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1990 ANNUAL REPORT

POWER REACTORS

U.S. NUCLEAR REGULATORY COMMISSION

JULY 1991

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
- *Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Power Reactors*, NUREG-1272, Vol. 4, No. 1, July 1990
- *Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Nonreactors*, NUREG-1272, Vol. 4, No. 2, July 1990

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 1990. The report is published in two separate parts. NUREG-1272, Vol. 5, No. 1, covers power reactors and presents an overview of the operating experience of the nuclear power industry from the NRC perspective, including comments about the trends of some key performance measures. The report also includes the principal findings and issues identified in AEOD studies over the past year and summarizes information from such sources as licensee event reports, diagnostic evaluations, and reports to the NRC's Operations Center.

NUREG-1272, Vol. 5, No. 2, covers nonreactors and presents a review of the events and concerns during 1990 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations. The reports contain a discussion of the Incident Investigation Team program and summarize the Incident Investigation Team and Augmented Inspection Team reports for that group of licensees. Each volume contains a list of the AEOD reports issued for 1980 through 1990.

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Abbreviations

ac	alternating current	CST	condensate storage tank
ACRS	Advisory Committee on Reactor Safeguards	CY	calendar year
ADM	Office of Administration	DBA	design-basis accident
ADS	automatic depressurization system	dc	direct current
ADV	atmospheric dump valve	DE	diagnostic evaluation
AEOD	NRC's Office for Analysis and Evaluation of Operational Data	DEP	Diagnostic Evaluation Program
AFW	auxiliary feedwater	DET	Diagnostic Evaluation Team
AHU	air handling unit	DG	diesel generator
AIT	Augmented Inspection Team	DHR	decay heat removal
ANO	Arkansas Nuclear One	ECCS	emergency core cooling system
AO	abnormal occurrence	EDG	emergency diesel generator
AOT	allowed outage time	EDO	Executive Director for Operations
APRM	average power range monitor	EED	Engineering Evaluations Department
APS	Arizona Public Service Co.	EER	engineering evaluation request
ASCO	Automatic Switch Company	EFIC	emergency feedwater initiation and control
ASME	American Society of Mechanical Engineers	EFO	equipment-forced outage
ASP	Accident Sequence Precursor	EHC	electrohydraulic control
ASTM	American Society for Testing and Materials	ENS	emergency notification system
ATOG	anticipated transient operating guideline	EOP	emergency operating procedure
ATWS	anticipated transient without scram	EP	emergency power
B&W	Babcock & Wilcox Co.	EPG	emergency procedure guideline
B&WOG	Babcock & Wilcox Owners Group	EPRI	Electric Power Research Institute
BNL	Brookhaven National Laboratory	ERDS	Emergency Response Data System
BWR	boiling-water reactor	ESF	engineered safety feature
BWROG	Boiling Water Reactor Owners Group	ESFAS	engineered safety features actuation system
CCP	centrifugal charging pump	ESW	emergency service water
CDPr	core damage probability	FAA	Federal Aviation Administration
CE	Combustion Engineering	FEF	Foreign Event File
CEI	Cleveland Electric Illuminating Co.	FOR	forced outage rate
CFR	<i>Code of Federal Regulations</i>	FSAR	Final Safety Analysis Report
CP&L	Carolina Power and Light Co.	FW	feedwater
CRDM	control rod drive mechanism	GDC	General Design Criterion
CRE	collective radiation exposure	GE	General Electric
CREV	control room emergency ventilation	GI	generic issue
CREVS	control room emergency ventilation system	GL	generic letter
CRGR	Committee To Review Generic Requirements	gpm	gallons per minute
		HELB	high-energy line break
		HOO	Headquarters Operations Officer
		HPCI	high-pressure coolant injection

Abbreviations (cont.)

HPCS	high-pressure core spray	NMSS	Office of Nuclear Material Safety and Safeguards
HVAC	heating, ventilation, and air conditioning	NNI	non-nuclear instrumentation
		NPRDS	Nuclear Plant Reliability Data System
IAEA	International Atomic Energy Agency	NPSH	net positive suction head
IAP	integrated action plan	NRC	U.S. Nuclear Regulatory Commission
ICS	integrated control system	NRR	NRC's Office of Nuclear Reactor Regulation
IE	NRC's Office of Inspection and Enforcement	NSO	nuclear station operator
IIP	Incident Investigation Program	NSSS	nuclear steam supply system
IIT	Incident Investigation Team	NTSB	National Transportation Safety Board
IN	information notice	NUBARG	Nuclear Utility Backfitting and Reform Group
INEL	Idaho National Engineering Laboratory	NUDOCS	Nuclear Documents System
INES	International Nuclear Event Scale	NUMARC	Nuclear Management and Resources Council
INPO	Institute of Nuclear Power Operations		
INR	Immediate Notification Report	OE	operating experience
IPE	individual plant examination	OECD/NEA	Organization for Economic Cooperation and Development/Nuclear Energy Agency
IRB	AEOD's Incident Response Branch		
IRM	intermediate range monitor	OGC	NRC's Office of the General Counsel
IRS	Incident Reporting System	OIG	NRC's Office of the Inspector General
IST	inservice testing	ORNL	Oak Ridge National Laboratory
		OTM	overspeed trip mechanism
LBB	leak before break	P	pressure
LER	licensee event report	PDP	positive displacement pump
LOCA	loss-of-coolant accident	PI	performance indicator
LOOP	loss-of-offsite power	PN	preliminary notification
LOSP	loss of station power	PORV	power-operated relief valve
LPCI	low-pressure coolant injection	PRA	probabilistic risk assessment
LTOP	low-temperature overpressure protection	PRB	plant review board
LWR	light-water reactor	PRISIM	plant risk statistical information management
LWRPA	light-water reactor plant analyzer	PSA	probabilistic safety assessment
		PTS	pressurized thermal shock
MCCB	molded-case circuit breaker	PUC	Public Utility Commission
ME	maintenance effectiveness	PWG	principal working group
MOV	motor-operated valve	PWR	pressurized-water reactor
MS	main steam	PZR	pressurizer
MSICV	main steam isolation check valve		
MSIV	main steam isolation valve	QA	quality assurance
NA	not available		
NASA	National Aeronautics and Space Administration		
NEA	Nuclear Energy Agency		
NED	Nuclear Engineering Department		

Abbreviations (cont.)

RAT	reserve auxiliary transformer	SOV	solenoid-operated valve
RBCCW	reactor building closed cooling water	SPDS	safety parameter display system
		SPIP	Safety and Performance Improvement Program
RCIC	reactor core isolation cooling	SRM	source range monitor
RCP	reactor coolant pump	SRV	safety-relief valve
RCS	reactor coolant system	SSA	safety system actuation
RES	NRC's Office of Nuclear Regulatory Research	SSES	Susquehanna Steam Electric Station
RHR	residual heat removal	SSF	safety system failure
RPS	reactor protection system	STA	shift technical advisor
RWCU	reactor water cleanup	STI	surveillance test interval
RWST	refueling water storage tank	SW	service water
RI	NRC Region I		
RII	NRC Region II	TAC	task action commitment
RIII	NRC Region III	TAP	Task Action Plan
RIV	NRC Region IV	TCM	technical committee meeting
RV	NRC Region V	TE	Toledo Edison Co.
		TMI	Three Mile Island
		TS	Technical Specification
		TTC	Technical Training Center
SAIC	Science Applications International Corporation		
SALP	Systematic Assessment of Licensee Performance	UAT	unit auxiliary transformer
SAT	startup auxiliary transformer	UNPEDE	Union of Producers and Distributors of Electrical Energy
SCATS	stripping, cycling, and thermal stratification	USI	unresolved safety issue
SCC	stress corrosion cracking	VCT	volume control tank
SCE	Southern California Edison Company		
SCS	Society for Computer Simulation	W	Westinghouse
SCSS	Sequence Coding and Search System	WANO	World Association of Nuclear Operators
SE	significant event	WNP	Washington Nuclear Plant
SER	Safety Evaluation Report	WPPSS	Washington Public Power Supply System
SFRCS	steam feedwater rupture control system		
SG or S/G	steam generator		
SGTR	steam generator tube rupture		
SI	safety injection		
SINET	Shared Information Network		
SIS	safety injection system		
SMUD	Sacramento Municipal Utility District		
SNUPPS	standardized nuclear unit power plant system		
SOER	significant operating experience report		
SONGS	San Onofre Nuclear Generating Station		

EXECUTIVE SUMMARY

Since its formation in 1979, the NRC Office for Analysis and Evaluation of Operational Data (AEOD) has provided, as one of its primary missions, a strong, independent capability for the analysis of operational data. The office serves as the focal point for the independent assessment of operational events and manages the review, analysis and evaluation of reactor plant safety performance. It is also responsible for the NRC's Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and the Incident Investigation Program. In addition, AEOD provides support for the Committee to Review Generic Requirements. The office's activities related to nonreactors and technical training are reported separately.

To accomplish this mission, the AEOD staff collects, analyzes, and disseminates operational data, develops performance indicators and assesses trends in performance from these data. Other elements that contribute to this mission include diagnostic evaluations, incident investigations and incident response, and technical training.

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. Aside from the recommendations made as a result of a recently completed case study, the outstanding AEOD recommendations continue to be addressed through generic issues and initiatives taken by industry.

AEOD provides support for the Committee to Review Generic Requirements (CRGR). This committee of senior NRC managers reviews proposed new generic requirements and staff

positions and then provides recommendations to the Executive Director of Operations before requirements are promulgated. In addition to generic backfitting, AEOD also provides oversight of NRC plant-specific backfitting, including training of NRC personnel, auditing of NRC practices, and obtaining industry feedback regarding these activities.

Through many activities of AEOD, the overall safety performance of power reactors may be inferred. Comparison of 1990 data with data from past years shows a continuing overall improving trend. Specifically, performance indicators for reactor trips, significant events and safety system actuations show an improving trend, while collective radiation exposure, equipment forced outages and the forced outage rate have remained relatively constant during the last three years. The industry average number of safety system failures increased slightly. This small increase is attributed to industry design-basis reconstitution programs which have identified conditions that could result in safety system failures. Data from completed analyses using the Accident Sequence Precursor (ASP) Program through 1989 indicate that the number of risk significant operating events remains small. In 1990, there was only one reactor event reported to Congress as an Abnormal Occurrence.

The Accident Sequence Precursor (ASP) Program continues to quantitatively evaluate reactor operational experience. This evaluation of significant operational experience complements qualitative evaluations described in this report. The precursor methodology provides a means to integrate actual initiating events and plant conditions, and the reliability of standby safety equipment into an overall quantitative assessment, expressed as the conditional core damage probability.

Overall ASP Program results, in terms of quantity and characteristics of precursors for 1989, are similar to results of preceding years. Although few

In number, steam generator tube leaks and ruptures, loss-of-offsite power events, emergency diesel generator reliability, and service water problems continue to be judged quantitatively risk significant.

ASP evaluations performed in prior years have been limited to events that occurred or could have occurred while the plants were "at power." The Vogtle event of March 20, 1990, which occurred while the plant was shut down focused NRC's attention to the risk significance of events during shutdown. The ASP methodology is being extended to cover a range of shutdown events.

AEOD administers the Commission's program for reporting abnormal occurrences (AOs) to Congress. AOs are incidents or events that the Commission determines are significant from the standpoint of public health and safety. In 1990, NRC reported one AO at a commercial power reactor (Vogtle) to Congress. The number of AOs has remained low since 1987 – fewer than four per year.

In 1990, AEOD issued 1 case study, 2 special reports, 10 engineering evaluations, and 18 technical review reports. These documents provide analyses of safety issues that arose from the study of nuclear plant operating experience. These reports are widely distributed to communicate the lessons learned within the NRC and to industry.

The case study, NUREG 1275, Volume 6, "Operating Experience Feedback Report—Solenoid-Operated Valve Problems," focused on the vulnerability of safety-related equipment to common-mode failures or degradations of solenoid operated valves (SOVs). The report identifies deficiencies that were found in design, application, manufacture, maintenance, surveillance testing, and feedback of failure data.

The other studies provided the basis for five information notices, advising licensees of (1) overspeed of turbine-driven auxiliary feedwater

pumps and overpressurization of associated piping, (2) the potential for residual heat removal pump damage caused by parallel pump interaction, (3) the factors affecting the lift set-point of pressurizer safety valves, (4) the effects of internal flooding, and (5) testing of power-operated relief valves with alternate pneumatic supplies. A report on thermal stratification provided information for a supplement to an NRC bulletin on that subject.

In 1990, about 2700 immediate notifications, approximately 400 fewer than last year, were telephoned to the NRC Operations Center. Of these 2700 events, 151 were classified as "Unusual Events," 10 as "Alerts," and 1 as a "Site Area Emergency" (Vogtle Unit 1). The latter two categories are events during which a substantial degradation of plant safety occurred or had the potential to occur. In addition to prompt screening of telephone notifications, AEOD reviewed and evaluated licensee event reports (about 2300 in 1990) to identify events or aspects of events with safety significance.

During 1990, one reactor event was judged to have a sufficiently high level of safety significance to warrant an onsite review by an NRC Incident Investigation Team (IIT). The event involved the loss of all vital ac power (and thus the residual heat removal system) during mid-loop operation at Vogtle Unit 1. The Vogtle IIT report identified several generic lessons or concerns. The most significant ones were related to the management of risk when a nuclear power plant is shut down.

NRC conducted nine Augmented Inspection Team (AIT) inspections at nuclear power plants. Examples of lessons learned and communicated to licensees from 1990 AIT inspections include possible failures of main steam isolation check valves, potential problems related to refueling activities, and potential problems resulting from the re-energization of Rosemount transmitter trip units.

AEOD provided oversight for review of performance at two licensed reactor facilities under the Diagnostic Evaluation Program: (1) the Zion Nuclear Power Station and (2) the Oyster Creek Nuclear Generating Station. These evaluations provided NRC senior management with an independent assessment of the licensee's overall performance in ensuring safe plant operations and identified plant-specific performance issues that the licensees should improve or correct.

AEOD continued to maintain and improve the exchange of operational experience and ideas with foreign countries as a contribution to nuclear power safety programs worldwide. In 1990, AEOD received 96 reports of events at foreign reactors through the Nuclear Energy Agency (NEA) Incident Reporting System (IRS). These reports were reviewed to identify issues of importance to the U.S. nuclear power program and to disseminate findings from studies of these events to U.S. licensees. During 1990, AEOD prepared and submitted to the international community about 60 reports of U.S. incidents that addressed operational events and generic concerns. From discussions with regulators in other countries, these reports have been valuable to their nuclear safety programs. In 1990, AEOD

became responsible for providing computer services for operation and maintenance of the IRS events database for the Paris-based NEA. The IRS includes reports from NEA members and those submitted to the International Atomic Energy Agency (IAEA) by its member countries under similar reporting criteria. Thus, the database contains information on worldwide nuclear power plant experience.

In addition to these programs and as a part of NRC's overall international program, AEOD exchanges information and ideas on a variety of topics of international interest. For example, the AEOD staff provided assistance to foreign countries and to the 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.

Through review, analysis and evaluation of domestic and foreign events, AEOD continues to identify and communicate lessons learned from operating experience.

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, Diagnostic Evaluation, Technical Training and Incident Investigation Programs. In addition to managing these programs, AEOD provides administrative and technical support for the NRC Committee To Review Generic Requirements, which is responsible for recommending to the Executive Director for Operations the disposition of proposed benefits, requests for information and new requirements to be applied to power reactors. 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 1990 AEOD Annual Report is published in two separate parts: *Power Reactors* and *Nonreactors*. The report on power reactors, Vol. 5, 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 1990 abnormal occurrences.

- Appendix C lists AEOD reports issued in 1990.
- Appendix D lists AEOD reports issued from 1980 through 1989.
- Appendix E presents the status of outstanding recommendations included in AEOD studies.
- Appendix F presents the status of NRC staff actions resulting from the findings of NRC Incident Investigation Teams (IITs).
- Appendix G presents the status of NRC staff actions involving potential generic issues resulting from the findings of NRC Diagnostic Evaluation Teams.

The report on nonreactors, Vol. 5, No. 2, presents a review of the events and concerns during 1990 associated with the use of licensed material in nonreactor applications, such as personnel overexposures and medical misadministrations.

2 Reactor Performance Trends

AEOD collects, stores and analyzes a wide range of operational data as described more fully in later sections of this report (See Section 5, Operating Experience Data Programs). A subset of this information has been selected for quarterly review in the NRC Performance Indicator Program.

The Performance Indicator (PI) Program is one aspect of the Commission's efforts to monitor performance of licensees who operate commercial nuclear power plants in the United States. Under the direction of AEOD, this program 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 safety system actuations, (3) the number of significant events, (4) the number of safety system failures, (5) the forced outage rate, (6) the number of equipment-forced outages per 1000 commercial critical hours, (7) the collective radiation exposure, and (8) cause codes. Each quarter, the AEOD staff provides a report containing plant-specific data for these eight PIs to the Commission and NRC senior managers. The reports are also placed in the NRC Public Document Room. In addition, the staff provides plant-specific information and industry average data extracted from each PI report to licensee managers.

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 that must be used in conjunction with other tools, such as the results of routine and special inspections and the Systematic Assessment of Licensee Performance, to provide data to NRC managers who must decide whether any plant-specific regulatory programs need adjusting. The PIs for a given plant, when viewed as a set, provide additional data for assessing changes in plant operational performance. The PIs focus attention on

the need to assess and understand underlying causes of identified changes by evaluating other available information.

This section provides selected industry trends and patterns insights provided by tracking and analyzing the Performance Indicators from 1986 through 1990. Figure 2.1 shows the industry averages for these seven indicators (cause code trends are not included because industry-wide averages are not calculated for this indicator). In calculating these industry-wide averages, data for certain periods have been excluded for plants that either (1) were in extended shutdowns requiring Commission approval before either startup or operation above low power or (2) have ceased commercial operation. Table 2.1 provides a list of these plants and the calendar quarters in which they were excluded.

Automatic Reactor Scrams

The Performance Indicator Program monitors the number of unplanned automatic reactor scrams per plant while the reactor is critical. Reactor scrams 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 that the average number of automatic scrams while critical per plant has continued to decline since 1986, although at a lower rate in 1990 than in previous years. The quarterly PI automatic reactor scram data for each plant for 1989 and 1990 are provided in Table A-1.1 of Appendix A-1.

The addition of 55 unplanned manual scrams while critical to the automatic scrams tracked by the Performance Indicator, results in an industry total of

Figure 2.1 Annual industry performance averages—1986-1990

231 scrams while critical for 1990. This was a 5 percent decrease from the total number of scrams while critical in 1989, and continues the long-term downward trend in the number of scrams. The total number of automatic and manual scrams and the scram rate while critical for 1986 through 1990 are listed in Table A-2.1 in Appendix A-2. The leading cause of scrams was equipment failure, which caused approximately three times as many events as the next leading cause (personnel error). Of the equipment failures, main feedwater system failures caused approximately three times as many events as failures in the next leading system (main steam). The component involved in the highest number of feedwater system failures was the feedwater control valve.

Table 2.1 Plants in an extended shutdown between 1986 and 1990

Plant Name	Quarters Excluded
Browns Ferry 1	85-2 through 90-4
Browns Ferry 2	85-4 through 90-4
Browns Ferry 3	85-2 through 90-4
Davis-Besse	85-3 through 86-4
Fort St. Vrain	86-1 through 87-1, 88-3 through 88-4 & after 89-3
LaCrosse	after 87-2
Palisades	86-3 through 87-1
Peach Bottom 2	87-2 through 89-1
Peach Bottom 3	87-2 through 89-3
Pilgrim	86-3 through 88-4
Rancho Seco	86-1 through 88-1 & after 89-2
Seabrook	86-4 through 89-1
Sequoyah 1	86-1 through 88-3
Sequoyah 2	86-4 through 88-1
Shoreham	after 85-3

Figure 2.2 shows the combined automatic and manual unplanned reactor scrams per 1000 hours by calendar year for the four Nuclear Steam Supply System (NSSS) vendors. While these data are aggregated by NSSS, since scram reduction efforts have historically been coordinated by NSSS owners

groups, there is no implication of a strong cause-effect relationship between scram rate and NSSS. Indeed, there is a large variance in individual plant scram rates within the NSSS groupings.

The combined automatic and manual reactor scram data for mature plants were evaluated independently of the data for new plants. The mature-plant scram data are summarized in Tables A-2.2 through A-2.5 in Appendix A-2. The new-plant scram data are summarized in Tables A-2.6 through A-2.9 in Appendix A-2. A new plant typically experiences more scrams during startup and early commercial operation. This can often be attributed to a causes that can only be uncovered as the plant proceeds through startup testing, and to lack of operator experience with power operation of the as-built plant. While a general cutoff of 24 months after licensing is used to distinguish new plant from mature, operating experience has indicated that in some cases a higher than average scram rate can persist beyond the two year cutoff.

Fifty-four percent of all scrams in mature plants in 1990 were caused by equipment failures. Problems in the main feedwater system continued to initiate the greatest number of scrams in mature plants.

Safety System Actuations

The PI Program monitors two types of engineered safety feature (ESF) actuations collectively designated as PI safety systems actuations (SSAs): (1) actuations of certain emergency core cooling systems and (2) actuations of the emergency ac electrical system that are caused by loss of power to a vital bus. In general, plant systems that are designated ESFs vary considerably among plants, and the circumstances of actuation range in immediate safety significance. Thus, the SSA PI focuses on two key safety features found at all plants in order to provide a fairly standardized measure of challenges to plant safety systems.

Unplanned Reactor Scrams

Idaho National Engineering Laboratory, EG&G Idaho, Inc.
05/17/91

Figure 2.2 Unplanned reactor scrams/1000 critical hours by calendar year

Figure 2.1 shows that the trend in the annual average number of SSAs per plant by year decreased from 1986 to 1988 but has been essentially constant since then. Table A-1.2 of Appendix A-1 contains plant-specific data for this PI for 1989 and 1990.

In 1990, the industry experienced a total of 773 actuations for all engineered safety feature (ESF) actuations, including the 119 actuations which were considered in the PI program. This total represented a 15 percent decrease from the 1989 number. Grouping plants by NSSS, this decrease is attributable to the reduction in the number of events reported by GE plants (357 events in 1990 as compared with 476 in 1989) and B&W plants (from 62 events in 1989 to 16 events in 1990). A single plant accounted for most of the decrease in the number of events at B&W plants, with only 3 control room emergency ventilation system (CREVs) actuations in 1990 compared to 49 in 1989. The decrease in the number of actuations at GE plants was broadly based, with no one plant or group of plants or systems accounting for the change. ESF actuation data are given in Appendix A-2 in Tables A-2.10 through A-2.13. The engineered safety features actuations by calendar year for the four vendors are shown in Figures 2.3 through 2.6. As for reactor scrams, the values in the figures should not be used to imply differences in safety performance among the NSSS plant types.

For each NSSS vendor, 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 "unnneeded." The latter are unnecessary challenges to the safety systems.

The emergency power (EP) and emergency core cooling system (ECCS) actuation plots are subsets of the "Total" ESF events. They show the trends of all unplanned actuations of these systems (i.e., including actuations beyond those counted in the Performance Indicator Program for those systems). 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. The plot labeled "More than ECCS, Emergency Power, and HVAC" accounts for actuations of ESFs other than those shown in their own plots, and contains contributions from a variety of systems.

For General Electric (GE) plants, an additional plot is provided for the Reactor Water Cleanup (RWCU) system, which is unique to GE NSSS plants, because that system accounts for a large fraction of all ESF events. However, the contribution has steadily declined through 1990.

Significant Events

Significant events are those events that the NRC staff identifies as meeting certain selection criteria. The identification 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, the primary coolant pressure boundary, or important associated structures; (4) a reactor trip with complications; (5) an unplanned release of radioactivity exceeding plant Technical Specifications (TS) or regulations; (6) operation outside the limits of TS; and (7) other events that are considered significant.

Figure 2.1 shows that the trend in the average number of significant events per plant by year decreased from 1986 to 1987. From 1987 through 1989, the number of significant events was relatively constant. In 1990, the number of significant events was about half of the 1989 number. Table A-1.3 of

Engineered Safety Feature Actuations Babcock & Wilcox

Idaho National Engineering Laboratory, EG&G Idaho, Inc.
08/17/91

Figure 2.3 Babcock & Wilcox reactor engineered safety feature actuations by calendar year

Engineered Safety Feature Actuations Combustion Engineering

Idaho National Engineering Laboratory, EG&G Idaho, Inc.
06/17/91

Figure 2.4 Combustion Engineering reactor engineered safety feature actuations by calendar year

Engineered Safety Feature Actuations General Electric

Isaiah National Engineering Laboratory, EC&G Isaiah, Inc.
05/17/91

Figure 2.5 General Electric reactor engineered safety feature actuations by calendar year

Engineered Safety Feature Actuations Westinghouse

Idaho National Engineering Laboratory, EG&G Idaho, Inc.
08/17/91

Figure 2.6 Westinghouse reactor engineered safety feature actuations by calendar year

Appendix A-1 provides the list of the significant events for 1990. Table A-1.4 of Appendix A-1 provides plant-specific data for this PI for 1989 and 1990.

Safety System Failures

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

Figure 2.1 shows that the industry average number of SSFs per plant per year increased slightly.

The industry design-basis reconstitution programs have identified problems which resulted in reports of SSFs. For those plants included in the industry average, the number of SSFs identified through review programs of all kinds (design, procedures, and technical specifications reviews) increased by 44 percent between 1987 and 1989, then remained constant in 1990. The industry review programs contributed about 60 percent of the total number of SSFs reported during those years.

Table A-1.5 of Appendix A-1 provides the quarterly plant-specific data for SSFs for 1989 and 1990. The annual number of safety system failures for each plant from 1987 through 1990 are listed in Table A-1.6 of Appendix A-1. The number of SSF events varies greatly from year to year at most plants. The number of SSF events per system per year is listed in Table A-1.7 of Appendix A-1. Two factors

significantly affected these data. First, at all multiunit sites, if a failure affected more than one unit, a separate event was counted for each affected unit. Second, if the failure of one system affected the ability of other systems to perform their safety functions, only the failure of the "root system" was classified as an SSF event.

As in previous years, three system groups accounted for almost half of all SSF events in 1990: the containment and containment isolation group, the emergency ac/dc power systems group, and the emergency core cooling systems (ECCS) group. The event counts for most groups and systems showed normal statistical variations over the last 4 years. One significant exception was noted in the 1990 counts for the residual heat removal (RHR) systems group. Only 4 SSF events in this group were reported in 1990, down from an average of 22 events per year for the previous 3 years, and down from the previous low for this group of 16 events in 1988.

Figure 2.7 shows the number of SSF events per plant for each nuclear steam supply system (NSSS) vendor. Reflecting the small change in overall industry trend in recent years, these figures do not show any definite trends.

Forced Outage Rate

The forced outage rate indicator 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.

Safety System Failures

Idaho National Engineering Laboratory, EG&G Idaho, Inc.
06/17/91

Figure 2.7 Safety system failures by calendar year

Figure 2.1 shows that the trend in the average forced outage rate by year declined from 1986 to 1988. The forced outage rate in 1989 appears to be unusually high when compared to the 1988 and 1990 rates. This relative peak resulted in large part from elevated forced outage rates at nine plants. Plant-specific data for 1989 and 1990 are provided in Table A-1.8 of Appendix A-1.

A sub-category of forced outages are those that result from plant conditions that cannot be corrected within the time limits provided in Technical Specifications and thus necessitate shutting down the plant. About 19 percent of all forced shutdown hours for 1990 were in this category. The number of these TS-required shutdowns and the amount of time spent shut down are given in Table A-2.14 of Appendix A-2 for each year since 1986 and for each plant. There were 55 shutdowns for a total of 11,319 hours in 1990 to meet TS requirements. Historically, a few plants contribute most of the shutdown hours each year, although the specific plants are different from year to year. Two plants were shut down for more than 1000 hours each in 1990, as compared to six in 1989, four in 1988, and four in 1987.

Equipment-Forced Outages per One Thousand Commercial Critical Hours

The equipment-forced outage (EFO) indicator is a measure of the number of forced outages caused by equipment failures per 1000 hours of commercial operation while the reactor is critical. The EFO rate is the inverse of the mean time between forced outages caused by equipment failures. AEOD monitors the EFO rate as an indicator that can provide a perspective on the effects of equipment problems on overall plant performance.

Figure 2.1 shows that the industry average EFO rate declined from 1986 to 1988 and has remained relatively constant since then. Table A-1.9 of Appendix A-1 contains the data for this PI for 1989 and 1990 for each plant.

Collective Radiation Exposure

With two site-specific exceptions, the collective radiation exposure (man-rem) per plant that AEOD monitors is the total radiation dose reported at a site for a certain period divided by the total number of units at the site. Some items that reduce collective radiation exposure, thereby contributing to a low 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 fuel clad integrity, good radiation exposure control programs, and an alert health physics staff.

Figure 2.1 shows that collective radiation exposure remained constant from 1989 to 1990, after slowly decreasing from 1985 through 1989. Table A-1.10 of Appendix A-1 contains quarterly radiation exposure data for 1989 and 1990 as provided by the Institute of Nuclear Power Operations (INPO) for each plant. Since INPO routinely receives radiation exposure data from the plants on a quarterly basis, to avoid unnecessary duplication of effort, AEOD uses the data that INPO supplies for the PI program.

Cause Codes

The cause codes indicator is intended to identify possible programmatic deficiencies. Cause code trends are developed from data in the Sequence Coding and Search System (SCSS) database. This database includes the causes associated with each event that a licensee reports as required by the regulations in Title 10, Part 50, Section 73 of the Code of Federal Regulations (10 CFR 50.73).

Industry averages for this indicator are not calculated. The quarterly cause code data used in the PI Program for each plant for 1989 and 1990 are shown in Tables A-1.11 through A-1.16. The cause code indicator captures the trends for administrative control problems (Table A-1.11); licensed operator errors (Table A-1.12); other personnel errors (Table

A-1.13); maintenance problems (Table A-1.14), design, construction, fabrication or installation problems (Table A-1.15); and equipment (electronic piece-part or environmentally related) failures (Table A-1.16).

Other Operational Data

Within the context of its safety mission, the NRC is not primarily concerned with the power production performance of the nuclear industry. However, because it also requires attention to detail and good management, sustained reliable power production can be a positive sign for safety performance. This section and the related appendices provide highlights of 1990 power production statistics.

The commercial nuclear power industry produced a gross electrical output of approximately 605 billion kilowatt hours (kWh) in 1990, which was an 8.8 percent increase over the 555 billion kWh produced in 1989. Unit availability factors ranged from 0 to 99.6 percent. (The unit availability factor includes both the number of hours the unit operated with the generator output breaker closed to the station bus and the number of hours the unit was off line for some other reason when the plant was capable of generating electricity.) The 1990 industry mean unit availability factor was 71.7 percent, which was a 3.4 percent increase over the 1989 figure of 68.3 percent. Unit availability, service, maximum dependable capacity, and design electrical rating capacity factors for 1990 are given in Table A-2.15 of Appendix A-2. Table A-2.16 lists the reactor critical hours by year.

3 Abnormal Occurrence Reporting

The Office for 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) 1990. These reports were for the third and fourth quarters of CY 1989 (designated NUREG-0090, Vol. 12, No. 3, and Vol. 12, No. 4, respectively) and the first and second quarters of CY 1990 (NUREG-0090, Vol. 13, No. 1, and Vol. 13, No. 2, respectively). The third and fourth quarter reports for CY 1990 (designated NUREG-0090, Vol. 13, No. 3, and Vol. 13, No. 4, respectively) were published during the first part of CY 1991.

During CY 1990, one AO occurred at a nuclear power plant. Table 3.1 shows the number of AOs occurring at nuclear power plants for each calendar year since 1986. The table shows a sharp decrease in the number of AOs beginning in 1987 to a level that has remained low for 1988 through 1990.

A summary of the AO that occurred at a nuclear power plant (Vogtle Unit 1) during 1990 is provided in Appendix B to this report. The event is also discussed in more detail in Section 7, "Incident Investigation Program," of this report. A summary of 1990 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. 5, No. 2, *Nonreactors*).

Table 3.1
U.S. nuclear power plants
Abnormal occurrences (AOs) per year

Calendar Year	Number of AOs
1986	11
1987	3
1988	3
1989	4
1990	1

4 Data From the Nuclear Regulatory Commission's Operations Center for 1990

The NRC's Operations Center in Bethesda, Maryland, provides a focal point for NRC communications with Commission licensees, State agencies, and other Federal agencies about operating events in the commercial nuclear sector. The Operations Center is staffed 24 hours a day by NRC Headquarters Operations Officers (HOOs), who are reactor systems specialists trained to receive, evaluate, and respond to events reported to the Operations Center. The following photograph shows participants in a typical exercise as they evaluate plant status and licensee actions to determine what NRC response is required and what guidance to offer State and local agencies.

The Operations Center received 2707 notifications in 1990 from licensees, primarily from nuclear power plant licensees, about events that must be reported under NRC's prompt reporting requirements. A small subset of these notifications involved events that licensees 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.

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

Table 4.1 shows the number of each type of event classified under licensee emergency plans for the past 6 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. An event classified as a "Site Area Emergency" or a "General Emergency" indicates a major failure of one or more systems required for public safety or an event with the potential for a major offsite radiological release. Exercises are held periodically to ensure that NRC's and the licensee's response organizations are proficient in dealing with each type of emergency. In 1990, NRC Headquarters and regional offices participated in a number of emergency planning exercises, drills and table tops including, the FERMI 2 Power Plant (February 14, 1990), Du Pont Medical Facility (August 8, 1990), Combustion Engineering Fuel Facility (September 6, 1990), the River Bend Station 2-day table top exercise with States and Parish participation (August 28 and September 18, 1990), and San Onofre 2 Power Plant (October 17, 1990). In addition, planning commenced for the next Federal Field Exercise (FFE-3), a multi-day demonstration of the full federal response to a radiological emergency at a nuclear power plant. In early 1991, agreement was reached with the Commonwealth of Pennsylvania and Pennsylvania Power and Light Company to conduct the FFE-3 at the Susquehanna facility the week of February 24, 1993.

With regard to the licensees' event classification system, AEOD representatives participated in the development of the International Nuclear Events Scale (INES) that the International Atomic Energy Agency (IAEA) made available in late 1990 to IAEA member countries for use in a trial period of about one year. The INES will be used by participating countries for prompt notification of the public and the media of the relative significance of events that occur at their power reactor facilities. The scale has seven numerical levels with a level 7 representing a major accident with a major release of radiological activity offsite down to a level one representing an anomaly with no offsite or onsite radiological consequences. The NRC decided not to participate in the trial period because of the existing event

classification and reporting system for U.S. reactor licensees that includes provisions for notifying offsite officials and notification of the media.

Table 4.1 Classification of events under licensee emergency plans—1986-1990

Event	1986	1987	1988	1989	1990
Unusual Event	209	231	212	197	151
Alert	9	9	6	13	10
Site Area Emergency	0	0	0	0	1
General Emergency	0	0	0	0	0

Figure 4.1 shows the average number of reports of events that the Operations Center received in 1990 at particular times during a 24-hour period. The data in the figure 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 peak in the afternoon and early evening 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 licensees are allowed up to 4 hours to report a non-emergency to the NRC Operations Center. 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 Regions I through V is 30, 33, 28, 10, and 10, respectively.

Events called into the NRC Operations Center by licensees of power reactors and certain fuel facilities are done so on a dedicated phone line, the Emergency Notification System (ENS). Since this single line limits the amount and timeliness of information that can be provided to the NRC, an Emergency Response Data System (ERDS) has been designed that will allow the simultaneous transmission of up to 60 selected parameters from the power reactors in the areas of primary systems, containment systems, meteorological conditions and effluent rates. The fuel facilities will be excluded from this program. The installation of ERDS is presently on a voluntary basis with 10 licensees, comprising 30 reactor units, having volunteered as of the end of 1990. However, a proposed rule was issued in October 1990 that will require the installation of ERDS within 18 months following the effective date of a final rule. That final rule is expected to be issued in July 1991.

With regard to the ENS, it has been recognized that eventually an upgrade to the agency's telecommunications system will be necessary to replace the aging ENS equipment and to provide additional communication paths other than the public switched network which is vulnerable to blockage during an emergency. Therefore, the NRC has initiated an effort to replace the ENS with multiple pathway routing. Options being considered are the Government - wide FTS 2000 terrestrial network as the primary system and a satellite system to provide a backup or diverse routing if a secondary system is determined to be necessary.

1990 Operations Center summary
Average number of events by time of day

Figure 4.1

Table 4.2 provides information on the "Site Area Emergency" event at a power facility that was reported in 1990 and the "Alert" events at power facilities and a fuel facility that were reported in 1990.

Actions taken by the NRC HOO in response to notifications of events ranged from a computer or log entry, followed by appropriate notifications, to establishing emergency conference calls among the HOO, the licensee, and the senior NRC regional and headquarters staff members. For very significant events, these conference calls would result in activation of the agency's Incident Response Plan. In 1990, the NRC Emergency Operations Center was placed in a Standby Response Mode during the Vogtle 1 "Site Area Emergency" event in accordance with the NRC Incident Response Plan (NUREG-0728, Revision 1).

**Table 4.2 "Site Area Emergency" and "Alert" events reported in 1990
at power facilities and a fuel facility**

Name of Facility	Event No.	Date	Description	Duration
<u>Site Area Emergency</u>				
Vogtle 1	18024	03/20/90	Loss of all vital ac power while in refueling mode	35 minutes
<u>Alerts</u>				
Power Reactor Facility				
Susquehanna 1	17688	02/03/90	Loss of shutdown cooling capability while in cold shutdown cooling	6 hours 32 minutes
Perry 1	18126	04/03/90	Loss of Emergency Service Water System at 100% power	3 hours 29 minutes
Cook 1	18562	05/24/90	Fire in 345 kV electrical switchyard circuit breaker at 70% power	5 hours 22 minutes
Zion 2	18849	07/07/90	Loss of control room annunciators while in cold shutdown mode	15 minutes
Zion 2	18851	07/07/90	Loss of control room annunciators while in cold shutdown mode	5 minutes
Beaver Valley 2	19243	08/30/90	Release of carbon dioxide from fire protection system at 52% power	2 hours 1 minute
Robinson 2	19333	09/11/90	Release of toxic freon gas into plant vital area while in refueling mode	35 minutes

Table 4.2 (cont.)

Name of Facility	Event No.	Date	Description	Duration
Sequoyah 2	19337	09/11/90	Loss of residual heat removal system while in cold shutdown mode	5 minutes
Waterford 3	20161	12/27/90	Chlorine gas release from nearby facility while at 100% power	48 minutes
Fuel Facility				
Babcock & Wilcox Co.	17543	01/12/90	Fire in plant laundry facility	32 minutes

5 Operating Experience Data Programs

5.1 Introduction

AEOD provides an independent capability to analyze and disseminate operational data at nuclear power plants. It reviews and evaluates operational data to identify (1) significant events and any associated safety concerns and root causes, (2) the trends and patterns displayed by these events, (3) the adequacy of the corrective actions taken to address these concerns, and (4) the generic applicability of events and concerns to other plants. AEOD also coordinates, directs and continues to develop the NRC Performance Indicator Program. In these activities, the AEOD staff uses 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 to the Nuclear Plant Reliability Data System (NPRDS), which is managed by the Institute of Nuclear Power Operations (INPO). In addition, AEOD uses plant operating data found in the licensee's 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 operational experience.

5.2 Performance Indicator Development

During 1990, the AEOD staff continued its efforts to enhance the Performance Indicator Program through (1) development of suitable peer groups for comparing the performance of individual plants to the average performance of a group of similar plants; (2) development of the methodology to account for the cyclic nature of cause code data during the operating cycle; (3) development of a (RES); and (4) participation in the International Atomic Energy Agency (IAEA) program for the development of performance indicators.

5.2.1 Peer Groups

In the staff requirements memorandum of August 10, 1989, that approved cause code trends as a new performance indicator, the Commission, prior to approving the display of individual plant cause code deviations from the industry average, directed the staff to address the comparability of LER data among licensees. In response to this concern, AEOD initiated a study to determine appropriate peer groups for comparing reported event data among licensees. In this study, AEOD considered many design and regulatory issues, then constructed preliminary peer groups and discussed them with NRC staff and industry groups. These talks led to revision of the preliminary peer groups into a set of 9 proposed groups which will be evaluated, along with the operating cycle methodology described below, in a test program during 1991.

5.2.2 Operating Cycles

Early in the peer group development discussed above, AEOD staff recognized that cause code data were cyclic with a period approximating the refueling interval. Further investigation of this phenomenon led to the conclusion that a plant's operating phase (startup, power operations, refueling, etc.) could have as much or more effect upon event reporting as a plant's peer group. AEOD, therefore, took two steps to address this effect. First, in the current PI report methodology, the plant and industry four-quarter moving averages displayed on the bar charts and used to calculate trends and deviations were changed to six-quarter moving averages to approximate the average length of a fuel cycle. Second, a study was begun to identify those phases of operation in which event reporting varies significantly and to develop the methodology to present plant trends and deviations as a function of operating phase. Upon completion of this study, the methodology will be evaluated,

along with the peer groups, in a test program during 1991.

5.2.3 Development of Safety-System Unavailability Indicators

In 1990, the Office of Nuclear Regulatory Research (RES) continued to evaluate an indicator of the unavailability of selected, risk-significant safety systems. The candidate indicator is the product of the fractions of time during plant operation that each train of the system would not have functioned on demand. A persistently high value of this indicator could indicate that plant programs have not prevented the safety-system availability from degrading.

In evaluation of the indicator with historical data from a small sample of plants, the indicator provides a reasonable indication of the lower bound of unavailability experienced by selected safety systems.*

5.2.4 International Atomic Energy Agency Performance Indicator Development Program

Since 1985, the International Atomic Energy Agency 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 of the IAEA Technical Committee Meeting held in September of

1989, the IAEA convened an Advisory Group meeting in March of 1990 to refine the guidance on and begin the development and validation of specific safety indicators. AEOD staff participated in this meeting. The Group described the needs for safety indicators at the senior plant management level, at the plant department level, and at the senior management level of regulatory bodies. The Group also defined the characteristics of ideal safety indicators; presented methods to develop such indicators, both those currently available and those requiring further development; and described techniques for validation of safety indicators. The Group recommended that IAEA convene meetings of experts in each of several areas to define specific indicators for plant use and to prepare validation programs for those indicators.

In November of 1990, the IAEA convened a Technical Committee Meeting (TCM) on "The Exchange of Experience in Managing Nuclear Power Plant Safety Performance Using Numerical Indicators." The purpose of the meeting was to discuss nuclear power plant management practices to improve plant performance using numerical indicators. AEOD sent a representative to this TCM. The meeting provided a useful exchange of information concerning current and future activities in the development of performance indicators. The set of indicators recently approved by the World Association of Nuclear Operators (WANO) has been accepted by all the operating organizations represented at the meeting, although there are some implementation problems in specific cases. There is also much work being done by nuclear power plant operators to develop living probabilistic safety assessment (PSA) techniques for evaluating in real-time the safety state of operating plants. None of the regulatory bodies represented at the TCM use the entire set of WANO indicators. The meeting highlighted the fact that the performance indicator needs of the regulatory bodies differ from those of the operators.

* System Unavailability Indicators Summary Report—NUREG/CR-5626, Volume 1 (Draft), March 1991

5.3 U.S. Operational Databases

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 1990, 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 cause code performance indicator. The SCSS database is also 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 1990. 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.

The Nuclear Plant Reliability Data System (NPRDS) is a data base of more than 560,000 component engineering records and 110,000 component failure records from commercial nuclear power plants.

Industry provides the data to the Institute of Nuclear Power Operations (INPO) which manages and directs the development of the data base. NPRDS provides the NRC's primary source of industry-wide component failure data. Since 1982, AEOD has conducted a program to evaluate industry participation in the system, INPO's management of the system, and the ability of the system to meet NRC's needs for component reliability data. AEOD reports its findings from these evaluations to the Commission.

5.3.1 Licensee Event Reporting

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.

It is important to note that safety performance is only one of several factors that cause the number of LERs submitted by individual plants to vary. Therefore, the assessment of the safety performance of a plant is not based on the number of LERs it has submitted. Rather, judgements about safety performance are based on an evaluation of the number and significance of operational events. The purpose of this section is to provide an overview of the reporting process as it relates to a major source of operational data.

The mean number of LERs per plant per year has consistently declined since 1986, as shown in Table 5.3.1. The number of LERs submitted by individual licensees in 1990 ranged from a low of 1 to a high of 47; the mean value for the industry was 19.

Approximately 92 percent of all LERs submitted in 1990 were in accordance with the reporting requirements of 10 CFR 50.73; the remaining 8 percent were submitted in accordance with other requirements, such as 10 CFR 50.36, or as voluntary reports.

Table 5.3.2 shows the percentage breakdown of LERs submitted under each section of 10 CFR 50.73.

Table 5.3.1 LERs submitted by year*

Year	No. of LERs	No. of Units	LERs per Unit
1986	2915	106	28
1987	2912	111	26
1988	2479	110	23
1989	2354	112	21
1990	2124	111	19

* Counts for 1986 through 1990 do not include the Dresden Unit 1, Humboldt Bay Unit 3, and Three Mile Island Unit 2 plants. Counts also do not include the Fort St. Vrain plant after 08/29/89, the LaCrosse plant after 04/30/87, the Rancho Seco Plant after 06/07/89, and the Shoreham plant after 06/06/87. Canceled, proprietary, and safeguard LERs were excluded from all counts.

Table 5.3.2 Percentage of LERs submitted in 1990 according to 10 CFR 50.73 requirements

CFR Title 10 Section	Requirement	Percent of Reports* (Approximate)
50.73(a)(2)(i)	TS shutdown or TS violation	43
50.73(a)(2)(iv)	ESF actuation (including reactor trips)	39
50.73(a)(2)(v)	Real or potential loss of safety system	9
50.73(a)(2)(ii)	Unanalyzed conditions	7
50.73(a)(2)(vii)	Failures in multiple systems	2
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)(A)	Airborne radioactive releases	0
50.73(a)(2)(viii)(B)	Liquid effluent radioactive releases	0

* This column reflects the licensees' categorization of LERs.

5.3.2 Workshops on Reporting of Events

In late 1989 and early 1990, the NRC staff conducted a survey of the effects of regulatory programs on licensees. Subsequently the NRC published draft NUREG-1395, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities," in March 1990 and requested public comments. A review of the survey and the comments received made apparent to the NRC staff the need both to clarify the threshold for event reporting and to review its backfitting policy.

To communicate NRC guidance and gather additional information, the NRC staff conducted four regional workshops between September and November 1990 on backfitting and event reporting. The NRC staff also interacted with the industry during these workshops to learn about their experience implementing the two rules for reporting events since they were issued in 1984 and to identify potential changes that may be needed for these rules and or existing guidance.

The workshops on the backfitting policy are discussed in Section 10, "Committee to Review Generic Requirements," of this report.

Sections 50.72 and 50.73, the two rules about reporting events, identify the types of reportable events and conditions that are considered to be significant or meaningful to the NRC in order to identify plant emergency conditions, plant specific issues and facilitate identification of generic safety issues. The Section 50.72 rule requires prompt telephone notification to the NRC for emergency response purposes as well as early notification of non-emergency events that the NRC considers significant or needs to understand to respond to public inquiries. The Section 50.73 rule requires written reports to provide information for more in-depth engineering studies of potential safety concerns. The existing NRC guidance primarily

consists of NUREG-1022, "Licensee Event Report System," and its two supplements.

