

Table A-1.9 Collective Radiation Exposure (Man-rem)—Quarterly PI Data*

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	185	133	134	242	64	33	30	228
Arkansas 2	185	133	134	242	64	33	30	228
Beaver Valley 1	483	23	10	13	59	133	198	276
Beaver Valley 2					59	133	198	276
Big Rock Point	25	106	14	11	16	59	60	13
Braidwood 1	55	10	5	5	50	8	65	173
Braidwood 2								
Browns Ferry 1	85	110	120	87	53	35	38	69
Browns Ferry 2	85	110	120	87	53	35	38	69
Browns Ferry 3	85	110	120	87	53	35	38	69
Brunswick 1	448	65	44	316	258	30	104	471
Brunswick 2	448	65	44	316	258	30	104	471
Byron 1	4	87	156	191	66	4	7	10
Byron 2					66	4	7	10
Callaway	3	5	6	13	6	259	9	8
Calvert Cliffs 1	19	105	9	12	20	68	47	38
Calvert Cliffs 2	19	105	9	12	20	68	47	38
Catawba 1	141	8	30	98	79	72	6	7
Catawba 2	141	8	30	98	79	72	6	7
Clinton 1					260	81	14	18
Cook 1	7	109	184	74	95	138	10	17
Cook 2	7	109	184	74	95	138	10	17
Cooper Station	60	149	20	21	28	274	19	21
Crystal River 3	16	3	5	39	130	70	8	11
Davis-Besse	26	162	102	17	7	11	9	10
Diablo Canyon 1	69	154	64	143	3	4	4	202
Diablo Canyon 2	69	154	64	143	3	4	4	202
Dresden 2	56	253	46	343	370	46	43	104
Dresden 3	56	253	46	343	370	46	43	104
Duane Arnold	22	17	50	526	45	28	46	67
Farley 1	64	192	10	11	34	127	44	169
Farley 2	64	192	10	11	34	127	44	169
Fermi 2	38	32	18	15	11	15	66	142
FitzPatrick	139	87	224	335	58	52	178	89
Fort Calhoun	20	17	30	213	48	16	19	13
Fort St. Vrain	0	0	0	0	1	1	NLI**	NLI
Ginna	226	12	13	21	124	440	24	20
Grand Gulf	54	20	36	37	143	312	25	18
Haddam Neck	170	39	13	14	19	14	313	251
Hatch 1	198	73	69	361	57	41	100	136

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	198	73	69	361	57	41	100	136
Hope Creek	215	35	18	29	110	21	96	233
Indian Point 2	76	44	82	32	220	1045	29	140
Indian Point 3	7	39	4	45	454	403	10	9
Kewaunee	175	26	5	5	208	26	4	2
LaSalle 1	173	413	90	560	178	62	94	360
LaSalle 2	173	413	90	560	178	62	94	360
Limerick 1	17	16	12	9	162	56	17	29
Limerick 2								
Maine Yankee	20	23	18	665	21	11	19	37
McGuire 1	17	190	63	281	30	49	222	6
McGuire 2	17	190	63	281	30	49	222	6
Millstone 1	96	32	6	11	41	377	17	24
Millstone 2	476	160	28	55	470	176	27	241
Millstone 3	55	19	3	6	8	146	7	7
Monticello	47	24	23	16	21	27	273	186
Nine Mile Point 1	283	236	152	133	56	81	92	33
Nine Mile Point 2					56	81	92	33
North Anna 1	31	10	8	10	174	511	28	28
North Anna 2	31	10	8	10	174	511	28	28
Oconee 1	129	42	100	24	62	62	14	62
Oconee 2	129	42	100	24	62	62	14	62
Oconee 3	129	42	100	24	62	62	14	62
Oyster Creek	82	82	205	1131	569	149	82	118
Palisades	95	29	338	279	57	16	18	211
Palo Verde 1	152	148	15	11	25	87	87	24
Palo Verde 2	152	148	15	11	25	87	87	24
Palo Verde 3					25	87	87	24
Peach Bottom 2	360	434	214	151	58	65	152	84
Peach Bottom 3	360	434	214	151	58	65	152	84
Perry 1	20	17	25	30	258	425	64	35
Pilgrim	163	57	96	75	49	54	45	57
Point Beach 1	11	74	11	98	9	83	11	137
Point Beach 2	11	74	11	98	9	83	11	137
Prairie Island 1	45	5	46	3	6	34	3	4
Prairie Island 2	45	5	46	3	6	34	3	4
Quad Cities 1	55	286	38	36	39	33	100	278
Quad Cities 2	55	286	38	36	39	33	100	278
Rancho Seco	29	12	22	19	34	12	NLI	NLI
River Bend	17	21	20	42	106	376	21	25

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	69	22	34	441	116	31	31	36
Salem 1	44	15	109	92	7	144	4	13
Salem 2	44	15	109	92	7	144	4	13
San Onofre 1	84	85	47	62	77	33	45	53
San Onofre 2	84	85	47	62	77	33	45	53
San Onofre 3	84	85	47	62	77	33	45	53
Seabrook								
Sequoyah 1	67	124	131	19	280	16	12	21
Sequoyah 2	67	124	131	19	280	16	12	21
Shearon Harris					6	4	5	140
Shoreham								
South Texas 1								
South Texas 2								
St. Lucie 1	16	16	232	18	144	21	58	9
St. Lucie 2	16	16	232	18	144	21	58	9
Summer	8	8	28	464	27	8	8	8
Surry 1	37	352	116	287	118	139	95	49
Surry 2	37	352	116	287	118	139	95	49
Susquehanna 1	97	125	18	17	28	168	73	82
Susquehanna 2	97	125	18	17	28	168	73	82
Three Mile Island 1	12	63	116	20	12	13	14	15
Trojan	10	346	12	33	7	346	62	6
Turkey Point 3	52	26	30	228	116	52	24	32
Turkey Point 4	52	26	30	228	116	52	24	32
Vermont Yankee	32	24	31	38	194	34	15	54
Vogtle 1					11	5	9	7
Vogtle 2								
Wash. Nuclear 2	43	201	64	44	36	361	44	52
Waterford 3	10	201	12	36	9	5	38	194
Wolf Creek	60	4	3	229	5	2	1	7
Yankee-Rowe	9	10	13	195	23	10	18	11
Zion 1	198	179	12	241	42	12	82	176
Zion 2	198	179	12	241	42	12	82	176
Total	9445	10209	6319	13376	9611	10762	5667	9302

*The data in this table was obtained from INPO.

**NLI means the plant is no longer included in the PI data. In the case of Rancho Seco, the unit ceased commercial operation in June 1989 and all performance indicator data after 89-2 will be NLI. In the case of Fort St. Vrain, the unit ceased all operation in August 1989 and all performance indicator data after 89-3 will be NLI.

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

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	3	1	2	8	3	4	2	7
Arkansas 2	1	4	2	3	3	4	1	4
Beaver Valley 1	3	2	0	1	1	1	0	1
Beaver Valley 2	2	0	1	1	3	5	1	2
Big Rock Point	1	1	0	0	1	0	2	0
Braidwood 1	2	5	3	1	0	0	1	5
Braidwood 2	1	2	3	3	2	0	0	3
Browns Ferry 1	8	3	12	11	9	5	7	3
Browns Ferry 2	7	3	12	12	10	9	8	3
Browns Ferry 3	7	3	12	10	7	4	7	3
Brunswick 1	4	2	1	5	4	5	1	4
Brunswick 2	5	1	1	6	0	4	3	3
Byron 1	0	1	1	1	2	2	1	0
Byron 2	0	3	1	1	2	0	1	0
Callaway	2	1	1	2	0	2	1	0
Calvert Cliffs 1	0	1	2	2	3	1	3	3
Calvert Cliffs 2	1	0	1	0	6	3	1	3
Catawba 1	7	2	1	3	4	3	3	1
Catawba 2	10	6	2	4	7	5	5	1
Clinton 1	2	4	1	4	10	7	1	5
Cook 1	1	2	4	2	2	3	3	1
Cook 2	1	2	3	3	2	2	2	1
Cooper Station	2	4	0	0	4	4	0	0
Crystal River 3	5	2	2	7	5	6	4	3
Davis-Besse	3	4	5	3	2	4	1	0
Diablo Canyon 1	5	5	3	3	3	0	2	6
Diablo Canyon 2	2	3	1	9	4	1	2	6
Dresden 2	3	5	0	3	7	2	3	1
Dresden 3	3	10	0	1	1	5	1	3
Duane Arnold	0	2	3	2	3	0	2	0
Farley 1	1	4	0	1	0	1	0	1
Farley 2	2	2	0	0	1	5	1	2
Fermi 2	5	2	1	1	2	3	2	8
FitzPatrick	1	1	0	4	0	2	2	5
Fort Calhoun	3	2	3	5	5	6	2	2
Fort St. Vrain*								
Ginna	1	1	1	1	0	4	0	2
Grand Gulf	5	0	2	1	0	5	1	1
Haddam Neck	0	1	1	0	2	3	1	0
Hatch 1	2	4	2	3	3	2	2	6

Footnotes at end of table.

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	6	6	1	1	2	1	4	6
Hope Creek	5	3	3	2	5	4	2	5
Indian Point 2	0	2	3	0	0	2	1	1
Indian Point 3	1	1	0	0	5	0	0	1
Kewaunee	1	2	2	0	5	2	0	0
LaSalle 1	1	0	1	1	3	3	0	4
LaSalle 2	2	0	1	2	5	4	2	2
Limerick 1	3	4	0	4	10	10	5	1
Limerick 2	NYL**	NYL	NYL	NYL	NYL	3	4	2
Maine Yankee	2	0	1	0	1	0	0	0
McGuire 1	3	3	10	9	4	2	7	1
McGuire 2	2	6	8	4	4	2	9	3
Millstone 1	0	0	0	3	2	1	1	1
Millstone 2	2	0	0	0	0	1	1	1
Millstone 3	7	3	1	4	3	4	7	8
Monticello	1	1	0	0	2	3	1	6
Nine Mile Point 1	4	4	2	2	4	4	3	5
Nine Mile Point 2	8	5	12	7	5	6	6	6
North Anna 1	5	3	0	8	1	2	2	0
North Anna 2	2	4	0	7	1	3	2	0
Oconee 1	5	3	2	1	7	4	5	2
Oconee 2	4	3	1	1	3	5	3	2
Oconee 3	3	4	3	1	4	4	4	2
Oyster Creek	2	3	3	7	5	3	1	0
Palisades	5	2	3	0	3	3	3	3
Palo Verde 1	10	2	2	3	2	3	1	3
Palo Verde 2	4	4	2	2	1	3	1	0
Palo Verde 3	3	1	0	0	0	3	1	2
Peach Bottom 2	2	3	4	7	3	7	2	4
Peach Bottom 3	1	1	2	4	1	3	3	6
Perry 1	7	4	4	3	7	7	2	3
Pilgrim	5	2	3	3	6	5	3	3
Point Beach 1	1	1	1	2	1	3	0	1
Point Beach 2	1	2	0	1	0	3	0	3
Prairie Island 1	1	0	1	3	1	3	1	1
Prairie Island 2	1	0	0	1	1	3	1	1
Quad Cities 1	5	1	2	1	1	0	2	5
Quad Cities 2	6	8	3	6	0	1	0	3
Rancho Seco	2	4	2	3	2	0	NLI†	NLI
River Bend	5	0	3	3	3	8	1	6

Footnotes at end of table.

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	2	3	4	2	2	1	0	1
Salem 1	4	1	6	1	4	11	1	1
Salem 2	4	1	6	3	4	4	0	2
San Onofre 1	7	1	1	1	4	4	7	3
San Onofre 2	1	4	4	2	4	2	9	2
San Onofre 3	3	3	5	1	4	3	3	1
Seabrook	1	1	1	1	1	2	1	2
Sequoyah 1	15	5	6	6	6	4	6	6
Sequoyah 2	20	6	7	1	5	5	5	5
Shearon Harris	3	4	5	2	3	3	2	2
Shoreham	2	2	3	3	2	0	0	1
South Texas 1	16	4	3	5	5	2	4	2
South Texas 2	NYL	NYL	NYL	0	5	3	1	1
St. Lucie 1	0	0	1	0	0	1	3	1
St. Lucie 2	2	1	0	0	1	2	1	0
Summer	0	1	2	0	2	3	3	1
Surry 1	1	4	6	2	5	4	4	0
Surry 2	2	5	3	2	3	4	5	4
Susquehanna 1	0	0	3	1	5	8	1	0
Susquehanna 2	3	1	2	0	3	4	2	3
Three Mile Island 1	0	1	4	0	0	0	0	2
Trojan	4	8	4	4	3	3	8	2
Turkey Point 3	3	2	7	3	4	1	2	1
Turkey Point 4	2	2	4	3	2	2	4	1
Vermont Yankee	2	1	2	2	7	0	3	1
Vogtle 1	4	8	3	5	4	3	1	1
Vogtle 2	NYL	NYL	NYL	NYL	3	4	1	0
Wash. Nuclear 2	6	6	5	4	5	7	5	3
Waterford 3	3	12	1	4	2	1	5	1
Wolf Creek	0	2	3	0	3	1	3	0
Yankee-Rowe	2	1	0	3	1	1	3	0
Zion 1	3	3	4	2	1	2	4	4
Zion 2	2	2	2	10	4	2	0	3

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

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

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	0	0	0	4	1	1	1	2
Arkansas 2	0	2	0	1	0	1	0	2
Beaver Valley 1	1	1	2	2	1	0	1	0
Beaver Valley 2	0	0	1	3	1	2	0	0
Big Rock Point	2	0	0	0	0	0	0	0
Braidwood 1	1	0	1	0	0	1	0	0
Braidwood 2	1	1	0	0	0	0	0	1
Browns Ferry 1	1	1	1	3	3	0	1	0
Browns Ferry 2	1	1	1	3	3	0	1	0
Browns Ferry 3	1	1	1	3	3	0	1	0
Brunswick 1	0	0	0	4	4	1	0	0
Brunswick 2	1	1	0	2	0	0	1	0
Byron 1	0	0	1	0	0	0	0	0
Byron 2	0	2	1	0	1	0	0	0
Callaway	2	2	1	1	1	4	1	0
Calvert Cliffs 1	0	0	2	0	2	0	0	0
Calvert Cliffs 2	0	1	1	0	0	1	0	0
Catawba 1	3	0	0	0	2	0	1	0
Catawba 2	4	3	0	0	2	0	3	0
Clinton 1	1	2	0	1	4	2	3	1
Cook 1	1	0	0	0	1	0	0	1
Cook 2	0	0	0	0	1	2	0	0
Cooper Station	1	3	0	0	1	0	0	0
Crystal River 3	0	0	1	1	1	2	1	0
Davis-Besse	0	1	0	4	1	0	0	0
Diablo Canyon 1	1	1	1	1	0	0	0	1
Diablo Canyon 2	2	0	0	1	0	1	0	0
Dresden 2	0	1	1	1	2	0	0	0
Dresden 3	0	0	0	0	2	0	0	0
Duane Arnold	0	0	0	1	0	0	0	0
Farley 1	2	0	0	0	0	0	1	0
Farley 2	2	0	0	0	0	0	0	1
Fermi 2	6	1	0	2	1	0	1	3
FitzPatrick	0	0	0	0	1	0	0	0
Fort Calhoun	0	0	0	0	0	1	0	0
Fort St. Vrain*								
Ginna	1	0	1	0	0	2	0	0
Grand Gulf	0	0	1	0	0	1	2	1
Haddam Neck	1	1	0	0	0	0	0	0
Hatch 1	0	0	0	0	1	0	0	2

Footnotes at end of table.

Table A-1.11 (cont.)

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	0	1	0	0	0	0	0	0
Hope Creek	0	1	0	0	1	1	0	0
Indian Point 2	1	0	0	0	0	0	0	0
Indian Point 3	1	0	0	0	0	0	0	0
Kewaunee	1	1	2	0	0	0	0	0
LaSalle 1	0	0	0	0	2	0	0	1
LaSalle 2	0	0	1	1	2	0	2	0
Limerick 1	1	1	0	1	0	0	0	0
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	1	0
Maine Yankee	0	0	0	1	0	0	0	0
McGuire 1	1	1	0	3	0	1	1	0
McGuire 2	1	1	0	0	0	1	2	0
Millstone 1	0	0	0	0	0	1	0	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	2	2	0	0	3	2	0	1
Monticello	0	0	0	1	0	2	0	2
Nine Mile Point 1	0	0	0	0	1	0	0	0
Nine Mile Point 2	3	1	3	0	1	5	0	4
North Anna 1	3	0	0	0	0	0	0	0
North Anna 2	0	0	0	0	1	0	0	0
Oconee 1	0	0	0	0	2	0	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	0	0	0	0	1	1
Oyster Creek	1	0	2	1	0	2	0	1
Palisades	1	0	0	1	0	0	0	0
Palo Verde 1	1	3	2	0	1	0	0	1
Palo Verde 2	4	0	0	1	1	1	0	0
Palo Verde 3	0	0	0	0	1	1	1	0
Peach Bottom 2	1	0	0	0	0	2	0	1
Peach Bottom 3	0	1	0	0	0	0	0	0
Perry 1	1	5	1	0	1	3	0	0
Pilgrim	1	0	0	3	1	3	0	0
Point Beach 1	0	0	0	0	1	0	0	0
Point Beach 2	0	1	0	1	0	0	0	0
Prairie Island 1	0	1	0	1	0	0	1	1
Prairie Island 2	0	1	1	0	0	0	1	1
Quad Cities 1	0	1	1	0	0	0	0	0
Quad Cities 2	0	0	1	0	0	0	0	0
Rancho Seco	1	1	1	1	0	0	NLI†	NLI
River Bend	2	0	1	1	1	0	0	0

Footnotes at end of table.

Table A-1.11 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0	0	0	0	1	0	0	0
Salem 1	0	0	0	0	2	1	0	1
Salem 2	0	1	0	1	0	0	0	0
San Onofre 1	0	0	0	0	1	1	0	0
San Onofre 2	0	0	0	0	1	1	3	1
San Onofre 3	1	1	1	0	0	1	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	3	0	0	1	1	0	0	0
Sequoyah 2	6	5	0	0	1	2	0	0
Shearon Harris	0	1	1	2	0	0	0	1
Shoreham	0	0	0	0	1	0	1	0
South Texas 1	3	1	1	2	0	0	0	0
South Texas 2	NYL	NYL	NYL	0	0	3	0	0
St. Lucie 1	1	0	1	0	0	0	1	0
St. Lucie 2	1	1	0	0	2	1	1	0
Summer	0	2	0	0	0	0	0	0
Surry 1	0	0	1	1	1	2	4	1
Surry 2	0	1	2	1	0	1	4	3
Susquehanna 1	0	0	0	0	2	0	0	0
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	2	0	0	0	0	1
Trojan	0	0	0	1	0	0	2	1
Turkey Point 3	2	0	2	0	0	0	0	1
Turkey Point 4	1	0	4	0	0	2	0	1
Vermont Yankee	0	0	0	0	1	0	0	0
Vogtle 1	2	0	0	1	1	0	0	0
Vogtle 2	NYL	NYL	NYL	NYL	3	2	0	0
Wash. Nuclear 2	1	4	2	0	0	1	0	0
Waterford 3	0	1	1	0	0	0	1	1
Wolf Creek	1	1	0	1	3	0	1	0
Yankee-Rowe	0	0	0	0	0	0	1	0
Zion 1	2	1	1	0	1	0	0	1
Zion 2	2	1	0	3	1	1	0	0

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

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

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	0	3	5	1	3	0	6
Arkansas 2	5	2	1	1	1	2	1	5
Beaver Valley 1	0	2	1	0	0	2	0	3
Beaver Valley 2	2	0	1	0	1	5	0	1
Big Rock Point	0	0	0	2	0	1	4	0
Braidwood 1	3	1	1	1	0	0	0	3
Braidwood 2	5	1	4	1	1	0	0	1
Browns Ferry 1	4	2	3	6	2	3	3	1
Browns Ferry 2	4	3	3	10	3	4	7	1
Browns Ferry 3	3	3	4	6	3	3	3	1
Brunswick 1	1	1	2	3	1	1	0	2
Brunswick 2	3	0	0	5	2	1	1	1
Byron 1	0	0	0	0	1	0	0	0
Byron 2	2	1	1	1	0	0	0	0
Callaway	2	1	1	1	1	1	1	1
Calvert Cliffs 1	0	4	2	0	1	1	0	2
Calvert Cliffs 2	1	1	1	0	2	1	0	1
Catawba 1	5	2	1	1	4	1	1	0
Catawba 2	9	4	2	1	3	5	3	1
Clinton 1	3	3	3	3	3	4	1	2
Cook 1	0	0	4	1	1	2	3	0
Cook 2	0	0	1	1	3	2	1	0
Cooper Station	3	2	1	0	3	0	0	0
Crystal River 3	3	0	2	5	0	3	3	1
Davis-Besse	2	4	1	2	0	1	5	1
Diablo Canyon 1	4	2	4	3	2	0	0	2
Diablo Canyon 2	0	2	1	6	0	0	1	0
Dresden 2	1	3	1	1	2	1	0	1
Dresden 3	1	3	1	0	0	2	0	2
Duane Arnold	0	1	0	2	2	1	2	0
Farley 1	2	1	0	4	0	2	0	2
Farley 2	1	0	0	2	0	3	1	2
Fermi 2	4	2	6	1	2	1	3	7
FitzPatrick	1	0	1	0	0	3	2	4
Fort Calhoun	4	3	2	7	2	1	1	2
Fort St. Vrain*								
Ginna	2	0	1	0	0	1	0	0
Grand Gulf	4	0	2	1	1	0	1	1
Haddam Neck	4	2	2	0	1	1	3	2
Hatch 1	0	2	0	3	1	0	0	2

Footnotes at end of table.

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	4	1	2	1	1	0	1	2
Hope Creek	1	3	3	7	0	2	1	2
Indian Point 2	0	3	1	3	3	0	0	0
Indian Point 3	1	0	1	1	4	1	0	1
Kewaunee	0	1	2	0	1	1	0	0
LaSalle 1	0	4	0	3	2	1	0	2
LaSalle 2	1	5	0	0	4	0	0	2
Limerick 1	2	3	1	5	5	7	3	3
Limerick 2	NYL**	NYL	NYL	NYL	NYL	1	5	5
Maine Yankee	0	0	1	1	0	1	0	0
McGuire 1	2	2	4	6	1	2	6	1
McGuire 2	1	4	3	2	1	1	4	3
Millstone 1	2	0	0	1	0	1	2	1
Millstone 2	5	0	1	1	1	2	2	1
Millstone 3	5	0	2	2	0	3	1	3
Monticello	3	0	0	0	1	2	1	3
Nine Mile Point 1	4	0	0	1	0	1	2	1
Nine Mile Point 2	3	3	5	5	3	1	2	3
North Anna 1	3	0	0	0	4	2	0	0
North Anna 2	1	2	1	1	1	2	0	0
Oconee 1	1	1	2	0	3	0	3	0
Oconee 2	0	1	0	0	1	1	1	0
Oconee 3	0	2	1	0	1	0	3	1
Oyster Creek	0	2	2	2	1	2	3	0
Palisades	1	1	4	3	1	2	0	0
Palo Verde 1	2	1	2	1	0	1	2	2
Palo Verde 2	2	3	1	1	1	1	0	2
Palo Verde 3	2	0	3	0	0	1	3	2
Peach Bottom 2	1	5	2	4	1	1	4	1
Peach Bottom 3	1	4	3	2	1	0	3	1
Perry 1	4	4	3	2	3	3	4	2
Pilgrim	2	3	0	0	4	3	3	3
Point Beach 1	0	0	1	0	1	1	1	1
Point Beach 2	0	0	0	0	1	1	1	2
Prairie Island 1	0	1	1	2	0	1	2	1
Prairie Island 2	0	1	1	2	0	1	1	1
Quad Cities 1	0	1	2	0	0	2	2	2
Quad Cities 2	0	3	2	4	0	0	0	1
Rancho Seco	3	0	4	2	1	1	NLI†	NLI
River Bend	1	2	3	1	5	7	2	4

Footnotes at end of table.

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0	0	1	0	3	0	0	0
Salem 1	4	0	1	1	2	4	0	4
Salem 2	1	3	0	3	1	0	0	3
San Onofre 1	0	1	1	2	1	0	1	0
San Onofre 2	2	1	3	3	1	1	4	0
San Onofre 3	3	1	5	0	2	2	1	0
Seabrook	1	0	1	1	1	1	3	1
Sequoyah 1	5	3	6	6	5	2	3	3
Sequoyah 2	5	3	7	2	1	4	4	2
Shearon Harris	1	4	4	6	6	3	1	4
Shoreham	2	5	2	2	0	0	1	2
South Texas 1	4	4	3	2	2	1	1	3
South Texas 2	NYL	NYL	NYL	1	1	1	0	2
St. Lucie 1	0	0	3	0	0	2	1	2
St. Lucie 2	0	1	0	0	1	1	1	0
Summer	2	0	1	2	3	0	3	1
Surry 1	3	6	2	4	4	5	2	0
Surry 2	2	2	1	3	2	4	4	1
Susquehanna 1	2	2	2	3	2	6	1	1
Susquehanna 2	2	3	1	4	1	3	2	3
Three Mile Island 1	0	1	0	0	0	0	0	0
Trojan	2	5	7	4	2	5	3	2
Turkey Point 3	2	4	3	2	2	1	2	0
Turkey Point 4	1	5	1	1	0	0	4	0
Vermont Yankee	0	2	1	1	2	2	1	0
Vogtle 1	3	0	4	7	6	0	1	1
Vogtle 2	NYL	NYL	NYL	NYL	4	2	1	3
Wash. Nuclear 2	3	4	5	2	2	5	1	0
Waterford 3	2	10	1	3	1	2	1	2
Wolf Creek	2	0	0	4	2	0	1	0
Yankee-Rowe	0	1	0	0	3	2	3	1
Zion 1	2	0	2	1	1	1	4	3
Zion 2	2	0	1	6	3	0	1	1

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

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

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	3	1	4	7	3	8	2	15
Arkansas 2	3	5	5	5	3	8	0	11
Beaver Valley 1	5	4	2	2	3	4	2	8
Beaver Valley 2	4	1	1	3	7	12	3	2
Big Rock Point	1	2	1	2	2	1	4	0
Braidwood 1	7	6	4	3	3	2	4	8
Braidwood 2	5	8	9	4	3	1	3	2
Browns Ferry 1	10	5	11	20	10	6	8	5
Browns Ferry 2	8	7	11	22	12	12	12	7
Browns Ferry 3	8	6	12	19	9	5	9	5
Brunswick 1	6	3	3	10	5	10	1	5
Brunswick 2	10	2	3	9	4	6	7	7
Byron 1	0	1	3	2	3	2	1	0
Byron 2	2	2	5	2	2	1	1	0
Callaway	2	3	3	2	2	5	2	1
Calvert Cliffs 1	0	4	1	3	4	1	3	4
Calvert Cliffs 2	3	1	1	1	6	4	1	4
Catawba 1	8	1	1	5	9	4	4	1
Catawba 2	18	12	3	7	10	9	5	1
Clinton 1	8	6	5	7	11	10	4	6
Cook 1	4	2	5	3	5	5	4	1
Cook 2	2	3	3	3	9	5	2	2
Cooper Station	3	9	6	1	4	8	2	1
Crystal River 3	8	2	2	7	4	9	7	1
Davis-Besse	3	4	4	4	2	4	4	2
Diablo Canyon 1	10	6	6	4	3	0	2	8
Diablo Canyon 2	4	6	3	12	4	2	3	4
Dresden 2	4	7	3	7	10	3	10	4
Dresden 3	4	12	1	1	2	4	3	6
Duane Arnold	1	3	3	1	6	1	4	1
Farley 1	4	4	0	5	0	3	0	2
Farley 2	2	2	0	3	1	7	2	2
Fermi 2	11	8	9	3	7	4	8	11
FitzPatrick	2	3	2	3	2	5	4	6
Fort Calhoun	5	5	4	8	7	6	2	3
Fort St. Vrain*								
Ginna	3	1	3	0	0	5	3	3
Grand Gulf	8	0	2	2	1	3	4	3
Haddam Neck	5	4	2	2	1	3	6	3
Hatch 1	2	4	2	4	4	0	2	6

Footnotes at end of table.

Table A-1.13 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	9	10	4	2	3	0	4	7
Hope Creek	5	6	6	9	5	7	3	5
Indian Point 2	1	4	5	3	2	1	3	1
Indian Point 3	2	2	1	1	6	1	0	2
Kewaunee	2	3	2	0	6	4	1	1
LaSalle 1	3	7	4	5	13	9	3	5
LaSalle 2	4	9	5	7	12	8	3	6
Limerick 1	5	9	0	5	11	13	7	7
Limerick 2	NYL**	NYL	NYL	NYL	NYL	2	6	9
Maine Yankee	2	0	2	1	1	1	0	2
McGuire 1	5	7	11	14	4	3	11	2
McGuire 2	2	7	10	6	6	2	11	5
Millstone 1	2	0	2	4	1	8	3	2
Millstone 2	5	2	1	1	2	3	3	2
Millstone 3	9	5	2	5	2	5	7	8
Monticello	2	1	0	0	4	7	4	8
Nine Mile Point 1	6	3	1	2	4	1	3	2
Nine Mile Point 2	12	5	13	11	6	8	8	7
North Anna 1	13	1	1	5	5	6	2	1
North Anna 2	4	3	1	5	3	5	1	0
Oconee 1	5	0	3	1	6	2	3	0
Oconee 2	4	0	2	1	4	2	1	1
Oconee 3	3	2	3	2	4	2	3	2
Oyster Creek	2	3	8	7	4	4	5	0
Palisades	5	2	6	5	3	5	4	2
Palo Verde 1	9	2	3	3	2	4	5	2
Palo Verde 2	8	4	1	3	2	4	2	4
Palo Verde 3	5	1	3	0	3	4	4	2
Peach Bottom 2	5	7	4	8	4	9	5	7
Peach Bottom 3	2	7	6	4	2	3	5	7
Perry 1	10	11	8	5	9	9	3	3
Pilgrim	7	4	3	4	9	5	7	8
Point Beach 1	1	1	1	1	1	4	1	1
Point Beach 2	1	2	0	0	2	2	2	5
Prairie Island 1	0	2	2	6	0	3	6	6
Prairie Island 2	0	2	2	5	0	2	6	6
Quad Cities 1	5	3	3	1	2	5	3	4
Quad Cities 2	7	12	3	6	0	3	0	1
Rancho Seco	4	2	2	6	3	1	NLI†	NLI
River Bend	6	2	7	5	13	14	3	7

Footnotes at end of table.

Table A-1.13 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	3	5	3	2	5	2	0	1
Salem 1	5	1	4	1	9	12	1	7
Salem 2	5	4	5	5	6	6	0	4
San Onofre 1	5	2	2	3	4	4	6	1
San Onofre 2	4	6	6	4	5	2	8	2
San Onofre 3	6	2	8	2	5	2	2	1
Seabrook	2	1	1	2	5	3	4	2
Sequoyah 1	16	7	9	12	6	7	7	9
Sequoyah 2	22	10	12	2	4	12	6	7
Shearon Harris	5	7	10	6	6	5	2	6
Shoreham	3	6	5	3	5	0	1	3
South Texas 1	17	8	7	5	7	3	2	5
South Texas 2	NYL	NYL	NYL	0	8	5	3	2
St. Lucie 1	3	1	4	0	0	2	2	3
St. Lucie 2	2	1	0	0	1	1	3	1
Summer	2	2	1	2	4	5	4	2
Surry 1	7	11	11	7	6	10	8	2
Surry 2	7	7	5	6	4	9	13	5
Susquehanna 1	3	4	6	2	3	10	2	3
Susquehanna 2	7	3	5	3	2	3	4	7
Three Mile Island 1	0	1	2	1	0	0	0	3
Trojan	5	9	10	10	4	5	8	5
Turkey Point 3	5	7	9	3	5	1	3	2
Turkey Point 4	3	9	9	2	3	3	7	2
Vermont Yankee	3	4	1	3	11	3	4	1
Vogtle 1	8	9	7	11	9	3	4	2
Vogtle 2	NYL	NYL	NYL	NYL	8	8	3	3
Wash. Nuclear 2	4	11	7	5	3	13	5	2
Waterford 3	3	16	3	3	1	3	6	4
Wolf Creek	4	3	3	3	7	3	5	0
Yankee-Rowe	2	1	0	4	3	3	4	1
Zion 1	7	4	5	3	4	1	5	5
Zion 2	4	2	4	13	4	2	1	2

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Table A-1.14 Cause Codes—Maintenance Sub-Categories—Maintenance Personnel Error—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	1	2	5	1	3	0	4
Arkansas 2	2	3	1	0	1	3	0	5
Beaver Valley 1	2	1	0	0	0	1	0	0
Beaver Valley 2	2	0	0	0	0	2	0	1
Big Rock Point	0	1	0	2	0	0	2	0
Braidwood 1	0	2	1	1	0	0	1	3
Braidwood 2	3	0	3	1	1	0	0	1
Browns Ferry 1	3	3	5	5	6	4	4	1
Browns Ferry 2	3	4	5	6	6	7	5	1
Browns Ferry 3	3	4	6	5	6	3	4	1
Brunswick 1	2	2	2	2	3	3	0	2
Brunswick 2	4	0	0	1	1	1	0	1
Byron 1	0	0	0	1	1	2	0	0
Byron 2	0	0	0	2	0	0	0	0
Callaway	1	1	1	1	0	0	0	0
Calvert Cliffs 1	0	4	1	1	2	1	1	1
Calvert Cliffs 2	2	0	0	0	3	2	0	1
Catawba 1	4	0	1	1	2	2	1	0
Catawba 2	5	5	2	2	2	5	1	0
Clinton 1	1	2	3	3	1	4	2	0
Cook 1	0	0	2	1	0	0	1	1
Cook 2	0	0	1	1	2	0	1	1
Cooper Station	2	2	0	0	2	5	0	0
Crystal River 3	3	0	0	3	1	5	1	0
Davis-Besse	2	3	2	1	0	1	2	1
Diablo Canyon 1	2	1	2	2	2	0	1	4
Diablo Canyon 2	1	1	1	7	2	1	2	1
Dresden 2	1	4	0	4	4	0	2	1
Dresden 3	1	8	0	1	1	3	0	2
Duane Arnold	0	3	1	0	2	1	1	0
Farley 1	2	0	0	3	0	2	0	2
Farley 2	1	0	0	1	0	4	1	2
Fermi 2	6	1	3	0	2	1	3	2
FitzPatrick	1	0	1	2	0	0	2	3
Fort Calhoun	2	2	1	3	0	4	0	1
Fort St. Vrain*								
Ginna	1	0	1	0	0	0	0	2
Grand Gulf	2	0	0	0	1	0	0	0
Haddam Neck	3	1	0	0	1	0	2	2
Hatch 1	0	1	1	2	0	0	0	0

Footnotes at end of table.

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	2	1	2	1	0	0	0	0
Hope Creek	2	4	0	4	1	3	0	1
Indian Point 2	0	2	1	3	2	0	0	1
Indian Point 3	2	0	1	1	3	0	0	0
Kewaunee	0	1	0	0	1	1	0	0
LaSalle 1	0	4	1	1	0	0	0	2
LaSalle 2	0	4	1	0	0	1	0	2
Limerick 1	2	3	0	3	4	3	0	2
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	2	2
Maine Yankee	1	0	0	1	0	0	0	0
McGuire 1	3	2	3	8	0	2	2	2
McGuire 2	2	6	4	2	0	0	1	4
Millstone 1	0	0	0	1	0	2	1	1
Millstone 2	4	0	0	1	0	1	1	1
Millstone 3	5	0	0	0	0	1	2	2
Monticello	0	0	0	0	2	1	1	3
Nine Mile Point 1	3	0	0	0	0	0	2	0
Nine Mile Point 2	3	0	3	2	0	1	1	2
North Anna 1	1	0	0	2	3	2	0	0
North Anna 2	0	1	0	2	1	2	0	0
Oconee 1	0	0	2	1	2	0	2	0
Oconee 2	1	0	1	1	1	1	0	0
Oconee 3	0	1	1	1	1	0	2	1
Oyster Creek	1	1	2	4	3	0	1	0
Palisades	0	0	4	1	1	1	1	0
Palo Verde 1	4	1	2	0	1	1	1	1
Palo Verde 2	2	0	0	0	0	1	0	2
Palo Verde 3	2	0	2	0	0	1	1	1
Peach Bottom 2	2	2	2	2	2	3	1	2
Peach Bottom 3	1	1	3	2	0	1	2	1
Perry 1	3	0	2	2	4	2	1	1
Pilgrim	2	4	2	0	2	1	1	2
Point Beach 1	0	0	1	1	1	0	1	0
Point Beach 2	0	1	0	0	1	0	1	1
Prairie Island 1	0	1	1	0	0	0	1	3
Prairie Island 2	0	1	1	0	0	0	1	3
Quad Cities 1	2	2	2	0	0	2	1	0
Quad Cities 2	4	6	2	2	0	1	0	1
Rancho Seco	0	1	1	2	1	0	NLI†	NLI
River Bend	0	0	3	0	3	5	1	3

Footnotes at end of table.

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	0	3	1	2	0	0	1
Salem 1	0	0	0	0	3	3	1	4
Salem 2	1	1	0	1	1	1	0	1
San Onofre 1	1	0	0	2	1	1	1	0
San Onofre 2	1	2	1	1	2	1	4	0
San Onofre 3	1	2	2	0	4	1	1	0
Seabrook	1	0	0	1	0	0	1	1
Sequoyah 1	5	6	2	5	3	2	0	3
Sequoyah 2	8	5	2	1	1	5	1	3
Shearon Harris	0	1	1	5	3	0	1	3
Shoreham	1	1	0	0	1	0	0	2
South Texas 1	1	2	2	3	2	1	0	1
South Texas 2	NYL	NYL	NYL	0	2	1	0	0
St. Lucie 1	1	0	2	0	0	2	2	2
St. Lucie 2	0	0	0	0	1	1	1	0
Summer	0	0	1	0	2	1	1	0
Surry 1	1	5	1	4	4	1	1	0
Surry 2	1	2	0	4	4	0	3	1
Susquehanna 1	1	0	1	0	2	1	0	2
Susquehanna 2	4	1	0	1	1	0	1	1
Three Mile Island 1	0	0	0	0	0	0	0	1
Trojan	2	2	5	1	1	2	2	0
Turkey Point 3	2	1	3	1	0	1	2	0
Turkey Point 4	1	1	1	1	0	1	3	0
Vermont Yankee	0	1	0	0	3	1	2	0
Vogtle 1	1	1	2	3	2	1	0	1
Vogtle 2	NYL	NYL	NYL	NYL	3	2	0	2
Wash. Nuclear 2	2	5	4	1	3	5	0	0
Waterford 3	2	4	0	1	0	0	2	2
Wolf Creek	2	0	1	2	1	0	0	0
Yankee-Rowe	0	0	0	0	2	0	2	0
Zion 1	2	0	1	1	1	0	1	3
Zion 2	3	0	1	4	4	0	0	1

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

**Table A-1.15 Cause Codes—Maintenance Sub-Categories—Personnel Error
During Surveillance/Testing—Quarterly PI Data**

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	2	0	1	4	1	3	2	6
Arkansas 2	1	2	2	3	2	2	0	3
Beaver Valley 1	1	2	1	2	1	2	0	4
Beaver Valley 2	1	0	1	3	3	4	1	1
Big Rock Point	0	0	0	0	1	1	2	0
Braidwood 1	5	3	3	0	0	0	0	4
Braidwood 2	2	2	2	2	1	0	0	1
Browns Ferry 1	6	1	5	11	3	1	2	1
Browns Ferry 2	5	1	5	13	5	2	4	1
Browns Ferry 3	5	1	5	11	1	1	2	1
Brunswick 1	2	0	0	4	1	3	1	1
Brunswick 2	2	1	1	6	1	2	4	2
Byron 1	0	0	1	0	1	0	1	0
Byron 2	2	1	2	0	1	0	1	0
Callaway	1	1	2	1	1	4	1	0
Calvert Cliffs 1	0	0	0	1	2	0	2	2
Calvert Cliffs 2	0	1	1	0	2	1	1	2
Catawba 1	5	1	1	2	2	2	1	0
Catawba 2	8	3	1	3	5	4	3	1
Clinton 1	4	4	1	3	8	5	0	4
Cook 1	2	2	3	1	2	4	3	0
Cook 2	1	2	2	2	2	4	1	0
Cooper Station	1	3	1	0	2	1	0	0
Crystal River 3	3	1	2	4	3	1	5	0
Davis-Besse	1	1	3	2	1	3	2	0
Diablo Canyon 1	5	4	4	2	1	0	1	3
Diablo Canyon 2	2	3	1	5	1	1	1	3
Dresden 2	3	2	2	1	3	1	0	0
Dresden 3	2	4	1	0	0	1	0	1
Duane Arnold	0	2	1	1	2	0	2	0
Farley 1	0	1	0	1	0	1	0	0
Farley 2	1	1	0	0	1	1	1	0
Fermi 2	5	3	3	2	3	3	2	5
FitzPatrick	1	1	0	2	1	3	0	3
Fort Calhoun	4	2	1	3	7	2	1	2
Fort St. Vrain*								
Ginna	1	0	0	0	0	4	0	0
Grand Gulf	3	0	1	1	0	2	1	1
Haddam Neck	2	2	1	0	0	2	2	0
Hatch 1	2	1	1	2	4	0	1	4

Footnotes at end of table.

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	8	6	1	0	3	0	2	5
Hope Creek	3	1	5	5	3	3	3	3
Indian Point 2	0	2	2	0	0	1	0	0
Indian Point 3	0	1	0	0	3	1	0	2
Kewaunee	1	1	2	0	5	2	0	0
LaSalle 1	0	0	0	1	4	4	0	2
LaSalle 2	2	1	0	3	6	3	1	1
Limerick 1	2	3	0	2	4	8	5	1
Limerick 2	NYL**	NYL	NYL	NYL	NYL	1	4	2
Maine Yankee	0	0	1	0	0	1	0	0
McGuire 1	1	2	5	4	3	1	7	0
McGuire 2	0	1	4	2	3	1	9	1
Millstone 1	1	0	0	1	1	2	2	1
Millstone 2	2	0	1	0	1	2	2	1
Millstone 3	3	3	2	3	2	2	5	5
Monticello	2	1	0	0	1	4	0	3
Nine Mile Point 1	2	2	1	2	4	1	0	1
Nine Mile Point 2	7	4	7	7	5	6	2	2
North Anna 1	5	1	0	3	1	2	1	0
North Anna 2	2	2	1	3	0	3	1	0
Oconee 1	4	0	1	0	3	1	1	0
Oconee 2	2	0	0	0	2	1	1	0
Oconee 3	2	1	2	0	2	1	1	1
Oyster Creek	0	1	4	2	1	2	3	0
Palisades	3	2	2	1	2	1	1	2
Palo Verde 1	5	0	1	2	1	2	1	1
Palo Verde 2	6	4	1	2	0	2	1	0
Palo Verde 3	3	1	1	0	1	3	2	1
Peach Bottom 2	1	5	1	5	2	5	1	2
Peach Bottom 3	0	3	1	2	2	2	2	4
Perry 1	5	3	3	1	3	6	2	1
Pilgrim	4	0	1	1	5	4	2	2
Point Beach 1	1	1	0	0	0	2	0	1
Point Beach 2	1	1	0	0	0	2	0	2
Prairie Island 1	0	0	1	3	0	1	1	0
Prairie Island 2	0	0	1	2	0	1	1	0
Quad Cities 1	3	0	1	1	1	0	0	2
Quad Cities 2	2	2	1	5	0	0	0	0
Rancho Seco	4	1	1	2	1	1	NLI†	NLI
River Bend	4	1	1	2	5	6	2	3

Footnotes at end of table.

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	2	0	0	2	1	0	0
Salem 1	4	1	3	0	5	7	0	1
Salem 2	3	2	4	3	2	3	0	1
San Onofre 1	4	2	1	1	1	2	3	1
San Onofre 2	0	2	4	0	3	0	4	2
San Onofre 3	2	0	7	1	1	1	1	1
Seabrook	1	1	1	0	2	2	3	1
Sequoyah 1	11	1	5	4	2	3	6	3
Sequoyah 2	13	4	7	1	2	5	5	2
Shearon Harris	3	3	7	3	3	4	1	1
Shoreham	2	5	3	2	2	0	1	1
South Texas 1	13	5	5	2	4	2	2	2
South Texas 2	NYL	NYL	NYL	0	4	1	0	1
St. Lucie 1	0	0	2	0	0	0	0	1
St. Lucie 2	0	1	0	0	0	0	1	0
Summer	2	2	1	2	2	1	2	2
Surry 1	2	3	2	2	2	7	4	0
Surry 2	1	1	1	1	0	7	4	3
Susquehanna 1	1	2	2	2	0	6	1	0
Susquehanna 2	0	2	2	2	0	3	1	3
Three Mile Island 1	0	1	2	0	0	0	0	1
Trojan	3	6	4	7	3	3	5	4
Turkey Point 3	1	5	5	2	4	0	1	0
Turkey Point 4	1	6	5	1	2	0	2	1
Vermont Yankee	1	1	0	3	4	1	1	0
Vogtle 1	6	7	3	6	4	1	2	1
Vogtle 2	NYL	NYL	NYL	NYL	2	3	1	1
Wash. Nuclear 2	2	4	4	4	0	5	4	1
Waterford 3	0	12	3	2	1	3	3	1
Wolf Creek	1	2	2	1	5	1	1	0
Yankee-Rowe	1	1	0	2	1	2	1	1
Zion 1	3	2	3	1	1	1	4	1
Zion 2	1	1	2	7	0	1	1	1

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Table A-1.16 Cause Codes—Maintenance Sub-Categories—Maintenance Equipment Failure—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	0	0	1	1	1	1	0	3
Arkansas 2	0	1	2	3	0	3	0	1
Beaver Valley 1	1	1	0	0	1	1	0	1
Beaver Valley 2	1	1	0	0	1	3	0	0
Big Rock Point	1	1	0	0	1	0	0	0
Braidwood 1	2	0	0	1	0	1	0	0
Braidwood 2	0	4	3	1	0	0	0	0
Browns Ferry 1	0	0	1	5	1	0	2	2
Browns Ferry 2	0	1	1	5	1	1	3	3
Browns Ferry 3	0	0	1	5	2	0	3	2
Brunswick 1	2	0	0	3	1	4	0	1
Brunswick 2	6	0	1	1	1	3	1	2
Byron 1	0	1	1	1	1	0	0	0
Byron 2	1	2	3	1	1	0	0	0
Callaway	1	0	1	0	1	0	0	0
Calvert Cliffs 1	0	1	0	0	0	0	0	0
Calvert Cliffs 2	1	0	0	0	1	0	0	0
Catawba 1	1	0	0	1	4	0	1	0
Catawba 2	7	5	2	3	1	0	1	0
Clinton 1	5	3	2	1	2	1	2	1
Cook 1	2	0	0	1	3	1	0	0
Cook 2	1	1	0	0	4	1	0	1
Cooper Station	0	3	5	1	0	0	1	1
Crystal River 3	2	0	0	0	0	1	1	0
Davis-Besse	0	0	0	1	0	0	0	0
Diablo Canyon 1	1	2	2	0	0	0	0	0
Diablo Canyon 2	2	2	0	2	0	0	0	0
Dresden 2	0	2	1	3	2	1	5	1
Dresden 3	1	4	0	0	1	0	1	1
Duane Arnold	1	0	1	0	1	0	1	1
Farley 1	2	2	0	2	0	0	0	0
Farley 2	0	1	0	1	0	1	0	0
Fermi 2	2	2	2	1	0	0	3	2
FitzPatrick	0	2	2	0	1	2	2	0
Fort Calhoun	1	0	0	1	0	0	1	0
Fort St. Vrain*								
Ginna	2	1	1	0	0	1	0	0
Grand Gulf	3	0	0	0	0	0	2	0
Haddam Neck	2	2	1	2	0	0	0	0
Hatch 1	0	1	1	1	0	0	0	0

Footnotes at end of table.

Table A-1.16 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	2	2	1	1	0	0	2	1
Hope Creek	1	2	1	0	0	0	0	0
Indian Point 2	1	1	1	0	0	0	1	0
Indian Point 3	0	1	0	0	0	0	0	0
Kewaunee	1	1	0	0	0	0	0	0
LaSalle 1	3	1	1	2	6	2	1	1
LaSalle 2	2	2	1	4	4	1	2	2
Limerick 1	0	2	0	0	1	1	0	2
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	0	0
Maine Yankee	1	0	1	0	0	0	0	2
McGuire 1	1	3	4	5	1	0	1	0
McGuire 2	1	1	4	3	2	0	1	0
Millstone 1	1	0	2	2	0	4	0	0
Millstone 2	0	2	0	0	1	0	0	0
Millstone 3	0	2	1	2	0	1	0	1
Monticello	0	0	0	0	0	1	0	0
Nine Mile Point 1	2	1	0	0	0	0	0	1
Nine Mile Point 2	2	1	4	1	1	1	1	2
North Anna 1	6	0	1	1	1	1	0	0
North Anna 2	1	0	0	1	1	0	0	0
Oconee 1	2	0	0	0	0	0	0	0
Oconee 2	1	0	1	0	0	0	0	0
Oconee 3	1	1	1	0	1	0	0	0
Oyster Creek	2	3	2	2	0	0	1	0
Palisades	0	0	0	2	0	0	1	0
Palo Verde 1	1	0	1	0	0	1	1	0
Palo Verde 2	1	0	0	1	2	1	1	2
Palo Verde 3	0	0	0	0	1	0	1	0
Peach Bottom 2	2	0	2	0	0	1	2	1
Peach Bottom 3	1	2	2	0	0	0	0	1
Perry 1	2	7	3	2	1	1	0	0
Pilgrim	1	2	0	1	1	0	1	2
Point Beach 1	0	0	0	0	0	1	0	0
Point Beach 2	0	0	0	0	0	0	0	1
Prairie Island 1	0	1	0	3	0	1	0	2
Prairie Island 2	0	1	0	2	0	1	0	2
Quad Cities 1	0	1	0	0	1	1	1	0
Quad Cities 2	1	6	0	0	0	2	0	0
Rancho Seco	1	0	0	1	1	0	NLI†	NLI
River Bend	0	1	1	3	3	2	0	0

Footnotes at end of table.

Table A-1.16 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	1	3	0	1	1	0	0	0
Salem 1	2	0	1	1	1	0	0	1
Salem 2	3	0	1	2	3	1	0	1
San Onofre 1	1	0	1	1	0	0	0	0
San Onofre 2	4	2	1	1	0	0	0	0
San Onofre 3	5	1	0	0	0	0	0	0
Seabrook	0	0	0	0	2	0	0	0
Sequoyah 1	5	1	3	5	1	0	0	1
Sequoyah 2	6	1	4	0	1	0	0	0
Shearon Harris	3	4	5	1	0	1	0	1
Shoreham	0	0	2	0	1	0	0	0
South Texas 1	2	1	3	0	0	0	0	1
South Texas 2	NYL	NYL	NYL	0	1	1	0	0
St. Lucie 1	2	0	1	0	0	0	0	0
St. Lucie 2	1	0	0	0	0	0	0	0
Summer	0	1	0	0	0	2	0	0
Surry 1	4	5	5	3	0	2	1	0
Surry 2	6	5	3	2	0	2	2	1
Susquehanna 1	0	2	3	0	0	1	1	1
Susquehanna 2	2	0	3	0	0	0	0	1
Three Mile Island 1	0	0	0	1	0	0	0	0
Trojan	1	2	4	1	0	0	1	1
Turkey Point 3	1	1	1	0	1	0	0	2
Turkey Point 4	0	1	3	0	1	0	0	1
Vermont Yankee	2	2	1	1	2	0	0	0
Vogtle 1	1	1	0	2	3	0	1	0
Vogtle 2	NYL	NYL	NYL	NYL	2	2	2	0
Wash. Nuclear 2	1	1	1	2	0	1	0	0
Waterford 3	1	2	0	0	0	0	0	0
Wolf Creek	1	1	1	1	0	0	1	0
Yankee-Rowe	0	0	0	1	0	0	0	0
Zion 1	2	2	1	1	1	0	0	0
Zion 2	2	1	1	5	0	0	0	0

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Table A-1.17 Cause Codes—Maintenance Sub-Categories—Potential Maintenance Problem—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	0	0	1	0	1	0	2
Arkansas 2	0	0	0	0	0	0	0	2
Beaver Valley 1	1	2	1	0	1	0	2	3
Beaver Valley 2	0	0	0	0	3	3	2	0
Big Rock Point	0	0	1	0	0	0	0	0
Braidwood 1	1	1	0	3	3	1	3	1
Braidwood 2	0	2	2	1	1	1	3	0
Browns Ferry 1	1	1	1	4	0	1	0	1
Browns Ferry 2	0	2	1	3	0	2	0	2
Browns Ferry 3	0	1	1	3	0	1	0	1
Brunswick 1	2	1	1	3	0	0	0	1
Brunswick 2	2	1	1	1	1	0	2	2
Byron 1	0	1	2	0	0	0	0	0
Byron 2	1	0	3	0	0	1	0	0
Callaway	1	1	0	0	0	1	1	1
Calvert Cliffs 1	0	1	0	1	0	0	0	1
Calvert Cliffs 2	1	0	0	1	0	1	0	1
Catawba 1	0	0	0	1	1	0	1	1
Catawba 2	2	0	0	1	2	0	0	0
Clinton 1	1	2	2	1	0	0	0	1
Cook 1	0	0	0	2	0	0	0	0
Cook 2	0	0	0	1	1	0	0	0
Cooper Station	0	1	1	0	0	2	1	0
Crystal River 3	2	1	0	0	0	2	0	1
Davis-Besse	1	1	0	0	1	0	0	1
Diablo Canyon 1	3	0	0	0	0	0	0	1
Diablo Canyon 2	1	1	1	1	1	0	0	0
Dresden 2	1	2	1	1	1	1	3	2
Dresden 3	0	0	0	0	0	0	2	2
Duane Arnold	1	0	1	0	1	0	0	0
Farley 1	1	1	0	1	0	0	0	0
Farley 2	0	0	0	1	0	1	0	0
Fermi 2	1	3	1	1	2	0	0	1
FitzPatrick	0	0	0	0	0	0	0	0
Fort Calhoun	0	2	2	2	0	0	0	0
Fort St. Vrain*								
Ginna	0	1	1	0	0	0	3	1
Grand Gulf	3	0	1	1	0	1	1	2
Haddam Neck	0	1	0	0	0	1	2	1
Hatch 1	0	2	0	1	0	0	1	1

Footnotes at end of table.