The workshops were well attended by licensees and industry representatives (e.g., Nuclear Management and Resources Council [NUMARC], vendors, consultants). In the workshops, the NRC staff from AEOD, NRR, NMSS, OGC and the respective regional offices communicated the need to ensure that all safety significant events are reported, and the need for consistency in reporting, and also described how the information from reported events is used. The importance of feedback of lessons learned from operating experience, especially about generic issues, in order to prevent recurrences of events or more significant events was stressed. Industry provided feedback on the need to clarify the criteria and the threshold for several types of reportable events, and on the requirements for their consistency and completeness.

As a followup to the workshops, a task group with representatives from AEOD, NRR, OGC, and ADM began preparing a revision to NUREG-1022 to consolidate the guidance in that document and its Supplements 1 and 2 and to provide guidance to clarify the existing requirements to improve the consistency and completeness of reports required by both 10 CFR 50.72 and 10 CFR 50.73.

5.3.3 Evaluation of the Nuclear Plant Reliability Data System

In June 1990, AEOD issued its report "Staff Evaluation of the Status of the Nuclear Reliability Data System in 1989," which covered primarily the calendar year 1989. The three major conclusions of the evaluation were that (1) industry participation in the system in 1989 remained about the same as in 1988, (2) INPO met several major milestones in making the NPRDS data more useful and the software easier to use in 1989 and (3) NPRDS continues to provide useful information for some NRC programs. However, the NPRDS has

limitations that hamper its ability to meet staff needs for component unavailability and reliability data. In addition, the inherent limitations of consistency, completeness and timeliness and the potential for reduction in voluntary participation by utilities, caused the data source to be inadequate to support a component failure based maintenance performance indicators.

In its evaluations, AEOD has traditionally measured industry participation in NPRDS by (1) the completeness of failure reporting, that is, the percentage of industry-wide reportable failures reported to the database; (2) the timeliness of reporting failures, that is, the average time between a failure's discovery and the availability of a report of the failure in the database; and (3) the quality of the data. The quality of the data is based on two aspects: the validity of the engineering data and the adequacy of the failure narratives. Do the failure narratives provide users with enough information to determine the cause of the failure and the corrective actions taken? The 1989 evaluation revealed that (1) average completeness remained about the same (65 percent to 70 percent of the NPRDS reportable failures are being reported), (2) the median time for a failure report to be available on the database after discovery of the failure remained between 3 and 4 months, and (3) the general quality of the data did not change noticeably.

The report stated that, during 1989, INPO completed several programs that will make the system more user-friendly and useful, and should improve the quality of the data. The release of NPRDS Plus in April 1989 marked the final step in providing (1) users a menu-driven, PC-retrieval system that provides new standardized information reports; (2) the ability to perform statistical analysis of the data; and (3) a method for easier entry and revision of the data in the system. INPO completed the addition of failure-mode codes to all failure reports and standardized manufacturer model identification numbers for about 285,000 records. INPO also approved the expansion of the scope of reportable components to new balance-of-plant

components. These new data elements will increase the usefulness of the data.

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

Also after 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.

5.4.1 Review of Foreign Events

The NRC received and reviewed 96 IRS reports from foreign reactors during 1990. AEOD routinely reviews these reports submitted to the IRS from

nuclear power reactors by both member countries of the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) in addition to information obtained through bilateral exchange programs with over 20 countries. Examples of issues of importance to domestic reactors that were highlighted by these information sources in 1990 are (1) bowed and damaged control rod guide tubes, (2) feedwater pipe ruptures caused by erosion and corrosion (pipe-wall thinning), (3) reactor vessel cold overpressure events, and (4) plant startup conducted while the scram accumulators were discharged. AEOD also developed important findings that were partly based on information about foreign events in several of its studies. AEOD disseminates, within the NRC, reports of selected foreign reactor events of particular interest to the staff regulating the U.S. program.

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

At the end of 1990, 112 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 Incident Reporting System (IRS). The United States and 12 other countries (Belgium, Canada, Finland, France, Germany, Italy, Japan, the Netherlands, Spain, Sweden, Switzerland, and the United Kingdom) are members of NEA. The NEA member countries also participate 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. operational experience feedback program. Such reports were based primarily on information earlier communicated to U.S. plant operators by NRC information notices, bulletins, and generic letters, AEOD reports on such subjects as debris blockage

of containment sumps, loss of vital AC power at Vogtle Unit 1, motor-operated valve testing, potential inter-systems loss-of-coolant accident (LOCA), and failures of high-pressure safety injection pumps.

A number of 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 various U.S. experiences with broad generic implications that resulted in actions being taken in their country's nuclear program.

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 of 1990, PWG-1 met jointly (annual meeting) with representatives from IAEA countries to discuss significant operating events. It was agreed that beginning in 1991, each NEA member country will present the high-priority safety concerns identified or studied over the past year. PWG-1 also met with representatives from various NEA member countries to discuss significant operating events and data. Discussions highlighted the NEA/IAEA relationship, emphasizing that joint efforts between NEA and IAEA should be tightly coordinated, particularly, in the area of event reporting and data management, data analysis and studies, and meetings. The United States cited several 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.

5.4.3 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. In 1990, this activity continued under a contract with ORNL, and includes the processing and encoding of all IRS events received from the NEA, and the quarterly transmittal of multiple copies of a computerized update of the database to NEA for further distribution to the member countries.

6 Results of AEOD Studies

6.1 Evaluations

In 1990, AEOD published one case study and 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 or industries' planned or scheduled corrective actions are adequate.

6.1.1 Operating Experience Feedback Report—Solenoid-Operated Valve Problems at U.S. Light Water Reactors

(Case Study AEOD/C90-01 - Published as NUREG-1275, Volume 6)

The AEOD staff analyzed recent U.S. light-water reactor experience with solenoid-operated valves (SOVs). It focused on the vulnerability of safety-related equipment to common-mode failures or degradations of SOVs. It presented information on many representative events in which common-mode failures or degradations affected, or had the potential to affect, multiple safety systems or multiple trains of individual safety systems. While plant safety analyses may not have addressed such common-mode failures or degradations, operating experience indicates they are continuing to occur.

Common-mode SOV failures and degradations have cut across multiple trains of safety systems as well as multiple safety systems. Common-mode SOV failures have compromised front-line safety systems and important support systems such as emergency alternating current (ac) power, auxiliary feedwater, high-pressure coolant injection, and scram systems, resulting in reductions in safety margins. Many of the common-mode SOV failures and degradations observed were beyond the conditions analyzed in plant Final Safety Analysis Report (FSAR) and are

not modeled in present-day probabilistic risk assessments (PRAs).

The AEOD staff sought the root causes of the observed failures and degradations of SOVs and examined the widespread deficiencies that were found in design and application, manufacture, maintenance, surveillance testing, and feedback of failure data.

Some of the more significant common-mode SOV events discussed in the report are as follows:

- Simultaneous common-mode SOV failures that resulted in the failure of both emergency diesel generators to start at the Perry plant.
- Simultaneous common-mode failures within the scram system at the Susquehanna plant.
- Common-mode scram pilot solenoid valve failures that resulted in primary system leakage outside primary containment at the Dresden plant.
- Simultaneous common-mode failures of two SOVs and the potential failures of 58 additional SOVs in multiple systems at the Kewaunee plant.
- Simultaneous common-mode failures of main steam isolation valves (MSIVs) to close upon demand at Perry and Brunswick.
- Simultaneous common-mode failures of safety relief valves in the automatic depressurization system at the Brunswick plant.

The AEOD staff concluded that correcting the root causes of the SOV problems would result in reducing the likelihood for common-mode SOV failures. Consequently, the staff recommended that for safety-related applications, licensees (1) verify the compatibility of SOV design and plant operating conditions, (2) verify the adequacy of plant

maintenance programs, (3) ensure SOVs are not subjected to fluid contamination, (4) review SOV surveillance testing practices, and (5) verify SOVs used in safety-related applications have been manufactured, procured, installed, and maintained commensurate with their safety functions. In addition, the staff recommended that an industry group take action to improve the mechanism for communicating SOV failure data to the manufacturers for early detection and resolution of potential generic problems.

The case study was forwarded to the NRC Office of Nuclear Reactor Regulation (NRR) for implementation in January 1991. NRR is now considering actions to be taken on the recommendations given in the case study. Major industry organizations (e.g., Electric Power Research Institute [EPRI], Institute of Nuclear Power Operations [INPO], Nuclear Management and Resources Council [NUMARC]) have been given copies of the case study for their consideration, especially in light of the last recommendation noted above for industry action. The case study was issued as NUREG-1275, Volume 6, in February 1991 and distributed to communicate the lessons of operating experience presented in the report.

6.1.2 Review of Thermal Stratification Operating Experience

(Special Study Report AEOD/S902)

The AEOD staff analyzed operating experience involving thermal stratification and the corrective actions taken by licensees. The study addressed representative operational events that occurred at domestic and foreign plants during the past 10 years.

Thermal stratification refers to a condition in which two streams of fluid at different temperatures flow in separate layers without appreciable mixing. Thermal

stratification can be classified as global thermal stratification, cyclic thermal stratification, and thermal striping. Global thermal stratification causes low-cycle bending stresses and is responsible for unexpected piping movement. Cyclic thermal stratification and thermal striping result in high-cycle stresses that could cause pipe cracks.

This study began as a result of operating experiences at the Farley and Trojan nuclear plants. On December 8, 1987, a pipe crack resulted in an unisolable leak in a high-pressure injection line at the Farley plant. The licensee installed instrumentation on the line and discovered large temperature differences between the top and bottom of the pipe. The licensee at the Trojan plant observed unexpected movement of the pressurizer surge line causing contact of some pipe-whip restraints and installed instrumentation during the 1988 fuel outage to measure the actual temperature distribution and pipe movement. Data indicated that thermal stratification occurred routinely in the surge line.

Additional thermal stratification events included damage to pipe supports at the Washington Nuclear Plant Unit 2 (WNP-2) in 1984 and cracks in pipe joints at several foreign plants.

The Farley Unit 2 event on December 9, 1987, prompted the NRC to issue Bulletin 88-08, "Thermal Stresses In Piping Connected to Reactor Coolant Systems." Supplements 1 and 2 to Bulletin 88-08 were issued subsequently to provide additional information and to alert licensees to an event at a French plant. Supplement 3 to Bulletin 88-08 was issued to discuss an event at a Japanese plant. As a result of the Trojan experience of unexpected movement of the pressurizer surge lines, the NRC issued Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," on December 20, 1988.

Although the AEOD review did not result in specific recommendations, an overview was provided of the

thermal stratification issues needing NRC and industry attention. The highlights are as follows:

- Thermal stratification has caused cracks and damaged supports and contributed to thermal fatigue in high-energy piping, affecting safety injection, residual heat removal, feedwater, and pressurizer surge line piping systems. Some cracks have resulted in unisolable leaks of the reactor coolant system. Thermal stratification has occurred in pressurized-water reactors and boiling-water reactors.
- Many interfacing systems can be subject to thermal stratification. Through-wall cracks generally occurred in the 90-degree elbow connecting the piping section containing stratified fluid and the downstream piping section containing well-mixed fluid. Apparently, this location provides the minimum margins against thermal fatigue.
- Searches of databases found over 2000 reports prepared during 1980 through 1988 about pipe cracks and damage to pipe supports, snubbers, and pipe-whip restraints. Failures were attributed to such causes as vibration-induced fatigue or water hammer. Although some of these failures probably resulted from the effects of thermal stratification, they were not identified as such. Thermal stratification is difficult to detect and usually only confirmed by measurements from thermocouples placed around a suspected section of pipe.
- Typically, programs to detect the presence of thermal stratification monitor only selected portions of lines that are considered to have a potential for thermal stratification.

Visual and dye penetrant examinations do not detect cracks caused by thermal stratification until they are through to the outer wall of the pipe because this form of thermal fatigue originates on the inner surface of the pipe with crack depth propagating to the outside. Fatigue cracks are difficult to detect using even the more

sophisticated ultrasonic testing and radiographic testing techniques.

- As plants age, locations susceptible to the effects of thermal stratification are subjected to an increasing number of thermal cycles and the likelihood of cracks increases accordingly. Because thermal stratification was not considered during the original design of most operating plants, it will be a factor in any consideration of life extension.
- Interfacing system leakage can cause thermal stratification in system piping, and leakage through check valves, block valves, and isolation valves has been common. Therefore, an effective preventative and corrective maintenance of check and motor-operated valves at system boundaries would be a good approach to minimize the potential for thermal stratification in many locations.
- The staff has accepted the leak-before-break (LBB) approach in specific instances in piping systems susceptible to thermal stratification, such as the pressurizer surge line. The fatigue and fracture mechanics analyses to justify a licensee's use of the LBB approach rely on assumptions regarding the thermal cycles, thermal gradients, material properties, and crack geometry. However, the validity of the analyses is best supported over the long term by continued monitoring of the effects of thermal stratification.
- Existing NRC guidance and industry codes and standards may need to be revised to reflect the lessons learned from operating experience, the results of ongoing NRC reviews of licensee responses to Bulletins 88-08, and 88-11, and the EPRI study of thermal stratification. Explicit acceptance criteria may be needed for analysis of systems subject to thermal stratification.

Plants that have identified thermal stratification as a concern have taken a variety of corrective actions

to deal with the thermal stratification problems, including: (1) increased monitoring of piping, (2) inservice inspection, (3) hardware modification, (4) operation procedural changes, and (5) changes to piping supports.

PWR and BWR owners groups began the SCATS program (striping, cycling, and thermal stratification) to investigate thermal stratification as a generic issue. The industry also requested EPRI to conduct a study of thermal stratification issues. Representatives from AEOD and EPRI met in June 1990 to discuss the preliminary results of EPRI's work, and EPRI made a presentation of its program to the NRC in December 1990.

6.1.3 Recurrence of Important Safety Issues Reported In LERs

(Special Report AEOD/S90-01)

The AEOD staff reviewed over 800 licensee event reports (LERs) having particularly important safety significance to identify issues previously covered in NRC generic communications. These LERs were selected from 9000 or so LERs submitted during 1986 through about the first quarter of 1989 that NRC staff had previously screened for significance. Over 1200 NRC generic communications have been published through 1988. Approximately 100 of the generic communications contained issues that matched 308 of the 800 selected LERs. Of the 308 LERs, 232 were submitted within 2 years of the generic communication they matched. (The licensees of only seven plants submitted 3 or more of the 76 LERs that matched generic communications that were more than 2 years old.)

Most of the matches were to information notices, which make up the bulk of the generic communications. Generic letters and bulletins had a much smaller share of all the matches, although bulletins had a disproportionately large share of the

older matched communications. Seismic design deficiencies and equipment qualification problems dominated the issues associated with the matched LERs. Several generic communications that matched specific LERs addressed each of these areas. These areas had been extensively addressed by the NRC and industry for several years before the study period.

The review indicated that licensees are making a good-faith effort to process generic communications responsibly. Inspection reports and the resident inspector's view on the licensees' feedback programs and NRC audits of the licensees' actions in response to generic letters and bulletins, as well as the small number of older matched LERs, do not indicate any serious programmatic-level deficiencies in the licensees' use of NRC generic communications. However, the 76 LERs that matched to generic communications more than 2 years old indicate the use of individual generic communications with important safety issues could be improved.

Several suggestions were made for improving generic communications and their use by the resident inspectors and for application to new plant designs.

6.1.4 Failures of Electrical Supply and Power Generation Equipment Which Disrupted Plant Function at Nuclear Power Plants

(Engineering Evaluation Report AEOD/E90-01)

This study addressed concerns about disruptions of normal station ac power supply. Faults caused by grid, switchyard, and major electrical equipment failures were tabulated and maintenance, surveillance, and procedural problems were identified as key contributors to these events. Failures in electrical supply or power generation equipment can affect safety equipment because

such failures often cause scrams or require shutdown of the unit, may cause operation of engineered safety features, may frequently cause interruptions of offsite power supply to station auxiliaries, may occasionally cause fires, and can complicate post-scram operations by rendering equipment unavailable or unreliable.

Interruptions of offsite power to safety buses generally cause transfer of the buses to the emergency diesel generators, while interruptions of offsite power to non-safety-related electrical buses cause scrams, loss of forced circulation, loss of preferred heat sink, and loss of service or instrument air.

The operating experiences discussed in this study, particularly maintenance, surveillance, and procedural problems, were made available to NRR for use in developing procedures for the upcoming electrical team inspections. Electrical equipment failures were addressed in engineering evaluation report AEOD/E905, "Electrical Bus Bar Failures," and in Information Notice 89-064, "Electrical Bus Bar Failures."

6.1.5 Crosby Low Pressure Relief Valves

(Engineering Evaluation Report AEOD/E90-02)

Premature opening of a relief valve on the residual heat removal system initiated an extended blowdown of the primary system at Braidwood nuclear plant while it was shut down. According to the licensee, the premature opening of the valve was apparently caused by foreign material in the stem region, which distorted the previous valve calibration. The delayed closing of the valve, which prolonged the event, was attributed to improper setting of the nozzle ring that surrounds the valve nozzle. If the low-pressure relief valve is not closed, uncontrolled blowdown of the primary system can result.

The manufacturer sets the nozzle ring and locks it in place before delivery. The licensee's procedure for subsequent maintenance activities that disturb the ring position required establishing a reference point by counting turns of the ring to meet some defined criteria. The ring could then be adjusted by moving the ring, using the reference point. The number of turns needed to establish a reference point was in the hundreds, and there were different reference points in different instructions for the same valve. The process was quite cumbersome and the maintenance personnel were generally clothed in anticontamination gear with gloves and face masks.

The AEOD staff concluded that the licensee's maintenance process was prone to errors and that a simpler mechanical reference point such as a scribe mark would be better. It was also pointed out that some plants have instrumentation to monitor when these low-pressure valves are open. An information notice was not issued because of a pre-existing industry report describing the Braidwood event.

6.1.6 Overpressurization of Auxiliary Feedwater Systems

(Engineering Evaluation Report AEOD/E90-03)

In January 1989, overpressurization of both auxiliary feedwater (AFW) pipe trains at the Rancho Seco plant had the potential to cause total loss of the AFW system. The overpressurization was a result of the AFW steam turbine overspeed caused by the simultaneous failures of the turbine governor and the overspeed trip mechanism (OTM).

Although the OTM was installed to provide protection for the AFW pump and turbine if the governor failed, both the governor and the OTM can fail simultaneously.

The overpressurization of the turbine-driven train caused by turbine overspeed could affect the other trains of the AFW system because the trains are cross-connected and the cross-connection isolation valves are open during normal system operation.

The AEOD staff concluded that maintenance programs should include appropriate preventive maintenance and testing programs for the governor and the OTM and the vendor's recommendations for maintenance and test. In addition, to limit overpressurization to a single train during testing, either a change in the piping arrangement or valve alignment should be considered to avoid a cross-connection between the AFW trains. As a result of the study, NRC issued Information Notice 90-45, "Overspeed of the Turbine-Driven Auxiliary Feedwater Pumps and Overpressurization of Associated Piping System," in July 1990.

6.1.7 Swelling and Cracking in Hafnium Control Rods

(Engineering Evaluation Report AEOD/E90-04)

Hafnium was introduced as a neutron absorber for use in control rods in 1980. It was meant to be used in addition to carbon tetraboride as a replacement for silver-indium-cadmium. Hafnium control rods were introduced as original equipment in several later pressurized-water reactors. While using eddy current testing, several licensees have noted swelling and cracking in these control rods resulting from the formation of hafnium hydrides. Of the 14 plants licensed to use hafnium control rods, six have replaced or will replace them, two never installed them, four have recently or may soon install them, and two are continuing to use them.

Westinghouse's analyses to assess the effects of swelling of control rods predicted that neither physical interference with insertion nor mechanical resistance, which would increase rod drop time, will

be of a large enough magnitude to invalidate safety analysis conclusions. Eddy current tests to date have upheld these analyses.

The AEOD staff concluded that plants licensed to use hafnium control rods should extend nondestructive examinations of these rods to determine the extent of cracking, swelling, and wear over the useable life of the hafnium rods. Since the use of these rods is limited to a few plants, this suggestion was forwarded to NRR project management. Recent feedback indicates that examinations are continuing.

6.1.8 Operational Experience on Bus Transfer

(Engineering Evaluation Report AEOD/E90-05)

Most electrical buses at a nuclear plant are provided with power fed from multiple power sources with provision for manual or automatic transfer from one source to another to provide maximum availability of power.

In 1988 and 1989, a PWR licensee issued two LERs indicating that, under certain postulated scenarios, (1) the existing 4160 V bus transfer practice could result in a common-mode failure of Class 1E loads of both trains and (2) repeated operation of the existing 4160 V bus transfer scheme could damage safety-related motors. These findings prompted an evaluation of bus transfer problems between 1985 and 1989.

The study indicated that bus transfers failed to take place on demand on at least 54 occasions. The failure of a bus to transfer results in partial or total loss of offsite power. A reliable bus transfer scheme is required to meet General Design Criterion (GDC) 17. In most plants if the bus transfer fails, the safety-related (Class 1E) loads must rely on emergency diesel generators for power. Such

failures also cause loss of power to reactor coolant pumps and, in turn, reliance on natural circulation cooldown. However, the study did not identify any equipment damage that could be directly attributed to bus transfers.

The report provided several suggestions for reducing the present rate of bus transfer failures, including the use of power distribution systems that require fewer bus transfers. About 82 percent of the U.S. nuclear plants use power distribution systems that require a bus transfer for every unit trip, which would be avoided by using a dedicated generator circuit breaker. This scheme is popular in Europe and has been used at eight U.S. nuclear plants. Another way to avoid bus transfer with every unit trip is to use power distribution systems in which the offsite power source is used both as the normal power source and alternate power source. This scheme is popular in France and has been used at six U.S. plants.

Equipment damage from bus transfers can be reduced by using supervised bus transfer schemes with static devices or by using faster bus transfer schemes such as parallel bus transfers or simultaneous bus transfers.

The AEOD staff presented the results of the study at the 1991 Winter Meeting of the Power Engineering Society of the Institute of Electrical and Electronic Engineers, and NRC is preparing an information notice.

6.1.9 Potential For Residual Heat Removal System Pump Damage

(Engineering Evaluation Report AEOD/E90-06)

On December 5, 1989, during a surveillance flow test at Sequoyah, both RHR pumps were in minimum flow, parallel operation when the

discharge check valve of the weaker pump was forced closed by the pressure of the stronger pump. The condition resulted in no flow through the weaker pump since its miniflow line was downstream of the discharge check valve. The loss of flow is referred to as "deadheading" and can result in pump damage.

Although Bulletin 88-04, "Potential Safety-Related Pump Loss," addressed the pump-to-pump interaction problem during miniflow operations, it did not address deadheading caused by loss of the miniflow path when the pump discharge check valve closed. The potential for the adverse pump-to-pump interaction, which can cause the discharge check valve to close, may not be detected during a surveillance test in which each pump is tested separately. In addition, the response to Bulletin 88-04 from the Sequoyah plant did not identify the deadheading problem.

The staff reviewed the residual heat removal (RHR) design of 19 randomly selected plants and identified 5 plants with system piping configuration similar to that of the RHR system at the Sequoyah plant. This suggested that the potential problem may not be recognized by other licensees with plants that have a similar RHR system configuration.

NRC issued Information Notice 90-61, "Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction," in September 1990.

6.1.10 Effect of Internal Flooding of Nuclear Power Plants on Safety-Equipment

(Engineering Evaluation Report AEOD/E90-07)

AEOD staff recognized certain operating events in which internal flooding had caused safety-related systems to be inoperable. Some of these systems are required to maintain core cooling. In addition;

one licensee's probabilistic risk assessment identified internal flooding to be a significant contributor to the risk of core damage.

The staff reviewed operational events in the past 10 years (1980-1989) that resulted in internal flooding in the secondary containment, auxiliary building, and turbine building of nuclear power plants. Some of these events caused a significant loss of safety-related equipment needed for decay heat removal or for plant shutdown and cooldown or resulted in precursors to potential severe core damage accidents.

The review indicated that internal flooding often resulted in common-mode failure of essential equipment for multiple safety-related systems. In some instances, the extent of damage could have challenged the plant's capability to achieve safe shutdown and cooldown. Other events resulted in unisolable flooding, resulting in the plants being flooded to the same water level as the source. Several events involving failure of a single component in a disassembled system during maintenance activities in cold shutdown or a refueling outage caused significant internal flooding that disabled other safety-related systems.

The AEOD staff concluded that

- The intent of Section 3.4.1, "Flood Protection," of the Standard Review Plan that requires that failures of fluid systems should not prevent safety-related equipment from performing their required functions has not been met in many instances.
- Deficiencies in flood protection design have caused unisolable flooding and common-mode flooding. Deficiencies in the design of flood barriers and floor drains often were the root cause of common-mode flooding. Some utilities have not taken corrective actions to prevent plant flooding resulting from floor drain backflow, despite IE Information Notice 83-44.

- Deficiencies in maintenance procedures and personnel errors have been the cause of significant internal flooding.

As a result of this study, NRC issued Supplement 1 to Information Notice 83-44, "Potential Damage to Redundant Safety Equipment As a Result of Backflow Through the Equipment and Floor Drain System" in August 1990.

6.1.11 Low Temperature Overpressure Protection: Testing PORVs With the Alternate Pneumatic Supply

(Engineering Evaluation Report AEOD/E90-08)

During a test of the low-temperature overpressure protection (LTOP) system at Point Beach, Unit 1, on May 3, 1989, the licensee declared the system to be inoperable because stroke time for the power-operated relief valve (PORV) was greater than assumed in the safety evaluation report for the LTOP system. The increase in stroking time was caused by inadequacies in the nitrogen gas supply used as a backup to the normal air supply to actuate the LTOP valves. There were four other plants (Turkey Point 3, San Onofre 1, Indian Point 2, and Catawba 1) that reported similar problems with backup pneumatic systems since 1987.

Inadequacies in the nitrogen supply were initially described in Information Notice 82-45, "PWR Low-Temperature Overpressure Protection," on November 19, 1982. Generally the problem is caused by excessive pressure drop or leakage in the backup gas supply lines, combined with a lack of surveillance on stroke time using the backup gas supply. Information Notice 89-32, "Surveillance Testing of Low-Temperature Overpressure-Protection Systems," on March 23, 1989, addressed surveillance requirements for LTOP systems, but did not specifically include testing the backup pneumatic supply system. As a result of this study,

testing of the backup pneumatic supply system for the LTOP system was then addressed by Supplement 1 to NRC Information Notice 89-32.

6.1.12 Additional Factors Affecting the Lift Setpoint of Pressurizer Safety Valves

(Engineering Evaluation Report AEOD/E90-09)

NRC information Notice 89-90, "Pressurizer Safety Valve Lift Setpoint Shift," was issued to inform licensees about potential problems resulting from operating pressurizer safety valves in environments different from those used to establish the lift setpoint of the valves. The IN primarily focused on problems encountered by setting the lift setpoint of safety valves with steam under the seat, and then operating the valves with a loop seal containing water.

Subsequent root cause analyses of unexpected shifts in the lift setpoint of pressurizer safety valves, conducted by seven licensees, identified additional factors contributing to the setpoint shifts. Some of these factors could cause a setpoint shift of as much as 150 psi. These factors include

- changes in temperature of the valve body and bonnet
- leakage across the valve seat
- excess loading of the safety valve tailpipe (nozzle loading)
- calibration of reactor coolant system (RCS) pressure control instrumentation
- presence of non-condensable gases in the pressurizer vapor space

The reduction in lift setpoints of pressurizer safety valves could result in primary system leakage and cause unnecessary challenges to plant safety systems. Conversely, an increase in lift setpoint causes a reduction in safety margin in the protection of the RCS against overpressure

transients. These additional factors affecting the lift setpoint of pressurizer safety valves are important and warrant further licensee attention.

A supplement to NRC Information Notice 89-90 was issued to alert licensees to the effect of temperature on the lift setpoint of safety valves.

6.1.13 Evaluation of Boiling Water Reactor Mode Switch Events

(Engineering Evaluation Report AEOD/E90-10)

AEOD staff reviewed 22 events involving reactor mode switches at boiling-water reactors from 1980 to 1989. Approximately 15 of the events involved the type SB-1 mode switch, while only 4 events involved the type SB-9 mode switch. Half of the events (11 of 22) were caused by an operator having difficulty in determining the actual position of the mode switch and 8 resulted from failure of some of the contacts to move to the proper position.

Most of these events occurred with the reactor at or near shutdown conditions. With the exception of two events, the safety significance of these events was generally low. However, the events represented unnecessary challenges to the reactor protection system and potential loss of the condenser as a heat sink.

Subsequent to issuance of the report, it was determined that all but six BWRs were going to change to non-SB1 type mode switches. Discussions with these licensees indicated that they were aware of the operating experience and were making decisions based on the information.

6.1.14 Preliminary Notifications and NPRDS Failure Reporting

(Technical Review Report AEOD/T90-01)

The AEOD staff performed a study to address the question of the Nuclear Plant Reliability Data System's (NPRDS) coverage of component failures that are described in Preliminary Notifications (PNs). In this study, the staff reviewed all of the PNs that were issued during the first quarter of 1989 (approximately 100) to identify component-level events that were potentially reportable to the NPRDS. The PN screening process was performed by engineers who were familiar with NPRDS conventions. For each case in which no corresponding NPRDS failure record could be found, the staff reviewed the licensee event report (LER) (if one had been issued) for additional detail to aid in determining whether the event contained any NPRDS-reportable component failures.

In all, the staff identified a total of 18 cases in which one or more NPRDS-reportable failures occurred. In 15 of these cases, they found matching failure reports in the NPRDS. The small sample examined in this study indicated that, after about 9 months following the failure discovery date, approximately 80 percent of NPRDS-reportable failures have been submitted to the system. Although this percentage was a little higher than past estimates (which had ranged between 65 percent and 75 percent), it was consistent with the earlier results.

PNs are designed to provide "early notice of events of possible safety or public-interest significance." Therefore, they cover a wide range of topics other than equipment failures, including drug testing, management changes, plant milestones reached (e.g., initial criticality), and nonreactor licensee events, such as lost radioactive sources and transportation incidents. For this reason, the relatively low yield of potentially reportable

component failure events—18 out of a total of 100—was not a surprising result.

6.1.15 Improper Installation of Heat-Shrinkable Tubing

(Technical Review Report AEOD/T90-03)

Improperly installed heat-shrinkable tubing of electrical cable splices and terminations can result in failure to meet environmental qualification requirements and can lead to common-cause degradation of redundant safety equipment.

On March 31, 1990, a PWR plant declared that one of the main feedwater suction valve electrical cable splices was incorrectly installed and failed to meet environmental qualification. The licensee's investigation concluded that the heat-shrinkable tubing was not of the correct size. The root cause was the absence of a procedure requiring the operator to measure the diameter of the wire and check the size of the sleeve.

This event prompted a review of problems with heat-shrinkable tubing. In 1986, following several reports of problems, NRC issued Information Notice (IN) 86-53, "Improper installation of Heat-Shrinkable Tubing." As a result of this IN, licensees conducted an investigation of this issue. Between 1986 and 1989, 23 LERs were submitted as a result of improper installation of heat-shrinkable tubing. The event data suggests that the discovery rates have come down significantly.

6.1.16 Reverse (Backward) Acting Valve Manual Handwheels

(Technical Review Report AEOD/T90-04)

Handwheels that must be rotated in a reverse direction to operate a valve by hand can be confusing to the operator and may complicate efforts to mitigate the effects of an ongoing transient. During an event at Palo Verde Unit 3 while the plant was at 98 percent power on March 3, 1989, unsuspected problems occurred because not all the valve handwheels turned in the same direction to be opened.

When all the turbine bypass valves and all four of the steam generator (S/G) atmospheric dump valves (ADV) failed to open after a grid fault, the auxiliary operators were dispatched to the main steam structure to manually open one ADV per S/G with the handwheels.

At Palo Verde the manual handwheels for the ADVs are configured facing each other. Although this makes the handwheels more accessible, it results in having identical valves that manually open and close in a reverse handwheel direction from each other. The event was further complicated by inadequate lighting. The auxiliary operators manually opened ADV on S/G No. 1. When they attempted to open an ADV on the other S/G, they were unable to get it more than 3 percent open when the handwheel came off because of a loose set screw. The auxiliary operators then tried to manually open the other ADV for that S/G by turning the manual handwheel clockwise, which was the direction required to open the other two ADVs. However, because of valve and handwheel orientation, the open direction for this ADV was counterclockwise. When they were unsuccessful in opening the ADV by hand, the auxiliary operators secured a 24-inch pipe wrench for mechanical advantage. Using the pipe wrench, they succeeded in rotating the handwheel one full turn in the wrong direction, which resulted in significant damage to the top of the ADV valve operator. It was subsequently discovered that 2 of the 12 types of air-operated valves at Palo Verde had unconventional handwheel operation.

Two somewhat similar events occurred at other plants in 1989. However, they were of minimal safety significance, and in our judgement, no generic action is warranted.

6.1.17 Effect of PUC Incentive Programs on Plant Utilization

(Technical Review Report AEOD/T90-05)

Incentive programs are being employed by a number of State public utility commissions (PUCs) to encourage improved plant utilization. The intent of these programs is to achieve better plant performance by having the utilities derive financial rewards for good performance and suffer penalties for poor performance.

Generally, these incentive plans focus on increased utilization as measured by capacity factor, availability, or heat rate. The plans vary widely among jurisdictions in terms of (1) the parameters measured, (2) the use of single or multiple measures of performance, (3) the objective or subjective evaluation criteria, and (4) the magnitude of the financial rewards or penalties. The mechanism for implementing financial rewards or penalties often involves adjustments in the utility's recovery of fuel replacement costs.

To determine what effect these incentive programs have had on plant utilization, the AEOD staff performed an exploratory examination of the association between incentive regulation by State public utility commissions and the utilization of nuclear power plants. This examination made no attempt to introduce statistical rigor or to assess causality. The results demonstrated little association between higher plant availability and capacity factor and incentive plans. For pressurized-water reactors (PWRs), the average 3-year plant availability and capacity factor appeared to be somewhat lower for plants under incentive plans than for plants that had

no incentive plans. No association between utilization and incentives was evident in the case of boiling-water reactors (BWRs).

6.1.18 Aquatic Life in Emergency Cooling Ponds

(Technical Review Report AEOD/T90-06)

In December 1989, an NRC diagnostic evaluation team became concerned about the presence of fish and other aquatic life in Arkansas Nuclear One (ANO) emergency cooling ponds. The fish population in the emergency cooling ponds had not been considered in the plant design basis, was not monitored, and was not controlled. During a postulated accident that would use the emergency cooling pond as the ultimate heat sink, the temperature of the pond could increase enough to kill fish and other aquatic life, possibly causing fouling of the pump intake screens and subsequent loss of emergency cooling capability.

The design and operations of other plants using similar cooling ponds were assessed. Ponds of interest were those with (1) no traveling screens at the intake structure (2) no monitoring or chemical treatment for control of aquatic life, and (3) small acre-feet of water in which a temperature increase could kill the resident fish population and clog stationary intake screens.

Of the 38 plant sites identified with reservoirs, cooling towers, or canals used for cooling plants, only 5 used small emergency cooling ponds with no traveling screens on the intake structure. Licensees at these 5 plants periodically monitored fish eradication or water chemistry control to preclude fish populations in the cooling ponds.

Additionally, Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related

Equipment," requested licensees to implement and maintain an ongoing program of surveillance and control techniques to significantly reduce the incidence of flow blockage problems as a result of biofouling. Licensee actions in response to this letter should further reduce the likelihood of biofouling.

6.1.19 Reversed Sensing Lines Connections

(Technical Review Report AEOD/T90-07)

On January 5, 1990, the Limerick Unit 2 hydrogen and oxygen analyzer was declared inoperable after a high concentration indication was received during nitrogen inerting of the containment. The licensee determined that connections of the analyzer sensing lines had been reversed because the analyzer ports were mislabeled.

The analyzer was installed by connecting the labeled inlet and outlet ports to the desired plant lines. Because of the labeling error, the normally closed sample point isolation valve was in the line exhausting to the containment while the sample line was capped, resulting in an indeterminate percentage of sample recirculation through the analyzer. Preoperational testing did not identify the improper configuration.

A review showed that 19 other events involving reversed sensing lines had occurred since 1984. The effects on plant operation were varied depending on the component and system involved, its intended function, and the amount of time before discovery. Inadequate design documents, deficient startup procedures, and weak verification methods were the primary causes of these events. All the events were of limited safety significance, were independent of reactor type (unlike those reviewed in a previous study in which BWRs were mostly involved), and the occurrence rate revealed no significant trends.

To reduce the number of events, licensees need to ensure adequate attention to installation details and modification and maintenance work performed on components. Additionally, proper quality control and post-modification or functional testing are needed. These aspects, although problematic, are within the scope of ongoing licensee Q/A, Q/C programs, and no additional generic action is warranted.

6.1.20 Turbine Bypass Malfunctions

(Technical Review Report AEOD/T90-08)

AEOD staff reviewed occurrences of turbine bypass malfunctions, inadvertent opening or closing of turbine bypass valves, and their resultant effect on plant startup, operation, and shutdown. Most of the 140 events over the 10 year period of the study relating to this issue led to a plant transient and 61 percent of the events ended in a reactor trip.

Turbine bypass valves are not a safety-related part of the main steam system. They are used to bypass a portion of the steam flow to the condenser, as a heat sink, during plant startup, shutdown, and some transients. Bypass system malfunctions lead to plant transient conditions involving feedwater and steam flow and reactor or steam generator water level and pressure.

The major cause of these events was not the malfunction or failure of the bypass valve itself (less than 15 percent) but failures of the steam bypass control system or other supporting components associated with these valves. These in turn led to valve malfunctions and challenge of a safety system in almost every case, although the risk of potential core damage was determined to be very small.

6.1.21 Inadvertent Partial Draining of Condensate Storage Tanks

(Technical Review Report AEOD/T90-09)

On February 9, 1988, 192,000 gallons of condensate storage tank (CST) inventory were lost at the Browns Ferry Nuclear Plant for Unit 1. Personnel in the control room were aware of the rapid level indication decrease (from 26.7 feet to 10.1 feet), but assumed the CST level instrumentation was malfunctioning, and CST inventory was actually lost. However, the licensee later determined that the CST level instrumentation was indicating correctly. A failed level sensor in another tank caused the continuous transfer of inventory from the CST. The condensate overflowed from the tank with the failed level sensor and eventually was discharged to the river.

AEOD staff searched NRC databases for other events of inadvertent CST drainage. Of an additional eight events, two had CST inventory losses greater than the Browns Ferry event: (1) Nine Mile Point Unit 2 (1987) lost 450,000 gallons from a large crack near the bottom of the CST and (2) Fermi Unit 2 (1986) lost 275,000 gallons from an overpressurization failure of piping below the CST. None of the events had adverse safety consequences. CST availability of analyzed inventory was guaranteed by either standpipes, tapping non-safety suction piping on the side of the CST at elevations that ensure minimum safeguard inventory, or level control valves that close at a preset level.

The event at Browns Ferry merely falls into a previously established pattern of inadvertent CST draindowns that did not affect plant safety.

6.1.22 Effect of High Energy Line Breaks on Chilled Water Systems at Nuclear Power Plants

(Technical Review Report AEOD/T90-11)

Gilbert/Commonwealth, the Architect Engineer for the Summer plant, identified that both chilled water trains could be rendered inoperable if a high-energy line break (HELB) occurred in the intermediate (auxiliary) building at Summer. A postulated HELB could provide additional heat loads on cooling coils in area air-handling units (AHUs) exposed to the steam environment. As the chilled water system supplies cooling water to these area AHUs, the heat from the HELB steam could place excessive heat loads on the chillers, rendering both trains of chilled water inoperable.

Vital equipment cooled by the chilled water system includes the charging pumps, component cooling water pump motors, and room-cooling AHUs for various vital components.

AEOD staff searched NRC databases for LERs from 1985 to late 1990 and identified no others relevant to this study. Crystal River 3 is the only other Gilbert/Commonwealth-designed plant with a problem similar to Summer's.

6.1.23 Loss of Offsite Power To Comply With NRC Regulations

(Technical Review Report AEOD/T90-12)

Brunswick Unit 2 manually shut down to comply with NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors," although the licensee knew the required scram would result in a loss-of-offsite power (LOOP) to the unit. Since a

loss-of-offsite power is not a desirable outcome, AEOD Investigated the circumstances of the event.

On June 17, 1989, Unit 2 was operating at 100 percent power when a malfunction in the startup auxiliary transformer (SAT) alarmed in the control room. Because of the possibility of inadvertent loss of the SAT while the cause of the alarm was being investigated, the licensee discussed the effect of an inoperable SAT on plant operations. The discussion centered around the NRC requirement to manually scram the reactor (per NRC Bulletin 88-07, Supplement 1) if power to the reactor recirculation pumps was lost, which would occur if the SAT became inoperable. The licensee decided that the most prudent action would be to decrease reactor power to less than 50 percent to ensure the reactor would not be operating in a region of core instability if recirculation flow was not available. Reactor power was decreased by reducing the reactor recirculation flow rate. The SAT was inadvertently tripped, causing the recirculation pumps to trip. At the time of the event, reactor power had been reduced to 76 percent, but plant power had not been decreased sufficiently to begin inserting the control rods; the unit was still operating with a 100 percent rod pattern (100 percent load line). In accordance with NRC Bulletin 88-07, Supplement 1, the reactor was manually scrammed to ensure the reactor would not operate with initiating conditions of potential core instability. Following the reactor scram and subsequent turbine trip, power to the unit from the unit auxiliary transformer (UAT) was lost, resulting in a LOOP to Unit 2. The diesel generators automatically started and powered the Unit 2 emergency buses.

NRC Bulletin No. 88-07, Supplement 1, was issued on December 30, 1988, and required BWR licensees without effective automatic scram protection to manually scram the reactor in response to every situation in which the recirculation pumps are tripped (or "no pumps operating") with the reactor in the RUN mode. This bulletin and its predecessor were issued in response to identified thermal hydraulic instability of BWR cores when the plant is

operating in certain unstable power and core flow regions as experienced at LaSalle Unit 2 on March 9, 1988. On March 2, 1989, the Brunswick licensee confirmed to the NRC that activities required to address the issues discussed in the bulletin were completed.

AEOD concluded that the plant entered a LOOP to comply with a regulatory requirement. However, the licensee's response to the problems with the SAT would not have been necessary if the recirculation pumps were powered from the UAT. In addition, if attempts to reduce power had been successful before loss of the SAT, a reactor scram would have been unnecessary.

6.1.24 Corrosion and Failure of Service Water Pump Impeller Snap Rings

(Technical Review Report AEOD/T90-13)

On November 8, 1988, the licensee at Arkansas Nuclear One, Unit 2, declared service water pump A inoperable after abnormal noise and high amperage conditions were observed. Disassembly of the pump revealed the impeller snap rings to be both heavily corroded and failed. Some of the snap rings were missing. Similar failures had previously occurred to the other two service water pumps. The failure of the snap rings raised concern about potential loss of the entire service water system by a single cause.

The snap rings were made of carbon steel and used to hold the impellers on the pump shaft. The licensee attributed the snap ring failure to corrosion from exposure to pumped chlorinated raw water. The pumps had not been inspected since the snap rings were last replaced between 1986 and 1987 because there was no routine periodic inspection program for these pumps.

Although carbon steel is seldom used for internal fittings of pumps handling raw water, the corrosion process requires a long time to cause destruction, and a proper periodic inspection can detect snap ring corrosion before pump performance is degraded. This event was judged to be unique and had no generic implications.

6.1.25 Seal Problems in Boric Acid Transfer Pumps

(Technical Review Report AEOD/T90-14)

Turkey Point reported a number of failures of the seals on the boric acid transfer pumps. The original seal design included a water pot to ensure lubrication of the seals with clean water rather than borated water, which could precipitate on the seal faces. Nitrogen was used to pressurize the water pot. A seal failure could lead to uncontrolled intrusion of nitrogen into the pump suction and loss of the boration function if the cover gas was not isolated beforehand.

The licensees of seven other plants disclosed that they used single mechanical seals for their pumps rather than the double seal used at Turkey Point. Seal failure does not appear to degrade pump performance, but it does require corrective action. Discussions with plant personnel indicated chronic housekeeping and maintenance problems with the 12 weight percent boric acid in the system. To minimize the housekeeping problem, one utility is changing to a pump without a seal as part of their modification to 4 weight percent boric acid in the system.

The seal water system used at Turkey Point appeared unique and the safety concern associated with the cover gas was previously addressed in NRC Information Notice 87-57, "Loss of Emergency Boration Capability Due to Nitrogen Gas Intrusion,"

(November 6, 1987) and by administrative procedures at the plant.

6.1.26 Impact of Pipe Liner Failure on Pump Operation

(Technical Review Report AEOD/T90-16)

Anomalous behavior of the component cooling water pump at Beaver Valley occurred when vibration-induced cyclic fatigue caused the weld joining liner and the expansion joint to fail. The vibration was caused by adverse flow conditions at the expansion joint that connected a reducing elbow to the pump suction. The failed liner was drawn into the pump suction causing binding and loss of flow.

One other pipe liner separation event was found in a search of NRC databases. However, there have been several events in which pipe coatings have come loose and restricted flow. The service water system would be the most likely system to have pipe coatings. Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," (July 18, 1989), contains guidance on this issue.

6.1.27 Inadvertent Containment Spray Actuations

(Technical Review Report AEOD/T90-17)

As a result of a containment spray system actuation that occurred at San Onofre Unit 2 on November 20, 1990, the AEOD staff performed a study of inadvertent containment spray system actuations. In this study, they searched for data regarding inadvertent containment spray system actuations using NUDOCS, the Sequence Coding and Search System (SCSS), and the engineered safety features

(ESF) data maintained for AEOD by INEL. For the period from 1984 to 1990, the staff identified six events in which the containment spray system actually sprayed into the containment and identified an additional six such events in operating experience data for the period 1980 to 1983.

Examination of the six more recent events indicated a limited amount of short-term damage. Some electrical equipment degraded because of grounds, shorts, or corrosion. Safety-related equipment did not appear to be damaged, but an inadvertent start of a reactor coolant pump at the Kewaunee plant was probably caused by a containment spray actuation that had occurred 3 weeks earlier. The event at San Onofre Unit 2 had caused degradation of non-safety-related electrical connectors. Because the Office of Nuclear Reactor Regulation (NRR) had identified potential long-term effects on equipment as an area that they would further review, we concluded that no further action by AEOD was necessary at this time.

6.1.28 Evaluation of Safety Equipment Outages for Significance at Zion (Revised)

(Technical Review Report AEOD/T925A)

AEOD staff evaluated the effects on plant safety of four Zion events reported November 22 and 23, 1989. The events involved failures or outages of diesel generators and the outage of one emergency service water pump. The conditions pertained to both plant units and potentially affected the station's ability to cope with a loss of offsite power (LOOP) initiating event. If a LOOP had occurred and emergency service water (ESW) had not been successfully provided to the operating diesel generator (DG), the DG would have failed within about 5 minutes. If that occurred and offsite power was not recovered within some critical time period, core damage could occur. The estimated conditional core damage probability (CDPr) for the

7 days exposure to LOOP assumed in the analysis was

$$\begin{aligned} \text{CDPr} &= (\text{exposure time} \\ &\text{fraction})(\text{LOOP freq/yr})(\text{fail} \\ &\text{recover LOOP short term})(\text{fail to} \\ &\text{recover LOOP long term or seal} \\ &\text{LOCA})(\text{DG 2B or ESW 2C failure}) \\ &= \text{about } 6\text{E-7}. \end{aligned}$$

The original analysis erroneously assumed that diesel generator 2B, the only DG assumed available, was wired to emergency service water pump 2B, which was out for maintenance. Therefore, the initial analysis was not correct. The revised analysis showed the event to be of low (about 6E-7) estimated conditional CDPr.