Table A-1.17 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	1	2	1	1	0	0	0	1
Hope Creek	0	0	0	0	1	1	0	1
Indian Point 2	0	0	1	0	0	0	2	0
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	1	1	0	0	0	1	1	1
LaSalle 1	2	2	3	2	3	3	2	0
LaSalle 2	1	2	5	2	2	3	0	1
Limerick 1	1	2	0	0	2	1	2	1
Limerick 2	NYL**	NYL	NYL	NYL	NYL	1	0	4
Maine Yankee	0	0	1	0	1	0	0	0
McGuire 1	2	3	0	3	0	0	1	0
McGuire 2	1	1	2	1	1	1	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	2	0	0	0	0	0	0
Millstone 3	1	0	0	1	0	1	0	0
Monticello	1	0	0	0	1	1	3	2
Nine Mile Point 1	1	0	0	0	0	0	1	0
Nine Mile Point 2	1	2	2	2	0	0	4	1
North Anna 1	4	0	1	0	0	1	1	1
North Anna 2	1	0	0	0	1	0	0	0
Oconee 1	0	0	0	0	1	1	0	0
Oconee 2	0	0	0	0	1	0	0	1
Oconee 3	0	0	0	1	0	1	0	0
Oyster Creek	0	0	3	1	0	2	0	0
Palisades	2	0	2	2	0	2	1	0
Palo Verde 1	1	2	1	1	0	0	2	0
Palo Verde 2	1	0	0	1	0	0	0	0
Palo Verde 3	0	0	0	0	1	0	0	0
Peach Bottom 2	0	0	1	2	0	0	1	2
Peach Bottom 3	0	3	1	0	0	0	1	1
Perry 1	0	4	2	2	1	0	0	1
Pilgrim	0	0	0	2	1	0	3	2
Point Beach 1	0	0	0	0	0	1	0	0
Point Beach 2	0	0	0	0	1	0	1	1
Prairie Island 1	0	0	0	0	0	1	4	1
Prairie Island 2	0	0	0	1	0	0	4	1
Quad Cities 1	1	0	0	0	0	2	1	2
Quad Cities 2	2	3	1	1	0	0	0	0
Rancho Seco	0	0	0	1	0	0	NLI†	NLI
River Bend	2	1	2	1	2	1	0	1

Footnotes at end of table.

Table A-1.17 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0	1	0	0	0	1	0	0
Salem 1	0	0	0	0	0	2	0	1
Salem 2	1	1	1	1	0	1	0	1
San Onofre 1	1	0	0	0	2	1	2	0
San Onofre 2	0	1	1	2	0	0	0	0
San Onofre 3	0	0	0	1	0	0	0	0
Seabrook	0	0	0	1	1	1	0	0
Sequoyah 1	2	1	1	2	0	2	1	2
Sequoyah 2	3	2	2	0	0	2	0	2
Shearon Harris	0	1	3	0	0	0	0	1
Shoreham	0	0	0	1	1	0	0	0
South Texas 1	1	1	1	1	1	0	0	1
South Texas 2	NYL	NYL	NYL	0	1	2	3	1
St. Lucie 1	1	1	2	0	0	0	0	0
St. Lucie 2	1	0	0	0	0	0	1	1
Summer	0	0	0	0	0	1	1	0
Surry 1	1	3	7	1	0	0	2	2
Surry 2	1	2	1	1	0	0	4	0
Susquehanna 1	1	0	1	0	1	2	0	0
Susquehanna 2	1	0	2	0	1	0	2	1
Three Mile Island 1	0	0	0	1	0	0	0	1
Trojan	0	0	1	2	0	0	0	0
Turkey Point 3	2	1	1	0	0	0	0	0
Turkey Point 4	1	2	2	0	0	1	2	0
Vermont Yankee	0	0	0	0	2	1	1	1
Vogtle 1	2	1	2	3	0	1	1	0
Vogtle 2	NYL	NYL	NYL	NYL	1	1	0	0
Wash. Nuclear 2	0	1	1	1	0	2	1	1
Waterford 3	2	0	0	0	0	0	1	1
Wolf Creek	0	0	0	0	1	2	3	0
Yankee-Rowe	1	0	0	1	0	1	1	0
Zion 1	2	1	1	1	1	0	0	1
Zion 2	1	1	1	2	0	1	0	0

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Table A-1.18 Cause Codes—Design/Fabrication/Installation—Quarterly PI Data

Plant Name	Calendar Year—Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	2	3	8	6	9	2	3	2
Arkansas 2	3	2	3	0	2	3	3	0
Beaver Valley 1	1	1	0	1	0	1	2	0
Beaver Valley 2	1	1	1	2	1	0	1	0
Big Rock Point	1	2	0	0	1	0	0	1
Braidwood 1	2	0	0	2	0	0	3	1
Braidwood 2	2	1	2	2	0	0	1	2
Browns Ferry 1	1	2	8	6	7	4	3	2
Browns Ferry 2	1	2	8	6	8	4	3	2
Browns Ferry 3	1	2	8	6	7	4	3	2
Brunswick 1	1	3	2	4	3	0	2	1
Brunswick 2	2	2	4	4	3	0	2	1
Byron 1	0	0	0	0	0	0	1	1
Byron 2	2	0	0	0	0	0	1	1
Callaway	1	0	1	1	1	0	0	0
Calvert Cliffs 1	0	0	2	3	2	5	4	1
Calvert Cliffs 2	1	1	0	1	0	4	3	1
Catawba 1	5	3	0	3	5	2	4	2
Catawba 2	6	5	0	4	5	2	3	3
Clinton 1	1	1	2	1	3	2	1	3
Cook 1	0	0	2	1	1	0	0	0
Cook 2	0	1	2	1	2	0	0	1
Cooper Station	3	1	0	0	6	6	0	1
Crystal River 3	1	2	3	3	7	5	5	2
Davis-Besse	4	1	4	1	0	2	0	0
Diablo Canyon 1	0	2	3	3	1	0	1	2
Diablo Canyon 2	2	1	2	6	1	0	1	1
Dresden 2	1	1	0	1	2	0	2	1
Dresden 3	2	1	0	0	2	1	1	2
Duane Arnold	0	2	4	5	1	0	2	0
Farley 1	1	2	1	0	0	2	0	2
Farley 2	4	0	1	1	0	2	0	1
Fermi 2	0	2	2	0	2	2	1	2
FitzPatrick	0	1	0	4	2	3	3	5
Fort Calhoun	1	1	3	4	2	3	0	1
Fort St. Vrain*								
Ginna	0	0	1	0	0	2	1	1
Grand Gulf	1	0	2	0	0	2	2	1
Haddam Neck	5	2	1	1	3	1	2	4
Hatch 1	1	1	2	2	0	1	0	2

Footnotes at end of table.

Table A-1.18 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	1	2	2	2	0	0	0	1
Hope Creek	3	5	5	2	0	0	3	3
Indian Point 2	1	2	1	4	3	0	0	1
Indian Point 3	0	1	0	2	0	4	0	1
Kewaunee	0	2	0	0	2	2	2	0
LaSalle 1	0	3	2	2	4	1	0	1
LaSalle 2	0	2	1	2	4	1	0	1
Limerick 1	7	5	3	8	10	8	0	0
Limerick 2	NYL**	NYL	NYL	NYL	NYL	1	1	1
Maine Yankee	0	2	1	1	1	0	0	1
McGuire 1	2	3	6	5	3	2	6	1
McGuire 2	1	3	4	2	1	2	6	1
Millstone 1	2	1	0	1	3	2	0	2
Millstone 2	1	0	0	0	2	0	0	1
Millstone 3	4	0	0	1	1	1	0	2
Monticello	0	1	1	1	1	3	2	5
Nine Mile Point 1	3	2	1	1	0	1	1	2
Nine Mile Point 2	7	5	12	6	2	1	1	2
North Anna 1	1	1	0	3	0	1	0	1
North Anna 2	1	2	0	3	0	1	0	1
Oconee 1	3	2	1	1	3	3	2	1
Oconee 2	3	3	1	1	3	2	2	2
Oconee 3	2	2	1	1	3	2	2	1
Oyster Creek	2	1	4	0	5	0	0	0
Palisades	0	3	1	2	3	1	4	1
Palo Verde 1	4	0	2	1	2	2	1	4
Palo Verde 2	3	1	4	2	2	3	1	3
Palo Verde 3	3	0	1	1	3	4	1	3
Peach Bottom 2	1	1	3	3	3	2	2	2
Peach Bottom 3	1	3	2	2	2	1	2	3
Perry 1	3	3	3	3	1	3	1	1
Pilgrim	3	2	1	1	4	0	1	0
Point Beach 1	2	1	2	0	1	3	0	1
Point Beach 2	1	1	2	2	1	2	1	2
Prairie Island 1	0	1	0	2	0	2	1	0
Prairie Island 2	0	1	0	1	0	2	0	0
Quad Cities 1	1	3	0	0	0	2	0	1
Quad Cities 2	1	9	0	0	0	2	1	1
Rancho Seco	1	3	1	1	3	0	NLI†	NLI
River Bend	2	1	0	2	2	4	0	0

Footnotes at end of table.

Table A-1.18 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	3	2	2	4	2	1	1	4
Salem 1	5	2	3	0	3	6	0	3
Salem 2	2	3	3	1	3	8	1	8
San Onofre 1	3	2	2	3	5	2	3	5
San Onofre 2	3	3	4	4	1	3	1	0
San Onofre 3	3	2	4	3	0	4	2	1
Seabrook	1	0	0	1	0	0	1	0
Sequoyah 1	6	3	3	1	0	0	0	1
Sequoyah 2	7	5	3	1	1	0	1	1
Shearon Harris	4	1	3	0	2	0	1	1
Shoreham	0	0	0	1	0	0	0	1
South Texas 1	6	6	8	4	5	1	4	0
South Texas 2	NYL	NYL	NYL	2	2	1	5	1
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	1	0	0	0
Summer	3	1	0	2	1	0	1	1
Surry 1	2	2	4	3	1	3	2	2
Surry 2	0	4	3	4	1	3	3	1
Susquehanna 1	0	1	2	1	1	1	2	0
Susquehanna 2	2	1	2	1	1	1	1	2
Three Mile Island 1	1	0	0	0	0	0	0	0
Trojan	0	1	2	2	2	3	2	1
Turkey Point 3	1	1	5	4	3	3	0	2
Turkey Point 4	1	0	3	3	2	3	3	1
Vermont Yankee	0	0	1	1	2	1	1	0
Vogtle 1	0	5	2	6	1	0	0	1
Vogtle 2	NYL	NYL	NYL	NYL	1	1	1	1
Wash. Nuclear 2	3	5	4	0	4	6	6	2
Waterford 3	1	5	0	4	2	0	3	0
Wolf Creek	1	4	4	6	1	1	0	1
Yankee-Rowe	0	1	0	2	0	1	1	0
Zion 1	2	0	2	2	0	1	0	1
Zion 2	1	0	2	3	0	2	0	0

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Table A-1.19 Cause Codes—Equipment Failures (Electronic Piece-Part or Environmentally Related Failures)—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Arkansas 1	1	0	0	0	0	1	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	2	0	0	0	0	0
Big Rock Point	0	0	0	1	0	0	0	0
Braidwood 1	1	0	2	0	0	0	0	0
Braidwood 2	0	0	1	1	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	0	0	0
Brunswick 2	0	0	0	0	0	0	0	0
Byron 1	0	0	2	0	1	0	1	0
Byron 2	0	1	2	0	0	0	1	0
Callaway	1	1	1	0	0	1	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	1	0	0	0	0	1	1	1
Catawba 2	1	1	0	0	0	1	1	0
Clinton 1	0	1	0	1	0	0	0	0
Cook 1	0	0	0	1	0	0	0	0
Cook 2	0	0	0	0	0	0	1	0
Cooper Station	0	0	1	0	0	0	0	0
Crystal River 3	0	0	0	0	0	1	0	0
Davis-Besse	0	0	1	0	1	1	0	0
Diablo Canyon 1	0	0	0	0	0	0	0	0
Diablo Canyon 2	0	0	0	0	1	0	0	0
Dresden 2	0	0	1	0	2	0	0	0
Dresden 3	0	0	0	0	1	0	0	0
Duane Arnold	0	0	0	0	1	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	1	0	0	0	0	0	0	0
FitzPatrick	0	0	0	0	0	0	0	1
Fort Calhoun	0	1	1	0	0	0	0	0
Fort St. Vrain*								
Ginna	0	1	2	0	0	0	0	0
Grand Gulf	1	0	0	0	1	0	0	0
Haddam Neck	0	0	0	0	0	0	0	0
Hatch 1	0	0	1	0	1	0	0	0

Footnotes at end of table.

Table A-1.19 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Hatch 2	0	0	1	0	0	0	0	0
Hope Creek	0	0	0	2	0	0	0	0
Indian Point 2	0	0	0	0	0	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	0
LaSalle 2	0	0	0	0	0	0	0	0
Limerick 1	0	0	0	0	0	0	0	0
Limerick 2	NYL**	NYL	NYL	NYL	NYL	0	1	0
Maine Yankee	0	0	0	0	0	0	0	0
McGuire 1	0	1	0	0	1	0	1	0
McGuire 2	0	0	1	0	1	0	0	0
Millstone 1	0	0	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	0	0	0	1	0	0	0	0
Monticello	0	0	0	0	0	0	0	0
Nine Mile Point 1	0	0	0	0	1	0	0	0
Nine Mile Point 2	0	1	4	0	1	1	0	1
North Anna 1	1	0	0	0	0	0	0	0
North Anna 2	1	0	0	0	0	1	0	0
Oconee 1	0	0	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	0	0	0	0	0	0
Oyster Creek	0	0	0	0	1	1	0	0
Palisades	0	0	0	0	0	0	0	0
Palo Verde 1	0	1	0	0	1	0	1	1
Palo Verde 2	0	0	0	0	0	0	1	0
Palo Verde 3	0	0	0	0	1	0	0	0
Peach Bottom 2	0	0	1	0	0	1	0	0
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry 1	0	0	0	1	0	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	1
Prairie Island 2	0	0	0	0	1	1	0	1
Quad Cities 1	0	0	0	0	0	1	0	0
Quad Cities 2	2	1	0	0	0	0	0	0
Rancho Seco	0	0	0	0	0	0	NLI†	NLI
River Bend	0	0	0	0	0	0	1	0

Footnotes at end of table.

Table A-1.19 (cont.)

Plant Name	Calendar Year-Quarter							
	88-1	88-2	88-3	88-4	89-1	89-2	89-3	89-4
Robinson 2	0	1	0	0	0	0	0	0
Salem 1	0	0	0	1	1	1	0	0
Salem 2	0	3	0	1	1	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	1	0	0	1	1	1	0	0
San Onofre 3	1	0	0	1	1	1	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	0	0	0
Sequoyah 2	1	0	2	0	0	0	0	0
Shearon Harris	0	0	0	0	1	0	0	0
Shoreham	0	0	0	0	0	0	0	0
South Texas 1	1	0	0	0	0	0	0	0
South Texas 2	NYL	NYL	NYL	0	0	0	2	1
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	1	0	1	0
Summer	0	0	0	0	0	0	0	1
Surry 1	0	0	1	1	1	0	1	2
Surry 2	1	1	0	1	1	0	0	0
Susquehanna 1	0	0	1	0	1	0	0	1
Susquehanna 2	0	0	1	0	1	0	0	0
Three Mile Island 1	0	0	0	0	0	0	0	0
Trojan	0	0	0	1	0	1	1	0
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	0	0	0	0
Vogtle 1	0	0	1	0	1	1	0	0
Vogtle 2	NYL	NYL	NYL	NYL	1	0	1	1
Wash. Nuclear 2	0	0	1	0	0	0	1	0
Waterford 3	0	0	0	0	0	0	1	0
Wolf Creek	0	0	0	0	0	0	0	0
Yankee-Rowe	1	0	0	0	0	0	0	0
Zion 1	0	0	0	0	1	0	0	0
Zion 2	0	0	0	1	0	1	0	0

*Fort St. Vrain is not included in the cause code data.

**NYL means the plant is not yet licensed.

†NLI means the plant is no longer included in the PI data.

Other Plant Operational Experience Data

Appendix A-2

Other Plant Operational Experience Data

This appendix presents selected Licensee Event Report (LER) Trends and Patterns database information by year in tabular and graphical forms. The information is discussed in Section 6 of this report.

Tables A-2.1 through A-2.5 present data regarding

reactor trips while the reactor is critical and reactor trip rates (trips per 1,000 critical hours).

Tables A-2.6 through A-2.10 and Figures A-2.1 through A-2.4 present data regarding emergency safety feature actuations.,

Table A-2.1 Total Automatic and Manual Reactor Trips While the Reactor Is Critical and the Reactor Trip Rates for 1985 Through 1989

Plant Name	Vendor	1985				1986				1987				1988				1989			
		Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate
Arkansas 1	B&W	8	6	2	1.14	2	2		0.36	2	2		0.25	1	1		0.16	5	5		0.83
Arkansas 2	CE	9	9		1.41	5	5		0.78	2	2		0.26	2	1	1	0.33	2	2		0.30
Beaver Valley 1	W	8	8		0.97	3	2	1	0.48	4	4		0.55	3	3		0.42	4	4		0.68
Beaver Valley 2*	W	NYC**				NYC				17	14	3	7.35	4	4		0.48	1	1		0.16
Big Rock Point	GE	3	1	2	0.46	1	1		0.12	1	1		0.16	3	2	1	0.47	1	1		0.14
Braidwood 1	W	NYC				NYC				7	5	2	2.04	3	2	1	0.52	2	2		0.36
Braidwood 2*	W	NYC				NYC				NYC				14	10	4	2.92	3	3		0.39
Browns Ferry 1	GE	2	1	1	1.21	N/A†				N/A				N/A				N/A			
Browns Ferry 2	GE	N/A				N/A				N/A				N/A				N/A			
Browns Ferry 3	GE	1		1	0.66	N/A				N/A				N/A				N/A			
Brunswick 1	GE	2	2		0.59	7	6	1	0.84	2	2		0.35	2	2		0.30	0			0.00
Brunswick 2	GE	3	2	1	0.42	2	2		0.47	2	2		0.24	2	1	1	0.35	1		1	0.17
Byron 1	W	22	22		4.40	5	4	1	0.64	4	4		0.64	3	3		0.46	1		1	0.11
Byron 2	W	NYC				NYC				11	9	2	1.61	6	5	1	0.69	0			0.00
Callaway	W	19	18	1	2.33	7	6	1	0.96	1	1		0.16	6	5	1	0.73	2	1	1	0.27
Calvert Cliffs 1	CE	6	6		1.12	4	4		0.58	6	4	2	0.91	3	2	1	0.47	0			0.00
Calvert Cliffs 2	CE	3	2	1	0.44	6	3	3	0.71	6	4	2	1.01	2	2		0.26	0			0.00
Catawba 1	W	11	9	2	1.91	8	8		1.47	5	3	2	0.82	0			0.00	3	1	2	0.40
Catawba 2	W	NYC				7	5	2	2.61	9	5	4	1.25	9	4	5	1.39	3	2	1	0.47
Clinton 1	GE	NYC				NYC				9	7	2	1.68	3	2	1	0.41	5	1	4	1.18
Cook 1	W	1	1		0.39	5	5		0.66	2	2		0.33	3	3		0.36	2	2		0.32
Cook 2	W	4	4		0.67	2	2		0.36	5	5		0.79	0			0.00	1	1		0.15
Cooper Station	GE	1		1	0.49	2	2		0.30	6	5	1	0.71	3	2	1	0.50	3	3		0.45
Crystal River 3	B&W	7	6	1	1.60	1	1		0.27	2	2		0.37	2	2		0.27	1	1		0.23
Davis-Besse	B&W	5	5		1.76	1	1		5.62	5	5		0.67	1	1		0.47	2	2		0.23
Diablo Canyon 1	W	9	8	1	1.38	3	3		0.50	5	5		0.59	5	5		0.88	1	1		0.14
Diablo Canyon 2	W	12	11	1	6.40	11	10	1	1.51	5	4	1	0.83	3	2	1	0.48	4	1	3	0.49
Dresden 2	GE	7	6	1	1.41	6	4	2	0.84	4	4		0.69	0			0.00	2	2		0.28
Dresden 3	GE	4	4		0.60	6	5	1	2.17	6	4	2	0.83	1	1		0.16	3	3		0.41
Duane Arnold	GE	0			0.00	2	1	1	0.27	0			0.00	1	1		0.15	5	4	1	0.72
Farley 1	W	4	4		0.53	4	4		0.55	4	3	1	0.48	1	1		0.13	1	1		0.13
Farley 2	W	5	5		0.73	4	4		0.53	2	2		0.31	0			0.00	6	5	1	0.83
Fermi 2	GE	8	7	1	4.20	4	2	2	2.85	7	7		1.36	5	4	1	0.94	4	2	2	0.67
FitzPatrick	GE	8	8		1.38	2	2		0.25	6	6		0.97	0			0.00	2	2		0.25
Fort Calhoun	CE	0			0.00	2	1	1	0.24	0			0.00	0			0.00	1		1	0.13
Fort St. Vrain††	GA	1	1		1.13	2	1	1	0.81	4	2	2	1.38	4	1	3	1.05	0			0.00

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1985				1986				1987				1988				1989			
		Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate
Ginna	W	8	7	1	1.02	4	3	1	0.52	0			0.00	2	2		0.26	1	1		0.15
Grand Gulf	GE	14	13	1	2.45	6	5	1	1.07	2	2		0.28	6	6		0.71	5	4	1	0.71
Haddam Neck	W	5	3	2	0.58	7	4	3	1.38	1	1		0.21	3	3		0.49	0			0.00
Hatch 1	GE	8	5	3	1.16	4	4		0.72	4	4		0.56	5	5		0.83	0			0.00
Hatch 2	GE	5	5		0.68	7	7		1.08	4	4		0.47	7	5	2	1.10	1	1		0.15
Hope Creek	GE	NYC				11	9	2	4.12	7	5	2	0.92	5	4	1	0.71	2	2		0.29
Indian Point 2	W	11	9	2	1.29	10	7	3	1.96	2	2		0.32	4	4		0.53	2	2		0.35
Indian Point 3	W	9	8	1	1.53	8	8		1.22	5	5		0.91	4	3	1	0.55	1		1	0.19
Kewaunee	W	4	3	1	0.55	3	3		0.40	2	2		0.25	3	3		0.39	1	1		0.13
La Crosse†††	AC	9	9		1.16	17	15	2	3.86	2	1	1	0.71	N/A				N/A			
LaSalle 1	GE	9	6	3	1.56	1	1		0.42	6	6		1.07	0			0.00	1	1		0.16
LaSalle 2	GE	1	1		0.26	4	4		0.60	1	1		0.21	2	1	1	0.30	1	1		0.15
Limerick 1	GE	4	4		0.91	1	1		0.14	2	2		0.33	1	1		0.12	0			0.00
Limerick 2*	GE	NYC				NYC				NYC				NYC				1	1		0.51
Maine Yankee	CE	8	6	2	1.14	6	4	2	0.77	2	1	1	0.35	3	3		0.43	2	2		0.24
McGuire 1	W	5	5		0.73	3	3		0.60	4	4		0.59	4	3	1	0.59	2	1	1	0.28
McGuire 2	W	9	8	1	1.64	6	5	1	1.04	5	5		0.71	2	1	1	0.27	3	3		0.43
Millstone 1	GE	3	3		0.41	3	2	1	0.36	4	4		0.57	1	1		0.12	3	3		0.41
Millstone 2	CE	1	1		0.22	4	4		0.61	5	5		0.61	1	1		0.14	0			0.00
Millstone 3	W	NYC				13	13		2.10	10	10		1.57	5	4	1	0.69	2	1	1	0.30
Monticello	GE	3	3		0.37	2	2		0.29	4	4		0.56	1	1		0.11	2	2		0.30
Nine Mile Point 1	GE	7	7		0.82	2	2		0.34	3	2	1	0.37	N/A				N/A			
Nine Mile Point 2	GE	NYC				NYC				6	6		2.22	9	7	2	1.99	6	6		1.15
North Anna 1	W	2		2	0.29	6	5	1	0.79	4	3	1	0.87	4	3	1	0.50	3	3		0.60
North Anna 2	W	2	1	1	0.23	4	4		0.55	0			0.00	0			0.00	0			0.00
Oconee 1	B&W	4	4		0.47	2	2		0.34	0			0.00	1	1		0.11	3	2	1	0.41
Oconee 2	B&W	4	4		0.59	5	5		0.69	3	3		0.35	1	1		0.14	3	3		0.41
Oconee 3	B&W	2	2		0.33	2	1	1	0.26	0			0.00	2	2		0.28	2	2		0.26
Oyster Creek	GE	6	6		0.88	3	2	1	1.26	3	3		0.53	0			0.00	5	4	1	1.00
Palisades	CE	2	2		0.27	2	2		1.34	6	1	5	1.42	0			0.00	1	1		0.17
Palo Verde 1	CE	9	9		3.12	13	13		2.36	3	3		0.65	5	5		0.87	1	1		0.66
Palo Verde 2	CE	NYC				10	10		2.63	4	3	1	0.57	1	1		0.17	3	3		0.71
Palo Verde 3*	CE	NYC				NYC				1	1		1.06	0			0.00	1	1		0.83
Peach Bottom 2	GE	6	6		2.06	5	4	1	0.69	0			0.00	N/A				4	4		0.75
Peach Bottom 3	GE	1	1		0.25	9	6	3	1.52	2	2		1.10	N/A				0			0.00
Perry 1	GE	NYC				4	3	1	2.80	12	6	6	2.66	8	8		1.15	0			0.00