6.2 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 structured means of evaluating the safety significance of operational experience. This program is administered and directed by AEOD and emphasizes evaluations of licensee event reports (LERs) of U.S. commercial light-water reactors to identify and categorize precursors to potential severe core damage accidents.

The ASP method models and evaluates plant equipment and human responses that could affect the progression of an accident, evaluating the actual failures that have occurred along with the probabilities for postulated additional failures that could occur. Precursors are important because they are combinations of events, actually experienced, which, if accompanied by additional events or failures, could result in core damage scenarios.

Precursors also can provide insights into the capability of a plant to respond to accidents or other incidents. Precursors identified in the ASP Program have an estimated probability of at least $1E-6$ of resulting in core damage.

A nuclear plant Accident Sequence Precursor is an observed situation, event, or combination of events which, had the precursor occurred along with other postulated events or failures, could have resulted in a plant condition leading to severe core damage.

As indicated by the above definition, a precursor can be more than a single event. A precursor can be a combination of events in a given plant situation that form a part of postulated sequences of events leading to severe core damage. Nuclear plant damage cannot occur unless an accident or initiating event occurs and plant safety equipment fails to respond adequately. Thus, an initiating event, such as a loss of offsite power (LOOP) or a loss-of-coolant accident (LOCA), is required along with actual or postulated failure of plant equipment intended to mitigate the initiator. In a precursor, not all parts of an accident scenario happen; therefore, the occurrence of the remaining parts of a scenario, either initiating event or plant equipment failures have to be postulated. The likelihood of these postulated occurrences are estimated using reliability data. The two major classes of precursor events are described below.

- (1) The first class of precursor events involves the unavailability of systems. These are events in which one or more safety system was found failed or degraded during surveillance of the equipment. These events can be important because while safety systems are degraded or unavailable, the plant can be at increased risk to initiator events (e.g., LOOP) which require the system or systems that are functionally unavailable. In these unavailability events, an initiating event and often additional system failures need to be postulated for core damage to occur.

- (2) The second class of precursor events involves initiators or plant challenges. These events may be accidents or other initiators requiring plant response to mitigate the effects of the event. The initiator can be a LOOP, a LOCA, or other plant upsets or transients requiring response. The significance of this type of event is determined by assessing the degree to which plant equipment worked properly while responding to mitigate the initiator event.

The precursor method uses event tree models to evaluate the likelihood of various possible outcomes (scenarios) for the events being modeled, resulting in a quantitative estimate of the significance of the event in terms of conditional core damage probability. The ASP methods and models are probabilistic and they have corresponding limitations and uncertainties of results. One limitation of the results of this method is that the evaluations require an estimation by the analyst of the likelihood of not recovering failed or unavailable equipment when needed.

Operational events for the years 1969 through 1989 were reviewed. The types of precursor found during calendar years 1989 and 1990 are generally similar to the types of precursors found in previous years; however, the reports tend to track the findings of other studies of reactor safety vulnerabilities, as the new knowledge becomes known. For example, the instrument air system and components associated with this system, such as solenoid-operated valves, have been shown by AEOD studies to be important.

The steam generator tube rupture (SGTR) events were more frequent during 1989 (two events) than in previous years. The SGTR was known to be important because (1) if the SG with the leaking tube could not be isolated, continuous flow from the primary to the secondary system would occur, depleting water available for core cooling, and

(2) risks of adverse health effects, though not evaluated by ASP, can be associated with these types of events. The concern is that until the SG with the leaking tube is isolated, primary side coolant discharges in a nearly direct path to the atmosphere with potential consequences from the radioactivity in the primary coolant water. The SGTR event that occurred at McGuire Unit 1 on March 7, 1989, had an estimated conditional core damage probability of 7.7 E-4 . This was the most significant precursor event for 1989.

Precursors are still being found that reflect hidden inherent equipment deficiencies, such as design errors, which are only uncovered during design reviews or exceptional demands on the equipment. For example, during the Palo Verde 3 event of March 3, 1989, manual operation of atmospheric dump valves (ADV) was hindered by equipment designs or conditions that increased the likelihood of human errors. In this event, the ADV equipment room lighting was poor, the opening direction to manually operate the ADVs was not consistent between valves, and the available procedure to operate the equipment was not adequate or was poorly written.

Operational experiences with equipment failures or initiating events while the plants are shut down for maintenance or refueling have been undergoing increased scrutiny over the last few years. For example, on March 20, 1990, while Vogtle 1 was shut down for refueling, critical ac power from the offsite power grid was inadvertently lost and the emergency diesel generators, which are intended to back up the offsite power for redundancy, initially failed, although one was subsequently recovered. The event had an estimated conditional core damage probability of slightly less than 1 E-3 .

Table 6.1 lists precursor events that occurred in 1989. Figure 6.1 shows the number of precursors by the year of occurrence. The precursors are further sorted in each year to show estimated conditional core damage probabilities.

Figure 6.1 1984 through 1989 precursor quantities vs year of occurrence

Table 6.1 Precursor events occurring in 1989

Plant	Event Date	Precursor Event Description	Core Damage Probability
McGuire 1	03/07	Steam generator tube rupture	7.7E-4
Arkansas 1	10/12	Two service water pump circuits inhibit auto-start	2.8E-4
Point Beach 2	03/29	LOOP due to Fire Deluge System actuation	2.5E-4
Three Mile Is. 1	11/14	Oil sludge renders emergency diesels inoperable	2.4E-4
North Anna 1	02/25	Steam generator tube leak and residual heat removal (RHR) suction valve failure	1.9E-4
Summer 1	07/11	Trip causes LOOP	1.5E-4
Arkansas 2	04/18	Trip with unavailability of one train of AFW	1.2E-4
Crystal River 3	06/29	LOOP with one DG unavailable and degraded emergency feedwater	6.3E-5
Palo Verde 3	03/03	Atmospheric dump valves and non-safety related buses unavailable after trip	4.9E-5
Brunswick 2	06/17	Reactor manual scram causes a LOOP	3.6E-5
Crystal River 3	06/16	LOOP with degraded emergency feedwater	3.5E-5
McGuire 2	09/05	Containment spray heat exchanger gasket failure	1.6E-5
Limerick 2	12/11	Reactor scram with degraded HPCI	1.5E-5
Dresden 3	03/25	LOOP with unavailable low-pressure coolant injection, degraded HPCI, and component cooling service water	1.3E-5
FitzPatrick	11/05	Reactor scram with HPCI system inoperable	1.3E-5
FitzPatrick	11/12	Scram with HPCI system inoperable while testing SRVs	1.3E-5
Harris 1	01/16	Reactor trip and loss of one train of AFW	1.3E-5
River Bend 1	05/02	Instrument air solenoid-operated valves installed backward impact ADS operability	1.3E-5
Cook 2	08/14	Trip due to control room instrumentation distribution (CRID-IV) failure	9.4E-6
Duane Arnold	03/04	Reactor scram and isolation with RCIC out of service	6.5E-6
FitzPatrick	03/06	Inadvertent SRV actuation with HPCI inoperable	6.5E-6
Sequoyah 1	12/16	RWST level and other instrument transmitters inoperable	4.8E-6
Harris 1	03/14	Reactor trip with one AFW pump out of service	4.4E-6
Harris 1	10/09	Reactor trip and trip of the turbine-driven AFW pump	4.4E-6
Sequoyah 2	12/16	RWST level and other instrument transmitters inoperable	3.6E-6
Trojan	10/30	High head safety injection inoperable due to unavailability of volume control tank isolation during testing	2.1E-6
Calvert Cliffs 1	03/14	Failure of instrument air boundary check valves	1.4E-6
Grand Gulf 1	12/06	Scram with RCIC unavailable	1.2E-6
Callaway	06/23	Trip, one DG out-of-service, and loss of ESF bus	1.2E-6
Farley 1	11/12	Motor driven AFW pumps fail to start under manual control	1.1E-6

6.3 Analyses of Human Performance in Operating Events

Studies of operating events have repeatedly shown the importance of human performance in reactor safety. To obtain additional information, AEOD began a new program to conduct onsite, in-depth studies of human performance during selected power reactor events. The primary goal of the program is to identify factors that affect human performance during events, whether the effect is positive or negative.

Under this program, teams of NRC staff and contractor specialists perform studies of selected events at plant sites shortly after the events occur. Interviews with the people involved in the event, instrument and computer recordings, and log book entries provide the basis for these reports. The results are presented in reports of site visits that are issued approximately 1 month after the event. Copies are provided to the affected plant managers and staff and to interested parties within the NRC. During 1990, six studies were completed (event date in parentheses):

1. Peach Bottom 3 Loss of EHC (1/28/90)
2. Catawba 1 Overpressurization (3/20/90)
3. Nine Mile Point 2 Partial Loss of Instrument Air (5/14/90)
4. Dresden 2 Safety Relief Valve Stuck Open (8/02/90)
5. Braidwood 1 Loss of Reactor Coolant (10/04/90)
6. Quad Cities 2 Reactor Scram Caused by Erroneous Control Rod Withdrawal (10/17/90)

The Peach Bottom Unit 3 event occurred while the plant was at 99.8 percent power. A major leak of electrohydraulic fluid (EHC) was observed from the main turbine control valve. In anticipation of a potential turbine trip without bypass transient, the shift manager ordered a fast power reduction. The plant was manually scrambled from 50 percent power. Although feedwater pumps A and B were manually tripped, the level transient following the

scram caused a reactor high-water-level trip of feedwater (FW) pump C. The operators were unable to restart the feedwater pumps and controlled the reactor water level by using reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI). These activities required many operator actions.

Peach Bottom findings: The operating crew experienced a high level of stress caused by the anticipation of a potential turbine trip without bypass and the need to overcome man-machine interface problems. Good control of HPCI flow to the vessel could not be achieved because of the lack of a direct-reading flow instrument. Prior training and good communications helped the crew shut down the plant safely.

The Catawba Unit 1 event occurred while the plant was in cold shutdown. During a reactor fill and vent operation, the reactor coolant system (RCS) and residual heat removal (RHR) system were inadvertently overpressurized because the operators were monitoring pressure instrumentation that was inoperable.

Catawba findings: Plant procedures for planning and scheduling did not ensure that the instrumentation was returned to service in a timely manner and plant tagging procedures did not require tagging of inoperable control room instruments. The operators vented the RCS longer than usual before system pressurization without considering that this venting would reduce the amount of air in the system and, consequently, would cause the pressure to rise. Also, they did not monitor redundant RCS pressure instrumentation.

At Nine Mile Point Unit 2, the operators manually scrambled the plant from 100 percent power when the plant experienced a partial loss of condenser vacuum. The operators quickly determined that the loss of the condenser vacuum was due to an offgas system problem that developed because of a ruptured instrument air line.

Nine Mile Point findings: Troubleshooting of the problem was hampered because the piping and Instrument diagrams for the air system were not complete. Procedures and training for loss of condenser vacuum and loss of air were weak for treating partial losses. The control room crew diagnosed the equipment problem accurately and responded quickly in spite of numerous nuisance annunciators, exemplifying good crew performance.

The Dresden Unit 2 event occurred while the plant was at 87 percent power. The operator manually scrammed the plant after trying unsuccessfully to shut a Target Rock Relief valve that had failed in the open position, and during the next hour, the plant cooldown rate reached 129° F an hour, exceeding the technical specification normal cooldown rate limit of 100° F an hour.

Dresden findings: The control room emergency organization provided little assistance to the shift engineer while the shift engineer was analyzing the condition of the plant and in evaluating operator activities. One order by the shift engineer was not carried out because of a communication problem. Procedures and training did not completely address problems that might be encountered when a relief valve is stuck open.

The Braidwood 1 event occurred while the plant was in cold shutdown, when approximately 600 gallons of reactor coolant were inadvertently discharged through a relief valve, resulting in contamination of licensee personnel. The study was conducted as part of a Region III Augmented Inspection Team (AIT) assignment. Region III issued the AIT report about this event October 23, 1990.

Braidwood findings: The findings from the Braidwood study report concerned task awareness and coordination and teamwork. At the time of the event, two different tests were simultaneously being performed on the same system under the direction of test engineers from the control room. However, the licensed senior control room operators were unaware that the testing was in progress. In addition, the test engineers had been working for more than 17 hours.

The Quad Cities Unit 2 event occurred while the plant was in hot standby. The reactor scrammed on hi-hi intermediate range flux because the operator withdrew rods to increase reactor pressure without recognizing the need to follow the normal procedures for reestablishing reactor criticality. NRC Information Notice 91-04, "Reactor Scram Following Control Rod Withdrawal Associated With Low-Power Turbine Testing," was later issued as a result of this event.

Quad Cities findings: The operations crew was not sufficiently aware that careful reactivity management would be required while installing and removing test equipment to avoid either supercriticality or short startup periods. The senior reactor operators did not adequately monitor control rod manipulations by the unit nuclear station operator (NSO). The procedure governing operations when going from power operation to hot standby did not caution personnel about problems that might be encountered with high rod worths. The crew had not been prepared and trained to handle the plant conditions. Finally, while an earlier shift had experienced high-notch worth, the information about their experience was neither recorded nor passed on.

Studies to date have revealed human performance issues such as task awareness, command and control, communications, stress, man-machine interface, simulator fidelity, shift technical advisor role, tagging, administrative controls, procedures, training, and the feedback of operating experience information. AEOD plans to continue this program to study events in which human performance plays an aggravating or a mitigating factor. An interim report will be issued during 1991 to summarize the results and potential generic findings of this program.

7 Incident Investigation Program

The Incident Investigation Program (IIP) ensures that NRC investigations of significant events are timely, thorough, well coordinated, and formally administered. The scope of the IIP includes investigations of significant operational events involving reactor and nonreactor 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.

Administration of the incident investigative activities of the NRC are prescribed in NUREG-1303, "Incident Investigation Manual." As described in the NUREG, AEOD has responsibility for overall administration of the IIP, while NRR is responsible for maintaining the procedures for an AIT response.

During 1990, one IIT was established to investigate the loss of vital ac power and the residual heat removal system during mid-loop operations at Vogtle Electric Generating Plant Unit 1. A second IIT was established for a nonreactor incident and is discussed in the AEOD 1990 annual report for nonreactors (NUREG-1405). The status of staff actions that the EDO assigned to various NRC offices associated with IIT reports are given in Appendix F.

Incident Investigation of the Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operation at Vogtle Unit 1 on March 20, 1990.

On March 20, 1990, the Vogtle Electric Generating Plant Unit 1, located in Burke County Georgia (about 25 miles southeast of Augusta), experienced a loss of all safety (vital) ac power. The plant was in cold shutdown with the reactor coolant level lowered to "mid-loop" for various maintenance tasks. Both the containment building personnel hatch and the equipment hatch were open. One emergency diesel generator and one reserve auxiliary transformer were out of service for maintenance, and the remaining reserve auxiliary transformer was supplying both Unit 1 safety buses. A fuel and lubricants truck in the low-voltage switchyard backed into the support column for an offsite power feed to the reserve auxiliary transformer that was supplying power. The insulator broke, a phase-to-ground fault occurred, and the feeder circuit breakers for the safety busses opened. The operable emergency diesel generator started automatically because of the undervoltage condition on the safety bus, but tripped off after about 1 minute. About 20 minutes later, the diesel generator automatically started a second time when its load sequencer was reset. The diesel generator operated for about 1 minute and tripped again. The diesel generator was restarted in the manual emergency mode 36 minutes after the loss of power. The generator then remained on line and provided power to its safety bus. During the 36 minutes without safety bus power, the reactor coolant system temperature rose from about 90° F to 136° F.

On March 23, 1990, the EDO sent an IIT to the site. The team was directed to determine what happened, identify probable causes, and make appropriate findings and conclusions to form a basis for any necessary follow-on actions. The team's report is documented in NUREG-1410.

The team found that nonconservative initial conditions, combined with a failure to adequately control switchyard activities, led to the incident. Shutdown electrical redundancy was limited to only

2 of 4 safety bus power supplies. The evaluation of the diesel generator trips indicated that the most probable cause involved the failure of Calcon jacket water high-temperature trip sensors. A significant number of Calcon sensor failures had occurred at Vogtle since 1985. In addition, the team concluded that the Vogtle staff had no effective control over the fuel and lubricants truck conducting routine operations in the switchyard. Moreover, because the truck carried fuel, the risk existed of a conflagration from ignition of fuel caused by electric arcing.

The summary conclusions of the team were as follows:

- Adequate precursor information was available to make this incident preventable.
- The Vogtle staff generally handled the incident well.
- Significant potential generic lessons were identified, including:
 - Approaches to shutdown risk management need to be developed.
 - There is incomplete implementation of existing analysis and guidance into procedures and training.
 - There is a need for additional analysis of reactor coolant system behavior following the loss of the residual heat removal system.
 - There is a need for further synthesis of existing operating information.

- Emergency classification guidance and implementation problems exist.
- The plant Technical Specifications do not take into consideration the risk associated with the various configurations of systems that may exist during shutdown conditions.
- At least some diesel generator control and annunciator systems are complex and may not be well understood.

IIT Training Program

The purpose of the IIT training program is to provide prospective IIT members with comprehensive guidance and methodology in conducting systematic and technically sound incident investigations. The training program was developed by AEOD following discussion with representatives of the National Transportation Safety Board (NTSB), the Federal Aviation Administration (FAA), and the National Aeronautics and Space Administration (NASA).

The first IIT refresher training course was conducted in January 1990. The class was composed of candidate IIT team members and leaders who had attended the first training course or previously participated in an IIT. The course covered IIT program changes and reviewed basic investigation techniques.

8 Augmented Inspection Team

The incident investigation program includes investigatory responses by an Incident Investigation Team (IIT) and the less formal response by an Augmented Inspection Team (AIT).

For events of less significance than those requiring an IIT, the cognizant NRC Regional Administrator may establish an Augmented Inspection Team (AIT) to investigate the event. During 1990, nine AIT inspections were conducted at power generation facilities (see Table 8.1). These AITs 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, 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 potential generic 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 engineering studies and reviews of operating experience, in generic communications to licensees.

The guidelines for administration of the investigative activities of the NRC are prescribed in NUREG-1303, "Incident Investigation Manual." The procedure for an AIT response is maintained by NRR, and is part of NUREG-1303.

Examples of lessons learned and communicated to licensees from events investigated by AITs in 1990 include the following:

- On October 8, 1990, the Tennessee Valley Authority noted that three of four main steam isolation check valves (MSICVs) failed at the Sequoyah Nuclear Power Plant Unit 1. Each failure involved the disc separating from the swing arm. As a result of the AIT evaluation of this event, NRC Information Notice (IN) 90-79 was issued. This notice alerted licensees to potential problems involving the design and location of MSICVs that could result in disc separation as a result of fatigue failure of the post (or stud).
- On October 4, 1990, Indian Point Nuclear Generating Unit 3 was in a refueling outage. The licensee was removing the upper core support structure (upper internals) from the reactor vessel to prepare for refueling.

Upon initiating a lateral move of the upper core support structure, movement was stopped upon the discovery that two peripheral fuel assemblies were attached to the bottom of the upper core plate, which is part of the upper internals package. As a result of the AIT evaluation of this event, NRC IN 90-77 was issued. This notice alerted licensees to potential problems pertaining to the removal of nuclear fuel from the reactor core.

- On February 11, 1990, at River Bend Station, the low-pressure coolant injection valves opened while at full power. The valves opened as a result of the reenergization of Rosemount transmitter trip units during maintenance. As a result of the AIT evaluation of this event, NRC IN 90-22 was issued. This notice alerted licensees to potential problems resulting from the reenergization of Rosemount transmitter trip units.

Table 8.1 Plants at Which AITs Conducted Inspections In 1990

Event Date	Plant	Event Description
01/16/90	Dresden 2	Loss of offsite power with multiple equipment failures
02/11/90	River Bend	LPCI injection valve received signal to open at full power
03/12/90	Comanche Peak 1	Inadvertent actuation of safety injection system
03/19/90	Brunswick 2	Plant trip when steam isolation valves close
09/26/90	Ginna 1	Reactor trip with complications
10/04/90	Braidwood 1	Reactor coolant leak outside of containment
10/04/90	Indian Point 3	Spent fuel assemblies remain attached to upper internals
10/08/90	Sequoyah 1	Main steam line check valves disc nut pin failures
12/31/90	Millstone 3	Condensate system piping ruptures

9 Diagnostic Evaluation Program

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

When a diagnostic evaluation (DE) is approved for a specific reactor facility, the EDO authorizes and establishes a Diagnostic Evaluation Team (DET). The DET consists of expert technical staff members from headquarters and the regional offices, experienced resident inspectors, and contractors, if appropriate. The DET manager and team members selected will not have had recent significant involvement in the licensing, inspection, or enforcement process at the facility to receive the DE. 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 effectiveness, operations, engineering, and quality programs.

During 1990 and early 1991, several significant improvements were made in the DEP. Manual Chapter 0520 and Appendix, "NRC Diagnostic Evaluation Program," was revised to address changes to the DEP requested by the EDO as well as other changes derived from the evolution and implementation experiences of the program since its inception in 1987. The revised Manual Chapter 0520 and Appendix was then reissued as NRC Directive and Handbook 8.7 in accordance with the NRC's new Management Directive System. The new directive and handbook include several changes in

DEP objectives and the responsibilities and authorities of the offices involved in the implementation of the DEP. In addition, the DET Guidelines were revised and issued and a roster of potential DET members was approved by the EDO. The guidelines provide additional information for the DET on the implementation of a DE at a reactor facility and this revision of the guidelines includes information to implement the above changes in NRC Directive and Handbook 8.7. The roster of potential DET members was selected by AEOD in coordination with senior management from the respective NRC Offices based upon the candidate's qualifications and experience.

During 1990, two reactor facility evaluations were conducted under the DEP. DETs were established to evaluate performance at the Zion Nuclear Power Station and Oyster Creek Nuclear Generating Station. Summaries of the major findings from these diagnostic evaluations are as follows:

Zion Nuclear Power Station Diagnostic Evaluation

The Zion Nuclear Power Station is composed of two 1040-MWe pressurized-water reactors located near Zion, Illinois, and operated by the Commonwealth Edison Company. Unit 1 and 2 began commercial operation in 1973 and 1974, respectively.

The DE, conducted from June 4 through June 29, 1990, concluded that the root causes of the licensee's performance problems were (1) insufficient corporate support for timely resolution of problems, (2) weak station management oversight and control, (3) the plant staff did not always recognize the need to make changes to improve safety performance, and (4) the plant design basis was not well documented or understood. Failures to effectively transfer lessons learned during the

Dresden diagnostic evaluation to Zion also contributed to performance problems found at Zion.

In the area of operations, the team found that management control over operator activities was weak, teamwork within the operating department was poor, operator overtime was not well managed or controlled, and operators and their management accepted equipment degradation and cumbersome administrative requirements and worked around these problems. In addition, nuclear station operators' knowledge of the current condition and configuration of plant equipment was limited because they seldom worked outside the control room. Operators lacked confidence in supervisors, did not feel accountable to management, and were not closely supervised.

The DET found major weaknesses in the maintenance area. Safety-related systems were poorly maintained. Corrective maintenance was not timely in several instances, preventive maintenance was limited, root causes of equipment failures were not identified in many instances, and motor-operated valves were poorly maintained. Although the DET focused its attention on the service water system, the identified programmatic deficiencies applied to all safety-related systems and motor-operated valve deficiencies were found in several systems. The team therefore concluded that other safety systems were likely to be poorly maintained.

The team determined that testing of safety-related components was not adequate to detect degradation. Many valves were not tested in accordance with American Society of Mechanical Engineers (ASME) Section XI. Service water heat exchangers were not routinely inspected, cleaned, or tested. Service water pump testing was not adequate. Adequate flow balancing of the service water system loads for accident conditions with the units cross-tied had not been demonstrated. Diagnostic testing of motor-operated valves was only recently initiated. Poor material condition was caused by years of operation without an effective

preventive maintenance program. Poor maintenance and testing were caused by deficient programs that did not clearly assign responsibility. Lack of effective direction and oversight from the corporate office managers also contributed to the above deficiencies.

In the areas of engineering design and technical support, the team concluded that seven main weaknesses continued to exist despite recent improvement programs: (1) the service water system design basis was not well documented or understood, (2) in-house design engineering expertise was weak due in part to corporate staff acting mainly as contract administrators, (3) no safety analysis or operating procedure had been developed for post-accident cooling with one unit operating and the other shut down, (4) accident vulnerabilities were discovered with additional potential accident conditions to be further evaluated by NRC, (5) analyses for some past modifications were weak or nonexistent, (6) weaknesses existed in identifying and correcting root causes of equipment problems, and (7) issues identified by the Dresden diagnostic evaluation had not been adequately addressed at Zion.

In the area of quality programs, the team noted that effective self-assessment programs had provided the licensee with a good understanding of performance problems. However, many corrective actions had not been timely or effective. The amount of work needed to resolve known problems appeared to be more than could be fully accomplished in a timely manner by the station staff. The licensee had not determined the resources needed. Most management attention and staff resources were directed toward identifying and resolving performance problems rather than addressing root causes.

Oyster Creek Nuclear Generating Station Diagnostic Evaluation

The Oyster Creek Nuclear Generating Station consists of a single 620 MWe boiling- water reactor located in Ocean County, New Jersey, and operated by GPU Nuclear Corporation. Oyster Creek began commercial operation in 1969.

The DE, conducted November 5 through December 7, 1990, found that at the time of the evaluation, Oyster Creek was in a state of transition; the licensee was still in the process of recovering the plant material condition resulting from minimal maintenance in the plant's early history and implementing a variety of improvement initiatives. Many of the strengths observed in the licensee's performance were primarily due to the experience and skills of individual personnel and managers. Continued improvements were anticipated, although improvement would be hampered by the weaknesses in work practices.

The weaknesses in the licensee's current performance generally applied to those tasks that were not perceived to be important to management. The root causes were (1) weak administrative control of work because of the lack of an infrastructure for the systematic completion of work, (2) weak supervision and independent verification which had caused some inadequate evaluations and poorly reasoned decisions, and (3) a general lack of rigor and inquisitiveness in the licensee's work practices that hampered the ability to achieve consistently effective solutions to problems.

The conduct of plant operations was good, and training facility improvements were evident. The operations department had an effective plan to ensure adequate staffing. Plant operators had a positive attitude, were professional and attentive, were knowledgeable of the plant and plant conditions, and conducted effective shift turnovers. Operator performance was hampered by having to

work around some equipment problems. Operations improvement initiatives were having a positive effect on the conduct of plant operation.

The maintenance and testing practices were adequate and improving. Current improvement initiatives were directed mostly toward recovery of the material condition of the plant, which was improving. The scope of preventive maintenance was relatively low, although preventive maintenance was increasing as reliability-centered maintenance techniques were implemented. Skipped preventive maintenance tasks were poorly controlled. Efforts to make substantive programmatic improvements to maintenance, including implementation of life-of-system maintenance plans and equipment failure trending, were impeded because these initiatives lacked formalization.

The quality of engineering and technical support varied. Tasks perceived to be important were performed well. However, the performance on other tasks often lacked rigor, lacked inquisitiveness, or failed to clearly identify the failure mechanism. Further, there was no systematic means of determining the operability or functional capability of equipment with deficiencies to establish how soon corrective actions were necessary. Engineering roles and responsibilities appeared to be understood, and appropriate technical resources were available when a need was identified.

Management effectiveness was improving but still had weaknesses. The plan for excellence in performance identified important goals and objectives that relate to most performance problems. Ample staffing and funding were available to support plant activities, and relations with the union had improved substantially. Considerable emphasis was placed on employee morale and development. Communication horizontally and vertically in the organization was improving. Performance indicators were used effectively. Work tracking was not always prioritized or tracked well; many of the implementation schedules in the excellence plan had slipped. Followup and closeout

of some deficient conditions, including audit findings and material condition issues, were often untimely or superficial because of poor accountability, the complexity of corrective action systems, and weak independent verification of the resolution of problems. Improvement Initiatives were similarly hampered.

10 Committee To Review Generic Requirements

The Committee To Review Generic Requirements (CRGR) reviews all generic requirements proposed by the NRC staff that involve one or more classes of power reactors. The CRGR is made up of senior managers from various headquarters program offices and, on a rotational basis, from one of the NRC regional offices. While performing the CRGR review function, a CRGR member expresses an individual professional opinion about each issue considered rather than representing the view of his or her respective office. The members of the CRGR determine whether proposed new generic requirements have sufficient merit in terms of safety and are justified in terms of cost (where appropriate), before reaching a consensus recommendation about each issue considered. Each independent CRGR recommendation is given to the NRC Executive Director for Operations (EDO) to consider. 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.

The CRGR held 21 meetings in 1990. During these meetings, they considered the following 26 issues:

- Revision to a regulatory guide on diesel generator reliability.
- Appeal by the Nuclear Utility Backfitting and Reform Group on the use of nuclear heat in hydrostatic testing at boiling-water reactors.
- Revision to a regulatory guide on quality assurance at operating reactors.
- Supplement to a generic letter on individual plant examinations that would address containment performance improvement programs.
- Proposed rule on emergency preparedness related to 10 CFR Part 52 licensing.
- Generic letter on interaction between licensees and vendors.
- Bulletin on loss of fill-oil in transmitters manufactured by Rosemont.
- Generic letter on critical power ratio determinations related to channel box bow in boiling-water reactors.
- NUREG-series reports on moving the technical specifications about radiological effluents to the offsite dose calculation manual.
- Proposed rule to endorse recent versions of the American Society of Mechanical Engineers' Code.
- Generic letter on non-code repairs to Class 3 piping.
- Supplement to a generic letter on service water systems to provide further guidance.
- Proposed rule on license renewal for nuclear power plants.
- Generic letter on individual plant examinations for external events.
- NUREG-series report on backfitting guidelines.
- Guidance for implementing a generic letter on testing motor-operated valves.
- Safety evaluation report about hydrogen control in BWR-6 Mark III containments.
- Generic letter on stress in reactor vessels during natural convection cooldown.
- Proposed rule on emergency response data system.

- Safety evaluation report on General Electric Co. fuel design.
 - Appeal by BWR Owners Group on diversity requirements under the rule for anticipated transients without reactor scram
 - Revision to the standard review plan on quality assurance
 - Supplement to a generic letter on testing of motor-operated valves to address specific valve deficiencies found in NRC-sponsored tests
 - Proposed regulatory guide and the standard review plan for nuclear power plant license renewal
 - Final rule on pressurized thermal shock
 - Proposed resolution of a generic issue about reactor coolant pump seal failure and implementing Regulatory Guide
2. Conduct of a series of workshops for industry and the NRC staff to exchange views on NRC's policy on backfitting and to identify suggestions for (1) improving NRC's backfit policy and (2) resolving the problems the industry and the NRC face in implementing that policy. Conduct of meetings with industry groups interested in backfitting.
 3. Revision of the CRGR Charter to increase its emphasis on considering the cumulative effect of generic backfits and considering Commission's safety goal policy in CRGR reviews.
 4. Conduct of backfit training sessions, including a series of sessions for personnel in the Office of Nuclear Reactor Regulation.

Other important initiatives and accomplishments of the CRGR and the CRGR staff during 1990 included the following:

1. Publication of NUREG-1409, "Backfitting Guidelines," dated July 1990, which provides a clearer understanding of the backfit rule and both the generic and plant-specific backfit policies and associated processes that have been adopted by the NRC.

The CRGR also periodically visits operating power reactors for discussions with the licensee's engineering, management, and operations staff as another means of assessing the effect of NRC generic communications and new generic requirements on the operation and safety of power reactor facilities. During 1990, the committee visited Indian Point Units 2 and 3, operated by the Consolidated Edison Company and the New York Power Authority, respectively.

Finally, the Director, AEOD, and the CRGR staff oversee 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

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Performance Indicator Program and Data

APPENDIX A-1

Performance Indicator Program and Data

Introduction

The NRC program for monitoring Performance Indicators (PIs) for Operating Commercial Nuclear Power Reactors includes the following eight indicators: (1) the number of unplanned automatic scrams while a reactor is critical, (2) the number of selected safety system actuations (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 1000 commercial critical hours, (7) the collective radiation exposure per plant, and (8) cause code trends.

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

This appendix provides the background of the development of the PI program and the definitions of the PIs. It also provides tables containing PI data for 1989 and 1990, and a listing of the significant events that occurred during 1990.

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 operating plants, testing 17 perspective performance indicators. For the most part, this trial program used data through calendar year 1985. The group then selected eight performance indicators as candidate PIs for initial implementation. After considering industry comments, the staff deleted one of the candidate PIs, the corrective maintenance backlog.

In October 1986, the NRC prepared a prototype report by expanding the trial program to include data for the first half of 1986 on 100 operating reactors. The staff discussed the recommended program, the task group report, and the prototype report in 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 differed as a result of differences in the reactor populations used to calculate the indicators and differences in data compilation. The trends in performance shown by the NRC and INPO indicators are consistent despite differences in specific values.

During 1989, the staff proposed and the Commission approved the use of cause code trends 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 actions 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 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 PI indicators in use at the time. The cause codes were of value for monitoring performance and diagnosing performance problems, thereby providing another screening tool. For monitoring performance, the cause codes could identify plant performance outliers that were not identified by the existing set of indicators. For diagnosing problems, the cause codes could provide a programmatic view of the trends in performance. Cause code trends were incorporated in the second quarter 1989 PI report published in August 1989.

Since February 1987, the AEOD staff has provided quarterly PI reports to the Commission and to NRC senior managers. The reports are also placed in the NRC's Public Document Room. Beginning with the issuance of the PI report for the fourth quarter of calendar year 1989, the staff routinely provides plant-specific information and industry average data extracted from each PI report to licensee managers.

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

Automatic Scrams While Critical (Scrams)

This indicator is the number of all unplanned automatic scrams that occur while the reactor is critical; a reactor scram means any actuation of the

reactor protection system that results in control rod motion. INPO uses an Indicator, "Unplanned Automatic Scrams Per 7000 Hours Critical," that is similar to this PI. AEOD also monitors the number of automatic scrams that occur while the reactor is critical at power levels equal to or below 15 percent power and the number of automatic scrams per 1000 critical hours that occur while the reactor is above 15 percent power.

Safety System Actuations (SSAs)

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

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

Significant Events (SEs)

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

Safety System Failures (SSFs)

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

Forced Outage Rate

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

Equipment-Forced Outages (EFOs) per 1000 Commercial Critical Hours (EFO Rate)

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

Collective Radiation Exposure

This indicator is the total radiation dose accumulated by unit personnel. With the exception of the Indian Point and Millstone nuclear plants, unit values at 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 administrative control problems; licensed operator errors; other personnel errors; maintenance problems; design, construction installation, or fabrication problems; equipment failures (electronic piece-part or environmentally related failures).

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

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1	1	0	3	0	0	0	0
Arkansas 2	0	1	0	1	0	1	1	0
Beaver Valley 1	2	1	0	1	1	0	0	0
Beaver Valley 2	1	0	0	0	0	0	1	0
Big Rock Point	0	0	1	0	0	0	0	0
Braidwood 1	1	0	1	0	1	1	1	2
Braidwood 2	0	1	2	0	0	1	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	1	0
Brunswick 2	0	0	0	0	0	0	4	1
Byron 1	0	0	0	0	1	1	1	1
Byron 2	0	0	0	0	1	0	0	0
Callaway	0	1	0	0	0	2	0	2
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	0	0	0
Catawba 2	2	0	0	0	0	0	0	1
Clinton 1	0	1	0	0	0	0	1	0
Comanche Peak 1	NYC*	NYC	NYC	NYC	NYC	2	4	0
Cook 1	2	0	0	0	0	0	0	0
Cook 2	0	0	1	0	0	1	0	2
Cooper Station	1	0	1	1	0	0	0	1
Crystal River 3	0	1	0	0	0	0	0	0
Davis-Besse	1	1	0	0	1	0	0	1
Diablo Canyon 1	0	0	0	1	0	1	0	2
Diablo Canyon 2	0	1	0	0	0	0	0	0
Dresden 2	1	0	1	0	2	0	0	0
Dresden 3	2	1	0	0	1	0	0	0
Duane Arnold	2	1	1	0	0	1	2	1
Farley 1	0	0	0	1	0	0	0	0
Farley 2	0	2	0	3	0	1	0	0
Fermi 2	1	0	0	1	0	1	0	1
FitzPatrick	0	0	0	2	2	0	0	2
Fort Calhoun	0	0	0	0	0	0	0	0
Fort St. Vrain	0	0	0	NLI**	NLI	NLI	NLI	NLI
Ginna	0	1	0	0	0	2	1	3
Grand Gulf	0	1	2	1	0	0	2	2
Haddam Neck	0	0	0	0	0	0	0	0

Footnotes at end of table.

Table A-1.1 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0	0	0	0	0	1	0	1
Hatch 2	0	0	1	0	2	0	0	0
Hope Creek	0	0	1	1	2	0	0	2
Indian Point 2	1	0	0	1	0	0	0	0
Indian Point 3	0	0	0	0	0	1	0	0
Kewaunee	0	0	0	1	0	0	0	0
Lasalle 1	1	0	0	0	1	1	0	0
Lasalle 2	0	0	1	0	1	0	1	0
Limerick 1	0	0	0	0	0	0	0	0
Limerick 2	NYC	NYC	0	1	0	0	2	0
Maine Yankee	1	1	0	0	0	0	0	0
McGuire 1	0	0	1	0	1	0	0	2
McGuire 2	2	1	0	0	0	0	0	0
Millstone 1	0	2	0	1	0	0	1	0
Millstone 2	0	0	0	0	0	0	1	0
Millstone 3	0	1	0	0	1	1	0	0
Monticello	0	1	0	1	0	0	0	1
Nine Mile Pt. 1	0	0	0	0	0	0	0	2
Nine Mile Pt. 2	0	2	1	3	0	0	1	0
North Anna 1	1	0	1	1	1	0	0	0
North Anna 2	0	0	0	0	0	0	0	1
Oconee 1	1	0	1	0	0	0	1	0
Oconee 2	2	1	0	0	0	0	0	0
Oconee 3	1	0	1	0	2	0	0	0
Oyster Creek	0	2	2	0	0	1	0	0
Palisades	0	0	1	0	1	0	0	0
Palo Verde 1	1	0	0	0	0	0	1	0
Palo Verde 2	1	0	1	1	0	0	0	0
Palo Verde 3	1	0	0	0	0	1	0	1
Peach Bottom 2	0	1	1	2	0	0	0	0
Peach Bottom 3	0	0	0	0	1	0	0	0
Perry	0	0	0	0	1	0	0	0
Pilgrim	1	1	1	1	0	1	0	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	1	0	1	0	0	0	0	0
Prairie Island 1	0	0	1	0	0	0	0	1
Prairie Island 2	0	1	0	2	3	0	0	2
Quad Cities 1	0	1	0	0	1	0	0	0
Quad Cities 2	0	1	0	1	0	0	0	2
Rancho Seco	1	0	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.1 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	2	0	1	1	1	1	0	1
Robinson 2	3	0	0	0	1	1	0	0
Salem 1	2	1	0	0	0	1	2	0
Salem 2	2	1	0	0	0	1	1	0
San Onofre 1	0	0	1	0	0	1	0	0
San Onofre 2	0	0	0	0	0	0	0	1
San Onofre 3	1	1	0	0	1	0	0	0
Seabrook	NYC	0	0	0	0	1	2	1
Sequoyah 1	1	0	0	1	0	1	2	0
Sequoyah 2	0	3	1	0	0	1	0	1
Shearon Harris	5	0	0	1	0	0	0	0
South Texas 1	2	0	1	0	1	2	2	1
South Texas 2	0	3	5	1	2	1	1	0
St. Lucie 1	0	0	2	0	0	0	0	0
St. Lucie 2	0	1	0	0	1	0	0	0
Summer	0	1	1	1	0	0	0	0
Surry 1	0	0	1	0	0	1	0	0
Surry 2	0	0	2	0	0	0	0	0
Susquehanna 1	2	0	0	1	0	0	0	0
Susquehanna 2	0	0	0	0	1	1	0	0
Three Mile Island 1	0	0	0	1	1	0	0	0
Trojan	0	0	1	0	0	0	1	0
Turkey Point 3	1	0	0	0	0	2	0	0
Turkey Point 4	0	0	0	1	0	1	1	0
Vermont Yankee	0	0	0	0	1	1	0	1
Vogtle 1	0	1	0	1	1	0	0	0
Vogtle 2	0	3	1	2	1	1	0	0
Wash. Nuclear 2	1	1	2	0	0	0	0	1
Waterford 3	0	0	1	0	2	0	1	0
Wolf Creek	2	0	0	0	1	3	0	0
Yankee-Rowe	0	1	1	0	0	0	0	1
Zion 1	1	0	0	0	1	0	1	0
Zion 2	0	0	0	0	0	0	2	1
Total	59	48	47	43	44	43	43	46

*NYC means the plant is not yet critical.

**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-1.2 Number of Safety System Actuations—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1	0	0	2	0	0	0	1
Arkansas 2	0	4	0	2	0	0	1	0
Beaver Valley 1	0	1	0	2	0	0	0	0
Beaver Valley 2	1	1	0	0	0	0	0	1
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	1	0	1	0	0	2	0
Braidwood 2	0	0	0	0	0	1	0	0
Browns Ferry 1	1	0	0	0	0	2	0	0
Browns Ferry 2	1	0	0	0	0	0	1	0
Browns Ferry 3	0	0	0	0	0	0	0	0
Brunswick 1	0	0	0	1	0	0	0	0
Brunswick 2	1	3	0	2	0	0	6	1
Byron 1	0	0	0	0	0	0	0	0
Byron 2	1	0	0	0	1	0	1	0
Callaway	0	2	0	0	0	0	0	1
Calvert Cliffs 1	2	0	0	0	1	0	1	0
Calvert Cliffs 2	0	0	0	0	0	0	0	0
Catawba 1	2	0	0	0	0	0	0	0
Catawba 2	2	0	0	0	0	0	0	0
Clinton 1	0	0	0	0	1	0	0	0
Comanche Peak 1	NYL*	NYL	NYL	NYL	1	0	2	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	0	0	1	0	0	0
Cooper Station	0	2	0	1	0	2	0	2
Crystal River 3	0	4	0	1	0	0	0	0
Davis-Besse	0	0	0	0	0	3	0	0
Diablo Canyon 1	0	0	0	1	0	1	0	1
Diablo Canyon 2	0	0	0	0	0	0	0	0
Dresden 2	1	0	0	0	1	0	0	2
Dresden 3	2	0	0	0	0	0	0	0
Duane Arnold	2	0	2	0	1	0	1	1
Farley 1	0	0	0	1	0	0	0	0
Farley 2	0	1	0	0	0	0	0	1
Fermi 2	1	0	1	1	0	0	0	0
FitzPatrick	0	0	0	1	1	0	0	0
Fort Calhoun	0	0	0	0	2	1	0	0
Fort St. Vrain	0	0	0	NLI**	NLI	NLI	NLI	NLI
Ginna	0	2	1	0	0	3	0	0
Grand Gulf	1	0	0	0	0	0	1	1
Haddam Neck	0	1	0	0	0	0	0	0

Footnotes at end of table.

Table A-1.2 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0	0	0	0	0	1	0	0
Hatch 2	0	0	1	0	1	0	0	0
Hope Creek	0	1	0	0	1	0	0	2
Indian Point 2	0	0	0	0	0	0	0	1
Indian Point 3	1	0	0	0	0	0	0	0
Kewaunee	0	0	0	0	1	0	0	0
Lasalle 1	0	0	0	1	0	0	0	0
Lasalle 2	2	1	0	0	0	0	0	0
Limerick 1	0	0	0	0	0	0	0	1
Limerick 2	NYL	0	0	1	1	0	1	0
Maine Yankee	0	0	0	0	0	1	0	0
McGuire 1	1	0	1	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	0
Millstone 1	0	1	0	0	0	0	0	0
Millstone 2	0	1	0	0	0	0	1	0
Millstone 3	1	0	0	2	1	0	0	0
Monticello	0	0	1	0	0	0	0	0
Nine Mile Pt. 1	1	0	0	0	0	0	3	2
Nine Mile Pt. 2	3	1	0	0	0	0	0	1
North Anna 1	1	1	0	0	0	0	0	0
North Anna 2	0	1	0	0	0	0	1	1
Oconee 1	0	0	0	0	0	1	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	0	0	0	0	0	1
Oyster Creek	0	1	0	0	0	0	0	0
Palisades	0	0	0	2	0	0	0	1
Palo Verde 1	0	0	1	0	0	0	0	0
Palo Verde 2	2	0	2	0	0	0	0	0
Palo Verde 3	1	0	0	0	0	0	0	0
Peach Bottom 2	0	0	0	1	0	1	0	0
Peach Bottom 3	0	0	0	0	1	0	1	1
Perry	0	0	0	0	1	0	0	0
Pilgrim	1	2	1	0	0	0	0	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	1	0	0	1	0	0	0	0
Prairie Island 1	0	0	0	0	0	1	0	0
Prairie Island 2	0	0	0	2	0	0	0	0
Quad Cities 1	0	0	0	0	0	0	0	0
Quad Cities 2	0	0	0	0	1	0	0	0
Rancho Seco	1	0	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.2 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	0	2	0	1	0	0	0	1
Robinson 2	1	0	0	0	0	0	0	0
Salem 1	1	1	0	0	2	0	0	0
Salem 2	1	0	0	1	0	2	1	0
San Onofre 1	0	0	0	0	0	0	1	0
San Onofre 2	0	0	0	1	0	0	0	1
San Onofre 3	1	0	0	0	0	0	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	2	0	0
Sequoyah 2	2	0	0	0	0	0	0	0
Shearon Harris	0	0	0	0	0	1	0	0
South Texas 1	1	0	0	0	0	1	0	1
South Texas 2	3	3	1	0	1	0	0	0
St. Lucie 1	0	0	0	0	0	1	0	0
St. Lucie 2	0	0	0	0	0	0	0	1
Summer	0	0	1	0	0	3	0	0
Surry 1	2	2	0	1	0	1	1	2
Surry 2	1	2	1	0	0	0	0	0
Susquehanna 1	0	0	1	0	0	0	0	1
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	0	1	0	0	1	0
Trojan	0	0	0	0	0	0	1	0
Turkey Point 3	0	2	0	0	0	1	0	0
Turkey Point 4	0	1	1	0	0	0	0	0
Vermont Yankee	2	0	0	0	0	0	0	0
Vogtle 1	0	0	0	0	1	0	0	0
Vogtle 2	1	0	1	0	1	0	0	0
Wash. Nuclear 2	1	2	0	0	0	0	0	1
Waterford 3	1	0	0	2	1	0	1	0
Wolf Creek	0	0	0	0	0	1	0	2
Yankee-Rowe	0	0	1	0	1	1	0	0
Zion 1	0	0	0	0	0	0	0	0
Zion 2	0	0	0	0	0	1	0	0
Total	53	47	18	33	25	32	29	33

*NYL means the plant is not yet licensed.

**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-1.3 Descriptions of Significant Events for 1990

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Beaver Valley 1	12/12/90	W	I	Unexpected thermal stratification of feedwater piping just upstream of SG inlet (AFW flow path) through a long stretch of horizontal piping leading to deformed pipe and dislodged supports.
Braidwood 1*	10/04/90	W	III	Coolant leak outside containment when RHR isolation valve was opened.
Brunswick 1 & 2	05/20/90	GE	II	Twenty-two of 47 licensed operators and 7 of 8 crews failed either the requalification examination or the operational evaluation.
Brunswick 2	08/19/90	GE	II	A technician testing the primary containment isolation logic tripped the next channel before the previously tripped channel was reset. Five SRVs failed to lift after the MSIVs closed.
Catawba 1	06/11/90	W	II	Misalignment of RHR system because of operator error.
Catawba 1	03/20/90	W	II	Inadvertent overpressurization of the RHR system during RCS fill and vent operations.
Clinton	01/24/90	GE	III	Essential service water flow to room coolers for various safety-related systems was set too low by a factor of 2 during preoperational testing.
Clinton	05/14/90	GE	III	After maintenance on service water system, outlet valves for EDG service water coolers were mispositioned. This created a situation where insufficient cooling water was available for Division I and II EDGs.
Dresden 2*	01/16/90	GE	III	Failure of reserve aux transformer (RAT). The RAT failed after condensate pump 2d tripped, pump motor caught fire. Loss of offsite power with multiple equipment failures.
Dresden 2	03/19/90	GE	III	Water hammer in HPCI probably caused by leaking valves.
Dresden 2	08/02/90	GE	III	A safety relief valve spuriously opened and remained stuck open. After a manual scram, there was a rapid cooldown of the system.
Duane Arnold	10/19/90	GE	III	Inadvertent pulling of potential transformer caused "sensed" loss of power on safety bus. Scram and unexpected plant response resulted.
FitzPatrick	04/11/90	GE	I	Many check valves in the emergency service water and interfacing systems were inoperable because of accumulations of silt and corrosion products.

*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
Fort Calhoun	09/13/90	CE	IV	Elevated ambient temperatures cause heat buildup in instrument cabinets, resulting in emergency diesel generator problems.
Ginna	12/12/90	W	I	Use of approved procedure caused loss of automatic and manual (single push button) initiation of an engineered safety features system.
Haddam Neck	08/20/90	W	I	Turbine-driven AFW pump steam admission valves open too fast, causing pump trip on overspeed. Flow controller setting necessary to achieve minimum required flowrate results in trip of AFW pump turbines on overspeed.
Indian Point 3	01/20/90	W	I	Two out of 3 EDGs were inoperable for 56 hours in Mode 1. EDGs are required for accident mitigation. This significant event was not briefed.
Indian Point 3*	10/04/90	W	I	Two non-rod fuel assemblies suspended from upper core support structure upon initial removal of upper internals.
Maine Yankee	06/04/90	CE	I	Loss of reactor coolant inventory while in a shutdown condition.
Maine Yankee	06/07/90	CE	I	Failure of control element assembly — end caps cracked and dislodged.
Millstone 1	05/11/90	GE	I	House heating steam lines passing through vital areas found not to be analyzed by HELB for failure. Seismic event could cause multiple line failures, resulting in loss of multiple trains of safety equipment.
Millstone 1	09/07/90	GE	I	The licensee revised the emergency operation procedures such that design flow through the LPCI heat exchangers would be exceeded, which could lead to tube failures.
Millstone 3*	12/31/90	W	I	Rupture of high-pressure condensate piping from corrosion/erosion.
Monticello	09/11/90	GE	III	Failure of a nonseismic fire suppression pipe could cause loss of both emergency diesel generators.
Nine Mile Pt. 1	11/12/90	GE	I	Phase-to-phase fault on one reserve transformer caused total loss of all offsite power.
Oconee 1, 2, & 3	04/24/90	B&W	II	Licensee discovered design deficiencies in electric power supply system (inadequate undervoltage protection and single failure vulnerability).

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

Table A-13 (cont.)

Plant Name	Event Date	NSSS Vendor	NRC Region	Description of Event
Peach Bottom 2	03/21/90	GE	I	Area coolers for ECCS equipment were not getting desired flow because of silting. It is unknown if the licensee found the problem because of GL 89-13 or what the safety significance is.
Quad Cities 1 & 2	07/07/90	GE	III	Floor drain check valve in all ECCS pump rooms inoperable would allow flood water flow to all redundant ECCS equipment rooms.
River Bend*	02/11/90	GE	IV	Partial ESF actuation occurred when a Topaz inverter reset and reenergized the Rosemount transmitter/trip units. Trip units reenergized before the transmitter, and resulted in opening of the two LPCI injection valves.
Sequoyah 1*	10/08/90	W	II	Three modified Atwood and Morrill main steam check valves failed. Valve disc separated from its hinge pin.
Sequoyah 2	08/22/90	W	II	Gas buildup in the charging system (hydrogen).
Surry 1 & 2	10/23/90	W	II	Degradation of SW cooling function caused a reduced flow to the recirculation spray heat exchangers.
Susquehanna 1	07/24/90	GE	I	A single failure could prevent automatic initiation of cooling to the emergency switchgear rooms during a loss-of-offsite power or a loss-of-coolant accident.
Vogtle 1	03/20/90	W	II	Loss of all ac 1E (vital) power and failure of single available EDG to start and load, subsequent RCS heatup from loss of RHR.
Wolf Creek	12/23/90	W	IV	Both trains of SI technically inoperable from freezing of recirculation to RWST.
Yankee-Rowe	06/23/90	W	I	Stuck control rod during rod drop test.

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

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

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1	0	1	1	0	0	0	0
Arkansas 2	0	2	1	0	0	0	0	0
Beaver Valley 1	0	0	0	0	0	0	0	1
Beaver Valley 2	0	0	0	0	0	0	0	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	1	0	1	0	0	0	1
Braidwood 2	0	1	0	0	0	0	0	0
Browns Ferry 1	2	0	0	0	0	0	0	0
Browns Ferry 2	2	0	0	0	0	0	0	0
Browns Ferry 3	2	0	0	0	0	0	0	0
Brunswick 1	0	0	0	0	0	1	0	0
Brunswick 2	0	1	0	0	0	1	1	0
Byron 1	0	0	0	0	0	0	0	0
Byron 2	1	0	0	0	0	0	0	0
Callaway	0	0	0	0	0	0	0	0
Calvert Cliffs 1	0	0	0	1	0	0	0	0
Calvert Cliffs 2	1	1	0	1	0	0	0	0
Catawba 1	0	0	0	0	1	1	0	0
Catawba 2	0	0	0	0	0	0	0	0
Clinton 1	0	1	0	0	1	1	0	0
Comanche Peak 1	NYL*	NYL	NYL	NYL	0	0	0	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	1	0	0	0	0	0
Cooper Station	0	0	0	0	0	0	0	0
Crystal River 3	1	1	0	1	0	0	0	0
Davis-Besse	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	0	0	0	0
Diablo Canyon 2	1	0	0	0	0	0	0	0
Dresden 2	1	0	0	0	2	0	1	0
Dresden 3	0	0	0	1	0	0	0	0
Duane Arnold	0	0	0	0	0	0	0	1
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	1	0	0
Fort Calhoun	0	0	0	0	0	0	1	0
Fort St. Vrain	0	0	0	NLI	NLI	NLI	NLI	NLI
Ginna	0	0	0	0	0	0	0	1
Grand Gulf	0	0	1	0	0	0	0	0
Haddam Neck	0	0	0	1	0	0	1	0

Footnotes at end of table.

Table A-1.4 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0	0	0	0	0	0	0	0
Hatch 2	0	0	0	0	0	0	0	0
Hope Creek	0	0	0	0	0	0	0	0
Indian Point 2	0	3	0	0	0	0	0	0
Indian Point 3	0	0	0	0	1	0	0	1
Kewaunee	0	0	0	0	0	0	0	0
Lasalle 1	1	0	0	0	0	0	0	0
Lasalle 2	1	0	0	0	0	0	0	0
Limerick 1	1	0	0	0	0	0	0	0
Limerick 2	NYL	0	0	0	0	0	0	0
Maine Yankee	0	0	0	0	0	2	0	0
McGuire 1	1	0	0	0	0	0	0	0
McGuire 2	0	0	1	0	0	0	0	0
Millstone 1	1	1	0	0	0	1	1	0
Millstone 2	0	0	0	0	0	0	0	0
Millstone 3	0	0	0	0	0	0	0	1
Monticello	0	0	0	0	0	0	1	0
Nine Mile Pt. 1	0	0	0	0	0	0	0	1
Nine Mile Pt. 2	2	0	0	0	0	0	0	0
North Anna 1	2	0	0	0	0	0	0	0
North Anna 2	1	0	0	0	0	0	0	0
Oconee 1	1	1	0	0	0	1	0	0
Oconee 2	0	1	0	0	0	1	0	0
Oconee 3	1	1	0	0	0	1	0	0
Oyster Creek	0	0	0	0	0	0	0	0
Palisades	0	0	0	1	0	0	0	0
Palo Verde 1	0	0	0	0	0	0	0	0
Palo Verde 2	0	0	0	0	0	0	0	0
Palo Verde 3	1	0	1	0	0	0	0	0
Peach Bottom 2	0	0	1	0	1	0	0	0
Peach Bottom 3	0	0	0	0	0	0	0	0
Perry	1	0	0	0	0	0	0	0
Pilgrim	1	1	0	0	0	0	0	0
Point Beach 1	0	0	0	1	0	0	0	0
Point Beach 2	0	0	0	1	0	0	0	0
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	2	0	0	0	0	1	0
Quad Cities 2	0	1	0	0	0	0	1	0
Rancho Seco	2	0	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.4 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	0	1	0	1	1	0	0	0
Robinson 2	1	0	1	0	0	0	0	0
Salem 1	0	1	0	0	0	0	0	0
Salem 2	0	0	0	0	0	0	0	0
San Onofre 1	2	0	0	0	0	0	0	0
San Onofre 2	0	0	0	0	0	0	0	0
San Onofre 3	0	0	0	0	0	0	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	0	0	1
Sequoyah 2	0	0	0	0	0	0	1	0
Shearon Harris	0	0	0	0	0	0	0	0
South Texas 1	0	0	0	0	0	0	0	0
South Texas 2	0	0	0	0	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	0	1	1	0	0	0	0	0
Surry 1	3	0	0	0	0	0	0	1
Surry 2	3	0	0	0	0	0	0	1
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	0
Trojan	0	1	1	0	0	0	0	0
Turkey Point 3	1	0	0	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	0	0	1	0	0	0
Vogtle 2	1	0	0	0	0	0	0	0
Wash. Nuclear 2	0	0	0	0	0	0	0	0
Waterford 3	0	0	0	0	0	0	0	0
Wolf Creek	0	0	0	0	0	0	0	1
Yankee-Rowe	0	0	0	0	0	1	0	0
Zion 1	0	0	0	1	0	0	0	0
Zion 2	0	0	0	0	0	0	0	0
Total	41	23	10	12	8	12	10	11

¹NYL means the plant is not yet licensed.

²NLI 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-1.5 Number of Safety System Failures—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	2	0	3	3	2	0	2	2
Arkansas 2	1	0	0	1	1	0	1	0
Beaver Valley 1	0	0	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	2	2
Big Rock Point	1	0	0	0	0	0	2	1
Braidwood 1	0	0	0	0	0	0	2	0
Braidwood 2	1	0	0	1	0	1	1	0
Browns Ferry 1	1	3	3	1	1	0	0	0
Browns Ferry 2	2	4	3	2	1	0	0	0
Browns Ferry 3	1	3	3	1	1	0	0	0
Brunswick 1	2	0	0	1	2	1	0	1
Brunswick 2	2	1	2	0	0	0	3	3
Byron 1	0	0	0	0	0	1	1	0
Byron 2	0	0	0	0	0	0	2	0
Callaway	0	0	0	0	1	0	0	0
Calvert Cliffs 1	2	1	1	2	1	1	1	2
Calvert Cliffs 2	1	2	1	2	1	1	0	1
Catawba 1	1	0	1	0	5	2	1	1
Catawba 2	1	0	2	0	6	0	1	1
Clinton 1	2	1	0	1	3	1	0	2
Comanche Peak 1	NYL*	NYL	NYL	NYL	1	2	2	2
Cook 1	0	0	1	0	0	1	0	1
Cook 2	0	0	0	0	1	1	0	0
Cooper Station	4	2	0	0	1	0	1	0
Crystal River 3	2	2	2	1	1	0	1	0
Davis-Besse	0	1	0	0	0	0	0	0
Diablo Canyon 1	1	0	0	0	1	1	0	0
Diablo Canyon 2	2	0	0	0	1	1	0	0
Dresden 2	2	1	1	1	0	1	1	1
Dresden 3	1	1	0	2	0	1	0	0
Duane Arnold	4	0	1	2	0	0	0	0
Farley 1	0	1	1	1	0	0	0	0
Farley 2	0	2	0	0	0	0	0	0
Fermi 2	3	0	2	1	1	0	2	1
FitzPatrick	4	3	2	4	3	0	0	0
Fort Calhoun	0	2	1	2	1	1	2	0
Fort St. Vrain	1	0	3	NLI	NLI	NLI	NLI	NLI
Ginna	0	0	0	1	0	0	0	1
Grand Gulf	0	2	1	0	1	0	3	1
Haddam Neck	2	4	2	2	3	1	9	4

Footnotes at end of table.

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	1	1	0	0	1	0	1	0
Hatch 2	0	1	0	0	1	1	1	0
Hope Creek	0	3	0	0	0	1	1	2
Indian Point 2	1	0	1	0	0	1	0	0
Indian Point 3	1	1	0	0	0	0	1	3
Kewaunee	1	0	0	0	0	0	1	1
Lasalle 1	1	3	1	0	0	2	1	1
Lasalle 2	1	1	1	2	1	0	0	0
Limerick 1	8	4	3	1	1	4	0	1
Limerick 2	NYL	0	3	2	2	4	1	0
Maine Yankee	0	0	0	0	0	1	0	0
McGuire 1	2	0	6	2	0	6	3	2
McGuire 2	2	0	6	2	1	2	3	2
Millstone 1	2	2	1	2	1	3	1	2
Millstone 2	0	0	1	1	0	1	0	4
Millstone 3	0	0	0	2	0	4	0	0
Monticello	1	3	0	2	0	0	1	1
Nine Mile Pt. 1	0	1	0	1	2	1	1	1
Nine Mile Pt. 2	0	0	0	0	0	0	0	1
North Anna 1	1	1	0	1	0	2	0	2
North Anna 2	0	1	0	0	0	1	0	2
Oconee 1	3	3	1	0	1	3	1	1
Oconee 2	2	3	1	0	1	2	1	1
Oconee 3	3	3	1	0	1	2	1	1
Oyster Creek	3	0	0	0	0	0	0	1
Palisades	1	0	0	0	0	1	2	0
Palo Verde 1	0	1	1	0	0	1	1	0
Palo Verde 2	0	1	1	0	0	1	1	1
Palo Verde 3	0	2	1	0	0	1	1	1
Peach Bottom 2	3	3	1	3	2	0	4	2
Peach Bottom 3	1	0	4	3	1	1	4	0
Perry	3	1	0	1	2	7	4	1
Pilgrim	2	0	3	1	0	0	1	1
Point Beach 1	1	3	0	1	0	2	2	1
Point Beach 2	0	1	0	1	0	1	2	0
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	1	1	0	1	1	1	2	3
Quad Cities 2	0	1	0	0	0	3	2	2
Rancho Seco	4	0	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.5 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	1	1	0	0	1	0	0	2
Robinson 2	0	1	1	2	0	0	0	0
Salem 1	2	2	0	0	0	2	0	0
Salem 2	0	0	0	1	1	3	1	1
San Onofre 1	4	0	2	2	1	1	1	0
San Onofre 2	0	0	0	0	0	0	0	0
San Onofre 3	0	0	0	0	0	0	1	0
Seabrook	1	0	0	1	1	0	0	0
Sequoyah 1	1	1	0	1	0	0	0	1
Sequoyah 2	1	1	0	1	0	0	0	0
Shearon Harris	0	0	1	2	0	2	2	1
South Texas 1	0	1	0	0	0	2	0	0
South Texas 2	1	0	0	0	0	0	0	0
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	0	0	0	0	0	0	0	0
Summer	2	0	1	0	1	1	0	0
Surry 1	2	0	1	0	2	0	1	2
Surry 2	2	0	1	2	1	0	1	1
Susquehanna 1	0	0	1	0	1	0	5	3
Susquehanna 2	1	0	1	0	2	1	3	1
Three Mile Island 1	0	0	0	1	1	0	0	0
Trojan	0	2	2	3	3	7	2	0
Turkey Point 3	1	0	0	2	1	2	0	1
Turkey Point 4	1	0	1	2	2	1	0	1
Vermont Yankee	4	1	2	0	0	0	0	1
Vogtle 1	3	0	0	0	2	0	0	0
Vogtle 2	2	0	0	0	0	0	1	0
Wash. Nuclear 2	1	5	4	2	0	1	4	3
Waterford 3	1	0	0	0	1	0	0	0
Wolf Creek	0	1	2	0	1	0	0	1
Yankee-Rowe	0	0	0	0	0	0	1	0
Zion 1	1	1	0	2	0	1	2	1
Zion 2	0	1	0	1	0	0	2	0
Total	127	99	93	86	81	102	112	88

*NYL means the plant is not yet licensed.

*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-1.6 Safety System Failures

Plant Name	Vendor	1987	1988	1989	1990	Total	Average Yearly Rate
Arkansas 1	BW	0	5	8	6	19	5
Arkansas 2	CE	0	4	2	2	8	2
Beaver Valley 1	WE	1	0	0	0	1	0
Beaver Valley 2	WE	2	0	0	4	6	2
Big Rock Point	GE	1	0	1	3	5	1
Braidwood 1	WE	5	0	0	2	7	2
Braidwood 2	WE	0	1	2	2	5	1
Browns Ferry 1	GE	4	5	8	1	18	5
Browns Ferry 2	GE	5	5	11	1	22	6
Browns Ferry 3	GE	5	6	8	1	20	5
Brunswick 1	GE	5	16	3	4	28	7
Brunswick 2	GE	7	11	5	5	28	7
Byron 1	WE	5	0	0	2	7	2
Byron 2	WE	3	0	0	2	5	1
Callaway	WE	7	2	0	1	10	3
Calvert Cliffs 1	CE	3	0	6	5	14	4
Calvert Cliffs 2	CE	1	0	6	3	10	3
Catawba 1	WE	3	4	2	9	18	5
Catawba 2	WE	2	3	3	9	17	4
Clinton 1	GE	4	7	4	6	21	5
Comanche Peak 1*	WE	NYL	NYL	NYL	7	7	7
Cook 1	WE	1	2	1	2	6	2
Cook 2	WE	1	2	0	2	5	1
Cooper Station	GE	2	1	6	2	11	3
Crystal River 3	BW	3	1	7	2	13	3
Davis-Besse	BW	1	3	1	0	5	1
Diablo Canyon 1	WE	1	1	1	2	5	1
Diablo Canyon 2	WE	4	1	2	2	9	2
Dresden 2	GE	4	6	5	3	18	5
Dresden 3	GE	8	0	4	1	13	3
Duane Arnold	GE	4	4	7	0	15	4
Farley 1	WE	2	1	3	0	6	2
Farley 2	WE	0	2	2	0	4	1
Fermi 2	GE	9	8	6	4	27	7
FitzPatrick	GE	4	6	13	3	26	7
Fort Calhoun	CE	5	5	5	4	19	5

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Vendor	1987	1988	1989	1990	Total	Average Yearly Rate
Fort St. Vrain ¹	GA	2	1	3	PSD	6	2
Ginna	WE	4	0	1	1	6	2
Grand Gulf	GE	1	2	3	5	11	3
Haddam Neck	WE	1	6	10	17	34	9
Hatch 1	GE	2	7	2	2	13	3
Hatch 2	GE	5	7	1	3	16	4
Hope Creek	GE	4	7	3	4	18	5
Indian Point 2	WE	4	2	2	1	9	2
Indian Point 3	WE	2	1	2	4	9	2
Kewaunee	WE	0	1	1	2	4	1
La Crosse	AC	0	PSD	PSD	PSD	0	0
LaSalle 1	GE	4	2	5	4	15	4
LaSalle 2	GE	2	4	5	1	12	3
Limerick 1	GE	6	4	16	6	32	8
Limerick 2 [*]	GE	NYL	NYL	5	7	12	6
Maine Yankee	CE	1	2	0	1	4	1
McGuire 1	WE	7	7	10	11	35	9
McGuire 2	WE	5	3	10	8	26	7
Millstone 1	GE	5	4	7	7	23	6
Millstone 2	CE	0	1	2	5	8	2
Millstone 3	WE	2	4	2	4	12	3
Monticello	GE	2	0	6	2	10	3
Nine Mile Pt. 1	GE	3	4	2	5	14	4
Nine Mile Pt. 2	GE	9	8	0	1	18	5
North Anna 1	WE	4	2	3	4	13	3
North Anna 2	WE	5	2	1	3	11	3
Oconee 1	BW	4	2	7	6	19	5
Oconee 2	BW	3	3	6	5	17	4
Oconee 3	BW	5	4	7	5	21	5
Oyster Creek	GE	8	6	3	1	18	5
Palisades	CE	6	4	1	3	14	4
Palo Verde 1	CE	1	5	2	2	10	3
Palo Verde 2	CE	1	4	2	3	10	3
Palo Verde 3	CE	1	3	3	3	10	3
Peach Bottom 2	GE	7	3	10	8	28	7
Peach Bottom 3	GE	3	2	8	6	19	5

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Vendor	1987	1988	1989	1990	Total	Average Yearly Rate
Perry	GE	20	11	5	14	50	13
Pilgrim	GE	4	2	6	2	14	4
Point Beach 1	WE	0	4	5	5	14	4
Point Beach 2	WE	1	3	2	3	9	2
Prairie Island 1	WE	2	1	0	0	3	1
Prairie Island 2	WE	0	2	0	0	2	1
Quad Cities 1	GE	8	4	3	7	22	6
Quad Cities 2	GE	2	5	1	7	15	4
Rancho Seco ^{II}	BW	10	1	4	NLO	15	5
River Bend	GE	4	1	2	3	10	3
Robinson 2	WE	13	6	4	0	23	6
Salem 1	WE	4	3	4	2	13	3
Salem 2	WE	5	5	1	6	17	4
San Onofre 1	WE	1	3	8	3	15	4
San Onofre 2	CE	0	5	0	0	5	1
San Onofre 3	CE	0	5	0	1	6	2
Seabrook*	WE	2	1	2	1	6	2
Sequoyah 1	WE	15	4	3	1	23	6
Sequoyah 2	WE	13	7	3	0	23	6
Shearon Harris	WE	5	8	3	5	21	5
Shoreham	GE	0	PSD	PSD	PSD	0	0
South Texas 1	WE	4	11	1	2	18	5
South Texas 2*	WE	NYL	0	1	0	1	0
St. Lucie 1	CE	4	0	0	0	4	1
St. Lucie 2	CE	0	0	0	0	0	0
Summer	WE	4	1	3	2	10	3
Surry 1	WE	6	6	3	5	20	5
Surry 2	WE	0	6	5	3	14	4
Susquehanna 1	GE	4	3	1	7	15	4
Susquehanna 2	GE	4	3	2	7	16	4
Three Mile Island 1	BW	0	0	1	1	2	1
Trojan	WE	5	5	7	12	29	7
Turkey Point 3	WE	6	6	3	4	19	5
Turkey Point 4	WE	8	6	4	4	22	6
Vermont Yankee	GE	2	1	7	1	11	3
Vogtle 1	WE	6	1	3	3	13	3

Footnotes at end of table.

Table A-1.6 (cont.)

Plant Name	Vendor	1987	1988	1989	1990	Total	Average Yearly Rate
Vogtle 2*	WE	NYL	NYL	2	2	4	2
Wash. Nuclear 2	GE	0	4	12	8	24	6
Waterford 3	CE	0	0	1	1	2	1
Wolf Creek	WE	5	1	3	2	11	3
Yankee-Rowe	WE	2	4	0	1	7	2
Zion 1	WE	2	3	4	4	13	3
Zion 2	WE	0	1	2	2	5	1
Total All Plants ⁱⁱⁱ		402	372	404	383	1561	
Total All WE Plants		180	145	134	173	632	
Total All GE Plants		171	169	196	152	688	
Total All CE Plants		23	38	30	33	124	
Total All BW Plants		26	19	41	25	111	

*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.

PSD Permanently shutdown plant.

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

ⁱⁱRancho Seco ceased commercial operations on June 6, 1989.

ⁱⁱⁱThis total includes Fort St. Vrain.

Table A-1.7 Safety System Failure—Group Counts By Year

	1987	1988	1989	1990	Total
Accident Monitoring Instrumentation					
Plant Protection System	0	0	0	1	1
Engineered Safety Features Actuation System	0	0	0	1	1
Post-Accident Monitoring System	0	2	2	1	5
Incore/Excore Neutron Monitoring System	0	2	0	0	2
Containment Environmental Monitoring System	0	0	0	2	2
Total	0	4	2	5	11
Auxiliary/Emergency Feedwater Systems Group					
Auxiliary/Emergency Feedwater System	9	5	10	14	38
Total	9	5	10	14	38
Combustible Gas Control Systems Group					
Containment Combustible Gas Control System	5	4	1	2	12
Emergency/Standby Gas Treatment System	4	13	11	7	35
Containment Purge System	1	0	4	3	8
Total	10	17	16	12	55
Component Cooling Water System Group					
Closed/Component Cooling Water System	9	7	4	5	25
Total	9	7	4	5	25
Containment and Containment Isolation Group					
Containment Isolation Control System	7	6	5	2	20
Containment Leakage Control System	0	0	1	3	4
Containment Vacuum Relief System	1	0	1	0	2
Reactor Containment Building	12	4	2	18	36
Primary Containment/Undetermined System	11	14	10	15	50
Secondary Containment/Undetermined System	7	12	10	0	29
Reactor Building (BWR)	2	1	12	10	25
Total	40	37	41	48	166
Containment Cooling Systems Group					
Reactor Building Environmental Control System	3	9	14	1	27
Drywell Environmental Control System	1	0	0	0	1
Shield Annulus Return and Exhaust System	0	1	6	11	18
Containment Ice Condenser/Refrigeration System	0	0	0	4	4
Containment Spray System	8	6	10	10	34
Containment Fan Cooling System	7	4	0	5	16
Total	19	20	30	31	100
Control Room Emergency Ventilation System Group					
Control Building/Control Complex Environmental Control System	43	33	28	29	133
Total	43	33	28	29	133

Table A-1.7 (cont.)

	1987	1988	1989	1990	Total
Emergency AC/DC Power Systems Group					
Diesel Cooling Water System	0	0	1	1	2
Diesel Generator Starting Air System	0	2	1	0	3
Medium-Voltage Power System - Class 1E	0	9	12	14	35
Low-Voltage Power System - Class 1E	7	3	5	5	20
Instrument and Uninterruptible Power System - Class 1E	8	4	1	1	14
DC Power System - Class 1E	5	8	11	6	30
Emergency Onsite Power Supply System	34	31	23	27	115
Emergency Onsite Power Supply Building Environmental Control System	0	3	3	0	6
Total	54	60	57	54	225
Emergency Core Cooling Systems Group					
High Pressure Coolant Injection System	38	30	40	37	145
High Pressure Core Spray System	1	5	14	9	29
Low Pressure Coolant Injection System	1	3	1	3	8
Low Pressure Core Spray System	2	2	7	0	11
Low Pressure Safety Injection System	11	10	8	3	32
High Pressure Safety Injection System	16	17	15	25	73
Upper Head Injection	1	1	0	0	2
Intermediate Head Injection	0	0	0	4	4
Total	70	68	85	81	304
Engineered Safety Features Instrumentation					
Engineered Safety Features Actuation System	11	14	4	10	39
Radiation Monitoring System	0	1	1	0	2
Integrated Control System	0	1	1	0	2
Feedwater/Steam Generator Water Level Control System	0	1	0	0	1
Reactor Power Control System	1	0	0	0	1
Solid State Control System/Auxiliary Logic Control System	0	1	1	0	2
Containment Environmental Monitoring System	0	0	1	0	1
Anticipated Transient Without Scram System	0	0	2	0	2
Total	12	18	10	10	50
Essential Compressed Air System Group					
Essential Air System	4	1	3	0	8
Total	4	1	3	0	8
Essential Service Water System Group					
Essential Service Water System	14	18	31	26	89
Total	14	18	31	26	89
Fire Detection/Suppression Systems Group					
Fire Detection System	2	0	1	0	3
Fire Protection System (water)	5	3	1	0	9
Fire Protection System (chemical)	4	2	0	0	6
Fire Protection System (passive)	1	0	0	0	1
Total	12	5	2	0	19

Table A-1.7 (cont.)

	1987	1988	1989	1990	Total
Isolation Condenser System Group					
Isolation Condenser System	1	2	0	0	3
Total	1	2	0	0	3
Low Temperature/Overpressure Protection Group					
Low Temperature/Overpressure System	1	4	3	4	12
Total	1	4	3	4	12
Main Steam Isolation Valves Group					
Main Steam Isolation Valves	13	11	7	4	35
Total	13	11	7	4	35
Primary Reactor Systems Group					
Control Rod Drive System	1	0	2	1	4
Reactor Coolant System	2	3	0	4	9
Reactor Recirculation System	2	1	0	1	4
Reactor Core System	0	0	0	3	3
Reactor Vessel System	0	1	1	0	2
Pressurizer System	0	0	0	1	1
Steam Generating System	1	0	4	0	5
Total	6	5	7	10	28
Radiation Monitoring Instrumentation					
Radiation Monitoring System	1	3	4	6	14
Incore/Excore Neutron Monitoring System	0	0	0	1	1
Leak Monitoring System	0	3	0	0	3
Containment Environmental Monitoring System	0	0	1	0	1
Total	1	6	5	7	19
Reactor Core Isolation Cooling Systems Group					
Reactor Core Isolation Cooling System	26	13	17	12	68
Total	26	13	17	12	68
Reactor Trip Instrumentation					
Plant Protection System	12	3	2	5	22
Feedwater/Steam Generator Water Level Control System	1	0	0	0	1
Reactor Power Control System	0	1	0	0	1
Incore/Excore Neutron Monitoring System	4	1	2	0	7
Total	17	5	4	5	31
Residual Heat Removal Systems Group					
Residual Heat Removal System	5	5	6	2	18
Residual Heat Removal System	16	11	19	2	48
Total	21	16	25	4	66

Table A-1.7 (cont.)

	1987	1988	1989	1990	Total
Safety and Relief Valves Group					
Reactor Coolant System	8	3	1	1	13
Main/Reheat Steam System	8	3	3	6	20
Automatic Depressurization System	0	3	5	1	9
Total	16	9	9	8	42
Spent Fuel Systems Group					
Fuel Pool Cooling and Purification System	1	2	0	2	5
Fuel Building Environmental Control System	0	1	2	5	8
Total	1	3	2	7	13
Standby Liquid Control System Group					
Standby Liquid Control System	2	3	1	1	7
Total	2	3	1	1	7
Ultimate Heat Sink System Group					
Ultimate Heat Sink System	1	2	5	6	14
Total	1	2	5	6	14

Table A-1.8 Forced Outage Rate (Percentage)—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	77	21	1	7	5	0	0	0
Arkansas 2	8	29	3	0	10	2	3	0
Beaver Valley 1	2	4	0	82	2	5	1	8
Beaver Valley 2	8	41	14	0	0	0	7	0
Big Rock Point	0	0	3	8	0	0	21	1
Braidwood 1	4	4	2	0	2	1	13	28
Braidwood 2	0	2	4	0	0	18	0	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	26	7	0	0	25	0	0
Brunswick 2	0	12	0	0	0	25	23	9
Byron 1	1	0	0	0	0	6	2	2
Byron 2	0	8	0	8	1	0	0	4
Callaway	0	3	2	0	0	5	0	14
Calvert Cliffs 1	4	0	0	0	0	15	0	0
Calvert Cliffs 2	11	0	0	0	0	0	0	0
Catawba 1	9	6	1	12	0	29	0	4
Catawba 2	9	18	1	0	3	0	0	2
Clinton 1	0	85	21	0	16	25	19	0
Comanche Peak 1	NYCO*	NYCO	NYCO	NYCO	NYCO	NYCO	17	2
Cook 1	1	0	4	0	0	0	0	0
Cook 2	0	0	5	0	17	3	0	13
Cooper Station	12	0	3	6	0	0	0	14
Crystal River 3	0	59	45	13	4	8	0	8
Davis-Besse	6	1	0	0	18	100	0	3
Diablo Canyon 1	0	0	0	1	2	6	0	7
Diablo Canyon 2	0	4	9	11	0	1	0	0
Dresden 2	9	0	2	4	14	0	7	0
Dresden 3	7	4	0	0	26	0	0	3
Duane Arnold	23	5	3	45	3	1	41	3
Farley 1	0	0	0	6	0	0	4	0
Farley 2	0	13	7	3	0	5	0	0
Fermi 2	28	0	2	20	0	6	0	0
FitzPatrick	0	0	0	14	9	0	2	7
Fort Calhoun	0	9	5	0	0	0	13	26
Fort St. Vrain	100	29	48	NLI	NLI	NLI	NLI	NLI
Ginna	1	12	14	0	0	4	3	11
Grand Gulf	0	0	11	3	0	0	10	49
Haddam Neck	0	0	0	0	0	0	25	21

Footnotes at end of table.

Table A-1.8 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0	0	0	0	0	42	0	5
Hatch 2	0	0	1	5	5	0	0	0
Hope Creek	0	0	3	3	10	0	0	16
Indian Point 2	1	0	0	2	0	0	4	0
Indian Point 3	0	8	0	5	0	2	2	13
Kewaunee	0	0	0	1	0	0	0	0
Lasalle 1	4	0	0	0	7	0	0	0
Lasalle 2	0	0	34	15	3	18	10	6
Limerick 1	0	0	0	0	0	12	0	34
Limerick 2	NYCO	NYCO	NYCO	NYCO	8	5	27	4
Maine Yankee	11	1	0	12	0	0	1	31
McGuire 1	27	43	2	0	48	1	4	35
McGuire 2	2	1	0	0	0	0	1	47
Millstone 1	0	11	0	5	0	1	5	16
Millstone 2	0	0	0	0	0	5	2	5
Millstone 3	13	15	0	10	6	37	0	0
Monticello	0	5	0	2	0	0	6	4
Nine Mile Pt. 1	100	100	100	100	100	100	53	8
Nine Mile Pt. 2	0	12	1	49	42	15	4	0
North Anna 1	10	0	2	17	2	0	0	0
North Anna 2	0	0	0	0	0	0	18	1
Oconee 1	11	0	1	6	0	0	1	0
Oconee 2	2	7	0	10	0	0	0	14
Oconee 3	4	0	0	1	2	0	0	1
Oyster Creek	85	55	24	9	15	22	4	0
Palisades	34	0	3	69	5	14	0	0
Palo Verde 1	30	100	0	0	0	0	18	0
Palo Verde 2	17	0	17	41	0	0	0	0
Palo Verde 3	31	0	0	97	22	7	2	2
Peach Bottom 2	0	6	8	18	4	31	10	6
Peach Bottom 3	0	0	0	2	11	0	7	0
Perry	2	0	1	0	4	0	0	0
Pilgrim	29	42	17	7	0	6	33	0
Point Beach 1	0	0	0	0	0	0	0	0
Point Beach 2	3	2	1	0	0	0	0	0
Prairie Island 1	0	0	1	0	3	0	0	1
Prairie Island 2	0	1	0	8	14	5	0	1
Quad Cities 1	0	13	6	3	4	0	1	0
Quad Cities 2	3	4	5	3	0	0	0	5
Rancho Seco	64	9	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.8 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	9	77	6	2	0	2	0	16
Robinson 2	7	14	43	91	0	5	0	0
Salem 1	12	71	0	12	5	72	55	0
Salem 2	23	14	0	0	8	45	11	0
San Onofre 1	0	87	25	3	0	9	0	0
San Onofre 2	34	28	0	23	0	0	5	10
San Onofre 3	3	12	9	0	9	0	0	0
Seabrook	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	3	20
Sequoyah 1	3	0	0	3	0	10	5	15
Sequoyah 2	0	16	5	0	0	2	0	6
Shearon Harris	6	0	0	2	0	7	0	11
South Texas 1	13	0	8	15	6	24	13	51
South Texas 2	NYCO	0	17	34	29	30	11	0
St. Lucie 1	0	0	1	1	2	10	40	1
St. Lucie 2	0	2	5	0	13	0	20	5
Summer	28	19	20	8	0	0	0	0
Surry 1	100	100	8	1	0	12	2	0
Surry 2	0	0	34	47	0	7	2	29
Susquehanna 1	22	0	4	2	12	0	0	4
Susquehanna 2	7	0	0	0	8	13	0	0
Three Mile Island 1	0	0	0	2	31	0	0	0
Trojan	0	0	13	0	1	0	13	6
Turkey Point 3	45	0	1	0	0	33	0	0
Turkey Point 4	0	7	24	24	0	20	23	0
Vermont Yankee	0	0	0	0	4	2	0	2
Vogtle 1	12	3	2	5	2	2	4	3
Vogtle 2	NYCO	4	1	5	2	4	0	0
Wash. Nuclear 2	4	0	11	0	0	0	9	13
Waterford 3	3	0	6	2	3	0	3	0
Wolf Creek	2	0	0	0	3	6	0	0
Yankee-Rowe	1	4	14	0	0	0	0	5
Zion 1	31	0	9	0	48	81	4	36
Zion 2	21	1	0	3	27	0	72	39
Total	1489	1589	1010	1328	960	1309	1019	1066

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-1.9 Equipment Forced Outages/1000 Commercial Critical Hours--Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1.92	0.57	0.91	0.00	1.44	0.00	0.00	0.00
Arkansas 2	0.50	1.29	0.00	0.95	0.51	0.93	1.03	0.00
Beaver Valley 1	0.47	0.00	0.00	6.72	0.96	0.00	0.52	0.97
Beaver Valley 2	0.59	1.99	0.00	0.00	0.00	0.00	0.00	0.00
Big Rock Point	0.00	0.00	0.87	0.49	0.00	0.00	1.25	1.23
Braidwood 1	1.21	0.51	0.00	0.00	0.00	0.00	1.04	1.86
Braidwood 2	0.00	0.46	0.00	0.00	0.00	0.00	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	0.66	0.00	0.00	0.00	0.00	0.00	0.00
Brunswick 2	0.00	0.52	0.00	0.00	0.00	0.00	1.12	0.48
Byron 1	0.47	0.00	0.00	0.00	0.00	0.99	0.46	0.00
Byron 2	0.00	0.49	0.00	0.49	0.48	0.00	0.00	1.02
Callaway	0.00	1.08	0.00	0.00	0.00	0.48	0.00	2.64
Calvert Cliffs 1	1.37	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Calvert Cliffs 2	0.57	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Catawba 1	3.10	0.99	0.46	1.52	0.00	2.91	0.00	0.47
Catawba 2	3.28	0.00	0.45	0.00	0.47	0.00	0.00	0.46
Clinton 1	0.00	4.69	1.10	0.00	0.96	0.61	1.09	0.00
Comanche Peak 1	NYCO*	NYCO	NYCO	NYCO	NYCO	NYCO	4.97	0.52
Cook 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cook 2	0.00	0.00	0.47	0.00	0.00	0.48	0.00	1.71
Cooper Station	0.52	0.00	0.00	0.48	0.00	0.00	0.00	0.00
Crystal River 3	0.00	3.58	0.00	0.00	0.00	4.26	0.00	0.00
Davis-Besse	0.51	0.46	0.00	0.00	0.00	0.00	0.00	0.46
Diablo Canyon 1	0.00	0.00	0.00	0.00	0.47	0.47	0.00	0.97
Diablo Canyon 2	0.00	0.51	0.98	0.50	0.00	0.66	0.00	0.00
Dresden 2	0.00	0.00	0.46	0.00	0.53	0.00	0.53	0.00
Dresden 3	1.47	1.29	0.00	0.00	2.00	0.00	0.00	0.46
Duane Arnold	1.72	0.95	0.00	0.77	0.00	0.47	5.07	0.00
Farley 1	0.00	0.00	0.00	0.00	0.00	0.00	0.47	0.00
Farley 2	0.00	0.00	0.00	0.92	0.00	0.54	0.00	0.00
Fermi 2	0.53	0.00	0.00	0.00	0.00	0.49	0.00	0.00
FitzPatrick	0.00	0.00	0.00	1.06	0.49	0.00	0.48	0.00
Fort Calhoun	0.00	0.00	0.47	0.00	0.00	0.00	0.51	0.60
Fort St. Vrain	0.00	1.52	0.86	NLI	NLI	NLI	NLI	NLI
Ginna	0.55	0.00	0.52	0.00	0.00	1.55	0.00	1.00
Grand Gulf	0.00	0.00	0.50	0.46	0.00	0.00	1.01	3.41
Haddam Neck	0.00	0.00	0.00	0.00	0.00	0.00	2.84	0.00

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.48
Hatch 2	0.00	0.00	0.64	1.68	0.48	0.00	0.00	0.00
Hope Creek	0.57	0.00	0.56	0.93	0.00	0.00	0.00	1.12
Indian Point 2	0.55	0.00	0.00	0.61	0.00	0.00	0.47	0.00
Indian Point 3	0.00	4.64	0.00	0.00	0.00	0.00	0.56	3.72
Kewaunee	0.00	0.00	0.00	0.46	0.00	0.00	0.00	0.00
Lasalle 1	0.48	0.00	0.00	0.00	0.50	0.48	0.00	0.00
Lasalle 2	0.00	0.00	0.00	0.00	0.00	4.40	0.50	0.47
Limerick 1	0.00	0.00	0.00	0.00	0.00	0.59	0.00	4.06
Limerick 2	NYCO	NYCO	NYCO	NYCO	1.06	0.52	1.29	0.91
Maine Yankee	1.02	0.00	0.00	0.54	0.00	0.00	0.91	2.42
McGuire 1	0.63	0.80	0.46	0.00	11.22	1.93	0.93	2.75
McGuire 2	0.93	0.46	0.00	0.00	0.00	0.00	0.68	8.53
Millstone 1	0.00	1.12	0.00	0.47	0.00	0.54	0.00	0.00
Millstone 2	0.00	0.00	0.00	0.00	0.00	0.77	0.00	0.77
Millstone 3	0.53	3.40	0.52	1.00	0.98	2.06	0.00	0.45
Monticello	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Nine Mile Pt. 1	0.00	0.00	0.00	0.00	0.00	0.00	0.77	0.48
Nine Mile Pt. 2	0.00	0.99	0.54	1.52	0.00	0.53	0.63	0.00
North Anna 1	0.75	0.00	0.54	0.54	0.47	0.00	0.00	0.00
North Anna 2	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Oconee 1	1.84	0.00	0.00	0.48	0.00	0.00	0.91	0.00
Oconee 2	0.93	2.62	0.00	1.54	0.00	0.00	0.00	0.72
Oconee 3	0.96	0.00	0.45	1.66	0.94	0.00	0.00	0.45
Oyster Creek	15.75	0.94	1.09	0.00	0.56	1.19	0.00	0.00
Palisades	0.69	0.00	0.46	3.69	0.00	0.82	0.00	0.00
Palo Verde 1	0.66	0.00	0.00	0.00	0.00	0.00	1.08	0.00
Palo Verde 2	0.68	0.00	1.83	2.82	0.00	0.00	0.00	0.00
Palo Verde 3	1.81	0.00	0.00	9.67	0.00	0.49	0.46	0.46
Peach Bottom 2	0.00	1.47	0.96	1.06	0.71	0.62	0.99	0.00
Peach Bottom 3	0.00	0.00	0.00	1.25	1.00	0.00	0.48	0.00
Perry	0.80	0.00	0.65	0.00	0.00	0.00	0.00	0.00
Pilgrim	0.00	0.00	0.54	0.00	0.00	0.63	0.58	0.00
Point Beach 1	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Point Beach 2	0.47	0.00	0.50	0.00	0.00	0.00	0.00	0.00
Prairie Island 1	0.00	0.00	0.00	0.00	0.79	0.00	0.00	0.46
Prairie Island 2	0.00	0.66	0.00	0.98	0.53	0.00	0.00	0.49
Quad Cities 1	0.00	2.03	1.22	1.17	0.00	0.00	0.48	0.00
Quad Cities 2	0.48	1.41	0.49	0.00	0.00	0.00	0.00	0.48
Rancho Seco	3.32	0.00	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.9 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	1.81	14.42	0.95	0.48	0.54	0.46	0.00	0.00
Robinson 2	1.13	0.53	0.00	0.00	0.00	0.54	0.00	0.00
Salem 1	1.07	8.97	0.00	0.00	0.49	2.77	3.74	0.00
Salem 2	1.68	1.55	0.00	0.58	0.50	13.76	1.00	0.00
San Onofre 1	0.00	4.81	1.78	0.59	0.00	0.50	0.00	0.00
San Onofre 2	0.69	0.63	0.00	1.48	0.00	0.00	0.00	1.00
San Onofre 3	0.47	1.04	0.00	0.00	0.51	0.00	0.00	0.00
Seabrook	NYCO	NYCO	NYCO	NYCO	NYCO	NYCO	0.00	1.61
Sequoyah 1	0.00	0.00	0.00	0.46	0.00	2.78	0.93	0.53
Sequoyah 2	0.00	0.59	0.47	0.00	0.00	0.00	0.00	0.00
Shearon Harris	1.44	0.00	0.00	4.05	0.00	0.52	0.00	0.64
South Texas 1	1.77	0.00	1.28	0.60	0.95	5.56	1.02	0.90
South Texas 2	NYCO	0.00	2.61	2.59	1.24	1.65	0.51	0.00
St. Lucie 1	0.00	0.00	0.00	0.45	1.97	1.96	1.48	0.00
St. Lucie 2	0.00	0.64	0.47	0.00	0.52	0.00	1.63	1.34
Summer	0.00	0.56	1.62	0.97	0.00	0.00	0.00	0.00
Surry 1	0.00	0.00	0.00	0.46	0.00	0.52	0.46	0.00
Surry 2	0.00	0.00	0.00	0.85	0.00	0.98	0.46	0.63
Susquehanna 1	1.16	0.00	0.00	0.00	0.51	0.00	0.00	0.96
Susquehanna 2	0.50	0.00	0.00	0.00	0.50	0.56	0.00	0.00
Three Mile Island 1	0.00	0.00	0.00	0.46	1.77	0.00	0.00	0.00
Trojan	0.00	0.00	1.00	0.00	0.00	0.00	1.08	0.00
Turkey Point 3	0.81	0.00	0.45	0.00	0.00	3.86	0.00	0.00
Turkey Point 4	0.00	1.78	2.31	1.62	0.00	1.10	1.16	0.00
Vermont Yankee	0.00	0.00	0.00	0.00	0.49	0.46	0.00	0.54
Vogtle 1	1.03	0.46	0.92	0.47	0.00	0.00	0.47	0.50
Vogtle 2	NYCO	1.05	0.45	1.46	0.47	0.94	0.00	0.00
Wash. Nuclear 2	0.49	0.00	1.08	0.00	0.00	0.00	0.77	1.02
Waterford 3	0.48	0.00	1.05	0.96	1.11	0.00	0.92	0.00
Wolf Creek	0.95	0.00	0.00	0.00	0.63	1.82	0.00	0.00
Yankee-Rowe	0.53	0.94	0.52	0.00	0.00	0.00	0.00	0.81
Zion 1	1.31	0.00	1.28	0.00	0.93	0.00	0.00	1.40
Zion 2	1.15	0.46	0.00	0.45	0.71	0.00	5.92	1.46

*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-1.10 Collective Radiation Exposure (Man-REM)—Quarterly PI Data*

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	64	33	30	228	16	16	13	336
Arkansas 2	64	33	30	228	16	16	13	336
Beaver Valley 1	59	133	198	299	8	8	104	49
Beaver Valley 2	59	133	198	299	8	8	104	49
Big Rock Point	16	59	60	13	9	13	60	142
Braidwood 1	50	8	65	173	29	56	5	4
Braidwood 2					29	56	5	4
Browns Ferry 1	53	35	38	82	86	156	123	61
Browns Ferry 2	53	35	38	82	86	156	123	61
Browns Ferry 3	53	35	38	82	86	156	123	61
Brunswick 1	258	30	104	471	188	32	52	499
Brunswick 2	258	30	104	471	188	32	52	499
Byron 1	66	4	7	10	136	5	39	36
Byron 2	66	4	7	10	136	5	39	36
Callaway	6	259	9	8	7	7	111	317
Calvert Cliffs 1	20	68	47	38	34	41	38	39
Calvert Cliffs 2	20	68	47	38	34	41	38	39
Catawba 1	79	72	6	10	256	54	86	10
Catawba 2	79	72	6	10	256	54	86	10
Clinton 1	260	81	14	18	85	26	51	408
Comanche Peak 1 ^{***}	NA	NA	NA	NA	NA	NA	NA	NA
Cook 1	95	138	10	10	15	6	121	177
Cook 2	95	138	10	10	15	6	121	177
Cooper Station	28	274	19	21	157	184	16	20
Crystal River 3	130	70	8	10	49	395	13	18
Davis-Besse	7	11	9	10	251	223	6	9
Diablo Canyon 1	3	4	4	207	104	37	4	4
Diablo Canyon 2	3	4	4	207	104	37	4	4
Dresden 2	370	46	43	105	168	46	76	410
Dresden 3	370	46	43	105	168	46	76	410
Duane Arnold	45	28	46	63	36	58	718	47
Farley 1	34	127	44	169	8	9	6	205
Farley 2	34	127	44	169	8	9	6	205
Fermi 2	11	15	66	142	21	28	25	10
FitzPatrick	58	52	178	89	101	664	63	56
Fort Calhoun	48	16	19	10	99	139	15	25
Fort St. Vrain	1	1	NLI ^{***}	NLI	NLI	NLI	NLI	NLI
Ginna	124	440	24	20	81	241	14	11
Grand Gulf	143	312	25	18	16	14	39	414
Haddam Neck	19	14	313	251	119	200	57	46

Footnotes at end of table.