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1985			Total Trip Rate	1986			Total Trip Rate	1987			Total Trip Rate	1988			Total Trip Rate	1989			Total Trip Rate
		Total Trips	Auto	Man		Total Trips	Auto	Man		Total Trips	Auto	Man		Total Trips	Auto	Man		Total Trips	Auto	Man	
Pilgrim	GE	4	4		0.49	4	4		2.33	N/A				0			0.00	5	4	1	0.89
Point Beach 1	W	1	1		0.14	2	2		0.25	2			0.27	0			0.00	0			0.00
Point Beach 2	W	1	1		0.13	2	2		0.28	1	1		0.13	1	1		0.13	2	2		0.28
Prairie Island 1	W	3	3		0.41	2	2		0.25	2	2		0.27	0			0.00	1	1		0.11
Prairie Island 2	W	0			0.00	3	3		0.38	0			0.00	0			0.00	3	3		0.38
Quad Cities 1	GE	3	2	1	0.36	4	4		0.65	1	1		0.16	1	1		0.12	3	1	2	0.45
Quad Cities 2	GE	4	4		0.63	2	2		0.35	5	5		0.72	3	2	1	0.48	2	2		0.24
Rancho Seco#	B&W	3	3		1.04	N/A			N/A					4	3	1	0.72	1	1		0.42
River Bend	GE	5	4	1	4.64	18	18		3.39	4	3	1	0.67	5	5		0.60	4	4		0.66
Robinson 2	W	12	12		1.53	10	10		1.40	4	4		0.63	3	3		0.52	3	3		0.70
Salem 1	W	1	1		0.12	9	9		1.27	1	1		0.16	3	2	1	0.43	3	3		0.48
Salem 2	W	10	9	1	1.91	9	9		1.60	4	4		0.62	7	7		1.17	4	3	1	0.52
San Onofre 1	W	2	1	1	0.29	2	2		0.67	1	1		0.14	0			0.00	3	1	2	0.84
San Onofre 2	CE	10	10		1.91	8	7	1	1.23	3	1	2	0.48	0			0.00	1		1	0.19
San Onofre 3	CE	5	5		1.04	6	5	1	0.81	2	2		0.28	1		1	0.17	2	2		0.24
Seabrook	W	NYC				NYC				NYC				NYC				1		1	5.14
Sequoyah 1	W	1	1		0.26	N/A			N/A					2	2		5.27	2	2		0.23
Sequoyah 2	W	6	6		1.13	N/A			N/A					5	5		0.96	4	4		0.63
Shearon Harris	W	NYC				NYC				19	9	10	3.06	3	1	2	0.46	6	6		0.86
Shoreham	GE	4	3	1	2.76	0			0.00	N/A				N/A				0			0.00
South Texas 1*	W	NYC				NYC				NYC				4	4		0.77	3	3		0.52
South Texas 2*	W	NYC				NYC				NYC				NYC				10	9	1	2.22
St. Lucie 1	CE	1	1		0.14	4	3	1	0.47	6	6		0.86	3	3		0.40	2	2		0.24
St. Lucie 2	CE	7	5	2	0.94	5	4	1	0.68	5	5		0.68	0			0.00	2	1	1	0.30
Summer	W	12	8	4	1.86	6	6		0.71	4	4		0.64	4	4		0.66	5	3	2	0.69
Surry 1	W	7	7		0.88	5	2	3	0.80	3	2	1	0.49	2	2		0.53	2	1	1	0.47
Surry 2	W	1	1		0.17	4	3	1	0.65	1	1		0.15	3	2	1	0.60	2	2		1.33
Susquehanna 1	GE	4	4		0.71	0			0.00	2	1	1	0.31	2	2		0.24	4	3	1	0.61
Susquehanna 2	GE	5	5		0.65	2	1	1	0.34	1	1		0.12	0			0.00	1		1	0.14
Three Mile Island 1	B&W	1	1		0.48	4	4		0.64	3	3		0.47	1	1		0.15	1	1		0.11
Trojan	W	4	4		0.59	4	4		0.57	3	2	1	0.63	4	4		0.68	1	1		0.18
Turkey Point 3	W	6	6		1.11	6	5	1	0.86	5	3	2	2.62	0			0.00	1	1		0.17
Turkey Point 4	W	9	9		1.14	3	2	1	0.98	1	1		0.22	1	1		0.20	2	1	1	0.48
Vermont Yankee	GE	1	1		0.16	2	2		0.46	3	3		0.41	3	3		0.36	0			0.00
Vogtle 1*	W	NYC				NYC				25	25		4.64	9	6	3	1.32	5	2	3	0.59
Vogtle 2*	W	NYC				NYC				NYC				NYC				7	6	1	1.14

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1985				1986				1987				1988				1989			
		Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate	Total Trips	Auto	Man	Total Trip Rate
Wash. Nuclear 2	GE	12	11	1	1.74	7	7		1.10	6	5	1	0.97	2	1	1	0.32	4	4		0.58
Waterford 3	CE	26	26		7.29	7	7		1.00	7	7		0.97	4	3	1	0.60	3	1	2	0.41
Wolf Creek	W	14	13	1	3.13	7	7		1.07	9	9		1.46	0			0.00	2	2		0.23
Yankee-Rowe	W	2	2		0.26	3	3		0.36	2	2		0.28	3	3		0.40	3	2	1	0.37
Zion 1	W	3	2	1	0.56	2	2		0.36	1	1		0.15	4	4		0.59	1	1		0.19
Zion 2	W	1	1		0.17	4	4		0.51	0			0.00	2	2		0.29	0			0.00
Total All Plants##		525	474	51	0.96	457	404	53	0.81	406	343	63	0.65	274	228	46	0.41	244	197	47	0.36
Total All W Plants		246	222	24	0.97	206	185	21	0.81	202	172	30	0.73	151	126	25	0.48	125	99	26	0.38
Total All GE Plants		148	129	19	0.87	133	115	18	0.77	125	108	17	0.63	81	68	13	0.41	80	65	15	0.39
Total All CE Plants		87	82	5	1.16	82	72	10	0.89	58	45	13	0.63	25	21	4	0.24	21	16	5	0.26
Total All B&W Plants		34	31	3	0.84	17	16	1	0.46	15	15		0.31	13	12	1	0.25	18	17	1	0.34

*Indicates a "new" plant, i.e., a plant with less than 2 years operating experience since receipt of its operating license.

**NYC means the plant is not yet critical.

†N/A means data are not applicable because the reactor was not critical during the year.

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

†††LaCrosse, a boiling water reactor designed by Allis Chalmers (AC), was permanently shut down on April 30, 1987.

#Rancho Seco ceased commercial operations on June 6, 1989.

##This total includes Fort St. Vrain and LaCrosse.

Table A-2.2 Reactor Trip Initiating Systems

Mature Westinghouse Reactors											
System	Total Trips						Trips/1000 Critical Hours¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	39	46	24	36	26	171	0.17	0.22	0.10	0.13	0.09
RPS	25	22	12	22	15	96	0.11	0.10	0.05	0.08	0.05
Turbine	10	27	20	17	17	91	0.04	0.13	0.09	0.06	0.06
Electrical	35	20	9	11	5	80	0.16	0.09	0.04	0.04	0.02
Control rod drive	12	11	8	5	7	43	0.05	0.05	0.03	0.02	0.02
Main generator	11	7	11	6	8	43	0.05	0.03	0.05	0.02	0.03
Main steam	4	7	5	4	14	34	0.02	0.03	0.02	0.01	0.05
Support	8	9	2	4	4	27	0.04	0.04	0.01	0.01	0.01
RCS	7	2	6	4	3	22	0.03	0.01	0.03	0.01	0.01
Condensate	8	2	2	3	2	17	0.04	0.01	0.01	0.01	0.01
Total	159	153	99	112	101	624	0.71	0.72	0.42	0.41	0.33

¹Critical hours: 1985 = 223,138; 1986 = 212,303; 1987 = 234,632; 1988 = 270,463; and 1989 = 303,837.

Mature General Electric Reactors											
System	Total Trips						Trips/1000 Critical Hours²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
RPS	24	15	9	10	24	82	0.17	0.10	0.05	0.05	0.12
Turbine	19	11	13	15	22	80	0.13	0.07	0.08	0.08	0.11
Feedwater	13	16	23	11	10	73	0.09	0.11	0.13	0.06	0.05
Electrical	13	12	7	3	5	40	0.09	0.08	0.04	0.02	0.02
Main steam	9	10	8	4	7	38	0.06	0.07	0.05	0.02	0.03
Condensate	9	11	6	6	5	37	0.06	0.07	0.04	0.03	0.02
Main generator	7	2	12	12	2	35	0.05	0.01	0.07	0.06	0.01
Support	6	11	5	3	4	29	0.05	0.07	0.04	0.02	0.02
RCS	6	3	2	3	0	14	0.04	0.02	0.01	0.02	0.00
Control rod drive	3	2	1	2	0	8	0.02	0.01	0.01	0.01	0.00
Total	109	93	86	69	79	436	0.76	0.61	0.50	0.37	0.39

²Critical hours: 1985 = 142,825; 1986 = 152,008; 1987 = 170,875; 1988 = 186,252; and 1989 = 202,523.

Table A-2.2 (cont.)

Mature Combustion Engineering Reactors

System	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	15	13	17	9	7	61	0.23	0.17	0.20	0.10	0.09
Electrical	3	8	6	3	2	22	0.05	0.11	0.07	0.03	0.03
Main generator	5	5	7	4	1	22	0.08	0.07	0.08	0.04	0.01
Turbine	6	4	7	0	2	19	0.09	0.05	0.08	0.00	0.03
RPS	2	6	2	3	5	18	0.03	0.08	0.02	0.03	0.06
RCS	5	5	3	2	2	17	0.08	0.07	0.04	0.02	0.03
Control rod drive	10	2	2	0	1	15	0.15	0.03	0.02	0.00	0.01
Main steam	4	3	2	1	0	10	0.06	0.04	0.02	0.01	0.00
Support	1	3	4	1	0	9	0.02	0.04	0.05	0.01	0.00
Condensate	1	3	3	2	0	9	0.02	0.04	0.04	0.02	0.00
Total	52	52	53	25	20	202	0.79	0.69	0.63	0.26	0.25

³Critical hours: 1985 = 66,061; 1986 = 75,738; 1987 = 84,587; 1988 = 94,354; and 1989 = 79,768.

Mature Babcock & Wilcox Reactors

System	Total Trips						Trips/1000 Critical Hours ⁴				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	17	5	7	7	4	40	0.42	0.14	0.14	0.14	0.08
Turbine	7	3	1	2	4	17	0.17	0.08	0.02	0.04	0.08
Electrical	5	2	1	1	4	13	0.12	0.05	0.02	0.02	0.08
RPS	2	2	2	0	4	10	0.05	0.05	0.04	0.00	0.08
Main generator	1	1	2	0	1	5	0.02	0.03	0.04	0.00	0.02
Condensate	0	3	1	1	0	5	0.00	0.08	0.02	0.02	0.00
Control rod drive	0	0	0	1	1	2	0.00	0.00	0.00	0.02	0.02
RCS	1	1	0	0	0	2	0.02	0.03	0.00	0.00	0.00
Main steam	1	0	0	1	0	2	0.02	0.00	0.00	0.02	0.00
Support	0	0	1	0	0	1	0.00	0.00	0.02	0.00	0.00
Total	34	17	15	13	18	97	0.84	0.46	0.31	0.25	0.34

⁴Critical hours: 1985 = 40,531; 1986 = 36,712; 1987 = 48,711; 1988 = 51,033; and 1989 = 52,332.

Table A-2.3 Activities at Time of Reactor Trip

Mature Westinghouse Reactors											
Activity	Total Trips						Trips/1000 Critical Hours¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	54	56	42	45	58	255	0.24	0.26	0.18	0.17	0.19
Testing	13	32	16	27	22	110	0.06	0.15	0.07	0.10	0.07
Raising power	30	26	13	13	2	84	0.13	0.12	0.06	0.05	0.01
Maintenance	21	16	10	10	10	67	0.09	0.08	0.04	0.04	0.03
Reducing power	23	12	11	11	5	62	0.10	0.06	0.05	0.04	0.02
Troubleshooting	12	7	5	5	2	31	0.05	0.03	0.02	0.02	0.01
Calibration	4	2	2	1	1	10	0.02	0.01	0.01	0.00	0.00
Special testing	0	2	0	0	1	3	0.00	0.01	0.00	0.00	0.00
Unknown	2	0	0	0	0	2	0.01	0.00	0.00	0.00	0.00
Total	159	153	99	112	101	624	0.71	0.72	0.42	0.41	0.33

¹Critical hours: 1985 = 223,138; 1986 = 212,303; 1987 = 234,632; 1988 = 270,463; and 1989 = 303,837.

Mature General Electric Reactors											
Activity	Total Trips						Trips/1000 Critical Hours²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	30	26	44	32	34	166	0.21	0.17	0.26	0.17	0.17
Testing	21	18	13	15	21	88	0.15	0.12	0.08	0.08	0.10
Reducing power	15	13	12	8	8	56	0.11	0.09	0.07	0.04	0.04
Raising power	10	19	7	8	7	51	0.07	0.12	0.04	0.04	0.03
Maintenance	17	13	9	5	3	47	0.12	0.09	0.05	0.03	0.01
Troubleshooting	3	3	1	0	3	10	0.02	0.02	0.01	0.00	0.01
Calibration	5	0	0	1	2	8	0.04	0.00	0.00	0.01	0.01
Unknown	8	0	0	0	0	8	0.06	0.00	0.00	0.00	0.00
Special testing	0	1	0	0	1	2	0.00	0.01	0.00	0.00	0.00
Total	109	93	86	69	79	436	0.76	0.61	0.50	0.37	0.39

²Critical hours: 1985 = 142,825; 1986 = 152,008; 1987 = 170,875; 1988 = 186,252; and 1989 = 202,523.

Table A-2.3 (cont.)

Mature Combustion Engineering Reactors

Activity	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	20	24	32	11	11	98	0.30	0.32	0.38	0.12	0.14
Maintenance	7	7	5	6	2	27	0.11	0.09	0.06	0.06	0.03
Raising power	9	7	2	3	3	24	0.14	0.09	0.02	0.03	0.04
Reducing power	3	5	8	2	3	21	0.05	0.07	0.09	0.02	0.04
Testing	3	8	5	2	0	18	0.05	0.11	0.06	0.02	0.00
Troubleshooting	5	1	0	1	1	8	0.08	0.01	0.00	0.01	0.01
Unknown	4	0	0	0	0	4	0.06	0.00	0.00	0.00	0.00
Calibration	1	0	1	0	0	2	0.02	0.00	0.01	0.00	0.00
Total	52	52	53	25	20	202	0.79	0.69	0.63	0.26	0.25

³Critical hours: 1985 = 66,061; 1986 = 75,738; 1987 = 84,587; 1988 = 94,354; and 1989 = 79,768.

Mature Babcock & Wilcox Reactors

Activity	Total Trips						Trips/1000 Critical Hours ⁴				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	11	7	9	6	7	40	0.27	0.19	0.18	0.12	0.13
Raising power	5	4	2	3	0	14	0.12	0.11	0.04	0.06	0.00
Testing	4	2	1	2	4	13	0.10	0.05	0.02	0.04	0.08
Maintenance	4	1	1	0	4	10	0.10	0.03	0.02	0.00	0.08
Troubleshooting	4	3	0	0	1	8	0.10	0.08	0.00	0.00	0.02
Reducing power	4	0	1	1	0	6	0.10	0.00	0.02	0.02	0.00
Calibration	1	0	1	1	2	5	0.02	0.00	0.02	0.02	0.04
Unknown	1	0	0	0	0	1	0.02	0.00	0.00	0.00	0.00
Total	34	17	15	13	18	97	0.84	0.46	0.31	0.25	0.34

⁴Critical hours: 1985 = 40,531; 1986 = 36,712; 1987 = 48,711; 1988 = 51,033; and 1989 = 52,332.

Table A-2.4 Reactor Trip Causes

Mature Westinghouse Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	81	89	59	61	52	342	0.36	0.42	0.25	0.23	0.17
Personnel error	51	44	26	33	27	181	0.23	0.21	0.11	0.12	0.09
Procedure	6	11	4	13	8	42	0.03	0.05	0.02	0.05	0.03
Unknown	10	2	4	2	8	26	0.04	0.01	0.02	0.01	0.03
Natural phenomenon	4	4	5	1	4	18	0.02	0.02	0.02	0.00	0.01
SG level	7	3	1	1	1	13	0.03	0.01	0.00	0.00	0.00
Other	0	0	0	1	1	2	0.00	0.00	0.00	0.00	0.00
Total	159	153	99	112	101	624	0.71	0.72	0.42	0.41	0.33

¹Critical hours: 1985 = 223,138; 1986 = 212,303; 1987 = 234,632; 1988 = 270,463; and 1989 = 303,837.

Mature General Electric Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	56	54	61	41	46	258	0.39	0.36	0.36	0.22	0.23
Personnel error	35	28	17	14	19	113	0.25	0.18	0.10	0.08	0.09
Procedure	10	7	5	9	5	36	0.07	0.05	0.03	0.05	0.02
Unknown	4	3	2	1	3	13	0.03	0.02	0.01	0.01	0.01
Natural phenomenon	3	0	0	4	2	9	0.02	0.00	0.00	0.02	0.01
Other	1	1	1	0	4	7	0.01	0.01	0.01	0.00	0.02
Total	109	93	86	69	79	436	0.76	0.61	0.50	0.37	0.39

²Critical hours: 1985 = 142,825; 1986 = 152,008; 1987 = 170,875; 1988 = 186,252; and 1989 = 202,523.

Table A-2.4 (cont.)

Mature Combustion Engineering Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	26	32	28	12	10	108	0.39	0.42	0.33	0.13	0.13
Personnel error	17	14	11	6	2	50	0.26	0.18	0.13	0.06	0.03
Procedure	5	2	5	5	3	20	0.08	0.03	0.06	0.05	0.04
Unknown	3	4	2	2	3	14	0.05	0.05	0.02	0.02	0.04
Natural phenomenon	0	0	2	0	1	3	0.00	0.00	0.02	0.00	0.01
SG level	1	0	2	0	0	3	0.02	0.00	0.02	0.00	0.00
Total	52	52	53	25	20	202	0.79	0.69	0.63	0.26	0.25

³Critical hours: 1985 = 66,061; 1986 = 75,738; 1987 = 84,587; 1988 = 94,354; and 1989 = 79,768.

Mature Babcock & Wilcox Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ⁴				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	24	10	10	9	5	58	0.59	0.27	0.21	0.18	0.10
Personnel error	3	5	4	2	10	24	0.07	0.14	0.08	0.04	0.19
Unknown	3	0	0	1	3	7	0.07	0.00	0.00	0.02	0.06
Procedure	3	1	0	1	0	5	0.07	0.03	0.00	0.02	0.00
Natural phenomenon	0	1	1	0	0	2	0.00	0.03	0.02	0.00	0.00
SG level	1	0	0	0	0	1	0.02	0.00	0.00	0.00	0.00
Total	34	17	15	13	18	97	0.84	0.46	0.31	0.25	0.34

⁴Critical hours: 1985 = 40,531; 1986 = 36,712; 1987 = 48,711; 1988 = 51,033; and 1989 = 52,332.

Table A-2.5 Reactor Trip Signals

Mature Westinghouse Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Low SG level	41	28	21	28	31	149	0.18	0.13	0.09	0.10	0.10
Turbine trip	38	35	25	29	20	147	0.17	0.16	0.11	0.11	0.07
Manual	17	18	9	16	24	84	0.08	0.08	0.04	0.06	0.08
High negative flux rate	6	6	9	6	7	34	0.03	0.03	0.04	0.02	0.02
Steam flow feedflow mismatch	7	12	4	1	2	26	0.03	0.06	0.02	0.00	0.01
Low reactor coolant flow	5	5	3	7	3	23	0.02	0.02	0.01	0.03	0.01
Safety injection	1	9	2	3	3	18	0.00	0.04	0.01	0.01	0.01
High RCS temperature	7	4	1	2	1	15	0.03	0.02	0.00	0.01	0.00
Loss of power	4	4	3	3	1	15	0.02	0.02	0.01	0.01	0.00
High reactor power	4	5	3	1	1	14	0.02	0.02	0.01	0.00	0.00
Other	1	2	3	5	0	11	0.00	0.01	0.01	0.02	0.00
Reactor trip breaker open	3	5	1	2	0	11	0.01	0.02	0.00	0.01	0.00
P-7 Permissive	3	3	2	0	2	10	0.01	0.01	0.01	0.00	0.01
IRM	1	2	1	4	1	9	0.00	0.01	0.00	0.01	0.00
PZR high pressure	4	4	0	0	0	8	0.02	0.02	0.00	0.00	0.00
High SG level	3	3	1	1	0	8	0.01	0.01	0.00	0.00	0.00
PZR low pressure	1	3	1	1	0	6	0.00	0.01	0.00	0.00	0.00
Unknown	6	0	0	0	0	6	0.03	0.00	0.00	0.00	0.00
RCP breaker open	2	2	0	2	0	6	0.01	0.01	0.00	0.01	0.00
Under voltage RCP	0	0	3	0	0	3	0.00	0.00	0.01	0.00	0.00
High RCS pressure	0	0	3	0	0	3	0.00	0.00	0.01	0.00	0.00
Over power delta temperature	0	0	0	0	2	2	0.00	0.00	0.00	0.00	0.01
Low RCS pressure	0	0	1	0	1	2	0.00	0.00	0.00	0.00	0.00
Low negative reactor power	0	1	0	1	0	2	0.00	0.00	0.00	0.00	0.00
MSIV closure	0	0	1	0	1	2	0.00	0.00	0.00	0.00	0.00
SRM	1	0	1	0	0	2	0.00	0.00	0.00	0.00	0.00
High steamline flow	2	0	0	0	0	2	0.01	0.00	0.00	0.00	0.00
High steamline pressure	1	0	0	0	0	1	0.00	0.00	0.00	0.00	0.00
High containment pressure	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Under frequency RCP speed	1	0	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Low steamline pressure	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.00
PZR high level	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Low condenser vacuum	0	0	1	0	0	1	0.00	0.00	0.00	0.00	0.00
Total	159	153	99	112	101	624	0.71	0.72	0.42	0.41	0.33

¹Critical hours: 1985 = 223,138; 1986 = 212,303; 1987 = 234,632; 1988 = 270,463; 1989 = 303,837.

Table A-2.5 (cont.)

Mature General Electric Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Turbine trip	29	21	27	27	28	132	0.20	0.14	0.16	0.14	0.14
Low reactor water level	24	18	27	12	10	91	0.17	0.12	0.16	0.06	0.05
Manual	14	12	6	10	15	57	0.10	0.08	0.04	0.05	0.07
MSIV closure	13	14	6	3	7	43	0.09	0.09	0.04	0.02	0.03
High reactor power	6	9	7	7	10	39	0.04	0.06	0.04	0.04	0.05
IRM	3	8	4	2	3	20	0.02	0.05	0.02	0.01	0.01
High RCS pressure	2	4	4	1	2	13	0.01	0.03	0.02	0.01	0.01
Low condenser vacuum	4	4	1	1	1	11	0.03	0.03	0.01	0.01	0.00
Other	5	0	1	2	1	9	0.04	0.00	0.01	0.01	0.00
MS line radiation	1	1	1	1	1	5	0.01	0.01	0.01	0.01	0.00
Loss of power	0	0	2	1	1	4	0.00	0.00	0.01	0.01	0.00
Trip discharge volume high	1	2	0	1	0	4	0.01	0.01	0.00	0.01	0.00
Unknown	4	0	0	0	0	4	0.03	0.00	0.00	0.00	0.00
High steamline flow	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
SRM	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
ATWS	0	0	0	1	0	1	0.00	0.00	0.00	0.01	0.00
Low reactor coolant flow	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
Total	109	93	86	69	79	436	0.76	0.61	0.50	0.37	0.39

²Critical hours: 1985 = 142,825; 1986 = 152,008; 1987 = 170,875; 1988 = 186,252; 1989 = 202,523.

Table A-2.5 (cont.)

Mature Combustion Engineering Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Turbine trip	14	11	8	5	5	43	0.21	0.15	0.09	0.05	0.06
Low SG level	9	10	13	7	4	43	0.14	0.13	0.15	0.07	0.05
Manual	6	9	14	4	5	38	0.09	0.12	0.17	0.04	0.06
Low DNBR	8	8	1	3	3	23	0.12	0.11	0.01	0.03	0.04
High RCS pressure	2	3	3	1	1	10	0.03	0.04	0.04	0.01	0.01
Loss of power	1	2	5	1	0	9	0.02	0.03	0.06	0.01	0.00
High SG level	2	2	3	0	1	8	0.03	0.03	0.04	0.00	0.01
Other	2	0	3	1	1	7	0.03	0.00	0.04	0.01	0.01
High reactor power	2	2	1	0	0	5	0.03	0.03	0.01	0.00	0.00
PZR high pressure	1	1	1	1	0	4	0.02	0.01	0.01	0.01	0.00
Low reactor coolant flow	1	2	1	0	0	4	0.02	0.03	0.01	0.00	0.00
MSIV closure	2	0	0	0	0	2	0.03	0.00	0.00	0.00	0.00
Steam flow feedflow mismatch	2	0	0	0	0	2	0.03	0.00	0.00	0.00	0.00
Low condenser vacuum	0	1	0	0	0	1	0.00	0.01	0.00	0.00	0.00
Safety injection	0	0	0	1	0	1	0.00	0.00	0.00	0.01	0.00
Under frequency RCP speed	0	1	0	0	0	1	0.00	0.01	0.00	0.00	0.00
Under voltage RCP	0	0	0	1	0	1	0.00	0.00	0.00	0.01	0.00
Total	52	52	53	25	20	202	0.79	0.69	0.63	0.26	0.25

³Critical hours: 1985 = 66,061; 1986 = 75,738; 1987 = 84,587; 1988 = 94,354; and 1989 = 79,768.

Table A-2.5 (cont.)

Mature Babcock & Wilcox Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ⁴				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
High RCS pressure	17	4	5	5	5	36	0.42	0.11	0.10	0.10	0.10
Turbine trip	3	8	5	2	7	25	0.07	0.22	0.10	0.04	0.13
Other	5	0	1	2	2	10	0.12	0.00	0.02	0.04	0.04
Manual	3	1	0	1	1	6	0.07	0.03	0.00	0.02	0.02
High reactor power	1	1	2	1	0	5	0.02	0.03	0.04	0.02	0.00
Under voltage RCP	0	0	1	1	1	3	0.00	0.00	0.02	0.02	0.02
Low SG level	2	1	0	0	0	3	0.05	0.03	0.00	0.00	0.00
Low reactor coolant flow	1	0	0	0	0	1	0.02	0.00	0.00	0.00	0.00
Loss of power	0	0	1	0	0	1	0.00	0.00	0.02	0.00	0.00
High SG level	0	1	0	0	0	1	0.00	0.03	0.00	0.00	0.00
Low RCS pressure	0	0	0	1	0	1	0.00	0.00	0.00	0.02	0.00
Steam flow feedflow mismatch	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.02
Under frequency RCP speed	1	0	0	0	0	1	0.02	0.00	0.00	0.00	0.00
Power imbalance flow	0	1	0	0	0	1	0.00	0.03	0.00	0.00	0.00
Unknown	1	0	0	0	0	1	0.02	0.00	0.00	0.00	0.00
High RCS temperature	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.02
Total	34	17	15	13	18	97	0.84	0.46	0.31	0.25	0.34

⁴Critical hours: 1985 = 40,531; 1986 = 36,712; 1987 = 48,711; 1988 = 51,033; and 1989 = 52,332.

Table A-2.6 Engineered Safety Features Actuations

Plant Name	Vendor	1985	1986	1987	1988	1989	Total	Average Yearly Rate
Arkansas 1	B&W	1	0	3	2	49	55	11.0
Arkansas 2	CE	6	1	0	4	7	18	3.60
Beaver Valley 1	W	5	2	13	11	11	42	8.40
Beaver Valley 2*	W	NYL**	NYL	15	7	15	37	7.40
Big Rock Point	GE	0	0	0	2	0	2	0.40
Braidwood 1	W	NYL	2	21	15	11	49	9.80
Braidwood 2*	W	NYL	NYL	0	11	1	12	2.40
Browns Ferry 1	GE	13	36	24	40	20	133	26.6
Browns Ferry 2	GE	15	32	16	39	17	119	23.8
Browns Ferry 3	GE	9	31	21	27	15	103	20.6
Brunswick 1	GE	86	27	21	13	14	161	32.2
Brunswick 2	GE	10	18	9	14	18	69	13.8
Byron 1	W	42	16	8	5	3	74	14.8
Byron 2	W	NYL	0	7	2	2	11	2.20
Callaway	W	38	20	12	7	14	91	18.2
Calvert Cliffs 1	CE	3	1	3	5	2	14	2.80
Calvert Cliffs 2	CE	3	2	3	0	0	8	1.60
Catawba 1	W	46	37	11	4	9	107	21.4
Catawba 2	W	NYL	44	19	13	4	80	16.0
Clinton 1	GE	NYL	24	43	10	18	95	19.0
Cook 1	W	33	6	1	0	0	40	8.00
Cook 2	W	27	3	2	1	1	34	6.80
Cooper Station	GE	7	22	17	22	14	82	16.4
Crystal River 3	B&W	29	3	6	6	7	51	10.2
Davis-Besse	B&W	8	4	4	4	2	22	4.40
Diablo Canyon 1	W	13	13	11	12	8	57	11.4
Diablo Canyon 2	W	13	12	12	17	5	59	11.8
Dresden 2	GE	12	6	5	7	13	43	8.60
Dresden 3	GE	11	15	8	8	5	47	9.40
Duane Arnold	GE	23	14	23	13	19	92	18.4
Farley 1	W	2	4	0	2	1	9	1.80
Farley 2	W	1	0	3	0	1	5	1.00
Fermi 2	GE	60	30	34	24	21	169	33.8
FitzPatrick	GE	10	6	13	4	11	44	8.80
Fort Calhoun	CE	9	1	12	10	2	34	6.80
Fort St. Vrain†	GA	8	9	46	5	6	74	14.8

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1985	1986	1987	1988	1989	Total	Average Yearly Rate
Ginna	W	8	1	2	5	10	26	5.20
Grand Gulf	GE	31	20	12	6	11	80	16.0
Haddam Neck	W	5	8	4	0	1	18	3.60
Hatch 1	GE	38	31	18	9	8	104	20.8
Hatch 2	GE	25	23	13	12	5	78	15.6
Hope Creek	GE	NYL	55	37	15	16	123	24.6
Indian Point 2	W	3	6	4	2	3	18	3.60
Indian Point 3	W	1	2	4	1	1	9	1.80
Kewaunee	W	8	4	10	13	9	44	8.80
LaSalle 1	GE	52	30	20	15	10	127	25.4
LaSalle 2	GE	24	9	11	7	11	62	12.4
Limerick 1	GE	75	39	49	28	27	218	43.6
Limerick 2*	GE	NYL	NYL	NYL	NYL	8	8	1.60
Maine Yankee	CE	2	0	0	3	0	5	1.00
McGuire 1	W	11	2	5	12	7	37	7.40
McGuire 2	W	8	8	1	6	2	25	5.00
Millstone 1	GE	6	5	8	6	4	29	5.80
Millstone 2	CE	7	7	2	5	1	22	4.40
Millstone 3	W	0	17	8	5	5	35	7.00
Monticello	GE	30	23	19	3	20	95	19.0
Nine Mile Point 1	GE	18	12	11	4	8	53	10.6
Nine Mile Point 2	GE	NYL	17	54	36	18	125	25.0
North Anna 1	W	1	3	5	1	4	14	2.80
North Anna 2	W	1	2	3	3	4	13	2.60
Oconee 1	B&W	0	0	0	0	0	0	0.00
Oconee 2	B&W	2	2	2	0	1	7	1.40
Oconee 3	B&W	4	1	1	0	1	7	1.40
Oyster Creek	GE	4	23	14	10	7	58	11.6
Palisades	CE	9	15	11	10	3	48	9.60
Palo Verde 1	CE	40	19	6	3	4	72	14.4
Palo Verde 2	CE	5	17	4	5	3	34	6.80
Palo Verde 3*	CE	NYL	NYL	3	1	7	11	2.20
Peach Bottom 2	GE	19	13	23	14	11	80	16.0
Peach Bottom 3	GE	18	19	12	12	3	64	12.8
Perry 1	GE	0	89	53	21	15	178	35.6
Pilgrim	GE	9	7	17	17	21	71	14.2

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1985	1986	1987	1988	1989	Total	Average Yearly Rate
Point Beach 1	W	6	3	1	0	2	12	2.40
Point Beach 2	W	2	2	2	2	4	12	2.40
Prairie Island 1	W	4	2	7	3	25	41	8.20
Prairie Island 2	W	0	4	1	0	2	7	1.40
Quad Cities 1	GE	29	19	13	6	6	73	14.6
Quad Cities 2	GE	17	8	12	13	2	52	10.4
Rancho Seco††	B&W	17	9	2	54	4	86	17.2
River Bend	GE	45	35	12	17	23	132	26.4
Robinson 2	W	1	3	1	2	1	8	1.60
Salem 1	W	0	0	0	0	11	11	2.20
Salem 2	W	1	3	0	2	11	17	3.40
San Onofre 1	W	2	3	3	2	3	13	2.60
San Onofre 2	CE	53	42	60	26	7	188	37.6
San Onofre 3	CE	66	7	20	7	6	106	21.2
Seabrook	W	NYL	2	9	3	7	21	4.20
Sequoyah 1	W	27	16	11	22	11	87	17.4
Sequoyah 2	W	7	14	9	16	5	51	10.2
Shearon Harris	W	NYL	4	26	8	6	44	8.80
Shoreham	GE	41	19	13	10	6	89	17.8
South Texas 1*	W	NYL	NYL	13	22	9	44	8.80
South Texas 2*	W	NYL	NYL	NYL	1	14	15	3.00
St. Lucie 1	CE	2	2	4	2	0	10	2.00
St. Lucie 2	CE	5	0	2	0	1	8	1.60
Summer	W	4	4	8	2	5	23	4.60
Surry 1	W	4	8	21	4	13	50	10.0
Surry 2	W	3	7	2	4	12	28	5.60
Susquehanna 1	GE	25	21	16	12	18	92	18.4
Susquehanna 2	GE	18	8	10	11	12	59	11.8
Three Mile Island 1	B&W	1	2	2	0	1	6	1.20
Trojan	W	7	1	7	17	10	42	8.40
Turkey Point 3	W	14	4	12	9	5	44	8.80
Turkey Point 4	W	9	20	14	5	4	52	10.4
Vermont Yankee	GE	5	2	7	2	14	30	6.00
Vogtle 1*	W	NYL	NYL	58	14	6	78	15.6
Vogtle 2*	W	NYL	NYL	NYL	NYL	14	14	2.80
Wash. Nuclear 2	GE	25	26	15	10	16	92	18.4

Footnotes at end of table.

Table A-2.6 (cont.)

Plant Name	Vendor	1985	1986	1987	1988	1989	Total	Average Yearly Rate
Waterford 3	CE	27	14	4	7	5	57	11.4
Wolf Creek	W	95	36	35	19	6	191	38.2
Yankee-Rowe	W	0	2	3	5	2	12	2.40
Zion 1	W	14	8	2	6	4	34	6.80
Zion 2	W	17	8	4	3	0	32	6.40
Total All Plants†††		1600	1338	1333	1004	928	6203	
Total All W Plants		483	366	430	326	324	1929	
Total All GE Plants		810	814	703	519	485	3331	
Total All CE Plants		237	128	134	88	48	635	
Total All B&W Plants		62	21	20	66	65	234	

*Indicates a "new" plant, i.e., a plant with less than 2 years operating experience since receipt of its operating license.

**NYL means the plant is not yet licensed.

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

††Rancho Seco ceased commercial operations on June 6, 1989.

†††This total includes Fort St. Vrain.

Table A-2.7 Engineered Safety Feature Actuation Causes

Westinghouse Reactors						
Cause	1985	1986	1987	1988	1989	Total
Equipment	253	167	207	141	139	907
Personnel error	99	98	144	106	98	545
Procedure	59	42	45	42	40	228
Other	44	24	11	15	15	109
Unknown	16	28	20	19	28	111
Natural phenomenon	12	7	3	3	4	29
Total	483	366	430	326	324	1929
General Electric Reactors						
Cause	1985	1986	1987	1988	1989	Total
Equipment	396	354	299	235	193	1477
Personnel error	235	231	219	155	154	994
Procedure	56	88	97	69	65	375
Unknown	61	74	40	23	46	244
Other	56	48	26	18	24	172
Natural phenomenon	6	19	22	19	3	69
Total	810	814	703	519	485	3331
Combustion Engineering Reactors						
Cause	1985	1986	1987	1988	1989	Total
Equipment	131	58	85	46	20	340
Personnel error	41	44	21	26	15	147
Other	39	9	10	1	2	61
Procedure	9	9	12	12	6	48
Unknown	14	7	6	3	3	33
Natural phenomenon	3	1	0	0	2	6
Total	237	128	134	88	48	635
Babcock & Wilcox Reactors						
Cause	1985	1986	1987	1988	1989	Total
Equipment	31	4	8	53	41	137
Personnel error	13	9	8	8	12	50
Unknown	3	6	2	4	9	24
Other	15	1	0	0	1	17
Procedure	0	1	2	1	1	5
Natural phenomenon	0	0	0	0	1	1
Total	62	21	20	66	65	234

Table A-2.8 Engineered Safety Feature Actuation Activities

Westinghouse Reactors						
Activity	1985	1986	1987	1988	1989	Total
Normal operation	332	230	262	174	179	1177
Testing	87	80	105	101	99	472
Maintenance	35	51	63	51	43	243
Unknown	28	1	0	0	2	31
Other	1	4	0	0	1	6
Total	483	366	430	326	324	1929
General Electric Reactors						
Activity	1985	1986	1987	1988	1989	Total
Normal operation	561	522	388	257	258	1986
Testing	192	181	207	160	151	891
Maintenance	55	108	108	102	71	444
Other	0	1	0	0	5	6
Unknown	2	2	0	0	0	4
Total	810	814	703	519	485	3331
Combustion Engineering Reactors						
Activity	1985	1986	1987	1988	1989	Total
Normal operation	187	85	91	46	25	434
Testing	32	29	23	19	12	115
Maintenance	17	14	20	23	11	85
Other	1	0	0	0	0	1
Total	237	128	134	88	48	635
Babcock & Wilcox Reactors						
Activity	1985	1986	1987	1988	1989	Total
Normal operation	49	13	14	61	35	172
Testing	8	5	1	4	25	43
Maintenance	5	1	5	1	5	17
Other	0	2	0	0	0	2
Total	62	21	20	66	65	234

Table A-2.9 Engineered Safety Feature Actuations of Selected Systems*

Westinghouse Reactors						
System	1985	1986	1987	1988	1989	Total
ECCS	52	45	38	36	34	205
Diesel	64	70	80	52	56	322
HVAC	211	133	191	147	151	833
Total	327	248	309	235	241	1360
General Electric Reactors						
System	1985	1986	1987	1988	1989	Total
ECCS	55	60	47	36	31	229
Diesel	65	65	53	45	43	271
HVAC	398	458	366	321	244	1787
RWCU	250	219	220	151	138	978
Total	768	802	686	553	456	3265
Combustion Engineering Reactors						
System	1985	1986	1987	1988	1989	Total
ECCS	26	15	8	13	15	77
Diesel	21	22	12	22	9	86
HVAC	191	73	104	57	24	449
Total	238	110	124	92	48	612
Babcock & Wilcox Reactors						
System	1985	1986	1987	1988	1989	Total
ECCS	15	6	6	4	4	35
Diesel	13	6	9	2	11	41
HVAC	6	5	6	52	44	113
Total	34	17	21	58	59	189

*Diesel – systems include all diesel starts.

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.

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-2.10 Engineered Safety Feature Actuation Signals

Westinghouse Reactors						
Signal	1985	1986	1987	1988	1989	Total
High-high radiation	143	82	105	89	78	497
Low volt	82	74	83	82	80	401
Other	38	50	46	38	55	227
Low SG level	42	21	37	15	9	124
High toxic gas	15	16	43	23	24	121
High SG level	16	25	25	21	18	105
Main feedwater pump trip	26	13	36	10	9	94
Unknown	31	19	5	4	7	66
Manual	24	10	8	7	12	61
Low steamline delta P	9	15	12	3	7	46
Reactor trip	4	10	11	7	4	36
Low temperature	15	10	1	7	1	34
Low reactor pressure	8	5	4	9	4	30
High steam flow	5	0	6	4	3	18
Any low level	5	6	0	2	5	18
High steamline delta P	5	3	1	2	2	13
High containment delta P	3	4	3	1	0	11
Low SG pressure	4	1	0	1	1	7
Low condenser vacuum	2	1	2	0	0	5
High temperature	0	1	1	0	2	4
Fire	1	0	1	0	2	4
High reactor pressure	2	0	0	1	1	4
Low sample flow	3	0	0	0	0	3
Total	483	366	430	326	324	1929

Table A-2.10 (cont.)

General Electric Reactors						
Signal	1985	1986	1987	1988	1989	Total
Low volt	148	218	203	201	188	958
High-high radiation	89	128	77	72	43	409
Other	120	55	59	49	61	344
Low reactor vessel level	80	71	71	47	46	315
High toxic gas	77	67	44	24	10	222
High RWCU flow	67	57	46	24	28	222
High room diff. temp.	46	33	41	22	11	153
Unknown	25	50	24	6	7	112
Manual	38	14	18	17	16	103
High temperature	9	21	34	8	16	88
High steam flow	5	21	16	10	23	75
High reactor pressure	17	8	18	5	8	56
Reactor trip	0	15	10	12	7	44
Low containment vacuum	8	13	12	2	6	41
Fire	36	3	0	0	0	39
High drywell pressure	5	18	4	6	4	37
High containment delta P	4	6	7	5	0	22
High RHR flow	9	3	2	0	4	18
Low condenser vacuum	5	3	5	3	2	18
Low steamline delta P	6	3	3	2	0	14
Low reactor pressure	5	0	3	0	3	11
High steamline delta P	5	1	3	1	0	10
Any low level	1	4	2	1	1	9
Low sample flow	2	0	0	1	1	4
High suppression tank level	3	1	0	0	0	4
Low temperature	0	1	1	0	0	2
Main feedwater pump trip	0	0	0	1	0	1
Total	810	814	703	519	485	3331

Table A-2.10 (cont.)

Combustion Engineering Reactors						
Signal	1985	1986	1987	1988	1989	Total
High-high radiation	144	57	79	37	8	325
Low volt	27	26	27	24	18	122
Other	12	13	7	5	8	45
High toxic gas	20	5	2	8	0	35
Manual	8	1	4	2	4	19
Low SG level	4	6	4	2	2	18
Low reactor pressure	8	2	2	3	3	18
Unknown	4	5	2	0	1	12
Reactor trip	1	5	3	1	0	10
High SG level	1	4	2	0	1	8
High reactor pressure	0	1	0	4	2	7
High containment delta P	2	1	1	1	1	6
Low SG pressure	4	0	0	0	0	4
Main feedwater pump trip	0	0	1	1	0	2
Low steamline delta P	0	1	0	0	0	1
Any low level	1	0	0	0	0	1
Low reactor vessel level	0	1	0	0	0	1
Fire	1	0	0	0	0	1
Total	237	128	134	88	48	635

Table A-2.10 (cont.)

Babcock & Wilcox Reactors						
Signal	1985	1986	1987	1988	1989	Total
High-high radiation	3	1	0	14	34	52
Low volt	9	6	9	3	12	39
High temperature	0	1	0	33	0	34
Low SG level	19	2	5	4	2	32
Manual	11	2	2	4	5	24
Other	7	3	1	1	2	14
Unknown	2	3	1	3	2	11
High toxic gas	0	0	0	1	6	7
High reactor pressure	5	0	0	0	1	6
High SG level	1	0	0	2	0	3
Reactor trip	1	2	0	0	0	3
Low reactor pressure	2	0	1	0	0	3
Main feedwater pump trip	0	0	0	1	1	2
Low steamline delta P	1	0	0	0	0	1
Low SG pressure	1	0	0	0	0	1
Any low level	0	1	0	0	0	1
High containment delta P	0	0	1	0	0	1
Total	62	21	20	66	65	234

Figure A-2.1
Engineered Safety Feature Actuations
Westinghouse NSSS

Figure A-2.2
Engined Safety Feature Actuations
Combustion Engineering NSSS

Figure A-2.3
Engineered Safety Feature Actuations
Babcock & Wilcox NSSS

Figure A-2.4
Engineered Safety Feature Actuations
General Electric NSSS

Appendix A-3

New Plant Operational Data

Appendix A-3

New Plant Operational Data

New Plant Reactor Trip and Engineered Safety Feature (ESF) Actuation Data

Tables A-3.1 through A-3.4 show reactor trip data for new plants (plants licensed less than 2 years) by major nuclear steam supply system (NSSS) vendor for the years 1985 through 1989.

Table A-3.5 shows ESF actuation data for new plants for the years 1987 through 1989.

New Plant Composite Graphs

Figures A-3.1 and A-3.2 show composite plots for unplanned reactor trips and ESF actuations, respectively, for the new plants. The plots contain upper and lower bounds, which are derived from data presented in NUREG-1275, Vol. 1.*

*R. L. Dennig and P. D. O'Reilly, *Operating Experience Feedback Report - New Plants*, NUREG-1275, Vol. 1, July 1987.

Table A-3.1 New Plant Reactor Trip Initiating Systems

New Plant Westinghouse Reactors											
System	Total Trips						Trips/1000 Critical Hours¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	32	27	41	11	4	115	1.01	0.75	0.98	0.26	0.16
Turbine	12	4	20	1	7	44	0.38	0.11	0.48	0.02	0.28
RPS	13	7	12	5	3	40	0.41	0.19	0.29	0.12	0.12
Control rod drive	8	4	4	5	4	25	0.25	0.11	0.10	0.12	0.16
Electrical	12	2	2	4	3	23	0.38	0.06	0.05	0.09	0.12
Main generator	4	2	5	7	2	20	0.13	0.06	0.12	0.16	0.08
Condensate	0	3	9	4	1	17	0.00	0.08	0.21	0.09	0.04
Main steam	3	2	6	0	0	11	0.09	0.06	0.14	0.00	0.00
Support	1	2	3	2	0	8	0.03	0.06	0.07	0.05	0.00
RCS	2	0	1	1	0	4	0.06	0.00	0.02	0.02	0.00
Total	87	53	103	40	24	307	2.74	1.46	2.46	0.94	0.97

¹Critical hours: 1985 = 31,733; 1986 = 36,194; 1987 = 41,926; 1988 = 42,555; 1989 = 24,658.

New Plant General Electric Reactors											
System	Total Trips						Trips/1000 Critical Hours²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	5	7	13	5	0	30	0.18	0.35	0.48	0.38	0.00
Turbine	8	8	5	3	0	24	0.29	0.40	0.19	0.23	0.00
RPS	8	6	3	0	0	17	0.29	0.30	0.11	0.00	0.00
Support	6	4	1	3	0	14	0.22	0.20	0.04	0.23	0.00
Electrical	6	3	2	0	0	11	0.22	0.15	0.07	0.00	0.00
Condensate	1	3	5	0	0	9	0.04	0.15	0.19	0.00	0.00
RCS	2	2	4	1	0	9	0.07	0.10	0.15	0.08	0.00
Main generator	2	3	2	0	1	8	0.07	0.15	0.07	0.00	0.51
Control rod drive	1	1	3	0	0	5	0.04	0.05	0.11	0.00	0.00
Main steam	0	3	1	0	0	4	0.00	0.15	0.04	0.00	0.00
Total	39	40	39	12	1	131	1.44	2.02	1.45	0.92	0.51

²Critical hours: 1985 = 27,161; 1986 = 19,816; 1987 = 26,932; 1988 = 13,041; 1989 = 1,962.

Table A-3.1 (cont.)

New Plant Combustion Engineering Reactors

System	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Feedwater	14	7	1	0	0	22	1.54	0.43	0.13	0.00	0.00
RPS	3	6	2	0	0	11	0.33	0.37	0.25	0.00	0.00
Control rod drive	4	5	0	0	0	9	0.44	0.31	0.00	0.00	0.00
Condensate	6	2	1	0	0	9	0.66	0.12	0.13	0.00	0.00
Electrical	2	3	0	0	0	5	0.22	0.18	0.00	0.00	0.00
Main generator	1	3	0	0	0	4	0.11	0.18	0.00	0.00	0.00
Main steam	1	2	0	0	1	4	0.11	0.12	0.00	0.00	0.90
Support	0	2	1	0	0	3	0.00	0.12	0.13	0.00	0.00
Turbine	2	0	0	0	0	2	0.22	0.00	0.00	0.00	0.00
RCS	2	0	0	0	0	2	0.22	0.00	0.00	0.00	0.00
Total	35	30	5	0	1	71	3.85	1.84	0.63	0.00	0.90

³Critical hours: 1985 = 9,082; 1986 = 16,313; 1987 = 7,931; 1988 = 8,370; 1989 = 1,106.

Table A-3.2 New Plant Activities at Time of Reactor Trip

New Plant Westinghouse Reactors											
Activity	Total Trips						Trips/1000 Critical Hours ¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	26	18	41	18	15	118	0.82	0.50	0.98	0.42	0.61
Raising power	16	12	18	6	2	54	0.50	0.33	0.43	0.14	0.08
Testing	19	5	18	6	3	51	0.60	0.14	0.43	0.14	0.12
Maintenance	7	5	11	6	1	30	0.22	0.14	0.26	0.14	0.04
Reducing power	6	4	4	1	1	16	0.19	0.11	0.10	0.02	0.04
Troubleshooting	6	2	3	3	2	16	0.19	0.06	0.07	0.07	0.08
Special testing	0	4	6	0	0	10	0.00	0.11	0.14	0.00	0.00
Calibration	4	3	2	0	0	9	0.13	0.08	0.05	0.00	0.00
Unknown	3	0	0	0	0	3	0.09	0.00	0.00	0.00	0.00
Total	87	53	103	40	24	307	2.74	1.46	2.46	0.94	0.97

¹Critical hours: 1985 = 31,733; 1986 = 36,194; 1987 = 41,926; 1988 = 42,555; 1989 = 24,658.

New Plant General Electric Reactors											
Activity	Total Trips						Trips/1000 Critical Hours ²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	10	10	16	5	1	42	0.37	0.50	0.59	0.38	0.51
Testing	9	7	9	1	0	26	0.33	0.35	0.33	0.08	0.00
Raising power	11	8	3	2	0	24	0.40	0.40	0.11	0.15	0.00
Reducing power	1	1	6	2	0	10	0.04	0.05	0.22	0.15	0.00
Troubleshooting	2	4	1	0	0	7	0.07	0.20	0.04	0.00	0.00
Maintenance	1	2	2	2	0	7	0.04	0.10	0.07	0.15	0.00
Special testing	0	5	1	0	0	6	0.00	0.25	0.04	0.00	0.00
Calibration	1	3	1	0	0	5	0.04	0.15	0.04	0.00	0.00
Unknown	4	0	0	0	0	4	0.15	0.00	0.00	0.00	0.00
Total	39	40	39	12	1	131	1.44	2.02	1.45	0.92	0.51

²Critical hours: 1985 = 27,161; 1986 = 19,816; 1987 = 26,932; 1988 = 13,041; 1989 = 1,962.

Table A-3.2 (cont.)

New Plant Combustion Engineering Reactors

Activity	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Normal operation	7	17	0	0	1	25	0.77	1.04	0.00	0.00	0.90
Testing	9	5	1	0	0	15	0.99	0.31	0.13	0.00	0.00
Raising power	6	4	1	0	0	11	0.66	0.25	0.13	0.00	0.00
Maintenance	5	1	0	0	0	6	0.55	0.06	0.00	0.00	0.00
Unknown	5	0	0	0	0	5	0.55	0.00	0.00	0.00	0.00
Special testing	0	2	1	0	0	3	0.00	0.12	0.13	0.00	0.00
Reducing power	1	1	1	0	0	3	0.11	0.06	0.13	0.00	0.00
Troubleshooting	1	0	1	0	0	2	0.11	0.00	0.13	0.00	0.00
Calibration	1	0	0	0	0	1	0.11	0.00	0.00	0.00	0.00
Total	35	30	5	0	1	71	3.85	1.84	0.63	0.00	0.90

³Critical hours: 1985 = 9,082; 1986 = 16,313; 1987 = 7,931; 1988 = 8,370; 1989 = 1,106.

Table A-3.3 New Plant Reactor Trip Causes

New Plant Westinghouse Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	55	20	51	24	14	164	1.73	0.55	1.22	0.56	0.57
Personnel error	20	19	30	9	3	81	0.63	0.52	0.72	0.21	0.12
Procedure	6	5	9	4	1	25	0.19	0.14	0.21	0.09	0.04
Unknown	4	4	5	1	3	17	0.13	0.11	0.12	0.02	0.12
SG level	0	3	6	0	0	9	0.00	0.08	0.14	0.00	0.00
Natural phenomenon	2	2	0	2	2	8	0.06	0.06	0.00	0.05	0.08
Other	0	0	2	0	1	3	0.00	0.00	0.05	0.00	0.04
Total	87	53	103	40	24	307	2.74	1.46	2.46	0.94	0.97

¹Critical hours: 1985 = 31,733; 1986 = 36,194; 1987 = 41,926; 1988 = 42,555; 1989 = 24,658.