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	57	41	100	136	315	358	43	39
Hatch 2	57	41	100	136	315	358	43	39
Hope Creek	110	21	96	238	49	32	24	103
Indian Point 2	220	1046	29	142	232	319	28	25
Indian Point 3	454	403	10	9	64	20	67	177
Kewaunee	208	26	4	2	104	36	2	3
Lasalle 1	178	62	94	360	117	259	43	55
Lasalle 2	178	62	94	360	117	259	43	55
Limerick 1	162	56	17	29	12	13	13	48
Limerick 2	NA	NA	NA	NA	NA	NA	NA	NA
Maine Yankee	21	11	19	38	28	563	29	62
McGuire 1	30	49	222	9	152	66	59	88
McGuire 2	30	49	222	9	152	66	59	88
Millstone 1	41	377	17	24	23	29	16	30
Millstone 2	470	176	27	241	6	158	133	275
Millstone 3	8	146	7	7	2	6	2	1
Monticello	21	27	273	186	25	13	32	23
Nine Mile Pt. 1	56	81	92	33	60	32	79	183
Nine Mile Pt. 2	56	81	92	33	60	32	79	183
North Anna 1	174	511	28	24	16	11	138	136
North Anna 2	174	511	28	24	16	11	138	136
Oconee 1	62	62	14	90	12	40	40	43
Oconee 2	62	62	14	90	12	40	40	43
Oconee 3	62	62	14	90	12	40	40	43
Oyster Creek	569	149	82	111	126	74	48	61
Palisades	57	16	18	208	24	132	105	426
Palo Verde 1	25	87	87	24	46	94	10	8
Palo Verde 2	25	87	87	24	46	94	10	8
Palo Verde 3	25	87	87	24	46	94	10	8
Peach Bottom 2	58	65	152	67	54	44	33	71
Peach Bottom 3	58	65	152	67	54	44	33	71
Perry	258	425	64	33	29	48	163	392
Pilgrim	49	54	45	61	65	63	50	37
Point Beach 1	9	83	11	134	8	69	14	86
Point Beach 2	9	83	11	134	8	69	14	86
Prairie Island 1	6	34	3	4	53	1	38	2
Prairie Island 2	6	34	3	4	53	1	38	2
Quad Cities 1	39	33	100	278	280	80	29	124
Quad Cities 2	39	33	100	278	280	80	29	124
Rancho Seco	34	12	NLI	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.10 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	106	376	21	25	42	15	29	368
Robinson 2	116	31	31	36	15	13	64	345
Salem 1	7	144	4	15	6	111	9	5
Salem 2	7	144	4	15	6	111	9	5
San Onofre 1	77	33	45	50	4	112	133	47
San Onofre 2	77	33	45	50	4	112	133	47
San Onofre 3	77	33	45	50	4	112	133	47
Seabrook	NA	NA	NA	NA	NA	NA	NA	NA
Sequoyah 1	280	16	12	22	111	361	197	188
Sequoyah 2	280	16	12	22	111	361	197	188
Shearon Harris	6	4	5	140	8	30	12	34
South Texas 1					1	67	4	134
South Texas 2	NA	NA	NA	NA	NA	NA	NA	NA
St. Lucie 1	144	21	58	9	227	39	30	70
St. Lucie 2	144	21	58	9	227	39	30	70
Summer	27	8	8	8	52	235	4	4
Surry 1	118	139	95	45	22	23	19	209
Surry 2	118	139	95	45	22	23	19	209
Susquehanna 1	28	168	73	83	27	24	76	92
Susquehanna 2	28	168	73	83	27	24	76	92
Three Mile Island 1	12	13	14	15	236	14	8	5
Trojan	7	346	62	6	37	220	6	4
Turkey Point 3	116	52	24	28	192	85	30	67
Turkey Point 4	116	52	24	28	192	85	30	67
Vermont Yankee	194	34	15	28	28	14	176	61
Vogtle 1	11	5	9	8	203	35	97	130
Vogtle 2	NA	NA	NA	NA	NA	NA	NA	NA
Wash. Nuclear 2	36	361	44	52	40	399	52	46
Waterford 3	9	5	38	194	17	5	2	15
Wolf Creek	5	2	1	7	84	91	4	3
Yankee-Rowe	23	10	18	11	8	21	193	26
Zion 1	42	12	82	176	57	262	11	18
Zion 2	42	12	82	176	57	262	11	18
Total	9611	10763	5667	9383	8353	10333	6253	11242

*The data in this table was obtained from INPO.

*NA means the plant has not yet been commercial for one calendar year. 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-1.11 Cause Codes-Administrative Control Problems—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	3	4	2	7	0	2	3	5
Arkansas 2	3	4	1	5	7	4	7	3
Beaver Valley 1	1	1	0	1	1	0	1	1
Beaver Valley 2	3	5	1	2	2	2	2	2
Big Rock Point	1	0	2	0	1	1	1	2
Braidwood 1	0	0	1	6	2	3	3	2
Braidwood 2	2	0	0	4	1	2	0	0
Browns Ferry 1	9	5	8	2	3	3	3	4
Browns Ferry 2	10	9	8	2	4	3	2	4
Browns Ferry 3	7	4	7	2	3	2	2	5
Brunswick 1	4	5	1	4	3	2	5	3
Brunswick 2	0	4	3	3	5	2	3	1
Byron 1	2	2	1	0	3	1	3	2
Byron 2	2	0	1	0	1	1	4	2
Callaway	0	2	1	0	2	1	1	3
Calvert Cliffs 1	3	1	4	4	7	5	1	1
Calvert Cliffs 2	6	3	2	4	4	2	0	2
Catawba 1	4	3	3	2	13	3	1	4
Catawba 2	7	5	5	2	9	2	3	4
Clinton 1	10	7	2	5	3	4	1	1
Comanche Peak 1	NYL*	NYL	NYL	NYL	3	10	6	6
Cook 1	2	3	3	1	1	2	2	1
Cook 2	2	2	2	1	1	1	1	1
Cooper Station	4	4	0	0	2	1	1	1
Crystal River 3	5	7	4	4	0	2	3	1
Davis-Besse	2	4	1	0	3	2	2	1
Diablo Canyon 1	3	0	2	6	3	1	0	0
Diablo Canyon 2	4	1	2	6	2	2	0	0
Dresden 2	7	2	3	1	2	1	2	5
Dresden 3	1	5	1	3	4	2	1	1
Duane Arnold	3	0	2	1	2	2	8	3
Farley 1	0	1	0	1	3	1	0	1
Farley 2	1	5	1	2	2	2	1	1
Fermi 2	2	3	2	8	1	0	1	2
FitzPatrick	0	2	2	5	6	2	1	3
Fort Calhoun	5	6	2	2	3	8	1	2
Fort St. Vrain**	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	4	0	2	0	1	0	3
Grand Gulf	0	5	1	1	2	4	2	6
Haddam Neck	2	3	1	0	2	1	6	4

Footnotes at end of table.

Table A-1.11 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	3	2	2	6	3	4	4	5
Hatch 2	2	1	4	6	5	3	2	5
Hope Creek	5	4	2	5	1	2	5	3
Indian Point 2	0	2	1	1	0	1	0	0
Indian Point 3	5	0	0	1	1	0	2	1
Kewaunee	5	2	0	0	2	1	0	2
Lasalle 1	3	3	0	4	2	1	0	2
Lasalle 2	5	4	2	2	3	6	0	3
Limerick 1	10	10	5	1	4	4	4	6
Limerick 2	NYL	3	4	2	5	3	4	3
Maine Yankee	1	0	0	0	0	0	1	1
McGuire 1	4	2	7	1	1	3	4	2
McGuire 2	4	2	9	3	3	1	4	1
Millstone 1	2	1	1	1	2	5	4	1
Millstone 2	1	1	1	1	1	2	1	1
Millstone 3	3	5	7	8	6	8	1	2
Monticello	2	3	1	6	0	3	6	2
Nine Mile Pt. 1	4	4	3	6	2	2	5	4
Nine Mile Pt. 2	5	7	6	6	2	5	1	5
North Anna 1	1	2	2	0	2	3	1	1
North Anna 2	1	3	2	0	2	3	1	3
Oconee 1	7	4	5	2	0	7	1	0
Oconee 2	3	5	3	2	0	3	1	0
Oconee 3	4	4	4	2	1	3	0	0
Oyster Creek	5	3	1	1	3	2	3	2
Palisades	3	3	3	3	2	3	1	2
Palo Verde 1	2	3	1	3	1	4	2	2
Palo Verde 2	1	3	1	0	2	3	1	0
Palo Verde 3	0	3	1	2	1	3	1	0
Peach Bottom 2	4	7	3	5	2	3	4	6
Peach Bottom 3	1	3	4	6	3	3	5	3
Perry	7	7	2	3	3	8	6	9
Pilgrim	6	5	3	3	4	1	5	2
Point Beach 1	1	3	0	1	0	0	0	0
Point Beach 2	0	3	0	4	0	0	0	1
Prairie Island 1	1	3	1	2	2	3	4	4
Prairie Island 2	1	3	1	2	2	3	5	4
Quad Cities 1	1	0	4	7	4	1	4	3
Quad Cities 2	0	1	0	4	4	0	2	3
Rancho Seco	3	0	NLI ¹	NLI	NLI	NLI	NLI	NLI

Footnotes at end of table.

Table A-1.11 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	3	8	1	7	3	3	0	9
Robinson 2	2	1	0	1	1	2	0	1
Salem 1	4	12	1	2	3	1	7	1
Salem 2	4	5	0	1	2	9	7	3
San Onofre 1	5	5	7	4	1	2	1	3
San Onofre 2	4	3	9	4	2	2	3	3
San Onofre 3	4	4	3	2	3	2	4	2
Seabrook	1	2	1	2	2	2	0	0
Sequoyah 1	6	4	6	6	2	6	3	2
Sequoyah 2	5	5	5	7	3	3	3	3
Shearon Harris	3	3	2	3	6	3	0	2
South Texas 1	5	2	4	3	3	6	4	0
South Texas 2	5	3	1	1	2	2	1	1
St. Lucie 1	0	1	3	1	2	2	1	0
St. Lucie 2	1	2	1	0	0	1	0	0
Summer	2	3	3	1	1	1	0	0
Surry 1	5	4	4	0	1	2	2	2
Surry 2	3	4	5	4	2	1	1	2
Susquehanna 1	6	8	1	2	3	3	5	8
Susquehanna 2	3	4	2	4	2	2	6	6
Three Mile Island 1	0	0	0	2	2	0	0	0
Trojan	3	3	9	2	6	11	7	5
Turkey Point 3	4	1	3	1	1	5	2	3
Turkey Point 4	2	2	5	1	0	1	5	3
Vermont Yankee	7	0	3	1	4	2	1	2
Vogtle 1	4	3	1	1	8	5	3	2
Vogtle 2	3	4	1	0	2	2	6	1
Wash. Nuclear 2	5	7	6	3	3	3	5	6
Waterford 3	2	1	5	1	0	2	3	2
Wolf Creek	3	1	3	0	1	6	1	3
Yankee-Rowe	1	1	3	0	0	0	0	1
Zion 1	1	2	4	4	3	4	4	0
Zion 2	3	2	0	3	4	2	2	1

*NYL means the plant is not yet licensed.

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

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

Table A-1.12 Cause Codes-Licensed Operator Errors—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1	1	1	2	0	0	0	1
Arkansas 2	0	1	0	2	0	0	0	0
Beaver Valley 1	1	0	1	0	1	0	0	0
Beaver Valley 2	1	2	0	0	0	0	1	2
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	1	0	0	1	0	1	0
Braidwood 2	0	0	0	1	2	1	0	0
Browns Ferry 1	3	0	1	0	1	0	0	0
Browns Ferry 2	3	0	1	0	1	0	0	0
Browns Ferry 3	3	0	1	0	1	0	0	0
Brunswick 1	4	1	0	0	0	0	0	0
Brunswick 2	0	0	1	0	0	0	0	0
Byron 1	0	0	0	0	0	0	0	2
Byron 2	1	0	0	0	0	0	1	2
Callaway	1	4	1	0	0	0	1	2
Calvert Cliffs 1	2	0	0	0	0	2	1	2
Calvert Cliffs 2	0	1	0	0	0	2	0	2
Catawba 1	2	0	1	0	3	2	0	0
Catawba 2	2	0	3	0	1	1	2	0
Clinton 1	4	2	3	1	1	1	1	1
Comanche Peak 1	NYL*	NYL	NYL	NYL	2	3	1	2
Cook 1	1	0	0	1	0	1	0	1
Cook 2	1	2	0	0	0	0	0	0
Cooper Station	0	0	0	0	0	1	0	0
Crystal River 3	1	2	1	0	0	0	0	0
Davis-Besse	1	0	0	0	1	0	1	0
Diablo Canyon 1	0	0	0	1	1	0	0	0
Diablo Canyon 2	0	1	0	0	0	1	0	0
Dresden 2	2	0	0	0	1	0	0	0
Dresden 3	2	0	0	0	0	0	0	0
Duane Arnold	0	0	0	0	0	0	0	0
Farley 1	0	0	1	0	0	0	0	0
Farley 2	0	0	0	1	0	0	0	0
Fermi 2	1	0	1	3	0	1	1	0
FitzPatrick	1	0	0	0	0	0	0	0
Fort Calhoun	0	1	0	0	1	0	1	0
Fort St. Vrain**	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	2	0	0	0	0	0	0
Grand Gulf	0	1	2	1	0	0	0	1
Haddam Neck	0	0	1	0	0	0	1	1

Footnote at end of table.

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	1	0	0	2	0	2	0	0
Hatch 2	0	0	0	0	0	0	0	0
Hope Creek	1	1	0	0	1	0	0	1
Indian Point 2	0	0	0	0	0	0	0	1
Indian Point 3	0	0	0	0	0	0	0	0
Kewaunee	0	0	0	0	0	0	0	0
Lasalle 1	2	0	0	1	0	0	0	0
Lasalle 2	2	0	2	0	0	0	0	0
Limerick 1	0	0	0	0	0	1	0	1
Limerick 2	NYL	0	1	0	0	1	1	0
Maine Yankee	0	0	0	0	0	1	0	0
McGuire 1	0	1	1	0	1	1	1	0
McGuire 2	0	1	2	0	1	0	1	0
Millstone 1	0	1	0	0	0	0	0	0
Millstone 2	0	0	0	0	0	0	2	0
Millstone 3	3	2	0	1	0	1	0	0
Monticello	0	2	0	2	0	0	0	1
Nine Mile Pt. 1	1	0	0	0	0	1	0	0
Nine Mile Pt. 2	1	5	0	4	2	1	0	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	1	0	0	0	0	0	1	1
Oconee 1	2	0	0	0	0	2	0	0
Oconee 2	0	0	0	0	0	0	0	0
Oconee 3	0	0	1	1	1	0	0	0
Oyster Creek	0	2	0	2	1	0	0	1
Palisades	0	0	0	0	3	0	2	0
Palo Verde 1	1	0	0	1	0	1	0	0
Palo Verde 2	1	1	0	0	1	1	1	1
Palo Verde 3	1	1	1	0	0	0	0	0
Peach Bottom 2	0	2	0	1	0	1	1	0
Peach Bottom 3	0	0	0	0	0	0	1	0
Perry	1	3	0	0	0	1	2	1
Pilgrim	1	3	0	0	0	0	0	0
Pont Beach 1	1	0	0	0	0	0	1	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	1	1	0	0	1	0
Prairie Island 2	0	0	1	1	1	0	2	0
Quad Cities 1	0	0	0	0	0	0	1	1
Quad Cities 2	0	0	0	0	0	0	0	2
Rancho Seco	0	0	NLI ¹	NLI	NLI	NLI	NLI	NLI

Footnote at end of table.

Table A-1.12 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	1	0	0	0	0	0	1	1
Robinson 2	1	0	0	0	0	0	0	0
Salem 1	2	1	0	1	0	1	1	0
Salem 2	0	0	0	0	0	1	1	1
San Onofre 1	1	1	0	0	1	0	0	0
San Onofre 2	1	1	3	1	0	1	0	0
San Onofre 3	0	1	1	0	0	3	2	1
Seabrook	0	0	0	0	1	0	1	0
Sequoyah 1	1	0	0	0	3	1	0	0
Sequoyah 2	1	2	0	0	2	0	0	1
Shearon Harris	0	0	0	2	0	0	0	1
South Texas 1	0	0	0	0	0	0	2	1
South Texas 2	0	3	0	0	0	0	0	0
St. Lucie 1	0	0	1	0	0	0	0	0
St. Lucie 2	2	1	1	0	1	0	0	0
Summer	0	0	0	0	0	0	0	0
Surry 1	1	2	4	1	0	0	1	0
Surry 2	0	1	4	3	0	0	0	0
Susquehanna 1	2	0	0	0	0	0	0	1
Susquehanna 2	0	0	0	0	0	0	0	0
Three Mile Island 1	0	0	0	1	2	0	1	0
Trojan	0	0	1	1	0	1	0	0
Turkey Point 3	0	0	0	1	1	2	0	0
Turkey Point 4	0	2	0	1	1	1	1	0
Vermont Yankee	1	0	0	0	0	0	0	0
Vogtle 1	1	0	0	0	2	1	0	1
Vogtle 2	3	2	0	0	0	2	1	1
Wash. Nuclear 2	0	1	0	0	0	0	1	0
Waterford 3	0	0	1	2	0	0	0	0
Wolf Creek	3	0	1	0	0	2	1	0
Yankee-Rowe	0	0	1	0	0	1	0	0
Zion 1	1	0	0	1	2	0	1	0
Zion 2	1	1	0	0	0	0	1	0

*NYL means the plant is not yet licensed.

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

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

Table A-1.13 Cause Codes-Other Personnel Errors—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	1	3	0	6	0	1	2	2
Arkansas 2	1	2	1	5	2	1	5	2
Beaver Valley 1	0	2	0	3	1	0	1	1
Beaver Valley 2	1	5	0	1	2	1	2	0
Big Rock Point	0	1	3	0	0	0	2	0
Braidwood 1	0	0	0	4	0	0	1	0
Braidwood 2	1	0	0	1	0	0	0	0
Browns Ferry 1	2	3	3	2	1	3	0	2
Browns Ferry 2	3	4	7	2	1	2	0	1
Browns Ferry 3	3	3	3	2	1	2	0	2
Brunswick 1	1	1	0	2	0	2	3	4
Brunswick 2	2	1	1	1	1	3	5	1
Byron 1	1	0	0	0	3	0	0	1
Byron 2	0	0	0	0	1	0	1	1
Callaway	1	1	1	1	1	1	1	3
Calvert Cliffs 1	1	1	0	1	4	1	2	0
Calvert Cliffs 2	2	1	0	0	2	0	1	1
Catawba 1	4	1	1	0	5	2	2	0
Catawba 2	3	5	3	1	4	0	1	0
Clinton 1	3	4	1	2	1	1	2	1
Comanche Peak 1	NYL ^a	NYL	NYL	NYL	0	4	3	2
Cook 1	1	2	3	0	1	0	1	1
Cook 2	3	2	1	0	2	1	1	2
Cooper Station	3	0	0	0	1	1	1	0
Crystal River 3	0	3	3	1	1	1	3	3
Davis-Besse	0	1	5	1	0	3	2	0
Diablo Canyon 1	2	0	0	2	2	0	0	5
Diablo Canyon 2	0	0	1	0	2	3	0	0
Dresden 2	2	1	0	2	0	0	0	0
Dresden 3	0	2	0	3	0	0	0	0
Duane Arnold	2	1	2	1	0	0	4	1
Farley 1	0	2	0	2	0	0	1	0
Farley 2	0	4	1	2	0	0	0	2
Fermi 2	2	1	3	7	0	0	0	0
FitzPatrick	0	3	2	4	3	2	0	2
Fort Calhoun	2	1	1	2	3	1	0	1
Fort St. Vrain ^{na}	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	1	0	0	2	3	1	0
Grand Gulf	1	0	1	1	0	2	3	2
Haddam Neck	1	1	3	2	0	1	1	3

Footnote at end of table.

Table A-1.13 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	1	0	0	2	0	3	1	2
Hatch 2	1	0	1	2	0	1	1	4
Hope Creek	0	2	1	2	1	2	2	2
Indian Point 2	3	1	0	0	0	0	0	2
Indian Point 3	4	1	0	1	0	0	0	2
Kewaunee	1	1	0	0	1	2	0	0
Lasalle 1	2	1	0	2	3	1	0	0
Lasalle 2	4	0	0	2	2	1	0	2
Limerick 1	5	7	3	3	5	1	3	2
Limerick 2	NYL	1	5	5	4	1	5	1
Maine Yankee	0	1	0	0	0	1	1	0
McGuire 1	1	2	6	1	2	2	0	1
McGuire 2	1	1	4	3	1	1	0	1
Millstone 1	0	1	2	1	0	1	1	1
Millstone 2	2	1	2	1	1	2	2	5
Millstone 3	0	3	1	3	5	1	1	0
Monticello	1	2	1	3	0	1	3	0
Nine Mile Pt. 1	0	1	2	1	0	4	1	1
Nine Mile Pt. 2	3	1	2	2	3	1	0	3
North Anna 1	4	2	0	0	2	2	0	1
North Anna 2	1	2	0	0	3	1	0	1
Oconee 1	4	0	3	0	0	1	0	0
Oconee 2	2	1	1	0	0	0	0	0
Oconee 3	2	0	3	1	1	0	0	0
Oyster Creek	1	2	3	0	1	0	0	0
Palisades	1	2	0	0	3	1	3	1
Palo Verde 1	0	1	2	2	1	0	0	1
Palo Verde 2	1	1	0	1	1	2	0	1
Palo Verde 3	0	1	3	1	2	0	0	1
Peach Bottom 2	1	1	4	1	0	0	0	4
Peach Bottom 3	1	0	3	1	1	2	1	4
Perry	3	3	4	3	1	5	4	4
Pilgrim	4	3	3	3	1	0	3	0
Point Beach 1	1	1	1	1	0	2	0	0
Point Beach 2	1	1	1	2	0	1	0	1
Prairie Island 1	0	1	2	1	2	2	1	0
Prairie Island 2	0	1	1	1	4	1	0	3
Quad Cities 1	0	2	3	2	1	0	0	2
Quad Cities 2	0	0	0	1	3	1	0	3
Rancho Seco	1	1	NLI ¹	NLI	NLI	NLI	NLI	NLI

Footnote at end of table.

Table A-1.13 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	5	7	2	4	3	3	3	6
Robinson 2	3	0	0	0	2	1	2	0
Salem 1	2	4	0	4	3	0	2	3
Salem 2	1	0	0	3	1	6	3	1
San Onofre 1	1	0	1	0	0	1	2	0
San Onofre 2	1	1	4	0	0	1	1	1
San Onofre 3	2	2	1	0	0	2	1	0
Seabrook	1	2	3	1	2	4	2	2
Sequoyah 1	5	2	3	3	1	5	1	1
Sequoyah 2	1	4	4	2	2	6	2	2
Shearon Harris	6	3	1	4	1	3	1	0
South Texas 1	2	1	1	3	2	0	2	1
South Texas 2	1	1	0	2	1	1	3	0
St. Lucie 1	0	2	1	2	0	1	0	0
St. Lucie 2	1	1	1	0	1	0	1	1
Summer	3	0	3	1	0	3	0	0
Surry 1	4	5	2	0	0	0	1	1
Surry 2	2	4	4	2	0	0	2	0
Susquehanna 1	2	6	1	2	1	1	1	1
Susquehanna 2	1	3	2	3	3	2	0	2
Three Mile Island 1	0	0	0	0	1	0	0	0
Trojan	2	5	4	2	5	2	1	2
Turkey Point 3	2	1	2	0	3	1	1	1
Turkey Point 4	0	0	4	0	2	1	0	1
Vermont Yankee	2	2	1	0	2	1	1	2
Vogtle 1	6	0	1	1	2	1	1	1
Vogtle 2	4	2	1	3	1	1	0	3
Wash. Nuclear 2	2	5	1	0	1	1	1	2
Waterford 3	1	2	1	2	1	2	0	0
Wolf Creek	2	0	1	0	0	1	1	1
Yankee-Rowe	3	2	3	1	0	1	0	3
Zion 1	1	1	4	3	2	2	2	1
Zion 2	4	0	1	1	1	4	2	1

*NYL means the plant is not yet licensed.

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

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

Table A-1.14 Cause Codes-Maintenance Related—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	3	8	2	15	1	1	4	5
Arkansas 2	3	8	0	13	9	5	10	4
Beaver Valley 1	3	4	2	8	4	2	5	6
Beaver Valley 2	7	12	3	2	3	2	8	8
Big Rock Point	2	1	3	0	0	0	2	1
Braidwood 1	3	2	2	9	3	3	9	5
Braidwood 2	3	1	1	2	1	7	1	1
Browns Ferry 1	10	6	9	5	5	7	3	3
Browns Ferry 2	12	12	12	7	6	6	3	2
Browns Ferry 3	9	5	9	5	5	5	3	4
Brunswick 1	5	10	1	4	2	5	8	12
Brunswick 2	4	6	7	7	6	6	5	6
Byron 1	3	2	1	0	3	3	2	3
Byron 2	2	1	1	0	2	1	5	3
Callaway	2	5	2	1	3	3	2	5
Calvert Cliffs 1	4	1	3	3	8	5	4	3
Calvert Cliffs 2	6	4	1	3	4	3	1	4
Catawba 1	9	4	4	2	11	4	2	4
Catawba 2	10	9	5	2	7	1	3	5
Clinton 1	11	10	4	6	4	3	3	2
Comanche Peak 1	NYL*	NYL	NYL	NYL	4	9	10	6
Cook 1	4	5	4	1	1	1	2	5
Cook 2	8	5	2	2	2	1	2	2
Cooper Station	4	8	2	1	2	3	3	1
Crystal River 3	4	9	6	1	1	3	2	4
Davis-Besse	2	4	4	2	2	3	4	2
Diablo Canyon 1	3	0	2	9	2	2	0	6
Diablo Canyon 2	4	2	3	4	1	5	1	0
Dresden 2	10	3	10	3	2	1	5	9
Dresden 3	2	4	3	5	4	1	1	1
Duane Arnold	6	1	4	2	4	2	8	2
Farley 1	0	3	0	2	2	1	1	1
Farley 2	1	8	2	2	2	2	0	3
Fermi 2	7	4	8	11	1	2	3	2
FitzPatrick	2	5	4	6	9	6	1	7
Fort Calhoun	7	6	2	3	5	5	2	1
Fort St. Vrain**	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	5	3	4	1	6	0	3
Grand Gulf	1	3	3	2	1	6	5	7
Haddam Neck	1	2	6	3	3	1	6	7

Footnote at end of table.

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	4	0	2	6	5	6	6	6
Hatch 2	3	0	4	7	5	3	5	7
Hope Creek	5	7	3	5	3	4	6	6
Indian Point 2	2	2	3	1	0	1	7	7
Indian Point 3	6	1	0	2	1	1	1	4
Kewaunee	6	4	1	1	1	2	0	4
Lasalle 1	13	9	3	5	5	4	1	2
Lasalle 2	12	8	3	6	7	5	1	6
Limerick 1	11	13	8	8	6	3	7	12
Limerick 2	NYL	2	6	9	7	3	9	8
Maine Yankee	1	1	0	2	0	3	1	3
McGuire 1	4	3	11	2	3	5	7	4
McGuire 2	6	2	12	5	2	2	7	3
Millstone 1	1	8	3	2	1	4	3	2
Millstone 2	3	2	3	2	0	4	4	4
Millstone 3	2	6	7	8	9	6	1	3
Monticello	4	7	4	8	0	5	5	1
Nine Mile Pt. 1	4	2	2	2	1	6	6	7
Nine Mile Pt. 2	6	8	7	6	6	6	2	7
North Anna 1	6	6	2	2	3	4	1	1
North Anna 2	5	5	1	0	3	3	2	3
Oconee 1	7	2	3	0	0	5	2	0
Oconee 2	5	2	1	1	0	2	1	0
Oconee 3	5	2	3	2	1	2	0	0
Oyster Creek	4	4	5	1	3	3	3	3
Palisades	3	5	4	2	6	3	2	2
Palo Verde 1	2	4	5	2	2	3	1	1
Palo Verde 2	2	4	2	3	2	5	0	1
Palo Verde 3	3	4	4	1	2	3	0	1
Peach Bottom 2	5	9	6	7	5	5	11	4
Peach Bottom 3	2	3	6	7	5	5	9	3
Perry	9	9	3	3	3	10	10	9
Pilgrim	9	5	7	8	6	3	5	2
Point Beach 1	1	4	1	1	0	3	1	0
Point Beach 2	2	2	2	6	0	1	0	3
Prairie Island 1	0	3	6	6	2	6	4	3
Prairie Island 2	0	2	6	6	5	5	5	7
Quad Cities 1	2	5	5	7	5	4	5	5
Quad Cities 2	0	3	0	2	6	4	3	6
Rancho Seco	4	1	NLI ¹	NLI	NLI	NLI	NLI	NLI

Footnote at end of table.

Table A-1.14 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	13	14	3	8	6	9	2	15
Robinson 2	5	2	0	1	4	3	2	0
Salem 1	9	13	1	7	7	7	9	4
Salem 2	6	7	0	3	4	12	10	7
San Onofre 1	5	5	6	2	3	7	3	3
San Onofre 2	5	3	8	3	1	7	3	4
San Onofre 3	5	3	3	1	2	5	4	3
Seabrook	5	4	4	2	11	5	2	2
Sequoyah 1	6	7	7	8	3	5	6	3
Sequoyah 2	4	12	6	8	6	5	4	4
Shearon Harris	6	5	2	7	4	2	1	4
South Texas 1	7	3	2	6	5	7	6	2
South Texas 2	8	5	2	2	2	5	4	3
St. Lucie 1	0	2	2	3	2	4	2	0
St. Lucie 2	1	1	3	1	1	1	1	1
Summer	4	5	4	2	1	4	0	1
Surry 1	6	10	8	2	2	2	7	6
Surry 2	4	9	13	6	2	2	5	4
Susquehanna 1	3	10	2	4	5	4	5	8
Susquehanna 2	2	3	3	7	3	4	7	7
Three Mile Island 1	0	0	0	3	3	0	1	0
Trojan	4	5	8	5	11	10	9	3
Turkey Point 3	5	1	3	2	5	5	3	4
Turkey Point 4	3	3	7	2	2	3	6	5
Vermont Yankee	11	3	4	1	4	3	3	4
Vogtle 1	9	3	4	2	6	5	5	2
Vogtle 2	8	8	3	3	2	4	6	4
Wash. Nuclear 2	3	13	5	2	5	5	4	7
Waterford 3	1	3	6	5	3	3	5	2
Wolf Creek	7	3	5	0	1	11	3	3
Yankee-Rowe	3	3	4	1	1	2	2	4
Zion 1	4	1	5	5	6	6	6	2
Zion 2	4	2	1	2	5	5	6	4

¹NYL means the plant is not yet licensed.

²Fort St. Vrain is not included in the cause code data.

³NLI means the plant is no longer included in the PI data.

Table A-1.15 Cause Codes-Design/Fabrication/Installation—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	9	2	3	2	1	1	5	3
Arkansas 2	2	3	3	1	1	2	2	2
Beaver Valley 1	0	1	2	0	3	0	0	0
Beaver Valley 2	1	0	1	0	1	0	1	2
Big Rock Point	1	0	0	1	0	0	2	0
Braidwood 1	0	0	3	1	0	0	1	1
Braidwood 2	0	0	1	2	0	1	1	0
Browns Ferry 1	7	4	3	2	2	1	1	0
Browns Ferry 2	8	4	3	2	2	1	1	0
Browns Ferry 3	7	4	3	2	2	1	1	2
Brunswick 1	3	0	2	1	1	2	3	1
Brunswick 2	3	0	2	1	1	2	3	2
Byron 1	0	0	1	1	0	0	1	1
Byron 2	0	0	1	1	0	0	1	1
Callaway	1	0	0	0	1	0	0	1
Calvert Cliffs 1	2	5	4	4	2	6	2	2
Calvert Cliffs 2	1	4	4	4	2	4	0	3
Catawba 1	5	2	4	2	5	1	0	1
Catawba 2	5	2	3	3	3	1	0	1
Clinton 1	3	2	1	3	4	1	0	1
Comanche Peak 1	NYL*	NYL	NYL	NYL	1	0	2	2
Cook 1	2	0	0	0	1	1	4	1
Cook 2	2	0	0	1	2	2	4	2
Cooper Station	6	6	0	1	2	0	1	0
Crystal River 3	7	5	5	2	4	2	1	2
Davis-Besse	0	3	0	0	2	0	0	0
Diablo Canyon 1	1	0	1	2	0	1	1	1
Diablo Canyon 2	1	0	1	1	1	2	0	0
Dresden 2	2	0	2	1	0	1	1	1
Dresden 3	2	1	1	2	2	1	0	0
Duane Arnold	1	0	2	1	0	1	5	2
Farley 1	0	2	0	2	0	0	0	1
Farley 2	0	2	0	1	0	0	0	1
Fermi 2	2	2	1	2	0	2	3	0
Fitzpatrick	2	3	3	5	1	5	0	1
Fort Calhoun	2	3	0	1	4	4	4	1
Fort St. Vrain™	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	2	1	1	0	1	0	2
Grand Gulf	0	2	2	1	1	0	3	1
Haddam Neck	3	2	3	4	1	2	5	2

Footnote at end of table.

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	0	1	0	2	1	2	1	1
Hatch 2	0	0	0	1	2	2	1	0
Hope Creek	0	0	3	3	0	2	3	9
Indian Point 2	3	0	0	1	0	0	1	0
Indian Point 3	0	4	0	1	1	0	0	1
Kewaunee	2	2	2	0	3	1	1	1
Lasalle 1	4	1	0	1	2	0	0	1
Lasalle 2	4	1	0	1	1	0	0	0
Limerick 1	10	8	1	0	1	2	3	3
Limerick 2	NYL	1	1	1	2	1	2	2
Maine Yankee	1	0	0	1	1	0	0	4
McGuire 1	3	2	7	1	3	2	1	5
McGuire 2	1	2	7	1	5	2	1	4
Millstone 1	3	2	0	2	1	2	0	0
Millstone 2	2	0	0	1	0	2	0	2
Millstone 3	1	1	0	2	3	2	0	1
Monticello	1	3	2	5	1	2	1	1
Nine Mile Pt. 1	0	1	2	2	2	4	0	1
Nine Mile Pt. 2	2	1	1	2	0	0	0	1
North Anna 1	0	1	0	1	0	1	0	1
North Anna 2	0	1	0	1	0	1	0	1
Oconee 1	3	3	2	1	1	3	1	0
Oconee 2	3	2	2	2	1	3	1	0
Oconee 3	3	2	2	1	2	3	1	0
Oyster Creek	5	0	0	1	0	0	1	1
Palisades	3	1	4	1	3	3	2	1
Palo Verde 1	2	2	1	4	0	1	1	1
Palo Verde 2	2	3	1	2	0	1	2	1
Palo Verde 3	3	4	1	2	0	1	2	2
Peach Bottom 2	3	2	2	4	0	2	2	1
Peach Bottom 3	2	1	2	3	0	2	3	1
Perry	1	3	1	1	2	1	1	1
Pilgrim	4	0	1	0	2	0	1	1
Point Beach 1	1	3	0	1	0	3	3	1
Point Beach 2	1	2	1	2	0	1	2	1
Prairie Island 1	0	2	1	0	0	0	1	1
Prairie Island 2	0	2	0	0	0	0	1	1
Quad Cities 1	0	2	1	2	1	1	3	2
Quad Cities 2	0	2	1	2	2	0	1	3
Rancho Seco	4	0	NLI ^a	NLI	NLI	NLI	NLI	NLI

Footnote at end of table.

Table A-1.15 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	2	4	0	0	4	2	0	2
Robinson 2	2	1	1	4	2	2	0	0
Salem 1	3	6	0	4	5	3	3	2
Salem 2	3	8	1	9	11	12	1	4
San Onofre 1	5	2	3	5	1	1	2	0
San Onofre 2	1	3	1	0	0	0	0	1
San Onofre 3	0	4	3	1	1	2	1	0
Seabrook	0	0	1	0	0	0	3	1
Sequoyah 1	0	0	0	1	0	1	1	2
Sequoyah 2	1	0	1	1	1	1	1	0
Shearon Harris	2	0	1	1	1	2	2	1
South Texas 1	5	1	4	0	0	2	0	0
South Texas 2	2	1	5	1	1	2	1	0
St. Lucie 1	0	0	0	0	1	0	1	0
St. Lucie 2	1	0	0	0	0	0	1	1
Summer	1	0	1	1	1	1	0	1
Surry 1	1	3	2	2	0	1	0	0
Surry 2	1	3	3	1	0	1	1	0
Susquehanna 1	1	1	2	0	1	1	4	1
Susquehanna 2	1	1	1	2	2	0	3	1
Three Mile Island 1	0	0	0	0	1	0	0	0
Trojan	2	2	2	1	2	6	0	2
Turkey Point 3	3	3	0	2	0	1	0	0
Turkey Point 4	2	3	3	1	0	1	0	1
Vermont Yankee	2	1	1	0	2	2	1	1
Vogtle 1	1	0	0	1	2	1	0	1
Vogtle 2	1	1	1	1	1	1	0	1
Wash. Nuclear 2	4	6	6	2	2	2	3	7
Waterford 3	2	0	3	0	0	0	1	1
Wolf Creek	1	1	0	1	1	0	0	0
Yankee-Rowe	0	1	1	0	0	0	1	0
Zion 1	0	1	0	1	0	0	3	0
Zion 2	0	2	0	0	1	0	2	0

¹NYL means the plant is not yet licensed.

²Fort St. Vrain is not included in the cause code data.

³NLI means the plant is no longer included in the PI data.

Table A-1.16 Cause Codes-Equipment Failure (Electronic Piece-Part or Environmentally Related Failures)—Quarterly PI Data

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Arkansas 1	0	1	0	0	0	0	0	0
Arkansas 2	0	0	0	0	0	0	0	0
Beaver Valley 1	0	0	0	0	0	0	0	0
Beaver Valley 2	0	0	0	0	0	0	1	0
Big Rock Point	0	0	0	0	0	0	0	0
Braidwood 1	0	0	1	0	0	1	0	0
Braidwood 2	0	0	1	0	0	0	0	0
Browns Ferry 1	0	0	0	0	0	0	1	0
Browns Ferry 2	0	0	0	0	0	0	1	0
Browns Ferry 3	0	0	0	0	0	0	1	0
Brunswick 1	0	0	0	0	0	0	1	0
Brunswick 2	0	0	0	0	0	0	2	1
Byron 1	1	0	1	0	1	0	2	0
Byron 2	0	0	1	0	0	0	0	0
Callaway	0	1	0	1	0	1	0	2
Calvert Cliffs 1	0	0	0	0	0	0	1	0
Calvert Cliffs 2	0	0	0	0	0	0	0	0
Catawba 1	0	1	1	1	0	0	1	0
Catawba 2	0	1	1	0	0	0	0	0
Clinton 1	0	0	0	0	0	0	0	0
Comanche Peak 1	NYL*	NYL	NYL	NYL	1	0	1	0
Cook 1	0	0	0	0	0	0	0	0
Cook 2	0	0	1	0	0	0	0	0
Cooper Station	0	0	0	0	0	1	0	0
Crystal River 3	0	1	0	0	0	0	0	0
Davis-Besse	1	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	1	0	0	1
Diablo Canyon 2	1	0	0	0	0	0	1	0
Dresden 2	2	0	0	0	0	0	0	0
Dresden 3	1	0	0	0	1	0	0	0
Duane Arnold	1	0	0	0	0	0	0	0
Farley 1	0	0	0	0	0	0	1	0
Farley 2	0	0	0	0	0	0	0	0
Fermi 2	0	0	0	0	1	0	0	2
FitzPatrick	0	0	0	1	2	0	0	0
Fort Calhoun	0	0	0	0	0	0	0	0
Fort St. Vrain**	NA	NA	NA	NA	NA	NA	NA	NA
Ginna	0	0	0	0	0	0	0	1
Grand Gulf	1	0	1	1	0	0	0	0
Haddam Neck	0	0	0	0	0	0	0	0

Footnote at end of table.

Table A-1.16 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
Hatch 1	1	0	0	0	1	2	1	1
Hatch 2	0	0	0	0	0	1	0	0
Hope Creek	0	0	0	0	0	0	0	1
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	1	0	1	0
Limerick 2	NYL	0	1	0	1	0	0	0
Maine Yankee	0	0	0	0	0	0	0	1
McGuire 1	1	0	1	0	0	0	0	0
McGuire 2	0	0	0	0	0	0	0	1
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	0	0	0	0	0
Monticello	0	0	0	0	0	0	0	0
Nine Mile Pt. 1	1	0	0	0	0	0	0	0
Nine Mile Pt. 2	1	1	1	2	0	0	0	0
North Anna 1	0	0	0	0	0	0	0	0
North Anna 2	0	1	0	0	0	0	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	1
Oyster Creek	1	1	0	0	0	0	0	0
Palisades	0	0	0	0	1	0	0	0
Palo Verde 1	1	0	1	1	0	0	0	0
Palo Verde 2	0	0	1	0	0	0	0	0
Palo Verde 3	1	0	0	0	1	1	1	1
Peach Bottom 2	0	1	0	0	0	0	1	0
Peach Bottom 3	0	0	0	0	0	0	1	0
Perry	0	0	0	0	0	1	1	0
Pilgrim	0	0	0	0	0	0	0	0
Point Beach 1	0	0	0	0	0	0	1	0
Point Beach 2	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	0	1	0	0	0	0
Prairie Island 2	1	1	0	1	0	0	0	0
Quad Cities 1	0	1	0	0	2	1	0	0
Quad Cities 2	0	0	0	0	2	1	0	1
Rancho Seco	0	0	NLI ¹	NLI	NLI	NLI	NLI	NLI

Footnote at end of table.

Table A-1.16 (cont.)

Plant Name	Calendar Year-Quarter							
	89-1	89-2	89-3	89-4	90-1	90-2	90-3	90-4
River Bend	0	0	1	0	0	0	0	0
Robinson 2	0	0	0	0	0	0	0	0
Salem 1	1	1	0	0	0	0	0	0
Salem 2	1	0	0	0	0	0	0	0
San Onofre 1	0	0	0	0	0	0	0	0
San Onofre 2	1	1	0	0	0	0	0	0
San Onofre 3	1	1	0	0	0	0	0	0
Seabrook	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	0	0	1	0
Sequoyah 2	0	0	0	0	1	0	0	0
Shearon Harris	1	0	0	0	0	0	0	0
South Texas 1	0	0	0	0	0	0	1	0
South Texas 2	0	0	3	1	1	0	0	1
St. Lucie 1	0	0	0	0	0	0	0	0
St. Lucie 2	1	0	1	0	0	0	0	1
Summer	0	0	0	1	0	0	0	0
Surry 1	1	0	1	2	0	0	0	0
Surry 2	1	0	0	0	1	0	0	0
Susquehanna 1	1	0	0	1	1	0	1	0
Susquehanna 2	1	0	1	0	1	1	0	0
Three Mile Island 1	0	0	0	0	0	0	0	0
Trojan	0	1	1	0	0	0	1	0
Turkey Point 3	0	0	0	0	1	0	0	0
Turkey Point 4	0	0	0	0	1	1	1	0
Vermont Yankee	0	0	0	0	0	0	0	0
Vogtle 1	1	1	0	0	0	0	0	0
Vogtle 2	1	0	1	1	0	1	1	0
Wash. Nuclear 2	0	0	1	0	0	0	1	0
Waterford 3	0	0	1	0	0	0	1	1
Wolf Creek	0	0	0	0	0	1	0	0
Yankee-Rowe	0	0	0	0	0	0	0	0
Zion 1	1	0	0	0	0	0	0	0
Zion 2	0	1	0	0	0	0	0	0

*NYL means the plant is not yet licensed.

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

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

Appendix A-2

Additional Plant Operational Experience Data

APPENDIX A-2

Other Plant Operational Experience Data

This appendix presents selected Licensee Event Report(LER) and plant operational experience data. This information is referenced in Sections 2 and 5 of this report.

Tables A-2.1 through A-2.9 present data regarding reactor scrams (trips). Tables A-2.10 through A-2.13 provide data regarding engineered safety feature actuations. Tables 2-14 through A-2.16 present information regarding shutdowns required by technical specifications, annual unit operating factors, and reactor critical hours.