New Plant General Electric Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	19	26	22	7	1	75	0.70	1.31	0.82	0.54	0.51
Personnel error	14	8	11	3	0	36	0.52	0.40	0.41	0.23	0.00
Procedure	3	3	5	2	0	13	0.11	0.15	0.19	0.15	0.00
Unknown	2	2	1	0	0	5	0.07	0.10	0.04	0.00	0.00
Natural phenomenon	1	0	0	0	0	1	0.04	0.00	0.00	0.00	0.00
Other	0	1	0	0	0	1	0.00	0.05	0.00	0.00	0.00
Total	39	40	39	12	1	131	1.44	2.02	1.45	0.92	0.51

²Critical hours: 1985 = 27,161; 1986 = 19,816; 1987 = 26,932; 1988 = 13,041; 1989 = 1,962.

Table A-3.3 (cont.)

New Plant Combustion Engineering Reactors											
Cause	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Equipment	20	21	3	0	1	45	2.20	1.29	0.38	0.00	0.90
Personnel error	5	3	1	0	0	9	0.55	0.18	0.13	0.00	0.00
Procedure	3	4	1	0	0	8	0.33	0.25	0.13	0.00	0.00
SG level	4	1	0	0	0	5	0.44	0.06	0.00	0.00	0.00
Unknown	3	1	0	0	0	4	0.33	0.06	0.00	0.00	0.00
Total	35	30	5	0	1	71	3.85	1.84	0.63	0.00	0.90

³Critical hours: 1985 = 9,082; 1986 = 16,313; 1987 = 7,931; 1988 = 8,370; 1989 = 1,106.

Table A-3.4 New Plant Reactor Trip Signals

New Plant Westinghouse Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ¹				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Low SG level	29	27	35	8	3	102	0.91	0.75	0.83	0.19	0.12
Turbine trip	13	11	24	8	6	62	0.41	0.30	0.57	0.19	0.24
Manual	7	4	21	9	3	44	0.22	0.11	0.50	0.21	0.12
High negative flux rate	5	3	2	7	3	20	0.16	0.08	0.05	0.16	0.12
High RCS temperature	4	0	4	1	3	12	0.13	0.00	0.10	0.02	0.12
High reactor power	6	1	1	1	2	11	0.19	0.03	0.02	0.02	0.08
Steam flow feedflow mismatch	2	1	3	0	1	7	0.06	0.03	0.07	0.00	0.04
Low reactor coolant flow	2	1	1	1	1	6	0.06	0.03	0.02	0.02	0.04
Other	2	1	0	1	1	5	0.06	0.03	0.00	0.02	0.04
SRM	0	1	3	1	0	5	0.00	0.03	0.07	0.02	0.00
Reactor trip breaker open	2	1	1	0	1	5	0.06	0.03	0.02	0.00	0.04
RCP breaker open	4	0	0	1	0	5	0.13	0.00	0.00	0.02	0.00
Loss of power	1	0	1	1	0	3	0.03	0.00	0.02	0.02	0.00
PZR low pressure	1	1	1	0	0	3	0.03	0.03	0.02	0.00	0.00
Safety injection	1	0	2	0	0	3	0.03	0.00	0.05	0.00	0.00
Low steamline pressure	0	0	2	0	0	2	0.00	0.00	0.05	0.00	0.00
P-7 permissive	1	0	1	0	0	2	0.03	0.00	0.02	0.00	0.00
IRM	1	0	1	0	0	2	0.03	0.00	0.02	0.00	0.00
High steamline flow	2	0	0	0	0	2	0.06	0.00	0.00	0.00	0.00
Under voltage RCP	0	1	0	1	0	2	0.00	0.03	0.00	0.02	0.00
Unknown	2	0	0	0	0	2	0.06	0.00	0.00	0.00	0.00
High SG level	2	0	0	0	0	2	0.06	0.00	0.00	0.00	0.00
Total	87	53	103	40	24	307	2.74	1.46	2.46	0.94	0.97

¹Critical hours: 1985 = 31,733; 1986 = 36,194; 1987 = 41,926; 1988 = 42,555; 1989 = 24,658.

Table A-3.4 (cont.)

New Plant General Electric Reactors

Signal	Total Trips						Trips/1000 Critical Hours ²				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Low reactor water level	13	12	5	4	0	34	0.48	0.61	0.19	0.31	0.00
Turbine trip	9	4	7	3	1	24	0.33	0.20	0.26	0.23	0.51
Manual	4	5	11	3	0	23	0.15	0.25	0.41	0.23	0.00
High reactor power	2	5	3	1	0	11	0.07	0.25	0.11	0.08	0.00
IRM	2	6	2	1	0	11	0.07	0.30	0.07	0.08	0.00
High RCS pressure	5	3	2	0	0	10	0.18	0.15	0.07	0.00	0.00
High reactor water level	0	0	5	0	0	5	0.00	0.00	0.19	0.00	0.00
Other	1	3	0	0	0	4	0.04	0.15	0.00	0.00	0.00
Low condenser vacuum	0	2	1	0	0	3	0.00	0.10	0.04	0.00	0.00
MSIV closure	1	0	1	0	0	2	0.04	0.00	0.04	0.00	0.00
MS line radiation	0	0	1	0	0	1	0.00	0.00	0.04	0.00	0.00
Trip discharge volume high	0	0	1	0	0	1	0.00	0.00	0.04	0.00	0.00
ATWS	1	0	0	0	0	1	0.04	0.00	0.00	0.00	0.00
Loss of power	1	0	0	0	0	1	0.04	0.00	0.00	0.00	0.00
Total	39	40	39	12	1	131	1.44	2.02	1.45	0.92	0.51

²Critical hours: 1985 = 27,161; 1986 = 19,816; 1987 = 26,932; 1988 = 13,041; 1989 = 1,962.

Table A-3.4 (cont.)

New Plant Combustion Engineering Reactors											
Signal	Total Trips						Trips/1000 Critical Hours ³				
	1985	1986	1987	1988	1989	Total	1985	1986	1987	1988	1989
Low SG level	11	6	1	0	0	18	1.21	0.37	0.13	0.00	0.00
Low DNBR	2	11	0	0	0	13	0.22	0.67	0.00	0.00	0.00
High SG level	8	0	0	0	0	8	0.88	0.00	0.00	0.00	0.00
PZR high pressure	3	4	0	0	0	7	0.33	0.25	0.00	0.00	0.00
Other	3	0	2	0	0	5	0.33	0.00	0.25	0.00	0.00
Low reactor coolant flow	1	3	0	0	0	4	0.11	0.18	0.00	0.00	0.00
Unknown	4	0	0	0	0	4	0.44	0.00	0.00	0.00	0.00
High reactor power	0	2	1	0	0	3	0.00	0.12	0.13	0.00	0.00
Manual	1	0	1	0	0	2	0.11	0.00	0.13	0.00	0.00
MSIV closure	1	1	0	0	0	2	0.11	0.06	0.00	0.00	0.00
Turbine trip	0	2	0	0	0	2	0.00	0.12	0.00	0.00	0.00
Loss of power	0	1	0	0	0	1	0.00	0.06	0.00	0.00	0.00
High RCS pressure	1	0	0	0	0	1	0.11	0.00	0.00	0.00	0.00
Low steamline pressure	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.90
Total	35	30	5	0	1	71	3.85	1.84	0.63	0.00	0.90

³Critical hours: 1985 = 9,082; 1986 = 16,313; 1987 = 7,931; 1988 = 8,370; 1989 = 1,106.

Table A-3.5 New Plant Engineering Safety Feature Actuations

Plant Name	Vendor	1987	1988	1989	Total
Beaver Valley 2	W	15	7	8	30
Braidwood 2	W	0	11	1	12
Limerick 2	GE	NYL*	NYL	8	8
Palo Verde 3	CE	3	1	4	8
South Texas 1	W	13	22	6	41
South Texas 2	W	NYL	1	14	15
Vogtle 1	W	58	14	1	73
Vogtle 2	W	NYL	NYL	14	14
Total All New Plants		89	56	56	201
Total All New W Plants		86	55	44	185
Total All New GE Plants		0	0	8	8
Total All New CE Plants		3	1	4	8

*NYL means the plant is not yet licensed.

Figure A-3.1
New Plants
Composite Unplanned Reactor Trips

Figure A-3.2
New Plants
Composite ESF Actuations

Appendix B

Summary of 1989 Abnormal Occurrences (Reactors)

Abnormal Occurrences (AOs) for CY 1989 (Reactors)

AO No.	Title of AO	Report No. NUREG-0090	Description of Event
89-1	Plug Failure Resulting in Steam Generator Tube Leak at North Anna Unit 1	Vol. 12, No. 1	<p>On February 25, 1989, Unit 1 automatically tripped from 76-percent power. This trip was caused by a main feedwater regulatory valve closing as a result of a problem in the instrument air supply line. During recovery operations, the licensee identified a primary-to-secondary leak of about 74 gallons per minute (gpm) in the "C" steam generator (S/G) and declared a Station Alert. The licensee continued cool-down and depressurization operations and the Station Alert was terminated when the plant entered cold shutdown. At the time of the event, Unit 2 was in a refueling shutdown.</p> <p>Investigation showed that the leak was caused by hot-leg mechanical plug failure in an S/G tube that had been plugged by Westinghouse in 1985. The top portion of the plug was severed from the body of the plug and was propelled up the inner diameter of the tube by the primary system pressure, where it punctured the tube just above the U-bend transition. The puncture was approximately 2-1/4 inches long and 3/4-inch wide. The plug also dented an adjacent tube. Failure of the plug was attributed to primary water stress corrosion cracking of heat-treated plug material (Inconel 600) as a result of low mill annealing temperatures that rendered the material highly susceptible to stress corrosion cracking. The primary-to-secondary leak was well within the normal primary system makeup capability; in addition, the radiological releases were well below Technical Specification limits. However, the event showed that major safety concerns existed because (1) this was an unexpected means of failure for S/G tubes, (2) this was a potential common mode failure with the possibility of multiple S/G tubes failing, and (3) there were generic implications for other plants using such susceptible tube plugs.</p> <p>The licensee's recovery plan was to investigate (with Westinghouse) the cause of the failure, determine corrective actions, and place Unit 1 into a refueling/maintenance outage so that all S/G repairs could be made simultaneously. The susceptible plugs were identified in the S/Gs for both Unit 1 and Unit 2. Repairs consisted either of removing susceptible plugs and replacing them or inserting a different type of plug into susceptible plugs. Repairs were completed for the S/Gs of both units, and the NRC agreed that the plants could be restarted. Unit 2 returned to power operation by the end of April 1989. Unit 1 restarted in July 1989. The NRC continued to investigate the potential generic implications of heat-treated mechanical plugs used at plants designed by Westinghouse, Combustion Engineering, and Babcock and Wilcox.</p>

AO No.	Title of AO	Report No. NUREG-0090	Description of Event
89-2	Steam Generator Tube Rupture at McGuire Unit 1	Vol. 12, No. 1	<p data-bbox="746 442 1451 587">On March 7, 1989, Unit 1 experienced an S/G tube rupture while operating at 100-percent power. The licensee declared a Station Alert, manually tripped the reactor, and continued to implement procedures for plant cooldown. The Station Alert was terminated when the plant entered cold shutdown.</p> <p data-bbox="746 612 1459 874">The maximum tube leak rate was estimated to be between 540 and 600 gpm. This leak rate considerably exceeded the normal primary system makeup capability (i.e., operation with the centrifugal charging pumps operating as part of the chemical and volume control system). To keep up with the leak, the licensee switched the pumps to their safety injection lineup. Steam generator tube rupture is one of the design-basis accidents considered in the NRC safety review of nuclear power plants.</p> <p data-bbox="746 900 1459 1012">Significant S/G tube ruptures in which the leak rate considerably exceeds the normal primary system makeup capability occasionally occur, as at McGuire Unit 1 and previously at Ginna and North Anna Unit 1.</p> <p data-bbox="746 1038 1467 1300">The NRC sent an Augmented Inspection Team (AIT) to investigate the event. The team concluded that the operating crew performed competently but identified weaknesses in both normal and emergency operating procedures. The tube failure did not result in a radiological release to the environment that exceeded regulatory limits. The event did not result in a Technical Specification (TS) safety limit being exceeded. The whole-body and thyroid doses from this event were well within the TS limits.</p> <p data-bbox="746 1325 1467 1736">Investigation by the licensee showed that the rupture was about 3-3/4 inches long and was caused by intergranular stress corrosion cracking. The licensee's recovery actions included inspection of all tubes in all four S/Gs, metallurgical analysis of the ruptured tube, removal or plugging of tubes as necessary, and revision of the procedures that the AIT identified as requiring upgrading. The licensee committed to a 100-percent inspection of the inservice tubes of all S/Gs at both McGuire Units 1 and 2 at their next refueling outages. The NRC staff concurred with the corrective actions taken and the commitments made by the licensee. The NRC gave permission to restart Unit 1 on May 5, 1989. The plant attained criticality on May 9, 1989, and reached full-power operation on May 13, 1989.</p>

AO No.	Title of AO	Report No. NUREG-0090	Description of Event
89-6	Significant Deficiencies in Management Controls at Surry Nuclear Power Station	Vol. 12, No. 2	<p>On May 18, 1989, the NRC issued a Notice of Violation and a Proposed Imposition of Civil Penalty in the amount of \$500,000 to the licensee. This significant enforcement action was taken on the basis of the findings of several NRC inspections conducted from September 1, 1988, through March 4, 1989. These findings, together with several other escalated enforcement actions since early 1988, demonstrated a major breakdown in the administrative and managerial control system at the Surry Power Station. Deficiencies were identified at both the site and corporate management levels. Serious safety concerns were raised about the licensee's ability to identify and correct deficiencies without NRC intervention. These deficiencies generally involved (1) original design deficiencies that the licensee should have identified; (2) inadequate evaluation by the licensee of operating events, identified deficiencies, and NRC bulletins and information notices; (3) failure of the licensee to take timely corrective action for known deficiencies; and (4) failure by the licensee to take adequate corrective actions.</p> <p>The inspections from September 1, 1988, through March 4, 1989, included (1) an NRC AIT inspection to review the facts and circumstances associated with the failure of the Unit 1 refueling cavity seal that occurred on May 17, 1988; (2) an NRC safety system functional inspection that focused on the safety-related service water system and the recirculation system, including associated electrical systems; and (3) various inspections by NRC resident inspectors.</p> <p>Both Unit 1 and Unit 2 had been shut down since September 1988 as a result of equipment problems and refueling operations, respectively. The licensee agreed not to restart either unit until the NRC concurred. The licensee implemented an extensive corrective action program, made a number of changes in site management, and paid the civil penalty in full. Unit 1 and Unit 2 were restarted in July and September 1989, respectively.</p>
89-8	Significant Deficiencies Associated With the Containment Building Recirculation Sump at the Trojan Nuclear Power Plant	Vol. 12, No. 3	<p>During July 1989, NRC and licensee inspectors found significant deficiencies associated with the containment building recirculation sump. The sump is a large collecting reservoir designed to provide an adequate supply of water with a minimum amount of particulate matter to the emergency core cooling system (ECCS) during the recirculation phase of a loss-of-coolant accident (LOCA). In the recirculation</p>

AO No.	Title of AO	Report No. NUREG-0090	Description of Event
89-8 (cont.)			<p data-bbox="765 432 1486 722">phase of a LOCA, injection of water from the refueling water storage tank has been essentially completed and the ECCS is configured to recirculate water from the sump to the reactor coolant system. The deficiencies consisted of significant amounts of debris in the sump and missing and damaged sump screens. Therefore, debris larger than design-basis material could have passed into the ECCS suction in the sump and degraded portions of the ECCS or made them inoperable during the recirculation phase of a postulated LOCA.</p> <p data-bbox="765 747 1486 863">It is believed that the damaged and missing sump screens and some of the debris in the sump had existed for an extended period (at least one operating cycle of approximately a year and possibly since initial operation in 1975).</p> <p data-bbox="765 888 1486 1058">The licensee corrected all installation and operational discrepancies, upgraded procedures and inspection criteria, and instituted organizational changes. On October 5, 1989, the NRC issued a Notice of Violation and a Proposed Imposition of Civil Penalty in the amount of \$280,000 to the licensee. The licensee paid the civil penalty in full.</p>

Appendix C

Reports Issued During 1989 (Reactors)

AEOD Reports Issued in 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	
06/89	Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Nonreactors	NUREG-1272, Vol. 3, No. 2	
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 Maintenance 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	Fires 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

Engineering Evaluations (cont.)			
Date	Title	No.	Author
10/89	Excessive Valve Body Erosion at Brunswick	E908	E. Brown
12/89	Operator Actions During Operational Events	E909	S. Israel
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 416OV 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	

Technical Review Reports (cont.)			
Date	Title	No.	Author
09/89	Check Valve Failure Rates From NPRDS Data	T918	E. Brown
09/89	Failure of Overcurrent Protective Device at Palisades Unit 1	T919	S. Mazumdar
	Not issued	T920	
10/89	Inadequate Capacity of 4160 V 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

Appendix D

Reports Issued From 1980 Through 1988 (Reactors)

Table D-1
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	
09/88	Report to the U.S. Nuclear Regulatory Commission on Analysis and Evaluation of Operational Data—1987 Nonreactors (NUREG-1272, Vol. 2, No. 2)	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	Engineering Evaluation Report—Pump Damage Due to Low Flow Cavitation	E807	C. Hsu

Table D-1 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
12/88	Operational Experience Review of Potential Large Openings in Containment	E808	T. Cintula
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
05/88	Technical Review Report—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-2
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

Table D-2 (cont.)

Technical Review Reports			
Date	Title	No.	Author
01/87	Compression Fitting Failures	T701	H. Ornstein
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 Oskarshamn 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. Nuclear 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-3
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 1985	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	R. Freeman
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
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

Table D-3 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
12/86	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	T612	R. Tripathi

Table D-3 (cont.)

Incident Investigation Program Reports			
Date	Title	No.	Author
01/86	Loss of Power and Water Hammer Event at San Onofre, Unit 1, on November 21, 1985	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-1.)

Table D-4
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 Systems 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
03/85	Service Water System Air Release Valve Failures	E505	S. Salah

Table D-4 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
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

Table D-4 (cont.)

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

Table D-5
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, <i>Preliminary Assessment of LER Reporting Under 10 CFR 50.73</i>	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

Table D-5 (cont.)

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
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-5 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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 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-5 (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 Lost 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-6
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 Westinghouse 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 of Residual Heat Removal Pumps at BWR Nuclear Power Plants	E309	S. Rubin

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
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

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
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	T307	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	T316	M. Chiramal
06/83	Problems With Diesel Driven Containment Spray Pump at Zion Unit 2 on December 16, 1982	T317	D. Zukor
06/83	Failure of Recirculation Spray Service Water Motor-Operated Valves	T318	D. Zukor

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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 1	T321	D. Zukor
06/83	Gouges in Main Coolant System Piping at Diablo Canyon on April 19, 1983	T322	D. Zukor
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

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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 a 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-7
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	E205	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

Table D-7 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
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
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

Table D-7 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
06/82	Repetitive Overspeed Trips of the Steam Driven Emergency Feedwater Pump on Initial Start at Arkansas Nuclear One, Unit 2	E228	E. Imbro
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

Table D-7 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
10/82	Events Involving Loss of Electrical Inverters Including Attendant Inverters to Vital Instrument Buses	E246	F. Ashe
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 Over-pressurization 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 82-015-01T 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-8
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

Table D-8 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
06/81	Deliberate Pump Trip at Browns Ferry Unit 2 on April 6, 1981	E113	W. Lanning
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 Preoperational 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

Table D-8 (cont.)

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

Table D-9
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 120 V 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

Table D-9 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
08/80	Tie Breaker Between Redundant Class 1E Buses—Point Beach Units 1 and 2	E010	M. Chiramal
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 <i>Corbicula</i> 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

For the third consecutive year, the total number of outstanding AEOD recommendations has declined. At the beginning of 1987, 80 AEOD recommendations were outstanding. As of December 31, 1987, the number had decreased to 63, and by December 31, 1988, only 39 AEOD recommendations were outstanding. Because of the expeditious resolution of most outstanding generic issues by both NRC and industry initiatives, only 10 AEOD recommendations were still outstanding on December 31, 1989. Also, during 1989 AEOD made no new recommendations. Table E-1 presents the status of AEOD recommendations for the last 3 years.

AEOD's tracking system ensures that all formal AEOD recommendations are tracked until resolution is achieved. 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 as NRC unresolved safety issues (USIs) and generic issues (GIs).

In addition to the formal recommendations that are tracked and listed in this appendix, NRC program offices routinely implement additional actions that are based on AEOD suggestions contained in engineering evaluations and special reports. These AEOD suggestions are not formally tracked or closed out by AEOD.

Information about each recommendation that is outstanding or that was resolved during 1989, including a description and status for each recommendation, follows.

Table E-1 Status of AEOD Recommendations, 1987-1989

Status of AEOD Recommendations:	12/31/87	12/31/88	12/31/89
Added since previous report	7	6	0
Resolved, deleted, transferred, or combined	24	30	29
Currently outstanding	63	39	10
Included in unresolved safety issues, generic issues, and Three Mile Island Action Plan items	37	25	7
Addressed by NRC and industry	26	14	3
Office of Nuclear Regulatory Research prioritization of issues			
High priority	37	25	8
Medium priority	8	8	0
Low priority	2	0	1
No priority	16	6	1
Functional areas addressed by issues			
Procedures, training, etc.	34	11	2
Hardware modification	16	8	3
Equipment testing	6	5	3
Other	7	15	2

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C101

Responsible
AEOD Engineer: T. Cintula

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: This recommendation is included in Generic Issue (GI) 65, "Probability of Core Melt Due to Component Cooling Water System Failures." GI 65 was incorporated into task 2 of GI 23, "Reactor Coolant Pump Seal Failures." GI 23 will address the reliability of RCP seal cooling systems. 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. The RES staff is coordinating the proposed resolution package with the Office of Nuclear Reactor Regulation (NRR), the Office for Analysis and Evaluation of Operational Data (AEOD), and the Office of the General Counsel (OGC). The Committee To Review Generic Requirements (CRGR) will review this package, which will then published for public comment in late 1990. The final resolution of GI 23 is scheduled for late 1991.

Recommendation Source: Memorandum, C. Michelson to Harold R. Denton, dated January 20, 1982

Responsible
AEOD Engineer: M. Chiramal

Title or Subject: "Safety Concerns Associated with Reactor Vessel Level Instrumentation in BWRs"

Recommendation 1: Safety-related low-low reactor vessel level start of high-pressure coolant injection and reactor core isolation cooling systems should not be prevented or delayed by non-safety-related high-level trip.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	D. Thatcher	High

Status: Resolved. This recommendation was considered as part of GI 101. GI 101 was resolved by RES in June 1989.

Recommendation 2: The NRC must be assured that the protective functions of narrow-range level instrumentation operate in spite of adverse control system and protection system interaction.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	A. Szukiewicz	High

AEOD Recommendation Tracking System

Recommendation Source: Memorandum, C. Michelson to Harold R. Denton, dated January 20, 1982 (cont.)

Status: Resolved. This recommendation was included in USI A-47. USI A-47 was resolved and closed out when Generic Letter (GL) 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47 'Safety Implications of Control Systems in Nuclear Power Plants,' " was issued on September 20, 1989.

Reference: Memorandum, H. R. Denton to C. Michelson, dated March 19, 1982

Recommendation Source: Case Study AEOD/C202

Responsible AEOD Engineer: T. Cintula

Title or Subject: "Flow Blockage by Bivalve Mollusks at Arkansas Nuclear One and Brunswick"

Recommendation 1: Capability to measure cooling water flow should be provided for all safety-related equipment.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	T. Y. Chang	Medium

Status: Resolved. The AEOD recommendations were included in GI 51, "Reliability of Open Cycle Service Water Systems." GI 51 was resolved and closed when GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment," was issued on July 18, 1989.

Recommendation 2: Develop and implement biofouling control strategies.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	T. Y. Chang	Medium

Status: Resolved. (See status of Recommendation 1.)

Recommendation 3: Periodic inspection of service water system piping.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	T. Y. Chang	Medium

Status: Resolved. (See status of Recommendation 1.)

Recommendation 4: Periodic verification of overall heat transfer coefficient on multiple pass heat exchangers.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	T. Y. Chang	Medium

Status: Resolved. (See status of Recommendation 1.)

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C202 (cont.)

Recommendation 5: Periodic verification of cooling water flow to all safety-related equipment should be specified in Technical Specifications.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	T. Y. Chang	Medium

Status: Resolved. (See status of Recommendation 1.)

Recommendation Source: Memorandum, C. Michelson to Chairman Ahearne, "New Unresolved Safety Issues," dated August 4, 1980; Memorandum, C. Michelson to H. Denton, "Resolution of Issue Concerning Steamline Break with Small LOCA," dated June 23, 1982; and Case Study AEOD/C205, "ATOG As Applied to the April 1981 Overfill Event at ANO-1"

**Responsible
AEOD Engineer:** H. Ornstein

Title or Subject: "Safety Implications of Steam Generator Transients and Accidents"

Recommendation 1: Combined primary/secondary side blowdown should be a USI for Babcock & Wilcox Co. (B&W) plants.

Responsible Office/Div/Br	Contact	Priority
NRR/DEST/SRXB	R. Jones	Low

Status: B&W licensees, the Electric Power Research Institute, and the NRC are jointly funding a test facility to obtain test data for integral systems. The data will be used to resolve the uncertainties associated with a B&W plant's response to small-break loss-of-coolant accidents and other transients and accidents. On the basis of the results of the integral testing, this issue has been given a low priority.

Recommendation 2: Task Action Plan (TAP) A-47 should focus on equipment modifications or additions to preclude steam generator (SG) overfill as a credible event.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	A. Szukiewicz	High

Status: Resolved. This recommendation was included in ongoing USI A-47 and the B&W Owners Group reassessment. Most B&W plants have committed to install safety-grade systems to provide SG overfill protection. USI A-47 was resolved when GL 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47 'Safety Implications of Control Systems at Nuclear Power Plants,'" was issued on September 20, 1989.