Table A-2.1 Total Automatic and Manual Reactor Scrams While the Reactor is Critical and the Reactor Scram Rates for 1986 through 1990

Plant Name	Vendor	1986				1987				1988				1989				1990			
		Total	Auto	Man	Total Scram Rate	Total	Auto	Man	Total Scram Rate	Total	Auto	Man	Total Scram Rate	Total	Auto	Man	Total Scram Rate	Total	Auto	Man	Total Scram Rate
Arkansas 1	BW	2	2		0.36	2	2		0.25	1	1		0.16	5	5		0.83	0			
Arkansas 2	CE	5	5		0.78	2	2		0.26	2	1	1	0.33	2	2		0.30	4	2	2	0.49
Beaver Valley 1	W	3	2	1	0.46	4	4		0.55	3	3		0.42	4	4		0.68	1	1		0.12
Beaver Valley 2	W	NYC ¹				17	14	3	7.35	4	4		0.46	1	1		0.16	1	1		0.15
Big Rock Point	GE	1	1		0.12	1	1		0.16	3	2	1	0.47	1	1		0.14	0			
Braidwood 1	W	NYC				7	5	2	2.04	3	2	1	0.52	2	2		0.36	5	5		0.64
Braidwood 2	W	NYC				NYC				14	10	4	2.92	3	3		0.39	1	1		0.14
Browns Ferry 1	GE	N/A ¹				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	N/A				N/A				N/A				N/A				N/A			
Brunswick 1	GE	7	6	1	0.84	2	2		0.35	2	2		0.30	0				1	1		0.17
Brunswick 2	GE	2	2		0.47	2	2		0.24	2	1	1	0.35	1		1	0.17	6	5	1	1.01
Byron 1	W	5	4	1	0.64	4	4		0.64	3	3		0.46	1		1	0.11	4	4		0.56
Byron 2	W	NYC				11	9	2	1.61	6	5	1	0.69	0				2	1	1	0.30
Callaway	W	7	6	1	0.96	1	1		0.16	6	5	1	0.73	2	1	1	0.27	4	4		0.54
Calvert Cliffs 1	CE	4	4		0.58	6	4	2	0.91	3	2	1	0.47	0				0			
Calvert Cliffs 2	CE	6	3	3	0.71	6	4	2	1.01	2	2		0.26	0				0			
Catawba 1	W	8	8		1.47	5	3	2	0.82	0				3	1	2	0.40	0			
Catawba 2	W	7	5	2	2.61	9	5	4	1.25	9	4	5	1.39	3	2	1	0.47	1	1		0.17
Clinton 1	GE	NYC				9	7	2	1.68	3	2	1	0.41	5	1	4	1.18	2	1	1	0.41
Comanche Peak 1**	W	NYC				NYC				NYC				NYC				9	6	3	1.70
Cook 1	W	5	5		0.66	2	2		0.33	3	3		0.36	2	2		0.32	0			
Cook 2	W	2	2		0.36	5	5		0.79	0				1	1		0.15	3	3		0.60
Cooper Station	GE	2	2		0.30	6	5	1	0.71	3	2	1	0.50	3	3		0.45	1	1		0.14
Crystal River 3	BW	1	1		0.27	2	2		0.37	2	2		0.27	1	1		0.23	0			
Davis-Besse	BW	1	1		5.62	5	5		0.67	1	1		0.47	2	2		0.23	2	2		0.40
Diablo Canyon 1	W	3	3		0.50	5	5		0.59	5	5		0.88	1	1		0.14	4	3	1	0.47
Diablo Canyon 2	W	11	10	1	1.51	5	4	1	0.63	3	2	1	0.48	4	1	3	0.49	0			
Dresden 2	GE	6	4	2	0.84	4	4		0.69	0				2	2		0.28	3	2	1	0.50
Dresden 3	GE	6	5	1	2.17	6	4	2	0.83	1	1		0.16	3	3		0.41	1	1		0.13
Duane Arnold	GE	2	1	1	0.27	0				1	1		0.15	5	4	1	0.72	6	4	2	0.90
Farley 1	W	4	4		0.55	4	3	1	0.48	1	1		0.13	1	1		0.13	1		1	0.12

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1986				1987				1988				1989				1990				
		Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	
Farley 2	W	4	4		0.53	2	2		0.31	0				6	5	1	0.83	1	1		0.15	
Fermi 2	GE	4	2	2	2.85	7	7		1.36	5	4	1	0.94	4	2	2	0.67	2	2		0.27	
FitzPatrick	GE	2	2		0.25	6	6		0.97	0				2	2		0.25	5	4	1	0.79	
Fort Calhoun	CE	2	1	1	0.24	0				0				1		1	0.13	1		1	0.18	
Fort St. Vrain ^a	GA	2	1	1	0.81	4	2	2	1.36	4	1	3	1.05	0				N/A				
Ginna	W	4	3	1	0.52	0				2	2		0.26	1	1		0.15	6	6		0.81	
Grand Gulf	GE	6	5	1	1.07	2	2		0.28	6	6		0.71	5	4	1	0.71	5	4	1	0.72	
Haddam Neck	W	7	4	3	1.38	1	1		0.21	3	3		0.49	0				2		2	0.71	
Hatch 1	GE	4	4		0.72	4	4		0.56	5	5		0.83	0				4	2	2	0.67	
Hatch 2	GE	7	7		1.08	4	4		0.47	7	5	2	1.10	1	1		0.15	2	2		0.23	
Hope Creek	GE	11	9	2	4.12	7	5	2	0.92	5	4	1	0.71	2	2		0.29	4	4		0.50	
Indian Point 2	W	10	7	3	1.98	2	2		0.32	4	4		0.53	2	2		0.35	0				
Indian Point 3	W	8	8		1.22	5	5		0.91	4	3	1	0.55	1		1	0.19	2	1	1	0.36	
Kewaunee	W	3	3		0.40	2	2		0.25	3	3		0.39	1	1		0.13	0				
La Crosse ^m	AC	17	15	2	3.86	2	1	1	0.71	N/A				N/A				N/A				
LaSalle 1	GE	1	1		0.42	6	6		1.07	0				1	1		0.16	2	2		0.24	
LaSalle 2	GE	4	4		0.60	1	1		0.21	2	1	1	0.30	1	1		0.15	2	2		0.32	
Limerick 1	GE	1	1		0.14	2	2		0.33	1	1		0.12	0				0				
Limerick 2 ^{**}	GE	NYC				NYC				NYC				1	1		0.51	2	2		0.26	
Maine Yankee	CE	6	4	2	0.77	2	1	1	0.35	3	3		0.43	2	2		0.24	0				
McGuire 1	W	3	3		0.60	4	4		0.59	4	3	1	0.59	2	1	1	0.28	3	3		0.62	
McGuire 2	W	6	5	1	1.04	5	5		0.71	2	1	1	0.27	3	3		0.43	1		1	0.17	
Millstone 1	GE	3	2	1	0.36	4	4		0.57	1	1		0.12	3	3		0.41	1	1		0.12	
Millstone 2	CE	4	4		0.61	5	5		0.61	1	1		0.14	0				2	1	1	0.31	
Millstone 3	W	13	13		2.10	10	10		1.57	5	4	1	0.69	2	1	1	0.30	7	2	5	0.89	
Monticello	GE	2	2		0.29	4	4		0.56	1	1		0.11	2	2		0.30	1	1		0.12	
Nine Mile Point 1	GE	2	2		0.34	3	2	1	0.37	0				0				4	2	2	1.19	
Nine Mile Point 2	GE	NYC				6	6		2.22	9	7	2	1.99	6	6		1.15	2	1	1	0.42	
North Anna 1	W	6	5	1	0.79	4	3	1	0.87	4	3	1	0.50	3	3		0.60	1	1		0.11	
North Anna 2	W	4	4		0.55	0				0				0				1	1		0.14	
Oconee 1	BW	2	2		0.34	0				1	1		0.11	3	2	1	0.41	1	1		0.13	
Oconee 2	BW	5	5		0.69	3	3		0.35	1	1		0.14	3	3		0.41	0				

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1986				1987				1988				1989				1990			
		Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate
Oconee 3	BW	2	1	1	0.26	0				2	2		0.26	2	2		0.26	3	2	1	0.34
Oyster Creek	GE	3	2	1	1.26	3	3		0.53	0				5	4	1	1.00	3	1	2	0.38
Palisades	CE	2	2		1.34	6	1	5	1.42	0				1	1		0.17	2	1	1	0.39
Palo Verde 1	CE	13	13		2.36	3	3		0.65	5	5		0.67	1	1		0.66	1	1		0.24
Palo Verde 2	CE	10	10		2.63	4	3	1	0.57	1	1		0.17	3	3		0.71	1		1	0.19
Palo Verde 3	CE	NYC				1	1		1.06	0				1	1		0.83	2	2		0.24
Peach Bottom 2	GE	5	4	1	0.69	0				N/A				4	4		0.75	0			
Peach Bottom 3	GE	9	6	3	1.52	2	2		1.10	N/A				0				3	1	2	0.38
Perry	GE	4	3	1	2.80	12	6	6	2.66	8	8		1.15	0				1	1		0.17
Pilgrim	GE	4	4		2.33	N/A				0				5	4	1	0.89	2	1	1	0.28
Point Beach 1	W	2	2		0.25	2	2		0.27	0				0				0			
Point Beach 2	W	2	2		0.28	1	1		0.13	1	1		0.13	2	2		0.28	0			
Prairie Island 1	W	2	2		0.25	2	2		0.27	0				1	1		0.11	1	1		0.13
Prairie Island 2	W	3	3		0.38	0				0				3	3		0.38	5	5		0.64
Quad Cities 1	GE	4	4		0.65	1	1		0.18	1	1		0.12	3	1	2	0.45	1	1		0.14
Quad Cities 2	GE	2	2		0.35	5	5		0.72	3	2	1	0.48	2	2		0.24	2	2		0.32
Rancho Seco ^a	BW	N/A				N/A				4	3	1	0.72	1	1		0.42	N/A			
River Bend	GE	18	18		3.39	4	3	1	0.87	5	5		0.60	4	4		0.66	3	3		0.44
Robinson 2	W	10	10		1.40	4	4		0.63	3	3		0.52	3	3		0.70	2	2		0.35
Salem 1	W	9	9		1.27	1	1		0.16	3	2	1	0.43	3	3		0.48	3	3		0.50
Salem 2	W	9	9		1.60	4	4		0.62	7	7		1.17	4	3	1	0.52	2	2		0.37
San Onofre 1	W	2	2		0.67	1	1		0.14	0				3	1	2	0.84	2	1	1	0.48
San Onofre 2	CE	6	7	1	1.23	3	1	2	0.48	0				1		1	0.19	1	1		0.13
San Onofre 3	CE	6	5	1	0.81	2	2		0.28	1		1	0.17	2	2		0.24	1	1		0.16
Seabrook ^b	W	NYC				NYC				NYC				1		1	5.14	4	4		0.72
Sequoyah 1	W	N/A				N/A				2	2		5.27	2	2		0.23	3	3		0.46
Sequoyah 2	W	N/A				N/A				5	5		0.96	4	4		0.63	2	2		0.29
Shearon Harris	W	NYC				19	9	10	3.06	3	1	2	0.46	8	8		0.66	0			
Shoreham ^c	GE	0				N/A				N/A				N/A				N/A			
South Texas 1	W	NYC				NYC				4	4		0.77	3	3		0.52	8	8	2	1.45
South Texas 2 ^d	W	NYC				NYC				NYC				10	9	1	2.22	4	4		0.67
St. Lucie 1	CE	4	3	1	0.47	6	6		0.86	3	3		0.40	2	2		0.24	1		1	0.18

Footnotes at end of table.

Table A-2.1 (cont.)

Plant Name	Vendor	1986				1987				1988				1989				1990			
		Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate	Total	Auto	Man	Total Soram Rate
St. Lucie 2	CE	5	4	1	0.68	5	5		0.68	0				2	1	1	0.30	1	1		0.15
Summer	W	6	6		0.71	4	4		0.64	4	4		0.66	5	3	2	0.69	0			
Surry 1	W	5	2	3	0.80	3	2	1	0.49	2	2		0.53	2	1	1	0.47	2	1	1	0.30
Surry 2	W	4	3	1	0.65	1	1		0.15	3	2	1	0.60	2	2		1.33	3		3	0.38
Susquehanna 1	GE	0				2	1	1	0.31	2	2		0.24	4	3	1	0.61	0			
Susquehanna 2	GE	2	1	1	0.34	1	1		0.12	0				1		1	0.14	2	2		0.24
Three Mile Isl. 1	BW	4	4		0.64	3	3		0.47	1	1		0.15	1	1		0.11	1	1		0.14
Trojan	W	4	4		0.57	3	2	1	0.63	4	4		0.68	1	1		0.18	1	1		0.17
Turkey Point 3	W	6	5	1	0.86	5	3	2	2.62	0				1	1		0.17	2	2		0.38
Turkey Point 4	W	3	2	1	0.98	1	1		0.22	1	1		0.20	2	1	1	0.48	3	2	1	0.44
Vermont Yankee	GE	2	2		0.46	3	3		0.41	3	3		0.36	0				3	3		0.40
Vogtle 1	W	NYC				25	25		4.64	9	6	3	1.32	5	2	3	0.59	4	1	3	0.56
Vogtle 2**	W	NYC				NYC				NYC				7	6	1	1.14	4	2	2	0.55
Wash. Nuclear 2	GE	7	7		1.10	6	5	1	0.97	2	1	1	0.32	4	4		0.58	2	1	1	0.34
Waterford 3	CE	7	7		1.00	7	7		0.97	4	3	1	0.60	3	1	2	0.41	3	3		0.37
Wolf Creek	W	7	7		1.07	9	9		1.46	0				2	2		0.23	4	4		0.56
Yankee-Rose	W	3	3		0.36	2	2		0.28	3	3		0.40	3	2	1	0.37	1	1		0.19
Zion 1	W	2	2		0.36	1	1		0.15	4	4		0.59	1	1		0.19	2	2		0.39
Zion 2	W	4	4		0.51	0				2	2		0.29	0				4	3	1	1.28
Total All Plants***		457	404	53	0.61	406	343	63	0.65	274	228	46	0.41	244	197	47	0.36	231	176	55	0.33
Total All W Plants		206	185	21	0.61	202	172	30	0.73	151	126	25	0.48	125	99	26	0.38	126	97	29	0.37
Total All GE Plants		133	115	18	0.77	125	108	17	0.63	81	68	13	0.41	80	65	15	0.39	78	60	18	0.34
Total All CE Plants		82	72	10	0.89	58	45	13	0.63	25	21	4	0.24	21	16	5	0.26	20	13	7	0.23
Total All B&W Plants		17	16	1	0.48	15	15	0	0.31	13	12	1	0.25	18	17	1	0.34	7	6	1	0.15

*NYC means the plant is not yet critical.

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

†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.

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

*Rancho Seco ceased commercial operation on June 6, 1989.

*Shoreham ceased commercial operations on June 7, 1987.

*** This total includes Fort St. Vrain and La Crosse.

Table A-2.2 Reactor Scram Initiating Systems

Mature Westinghouse Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	45	24	35	27	31	162	0.21	0.10	0.13	0.09	0.10
RPS	22	12	22	15	13	84	0.10	0.05	0.08	0.05	0.04
Turbine	27	20	17	16	10	90	0.12	0.09	0.06	0.05	0.03
Electrical	20	9	10	5	10	54	0.09	0.04	0.04	0.02	0.03
Control Rod Drive	11	8	5	7	7	38	0.05	0.03	0.02	0.02	0.02
Main generator	7	11	6	8	8	40	0.03	0.05	0.02	0.03	0.03
Main steam	7	5	4	13	8	37	0.03	0.02	0.01	0.04	0.03
Support	9	2	4	4	8	27	0.04	0.01	0.01	0.01	0.03
RCS	2	6	4	3	8	23	0.01	0.03	0.01	0.01	0.03
Condensate	2	2	3	2	2	11	0.01	0.01	0.01	0.01	0.01
Total	152	99	110	100	105	566	0.70	0.42	0.41	0.33	0.33

¹Critical hours: 1986 = 218,327.8; 1987 = 234,631.5; 1988 = 270,463.7; 1989 = 303,642.2; and 1990 = 316,721.8.

Mature General Electric Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	16	23	11	10	15	75	0.11	0.13	0.06	0.05	0.07
RPS	15	9	10	24	12	70	0.10	0.05	0.05	0.12	0.05
Turbine	12	13	15	22	17	79	0.08	0.08	0.08	0.11	0.08
Electrical	11	7	3	5	7	33	0.07	0.04	0.02	0.02	0.03
Control Rod Drive	2	1	2	0	1	6	0.01	0.01	0.01	0.00	0.00
Main generator	2	12	12	2	6	34	0.01	0.07	0.06	0.01	0.03
Main steam	10	8	4	7	6	35	0.07	0.05	0.02	0.03	0.03
Support	11	5	2	4	9	31	0.07	0.03	0.01	0.02	0.04
RCS	3	2	3	0	1	9	0.02	0.01	0.02	0.00	0.00
Condensate	11	6	6	5	2	30	0.07	0.04	0.03	0.02	0.01
Total	93	86	68	79	76	402	0.61	0.50	0.37	0.39	0.34

²Critical Hours: 1986 = 152,007.9; 1987 = 170,556.6; 1988 = 186,252.0; 1989 = 202,523.0; and 1990 = 224,059.4.

Table A-2.2 (cont.)

Mature Combustion Engineering Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	13	17	9	7	2	48	0.17	0.20	0.10	0.09	0.02
RPS	6	2	3	5	7	23	0.08	0.02	0.03	0.06	0.08
Turbine	4	7	0	2	2	15	0.05	0.08	0.00	0.03	0.02
Electrical	8	6	3	2	2	21	0.11	0.07	0.03	0.03	0.02
Control Rod Drive	2	2	0	1	2	7	0.03	0.02	0.00	0.01	0.02
Main generator	5	7	4	1	0	17	0.07	0.08	0.04	0.01	0.00
Main steam	3	2	1	0	2	8	0.04	0.02	0.01	0.00	0.02
Support	3	4	1	0	1	9	0.04	0.05	0.01	0.00	0.01
RCS	5	3	2	2	1	13	0.07	0.04	0.02	0.03	0.01
Condensate	3	3	2	0	1	9	0.04	0.04	0.02	0.00	0.01
Total	52	53	25	20	20	170	0.69	0.63	0.26	0.25	0.23

³Critical hours: 1986 = 75,737.5; 1987 = 84,585.2; 1988 = 94,353.6; 1989 = 79,767.5; and 1990 = 85,829.4.

Mature Babcock & Wilcox Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ⁴				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	5	7	7	4	2	25	0.14	0.14	0.14	0.08	0.04
RPS	2	2	0	4	3	11	0.05	0.04	0.00	0.08	0.06
Turbine	3	1	2	4	0	10	0.08	0.02	0.04	0.08	0.00
Electrical	2	1	1	4	0	8	0.05	0.02	0.02	0.08	0.00
Control Rod Drive	0	0	1	1	2	4	0.00	0.00	0.02	0.02	0.04
Main generator	1	2	0	1	0	4	0.03	0.04	0.00	0.02	0.00
Main steam	0	0	1	0	0	1	0.00	0.00	0.02	0.00	0.00
Support	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
RCS	1	0	0	0	0	1	0.03	0.00	0.00	0.00	0.00
Condensate	3	1	1	0	0	5	0.08	0.02	0.02	0.00	0.00
Total	17	15	13	18	7	70	0.46	0.31	0.25	0.34	0.15

⁴Critical hours: 1986 = 36,712.2; 1987 = 48,711.0; 1988 = 51,033.2; 1989 = 52,332.1; and 1990 = 48,234.5.

Table A-2.3 Activities at Time of Reactor Scram

Mature Westinghouse Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Normal Operation	55	42	44	56	63	260	0.25	0.18	0.16	0.18	0.20
Testing	32	16	27	22	17	114	0.15	0.07	0.10	0.07	0.05
Raising Power	26	13	13	4	5	61	0.12	0.06	0.05	0.01	0.02
Maintenance	16	10	10	10	7	53	0.07	0.04	0.04	0.03	0.02
Reducing Power	12	11	10	5	9	47	0.05	0.05	0.04	0.02	0.03
Troubleshooting	7	5	5	2	3	22	0.03	0.02	0.02	0.01	0.01
Calibration	2	2	1	1	1	7	0.01	0.01	0.00	0.00	0.00
Special Testing	2	0	0	0	0	2	0.01	0.00	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	152	99	110	100	105	566	0.70	0.42	0.41	0.33	0.33

¹Critical hours: 1986 = 218,327.8; 1987 = 234,631.5; 1988 = 270,463.7; 1989 = 303,642.2; and 1990 = 316,721.8.

Mature General Electric Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Normal Operation	26	44	32	34	48	184	0.17	0.26	0.17	0.17	0.21
Testing	18	13	15	21	13	80	0.12	0.08	0.08	0.10	0.06
Raising Power	19	7	8	7	5	46	0.12	0.04	0.04	0.03	0.02
Reducing Power	13	12	7	8	6	46	0.09	0.07	0.04	0.04	0.03
Maintenance	13	9	5	3	3	33	0.09	0.05	0.03	0.01	0.01
Troubleshooting	3	1	0	3	0	7	0.02	0.01	0.00	0.01	0.00
Calibration	0	0	1	2	1	4	0.00	0.00	0.01	0.01	0.00
Special Testing	1	0	0	1	0	2	0.01	0.00	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	93	86	68	79	76	402	0.61	0.50	0.37	0.39	0.34

²Critical Hours: 1986 = 152,007.9; 1987 = 170,556.6; 1988 = 186,252.0; 1989 = 202,523.0; and 1990 = 224,059.4.

Table A-2.3 (cont.)

Mature Combustion Engineering Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Normal Operation	24	32	10	11	11	88	0.32	0.38	0.11	0.14	0.13
Reducing Power	5	8	2	3	4	22	0.07	0.09	0.02	0.04	0.05
Maintenance	7	5	6	2	0	20	0.09	0.06	0.06	0.03	0.00
Testing	8	5	3	0	2	18	0.11	0.06	0.03	0.00	0.02
Raising Power	7	2	3	3	1	16	0.09	0.02	0.03	0.04	0.01
Troubleshooting	1	0	1	1	2	5	0.01	0.00	0.01	0.01	0.02
Calibration	0	1	0	0	0	1	0.00	0.01	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Special Testing	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	52	53	25	20	20	170	0.69	0.63	0.26	0.25	0.23

³Critical hours: 1986 = 75,737.5; 1987 = 84,585.2; 1988 = 94,353.6; 1989 = 79,767.5; and 1990 = 85,829.4.

Mature Babcock & Wilcox Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours ⁴				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Normal Operation	7	9	6	7	3	32	0.19	0.18	0.12	0.13	0.06
Testing	2	1	2	4	3	12	0.05	0.02	0.04	0.08	0.06
Raising Power	4	2	3	0	0	9	0.11	0.04	0.06	0.00	0.00
Maintenance	1	1	0	4	0	6	0.03	0.02	0.00	0.08	0.00
Calibration	0	1	1	2	1	5	0.00	0.02	0.02	0.04	0.02
Troubleshooting	3	0	0	1	0	4	0.08	0.00	0.00	0.02	0.00
Reducing Power	0	1	1	0	0	2	0.00	0.02	0.02	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Special Testing	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	17	15	13	18	7	70	0.46	0.31	0.25	0.34	0.15

⁴Critical hours: 1986 = 36,712.2; 1987 = 48,711.0; 1988 = 51,033.2; 1989 = 52,332.1; and 1990 = 48,234.5.

Table A-2.4 Reactor Scram Causes

Mature Westinghouse Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	88	59	60	53	56	316	0.40	0.25	0.22	0.17	0.18
Personnel Error	44	26	32	27	23	152	0.20	0.11	0.12	0.09	0.07
Procedure	11	4	13	8	4	40	0.05	0.02	0.05	0.03	0.01
Unknown	2	4	2	6	9	23	0.01	0.02	0.01	0.02	0.03
Natural Phenomenon	4	5	1	4	4	18	0.02	0.02	0.00	0.01	0.01
Other	0	0	1	1	8	10	0.00	0.00	0.00	0.00	0.03
SG Level	3	1	1	1	1	7	0.01	0.00	0.00	0.00	0.00
Total	152	99	110	100	105	566	0.70	0.42	0.41	0.33	0.33

¹Critical hours: 1986 = 218,327.8; 1987 = 234,631.5; 1988 = 270,463.7; 1989 = 303,642.2; and 1990 = 316,721.8.

Mature General Electric Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	54	61	40	47	41	243	0.36	0.36	0.21	0.23	0.18
Personnel Error	28	17	14	19	10	88	0.18	0.10	0.08	0.09	0.04
Procedure	7	5	9	5	4	30	0.05	0.03	0.05	0.02	0.02
Unknown	3	2	1	3	10	19	0.02	0.01	0.01	0.01	0.04
Other	1	1	0	3	11	16	0.01	0.01	0.00	0.01	0.05
Natural Phenomenon	0	0	4	2	0	6	0.00	0.00	0.02	0.01	0.00
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	93	86	68	79	76	402	0.61	0.50	0.37	0.39	0.34

²Critical Hours: 1986 = 152,007.9; 1987 = 170,556.6; 1988 = 186,252.0; 1989 = 202,523.0; and 1990 = 224,059.4.

Table A-2.4 (cont.)

Mature Combustion Engineering Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	32	28	12	10	12	94	0.42	0.33	0.13	0.13	0.14
Personnel Error	14	11	6	2	2	35	0.18	0.13	0.06	0.03	0.02
Procedure	2	5	5	3	0	15	0.03	0.06	0.05	0.04	0.00
Unknown	4	2	2	3	2	13	0.05	0.02	0.02	0.04	0.02
Other	0	3	0	1	3	7	0.00	0.04	0.00	0.01	0.03
Natural Phenomenon	0	2	0	1	1	4	0.00	0.02	0.00	0.01	0.01
SG Level	0	2	0	0	0	2	0.00	0.02	0.00	0.00	0.00
Total	52	53	25	20	20	170	0.69	0.63	0.26	0.25	0.23

³Critical hours: 1986 = 75,737.5; 1987 = 84,585.2; 1988 = 94,353.6; 1989 = 79,767.5; and 1990 = 85,829.4.

Mature Babcock & Wilcox Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours ⁴				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	10	10	9	5	2	36	0.27	0.21	0.18	0.10	0.04
Personnel Error	5	4	2	10	1	22	0.14	0.08	0.04	0.19	0.02
Unknown	0	0	1	3	2	6	0.00	0.00	0.02	0.06	0.04
Procedure	1	0	1	0	0	2	0.03	0.00	0.02	0.00	0.00
Other	0	0	0	0	2	2	0.00	0.00	0.00	0.00	0.04
Natural Phenomenon	1	1	0	0	0	2	0.03	0.02	0.00	0.00	0.00
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	17	15	13	18	7	70	0.46	0.31	0.25	0.34	0.15

⁴Critical hours: 1986 = 36,712.2; 1987 = 48,711.0; 1988 = 51,033.2; 1989 = 52,332.1; and 1990 = 48,234.5.

Table A-2.5 Reactor Scram Signals

Mature Westinghouse Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Low SG level	27	21	28	30	27	133	0.12	0.09	0.10	0.10	0.09
Turbine trip	35	25	29	19	21	129	0.16	0.11	0.11	0.06	0.07
Manual	18	9	15	23	23	88	0.08	0.04	0.06	0.08	0.07
High neg. flux rate	6	9	6	7	6	34	0.03	0.04	0.02	0.02	0.02
Stm/feed flow mismch	12	4	1	2	3	22	0.05	0.02	0.00	0.01	0.01
Low RCS flow	5	3	7	3	2	20	0.02	0.01	0.03	0.01	0.01
Safety injection	9	2	3	3	1	18	0.04	0.01	0.01	0.01	0.00
Other	2	3	5	0	7	17	0.01	0.01	0.02	0.00	0.02
High reactor power	5	3	1	1	2	12	0.02	0.01	0.00	0.00	0.01
Loss of power	4	3	2	1	1	11	0.02	0.01	0.01	0.00	0.00
IRM	2	1	4	1	2	10	0.01	0.00	0.01	0.00	0.01
High RCS temp.	4	1	2	1	1	9	0.02	0.00	0.01	0.00	0.00
PZR low pressure	3	1	1	0	3	8	0.01	0.00	0.00	0.00	0.01
Rx trip breaker open	5	1	2	0	0	8	0.02	0.00	0.01	0.00	0.00
High SG level	3	1	1	1	1	7	0.01	0.00	0.00	0.00	0.00
P-7 permissive	3	2	0	2	0	7	0.01	0.01	0.00	0.01	0.00
RCP breaker open	2	0	2	0	2	6	0.01	0.00	0.01	0.00	0.01
PZR high pressure	4	0	0	0	1	5	0.02	0.00	0.00	0.00	0.00
RCP under voltage	0	3	0	0	0	3	0.00	0.01	0.00	0.00	0.00
High RCS pressure	0	3	0	0	0	3	0.00	0.01	0.00	0.00	0.00
MSIV closure	0	1	0	2	0	3	0.00	0.00	0.00	0.01	0.00
Over power Δ temp.	0	0	0	2	1	3	0.00	0.00	0.00	0.01	0.00
Low neg. Rx power	1	0	1	0	0	2	0.00	0.00	0.00	0.00	0.00
Low RCS pressure	0	1	0	1	0	2	0.00	0.00	0.00	0.00	0.00
Low steamline press.	0	0	0	1	1	2	0.00	0.00	0.00	0.00	0.00
SRM	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
PZR high level	1	0	0	0	0	1	0.00	0.00	0.00	0.00	0.00
High contain. press.	1	0	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Low condenser vac.	0	1	0	0	0	1	0.00	0.00	0.00	0.00	0.00
Total	152	99	100	100	105	566	0.70	0.42	0.41	0.33	0.33

¹Critical hours: 1986 = 218,327.8; 1987 = 234,631.5; 1988 = 270,463.7; 1989 = 303,642.2; and 1990 = 316,721.8.

Table A-2.5 (cont.)

Mature General Electric Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Turbine trip	21	27	27	28	21	124	0.14	0.16	0.14	0.14	0.09
Low Rx water level	18	27	12	10	13	80	0.12	0.16	0.06	0.05	0.06
Manual	12	6	9	15	18	60	0.08	0.04	0.05	0.07	0.08
High Rx power	9	7	6	10	6	36	0.06	0.04	0.03	0.05	0.03
MSIV closure	14	6	3	7	6	36	0.09	0.04	0.02	0.03	0.03
IRM	8	4	3	3	2	20	0.05	0.02	0.02	0.01	0.01
High RCS pressure	4	4	1	2	4	15	0.03	0.02	0.01	0.01	0.02
Low condenser vac.	4	1	1	1	1	8	0.03	0.01	0.01	0.00	0.00
Other	0	1	2	1	4	8	0.00	0.01	0.01	0.00	0.02
Loss of power	0	2	1	1	0	4	0.00	0.01	0.01	0.00	0.00
MS line radiation	1	1	1	1	0	4	0.01	0.01	0.01	0.00	0.00
ATWS	0	0	1	0	0	1	0.00	0.00	0.01	0.00	0.00
High Rx water level	0	0	0	0	1	1	0.00	0.00	0.00	0.00	0.00
Total	93	86	68	79	76	402	0.61	0.50	0.37	0.39	0.34

²Critical Hours: 1986 = 152,007.9; 1987 = 170,556.6; 1988 = 186,252.0; 1989 = 202,523.0; and 1990 = 224,059.4.

Mature Combustion Engineering Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Manual	9	14	4	5	7	39	0.12	0.17	0.04	0.06	0.08
Low SG level	10	13	7	4	1	35	0.13	0.15	0.07	0.05	0.01
Turbine trip	11	8	5	5	1	30	0.15	0.09	0.05	0.06	0.01
Low DNBR	8	1	3	3	5	20	0.11	0.01	0.03	0.04	0.06
High RCS pressure	3	3	1	1	0	8	0.04	0.04	0.01	0.01	0.00
Loss of power	2	5	1	0	0	8	0.03	0.06	0.01	0.00	0.00
Other	0	3	1	1	1	6	0.00	0.04	0.01	0.01	0.01
High SG level	2	3	0	1	0	6	0.03	0.04	0.00	0.01	0.00
PZR high pressure	1	1	1	0	2	5	0.01	0.01	0.01	0.00	0.02
High reactor power	2	1	0	0	1	4	0.03	0.01	0.00	0.00	0.01
Low RCS flow	2	1	0	0	0	3	0.03	0.01	0.00	0.00	0.00
MSIV closure	0	0	0	0	2	2	0.00	0.00	0.00	0.00	0.02
RCP under voltage	0	0	1	0	0	1	0.00	0.00	0.01	0.00	0.00
RCP under freq./spd.	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
Low condenser vac.	1	0	0	0	0	1	0.01	0.00	0.00	0.00	0.00
Safety injection	0	0	1	0	0	1	0.00	0.00	0.01	0.00	0.00
Total	52	53	25	20	20	170	0.69	0.63	0.26	0.25	0.23

³Critical hours: 1986 = 75,737.5; 1987 = 84,585.2; 1988 = 94,353.6; 1989 = 79,767.5; and 1990 = 85,829.4.

Table A-2.5 (cont.)

Mature Babcock & Wilcox Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ⁴				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Turbine trip	8	5	2	7	0	22	0.22	0.10	0.04	0.13	0.00
High RCS pressure	4	5	5	5	2	21	0.11	0.10	0.10	0.10	0.04
High reactor power	1	2	1	0	2	6	0.03	0.04	0.02	0.00	0.04
Other	0	1	2	2	0	5	0.00	0.02	0.04	0.04	0.00
Manual	1	0	1	1	1	4	0.03	0.00	0.02	0.02	0.02
RCP under voltage	0	1	1	1	0	3	0.00	0.02	0.02	0.02	0.00
Low RCS pressure	0	0	1	0	2	3	0.00	0.00	0.02	0.00	0.04
Stm/feed flow mismch	0	0	0	1	0	1	0.00	0.00	0.00	0.02	0.00
High SG level	1	0	0	0	0	1	0.03	0.00	0.00	0.00	0.00
Low SG level	1	0	0	0	0	1	0.03	0.00	0.00	0.00	0.00
High RCS temp.	0	0	0	1	0	1	0.00	0.00	0.00	0.02	0.00
Power/flow imbalance	1	0	0	0	0	1	0.03	0.00	0.00	0.00	0.00
Loss of power	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
Total	17	15	13	18	7	70	0.46	0.31	0.25	0.34	0.15

⁴Critical hours: 1986 = 36,712.2; 1987 = 48,711.0; 1988 = 51,033.2; 1989 = 52,332.1; and 1990 = 48,234.5.

Table A-2.6 New Plant Reactor Scram Initiating Systems

New Plant Westinghouse Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	28	41	12	4	10	95	0.77	0.98	0.28	0.16	0.41
RPS	7	12	5	3	4	31	0.19	0.29	0.12	0.12	0.17
Turbine	4	20	1	7	3	35	0.11	0.48	0.02	0.28	0.12
Electrical	2	2	4	3	1	12	0.06	0.05	0.09	0.12	0.04
Cntrl. Rod Drive	4	4	5	4	0	17	0.11	0.10	0.12	0.16	0.00
Main generator	2	5	7	2	1	17	0.06	0.12	0.16	0.08	0.04
Main steam	2	6	0	1	2	11	0.06	0.14	0.00	0.04	0.08
Support	2	3	2	0	0	7	0.06	0.07	0.05	0.00	0.00
RCS	0	1	1	0	0	2	0.00	0.02	0.02	0.00	0.00
Condensate	3	9	4	1	0	17	0.08	0.21	0.09	0.04	0.00
Total	54	103	41	25	21	244	1.49	2.46	0.96	1.01	0.87

¹Critical Hours: 1986 = 36,193.6, 1987 = 41,916.9, 1988 = 42,555.3, 1989 = 24,852.4, and 1990 = 24,157.7.

New Plant General Electric Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	7	13	5	0	1	26	0.35	0.48	0.38	0.00	0.13
RPS	6	3	0	0	0	9	0.30	0.11	0.00	0.00	0.00
Turbine	8	5	3	0	0	16	0.40	0.19	0.23	0.00	0.00
Electrical	3	2	0	0	0	5	0.15	0.07	0.00	0.00	0.00
Cntrl. Rod Drive	1	3	0	0	0	4	0.05	0.11	0.00	0.00	0.00
Main generator	3	2	0	1	0	6	0.15	0.07	0.00	0.51	0.00
Main steam	3	1	0	0	1	5	0.15	0.04	0.00	0.00	0.13
Support	4	1	4	0	0	9	0.20	0.04	0.31	0.00	0.00
RCS	2	4	1	0	0	7	0.10	0.15	0.08	0.00	0.00
Condensate	3	5	0	0	0	8	0.15	0.19	0.00	0.00	0.00
Total	40	39	13	1	2	95	2.02	1.45	1.00	0.51	0.26

²Critical Hours: 1986 = 19,816.1, 1987 = 26,932.9, 1988 = 13,041.3, 1989 = 1,961.8, and 1990 = 7,727.3.

Table A-2.6 (cont.)

New Plant Combustion Engineering Reactors											
System	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Feedwater	7	1	0	0	NA	8	0.43	0.13	0.00	0.00	NA
RPS	6	2	0	0	NA	8	0.37	0.25	0.00	0.00	NA
Turbine	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Electrical	3	0	0	0	NA	3	0.18	0.00	0.00	0.00	NA
Cntrl. Rod Drive	5	0	0	0	NA	5	0.31	0.00	0.00	0.00	NA
Main generator	3	0	0	0	NA	3	0.18	0.00	0.00	0.00	NA
Main steam	2	0	0	1	NA	3	0.12	0.00	0.00	0.90	NA
Support	2	1	0	0	NA	3	0.12	0.13	0.00	0.00	NA
RCS	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Condensate	2	1	0	0	NA	3	0.12	0.13	0.00	0.00	NA
Total	30	5	0	1	NA	36	1.84	0.63	0.00	0.90	NA

³Critical Hours: 1986 = 16,312.7, 1987 = 7,931.1, 1988 = 8,369.7, 1989 = 1,106.1, and 1990 = 0.0.

Table A-2.7 New Plant Activities at Time of Reactor Scram

New Plant Westinghouse Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Nrml. Operatn.	19	41	18	15	15	108	0.52	0.98	0.42	0.60	0.62
Raising Power	12	18	6	2	2	40	0.33	0.43	0.14	0.08	0.08
Testing	5	18	6	3	2	34	0.14	0.43	0.14	0.12	0.08
Maintenance	5	11	6	1	1	24	0.14	0.26	0.14	0.04	0.04
Reducing Power	4	4	2	1	0	11	0.11	0.10	0.05	0.04	0.00
Troubleshooting	2	3	3	2	1	11	0.06	0.07	0.07	0.08	0.04
Special Testing	4	6	0	1	0	11	0.11	0.14	0.00	0.04	0.00
Calibration	3	2	0	0	0	5	0.08	0.05	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	54	103	41	25	21	244	1.49	2.46	0.96	1.01	0.87

¹Critical Hours: 1986 = 36,193.6, 1987 = 41,916.9, 1988 = 42,555.3, 1989 = 24,852.4, and 1990 = 24,157.7.

New Plant General Electric Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Nrml. Operatn.	10	16	5	1	1	33	0.50	0.59	0.38	0.51	0.13
Testing	7	9	1	0	0	17	0.35	0.33	0.08	0.00	0.00
Raising Power	8	3	2	0	0	13	0.40	0.11	0.15	0.00	0.00
Reducing Power	1	6	3	0	1	11	0.05	0.22	0.23	0.00	0.13
Maintenance	2	2	2	0	0	6	0.10	0.07	0.15	0.00	0.00
Special Testing	5	1	0	0	0	6	0.25	0.04	0.00	0.00	0.00
Troubleshooting	4	1	0	0	0	5	0.20	0.04	0.00	0.00	0.00
Calibration	3	1	0	0	0	4	0.15	0.04	0.00	0.00	0.00
Unknown	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Other	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	40	39	13	1	2	95	2.02	1.45	1.00	0.51	0.26

²Critical Hours: 1986 = 19,816.1, 1987 = 26,932.9, 1988 = 13,041.3, 1989 = 1,961.8, and 1990 = 7,727.3.

Table A-2.7 (cont.)

New Plant Combustion Engineering Reactors											
Activity	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Nrml. Opertn.	17	0	0	1	NA	18	1.04	0.00	0.00	0.90	NA
Testing	5	1	0	0	NA	6	0.31	0.13	0.00	0.00	NA
Raising Power	4	1	0	0	NA	5	0.25	0.13	0.00	0.00	NA
Special Testing	2	1	0	0	NA	3	0.12	0.13	0.00	0.00	NA
Reducing Power	1	1	0	0	NA	2	0.06	0.13	0.00	0.00	NA
Maintenance	1	0	0	0	NA	1	0.06	0.00	0.00	0.00	NA
Troubleshooting	0	1	0	0	NA	1	0.00	0.13	0.00	0.00	NA
Unknown	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Other	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Calibration	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Total	30	5	0	1	NA	36	1.84	0.63	0.00	0.90	NA

³Critical Hours: 1986 = 16,312.7 1987 = 7,931.1, 1988 = 8,369.7, 1989 = 1,106.1, and 1990 = 0.0.

Table A-2.8 New Plant Reactor Scram Causes

New Plant Westinghouse Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	21	51	23	15	9	119	0.58	1.22	0.54	0.60	0.3
Personnel Error	19	30	11	4	7	71	0.52	0.72	0.26	0.16	0.2
Procedure	5	9	4	1	0	19	0.14	0.21	0.09	0.04	0.0
Unknown	4	5	1	2	3	15	0.11	0.12	0.02	0.08	0.1
SG Level	3	6	0	0	0	9	0.08	0.14	0.00	0.00	0.0
Natural Phenom.	2	0	2	2	1	7	0.06	0.00	0.05	0.08	0.0
Other	0	2	0	1	1	4	0.00	0.05	0.00	0.04	0.0
Total	54	103	41	25	21	244	1.49	2.46	0.96	1.01	0.8

¹Critical Hours: 1986 = 36,193.6, 1987 = 41,916.9, 1988 = 42,555.3, 1989 = 24,852.4, and 1990 = 24,157.7.

New Plant General Electric Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	26	22	8	1	2	59	1.31	0.82	0.61	0.51	0.26
Personnel Error	8	11	3	0	0	22	0.40	0.41	0.23	0.00	0.00
Procedure	3	5	2	0	0	10	0.15	0.19	0.15	0.00	0.00
Unknown	2	1	0	0	0	3	0.10	0.04	0.00	0.00	0.00
Other	1	0	0	0	0	1	0.05	0.00	0.00	0.00	0.00
Natural Phenom.	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
SG Level	0	0	0	0	0	0	0.00	0.00	0.00	0.00	0.00
Total	40	39	13	1	2	95	2.02	1.45	1.00	0.51	0.26

²Critical Hours: 1986 = 19,816.1, 1987 = 26,932.9, 1988 = 13,041.3, 1989 = 1,961.8, and 1990 = 7,727.3.

New Plant Combustion Engineering Reactors											
Cause	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Equipment	21	3	0	1	NA	25	1.29	0.38	0.00	0.90	NA
Procedure	4	1	0	0	NA	5	0.25	0.13	0.00	0.00	NA
Personnel Error	3	1	0	0	NA	4	0.18	0.13	0.00	0.00	NA
Unknown	1	0	0	0	NA	1	0.06	0.00	0.00	0.00	NA
SG Level	1	0	0	0	NA	1	0.06	0.00	0.00	0.00	NA
Other	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Natural Phenom.	0	0	0	0	NA	0	0.00	0.00	0.00	0.00	NA
Total	30	5	0	1	NA	36	1.84	0.63	0.00	0.90	NA

³Critical Hours: 1986 = 16,312.7, 1987 = 7,931.1, 1988 = 8,369.7, 1989 = 1,106.1, and 1990 = 0.0.

Table A-2.9 New Plant Reactor Scram Signals

New Plant Westinghouse Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ¹				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Low SG level	28	35	8	3	5	79	0.77	0.83	0.19	0.12	0.21
Turbine trip	11	24	8	6	6	55	0.30	0.57	0.19	0.24	0.25
Manual	4	21	10	4	5	44	0.11	0.50	0.23	0.16	0.21
High neg. flux rate	3	2	7	3	1	16	0.08	0.05	0.16	0.12	0.04
High RCS temp.	0	4	1	3	0	8	0.00	0.10	0.02	0.12	0.00
SRM	1	3	1	0	1	6	0.03	0.07	0.02	0.00	0.04
High reactor power	1	1	1	2	0	5	0.03	0.02	0.02	0.08	0.00
Stm/feed flow mismch	1	3	0	1	0	5	0.03	0.07	0.00	0.04	0.00
Rx trip breaker open	1	1	0	1	2	5	0.03	0.02	0.00	0.04	0.08
Low RCS flow	1	1	1	1	0	4	0.03	0.02	0.02	0.04	0.00
Other	1	0	1	1	1	4	0.03	0.00	0.02	0.04	0.04
PZR low pressure	1	1	0	0	0	2	0.03	0.02	0.00	0.00	0.00
RCP under voltage	1	0	1	0	0	2	0.03	0.00	0.02	0.00	0.00
Loss of power	0	1	1	0	0	2	0.00	0.02	0.02	0.00	0.00
Safety injection	0	2	0	0	0	2	0.00	0.05	0.00	0.00	0.00
Low steamline press.	0	2	0	0	0	2	0.00	0.05	0.00	0.00	0.00
IRM	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
RCP breaker open	0	0	1	0	0	1	0.00	0.00	0.02	0.00	0.00
P-7 permissive	0	1	0	0	0	1	0.00	0.02	0.00	0.00	0.00
Total	54	103	41	25	21	244	1.49	2.46	0.96	1.01	0.87

¹Critical Hours: 1986 = 36,193.6, 1987 = 41,916.9, 1988 = 42,555.3, 1989 = 24,852.4, and 1990 = 24,157.7.

Table A-2.9 (cont.)

New Plant General Electric Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ²				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Low Rx wtr. level	12	5	4	0	0	21	0.61	0.19	0.31	0.00	0.00
Manual	5	11	4	0	0	20	0.25	0.41	0.31	0.00	0.00
Turbine trip	4	7	3	1	1	16	0.20	0.26	0.23	0.51	0.13
IRM	6	2	1	0	0	9	0.30	0.07	0.08	0.00	0.00
High Rx power	5	3	1	0	0	9	0.25	0.11	0.08	0.00	0.00
High RCS press.	3	2	0	0	0	5	0.15	0.07	0.00	0.00	0.00
High Rx wtr. level	0	5	0	0	0	5	0.00	0.19	0.00	0.00	0.00
Low condnsr. vac.	2	1	0	0	0	3	0.10	0.04	0.00	0.00	0.00
Other	3	0	0	0	0	3	0.15	0.00	0.00	0.00	0.00
MSIV closure	0	1	0	0	1	2	0.00	0.04	0.00	0.00	0.13
Hi Trp dischr. vol	0	1	0	0	0	1	0.00	0.04	0.00	0.00	0.00
MS line radiation	0	1	0	0	0	1	0.00	0.04	0.00	0.00	0.00
Total	40	39	13	1	2	95	2.02	1.45	1.00	0.51	0.26

²Critical Hours: 1986 = 19,816.1 1987 = 26,932.9, 1988 = 13,041.3, 1989 = 1,961.8, and 1990 = 7,727.3.