AEOD Recommendation Tracking System

Recommendation Source:	Case Study AEOD/C401 and Memorandum, C. J. Heltemes, Jr., to H. Denton, dated March 16, 1984		
Responsible AEOD Engineer:	S. Salah		
Title or Subject:	"Low Temperature Overpressure Events at Turkey Point Unit 4"		
Recommendation 1:	Correct the low-temperature overpressure protection Technical Specifications in the five areas identified in the case study report.		
	Responsible Office/Div/Br	Contact	Priority
	NRR/PD4	D. Pickett	High
Status:	This recommendation was included in GI 94. After Committee To Review Generic Requirements (CRGR) and Advisory Committee on Reactor Safeguards (ACRS) review and resolution of their final response, the GI was transferred to NRR in November 1989 for final resolution and closeout of the issue. The NRR staff is evaluating issuance of a generic letter to close out this GI.		
Recommendation Source:	Case Study AEOD/C403		
Responsible AEOD Engineer:	T. Cintula		
Title or Subject:	"Edwin I. Hatch Unit 2 Plant Systems Interaction Event on August 25, 1982"		
Recommendation 1:	Evaluate the common-mode failure potential of safety systems caused by the harsh environment of water or steam from breaks outside containment being back-channelled through floor drain systems.		
	Responsible Office/Div/Br	Contact	Priority
	RES/DSIR/EIB	D. Thatcher	High
Status:	Resolved. This recommendation originally was to be evaluated as part of GI 77, "Back-flow Protection in Common Equipment and Floor Drain Systems." In 1986, NRR incorporated GI 77 into GI A-17, "Systems Interactions." The staff prepared a proposed resolution package for GI A-17 that included consideration of water intrusion and flooding of nuclear plant equipment. The <i>Federal Register</i> notice of the resolution of GI A-17 was published on August 22, 1989. Subsequently, a generic letter (GL 89-18) was issued to all licensees on September 6, 1989. The generic letter provides lessons learned for use by licensees as part of their individual plant examination evaluation of flooding from internal plant sources.		
Recommendation 2:	Supplemental arrangements should be provided to ensure timely isolation of the affected floor drain system if the results of the above evaluation result in unacceptable common-mode safety system failures.		

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C403 (cont.)

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	D. Thatcher	High

Status: Resolved. (See status of Recommendation 1.)

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/RPSIB	C. Graves	Medium

Status: 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. 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 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 are being examined for similarity to Indian Point 2. A scoping analysis is also under way for risks from hydrogen at boiling water reactors (BWRs). RES plans to resolve GI 106 by mid-1990.

Recommendation Source: Case Study AEOD/C502

Responsible
AEOD Engineer: P. Lam

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.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/RPSIB	H. Woods	High

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C502 (cont.)

Status: This recommendation was incorporated in GI 105, "Interfacing System LOCA [Loss-of-Coolant-Accident] at BWRs." The NRC has published two reports about this issue: NUREG/CR-5124, "Interfacing System LOCA in PWRs," and NUREG/CR-5154, "Interfacing System LOCA in BWRs." NRR is developing an NRC intersystem LOCA action plan. RES plans to complete GI 105 in late 1990.

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.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/RPSIB	H. Woods	High

Status: (See status of Recommendation 1.)

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

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/RPSIB	H. Woods	High

Status: (See status of Recommendation 1.)

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	H. Woods	High

Status: (See status of Recommendation 1.)

Recommendation Source: Case Study AEOD/C503

Responsible
AEOD Engineer: H. Ornstein

Title or Subject: "Decay Heat Removal Problems at U.S. PWRs"

Recommendation 5: NRR should consider removal of autoclosure interlocks to minimize loss-of-decay-heat-removal events.

Responsible Office/Div/Br	Contact	Priority
NRR/DEST/SRXB	R. Jones	High

Status: Resolved. GL 88-17 recommended that licensees review or consider removal of autoclosure interlocks. NRR is reviewing this issue on a plant-by-plant basis. As a result, some licensees have removed their autoclosure interlocks.

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C602

**Responsible
AEOD Engineer:** C. Hsu

Title or Subject: "Operational Experience Involving Turbine Overspeed Trips"

Recommendation 1: Alleviate the effect of slow response of the governor valve during the turbine startup transient by implementing a steam bypass modification for turbines equipped with a Woodward Model EG governor.

Responsible Office/Div/Br	Contact	Priority
RES/DREP/PRAB	S. Diab	High

Status: Resolved. This recommendation was included in GI 124, "Auxiliary Feedwater System Reliability." Reliability analyses have been received from the nuclear steam supply system vendors, site visits have been completed for all designated PWR licensees, a summary report was issued, and the generic issue was closed out in January 1989.

Recommendation 2: Avoid entrapped oil in the speed-setting cylinder by establishing administrative controls for bleeding off the entrapped oil, installing a controllable dump valve, or providing indication in the control room of a spinning turbine.

Responsible Office/Div/Br	Contact	Priority
RES/DREP/PRAB	S. Diab	High

Status: Resolved. (See status of Recommendation 1.)

Recommendation 3: Ensure the adequacy of the existing vendor-supplied calibration procedures for governor speed setting.

Responsible Office/Div/Br	Contact	Priority
RES/DREP/PRAB	S. Diab	High

Status: Resolved. (See status of Recommendation 1.)

Recommendation 4: Prevent water induction into the turbine by providing adequate provisions for condensate removal from the steam supply line to the auxiliary feedwater turbine.

Responsible Office/Div/Br	Contact	Priority
RES/DREP/PRAB	S. Diab	High

Status: Resolved. (See status of Recommendation 1.)

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C602 (cont.)

Recommendation 5: Minimize trip and reset problems by assessing the adequacy of the existing procedural instruction, upgrading the training program, and providing local indication as well as control room indication for trip and reset conditions.

Responsible Office/Div/Br	Contact	Priority
RES/DREP/PRAB	S. Diab	High

Status: Resolved. (See status of Recommendation 1.)

Reference: Memorandum, H. R. Denton to C. J. Heltemes, Jr., dated October 22, 1986.

Recommendation Source: Case Study AEOD/C603

Responsible
AEOD Engineer: E. Brown

Title or Subject: "A Review of Motor-Operated Valve Performance"

Recommendation 1: Expeditiously implement the recommendations in AEOD/C203 (May 1982) and AEOD/S503 (September 1985).

Responsible Office/Div/Br	Contact	Priority
Nuclear Management and Resources Council (NUMARC)	T. Tipton	High
RES/DSIR/EME	S. Aggarwal	
NRR/DET/EMEB	L. Marsh	

Status: Transferred. Although AEOD intends to trend the performance of motor-operated valves (MOVs) from analysis of licensee event report and nuclear plant reliability data system data, responsibility for the overall NRC coordination and oversight of the industry efforts was transferred to NRR by SECY-89-165 on May 31, 1989.

Recommendation 2: Require licensees to establish procedures and diagnostic capability to determine root causes of MOV failure to operate.

Responsible Office/Div/Br	Contact	Priority
NUMARC	T. Tipton	High

Status: Transferred. (See status of Recommendation 1.)

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C603 (cont.)

Recommendation 3: Require licensees to develop a strong training program to ensure that appropriate information and instructions are disseminated to operating and maintenance personnel.

Responsible Office/Div/Br	Contact	Priority
NUMARC	T. Tipton	High

Status: Transferred. (See status of Recommendation 1.)

Recommendation 4: The scope of Inspection and Enforcement (IE) Bulletin 85-03 should be extended to cover all safety-related MOV assemblies required to be tested in accordance with 10 CFR 50.55a(g).

Responsible Office/Div/Br	Contact	Priority
NUMARC	T. Tipton	High
NRR/DET/EMEB	L. Marsh	

Status: Transferred. (See status of Recommendation 1.)

Reference: Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," was issued on June 28, 1989, to implement the recommendation.

Recommendation Source: Case Study AEOD/C701 (March 1987)/NUREG-1275, Vol. 2 (December 1987)

**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	J. Wermiel	High
RES/DSR	G. Burdick	

Status: Information Notice (IN) 87-28, 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 (now resolved), whereas Recommendation 5 will be subsequently addressed. As a result of the generic letter, licensees have conducted reviews at their plants and have found numerous problems with air systems similar to those outlined in the case study. Many licensees have initiated corrective actions in accordance with guidance in the case study. NRR issued IN 89-26, "Instrument Air Supply Affecting Safety-Related Equipment," on March 7, 1989. This information notice addressed some inadequacies identified by licensees in their responses to GL 88-14. AEOD is working with industry (American Society of Mechanical Engineers and the American Nuclear Society) to formulate an air systems performance standard to ensure that licensees maintain their plant instrument air systems adequately.

AEOD Recommendation Tracking System

Recommendation Source: Case Study AEOD/C801 (NUREG-1275, Vol. 3)

Responsible
AEOD Engineer: P. Lam

Title or Subject: "Service Water System Failures and Degradations in LWRs"

Recommendation 1: Conduct, on a regular basis, performance testing of all heat exchangers that are cooled by the service water system and perform a safety function to verify heat exchanger heat transfer capability.

Responsible Office/Div/Br	Contact	Priority
NRR/DOEA/OGCB	C. V. Hodge	High

Status: Resolved. NRR issued Generic Letter 89-13, "Service Water Systems Problems Affecting Safety-Related Equipment," on July 18, 1989. The generic letter addressed all AEOD recommendations, as well as those from Region II and RES.

Recommendation 2: Require licensees to verify that their service water systems are not vulnerable to a single failure of an active component.

Status: Resolved. (See status of Recommendation 1.)

Recommendation 3: Inspect, on a regular basis, important portions of the piping of the service water system for corrosion, erosion, and biofouling.

Responsible Office/Div/Br	Contact	Priority
NRR/DOEA/OGCB	C. V. Hodge	High

Status: Resolved. (See status of Recommendation 1.)

Recommendation 4: Reduce human errors in the operation, repair, and maintenance of the service water system.

Responsible Office/Div/Br	Contact	Priority
NRR/DOEA/OGCB	C. V. Hodge	High

Status: Resolved. (See status of Recommendation 1.)

AEOD Recommendation Tracking System

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: M. Chiramal

Title or Subject: "Operational Restrictions for Class 1E 120 Vac 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.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	D. Thatcher	High

Status: This recommendation is currently being pursued under GI 128. GI 128 was reviewed by the CRGR in July 1989, but the ACRS has not agreed to the proposed resolution. Final closeout of the generic issue and issuance of the generic letter should be completed in 1990.

References: 1. Memorandum, H. Denton to C. Michelson, dated September 29, 1980
 2. Memorandum, H. Denton to D. Eisenhower, dated October 25, 1983
 3. Memorandum, H. Denton to C. Michelson, dated October 16, 1980

Recommendation Source: Engineering Evaluation AEOD/E304

Responsible AEOD Engineer: T. Cintula

Title or Subject: "Backflow Protection in Common Equipment and Floor Drain Systems"

Recommendation 1: Provide backflow protection for drain systems in older operating plants.

Responsible Office/Div/Br	Contact	Priority
RES/DSIR/EIB	D. Thatcher	High

Status: Resolved. This recommendation was originally to be evaluated as part of GI 77, "Backflow Protection in Common Equipment and Floor Drain Systems." In 1986, NRR incorporated GI 77 into GI A-17, "Systems Interactions." The staff prepared a proposed resolution package for GI A-17 that included consideration of water intrusion and flooding of nuclear plant equipment. The *Federal Register* notice of the resolution of GI A-17 was published on August 22, 1989. Subsequently, a generic letter (GL 89-18) was issued to all licensees on September 6, 1989. The generic letter provides lessons learned for use by licensees as part of their individual plant examination (IPE) evaluation of flooding from internal plant sources.

AEOD Recommendation Tracking System

Recommendation Source: Special Study Report AEOD/S602

**Responsible
AEOD Engineer:** J. Crooks

Title or Subject: "An Overview of Nuclear Power Plant Operating Experience Feedback Programs"

Recommendation 3: The NRC should continue to monitor operating experience (OE) and industry performance with regard to OE activities for some time (perhaps 2 years or longer). If sufficient improvement in the effectiveness of OE activities has not been observed after this period of time, further regulatory action would be appropriate. In this regard, AEOD believes the following actions should be initiated at this time:

1. NRR, in conjunction with AEOD, should reappraise the way in which the OE feedback activities might be improved through further guidance. The guidance could address the shortcomings noted in this study. The elements of an acceptable program could be defined in sufficient detail to permit determinations of program adequacy. Further, the guidance could address the need to (1) thoroughly assess both in-house and industry-wide experience; (2) disseminate applicable lessons to be used for the various training programs for nonsupervisory operating personnel; and (3) take corrective actions in the areas of procedural changes, training, and plant modifications. The guidance developed should, however, not be overly prescriptive and should be consistent with present requirements.
2. Consistent with Item 1 of this recommendation, NRR should develop a draft inspection module for reviewing the effectiveness of licensee practices for implementing OE feedback activities, including verification of such implementation by NRC on a plant-by-plant basis (at least annually).
3. NRR should consider revising current OE feedback reviews to include routinely a sampling of industry feedback documents.

Responsible Office/Div/Br	Contact	Priority
AEOD/DSP/TPAB	J. Crooks	Various
NRR/DOEA/LLPB	F. Hebdon	

Status: Resolved.

1. NRR and AEOD are monitoring the OE feedback programs as part of their normal functions.
 2. NRR and AEOD assessed the recommendations of the contract study on proposed improvements to generic communications. Any further actions will be based on current reviews.
 3. A sampling of industry feedback documents is currently available in the inspection procedures.
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AEOD Recommendation Tracking System

Recommendation Source: Special Study Report AEOD/S803

Responsible
AEOD Engineer: G. Lanik

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 and B-29 and anticipated transient without scram (ATWS) mitigation.

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

Status: The Boiling Water Reactor Owners Group (BWROG) submitted a letter (BWROG-9009) dated February 2, 1990, with the attachment "Licensing Basis for Long-Term Solutions to BWR Stability" that proposes long-term solutions, including hardware and protection system modifications, to avoid violation of safety limits as a result of power oscillations. The proposed solutions are derived from the analytical effort in response to Bulletin 88-07, Supplement 1, and are being reviewed by the NRC staff.

Both Brookhaven National Laboratory (BNL) and the BWROG have completed preliminary analyses to evaluate ATWS scenarios. The BWROG concluded that previous ATWS analyses are valid. BNL results indicate that some emergency procedure guideline modifications should be considered. The NRC staff is planning additional analyses to resolve the issue.

Appendix F

Status of NRC Staff Actions for Events Investigated by Incident Investigation Teams at Davis-Besse, San Onofre, and Rancho Seco in 1985

Appendix F

Status of NRC Staff Actions for Events Investigated by Incident Investigation Teams at Davis-Besse, San Onofre, and Rancho Seco in 1985

In accordance with NRC Manual Chapter 0513, "NRC Incident Investigation Program," dated August 5, 1987, upon receipt of an Incident Investigation Team (IIT) report, the Executive Director for Operations (EDO) shall identify and assign NRC office responsibility for 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 provides a written disposition and/or status, along with appropriate references, for each

of the NRC staff action items that were assigned by the EDO to the various NRC offices associated with the IIT reports on the 1985 events at Davis-Besse Nuclear Power Station, San Onofre Nuclear Generating Station, and Rancho Seco Nuclear Generating Station, respectively. (Details of the resolution of plant-specific items were not included for Rancho Seco because of the plant's shutdown status and the utility's decision to decommission.) All actions assigned by the EDO in response to these three IIT reports have been adequately addressed.

Under the "Disposition" section, an entry has been made to indicate whether action items are resolved or ongoing. Also, as a result of an NRC reorganization in 1987, the former Office of Inspection and Enforcement (IE) was abolished and its functions were assumed by other offices. For ongoing action items, these new offices are specifically designated.

AEOD IIT Action Tracking System

Action Source: IIT Report on Davis-Besse Event of June 9, 1985 (Reference 1)

Item 1: Adequacy of the licensee's management and maintenance practices.

- Actions:**
- (a) Evaluate and take action on the licensee's response to findings relating to corrective actions and preventive maintenance problems (including testing, root cause determination of equipment misoperation, and operating experience). (Responsible Office: Office of Nuclear Reactor Regulation (NRR))
 - (b) Evaluate and take action on the licensee's response to findings concerning management practices (e.g., control of maintenance programs and post-trip reviews). (Responsible Office: Region III (RIII))

Disposition: Resolved.

Reference 2 documents the NRC staff's evaluation of the licensee's response to all plant-specific findings regarding the event. In a number of cases, the staff relied on commitments made by the licensee to complete certain actions to address certain issues adequately (Reference 3). All licensee actions required before restart were completed. All other plant-specific actions required to resolve the IIT findings have been completed except for the human engineering enhancements in the control room that were completed during the January through May 1990 refueling outage in response to Item 10a, and the installation of a plant-specific simulator in response to Item 10c, which is tentatively scheduled for completion in 1991.

Item 2: Completion of analyses for loss-of-feedwater events.

- Actions:** Evaluate the time margins and consequences of alternative sequences for a loss-of-feedwater event at Davis-Besse. (Responsible Office: NRR)

Disposition: Resolved.

For plant-specific actions in response to Item 2, see the disposition of Item 1.

Item 3: Adequacy of the steam feedwater rupture control system (SFRCS).

- Actions:** Review the design for the SFRCS and the susceptibility of the SFRCS to (a) spurious actuations involving such items as main steam isolation valve (MSIV) closure and (b) single failures. (Responsible Office: NRR)

Disposition: Resolved.

For plant-specific actions in response to Item 3, see the disposition of Item 1.

Item 4: Interaction of plant security features and operator actions.

- Actions:** Evaluate the effect of security features (locked doors, locked equipment, etc.) on the operator's ability to gain prompt access to equipment required to perform safety actions outside the control room in accordance with emergency procedures. (Responsible Office: NRR)

Disposition: Resolved.

AEOD IIT Action Tracking System

Item 4 (cont.)

For plant-specific actions in response to Item 4, see the disposition of Item 1.

Reference 4 documents the staff action taken regarding Item 4 and a related operational experience evaluation performed by AEOD (Reference 5), which subsequently resulted in the issuance of an NRC information notice (Reference 6). In addition, Reference 7 documents that Item 4 was also evaluated under Generic Issue 122.3, "Physical Security System Constraints," which resulted in a low priority ranking.

Item 5: Availability of the shift technical advisor (STA).

Actions: Evaluate the time available and the role for STA assistance during a complex operating event. (Responsible Office: NRR)

Disposition: Resolved.

For plant-specific actions in response to Item 5, see the disposition of Item 1.

References 8 and 9 document the NRR review of the IIT report, which resulted in the identification of a number of short-term and long-term generic issues for staff action. The status of these generic actions is documented in Reference 7. The generic issue of the availability of the STA (Generic Issue 125.I.1, "Availability of the Shift Technical Advisor") was dropped on the basis of the results of a survey of licensees indicating that STAs are in the control room or are immediately available at the majority of operating plants. For the three plants identified with a deficiency, licensee corrective action was reviewed by the NRC staff on a plant-specific basis.

Item 6: Reliability of the auxiliary feedwater (AFW) containment isolation valves and other safety-related valves.

- Actions:**
- (a) Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)
 - (b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
 - (c) Determine if the safety function of the AFW containment isolation valves has been properly specified; that is, are the valves required to open as well as close for design-basis events. (Responsible Office: NRR)
 - (d) Verify that these valves constitute a single failure point of the AFW system for certain design-basis events. (Responsible Office: NRR)
 - (e) Determine that the procedures for adjustments of the AFW isolation valves, such as torque switch bypass switches, are clear and proper, and that the associated training programs are adequate. Confirm that adjustment settings are consistent with plant procedures. (Responsible Office: RIII)
 - (f) Determine if the engineering basis for the specification of the adjustments for safety-related valves, such as the torque switch and the torque switch bypass switch setting, is adequate for all design-basis events. (Responsible Office: NRR)
 - (g) Evaluate the test program for the AFW containment isolation valves to confirm operability for all design-basis events. (Responsible Office: NRR)

AEOD IIT Action Tracking System

Item 6 (cont.)

- (h) Evaluate whether other safety-related valves at Davis-Besse may be subject to the same type/cause of failure. (Responsible Office: NRR)
- (i) Conduct a review of failures of safety-related motor-operated valves and provide an assessment of pertinent failure modes affecting valve performance under design-basis conditions. (Responsible Office: AEOD)
- (j) Determine if further generic correspondence, such as an NRC bulletin, is warranted on this type/cause of failure of safety-related valves. (Responsible Office: IE)

Disposition: Resolved.

For plant-specific actions in response to Items 6a through 6h, see the disposition of Item 1.

Reference 10 documents the results of AEOD's review of safety-related motor-operated valve experience in response to Item 6i.

Reference 3 documents the IE evaluation of Item 6j. This evaluation determined that improper setting of torque and limit switches on motor-operated valves was a significant contributor to the event. As a result, the NRC issued a bulletin regarding this subject (Reference 11). The NRC subsequently issued a generic letter (Reference 12) that expanded the scope of the bulletin on the basis of an NRC staff assessment that indicated that the program to verify switch settings should be extended to ensure the operability of all safety-related fluid systems.

Item 7: Adequacy of emergency notifications.

- Actions:**
- (a) Verify the adequacy of the licensee's procedures and training for reporting of events to the NRC Operations Center. (Responsible Office: RIII)
 - (b) Review the adequacy of NRC guidance for determination of severity levels when plant conditions vary and may be stable when the licensee has an opportunity to report. (Responsible Office: IE)
 - (c) Review the adequacy of shift staffing for ensuring that knowledgeable individuals will be available for properly implementing the emergency plan during complex and long operational events. (Responsible Office: IE)

Disposition: Resolved.

For plant-specific actions in response to Item 7a, see the disposition of Item 1.

Reference 13 documents the IE review of Item 7b. As a result of this review, an NRC information notice (Reference 14) was issued to clarify the requirement that licensees notify the NRC Operations Officer of emergencies.

Reference 13 documents the IE review of Item 7c and concludes that it is not necessary or desirable to require additional personnel on shift to implement the emergency plan. NRC Information Notice 85-80 (Reference 14) addresses the specific corrective actions undertaken at Davis-Besse and provides generic recommendations to licensees to minimize the potential emergency planning problem raised by the Davis-Besse event.

AEOD IIT Action Tracking System

Item 8: Reliability of the AFW pump turbines.

- Actions:**
- (a) Monitor the licensee's troubleshooting activities, including possible hot plant operation to confirm failure mode. (Responsible Office: RIII)
 - (b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
 - (c) Evaluate the licensee's response and corrective actions relating to the unreliability of the AFW system (including the need for a third pump and turbine trip reset capability). (Responsible Office: NRR)
 - (d) Verify that the AFW system has been adequately tested to confirm the system configuration involved with design-basis events. (Responsible Office: RIII)
 - (e) Review the implementation of the operator training program to ensure proper operator actions, such as resetting of trip throttle valve. (Responsible Office: RIII)
 - (f) Conduct a review of past operating experience and determine the causes for overspeed turbine trips. (Responsible Office: AEOD)
 - (g) Determine the need for further generic correspondence on this mode/cause. (Responsible Office: IE)

Disposition: Resolved.

For plant-specific actions in response to Items 8a through 8e, see the disposition of Item 1.

Reference 15 documents AEOD's review of operating experience in response to Item 8f.

Reference 13 documents the IE assessment of the need for generic correspondence in response to Item 8g. It was determined that an NRC information notice was warranted (Reference 16). Following AEOD review of operating experience, a supplemental NRC information notice was issued (Reference 17) to address and communicate the results of the AEOD evaluation.

Item 9: Reliability of the pilot-operated relief valve (PORV).

- Actions:**
- (a) Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)
 - (b) Evaluate the licensee's engineering report on root cause analysis and proposed corrective actions. (Responsible Office: NRR)
 - (c) Determine the need for a test program to establish reliability. (Responsible Office: NRR)
 - (d) Determine if surveillance tests are necessary to confirm operational readiness. (Responsible Office: NRR)
 - (e) Determine if additional protection against PORV failure is necessary, that is, automatic block valve closure. (Responsible Office: NRR)

Disposition: Ongoing.

For plant-specific actions in response to Items 9a and 9b, see the disposition of Item 1.

AEOD IIT Action Tracking System

Item 9 (cont.)

Reference 7 documents the status of two generic issues (Generic Issue 125.I.2.A, "Need for a Test Program To Establish Reliability of the PORV," in response to Item 9c, and Generic Issue 125.I.2.B, "Need for PORV Surveillance Tests To Confirm Operational Readiness," in response to Item 9d). Both issues are being addressed in the resolution of Generic Issue 70, "PORV and Block Valve Reliability." Generic Issue 70 is considered resolved, based on the development of an NRC generic letter, which was presented and subsequently approved during Committee to Review Generic Requirements (CRGR) Meetings 167 and 168. The NRC generic letter is expected to be issued in 1990.

Reference 7 documents another generic issue (Generic Issue 125.I.2.C, "Need for Additional Protection Against PORV Failure") developed in response to Item 9e. This generic issue was dropped on the basis of the results of safety evaluation reports on the responses of three vendor groups (Combustion Engineering (CE), Babcock & Wilcox (B&W), and Westinghouse (W)) to TMI Action Plan Item II.K.3(2), "Report on Overall Safety Effect of PORV Isolation System." The safety evaluation reports conclude that an automatic PORV isolation system is not necessary and, in light of the control room indications available to the operators, the safety concerns pertaining to Item 9e have been adequately resolved.

Item 10: Adequacy of control room instrumentation and controls.

- Actions:**
- (a) Evaluate the adequacy of the SFRCS actuation controls and associated training program. (Responsible Office: NRR)
 - (b) Evaluate the adequacy of the installed control room instrumentation to allow operators to make the necessary and prompt determination with regard to procedure conformance and PORV position. (Responsible Office: NRR)
 - (c) Determine if NRC requirements should be revised regarding (1) safety parameter display system (SPDS) availability and (2) the need for a plant-specific simulator. (Responsible Office: NRR)

Disposition: Ongoing.

For plant-specific actions in response to Items 10a, 10b, and 10c, see the disposition of Item 1.

Reference 7 documents the status of the generic issue (Generic Issue 125.I.3, "SPDS Availability") developed in response to Item 10c. The NRC staff evaluated licensee and applicant implementation of the SPDS requirements at 57 units and found that a large percentage of designs did not satisfy the requirements identified in Supplement 1 to NUREG-0737 ("Clarification of TMI Action Plan Requirements," Supplement 1, published January 1983). As a result, NRC Generic Letter 89-06 ("Task Action Plan Item I.D.2—Safety Parameter Display System—10 CFR 50.54(f)," April 12, 1989) was issued to inform licensees of the staff's findings to aid in implementing SPDS requirements. With the issuance of NRC Generic Letter 89-06, the generic issue of SPDS availability was resolved and requirements were established. Reference 7 documents another generic issue (Generic Issue 125.I.4, "Plant-Specific Simulator") developed in response to Item 10c, which was addressed in the rulemaking amendments to 10 CFR Part 55 under TMI Action Plan Item I.A.4.2(4).

Item 11: Need for isolation of the startup feedwater pump.

- Actions:** Reassess acceptability of the provisions that resulted in the inability to place the startup feedwater pump in service from the control room. (Responsible Office: NRR)

AEOD IIT Action Tracking System

Item 11 (cont.)**Disposition:** Resolved.

For plant-specific actions in response to Item 11, see the disposition of Item 1.

Item 12: Resolution of equipment deficiencies.**Actions:** (a) Monitor the licensee's troubleshooting activities. (Responsible Office: RIII)

(b) Evaluate the licensee's report on the root cause analysis and corrective action for the equipment listed in Table 5.1 of that report and not addressed by other items in this action plan. (Responsible Office: NRR)

(c) Determine the need for generic correspondence on equipment problems. (Responsible Office: IE)

Disposition: Resolved.

For plant-specific actions in response to Items 12a and 12b, see the disposition of Item 1.

Reference 13 documents the IE evaluation of the need for generic correspondence on equipment problems in response to Item 12c. The evaluation determined that no further generic correspondence was warranted.

Item 13: Adequacy of plant procedures.**Actions:** Verify that plant procedures involving "drastic" actions are required to be sufficiently precise and clear to ensure prompt implementation. (Responsible Office: NRR)**Disposition:** Resolved.

Reference 7 documents the status of the staff's evaluation of the generic issue (Generic Issue 122.2, "Initiating Feed-and-Bleed") developed in response to Item 13. The staff concluded that no new regulatory requirements and guidance were needed. The staff determined that there was adequate reactor safety and ongoing industry initiatives would continue to enhance safety involving feed-and-bleed procedures. More specifically, the staff's conclusion was based on the following information: (1) as a result of the accident at Three Mile Island, the NRC requires licensees to implement new emergency operating procedures (EOPs) to prevent or to mitigate accidents; (2) licensees have EOPs in place that incorporate nuclear steam supply system (NSSS) vendor guidance for feed-and-bleed procedures; (3) licensees are continuing to enhance feed-and-bleed procedures, taking into account current NSSS vendor recommendations; and (4) NRC maintains ongoing licensing review and inspection activities concerning NSSS vendor and licensee enhancement of EOPs, including feed-and-bleed procedures. Thus, this issue was considered resolved, and no new requirements were established.

Item 14: Adequacy of safety system testing.**Actions:** Evaluate the NRC requirements and guidance to ensure that safety systems are tested in all configurations required by the design-basis analysis. (Responsible Office: NRR)**Disposition:** Resolved.

AEOD IIT Action Tracking System

Item 14 (cont.)

Reference 7 documents the status of the staff's evaluation of the generic issue (Generic Issue 125.I.5, "Safety Systems Tested in All Conditions Required by Design-Basis Accident (DBA)") developed in response to Item 14, and concludes that using integral plant and system testing, as near as practical to DBA conditions, to detect common mode design deficiencies seems too limited in its goal, considering the potentially large expenditure of time and resources that may be needed to develop the program. The estimated success probability of the tests to detect all unforeseen common mode design deficiencies results in a potential reduction in core-melt frequency of 3.4×10^{-7} /reactor-year, which borders between the generic issue ratings of "drop" and "low priority." On the basis of the results of the staff's evaluation and other considerations, this generic issue was dropped.

Item 15: Acceptability of current safety assessment methods.

Actions: Assess the implications of multiple independent and common mode failures as they relate to departures from design assumptions and specifications used in probabilistic safety analyses. (Responsible Office: Office of Nuclear Regulatory Research (RES))

Disposition: Resolved.

Reference 18 documents the RES actions to be taken in response to Item 15. References 19 through 24 describe the research results. This research resulted in the development of reliability technology applicable to the operational safety of nuclear power plants, and the results are being applied to improve the regulatory program.

- References:**
1. NUREG-1154, "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," dated July 1985.
 2. NUREG-1177, "Safety Evaluation Report Related to the Restart of Davis-Besse Nuclear Power Station, Unit 1, Following the Event of June 9, 1985," dated June 1986.
 3. Letter from J. Stolz (NRC) to J. Williams, Jr. (Toledo Edison), "Safety Evaluation Report Supporting Davis-Besse Restart—NUREG-1177; License Action Commitments," dated June 10, 1986.
 4. Memorandum from R. Vollmer to F. Hebdon (NRC), "Delayed Access to Safety-Related Areas During Plant Operation," dated March 19, 1986.
 5. AEOD Engineering Evaluation E403, "Delayed Access to Safety-Related Areas During Plant Operation," dated February 19, 1986.
 6. IE Information Notice 86-55, "Delayed Access to Safety-Related Areas and Equipment During Plant Emergencies," dated July 10, 1986.
 7. NUREG-0933, Supplements 1 to 10, "A Prioritization of Generic Safety Issues," December 1983 to July 1989.
 8. Memorandum from D. Crutchfield to H. Thompson (NRC), "Potential Immediate Generic Actions As a Result of the Davis-Besse Event of June 9, 1985," dated August 5, 1985.
 9. Memorandum from H. Thompson to T. Speis (NRC), "Longer Term Generic Actions As a Result of the Davis-Besse Event of June 9, 1985," dated November 6, 1985.

AEOD IIT Action Tracking System

References (cont.)

10. AEOD Case Study C603, "A Review of Motor-Operated Valve Performance," dated December 1986.
 11. IE Bulletin 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," dated November 15, 1985.
 12. NRC Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989.
 13. Memorandum from J. Taylor to V. Stello, Jr. (NRC), "Staff Actions Resulting From the Investigation of the June 9, 1985 Davis-Besse Event (NUREG-1154)," dated March 7, 1986.
 14. IE Information Notice 85-80, "Timely Declaration of an Emergency Class, Implementation of an Emergency Plan, and Emergency Notification," dated October 15, 1985.
 15. AEOD Case Study C602, "Operational Experience Involving Turbine Overspeed Trips," dated August 1986.
 16. IE Information Notice 86-14, "PWR Auxiliary Feedwater Pump Turbine Control Problems," dated March 10, 1986.
 17. IE Information Notice 86-14, Supplement 1, "Overspeed Trips of AFW, HPCI, and RCIC Turbines," dated December 17, 1986.
 18. Memorandum from R. Minogue to W. Dircks (NRC), "RES Actions Resulting From Investigation of Davis-Besse Event," dated October 4, 1985.
 19. NUREG/CR-4618, "Evaluation of Reliability Technology Applicable to LWR Operational Safety," dated August 1988.
 20. NUREG/CR-5425, "Evaluation of Allowed Outage Times (AOTs) From a Risk and Reliability Standpoint," dated July 1989.
 21. NUREG/CR-5200, "Evaluation of Risks Associated With AOT and STI Requirements at the ANO-1 Nuclear Power Plant," dated August 1988.
 22. NUREG/CR-5021, "Users' Guide for PRISIM Arkansas Nuclear One Unit 1," Volumes 1 and 2, dated March 1988.
 23. NUREG-1377, "NRC Research Program on Plant Aging: Listing and Abstracts of Reports Issued Through February 1, 1989," dated August 1989.
 24. "Engineering Simulator Applications to Emergency Response Training," R.J. Beelman et al., *Proceedings of the SCS Multi-Conference on Simulators VI, 28-31 March 1989*, Tampa, Florida (available from Society for Computer Simulation, P.O. Box 17900, San Diego, California).
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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)

Note: Following the loss-of-power and water-hammer event that occurred on November 21, 1985, the NRC developed an action plan for the review of licensee actions required as a result of the event and the subsequent IIT report. This plan involved actions by Region V, NRR, and IE. The NRC concerns and the licensee commitments were largely addressed before the startup of Unit 1 (July 15, 1986). This effort was documented in Inspection Report No. 50-206/86-34 (Reference 2).

Item 1: Adequacy of feedwater check valves to perform safety function.

Actions: Implement and coordinate the staff and industry actions necessary to ensure the reliability of safety-related check valves. (Responsible Office: IE; other offices to assist as requested)

Areas to be evaluated include

- Licensee's engineering report on root cause analysis and proposed corrective actions
- Adequacy of check valve design for this application
- Adequacy of Inservice Testing (IST) Program and procedures to detect degraded and failed valves
- Adequacy of check valves (and related testing programs) in other systems such as the residual heat removal (RHR) system

Disposition: Resolved.

Reference 3 documents the IE review of plant-specific aspects of the event in response to Item 1 and concludes that plant-specific check valve problems have been adequately resolved. The generic adequacy of check valve designs and IST programs was coordinated with utility owners groups and the Institute of Nuclear Power Operations (INPO), and the results were disseminated to the industry. This activity is documented in References 4 and 5. Concurrent with this industry activity, the NRC staff conducted in-depth plant inspections to assess check valves in accordance with NRC Inspection Manual Inspection Procedure 73756, "Inservice Testing of Pumps and Valves," dated March 16, 1987. These inspections found that licensee programs to address check valves ranged from little attention to reasonably good attention. In April 1989, the Nuclear Industry Check Valve Group was formed.

This group plans to develop specific guidelines for plant actions pertaining to database information on check valve performance, non-intrusive examination procedures to assess check valve performance, and maintenance procedures. In addition, the NRC recently issued a generic letter (Reference 6), which covers certain aspects of IST requirements for check valves.

Item 2: Completeness of resolved Unresolved Safety Issue (USI) A-1, "Water Hammer."

Actions: Assess the need to reevaluate USI A-1 to address specifically the potential for and prevention of condensation-induced water hammers in feedwater piping (assuming the issue concerning check valve integrity will be resolved in Item 1). (Responsible Office: NRR)

AEOD IIT Action Tracking System

Item 2 (cont.)

Disposition: Resolved.

Reference 7 documents the results of the NRR reassessment of USI A-1 in response to Item 2 and concludes that the event at San Onofre was due to grossly failed check valves in the feedwater system and a basis did not exist for reopening the water-hammer safety issue. Additionally, resolution of USI A-1 recognized that water-hammer events would continue to occur and that such events would not produce unacceptably high contributions to core-melt frequency or public risk. The imposition of new requirements to reduce water-hammer events is not supportable by current cost-benefit guidelines and, thus, no further action is warranted at this time.

Item 3: Adequacy of San Onofre Unit 1 design.

Actions: Implement and coordinate the staff's actions to reevaluate the following San Onofre design features. (Responsible Office: NRR)

- Manual loading of the diesel generators following a loss-of-power event
- Manual actuation of steamline isolation valves and assurance of steam generator availability to remove decay heat
- Lack of steam generator blowdown status in control room
- Adequacy of the licensee's design change to eliminate spurious safety injection (SI) indication upon loss of power

Disposition: Resolved.

Reference 8 documents NRR's resolution of design issues in response to Item 3. Reference 9 documents the resolution of the last outstanding item identified in Reference 8 and, thus, all plant-specific design issues have been resolved.

Item 4: Adequacy of post-trip review.

- Actions:**
- (a) Evaluate NRC requirements for ensuring that sufficient event data are retrievable to reconstruct accurately the event following a loss of offsite power. (Responsible Office: NRR)
 - (b) Evaluate the licensee's process for post-trip review and evaluation, including the thoroughness of the review and oversight provided by the onsite and offsite nuclear safety review groups. (Responsible Office: Region V (RV))

Disposition: Resolved.

Reference 10 documents the NRR evaluation in response to Item 4a and concludes that this issue was adequately addressed as part of Item 1.2 of NRC Generic Letter 83-28 ("Required Actions Based on Generic Implications of Salem ATWS Events," issued July 8, 1983).

Reference 11 documents the RV assessment of the adequacy of the licensee's safety review process in response to Item 4b and concludes that the licensee's process for post-trip review and evaluation meets regulatory requirements.

AEOD IIT Action Tracking System

Item 5: Adequacy of licensee's recordkeeping practices.

Actions: Evaluate the adequacy of the licensee's maintenance records. (Responsible Office: RV)

Disposition: Resolved.

Reference 12 documents the RV assessment of the adequacy of the licensee's maintenance records in response to Item 5 and concludes that the licensee's recordkeeping practices were satisfactory.

Item 6: Adequacy of operator training and/or procedures.

Actions: Review the implementation of the training program regarding operator understanding and actions in the areas of electrical systems and invoking Technical Specification action statements. (Responsible Office: RV)

Disposition: Resolved.

Reference 11 documents the RV assessment of the licensee's training program in response to Item 6 and concludes that operator training and/or procedures were adequate.

Item 7: Adequacy of emergency notifications and NRC response.

- Actions:**
- (a) Verify the adequacy of the licensee's procedures and training for reporting of events to the NRC Operations Center. (Responsible Office: RV)
 - (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 the overall plant status. (Responsible Office: AEOD)

Disposition: Ongoing.

Reference 13 documents the RV assessment of the adequacy of the licensee's reporting of events to the NRC in response to Item 7a and concludes that it was adequate.

References 14 and 15 document the IE evaluation in response to Item 7b and conclude that the NRC policy on the use of the ENS was clear. To ensure that the NRC policy regarding the use of NRC personnel as ENS communicators would be followed, action was taken, as documented in Reference 15, to communicate NRC's policy and guidance regarding the resident inspector's role during licensee events to all regions. Actions that were taken to improve the ability to determine the overall plant status included development of the Emergency Response Data System (ERDS), increased emphasis on site-specific training for NRC Headquarters Operations Officers (HOOs), and development of various site-specific information systems for use by the HOOs. These actions, as documented in References 15 and 16, are in various stages of implementation and/or are being upgraded, and with the exception of the ERDS, are expected to be fully implemented within a year. The ERDS is expected to be fully operational in about 3 years.

Item 8: Significance of backlog of license amendments.

Actions: Evaluate whether a backlog of license amendments and Technical Specification changes contributed to delays in approving the licensee's IST program. (Responsible Office: NRR)

AEOD IIT Action Tracking System

Item 8 (cont.)

Disposition: Resolved.

Reference 17 documents the NRR evaluation of the backlog of licensee amendments in response to Item 8 and concludes that the backlog of license amendments did not delay approval of the licensee's IST program.

- References:**
1. NUREG-1190, "Loss of Power and Water Hammer Event at San Onofre, Unit 1, on November 21, 1985," dated January 1986.
 2. NRC Inspection Report No. 50-206/86-34, dated July 28, 1986.
 3. Memorandum from J. Taylor to J. Martin (NRC), "Staff Actions Resulting From the November 21, 1985 San Onofre Unit 1 Water Hammer Event," dated June 6, 1986.
 4. Institute of Nuclear Power Operations (INPO), Significant Operating Experience Report (SOER) 86-03, "Check Valve Failures or Degradations," dated October 1986. (Proprietary)
 5. Electric Power Research Institute (EPRI), Report NP-5479, "Check Valve Application Guidelines," dated January 1988.
 6. NRC Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," dated April 3, 1989.
 7. Memorandum from H. Denton to V. Stello, Jr. (NRC), "Review and Assessment of Waterhammer Occurrences Since CY 1981," dated July 7, 1986.
 8. Letter from T. M. Novak (NRC) to K. P. Baskin (Southern California Edison Company), "Technical Evaluation of Issues From the NRC Staff Action Plan in Response to the November 21, 1985 Event at San Onofre Unit 1," dated November 12, 1986.
 9. Letter from C. M. Trammell (NRC) to H. B. Ray (Southern California Edison Company), "Long-Term Electrical Cable Monitoring Program—San Onofre Nuclear Generating Station, Unit 1 (TAC 73782)," dated December 8, 1989.
 10. Memorandum from H. Denton to V. Stello, Jr. (NRC), "Post-Trip Data Following Loss of Offsite Power," dated August 4, 1986.
 11. NRC Inspection Report No. 50-206/86-20, dated June 9, 1986.
 12. NRC Inspection Report No. 50-206/86-22, dated June 9, 1986.
 13. NRC Inspection Report No. 50-206/86-16, dated June 2, 1986.
 14. Memorandum from K. Perkins to G. Zech (NRC), "Response to 01/04/86 Memo From Stello to Denton on the 11/21/86 San Onofre Investigation," dated February 10, 1986.
 15. Memorandum from G. Grimes to D. Kirsch (NRC), "IRB Response to San Onofre 1 IIT Action Item List," dated June 11, 1986.
 16. Memorandum from K. Perkins to G. Zech (NRC), "IRB Response to San Onofre 1 IIT Action Item List," dated February 13, 1986.

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References (cont.)

17. Memorandum from H. Denton to V. Stello, Jr. (NRC), "San Onofre Nuclear Generating Station, Unit 1 (SONGS-1) Inservice Testing (IST) Review," dated April 28, 1986.
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Action Source: IIT Report on Rancho Seco Incident of December 26, 1985 (Reference 1)

Note: Only the disposition and/or status of generic actions for Rancho Seco are addressed because of the plant's shutdown status and the utility's decision to decommission the facility. However, it should be noted that all plant-specific issues that require resolution for plant restart were resolved, as documented in References 2 and 3.

Item 1: Adequacy of the auxiliary feedwater (AFW) system.

- Actions:**
- (a) Verify the acceptability of the existing initiation and control of the AFW system. (Responsible Office: NRR)
 - (b) Determine the status of any licensee commitment to install the emergency feedwater initiation and control (EFIC) system, and determine the acceptability of the current schedule for installation. (Responsible Office: NRR)

Disposition: Resolved.

Reference 4 documents the status of licensee commitments deemed acceptable by the staff in response to Item 1b and states that all other plants designed by Babcock & Wilcox (B&W) have completed the safety-grade upgrade of the AFW system, except Three Mile Island (TMI) Unit 1, which completed the upgrade during the November 1986 refueling outage.

Item 2: Completeness of various staff and licensee actions associated with control systems.

- Actions:**
- (a) In light of the ongoing B&W generic review, assess the need to reevaluate the actions taken by the staff and by the licensees in response to the findings, conclusions, and recommendations associated with BAW-1564 (B&W technical report, "Integrated Control System Reliability Analysis," issued August 1979); Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control System Bus During Operation," issued November 30, 1979); NUREG-0667 ("Transient Response to Babcock & Wilcox-Designed Reactors," issued May 1980); the February 1980 loss of non-nuclear instrumentation (NNI) power at Crystal River; the March 19, 1984, partial loss of NNI at Rancho Seco; and BAW-1791 (B&W technical report, "B&W Owners Group Probabilistic Evaluation of Pressurized Thermal Shock, Phase 1 Report," issued June 1983). (Responsible Office: NRR)
 - (b) Assess the need to expand the scope of Unresolved Safety Issue (USI) A-47 to include additional consideration of frequent events with undesirable consequences even if the consequences of a particular event are bounded by the final safety analysis report (FSAR) analysis, and the degree to which events that are not significant at the referenced plants might be significant at other plants. (Responsible Office: NRR)
 - (c) Consider the need for issuing further generic communications. (Responsible Office: NRR)

Disposition: Ongoing.

Reference 5 documents the NRR assessment of prior actions taken by the licensees and the NRC staff and details the NRC staff's plans and ongoing activities by the B&W Owners Group (B&WOG) in response to Item 2a. This item, along with others, was examined under the B&WOG Plant Reassessment Program, which was initiated in response to a number of significant events at B&W-designed reactors that led the NRC staff to conclude that the basic design requirements for B&W reactors needed to be reexamined. The B&WOG assumed a leadership role in this effort. The 208 specific

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Item 2 (cont.)

tasks that resulted from this effort were evaluated by the NRC staff. References 6 and 7 document the results of the NRC staff's evaluation. In addition, the staff is auditing the implementation of these tasks at each B&W-designed facility.

Reference 4 documents the NRR assessment of the need to expand the scope of USI A-47 in response to Item 2b and concludes that further expansion does not appear appropriate.

Reference 8 documents the IE assessment of the need for further generic communications in response to Item 2c and concludes that further generic communications do not appear warranted as the issues are specific to B&W-designed plants and are already being addressed by the B&WOG and the NRC staff.

Item 3: Adequacy of the design of the integrated control system (ICS).

Actions: Assess the adequacy of the design of the ICS. (Responsible Office: NRR)

Particular features to be included are

- (a) Whether the ICS is sufficiently reliable to ensure that the frequency of unnecessary safety challenges is acceptably low (i.e., is the ICS properly classified as a non-safety-related system?)
- (b) Loss of remote (i.e., hand) power coincident with loss of automatic control
- (c) Results of Sacramento Municipal Utility District's (SMUD's) analysis to date of the power supply monitor
- (d) Results of SMUD's contractor analysis of the power supply monitor
- (e) Role of the power supply monitor as a potential single failure in the ICS and/or NNI
- (f) Results of a study of the response of the ICS upon restoration of power
- (g) Acceptability of the failure mode of meters and recorders that are affected by a loss of ICS power

Disposition: Resolved.

The evaluation of the adequacy of the ICS in response to Items 3a through 3g was addressed by the B&WOG Plant Reassessment Program, and the improvements proposed by the B&WOG were evaluated by the staff in Section 6.1 of Reference 6. The staff concluded in its safety evaluation report that with proper implementation of the B&WOG recommendations, the performance of the ICS will be significantly improved.

Item 4: Adequacy of the maintenance program for manual isolation valves.

- Actions:**
- (a) Evaluate the need for a maintenance program for manual isolation valves in safety-related systems. (Responsible Office: RV)
 - (b) Evaluate the adequacy of industry standards and NRC requirements regarding periodic testing and maintenance of manual valves in safety-related systems. (Responsible Office: NRR)

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Item 4 (cont.)

Disposition: Resolved.

For disposition of plant-specific Item 4a, see the note at the beginning of this section.

Reference 4 documents the NRR evaluation of the adequacy of industry standards and NRC requirements in response to Item 4b and concludes that the industry does not have a formal maintenance program that applies to every manual valve, nor are there NRC requirements for maintenance and testing of “convenience” valves. Actions taken to address this item included issuance of an NRC information notice (Reference 9) and development of a generic issue (Generic Issue 127, “Maintenance and Testing of Manual Valves in Safety-Related Systems”). Evaluation of this generic issue indicates that there is little or no prospect of safety improvements that are both substantial and worthwhile and that it is not clear from currently available information whether the generic issue merits further attention.

Item 5: Adequacy of procedures and training.

Actions: Evaluate the adequacy of the procedures and operator (licensed and nonlicensed) training, particularly with regard to

- (a) The degree to which event-specific procedures (e.g., loss of ICS, station blackout) are needed to recover quickly from events that have been diagnosed by the operators, and to mitigate such events if the initiating condition cannot be immediately corrected. (Responsible Office: NRR)
- (b) The consistency between emergency operating procedures (EOPs) and NRC-approved procedure generation packages. (Responsible Office: RV)
- (c) The consistency between EOPs and NRC-approved procedure generation packages. (Responsible Office: IE)
- (d) The adequacy of procedural guidance concerning (1) when to trip auxiliary feedwater pumps and (2) the relative priorities of avoiding the pressurized thermal shock (PTS) region and maintaining pressurizer level. (Responsible Offices: NRR and RV)
- (e) The adequacy of the training of nonlicensed operators in the use of valve wrenches, the methods for manually overriding and operating valves, and the use of various indications of valve position. (Responsible Office: RV)
- (f) The adequacy of the annunciator response procedures (e.g., the Annunciator Procedures Manual). (Responsible Office: RV)
- (g) The adequacy of the program to ensure that all applicable procedures and operator training are reviewed and updated when plant modifications are made. (Responsible Office: RV)

Disposition: Resolved.

For disposition of plant-specific Items 5b and 5d through 5g, see the note at the beginning of this section.

Reference 4 documents the NRR evaluation of event-specific procedures and concludes that there appears to be no basis for the generic concern regarding the need for event-specific procedures expressed in Item 5a. This item, however, was included under the B&WOG Plant Reassessment Program.

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Item 5 (cont.)

Reference 4 documents the actions taken by IE and NRR in response to Item 5c, which included the issuance of an NRC information notice (Reference 10), implementation of an EOP inspection program by NRR, and the issuance in June 1986 of a temporary instruction to the regions to inspect EOPs at operating plants.

Reference 4 documents that the generic aspects of Item 5d were addressed by the B&WOG Plant Reassessment Program as part of B&WOG's procedural review effort.

Item 6: Adequacy of the radiological control and emergency preparedness program.

Actions: Evaluate the adequacy of the radiological control and emergency preparedness program, including procedures, operator training, equipment availability, and coordination between operating and health physics personnel. (Responsible Office: RV)

Disposition: For disposition of plant-specific Item 6, see the note at the beginning of this section.

Item 7: Adequacy of the FSAR accident analysis.

Actions: Verify the adequacy of the Rancho Seco FSAR accident analysis, particularly the degree to which credit is given for the non-safety-related ICS and the non-safety-related main steamline failure logic. (Responsible Office: NRR)

Disposition: Resolved.

Reference 4 documents that Item 7 was handled as a plant-specific issue and was subsequently addressed as part of the licensee restart evaluation documented in Reference 2. The scope of the review performed focused primarily on the effects of the ICS on the FSAR accident analysis. In addition, Reference 3 documents the resolution of all the remaining plant-specific issues that had not been resolved in Reference 2. Thus, all plant-specific issues that required resolution for the restart of Rancho Seco were resolved.

Item 8: Adequacy of required staffing.

Actions: Evaluate the adequacy of plant staffing to deal with expected operational transients. (Responsible Office: NRR)

Disposition: Resolved.

Reference 4 documents that Item 8 was addressed by the B&WOG Plant Reassessment Program as part of the procedural review effort.

Item 9: Adequacy of the troubleshooting program.

Actions: Evaluate the adequacy of the licensee's program to troubleshoot damaged equipment in a controlled and systematic manner to determine the root cause and appropriate corrective actions. (Responsible Office: RV)

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Item 9 (cont.)

Disposition: For disposition of plant-specific Item 9, see the note at the beginning of this section.

- References:**
1. NUREG-1195, "Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985," dated February 1986.
 2. NUREG-1286, "Safety Evaluation Report Related to the Restart of Rancho Seco Nuclear Generating Station, Unit 1, Following the Event of December 26, 1985," dated October 1987.
 3. NUREG-1286, Supplement 1, "Safety Evaluation Report Related to the Restart of Rancho Seco Nuclear Generating Station, Unit 1, Following the Event of December 26, 1985," dated March 1988.
 4. Memorandum from H. Denton to V. Stello, Jr. (NRC), "Staff Actions Resulting From the Investigation of the December 26, 1985 Incident at Rancho Seco (NUREG-1195)," dated October 7, 1987.
 5. Memorandum from H. Denton to V. Stello, Jr. (NRC), "Completeness of Various Staff and Licensee Actions Associated With Control Systems," dated July 23, 1986.
 6. NUREG-1231, "Safety Evaluation Report Related to the Babcock and Wilcox Owners Group Plant Reassessment Program," dated November 1987.
 7. NUREG-1231, Supplement 1, "Safety Evaluation Report Related to Babcock and Wilcox Owners Group Plant Reassessment Program," dated March 1988.
 8. Memorandum from J. Taylor to V. Stello, Jr. (NRC), "Staff Actions Resulting From the Investigation of the December 26, 1985 Incident at Rancho Seco (NUREG-1195)," dated September 24, 1986.
 9. IE Information Notice 86-61, "Failure of Auxiliary Feedwater Manual Isolation Valve," dated July 28, 1986.
 10. IE Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures," dated August 14, 1986.