New Plant Combustion Engineering Reactors											
Signal	Total Scrams						Scrams/1000 Critical Hours ³				
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990
Low DNBR	11	0	0	0	NA	11	0.67	0.00	0.00	0.00	NA
Low SG level	6	1	0	0	NA	7	0.37	0.13	0.00	0.00	NA
PZR high pressure	4	0	0	0	NA	4	0.25	0.00	0.00	0.00	NA
High Rx power	2	1	0	0	NA	3	0.12	0.13	0.00	0.00	NA
Low RCS flow	3	0	0	0	NA	3	0.18	0.00	0.00	0.00	NA
Turbine trip	2	0	0	0	NA	2	0.12	0.00	0.00	0.00	NA
Other	0	2	0	0	NA	2	0.00	0.25	0.00	0.00	NA
Manual	0	1	0	0	NA	1	0.00	0.13	0.00	0.00	NA
Loss of power	1	0	0	0	NA	1	0.06	0.00	0.00	0.00	NA
MSIV closure	1	0	0	0	NA	1	0.06	0.00	0.00	0.00	NA
Low stmlne press.	0	0	0	1	NA	1	0.00	0.00	0.00	0.90	NA
Total	30	5	0	1	NA	36	1.84	0.63	0.00	0.90	NA

³Critical Hours: 1986 = 16,312.7 1987 = 7,931.1, 1988 = 8,369.7, 1989 = 1,106.1, and 1990 = 0.0.

Table A-2.10 Engineered Safety Features Actuations

Plant Name	Vendor	1986	1987	1988	1989	1990	Total	Average Yearly Rate
Arkansas 1	BW	0	3	2	49	4	58	11.60
Arkansas 2	CE	1	0	4	7	3	15	3.00
Beaver Valley 1	WE	2	13	11	11	11	48	9.60
Beaver Valley 2	WE	NYL	15	7	15	17	54	13.50
Big Rock Point	GE	0	0	2	0	0	2	0.40
Braidwood 1	WE	2	21	15	11	13	62	12.40
Braidwood 2	WE	NYL	0	11	1	6	18	4.50
Browns Ferry 1	GE	36	24	40	20	13	133	26.60
Browns Ferry 2	GE	32	16	39	17	7	111	22.20
Browns Ferry 3	GE	31	21	27	15	7	101	20.20
Brunswick 1	GE	27	21	13	14	15	90	18.00
Brunswick 2	GE	18	9	14	17	20	78	15.60
Byron 1	WE	16	8	5	3	3	35	7.00
Byron 2	WE	0	7	2	2	6	17	3.40
Callaway	WE	20	12	7	14	4	57	11.40
Calvert Cliffs 1	CE	1	3	5	2	4	15	3.00
Calvert Cliffs 2	CE	2	3	0	0	1	6	1.20
Catawba 1	WE	37	11	5	9	2	64	12.80
Catawba 2	WE	44	19	13	4	3	83	16.60
Clinton 1	GE	24	43	10	17	4	98	19.60
Comanche Peak 1*	WE	NYL	NYL	NYL	NYL	14	14	14.00
Cook 1	WE	6	1	0	0	1	8	1.60
Cook 2	WE	3	2	1	1	2	9	1.80
Cooper Station	GE	22	17	22	14	7	82	16.40
Crystal River 3	BW	3	6	6	7	1	23	4.60
Davis-Besse	BW	4	4	4	1	7	20	4.00
Diablo Canyon 1	WE	13	11	12	8	4	48	9.60
Diablo Canyon 2	WE	12	12	17	5	6	52	10.40
Dresden 2	GE	6	5	7	13	9	40	8.00
Dresden 3	GE	15	8	8	5	4	40	8.00
Duane Arnold	GE	14	23	13	19	16	85	17.00
Farley 1	WE	4	0	2	1	0	7	1.40
Farley 2	WE	0	3	0	1	1	5	1.00
Fermi 2	GE	30	34	24	21	7	116	23.20
FitzPatrick	GE	6	13	4	11	15	49	9.80
Fort Calhoun	CE	1	12	10	2	7	32	6.40

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990	Total	Average Yearly Rate
Fort St. Vrain ¹	GA	10	46	5	6	NLI	67	16.75
Ginna	WE	1	2	5	10	5	23	4.60
Grand Gulf	GE	20	12	6	11	11	60	12.00
Haddam Neck	WE	8	4	0	1	2	15	3.00
Hatch 1	GE	31	18	9	8	9	75	15.00
Hatch 2	GE	23	13	12	5	6	59	11.80
Hope Creek	GE	55	38	15	16	14	138	27.60
Indian Point 2	WE	6	4	2	3	22	37	7.40
Indian Point 3	WE	2	4	1	1	0	8	1.60
Kewaunee	WE	4	10	13	9	7	43	8.60
LaSalle 1	GE	30	20	16	10	5	81	16.20
LaSalle 2	GE	9	11	7	11	8	46	9.20
Limerick 1	GE	39	49	28	27	12	155	31.00
Limerick 2*	GE	NYL	NYL	NYL	7	14	21	10.50
Maine Yankee	CE	0	0	3	0	1	4	0.80
McGuire 1	WE	2	5	12	7	3	29	5.80
McGuire 2	WE	8	1	6	2	2	19	3.80
Millstone 1	GE	5	8	6	4	2	25	5.00
Millstone 2	CE	7	2	5	1	5	20	4.00
Millstone 3	WE	17	8	5	5	5	40	8.00
Monticello	GE	23	19	3	20	11	76	15.20
Nine Mile Pt. 1	GE	12	11	4	8	12	47	9.40
Nine Mile Pt. 2	GE	17	54	36	18	13	138	27.60
North Anna 1	WE	3	5	1	4	0	13	2.60
North Anna 2	WE	2	3	3	4	4	16	3.20
Oconee 1	BW	0	0	0	0	1	1	0.20
Oconee 2	BW	2	2	0	1	0	5	1.00
Oconee 3	BW	1	1	0	1	1	4	0.80
Oyster Creek	GE	23	14	10	7	2	56	11.20
Palisades	CE	15	11	10	3	6	45	9.00
Palo Verde 1	CE	19	6	3	4	3	35	7.00
Palo Verde 2	CE	17	4	5	3	3	32	6.40
Palo Verde 3	CE	NYL	3	1	7	1	12	3.00
Peach Bottom 2	GE	13	23	14	11	15	76	15.20
Peach Bottom 3	GE	19	12	12	3	10	56	11.20
Perry	GE	89	53	21	15	15	193	38.60

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990	Total	Average Yearly Rate
Pilgrim	GE	7	17	17	21	10	72	14.40
Point Beach 1	WE	3	1	0	2	3	9	1.80
Point Beach 2	WE	2	2	2	4	2	12	2.40
Prairie Island 1	WE	2	7	3	25	7	44	8.80
Prairie Island 2	WE	4	1	0	2	4	11	2.20
Quad Cities 1	GE	19	14	6	6	9	54	10.80
Quad Cities 2	GE	8	12	13	2	3	38	7.60
Rancho Seco ^{II}	BW	9	2	54	2	NLI	67	16.75
River Bend	GE	35	12	17	23	21	108	21.60
Robinson 2	WE	3	1	2	1	0	7	1.40
Salem 1	WE	0	0	0	11	36	47	9.40
Salem 2	WE	3	0	2	11	40	56	11.20
San Onofre 1	WE	3	3	2	3	2	13	2.60
San Onofre 2	CE	42	60	26	7	6	141	28.20
San Onofre 3	CE	7	20	7	6	2	42	8.40
Seabrook [*]	WE	2	9	3	7	10	31	6.20
Sequoyah 1	WE	16	11	22	11	13	73	14.60
Sequoyah 2	WE	14	9	16	5	9	53	10.60
Shearon Harris	WE	4	26	8	6	3	47	9.40
Shoreham	GE	19	10	NLI	NLI	NLI	29	14.50
South Texas 1	WE	NYL	14	22	9	6	51	12.75
South Texas 2 [*]	WE	NYL	NYL	1	14	8	23	7.67
St. Lucie 1	CE	2	4	2	0	3	11	2.20
St. Lucie 2	CE	0	2	0	1	2	5	1.00
Summer	WE	4	8	2	5	4	23	4.60
Surry 1	WE	8	21	4	13	4	50	10.00
Surry 2	WE	7	2	4	12	0	25	5.00
Susquehanna 1	GE	21	16	12	18	7	74	14.80
Susquehanna 2	GE	8	10	11	12	1	42	8.40
Three Mile Isl. 1	BW	2	2	0	1	2	7	1.40
Trojan	WE	1	7	17	9	9	43	8.60
Turkey Point 3	WE	4	12	9	5	3	33	6.60
Turkey Point 4	WE	20	14	5	4	3	46	9.20
Vermont Yankee	GE	2	7	2	14	8	33	6.60
Vogtle 1	WE	NYL	58	14	6	4	82	20.50
Vogtle 2 [*]	WE	NYL	NYL	NYL	14	4	18	9.00

Footnotes at end of table.

Table A-2.10 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990	Total	Average Yearly Rate
Wash. Nuclear 2	GE	26	15	10	16	12	79	15.80
Waterford 3	CE	14	4	7	5	10	40	8.00
Wolf Creek	WE	36	35	19	6	12	108	21.60
Yankee-Rowe	WE	2	3	5	2	2	14	2.80
Zion 1	WE	8	2	6	4	2	22	4.40
Zion 2	WE	8	4	3	0	8	23	4.60
Total All Plants ⁱⁱⁱ		1362	1334	996	915	768	5375	
Total All WE Plants		366	431	327	323	341	1788	
Total All GE Plants		814	702	510	476	354	2856	
Total All CE Plants		128	134	88	48	57	455	
Total All BW Plants		21	20	66	62	16	185	

ⁱIndicates 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.

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

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

ⁱⁱⁱRancho Seco ceased commercial operations on June 6, 1989.

^{iv}This total includes Fort St. Vrain.

Table A-2.11 Engineered Safety Feature Actuations of Selected Systems*

Westinghouse Reactors						
System	1986	1987	1988	1989	1990	Total
ECCS	45	38	36	35	21	175
Emergency Power	70	81	52	56	53	312
HVAC	133	191	146	151	169	790
Total	248	310	234	242	243	1277
General Electric Reactors						
System	1986	1987	1988	1989	1990	Total
ECCS	60	47	35	31	36	209
Emergency Power	65	52	44	42	26	229
HVAC	458	365	315	242	198	1578
RWCU	219	219	145	137	100	820
Total	802	683	539	452	360	2836
Combustion Engineering Reactors						
System	1986	1987	1988	1989	1990	Total
ECCS	15	8	13	15	7	58
Emergency Power	22	12	22	9	15	80
HVAC	73	104	57	29	32	295
Total	110	124	92	53	54	433
Babcock & Wilcox Reactors						
System	1986	1987	1988	1989	1990	Total
ECCS	6	6	4	4	6	26
Emergency Power	6	9	2	9	4	30
HVAC	5	6	52	39	10	112
Total	17	21	58	52	20	168

* ECCS - systems include high pressure coolant injection, low pressure core spray, low pressure coolant injection, low pressure safety injection, high pressure safety injection, high pressure core spray, accumulators, and isolation condensers.

Emergency Power - systems include all diesel starts.

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-2.12 Engineered Safety Feature Actuation Activities

Westinghouse Reactors						
Activity	1986	1987	1988	1989	1990	Total
Normal Operation	230	262	174	179	210	1055
Testing	80	106	101	98	90	475
Maintenance	51	63	51	43	41	249
Other	4	0	0	1	0	5
Unknown	1	0	0	2	0	3
Total	366	431	326	323	341	1787

General Electric Reactors						
Activity	1986	1987	1988	1989	1990	Total
Normal Operation	522	386	253	254	212	1627
Testing	181	206	155	147	77	766
Maintenance	108	108	101	71	65	453
Other	1	0	0	4	0	5
Unknown	2	0	0	0	0	2
Total	814	700	509	476	354	2853

Combustion Engineering Reactors						
Activity	1986	1987	1988	1989	1990	Total
Normal Operation	85	91	46	29	27	278
Testing	29	23	19	12	18	101
Maintenance	14	20	23	12	12	81
Unknown	0	0	0	0	0	0
Other	0	0	0	0	0	0
Total	128	134	88	53	57	460

Babcock & Wilcox Reactors						
Activity	1986	1987	1988	1989	1990	Total
Normal Operation	13	14	61	30	8	126
Testing	5	1	4	22	3	35
Maintenance	1	5	1	5	5	17
Other	2	0	0	0	0	2
Unknown	0	0	0	0	0	0
Total	21	20	66	57	16	180

Table A-2.13 Engineered Safety Feature Actuation Causes

Westinghouse Reactors						
Cause	1986	1987	1988	1989	1990	Total
Equipment	167	207	141	147	134	796
Personnel Error	98	144	106	99	91	538
Procedure	42	45	42	39	42	210
Unknown	28	21	19	18	43	129
Other	24	11	15	14	27	91
Natural Phenomenon	7	3	3	6	4	23
Total	366	431	326	323	341	1787
General Electric Reactors						
Cause	1986	1987	1988	1989	1990	Total
Equipment	354	298	232	197	131	1212
Personnel Error	231	217	150	151	94	843
Procedure	88	97	68	63	46	362
Unknown	74	40	23	39	35	211
Other	48	26	17	23	43	157
Natural Phenomenon	19	22	19	3	5	68
Total	814	700	509	476	354	2853
Combustion Engineering Reactors						
Cause	1986	1987	1988	1989	1990	Total
Equipment	58	85	46	26	17	232
Personnel Error	44	21	26	14	19	124
Procedure	9	12	12	6	6	45
Unknown	7	6	3	3	10	29
Other	9	10	1	2	5	27
Natural Phenomenon	1	0	0	2	0	3
Total	128	134	88	53	57	460
Babcock & Wilcox Reactors						
Cause	1986	1987	1988	1989	1990	Total
Equipment	4	8	53	34	5	104
Personnel Error	9	8	8	12	8	45
Unknown	6	2	4	8	0	20
Procedure	1	2	1	1	3	8
Other	1	0	0	1	0	2
Natural Phenomenon	0	0	0	1	0	1
Total	21	20	66	57	16	180

Table A-2.14 Technical Specification Shutdowns

Plant Name	Number of TS Required Shutdowns						TS Required Shutdown Hours					
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990	Total
Arkansas 1	1	0	1	1	2	5	381	0	183	425	323	1292
Arkansas 2	0	2	1	0	1	4	0	1024	362	0	79	1465
Beaver Valley 1	0	0	2	0	2	4	0	0	240	0	29	269
Beaver Valley 2	NYL	2	1	1	0	4	NYL	83	19	166	0	268
Big Rock Point	1	0	3	0	1	5	137	0	221	0	59	417
Braidwood 1	0	2	1	1	3	7	0	41	683	81	1075	1880
Braidwood 2	NYL	0	2	0	0	2	NYL	0	75	0	0	75
Browns Ferry 1	0	0	0	0	0	0	0	0	0	0	0	0
Browns Ferry 2	0	0	0	0	0	0	0	0	0	0	0	0
Browns Ferry 3	0	0	0	0	0	0	0	0	0	0	0	0
Brunswick 1	0	0	1	1	0	2	0	0	200	81	0	261
Brunswick 2	1	0	0	0	0	1	60	0	0	0	0	60
Byron 1	1	0	1	0	1	3	508	0	105	0	40	653
Byron 2	0	0	0	0	0	0	0	0	0	0	0	0
Callaway	0	1	0	1	0	2	0	39	0	20	0	59
Calvert Cliffs 1	1	1	0	0	0	2	152	130	0	0	0	282
Calvert Cliffs 2	0	0	0	1	0	1	0	0	0	35	0	35
Catawba 1	2	4	2	2	1	11	739	420	93	181	220	1653
Catawba 2	0	5	2	0	0	7	0	1214	347	0	0	1561
Clinton 1	0	0	1	1	2	4	0	0	0	561	361	942
Comanche Peak 1*	NYL	NYL	NYL	NYL	1	1	NYL	NYL	NYL	NYL	32	32
Cook 1	0	1	1	1	0	3	0	330	68	75	0	473
Cook 2	0	0	0	0	2	2	0	0	0	0	364	364
Cooper Station	0	1	0	0	0	1	0	194	0	0	0	194
Crystal River 3	0	0	0	3	2	5	0	0	0	1293	199	1492
Davis-Besse	0	0	0	0	0	0	0	0	0	0	0	0
Diablo Canyon 1	0	0	0	0	0	0	0	0	0	0	0	0
Diablo Canyon 2	0	0	0	0	0	0	0	0	0	0	0	0
Dresden 2	1	3	0	0	0	4	140	121	0	0	0	261
Dresden 3	2	5	0	0	0	7	132	1547	0	0	0	1679
Duane Arnold	1	0	0	0	0	1	85	0	0	0	0	85
Farley 1	0	0	0	0	0	0	0	0	0	0	0	0

Footnotes at end of table.

Table A-2.14 (cont.)

Plant Name	Number of TS Required Shutdowns						TS Required Shutdown Hours					
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990	Total
Farley 2	1	1	0	0	0	2	103	436	0	0	0	539
Fermi 2	1	3	3	0	0	7	63	407	1444	0	0	1914
Fitzpatrick	0	0	0	0	0	0	0	0	0	0	0	0
Fort Calhoun	0	0	0	0	0	0	0	0	0	0	0	0
Fort St. Vrain ¹	0	1	0	2	PSD	3	0	1044	0	496	PSD	1540
Ginna	0	0	1	1	0	2	0	0	230	301	0	531
Grand Gulf	3	0	0	0	0	3	365	0	0	0	0	365
Haddam Neck	1	0	0	0	1	2	496	0	0	0	455	953
Hatch 1	1	0	0	0	0	1	83	0	0	0	0	83
Hatch 2	1	1	1	0	0	3	47	77	30	0	0	154
Hope Creek	1	1	1	1	0	4	64	70	0	72	0	226
Indian Point 2	2	0	1	0	0	3	67	0	121	0	0	188
Indian Point 3	0	0	1	0	1	2	0	0	810	0	38	848
Kewaunee	0	0	0	0	0	0	0	0	0	0	0	0
La Crosse ²	0	0	PSD	PSD	PSD	0	0	0	PSD	PSD	PSD	0
LaSalle 1	0	1	0	0	0	1	0	137	0	0	0	137
LaSalle 2	0	0	1	0	0	1	0	0	60	0	0	60
Limerick 1	0	0	0	0	1	1	0	0	0	0	175	175
Limerick 2 ³	NYL	NYL	NYL	0	0	0	NYL	NYL	NYL	0	0	0
Maine Yankee	0	0	0	1	1	2	0	0	0	13	540	553
McGuire 1	3	1	1	1	1	7	847	18	430	1507	21	2623
McGuire 2	3	1	0	0	1	5	542	48	0	0	22	612
Millstone 1	0	0	1	1	1	3	0	0	141	152	102	395
Millstone 2	0	1	2	0	0	3	0	435	332	0	0	767
Millstone 3	0	1	0	1	0	2	0	126	0	221	0	347
Monticello	0	0	0	0	0	0	0	0	0	0	0	0
Nine Mile Point 1	2	0	0	0	1	3	250	0	0	0	34	284
Nine Mile Point 2	0	1	0	0	0	1	0	36	0	0	0	36
North Anna 1	1	0	0	1	0	2	64	0	0	3317	0	3381
North Anna 2	2	0	0	0	0	2	316	0	0	0	0	316
Oconee 1	0	0	0	0	0	0	0	0	0	0	0	0
Oconee 2	0	0	0	0	0	0	0	0	0	0	0	0

Footnotes at end of table.

Table A-2.14 (cont.)

Plant Name	Number of TS Required Shutdowns						TS Required Shutdown Hours					
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990	Total
Oconee 3	0	1	0	0	0	1	0	100	0	0	0	100
Oyster Creek	1	1	2	3	1	8	86	531	2290	242	319	3468
Palisades	1	1	0	1	0	3	72	2	0	72	0	146
Palo Verde 1	0	1	0	0	0	1	0	117	0	0	0	117
Palo Verde 2	2	1	0	0	0	3	23	132	0	0	0	155
Palo Verde 3	NYL	0	0	1	1	2	NYL	0	0	370	43	413
Peach Bottom 2	1	0	0	1	3	5	178	0	0	172	342	692
Peach Bottom 3	2	0	0	0	0	2	227	0	0	0	0	227
Perry	1	1	0	0	0	2	27	252	0	0	0	279
Pilgrim	2	0	0	3	1	6	515	0	0	1053	192	1760
Point Beach 1	0	0	0	0	1	1	0	0	0	0	218	218
Point Beach 2	0	0	0	0	0	0	0	0	0	0	0	0
Prairie Island 1	0	0	0	0	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0	0	0	0	0
Quad Cities 1	1	0	1	0	1	3	97	0	42	0	80	219
Quad Cities 2	1	1	0	1	0	3	28	59	0	23	0	110
Rancho Seco ^a	0	0	1	1	NLO	2	0	0	111	1074	PSD	1185
River Bend	1	2	1	0	0	4	80	140	54	0	0	274
Robinson 2	0	3	4	1	0	8	0	179	1173	2986	0	4318
Salem 1	0	0	0	2	3	5	0	0	0	272	1588	1858
Salem 2	0	0	0	2	1	3	0	0	0	185	182	327
San Onofre 1	2	0	0	1	0	3	638	0	0	48	0	686
San Onofre 2	0	0	1	1	1	3	0	0	58	728	41	827
San Onofre 3	0	0	0	1	0	1	0	0	0	220	0	220
Seabrook ^a	0	0	0	0	0	0	0	0	0	0	0	0
Sequoyah 1	0	0	0	0	1	1	0	0	0	0	324	324
Sequoyah 2	0	0	3	0	0	3	0	0	96	0	0	96
Shearon Harris	0	4	1	0	1	6	0	132	54	0	96	282
Shoreham ^a	1	0	PSD	PSD	PSD	1	19	0	PSD	PSD	PSD	19
South Texas 1	NYL	0	2	1	1	4	NYL	0	198	271	71	540
South Texas 2 ^a	NYL	NYL	0	1	2	3	NYL	NYL	0	60	516	576
St. Lucie 1	0	1	0	0	1	2	0	205	0	0	14	219

Footnotes at end of table.

Table A-2.14 (cont.)

Plant Name	Number of TS Required Shutdowns						TS Required Shutdown Hours					
	1986	1987	1988	1989	1990	Total	1986	1987	1988	1989	1990	Total
St. Lucie 2	0	1	0	0	0	1	0	20	0	0	0	20
Summer	1	1	0	1	0	3	56	24	0	232	0	312
Surry 1	0	2	3	0	0	5	0	486	2878	0	0	3144
Surry 2	1	0	0	1	1	3	318	0	0	313	618	1249
Susquehanna 1	1	2	0	2	0	5	637	161	0	241	0	1039
Susquehanna 2	2	1	0	1	0	4	727	99	0	137	0	963
Three Mile Isl 1	0	0	0	0	0	0	0	0	0	0	0	0
Trojan	0	0	0	0	2	2	0	0	0	0	50	50
Turkey Point 3	2	0	1	1	0	4	707	0	102	188	0	977
Turkey Point 4	4	3	1	0	0	8	122	175	412	0	0	709
Vermont Yankee	0	0	0	0	0	0	0	0	0	0	0	0
Vogtle 1	NYL	0	0	1	1	2	NYL	0	0	203	3	208
Vogtle 2 [*]	NYL	NYL	NYL	1	0	1	NYL	NYL	NYL	11	0	11
Wash. Nuclear 2	2	0	2	1	1	6	172	0	302	115	208	797
Waterford 3	0	0	1	0	0	1	0	0	39	0	0	39
Wolf Creek	1	3	0	0	0	4	141	55	0	0	0	196
Yankee-Rowe	1	1	0	1	1	4	20	3	0	100	865	988
Zion 1	1	0	1	2	3	7	10	0	232	155	903	1300
Zion 2	0	0	0	0	1	1	0	0	0	0	480	480
Total ^{***}	65	71	57	55	55	303	10563	10829	14015	18388	11319	65114

NYL Not Yet Licensed

NLO No Longer Operational

PSD Permanently Shut Down

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

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

^{*}La Crosse, a boiling water reactor designed by Allis Chalmers (AC), was permanently shut down on April 30, 1987.

^{*}Rancho Seco ceased commercial operation on June 8, 1989.

^{*}Shoreham ceased commercial operations on June 7, 1987.

^{***}This total includes Fort St. Vrain and La Crosse.

Table A-2.15 Annual Unit Operating Factors for 1990

Plant Name	Docket	Availability Factor	Service Factor	Maximum Dependable Capacity Factor	Design Electrical Rating Capacity Factor
Arkansas 1	313	73.5	73.5	56.3	55.4
Arkansas 2	368	93.8	93.8	94.9	89.2
Beaver Valley 1	334	92.2	92.2	86.9	84.3
Beaver Valley 2	412	76.9	76.9	59.1	58.5
Big Rock Point	155	76.7	76.7	72.5	67.5
Braidwood 1	456	88.8	88.8	84.2	84.2
Braidwood 2	457	78.2	78.2	64.6	64.6
Browns Ferry 1	259	0.0	0.0	0.0	0.0
Browns Ferry 2	260	0.0	0.0	0.0	0.0
Browns Ferry 3	296	0.0	0.0	0.0	0.0
Brunswick 1	325	67.5	67.5	62.4	60.1
Brunswick 2	324	64.1	64.1	58.5	56.3
Byron 1	454	80.6	80.6	71.6	70.7
Byron 2	455	75.3	75.3	62.1	61.3
Callaway	483	81.8	81.8	81.2	78.0
Calvert Cliffs 1	317	21.0	21.0	18.6	18.2
Calvert Cliffs 2	318	0.0	0.0	0.0	0.0
Catawba 1	413	71.7	71.7	69.5	68.5
Catawba 2	414	68.3	68.3	65.1	64.2
Clinton 1	461	52.6	52.6	44.1	44.0
Comanche Peak 1*	445	84.9	84.9	65.5	64.7
Cook 1	315	79.2	79.2	70.5	69.8
Cook 2	316	55.4	55.4	51.8	50.0
Cooper Station	298	78.9	78.9	76.4	75.0
Crystal River 3	302	61.9	61.9	57.6	57.3
Davis-Besse	346	55.6	55.6	54.4	52.4
Diablo Canyon 1	275	96.2	96.2	92.7	91.6
Diablo Canyon 2	323	83.2	83.2	79.4	77.1
Dresden 2	237	67.6	67.6	60.3	58.7
Dresden 3	249	83.0	83.0	75.9	73.9
Duane Arnold	331	74.2	74.2	63.9	63.9
Farley 1	348	99.1	99.1	95.7	95.1
Farley 2	364	74.0	74.0	72.4	72.3
Fermi 2	341	83.0	83.0	74.2	75.6
FitzPatrick	333	69.1	69.1	67.2	64.4
Fort Calhoun	285	61.9	61.9	57.7	57.7
Ginna	244	83.6	83.6	83.8	83.8
Grand Gulf	416	77.2	77.2	74.0	67.6
Haddam Neck	213	29.6	29.6	23.3	22.6
Hatch 1	321	65.3	65.3	61.8	59.9

Footnotes at end of table.

Table A-2.15 (cont.)

Plant Name	Docket	Availability Factor	Service Factor	Maximum Dependable Capacity Factor	Design Electrical Rating Capacity Factor
Hatch 2	366	98.7	98.7	97.3	95.0
Hope Creek	354	90.7	90.7	89.7	86.7
Indian Point 2	247	66.0	66.0	66.5	61.6
Indian Point 3	286	61.4	61.4	59.5	59.5
Kewaunee	305	87.5	87.5	88.5	83.2
LaSalle 1	373	95.1	95.1	95.2	91.5
LaSalle 2	374	70.4	70.4	68.1	65.5
Limerick 1	352	65.4	65.4	60.8	60.8
Limerick 2*	353	83.5	83.5	79.8	79.8
Maine Yankee	309	66.9	66.9	66.8	66.0
McGuire 1	369	53.9	53.9	47.9	45.8
McGuire 2	370	67.1	67.1	65.3	62.5
Millstone 1	245	90.2	90.2	88.9	88.1
Millstone 2	336	73.0	73.0	70.2	69.6
Millstone 3	423	89.0	89.0	82.5	81.3
Monticello	263	96.1	96.1	96.0	94.4
Nine Mile Pt. 1	220	34.8	34.8	23.9	23.5
Nine Mile Pt. 2	410	53.6	53.6	43.5	43.3
North Anna 1	338	99.6	99.6	90.5	91.0
North Anna 2	339	79.7	79.7	74.9	75.2
Oconee 1	269	88.5	88.5	87.0	83.1
Oconee 2	270	85.3	85.3	84.6	80.8
Oconee 3	287	99.5	99.5	100.2	95.7
Oyster Creek	219	87.7	87.7	79.3	75.6
Palisades	255	57.9	57.9	47.0	42.7
Palo Verde 1	528	44.8	44.8	44.1	42.4
Palo Verde 2	529	60.2	60.2	58.4	56.1
Palo Verde 3	530	91.9	91.9	90.1	86.6
Peach Bottom 2	277	79.7	79.7	72.5	71.8
Peach Bottom 3	278	87.8	87.8	83.1	80.8
Perry	440	65.3	65.3	65.9	63.2
Pilgrim	293	77.5	77.5	72.3	74.0
Point Beach 1	266	84.1	84.1	83.1	81.1
Point Beach 2	301	88.1	88.1	89.3	87.1
Prairie Island 1	282	88.6	88.6	86.9	82.5
Prairie Island 2	306	86.8	86.8	86.8	81.9
Quad Cities 1	254	83.1	83.1	79.2	77.2
Quad Cities 2	265	70.6	70.6	64.6	63.0
River Bend	458	75.8	75.8	68.2	68.2
Robinson 2	261	64.1	64.1	56.9	54.0

Footnotes at end of table.

Table A-2.15 (cont.)

Plant Name	Docket	Availability Factor	Service Factor	Maximum Dependable Capacity Factor	Design Electrical Rating Capacity Factor
Salem 1	272	67.0	67.0	61.5	61.0
Salem 2	311	59.0	59.0	55.8	55.4
San Onofre 1	206	47.3	47.3	40.4	40.4
San Onofre 2	361	87.4	87.4	88.7	88.7
San Onofre 3	362	70.3	70.3	69.6	69.6
Seabrook*	443	85.5	85.5	82.4	82.6
Sequoyah 1	327	73.1	73.1	67.9	67.9
Sequoyah 2	328	78.4	78.4	71.4	71.4
Shearon Harris	400	89.2	89.2	84.1	80.4
South Texas 1	498	59.4	59.4	54.7	54.7
South Texas 2*	499	62.8	62.8	58.7	58.7
St. Lucie 1	335	62.4	62.4	61.1	61.8
St. Lucie 2	389	74.1	74.1	72.4	73.2
Summer	395	82.9	82.9	78.8	77.5
Surry 1	280	76.0	76.0	69.8	69.1
Surry 2	281	90.4	90.4	85.3	84.6
Susquehanna 1	387	74.6	74.6	71.1	70.0
Susquehanna 2	388	93.0	93.0	91.1	90.1
Three Mile Isl. 1	289	81.3	81.3	74.9	73.9
Trojan	344	64.6	64.6	63.3	61.4
Turkey Point 3	250	59.4	59.4	57.6	55.4
Turkey Point 4	251	76.4	76.4	75.1	72.2
Vermont Yankee	271	84.4	84.4	81.9	80.3
Vogtle 1	424	79.7	79.7	77.7	76.1
Vogtle 2*	425	81.4	81.4	70.5	71.1
Wash. Nuclear 2	397	65.7	65.7	59.9	59.6
Waterford 3	382	92.3	92.3	91.4	89.0
Wolf Creek	482	80.3	80.3	79.2	76.8
Yankee-Rowe	029	60.2	60.2	56.5	53.9
Zion 1	295	54.2	54.2	48.8	48.8
Zion 2	304	33.8	33.8	29.0	29.0
Industry Average		71.7	71.7	67.5	66.1

* An asterisk indicates a new plant.

Table A-2.16 All Licensed Reactor Critical Hours by Year

Plant Name	Vendor	1986	1987	1988	1989	1990
Arkansas 1	BW	5536.7	7855.7	6156.6	5999.1	6500.2
Arkansas 2	CE	6370.0	7715.4	6032.0	6610.1	8246.6
Beaver Valley 1	WE	6243.8	7339.4	7066.7	5887.6	8155.9
Beaver Valley 2	WE	NYC	2312.5	8283.8	6307.5	6790.5
Big Rock Point	GE	8387.3	6215.5	6394.2	6920.8	6759.0
Braidwood 1	WE	NYC	3426.2	5746.1	5586.8	7830.1
Braidwood 2	WE	NYC	NYC	4796.1	7618.1	6904.2
Browns Ferry 1	GE	0.0	0.0	0.0	0.0	0.0
Browns Ferry 2	GE	0.0	0.0	0.0	0.0	0.0
Browns Ferry 3	GE	0.0	0.0	0.0	0.0	0.0
Brunswick 1	GE	8317.7	5788.7	6660.7	5749.3	5948.2
Brunswick 2	GE	4232.4	8328.5	5645.8	5779.9	5926.6
Byron 1	WE	7820.9	6210.3	6485.1	8742.7	7144.2
Byron 2	WE	NYC	6812.5	8676.0	7060.4	6667.1
Callaway	WE	7307.6	6227.7	8202.1	7481.6	7365.0
Calvert Cliffs 1	CE	6906.2	6615.6	6398.5	1806.6	1924.5
Calvert Cliffs 2	CE	8442.0	5957.8	7827.1	1766.4	0.0
Catawba 1	WE	5425.2	6076.4	7070.3	7485.1	6348.9
Catawba 2	WE	2678.5	7212.8	6496.8	6448.2	6047.5
Clinton 1	GE	NYC	5350.3	7399.4	4244.3	4826.8
Comanche Peak 1*	WE	NYC	NYC	NYC	NYC	5302.5
Cook 1	WE	7536.4	6012.2	8433.8	6169.8	6944.8
Cook 2	WE	5560.5	6290.3	2715.5	6580.9	4958.9
Cooper Station	GE	6570.1	8424.2	5967.9	6672.9	6953.3
Crystal River 3	BW	3691.1	5333.6	7457.3	4274.4	5591.1
Davis-Besse	BW	178.0	7425.7	2126.7	8547.1	4966.6
Diablo Canyon 1	WE	5967.4	8475.7	5682.3	7189.1	8504.3
Diablo Canyon 2	WE	7296.3	6058.8	6190.7	8136.8	7432.9
Dresden 2	GE	7110.1	5763.7	6974.7	7252.5	5958.8
Dresden 3	GE	2766.4	7208.7	6346.3	7311.6	7453.4
Duane Arnold	GE	7348.2	5668.3	6609.9	6921.1	6641.2
Farley 1	WE	7276.4	8307.2	7428.3	7613.4	8695.9
Farley 2	WE	7549.7	6537.7	8784.0	7205.2	6501.1
Fermi 2	GE	1403.0	5147.5	5325.8	6002.4	7420.8
FitzPatrick	GE	8075.8	6161.3	6060.6	8086.8	6356.0
Fort Calhoun	CE	8485.2	6608.3	6510.0	7816.5	5622.4

Footnotes at end of table.

Table A-2.16 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990
Fort St. Vrain ¹	GA	2456.6	2908.9	3798.1	3331.9	PSD
Ginna	WE	7716.3	8014.5	7679.2	6648.5	7393.2
Grand Gulf	GE	5624.6	7203.3	8498.1	7005.5	6911.1
Haddam Neck	WE	5060.9	4728.9	6177.0	5883.3	2824.5
Hatch 1	GE	5521.2	7191.7	6008.8	8760.0	5939.6
Hatch 2	GE	6451.9	8519.6	6359.2	6495.8	8684.7
Hope Creek	GE	2669.8	7570.1	7089.5	6813.9	8020.0
Indian Point 2	WE	5101.9	6347.3	7491.8	5644.2	5837.0
Indian Point 3	WE	6581.6	5496.5	7312.7	5352.0	5511.3
Kewaunee	WE	7584.3	7860.9	7755.6	7436.2	7700.5
La Crosse	AC	4400.8	2805.4	PSD	PSD	PSD
LaSalle 1	GE	2395.7	5609.1	5931.1	6114.8	8475.3
LaSalle 2	GE	6614.0	4781.4	6648.2	6693.0	6343.1
Limerick 1	GE	7146.1	6151.0	8476.3	5784.5	6002.8
Limerick 2 ^o	GE	NYC	NYC	NYC	1961.8	7727.3
Maine Yankee	CE	7791.0	5724.4	6949.7	8210.0	6215.9
McGuire 1	WE	5022.2	6835.7	6783.8	7210.8	4807.9
McGuire 2	WE	5770.4	7046.9	7313.5	6943.4	5937.3
Millstone 1	GE	8276.5	6970.7	8661.6	7377.3	8021.0
Millstone 2	CE	6599.6	8242.0	6953.1	6027.7	6551.5
Millstone 3	WE	6201.1	6350.7	7196.3	6716.1	7909.2
Monticello	GE	6984.9	7173.6	8768.7	6679.1	8487.3
Nine Mile Pt. 1	GE	5823.5	8171.2	0.0	0.0	3365.8
Nine Mile Pt. 2	GE	NYC	2702.6	4525.4	5206.2	4810.4
North Anna 1	WE	7560.0	4585.4	8019.5	5023.1	8748.4
North Anna 2	WE	7301.3	6842.2	8734.9	6918.9	7012.2
Oconee 1	BW	5948.7	6913.9	8769.0	7371.0	7774.7
Oconee 2	BW	7253.7	8604.9	6989.2	7385.8	7505.7
Oconee 3	BW	7835.4	6142.2	7229.7	7682.9	8730.6
Oyster Creek	GE	2389.1	5620.0	5789.0	5015.2	7804.6
Palisades	CE	1490.5	4226.6	4990.4	6050.6	5143.1
Palo Verde 1	CE	5498.0	4589.1	5762.9	1522.0	4198.1
Palo Verde 2	CE	3803.1	6985.2	5750.0	4226.0	5376.2
Palo Verde 3	CE	NYC	945.9	8369.7	1209.5	8168.5
Peach Bottom 2	GE	7272.8	1729.8	0.0	5330.5	7173.4
Peach Bottom 3	GE	5919.3	1823.2	0.0	801.4	7844.1

Footnotes at end of table.

Table A-2.16 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990
Perry	GE	1429.6	4504.8	6939.2	4997.0	5880.4
Pilgrim	GE	1715.5	0.0	0.3	5613.8	7195.5
Point Beach 1	WE	7905.4	7389.4	7847.7	7728.3	7423.8
Point Beach 2	WE	7262.7	7583.1	7707.8	7243.6	7738.8
Prairie Island 1	WE	7898.4	7287.6	7835.6	8740.7	7840.4
Prairie Island 2	WE	7972.1	8760.0	7813.9	7852.4	7785.7
Quad Cities 1	GE	6151.3	6251.6	8477.9	6621.4	7318.1
Quad Cities 2	GE	5728.0	6941.4	6292.8	8434.7	6304.6
Rancho Seco ^{II}	BW	0.0	0.0	5543.8	2354.6	PSD
River Bend	GE	5305.7	5995.1	8280.0	6051.7	6835.2
Robinson 2	WE	7118.8	6354.3	5791.5	4262.0	5674.7
Salem 1	WE	7097.2	6412.5	6937.1	6276.4	6055.0
Salem 2	WE	5629.3	6423.0	5992.9	7650.0	5350.5
San Onofre 1	WE	2975.0	7382.9	3817.7	3582.8	4162.8
San Onofre 2	CE	6480.0	6192.6	8286.3	5227.0	7692.8
San Onofre 3	CE	7422.3	7135.2	5930.8	8251.6	6297.7
Seabrook*	WE	NYC	NYC	NYC	194.4	5524.9
Sequoyah 1	WE	0.0	0.0	379.5	8671.3	6576.8
Sequoyah 2	WE	0.0	0.0	5202.1	6343.8	6940.8
Shearon Harris	WE	NYC	6214.0	6585.1	6962.6	7848.6
Shoreham	GE	1299.5	0.0	0.0	2.7	0.0
South Texas 1	WE	NYC	NYC	5172.3	5750.7	5533.8
South Texas 2*	WE	NYC	NYC	NYC	4514.1	6004.6
St. Lucie 1	CE	8424.0	6971.6	7554.3	8290.1	5569.7
St. Lucie 2	CE	7326.7	7382.3	8784.0	6626.9	6691.4
Summer	WE	8453.2	6222.4	6067.7	7276.2	7346.3
Surry 1	WE	6233.2	6178.3	3755.2	4272.2	6723.4
Surry 2	WE	6171.1	6555.2	5028.3	1504.3	7973.7
Susquehanna 1	GE	6196.3	6464.6	8289.7	6592.5	6769.1
Susquehanna 2	GE	5946.6	8484.0	6156.9	6916.4	8197.5
Three Mile Isl. 1	BW	6268.6	6435.0	6760.9	8717.2	7165.6
Trojan	WE	7064.1	4730.5	5925.3	5423.2	5810.5
Turkey Point 3	WE	6988.4	1909.7	5408.1	5806.6	5283.7
Turkey Point 4	WE	3048.1	4503.2	5050.1	4147.1	6802.7
Vermont Yankee	GE	4359.6	7374.6	8404.4	7416.2	7522.8
Vogtle 1	WE	NYC	5385.7	6822.3	8413.0	7170.5

Footnotes at end of table.

Table A-2.16 (cont.)

Plant Name	Vendor	1986	1987	1988	1989	1990
Vogtle 2 ^a	WE	NYC	NYC	NYC	6134.5	7325.7
Wash. Nuclear 2	GE	6391.5	6199.4	6310.9	6857.8	5908.9
Waterford 3	CE	7011.6	7224.3	6624.5	7232.6	8131.0
Wolf Creek	WE	6523.7	6152.6	6117.6	8715.3	7096.1
Yankee-Rowe	WE	8343.5	7248.3	7486.8	8137.2	5390.7
Zion 1	WE	5491.0	6877.3	6747.9	5268.3	5097.0
Zion 2	WE	7783.5	5569.7	7004.6	8333.9	3122.7
Total All Plants ⁱⁱⁱ		561965.2	620979.5	669866.9	669517.0	706730.1
Total All WE Plants		254521.4	276548.4	313019.0	328494.6	340879.5
Total All GE Plants		171824.0	197489.5	199293.3	204484.8	231786.7
Total All CE Plants		92050.2	92516.3	102723.3	80873.6	85829.4
Total All BW Plants		30443.6	42276.0	44272.3	43614.9	41068.9

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

NYC Not Yet Critical.

PSD Permanently shutdown plant.

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

ⁱⁱRancho Seco ceased commercial operations on June 6, 1989.

ⁱⁱⁱThis total includes Fort St. Vrain.

Appendix B

Summary of 1990 Abnormal Occurrences (Reactors)

Abnormal Occurrences (AOs) for CY 1990 (Reactors)

NUREG-0090, Volume 13, No. 1,

Report No. 90-1

Loss of Vital ac Power with Subsequent Reactor Coolant System Heat-up at Vogtle Unit 1 During Shutdown

Early on March 20, 1990, Vogtle Unit 2 was operating at 100 percent power and Unit 1 had been in a refueling outage for about 25 days with its reactor coolant system (RCS) level reduced to "mid-loop" (below the top of the pressure vessel nozzles) to facilitate maintenance. Decay heat in the core was being removed by one train of the residual heat removal (RHR) system. Several pieces of equipment were out of service for maintenance, including one of Unit 1's two emergency diesel generators (EDGs). At 9:20 a.m., EST, a truck in the plant low-voltage switchyard, carrying gasoline, diesel oil, and lubricants, backed into a support column for the feeder line supplying power to the Unit 1 A reserve auxiliary transformer (RAT) and the Unit 2 B RAT. The insulator for the C phase of the feeder line fractured and initiated a phase-to-ground electrical fault. The fault resulted in a loss of power to the Unit 1 A RAT and the Unit 2 B RAT.

Because of an improperly connected differential current transformer, Unit 2 experienced a turbine trip and reactor trip; the trip was relatively uncomplicated and the licensee (Georgia Power Company) subsequently restarted Unit 2 without incident. However, considerable complications were experienced on Unit 1. The event involved a loss of all safety ac power to Unit 1 and difficulties in starting the one available EDG with a consequent loss of shutdown cooling. (Non-safety power remained available.) The significance of the event is that had the licensee been unable to restore power within 1.8 hours, the plant would have been in an inadequately analyzed condition and personnel would not have had adequate procedures and training to deal with the situation. (Had the incident

occurred 2 days after shutdown, the 1.8 hours would have been reduced to less than 15 minutes.) This status of analysis, procedures, and training is believed to be typical of industry preparedness. One alternate source of RCS makeup was available that did not require ac power and would increase the time available for the licensee to restore ac power. This involved gravity feed of borated water from the refueling water storage tank to the RCS.

Because of the number of past incidents that have occurred at various plants while in shutdown conditions and the potential for regulatory concerns, NRC management decided that the Vogtle event warranted the formal and detailed review of an Incident Investigation Team (IIT). An IIT consisting of NRC members (Headquarters, Regions I and V) and industry members (a member of the Institute of Nuclear Power Operations, and two consultants) was formed on March 23, 1990. The team's evaluations and findings are contained in NUREG-1410 ("Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operations at Vogtle Unit 1 on March 20, 1990"), which was issued during June 1990. Meanwhile, the licensee took corrective actions and completed refueling operations. On April 13, 1990, the NRC agreed with the licensee that Unit 1 could be restarted; the plant attained criticality on April 16, 1990, and was connected to the electrical grid on April 21, 1990. This incident is also discussed in more detail in Section 7, "Incident Investigation Program," of this report.

Appendix C

Reports Issued During 1990 (Reactors)

AEOD Reports Issued in 1990

Case Studies			
Date	Title	No.	Author
10/90	Operating Experience Feedback Report—Solenoid Operated Valve Problems at U.S. Light Water Reactors (Reprinted as NUREG-1275, Vol. 6)	C90-01	H. Ornstein
Special Study Reports			
Date	Title	No.	Author
07/90	Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Power Reactors	NUREG-1272, Vol. 4, No. 2	
07/90	Office for Analysis and Evaluation of Operational Data 1989 Annual Report, Nonreactors	NUREG-1272, Vol. 4, No. 2	
03/90	Review of Thermal Stratification Operating Experience	S902	T. Su
08/90	Recurrence of Important Safety Issues Reported in LERs	S90-01	S. Israel
Engineering Evaluations			
Date	Title	No.	Author
02/90	Failures of Electrical Supply and Power Generation Equipment Which Disrupted Plant Function at Nuclear Power Plants	E90-01	M. Wegner
02/90	Crosby Low Pressure Relief Valves	E90-02	S. Israel
05/90	Overpressurization of Auxiliary Feedwater Systems	E90-03	C. Hsu
04/90	Swelling and Cracking in Hafnium Control Rods	E90-04	M. Wegner
05/90	Operational Experience on Bus Transfer	E90-05	S. Mazumdar
07/90	Potential for Residual Heat Removal System Pump Damage	E90-06	C. Hsu
07/90	Effects of Internal Flooding of Nuclear Power Plants on Safety Equipment	E90-07	T. Su
09/90	Low Temperature Overpressure Protection: Testing PORVs With the Alternate Pneumatic Supply	E90-08	S. Israel S. Salah

Engineering Evaluations (cont.)			
Date	Title	No.	Author
10/90	Additional Factors Affecting the Lift Setpoint of Pressurizer Safety Valves	E90-09	L. Padovan
12/90	Evaluation of Boiling Water Reactor Mode Switch Events	E90-10	W. Jones
Technical Review Reports			
Date	Title	No.	Author
01/90	PNO's Issued In First Quarter of 1989	T90-01	R. Dennig T. Wolf
01/90	Insights Regarding Commonwealth Edison Plant Root-Cause Determinations Related to Maintenance Effectiveness (Proprietary)	T90-02	N. Thomasson
03/90	Improper installation of Heat Shrinkable Tubing	T90-03	S. Mazumdar
03/90	Reverse (Backward) Acting Valve Manual Handwheels	T90-04	T. Cintula
03/90	Association Between Nuclear Plant Utilization and Incentive Regulation by Station Public Utility Commissions	T90-05	S. Stern
05/90	Aquatic Life In Emergency Cooling Ponds	T90-06	L. Padovan
05/90	Reversed Sensing Lines Connections	T90-07	B. Kaufer
06/90	Turbine Bypass Malfunctions	T90-08	B. Kaufer
06/90	Inadvertent Partial Draining of Condensate Storage Tanks	T90-09	T. Cintula
07/90	Evaluation of Maintenance Trends at Five Selected Sites (Proprietary)	T90-10	P. O'Reilly
07/90	Evaluation of Safety Equipment Outages For Significance at Zion (Revised)	T925A	F. Manning
08/90	Effect of High Energy Line Breaks on Chilled Water Systems at Nuclear Power Plants	T90-11	L. Padovan
09/90	Loss of Offsite Power to Comply With NRC Regulations	T90-12	T. Cintula
10/90	Corrosion and Failure of Service Water Pump Impeller Snap Rings	T90-13	C. Hsu
10/90	Seal Problems in Boric Acid Transfer Pumps	T90-14	S. Israel

Technical Review Reports (cont.)			
Date	Title	No.	Author
10/90	Salem 1 and 2 Evaluation of Operating Experience (Proprietary)	T90-15	P. O'Reilly
11/90	Impact of Pipe Liner Failure of Pump Operation	T90-16	S. Israel
12/90	Inadvertent Containment Spray Actuations	T90-17	M. Harper

Appendix D

Reports Issued From 1980 Through 1989 (Reactors)

Table D-1
Reports Issued in CY 1989

Case Studies			
Date	Title	No.	Author
None			
Special Study Reports			
Date	Title	No.	Author
06/89	Office for Analysis and Evaluation of Operational Data 1988 Annual Report, Power Reactors	NUREG-1272, Vol. 3, No. 1	
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 Effectiveness Monitoring (Appendices A and B are Proprietary)	S804B	P. O'Reilly T. Wolf P. Cross-Prather
02/89	Maintenance Programs at Nuclear Power Plants (Table 2 is Proprietary)	S901 Revision 1	M. Chiramal S. Israel M. Wegner S. Stern
Engineering Evaluations			
Date	Title	No.	Author
02/89	Problems With Oils, Greases, Solvents and Other Chemical Materials	E901	S. Israel
03/89	Fire and Explosive Mixtures Resulted From Introduction of Hydrogen Into Plant Air Systems	E902	H. Ornstein
	Not issued	E903	
04/89	On Demand Malfunctions of HPCI and RCIC	E904	T. Cintuia

Table D-1 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
06/89	Electrical Bus Bar Failures	E905	M. Padovan
08/89	Failure of Steam Generator Isolation Check Valve	E906	T. Cintula
09/89	Diversion of Seal Cooler Flow for RHR Pumps	E907	S. Israel
10/89	Excessive Valve Body Erosion at Brunswick	E908	E. Brown
12/89	Operator Actions During Operational Events	E909	S. Israel
12/89	Potential for Gas Binding of High Head Safety Injection Pumps Resulting From Inservice Testing of VCT Outlet Isolation Valves	E910	M. Padovan
Technical Review Reports			
Date	Title	No.	Author
01/89	Millstone Unit 1-Safety/Relief Valve Discharge Line Vacuum Breakers Failed Open	T901	T. Su
02/89	Inadvertent Reactor Trips Due to RCS Flow Instrumentation Maintenance Activities	T902	M. Padovan
03/89	Generic Implication of Browns Ferry Fire on November 2, 1987	T903	T. Su
04/89	Design Deficiency of Safety Injection Block Switch	T904	S. Mazumdar
04/89	Failure of 4160V GE Magneblast Breaker to Trip Open	T905	S. Mazumdar
04/89	Broken Lifting Beam Bolts In HPCI Terry Turbine	T906	T. Cintula
04/89	Component Degradation Due to Indiscriminate Painting	T907	M. Padovan
	A nonreactor report; See NUREG-1272, Vol. 4, No. 2	T908	
05/89	Operating Events Involving Dampers	T909	S. Israel
06/89	Investigation of Cracked Control Rod Drive Seal Housings at Palisades	T910	W. Jones
06/89	Evaluation of Individually Reported Safety System LERs for Their Combined Significance	T911	F. Manning
06/89	Selected Maintenance Rework	T912	S. Israel

Table D-1 (cont.)

Technical Review Reports (cont.)			
Date	Title	No.	Author
07/89	Comparison of the Proposed Maintenance Effectiveness (ME) indicator With Catawba and Farley Nuclear Plants Regarding Inspections (Proprietary)	T913	N. Thomasson T. Wolf M. Harper
09/89	Overview of Design/Installation Fabrication Errors in 1988	T914	S. Israel
09/89	EDG Ground Fault Detection and Trip Circuit at Perry Unit 1	T915	S. Mazumdar
09/89	Debris in Containment Recirculation Sumps	T916	M. Padovan
	Not issued (refer to E908)	T917	
09/89	Check Valve Failure Rates From NPRDS Data	T918	E. Brown
09/89	Failure of Overcurrent Protective Device at Palisades Unit 1	T919	S. Mazumdar
	Not issued	T920	
10/89	Inadequate Capacity of 4160V Switchgear at FitzPatrick	T921	S. Mazumdar
11/89	Failure of HPCI Turbine Due to High Moisture in Lube Oil	T922	C. Hsu
11/89	Delaminating Foil Insulation in Primary Containment	T923	T. Cintula
	Not issued	T924	
12/89	Evaluation of Safety Equipment Outages for Significance at Zion	T925	F. Manning
12/89	Evaluation of Two Beaver Valley 2 Nuclear Plant Equipment Degradation Events for Their Combined Significance	T926	F. Manning
12/89	Follow-up on Steam Binding of AFW Pumps	T927	C. Hsu
12/89	Inadequate Overpressure Protection for Auxiliary Steam Headers at the Oconee Plants	T928	S. Salah

Table D-2
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 Trail 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

Table D-2 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
08/88	Reliability of Recirculation Pump Breaker During an ATWS	E804	T. Su
09/88	Potential LOCA Due to Energized Uncovered Pressurizer Heaters	E805	T. Cintula
10/88	Loss of Decay Heat Removal Due to Rapid Refueling Cavity Pumpdown	E806	M. Padovan
10/88	Pump Damage Due to Low Flow Cavitation	E807	C. Hsu
12/88	Operational Experience Review of Potential Large Openings In Containment	E808	T. Cintula
Technical Review Reports			
Date	Title	No.	Author
01/88	Perry Nuclear Power Plant Unit 1-Unexpected MSIVs Closure and Reopening	T801	T. Su
05/88	Summary of Early Operational Experience of Foreign Commercial Nuclear Reactors (Proprietary)	T803	P. O'Reilly
05/88	"Precursor" Operational Events That Occurred From November 1, 1987, Through March 1988	T804	F. Manning
05/88	Insights From Significant Events in 1987	T805	S. Israel
05/88	Recent Operational Experience Trends at Fermi 2	T806	T. Wolf
06/88	Recent Operational Experience Trends at Indian Point 2	T807	T. Wolf
06/88	A Technical Basis for Granting Test Frequency Relief	T808	G. Plumlee
06/88	Blocked Thimble Tubes/Stuck Incore Detector	T809	M. Wegner
07/88	An Analysis of NPRDS Data for Hatch Plant (Proprietary)	T810	T. Wolf P. Cross-Prather
11/88	Degradation of Ice Condenser Containment Functional Capability	T811	F. Manning

Table D-2 (cont.)

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

Table D-3
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. Cintuia
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

Table D-3 (cont.)

Program Support Reports (cont.)			
Date	Title	No.	Author
09/87	Trends and Patterns Program Report-Operational Experience Feedback on Main Feedwater Flow Control and Main Feedwater Flow Bypass Valves and Valve Operators	P701	G. Plumlee
Technical Review Reports			
Date	Title	No.	Author
01/87	Compression Fitting Failures	T701	H. Ornstein
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

Table D-3 (cont.)

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

Table D-4
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	
12/86	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	C604	M. Chiramal
12/86	Operational Experience Involving Losses of Electrical Inverters	C605	F. Ashe
01/86	Trends and Patterns Program Plan-FY86-FY88	P601	R. Dennig
08/86	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1985	P602	L. Bell
08/86	Trends and Patterns Report of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants	P603	M. Harper
08/86	Trends and Patterns Report of the Operational Experience of Newly Licensed U.S. Nuclear Power Reactors	P604	T. Wolf
04/86	AEOD Annual Report for 1986	S601	J. Helternes
05/86	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	S602	J. Crooks
06/86	Adequacy of the Scope of IE Bulletin 86-01	S603	E. Leeds
Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
05/86	Core Damage Precursor Event at Trojan	E514 Revision 1	D. Zukor
01/86	Deficient Operator Actions Following Dual Function Valve Failures	E601	E. Leeds
01/86	Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures	E602	E. Leeds

Table D-4 (cont.)

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

Table D-4 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
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, 1986	NUREG-1190	
02/86	Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985	NUREG-1195	
08/86	Incident Investigation Manual*		
12/86	Incident Investigation Manual, Revision 1*		

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

Table D-5
Reports Issued in CY 1985

Case and Special Studies			
Date	Title	No.	Author
09/85	Licensee Event Report System, Evaluation of First Year Results and Recommendations for Improvements	NUREG-1022, Supp. No. 2	
06/85	Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	C501	H. Ornstein
09/85	Overpressurization of Emergency Core Cooling in Boiling Water Reactors	C502	P. Lam
12/85	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	C503	H. Ornstein
12/85	Loss of Safety System Function Events	C504	E. Trager
07/85	Feedwater Transient Incidents in Westinghouse PWRs	P501	R. Dennig
06/85	Trends and Patterns Analysis of 1981 Through 1983 LER Data (NUREG/CR-4129)	P502	B. Brady
08/85	Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Reactors-January 1 Through June 30, 1984	P503	T. Wolf
08/85	Trends and Patterns Report of Unplanned Reactor Trips at U.S. Light Water Reactors in 1984	P504	L. Bell
03/85	Review of Operational Experience From Non-Power Reactors	S501	D. Zukor
04/85	AEOD Semiannual Report for July-December 1984	S502	J. Heltemes
09/85	Evaluation of Recent Valve Operator Motor Burnout Events	S503	E. Brown
Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
01/85	Motor-Operated Valve Failures Due to Hammering Problem	E501	M. Chiramal

Table D-5 (cont.)

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

Table D-5 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
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-6
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

Table D-6 (cont.)

Case and Special Studies (cont.)			
Date	Title	No.	Author
09/84	Semiannual Report on AEOD Activities	S405	J. Heltemes
10/84	Application of Risk Perspectives: A Procedures Guide	S406	P. Lam
Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
01/84	Temporary Loss of All AC Power Due to Relay Failure in Diesel Generator Load Shedding Circuitry at Fort St. Vrain	E401	M. Chiramal
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

Table D-6 (cont.)

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

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
03/84	Failures of Containment Air Monitors at Farley Units 1 and 2	T401	D. Zukor
03/84	Chemical Contamination of Primary and Secondary Systems in Light Water Reactors	T402	M. El-Zeftawy
03/84	Setpoint Drift of Barton Model 288 Switches	T403	M. Chiramal
04/84	Cable Fire and Loss of Control Power to Engineered Safeguards Valves	T404	M. Chiramal
04/84	Cold Weather Events 1983-1984	T405	T. Cintula
04/84	Improper Spare Parts Procurement Event at Grand Gulf Unit	T406	T. Wolf
04/84	Failure of a 4 kV Circuit Breaker to Trip	T407	M. Chiramal
05/84	Diesel Generator Inoperability Due to Overheating of Ventilation Cowling	T408	M. Chiramal
05/84	Multiple Failure of Bell and Howell Dual Potentiometer Modules That Occurred at the Fort Calhoun Nuclear Station	T409	F. Ashe
05/84	Failure of Injection Valve for the High Pressure Coolant Injection System to Open During a Surveillance Test	T410	E. Brown
06/84	Contamination of the Nitrogen System at Sacramento Municipal Utility District	T411	M. El-Zeftawy
06/84	Failure of an Access Door Between the Drywell and the Wetwell	T412	T. Wolf
06/84	Failure of Fire Damper in Safeguards Ventilation System	T413	W. Lanning
07/84	Station Operating Restrictions for Loss or Out-Of-Service Power Transformers Through Which Electrical Power is Supplied to the Emergency Buses	T414	F. Ashe
07/84	Destruction of Charging Pump	T415	W. Lanning
08/84	Loss of Engineered Safety Feature Auxiliary Feedwater Pump Capability at Trojan on January 22, 1983	T416	D. Zukor
08/84	Excessive Cooledown Rate Event at LaSalle Unit 1	T417	S. Salah

Table D-6 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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-7
Reports Issued In CY 1983

Case and Special Studies			
Date	Title	No.	Author
09/83	Licensee Event Report System, Description of System and Guidelines for Reporting	NUREG-1022	
09/83	Potentially Damaging Failure Modes of High and Medium Voltage Electrical Equipment	NUREG/CR-3122	
M. Chiramal			
04/83	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	C301	M. Chiramal
07/83	Report on the Implications of the Anticipated Transient Without Scram Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience	P301	J. Crooks
Engineering Evaluations and Technical Reviews			
Date	Title	No.	Author
01/83	Fuel Degradation at Westinghouse Plants	E301	D. Zukor
04/83	Update to AEOD/E301 (Fuel Degradation at Plants)	E301 Revision 1	D. Zukor
01/83	Potential Loss of Service Water Flow Resulting From a Loss of Instrument Air	E302	E. Imbro
02/83	Valve Flooding Event at Surry	E303	D. Zukor
03/83	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	E304	T. Cintula
04/83	Inoperable Motor-Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	E305	E. Brown F. Ashe
04/83	Cooldown During Loss of Control Room Test at McGuire Unit 1	E306	D. Zukor
04/83	Degradation of Safety-Related Batteries Due to Cracking of Battery Cell Cases and/or Other Possible Aging-Related Mechanisms	E307	F. Ashe

Table D-7 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
04/83	Cracks and Leaks in Small-Diameter Piping	E308	E. Brown
04/83	The Potential for Water Hammer During the Restart Residual Heat Removal Pumps at BWR Nuclear Power Plants	E309	S. Rubin
04/83	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E310	T. Cintula
04/83	Loss of Salt Water Flow to the Service Water Heat Exchangers for 23 Minutes at Calvert Cliffs Unit 2	E311	T. Cintula
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

Table D-7 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
09/83	Review of Events Involving Failures of Power Supply in Instrumentation and Control Systems	E324	M. Chiramal
11/83	Vapor Binding of Auxiliary Feedwater Pumps at Robinson Unit 2	E325	W. Lanning
11/83	Steam Voiding in Oconee Unit 3 on June 13, 1975: A Precursor Event to the TMI-2 Accident	E326	H. Ornstein
11/83	Gaseous Releases From Waste Gas Disposal System	E327	N. Trehan
11/83	Human Factors Involvement in Events at Oconee Units 1, 2, and 3	NT304	K. Black
08/83	Human Factors Contributions to Accident Sequence Precursor Events	N305	E. Trager
01/83	Diesel Generator Load Sequencer Design Deficiency- LER 82-025/OIT	T301	M. Chiramal
02/83	Postulated Loss of Auxiliary Feedwater System Resulting From a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	T302	E. Imbro
03/83	Seat Degradation in Henry Pratt Butterfly Valves	T303	E. Brown
03/83	Cause of Containment Isolation Valve F042A to Close	T304	S. Salah
03/83	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Intrusion at Sequoyah Unit 1	T305	E. Imbro
04/83	Scram Discharge Volume Level Switch Failure at Hatch Unit 2	T306	J. Pellet
04/83	Condensate Demineralizer Resin Migration Through the Plant Vent and the Standby Gas Treatment System at Pilgrim Unit 1	E307	J. Pellet
04/83	Undetectable Failure in Westinghouse Solid State Protection System	T308	M. Chiramal
04/83	Air in Reactor Water Cleanup System Instrument Sensing Lines at Brunswick Unit 2	T309	S. Salah
04/83	Blocking of Automatic Safety Injection Signals	T310	M. Chiramal

Table D-7 (cont.)

Engineering Evaluations and Technical Reviews (cont.)			
Date	Title	No.	Author
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
06/83	Design Deficiency in Control Circuits of Feedwater Isolation Valves and Boron Injection Tank Recirculation Valves	T319	M. Chiramal
06/83	inadvertent Safety Injections Attributed to Personnel Error at Summer	T320	F. Ashe
06/83	Check Valve Installed Backwards in Instrument Air Line to the Power-Operated Relief Valve at Surry Unit 2	T321	D. Zukor
06/83	Gouges in Main Coolant System Piping at Diablo Canyon on April 19, 1983	T322	D. Zukor
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

Table D-7 (cont.)

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

Table D-8
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

Table D-8 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
02/82	LER 50-336/81-26: Investigation of the Relative Frequency of Valve Overtravel Anomalies That Could Result in a Potential Centrifugal Pump Runout Exceeding Net Positive Suction Head	E207	E. Imbro
02/82	An Observed Difference in Lift Setpoint for Steam Generator and Pressurizer Safety Valves	E208	T. Cintula
02/82	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback Unit 1 on April 13, 1979)	E209	M. Chiramal
02/82	Inadequate Switchgear Cooling at Beaver Valley Unit 1	E210	W. Lanning
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

Table D-8 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
04/82	Indian Point Unit 2 Flooding Event	E221	W. Lanning
05/82	Loss of Reserve Station Service Transformer "B" on January 18, 1982, at Surry Unit 2	E222	N. Trehan
05/82	Inadvertent Loss-Of-Coolant Events at Sequoyah Units 1 and 2	E223	W. Lanning
05/82	Generic Concerns Associated With the Ginna Steam Generator Tube Rupture Event	E224	W. Lanning
06/82	Degradation of BWR Scram Pilot Solenoid Valves Due to Abnormal Power Supply Voltage	E225	M. Chiramal
06/82	Inoperability of Instrumentation Due to Extreme Cold Weather	E226	M. Chiramal
06/82	Failure of Engineered Safety Features Manual Initiation Pushbutton Switches	E227	F. Ashe
06/82	Repetitive Overspeed Trips of the Steam Driven Emergency Feedwater Pump on Initial Start at Arkansas Nuclear One, Unit 2	E228	E. Imbro
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

Table D-8 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
08/82	Brunswick Steam Electric Plant Unit 2 Loss of Residual Heat Removal Service Water on January 16, 1982	E236	T. Wolf
08/82	Power-Operated Relief Valve Failure at Robinson	E237	E. Brown
08/82	Water In the Lube Oil In Safety Injection Pump IA-A at Sequoyah-LER 81-076	E238	N. Trehan
09/82	Main Steam Isolation Valve Closures and Pressurizer Safety Valve Actuations at St. Lucie Unit 1 on December 19, 1981	E239	T. Cintula
09/82	Preliminary Account of Events Associated With a Reactor Trip at Hatch Unit 2 on August 25, 1982	E240	S. Rubin
10/82	Emergency Diesel Generator System Problems at FitzPatrick	E241	M. Chiramal
10/82	Fuel Assembly Degradation While In the Spent Fuel Storage Pool	E242	E. Brown
10/82	Plant Trip Followed by a Safety Injection Due to Loss of "A" Cooling Tower Pump at Palisades on February 4, 1982	E243	T. Cintula
10/82	Loss of Residual Heat Removal System Event at Pilgrim Nuclear Power Station on December 21, 1981	E244	T. Wolf
10/82	Failure of Westinghouse Type SC-1 No. 1876-072 Relays	E245	F. Ashe
10/82	Events Involving Loss of Electrical Inverters Including Attendant Inverters to Vital Instrument Buses	E246	F. Ashe
10/82	Engineering Evaluation of Turbine/Reactor Trip Rancho Seco on August 7, 1981	E247	J. Pellet
11/82	Engineering Evaluation Report on McGuire Overpressurization Event of August 27, 1981	E248	D. Zukor
11/82	Engineering Evaluation Memorandum-Licensee Reporting of the Turbine/Reactor Trip at Rancho Seco on August 7, 1981	E249	H. Ornstein
11/82	Quad Cities Unit 2 Loss of Auxiliary Electrical Power Event on June 22, 1982	E250	M. Chiramal

Table D-8 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
11/82	Salem Unit 2 Loss of Vital Bus No. 2A	E251	M. Chiramal
11/82	Potential Control Logic Problem Resulting in Inoperable Auto-Start of Diesel Generator Units Under the Conditions of Loss-of-Coolant Accident and Loss of Station Power (LOSP)	E253	F. Ashe
11/82	Review of Prairie Island Unit 1 LER 82-015-O1T on Diesel Generator Operability	E254	M. Chiramal
11/82	Failure of the Vent Line on the Common Discharge of the Two Motor-Driven Auxiliary Feedwater Pumps at San Onofre Unit 2 From an Improper Valve Lineup	E255	T. Cintula
11/82	Loss of Shutdown Cooling and Subsequent Boron Dilution at San Onofre Unit 2	E256	T. Cintula
12/82	Insufficient Net Positive Suction Head for Charging Pump Service Water Pumps at Surry Nuclear Power Station	E257	D. Zukor

Table D-9
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

Table D-9 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
05/81	Evaluation of High Pressure Safety Injection Pump Operability Without Service Water	E111	E. Imbro
06/81	Inoperability of Instrumentation Due to Extreme Cold Weather	E112	M. Chiramal
06/81	Deliberate Pump Trip at Browns Ferry Unit 2 on April 6, 1981	E113	W. Lanning
06/81	Control System Failures That Could Cause or Exacerbate Nuclear Power Plant Accidents	E114	F. Ashe
07/81	Additional Information on Events at TMI-2 During Preoperational Testing (September 5-12, 1977)	E115	H. Ornstein
07/81	Failure of B Phase Main Transformer and Subsequent Fire in the Transformer Area-North Anna Unit 2	E116	M. Chiramal
07/81	Events at TMI-2 During Preoperation Testing	E117	H. Ornstein
07/81	Setpoint Drift Occurrences for the Barton Model 288 Instrument	E118	F. Ashe
07/81	Loss of Residual Heat Removal Capability at Brunswick Units 1 and 2	E119	E. Imbro
08/81	Ignition of Gaseous Waste Decay Tank at San Onofre Unit 1-July 17, 1981	E120	H. Ornstein
08/81	Crystal River 3 Engineered Safeguards Relay Failures	E121	M. Chiramal
09/81	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of Integrated Control System Nonnuclear/Instrumentation	E122	H. Ornstein
09/81	Immediate Action Memo: Common Cause Failure Potential at Rancho Seco-Desiccant Contamination of Air Lines	E123	H. Ornstein
09/81	Review of Information on Purge Valves	E124	E. Brown
10/81	Engineering Evaluation Report on Shutdown Cooling System Heat Exchanger Failures at Oyster Creek, August 1981	E125	G. Lanik
10/81	Event Sequences Not Considered in the Design of Emergency Bus Control Logic	E126	F. Ashe

Table D-9 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
10/81	Pressure Boundary Degradation Due To Pump Seal Failure at Arkansas Nuclear One	E127	W. Lanning
11/81	Inoperable Teledyne Solenoid Valves	E128	F. Ashe
12/81	Brunswick Unit 2 Diesel Generator Jacket Water Temperature Control Valve and Manual Bypass Valve	E129	M. Chiramal
12/81	Davis Besse LER 79-062 on Auxiliary Feedwater System Pressure Switches	E130	M. Chiramal
12/81	High Circulating Current Associated With Inverter Output Due to Lack of Circuit Tuning	E131	F. Ashe
12/81	Abnormal Wear Encountered on Aloyco Swing Check Valves Installed in the Low Pressure Safety Injection System at Palisades	E132	T. Cintula
04/81	Inadequacies In Periodic Testing of Combustion Engineering PWR Reactor Protection System	E133	M. Chiramal

Table D-10
Reports Issued in CY 1980

Case Studies			
Date	Title	No.	Author
07/80	Report on the Browns Ferry Unit 3 Partial Failure to Scram Event on June 28, 1980	C001	S. Rubin
09/80	Report on the Interim Equipment and Procedures at Browns Ferry Unit 3 to Detect Water in the Scram Discharge Volume	C002	G. Lanik
10/80	Report on Loss-of-Offsite-Power Event at Arkansas Nuclear One, Units 1 and 2	C003	W. Lanning
11/80	AEOD Actions Concerning the Crystal River Unit 3 Loss of Nonnuclear Instrumentation and Integrated Control System Power on February 26, 1980	C004	H. Ornstein
12/80	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	C005	E. Imbro
Engineering Evaluations			
Date	Title	No.	Author
03/80	Crystal River Nuclear Power Plant Decay Heat Closed Cycle Cooling Water Pumps/DCP-1A and DCP-1B	E001	H. Ornstein
05/80	BWR Jet Pump Integrity	E002	S. Rubin
06/80	Comparison of Reactor Coolant Pump Events Contained in LERs, NPRDS, RECON, and Plant Records	E003	E. Brown
07/80	Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants, January 1, 1972 to April 30, 1978	E004	H. Ornstein
07/80	Operational Restrictions for Class 1E 120V AC Vital Instrument Buses	E005	M. Chiramal
08/80	Loss of Residual Heat Removal at Beaver Valley, LER 80-031	E006	W. Lanning

Table D-10 (cont.)

Engineering Evaluations (cont.)			
Date	Title	No.	Author
08/80	Potential for Unacceptable Interaction Between the Control Rod Drive System and Nonessential Control Air System at Browns Ferry	E007	S. Rubin
08/80	Operational Restrictions During Surveillance Testing of Emergency Diesel Generators	E008	M. Chiramal
08/80	Failures of Containment Isolation Valves at Zion	E009	W. Lanning
08/80	Tie Breaker Between Redundant Class 1E Buses-Point Beach Units 1 and 2	E010	M. Chiramal
08/980	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

This section summarizes the year-end status of all AEOD recommendations that are either new or still outstanding since the last report. At the beginning of 1990, 11 AEOD recommendations were outstanding. During 1990, two recommendations were resolved, and six new recommendations were added by Case Study C90-01. Therefore, as of December 31, 1990, 15 AEOD recommendations were outstanding. Table E-1 presents the status of AEOD recommendations at the end of 1990.

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

Information about each recommendation that is outstanding or that was resolved during 1990, including a description and status for each, follows.

Table E-1 Status of AEOD Recommendations, 1990

Status of AEOD Recommendations	12/31/90
• Added since previous report	6
• Resolved	2
• Currently outstanding	15
- Included In unresolved safety Issues, generic issues, and Three Mile Island Action Plan items	7
- Addressed by NRC and Industry	2
- Too recent to be assigned	6
• Office of Nuclear Regulatory Research prioritization of Issues	
- High priority	6
- Medium priority	1
- Low priority	0
- No priority	8
• Functional areas addressed by issues	
- Procedures, training, etc.	4
- Hardware modification	6
- Equipment testing	3
- Other	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:

Active. 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. During 1990, the RES staff coordinated 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) reviewed this package and recommended modifying of the package to encourage industry comments to facilitate a decision on final resolution. This package will be issued for public comment, along with specific questions for possible alternative solutions, in mid-1991. The final resolution of GI 23 is scheduled for late 1992.

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.

AEOD Recommendation Tracking System

	Responsible Office/Div/BR	Contact	Priority
	NRR/DEST/SRXB	R. Jones	Low
Status:	<p>Resolved. B&W licensees, the Electric Power Research Institute, and the NRC have jointly funded 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 test results thus far, the concerns raised in the report have been addressed. Since the results appear acceptable, there appears to be no need for additional work.</p>		

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" 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/DRP/PD4-2	D. Pickett	High
Status:	<p>Resolved. This recommendation was included in GI 94. After the Committee To Review Generic Requirements (CRGR) and the Advisory Committee on Reactor Safeguards (ACRS) review and resolution of their final response, the GI was transferred to NRR in November 1989 to be resolved and closed. This recommendation was technically resolved by issuance of Generic Letter 90-06 on June 25, 1990 which required improvements to PORVs and block valves.</p>		

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.

AEOD Recommendation Tracking System

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

Status: Active. GI 106, "Piping and the Use of Highly Combustible Gases in Vital Areas," was originally given a low priority. The priority of this issue was re-evaluated, 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) and hydrogen storage facilities. RES plans to resolve GI 106 in 1991.

Recommendation

Source: Case Study AEOD/C502

**Responsible
AEOD Engineer:** S. Salah

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	G. Burdick	High

Status: Active. This recommendation was incorporated in GI 105, "Interfacing Systems LOCA [Loss-of-Coolant-Accident] at LWRs." 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 has completed an NRC inter-system LOCA action plan for identifying individual plant susceptibility to interfacing systems LOCA. This plan includes a Davis-Besse, Catawba, and Waterford-3 analysis to identify PWR resolution alternatives and Impel Corporation component fragility information for BWRs. RES plans to complete GI 105 in late 1991.

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	G. Burdick	High

AEOD Recommendation Tracking System

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	G. Burdick	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	G. Burdick	High

Status: (See status of Recommendation 1.)

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	W. LeFave	High

Status: Active. 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 was not addressed. However, AEOD is working with industry (American Society of Mechanical Engineers and the American Nuclear Society) to formulate a performance standard to ensure that licensees adequately maintain their plant instrument air systems.

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

AEOD Recommendation Tracking System

Responsible AEOD Engineer: S. Mazumdar

Title or Subject: "Operational Restrictions for Class 1E 120 V ac Vital Instrument Buses"
"Tie Breaker Between Redundant Class 1E Buses—Point Beach Nuclear Plant Units 1 and 2"

Recommendation 1: Interconnection between redundant safety-related electrical load groups should comply with requirements of Regulatory Guide 1.6.

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

Status: Active. This recommendation was being pursued under GI 128. GI 128 was reviewed and approved by the CRGR in July 1989, but the ACRS did not agree to the proposed resolution. This recommendation and a number of GSI's directly related to onsite electrical power systems (NUREG-0933) are near resolution. These issues include A-30, "Adequacy of Safety-Related DC Power Supplies," and 48, "LCOs for Class 1E Vital Instrument Bus in Operating Reactors," and 49, "Interlocks and LCOs for Redundant Class 1E Tie Breaker." These issues are scheduled to be resolved by issuing of the generic letters in April 1991.

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: Active. The Bolling Water Reactor Owners Group (BWROG) is continuing work on their "Long-Term Solution" options to provide protection against violation of the fuel safety limit, particularly from asymmetric oscillations. Recently, the LPRM-based microprocessor system to detect and suppress oscillations has been emphasized. The BWROG expects to complete and submit for NRC review a licensing topical report describing the relevant methodologies of proposed solutions by March 1991, which should permit staff review by June 1991. Further effort will be needed for plant-specific implementations.

AEOD Recommendation Tracking System

Both Brookhaven National Laboratory (BNL) and the BWROG are continuing to evaluate ATWS scenarios. The BWROG has proposed revised emergency operating procedures (EOPs) to provide quicker power reduction. The BWROG and BNL analyses that use these procedures appear to demonstrate that they would be effective in terminating the oscillations and in lowering the power level before reaching suppression pool and containment limits. However, recent BWROG calculations have produced sufficiently large oscillations (1250% neutron flux spikes) to raise concerns about the possibility of high fuel-oscillation peaks and failure to rewet after the peaks. The BWROG is concerned that margins to this problem may be insufficient to bound uncertainties in initial reactor operating conditions.

A task force and peer review group has been established to investigate and recommend appropriate action. This will delay the BWROG submittal of the ATWS report and may affect the staff goal to close this issue before June 1991.

Recommendation

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

**Responsible
AEOD Engineer:** H. Ornstein

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

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

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

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

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

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

Responsible Office/Div/BR	Contact	Priority
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- Not yet designated -

Status: Active. Recommendations 1 through 5.
The case study was issued in December 1990, and NRR is presently considering actions to be taken. NRR has not yet designated a responsible engineer to follow the recommendations in this study.

AEOD Recommendation Tracking System

Recommendation 6: An industry group such as INPO should initiate a SOV failure-feedback program.

**Responsible
Office/Div/Br**

Contact

Priority

- Not yet designated -

Status:

Active. The report was recently released to industry and no definitive action plan has yet been proposed.

Appendix F

Status of NRC Staff Actions For Events Investigated by Incident Investigation Teams

Appendix F

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

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 or warrant additional attention or action. Office Directors designated by the EDO as having responsibility for resolving issues or concerns are responsible for providing written status reports on the disposition of assigned actions. In addition, followup actions associated with the IIT report do not necessarily include all licensee actions, nor do they cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or

possible enforcement actions. These items are expected to be defined and implemented through the normal organizational structure and procedures.

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

For the Davis-Besse, San Onofre, and Rancho Seco IITs, the appendix addresses the status of staff actions that had not yet been reported as resolved in the AEOD 1989 Annual Report, and for the Vogtle IIT, the appendix addresses the status of all associated staff actions.

AEOD IIT Action Tracking System

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

Item 9: Reliability of the pilot-operated relief valve (PORV)

- Actions:**
- (a) Monitor the licensee's troubleshooting activities. (Responsible Office: Region III)
 - (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: Resolved

The 1989 AEOD Annual Report, Reference 2, documented the resolution of Actions (a), (b), and (e). The status of Actions (c) and (d) was documented as ongoing, pending the issuance of an NRC generic letter that will address PORV and block-valve reliability. The generic letter, Reference 3, was issued on June 25, 1990.

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

The 1989 AEOD Annual Report, Reference 2, documents the status of these actions as pending the installation of a plant-specific simulator. Delivery of the simulator is scheduled for January 10, 1991. The simulator is expected to be operational by July 1991.

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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-1272, Volume 4, No. 1, "Office for Analysis and Evaluation of Operational Data 1989 Annual Report," dated July 1990.
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Action Source: IIT Report on San Onofre Nuclear Generating Station (SONGS) Unit 1 Event of November 21, 1985 (Reference 1)

Item 7: Adequacy of emergency notifications and NRC response.

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 overall plant status. (Responsible Office: AEOD)

Disposition: Ongoing

The 1989 AEOD Annual Report, Reference 2, documented the resolution of Action (a). The NRC policy regarding use of the ENS line and NRC personnel as ENS communicators was evaluated and actions taken as documented in Reference 2. The status of Action (b) was ongoing pending the implementation of site-specific information systems for use by NRC Headquarters Operations Officers and implementation of the Emergency Response Data System (ERDS). The site-specific information system is expected to become operational during 1991 and the ERDS in 1993.

References:

- 1. NUREG-1190, "Loss of Power and Water Hammer Event at San Onofre, Unit 1, on November 21, 1985," dated January 1986.
- 2. NUREG-1272, Volume 4, No. 1, "Office for Analysis and Evaluation of Operational Data 1989 Annual Report," dated July 1990.

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Action Source: IIT Report on Rancho Seco Incident of December 26, 1985 (Reference 1)

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 licensee 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 Instrument and Control System Bus During Operation," issued November 30, 1979); NUREG-0667 ("Transient Response to Babcock and 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: Resolved

The 1989 AEOD Annual Report, Reference 2, documented the resolution of Actions (b) and (c). The status of Action (a) was ongoing pending the completion of the staff's audit of the implementation of the Babcock and Wilcox Owners Group (BWO) recommendations resulting from their plant reassessment program. The staff completed Safety and Performance Improvement Program (SPIP) implementation audits to verify the implementation of the BWO recommendations as documented in References 3 through 7.

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- 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-1272, Volume 4, No. 1, "Office for Analysis and Evaluation of Operational Data 1989 Annual Report," dated July 1990.
 3. Safety and Performance Improvement Program (SPIP) implementation Audit at Arkansas Nuclear One, Unit 1, dated December 19, 1990.
 4. SER on Crystal River Unit 3 SPIP Programmatic Audit, dated September 17, 1990.
 5. SER on Davis Besse SPIP Implementation Audit, dated November 6, 1990.

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6. SPIP Implementation Audit at the Oconee Nuclear Station, dated November 15, 1990.
 7. Results of Three Mile Island Unit 1 Safety and Performance Improvement Program (SPIP) Implementation Audit, dated May 25, 1990.
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Action Source: IIT Report on Vogtle Unit 1 Report of March 20, 1990
(Reference 1)

Item 1: Adequacy of shutdown risk management

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

Disposition: Ongoing

The staff is conducting the following assessments of shutdown risk: (1) review of shutdown probabilistic risk assessments (PRAs), (2) performance of new shutdown PRAs, (3) evaluation of potential accident precursors, and (4) field studies of shutdown operations and activities. Completion is expected by late 1991.

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

Disposition: Ongoing

The Office of Nuclear Regulatory Research (RES) is developing risk methods for non-full-power operating modes, including the shutdown mode. Demonstration of the applicability of these methods will be performed for two plants: Surry and Grand Gulf. The set of initiating events to be considered include those internal events normally considered in risk analyses (e.g., pipe ruptures) as well as events special to low-power and shutdown conditions. Fires and floods initiated within the plant are included in the set. Screening analyses will be performed. This work is being performed in coordination with NRR's plan for shutdown risk management analysis.

NUREG-1410 proposes two alternative methods for continued core cooling, after a loss of RHR during mid-loop operations. The first method is to use gravity feed of water from the refueling water storage tank into the reactor vessel. The staff has determined that research is not required to confirm this method, as long as a continuous flow can be ensured. The other method, reflux cooling in the presence of non-condensable gasses, is a cooling mode that has not been studied extensively and is not completely understood. RES planning efforts will be concentrated on this method.

The staff is currently collecting and evaluating experimental and analytical data applicable to the thermal-hydraulic processes involved in loss of RHR during mid-loop operation. These data are being studied to evaluate (1) the potential plant

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conditions that may result and the cooling methods that may exist; (2) the controlling thermal-hydraulic phenomena associated with these conditions as well as a determination of relevant existing data; and (3) the conditions and phenomena requiring additional research to provide a valid basis for regulatory actions. The schedule for completing these actions is as follows: (1) complete screening analyses and develop perspectives by May 1991, (2) complete detailed risk analysis of the most important event by March 1991, (3) complete analysis of other important events by December 1992, and (4) complete analysis on the effectiveness of alternative cooling methods by June 1991.

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

Disposition: Ongoing

The staff will analyze the results of work conducted under Action (a) above and will determine the need for and develop, as appropriate, proposed new requirements for shutdown risk management, proposed new or revised Standard Technical Specifications, and/or guidelines for emergency operating procedures. The expected completion date is September 1991.

Item 2: Adequacy of control over switchyard activities

Action: (a) Evaluate the adequacy of existing regulatory guidance and requirements for the control of activities and hazardous materials in switchyards and protected areas. Issue new guidance as necessary. (Responsible Office: NRR)

Disposition: Ongoing

The staff determined that, with the exception of safeguards control, little if any regulatory guidance and requirements exist for the control of activities and hazardous materials in switchyards and protected areas. Supplement 1 to Information Notice (IN) 90-25, "Loss of Vital AC Power with Subsequent Reactor Coolant System Heat-Up," will be issued by March 15, 1991, to notify licensees of concerns about switchyard controls that resulted from the Vogtle and two previous events to enable appropriate corrective actions to be considered.

Action: (b) Evaluate the corrective actions taken at Vogtle to ensure adequate control of activities and hazardous materials in the switchyard. (Responsible Office: Region II)

Disposition: Resolved

Reference 5 documents the licensee's immediate corrective actions for access control to the switchyard. Reference 6 documents the region's review of the licensee's revisions to procedures for controls for vehicular movement, spotting of

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portable equipment, and topping off of portable equipment. The application of the guidance in these procedures was inspected and documented in Reference 7. References 6 and 8 document the region's inspection of the licensee's General Employee Training Program with regard to vehicular movement controls. All of the corrective actions were deemed acceptable.

Item 3: Adequacy of diesel generator instrument and control systems

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

Disposition: Ongoing

By the end of January 1991, the Electrical Systems Branch, NRR, will complete its recommendations regarding this action. Following this effort, NRR management will review the recommendations to determine what level of generic action, if any, is appropriate.

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

Disposition: Ongoing

Same as Action 3(a).

Action: (c) Evaluate Vogtle's procedures and training for emergency diesel generators and local sequencer panel operations including response to malfunctions. (Responsible Office: Region II)

Disposition: Resolved

Reference 5 documents the region's review of the licensee's actions to ensure that sequencer start and restart features and guidance about logging malfunctions were disseminated to all operators. The region's review of the licensee's surveillance instructions to ensure that design changes and calibration "lessons learned" have been incorporated is documented in Reference 9. The region's verification of operator proficiency in emergency diesel generator and sequencer operations via job performance measures during requalification examinations is documented in Reference 10. Reference 8 documents the region's performance-based training inspection, using the March 20, 1990, event as the entry point. The results of the above were deemed satisfactory.

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Item 4: Adequacy of emergency preparedness

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

Disposition: Ongoing

The staff has integrated this action into its evaluation of safety risks during shutdown and low-power operation. The staff will analyze the results of the evaluations performed for Item 1 and determine the need for and develop, as appropriate, proposed new or revised guidance pertaining to emergency preparedness (NUREG-0654).

Action: (b.1) Evaluate the policy and guidance to NRC managers for determining when the NRC will enter the standby mode. Initiate followup actions, as appropriate. (Responsible Office: AEOD)

Disposition: Resolved

The policy was reviewed, and after considering the Vogtle event and a number of past events, AEOD concluded that sufficient guidance exists for appropriately making a determination for the agency to go to the standby mode. The criteria and purpose for entering the standby mode were reemphasized in a memorandum from the Director, AEOD, to NRC's senior executives involved in decisions regarding the agencies response mode (Reference 4).

Action: (b.2) Identify and correct Vogtle's recent ENS problem. (Responsible Office: AEOD)

Disposition: Resolved

Repairs to the ENS system were completed and a complete system test was conducted with satisfactory results.

Item 5: Adequacy of operating experience feedback

Action: (a) Evaluate the procedures and practices for events analysis and feedback of lessons to industry and recommend appropriate improvements. Consideration should be given to: (a) past NRC reports not the subject of "generic communications" and (b) the development of the proper scope of generic action based on the individual events. Also, address the need for feedback regarding the Calcon sensor failures and calibration issues arising from the Vogtle event. (Responsible Office: NRR)

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Disposition: Ongoing

The staff is developing recommendations for improvement, which will be incorporated into a revision to NRC Inspection Manual Chapter 0720, "NRC Bulletins and Information Notices." This revision is expected to be completed by June 1991.

The staff sent the information on the Calcon sensor failures to the TDI owners group.

Action: (b) Review the industry's NPRDS program regarding: (1) the need for improvements in reporting guidance or practices for diesel generator components, and (2) industry practice in providing NPRDS reported component failures to vendors or the level of vendor use of NPRDS. Interface with INPO as appropriate for action on this and the staff's overall evaluation of NPRDS to stimulate improvements. (Responsible Office: AEOD)

Disposition: Resolved

At an August 29, 1990, meeting, INPO informed AEOD that the diesel generator reporting issue would be pursued, using an existing process for dealing with NPRDS reporting inconsistencies. INPO would involve the NPRDS Users Group of which NRC is a member. INPO's actions to date have had a discernible effect on improving the reporting of diesel-generator component failures. INPO has several programs for interacting with vendors that include providing the vendors with NPRDS data. INPO also has several programs under way to improve NPRDS quality, completeness, and timeliness. Results of the above activities will be monitored through the staff's routine NPRDS activities (e.g., annual evaluations, special component studies, failure reporting trends). AEOD concluded that these actions were an acceptable approach to address this action item.

- References:**
1. NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Mid-Loop Operation at Vogtle Unit 1 on March 20, 1990," dated June 1990.
 2. NRC Information Notice No. 90-25, "Loss of Vital AC Power with Subsequent Reactor Coolant System Heat-Up," dated April 16, 1990.
 3. Memorandum from J. Taylor to NRC Staff, "Staff Actions Resulting From the Investigation of the March 20, 1990, incident at Vogtle Unit 1 (NUREG-1410)," dated June 21, 1990.
 4. Memorandum from E. Jordan to NRC Staff, "Evaluation of Policy for Determining When the NRC Will Enter the Standby Mode," dated November 29, 1990.
 5. Letter from W. G. Hariston, III, to S. D. Ebnetter, "Vogtle Electric Generating Plant Confirmation of Action Letter," dated April 9, 1990.
 6. NRC Inspection Report Nos. 50-424/90-25 and 50-425/90-25, dated November 21, 1990.

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7. NRC Inspection Report Nos. 50-424/90-28 and 50-425/90-28, dated December 13, 1990.
 8. NRC Inspection Report Nos. 50-424/90-29 and 50-425/90-29, dated December 24, 1990.
 9. NRC Inspection Report Nos. 50-424/90-20 and 50-425/90-20, dated July 31, 1990.
 10. NRC Examination Report No. 50-424/OL-90-03, dated August 23, 1990.
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Appendix G

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

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

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

Note: Only the disposition and/or status of generic actions for Zion are addressed.

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

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

Disposition: Ongoing

Reference 1 documents the status. The feasibility of using NPRDS is being reviewed. Initial data retrieval indicates that MCCB failures cannot be isolated using standard searches. Manufacturer and model number data is therefore being compiled, using, for example, the responses to Bulletin 88-10, "Non-Conforming Molded-Case Circuit Breakers," the data from Zion, and contacts with vendors. The degree of success in data retrieval, using this approach, will determine the value of NPRDS in addressing this item. If NPRDS is sufficient, AEOD, working with NRR, will provide an analysis to determine whether a generic communication is warranted.

Reference: 1. Memorandum from E. Jordan to J. Taylor (NRC), "Staff Actions Resulting from the Diagnostic Evaluation at Zion Nuclear Station—Six Month Status Report," dated March 1, 1991.

Action Source: Memorandum from J. Taylor to Office Directors and Region I Administrator, "Staff Actions Resulting from the Diagnostic Evaluation at Oyster Creek Nuclear Generating Station," dated February 14, 1991.

Note: Only the disposition and/or status of generic actions for Oyster Creek are addressed.

Item 1: Adequacy of the torus vacuum-breaker system.

Action: Assess the generic implications regarding the capability to satisfy both the vacuum breakers and containment isolation functions in similar Mark I plants. (Responsible Office: NRR)

Disposition: Ongoing - Assigned—A status report on the disposition of this item is scheduled to the EDO on February 4, 1992.

- References:**
1. Diagnostic Evaluation Team Report for the Oyster Creek Nuclear Generating Station, dated January 1991.
 2. Memorandum from Thomas E. Murley to James M. Taylor (NRC), NRR Action Plan Resulting From the Diagnostic Evaluation at the Oyster Creek Nuclear Generating Station, dated April 18, 1991.